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

July 31, 1997

Dr. Gunter Kegel, Director
Radiation Laboratory
University of Massachusetts Lowell
One University Avenue
Lowell, Massachusetts 01854

SUBJECT: ISSUANCE OF ORDER MODIFYING LICENSE NO. R-125 TO CONVERT FROM HIGH- TO LOW-ENRICHED URANIUM (AMENDMENT NO. 12) - UNIVERSITY OF MASSACHUSETTS LOWELL (TAC NO. M86788)

Dear Dr. Kegel:

The U.S. Nuclear Regulatory Commission (NRC) is issuing the enclosed Order, Amendment No. 12 to Facility Operating License No. R-125, to authorize the conversion from high-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel. This Order modifies the license in accordance with Section 50.64 of Title 10 of the Code of Federal Regulations (10 CFR 50.64), which requires that non-power reactors, such as the reactor at the University of Massachusetts Lowell, 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 May 21, 1993, as supplemented on March 17, 1994, May 16, 1997, and June 6, 1997.

The portions of the Order that allow possession of the LEU fuel [License Condition 2.B.(2)] and require submission of a startup report within 6 months of achieving initial criticality with LEU fuel [License Condition 2.C.(4)] are to be implemented 30 days after the date of publication of this Order in the Federal Register. The portions of the Order that change License Condition 2.B.(4) to allow the possession but not the use of the HEU fuel and that change Licence Condition 2.C.(2) and the technical specifications to apply to LEU fuel are to be implemented on the day of receipt of an adequate number and type of LEU fuel elements that are necessary to operate the facility as specified in your submittal and supplements.

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

Sincerely,

Theodore S. Michaels

Theodore S. Michaels, Senior Project Manager
Non-Power Reactors and Decommissioning
Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Docket No. 50-223
Enclosures: 1. Order
2. Replacement pages for Technical Specifications
3. Safety Evaluation

cc w/enclosures:
See next page

cc:

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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| In the Matter of<br><br>UNIVERSITY OF MASSACHUSETTS LOWELL<br><br>(University of Massachusetts Lowell<br>Research Reactor) | )<br>)<br>)<br>)<br>)<br>) | Docket No. 50-223 |

ORDER MODIFYING FACILITY OPERATING LICENSE NO. R-125

I.

The University of Massachusetts Lowell (the licensee) is the holder of Facility Operating License No. R-125 (the license) issued on December 24, 1974, by the U.S. Atomic Energy Commission, and subsequently renewed on November 21, 1985, by the U.S. Nuclear Regulatory Commission (the NRC or the Commission). The license authorizes operation of the University of Massachusetts Lowell Research Reactor (the facility) at a power level up to 1 megawatt thermal (MW(t)). The facility is a research reactor located in the center of the North Campus of the University of Massachusetts Lowell, in the city of Lowell, Middlesex County, in northeastern Massachusetts, approximately 5 miles from the New Hampshire border. The mailing address is Radiation Laboratory, University of Massachusetts Lowell, One University Avenue, Lowell, Massachusetts 01854.

II.

On February 25, 1986, 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 if Federal Government funding for conversion-related costs is available, each licensee of a non-power reactor replace HEU fuel at its facility with low-enriched uranium (LEU) fuel acceptable to the Commission unless the Commission has determined that the reactor has a unique purpose. The Commission issued the rule to reduce the risk of theft and diversion of HEU fuel used in non-power reactors.

Paragraphs 50.64(b)(2)(i) and (ii) require that a licensee of a non-power reactor (1) not acquire more HEU fuel if LEU fuel that is acceptable to the Commission for that reactor is available when the licensee proposes to acquire HEU fuel and (2) replace all HEU fuel in its possession with available LEU fuel acceptable to the Commission for that reactor in accordance with a schedule determined pursuant to 10 CFR 50.64(c)(2).

Paragraph 50.64(c)(2)(i) requires, among other things, that each licensee of a non-power reactor authorized to possess and to use HEU fuel develop and submit to the Director of the Office of Nuclear Reactor Regulation (Director) by March 27, 1987, and at 12-month intervals thereafter, a written proposal for meeting the requirements of the rule. The licensee shall include in its proposal a certification that Federal Government funding for conversion is available through the U.S. Department of Energy or other appropriate Federal agency and a schedule for conversion, based upon availability of replacement fuel acceptable to the Commission for that reactor and upon consideration of other factors such as the availability of shipping casks, implementation of arrangements for available financial support, and reactor usage.

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

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

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

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 May 21, 1993, as supplemented on March 17, 1994, May 16, 1997, and June 6, 1997, the NRC staff received the licensee's conversion proposal, including its proposed modifications and supporting safety analyses. HEU fuel elements are to be replaced with LEU fuel elements. The fuel elements contain fuel plates, typical of materials test reactors, with the fuel meat consisting of uranium silicide dispersed in an aluminum matrix. These plates contain the uranium-235 isotope at an enrichment of less than 20 percent. The NRC staff reviewed the licensee's proposal and the requirements of 10 CFR 50.64 and has determined that public health and safety and common defense and security require the licensee to convert the facility from the use of HEU to LEU fuel in accordance with the attachment to this Order and the schedule included herein. 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 10 CFR 50.64, IT IS HEREBY ORDERED THAT:

Facility Operating License No. R-125 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 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, DC 20555-0001, 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 manner in which his or her 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 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

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland,
this 31st day of July 1997

Attachment:
As stated

ATTACHMENT TO ORDER

MODIFYING FACILITY OPERATING LICENSE NO. R-125

A. License Conditions Revised and Added by This Order

2.B.(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess, and use at any one time up to 6.0 kilograms of contained uranium-235 at enrichments equal to or less than 20 percent in the form of material test reactor (MTR) type reactor fuel in connection with operation of the reactor and 5 Ci Am-Be and 10 Ci Sb-Be neutron sources for use in connection with operation of the reactor.

2.B.(4) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to possess, but not use, up to 4.80 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.

2.C.(2) Technical Specifications

The technical specifications contained in Appendix A, as revised through Amendment No. 12, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the technical specifications.

2.C.(4) The licensee shall submit a startup test report within six months of the initial criticality with low-enriched uranium reactor fuel in accordance with Amendment No. 12. 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. 12, Facility Operating License No. R-125, Docket No. 50-223, Replacement Pages for Technical Specifications," and as discussed in the safety evaluation for this Order.

ENCLOSURE TO LICENSE AMENDMENT NO. 12

FACILITY OPERATING LICENSE NO. R-125

DOCKET NO. 50-223

REPLACEMENT PAGES FOR TECHNICAL SPECIFICATIONS

Replace the Appendix A technical specifications in their entirety with the enclosed papers. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

# APPENDIX A

# TO

# FACILITY OPERATING LICENSE NO. R-125

# TECHNICAL SPECIFICATIONS

# FOR THE

# UNIVERSITY OF LOWELL

# TABLE OF CONTENTS

# 1.0 DEFINITIONS

## 1.1 ABNORMAL OCCURRENCES -

An *abnormal occurrence* is any of the following:

a. Any actual safety system setting less conservative than specified in Paragraph 2.2 of these Technical Specifications;

b. Operation in violation of a limiting condition for operation;

c. Safety system component malfunction or other component or system malfunction which could, or threatens to, render the system incapable of performing its intended function;

d. Release of fission products from a fuel element in a quantity that would indicate a fuel element cladding failure;

e. An uncontrolled or unanticipated change in reactivity greater than 0.5% delta k/k;

f. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of an unsafe condition in connection with the operation of the reactor;

g. Conditions arising from natural or offsite manmade events that affect or threaten to affect the safe operation of the facility.

## 1.2 CHANNEL -

A *channel* is the combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter. Such a channel is also referred to as a measuring channel. It may or may not be a safety channel.

1.3 CHANNEL CALIBRATION -

A *channel calibration* is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include a Channel Test.

1.4 CHANNEL CHECK -

A *channel check* is a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

1.5 CHANNEL TEST -

A *channel test* is the introduction of a signal into the channel for verification that it is operable.

1.6 CONTAINMENT BUILDING INTEGRITY -

Integrity of the containment building is said to be maintained when all isolation system equipment is operable or secured in an isolating position.

1.7 CONTROL ROD -

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

1.8 <u>EXCESS REACTIVITY</u> -

*Excess reactivity* is that amount of reactivity that would exist if all control rods (control and regulating) were moved to the maximum reactive condition from the point where the reactor is exactly critical ($k_{eff}=1$).

1.9 <u>EXPERIMENT</u> -

An *experiment* is any operation, hardware, or target which is designed to investigate non-routine reactor characteristics or which is intended for irradiation within the pool, on or in a beamport or irradiation facility and which is not rigidly secured to a core or shield structure so as to be a part of their design.

1.10 <u>MEASURED VALUE</u> -

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

1.11 <u>MOVABLE EXPERIMENT</u> -

A *movable experiment* is one in which the entire experiment may be moved into or out of the core or core region while the reactor is operating.

1.12 <u>OPERABLE</u> -

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

1.13 <u>OPERATING</u> -

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

1.14 PROTECTIVE CHANNEL -

A *protective channel* is a channel in the reactor safety system which is not merely a measuring channel.

1.15 REACTOR OPERATING MODE -

*Reactor operating mode* refers to the method by which the core is cooled, either natural convection mode of operation or forced convection mode of operation.

1.16 REACTOR OPERATING -

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

1.17 REACTOR SAFETY SYSTEM -

The *reactor safety system* consists of those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

1.18 REACTOR SECURED -

The reactor is secured when:

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

(2) A combination of the following:

a. The minimum number of neutron absorbing control rods are fully inserted or other safety devices are in the shutdown position, as required by technical specifications, and

b. The console key switch is in the off position and the key is removed from the lock, and

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

d. No experiments in or near the reactor are being moved or serviced that have, on movement, a reactivity worth exceeding that maximum value allowed for a single experiment or one dollar, whichever is smaller.

1.19 <u>REACTOR SHUTDOWN</u> -

The reactor is shut down if it is subcritical by at least 0.7% delta k/k in the Reference Core Condition plus the absolute reactivity worth of all experiments.

1.20 <u>REFERENCE CORE CONDITION</u> -

The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible <.2% delta k/k.

1.21 <u>REGULATING ROD</u> -

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

1.22 <u>SAFETY CHANNEL</u> -

A *safety channel* is a measuring or protective channel in the reactor safety system.

1.23 SECURED EXPERIMENT -

A *secured experiment* is an experiment, experimental facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The retaining devices must be able to withstand the hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or forces which can arise as a result of credible malfunctions.

1.24 SHALL, SHOULD, AND MAY -

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

1.25 SHUTDOWN MARGIN -

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

1.26 SURVEILLANCE INTERVALS -

The average over any extended period for each surveillance time interval shall be closer to the normal surveillance time than the extended time. Any extension of these intervals shall be occasional and for a valid reason, and shall not affect the average as defined. Allowable *surveillance intervals* shall not exceed the following:

a. Five year (interval not to exceed six years)

b. Two year (interval not to exceed two and one half years)

c. Annual (interval not to exceed 15 months)

d. Semi-annual (interval not to exceed seven and one half months)

e. Quarterly (interval not to exceed four months)

f. Monthly (interval not to exceed six weeks)

g. Weekly (interval not to exceed ten days)

h. Daily (must be done during the calendar day).

1.27 TRUE VALUE -

The *true value* is the actual value of a parameter.

# 2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

## 2.1 SAFETY LIMITS

### 2.1.1 Safety limits in the forced convection mode of operation.

Applicability

This specification applies to the interrelated variables associated with core thermal and hydraulic performance with forced convection flow. These variables are:

P = Reactor thermal power

W = Reactor coolant flow rate

$T_i$ = Reactor coolant inlet temperature

L = Height of water above the center line of the core

Objective

To assure that the integrity of the fuel cladding is maintained.

Specification

Under the conditions of forced convection flow:

1. The combination of true values of reactor thermal power (P) and reactor coolant flow rate (W) shall not exceed the limits shown in Figure 2.1 T.S under any operating conditions. The limits are considered exceeded if the point defined by the true values of P and W is at any time above the curve shown in Figure 2.1 T.S.
2. The true value of the pool water level (L) shall not be less than 24 feet above the center line of the core.

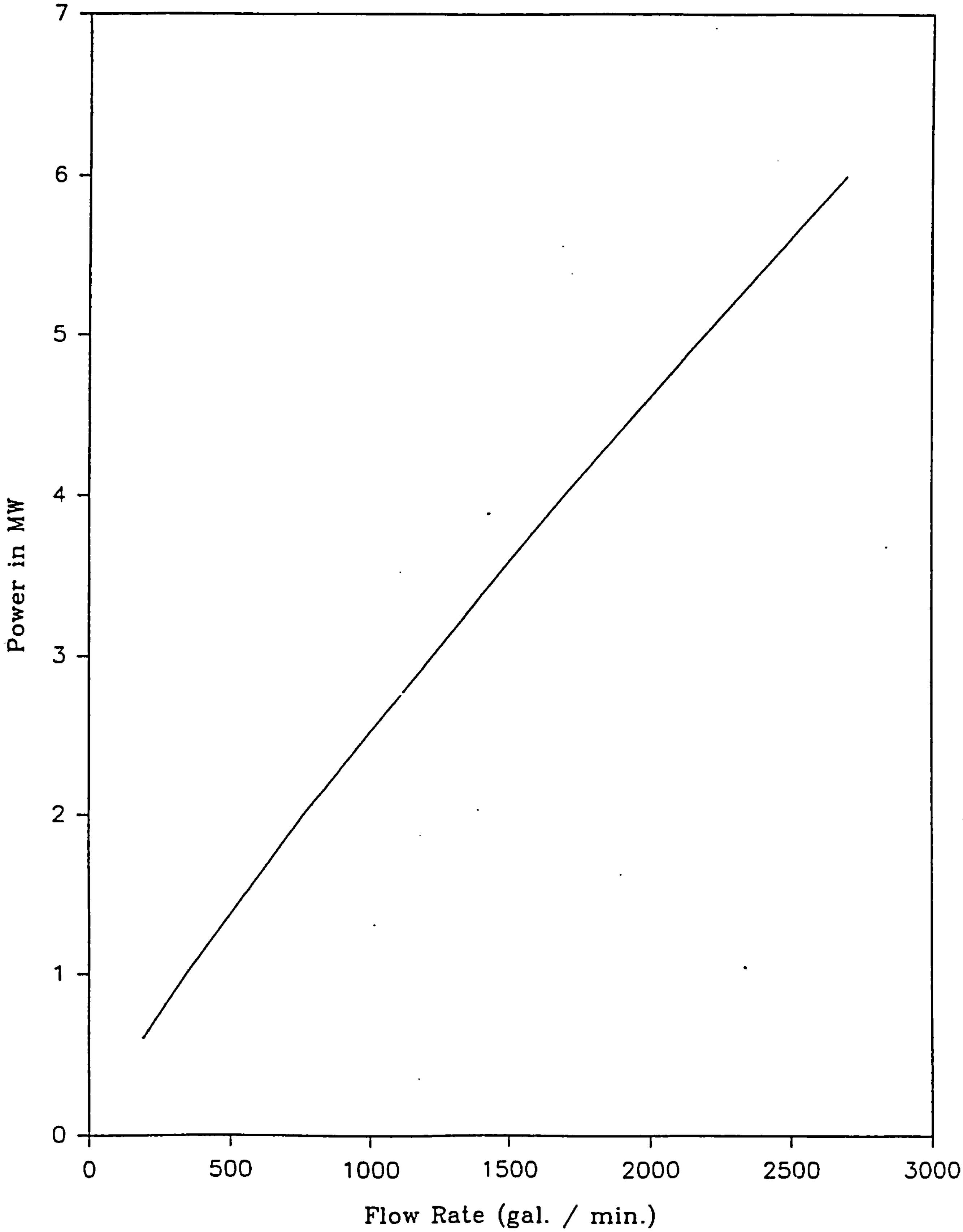


**Figure 2.1 T.S. Power-Flow Safety Limit Curve**

3. The true value of the reactor coolant inlet temperatu (pool temperature, $T_P$) shall not be greater than 110°F.

Bases

In the region of full power operation, the criterion used to establish the safety limit was the onset of nucleate boiling (ONB) at the hot spot in the hot channel. The analysis is given in Section 3.1.2.2 of the FSAR Supplement for Conversion to Low Enrichment Uranium (LEU) Fuel.

2.1.2 Safety Limits in the natural convection mode of operation.

Applicability

This specification applies to the interrelated variables associated with core thermal and hydraulic performance with natural convection flow. These variables are:

P = Reactor thermal power

$T_P$ = Reactor pool temperature

L = Height of water above the center line of the core

Objective

To assure that the integrity of the fuel cladding is maintained.

Specification

Under conditions of natural convection flow:

1. The true value of the reactor thermal power (P) shall not exceed 0.335 MW.
2. The true value of the reactor thermal power (P) shall not exceed 1.33 kW when the true value of the pool

water level (L) is less than 24 feet above the center line of the core.

3. The reactor shall not be taken critical when the true value of the pool water level (L) is less than 2 feet above the center line of the core.

4. The true value of the reactor coolant inlet temperature (pool temperature, $T_p$) shall not be greater than 110°F.

Bases

The criterion for establishing a safety limit with natural convection flow is the onset of nucleate boiling at the hot spot on the hot channel. The analysis of natural convection flow given in Section 3.1.2.1 of the FSAR Supplement for Conversion to LEU Fuel shows that ONB occurs at 0.335 MW with a corresponding fuel clad temperature of 118.6°C (245.5°F) which is well below the temperature at which fuel clad damage could occur.

Operation of the reactor with less than full water height above the core is limited to a power about 250 times lower than the limit with full water height; there is no possibility of fuel clad damage under water immersion at 1.33 kW.

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

2.2.1 Limiting Safety System Settings in the forced convection mode of operation.

Applicability

This specification applies to the setpoints for the safety channels monitoring reactor thermal power (P), coolant flow rate (W), reactor coolant inlet temperature ($T_i$), and the height of water above the center line of the core (L).

Objective

To assure that automatic protective action is initiated in order to prevent a Safety Limit from being exceeded.

Specification

Under conditions of forced convection flow the values of the Limiting Safety System Settings shall be as follows:

P = 1.25 MWt (max)

W = 1170 GPM (min)

$T_i$ = 108°F (max)

L = 24.25 ft (min)

Bases

The Limiting Safety System Settings that are given in Specification 2.2.1 represent values of the interrelated variables which, if exceeded, shall result in automatic protective action that will prevent Safety Limits from being exceeded during the course of the most adverse anticipated transient. To determine the LSSS given above, an analysis of the uncertainties in the instruments and measurements was taken into account. These safety settings are adjusted so that the true value of the measured parameter will not exceed the specified Safety Limits. The results of these adjustments included a flow variation of 4%, a temperature variation of $2^{o}F$, a power level variation of 6%, and a pool water level variation of three inches. (See Section 3.1.2.5 of the FSAR Supplement for Conversion to LEU Fuel and Paragraph 9.1.2 of the FSAR).

2.2.2 Limiting Safety System Settings in the natural convection flow mode of operation.

Applicability

This specification applies to the setpoints for the safety channels monitoring reactor thermal power (P), reactor pool temperature ($T_p$), and the height of water above the center line of the core (L).

Objective

To assure that automatic protective action is initiated in order to prevent undesirable radiation levels on the surface of the pool.

Specification

Under conditions of natural convection flow the measured values of the Limiting Safety System Settings shall be as follows:

| Full Pool Level | Low Pool Level |
|---|---|
| P = 125 kW (max) | P = 1.25 kW (max) |
| $T_p$ = $108^{o}F$ (max) | $T_p$ = $108^{o}F$ (max) |
| L = 24.25 ft (min) | L = 2.25 ft (min) |

Bases

The Limiting Safety System Settings that are given in Specification 2.2.2 represent values of the interrelated variables which, if exceeded, shall result in automatic protective action that will prevent undesirable radiation levels on the surface of the pool due to: a) the production and escape of $^{16}N$ during the natural convection mode of operation with full pool level, and b) direct radiation from the core during low pool level operation. The specifications given above assure that an adequate safety margin exists between the LSSS and the SL for natural convection, because the values of the power LSSS would be much higher (335 kW, Section 3.1.2.1 of the FSAR Supplement for Conversion to LEU Fuel) if the specifications were based on Safety Limits rather than on $^{16}N$ production. The $^{16}N$ criterion is not related to the ONB which was the criterion used in establishing the Safety Limits (see Section 3.1.2.1 of the FSAR Supplement for Conversion to LEU Fuel).

# 3.0 LIMITING CONDITIONS FOR OPERATION

## 3.1 REACTIVITY

Applicability

These specifications apply to the reactivity condition of the reactor and the reactivity worths of control rods, regulating rod, and experiments.

Objective

To assure that the reactor can be safely shutdown and maintained in a safe shutdown condition at all times and that the Safety Limits will not be exceeded.

Specification

The reactor shall not be operated unless the following conditions exist:

1. The minimum shutdown margin relative to the cold, clean (xenon-free) critical condition, with the most reactive control rod in the fully withdrawn position, is greater than 2.7% delta k/k.
2. The reactor core is loaded so that the excess reactivity in the cold clean (xenon-free) critical condition does not exceed 4.7% delta k/k.
3. All core grid positions are filled with fuel elements, irradiation baskets, source holders, regulating rod, graphite reflector elements or grid plugs. All but 5 of the peripheral radiation baskets must contain flow restricting devices. This specification will not apply for low power operation, <10 kW, without forced flow.

4. The drop time of each control rod from a fully withdrawn position is less than 1.0 second.

5. The isothermal temperature coefficient of reactivity is negative at temperatures >70°F.

6. The reactivity insertion rates of the control rods are less than 0.025% delta k/k per second.

7. The total reactivity worth of the regulating rod is less than the effective delayed neutron fraction.

8. The reactivity insertion rate of the regulating rod is less than 0.054% delta k/k per second.

9. The reactivity worth of experiments shall not exceed the values indicated in the following table:

| Kind | Single Experiment Worth | Total Worth |
|---|---|---|
| Movable (including the pneumatic rabbit) summed together for all experiments | 0.1% delta k/k | 0.5% delta k |
| Secured experiments | 0.5% delta k/k | 2.5% delta k/k |

10. The total reactivity worth of all experiments shall not be greater than 2.5% delta k/k.

Bases

1. The shutdown margin required by Specification 1 assures that the reactor can be shut down from any operating condition and will remain shutdown after cooldown and xenon decay, even if the highest worth control rod should be in the fully withdrawn position.

2. The maximum allowed excess reactivity of 4.7% delta k/k provides sufficient reactivity to accommodate fuel burnup, xenon and samarium poisoning buildup, experiments, and control requirements, but gives a sufficient shutdown margin even with the highest worth rod fully withdrawn.
3. The requirement that all grid plate positions be filled and the restriction on radiation baskets during reactor operation assures that the quantity of primary coolant which bypasses the heat producing elements will be kept within the limits used in establishing Safety Limits in Section 3.1.2 of the FSAR Supplement for Conversion to LEU Fuel. This requirement does not apply under natural circulation conditions at low power
4. The control rod drop time required by Specification 4 assures that the Safety Limit will not be exceeded during the flow coast down which occurs upon loss of forced convection coolant flow. The analysis of this situation, which is given in Section 3.1.2.5 of the FSAR Supplement for Conversion to LEU Fuel, assumes a 1 second rod drop time.
5. The requirement for a negative temperature coefficient of reactivity assures that any temperature rise caused by a reactor transient will not cause a further increase in reactivity.

6. The maximum rate of reactivity insertion by the control rods which is allowed in Specification 6 assures that the Safety Limit will not be exceeded during a startup accident due to a continuous linear reactivity insertion. Analysis in Section 3.1.2.9 of the FSAR Supplement for Conversion to LEU Fuel shows that a maximum power of less than 1.3 MW would be reached assuming a continuous linear reactivity insertion rate of 0.035% delta k/k per second, which is greater than the maximum allowed.
7. Limiting the reactivity worth of the regulating rod to a value less than the effective delayed neutron fraction assures that a failure of the automatic servo control system could not result in a prompt critical condition.
8. The maximum rate of reactivity insertion by the regulating rod which is allowed in Specification 8 assures that the Safety Limit on reactor power will not be exceeded during an operational accident involving the continuous withdrawal of the regulating rod. The analysis, in Section 3.1.2.9 of the FSAP Supplement for Conversion to LEU Fuel, shows that the maximum power reached would be about 1.3 MW.
9. Specification 9 assures that the failure of a single experiment will not result in the exceeding of a Safety Limit; the analysis of the step insertion of 0.5% delta k/k is given in Section 3.1.2.8 of the FSAR Supplement for Conversion to LEU Fuel. Limiting a movable experiment such as the pneumatic rabbit to 0.1% delta k/k assures that the prompt

jump, which is about 17%, will result in a power below the power level scram setting, i.e., below 125% of power.

10. The total reactivity of 2.5% in Specification 10 places a reasonable upper limit on the worth of all experiments which is compatible with the allowable excess reactivity and the shutdown margin and is consistent with the functional mission of the reactor.

## 3.2 REACTOR INSTRUMENTATION

Applicability

This specification applies to the instrumentation which must be available and operable for safe operation of the reactor.

Objective

The objective is to require that sufficient information be available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless the measuring channels listed in the following table are operable:

| Measuring Channel | Minimum Required | Operating Mode in Which Required |
|---|---|---|
| Startup Count Rate | 1 | All modes (during reactor startup) |
| Log N (Period) | 1 | All modes |
| Power Level (Linear N) | 2 | All modes |
| Reactor Coolant Inlet Temperature | 1 | Forced convection |
| Coolant Flow Rate | 1 | Forced convection |

| | | |
|---|---|---|
| Reactor Pool Temperature | 1 | All modes |

Bases

The neutron detectors assure that measurements of the reactor power level are adequately displayed during reactor startup and low and high power operation. The temperature and flow detectors give information to the operator to prevent the exceeding of a Safety Limit.

## 3.3 REACTOR SAFETY SYSTEM

Applicability

This specification applies to the reactor safety system channels.

Objective

To require the minimum number of reactor safety system channels that must be operable in order to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless the reactor safety system channels described in the following table are operable.

| Reactor Safety System Component/Channel | Minimum Required | Function | Operating Mode in Which Required |
|---|---|---|---|
| Startup Count Rate | 1 | Prevent blade withdrawal when N count rate $\leq$ 2 cps | Reactor startup in all modes |
| Reactor Period | 1 | Automatic reactor scram with $\leq$ 3 sec period<br>Control blade inhibit $\leq$ 15 sec period | All modes |

| Reactor Safety System Component/Channel | Minimum Required | Function | Operating Mode in Which Required |
|---|---|---|---|
| Reactor Power Level | 2 | Automatic scram when $\geq$ 125% of range scale | All modes |
| Coolant Flow Rate | 1 | Automatic scram at 1170 gpm | Forced convection above 0.1 MW |
| Seismic Disturbance | 1 | Automatic scram at Modified Mercalli Scale IV | All modes |
| Primary Piping Alignment | 1 | Automatic scram | Forced Convection above 0.1 MW |
| Pool Water Level | 1 | Automatic scram at:<br>(1) 24.25 ft above core center line;<br>(2) 2.25 ft above core center line | <br>(1) All modes with full water height;<br>(2) operation with limited water height |
| Pool Temperature | 1 | Automatic scram $\geq$ 108°F | All modes |
| Coolant Inlet Temperature | 1 | Automatic scram $\geq$ 108°F | Forced convection above 0.1 MW |
| Bridge Movement | 1 | Automatic scram if moved >1 inch | All modes |

| Reactor Safety System Component/Channel | Minimum Required | Function | Operating Mode in Which Requi |
|---|---|---|---|
| Coolant Gate Opens | 2 | Automatic scram if either the coolant riser or coolant downcomer gate opens | Forced convection above 0.1 MW; down comer flow pattern |
| | 1 | Automatic scram if the coolant riser gate opens | Forced convection above 0.1 MW; cross pool flow pattern |
| Detector High Voltage Failure | 1 | Automatic scram if Voltage <500V | All modes |
| Thermal Column Door Open | 1 | Automatic scram | All modes |
| Truck Door and/or Air lock Integrity | 3 | Automatic scram | All modes |
| Manual Scram Button | 1 | Manual scram | All modes |
| "Reactor On" Key-Switch | 1 | Manual scram if "off" | All modes |

Bases

The inhibit function on the startup channel assures that the required startup neutron source is sufficient and in a proper location for the reactor startup, such that a minimum source multiplication count rate level is being detected to ensure proper operation of the startup channel.

The automatic protective action initiated by the reactor period channel, high flux channels, flow rate channels, coolant inlet temperature channel, pool temperature channel, and pool water level channel provides the redundant protection to assure that a Safety Limit is not exceeded.

Automatic protection action initiated by the seismic detector, bridge misalignment, opening of coolant gates, high voltage failure, and opening of thermal column door assures shutdown of the reactor under conditions that could lead to a safety problem.

The automatic protective action covering the condition of the air lock doors assures that containment capability is maintained.

The manual scram button and the "Reactor On" Key-Switch provide two manual scram methods to the operator if any abnormal condition should occur.

## 3.4 RADIATION MONITORING EQUIPMENT

Applicability

This specification applies to the availability of radiation monitoring equipment which must be operable during reactor operation.

Objective

To assure that radiation monitoring equipment is available for evaluation of radiation conditions in restricted and unrestricted areas.

Specification

1. When the reactor is operating, gaseous and particulate sampling of the stack effluent shall be monitored by a stack monitor with readouts in the control room.
2. When the reactor is operating, at least one constant air monitoring unit located in the containment building on the reactor pool level and having a readout in the control room shall be operating.
3. The reactor shall not be continuously* operated without a minimum of one radiation monitor on the experimental level of the reactor building and one monitor over the reactor pool operating and capable of warning personnel of high radiation levels.

Bases

A continuing evaluation of the radiation levels within the reactor building will be made to assure the safety of personnel. This is accomplished by the area monitoring system of the type described in Chapter 10 of the FSAR.

*In order to continue operation of the reactor, replacement of an inoperative monitor must be made within 15 minutes of recognition of failure, except that the reactor may be operated in a steady-state power mode if the installed systems are replaced with portable gamma-sensitive instruments having their own alarm.

A continuing evaluation of the stack effluent will be made using the information recorded from the particulate and gas monitors.

## 3.5 CONTAINMENT AND EMERGENCY EXHAUST SYSTEM

### Applicability

This specification applies to the operation of the reactor containment and emergency exhaust system.

### Objective

To assure that the containment and emergency exhaust system is in operation to mitigate the consequences of possible release of radioactive materials resulting from reactor operation.

### Specification

The reactor shall not be operated unless the following equipment is operable, and conditions met:

| | Equipment/Condition | Function |
|---|---|---|
| 1. | At least one door in each of the personnel air locks is closed and the truck door is closed. | To maintain containment system integrity |
| 2. | All isolation valves, except that reactor operation can proceed if a failed isolation valve is in the closed (isolated) position. | To maintain containment system integrity |

| | Equipment/Condition | Function |
|---|---|---|
| 3. | Initiation system for containment isolation. | To maintain containment system integrity |
| 4. | Emergency exhaust system | To maintain the ability to tend toward a negative building pressure without unloading any large fraction of possible airborne activity. |
| 5. | Vacuum relief device | To ensure that building vacuum will not exceed 0.2 psi. |
| 6. | Reactor Alarm system* | To assure that proper emergency action is taken. |

Bases

In the unlikely event of a release of fission products, or other airborne radioactivity, the containment isolation initiation system will secure the normal ventilation exhaust fan, will bypass the normal ventilation supply up the stack, and will close the normal inlet and exhaust valves. In containment, the emergency exhaust system will tend to maintain a negative building pressure with a combination of controls intended to prevent unloading any large fraction of airborne activity if the internal building pressure is high. The emergency exhaust purges the building air through charcoal and absolute filters and controls the discharge, which is diluted by supply air, through a 100-foot stack on site. Chapter 3

*The public address system can serve as a temporary substitute for reactor evacuation and formation of the Emergency Team during short periods of maintenance.

of the FSAR describes the system's sequence of operation; Chapter 9 provides the analysis.

## 3.6 LIMITATIONS OF EXPERIMENTS

Applicability

This specification applies to experiments to be installed in the reactor and associated experimental facilities.

Objectives

To prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specification

The reactor shall not be operated unless the following conditions governing experiments exist:

1. All materials to be irradiated shall be either corrosion resistant or encapsulated within corrosion resistant containers to prevent interaction with reactor components or pool water. Corrosive materials shall be doubly encapsulated.
2. Irradiation containers to be used in the reactor, in which a static pressure will exist or in which a pressure buildup is predicted, shall be designed and tested for a pressure exceeding the maximum expected by a factor of 2.
3. Explosive material such as (but not limited to) gunpowder, dynamite, TNT, nitroglycerine, or PETN in quantities <25 mg may be irradiated in the reactor or experimental facilities provided out-of-core tests indicate that, with the containment provided, no damage to the explosive

containers, the reactor, the reactor components or the Co-( Source shall occur upon detonation of the explosive.

4. Explosive materials, in quantities >25 mg shall not be allowed in the reactor or the reactor pool without rigorous safety evaluation, and special authorization from the USNRC.
5. All experiments shall be designed against failure from internal and external heating at the true values associated with the LSSS for reactor power level and other process variables.
6. The outside surface temperature of a submerged experiment or capsule shall not exceed the saturation temperature of the reactor coolant during operation of the reactor.
7. Experimental apparatus, material or equipment to be irradiated shall be positioned so as not to cause shadowing of the nuclear instrumentation, interference with control rods, or other perturbations which may interfere with safe operation of the reactor.
8. Cryogenic liquids shall not be used in any experiment within the reactor pool.
9. The reactor shall not be operated whenever the reactor core is in the same end of the reactor pool as any portion of the Cobalt-60 Source.

Bases

Specifications 1 through 6 are intended to reduce the likelihood of damage to reactor components and/or radioactivity releases resulting from experiment failure, including any experiment involving the Co-60 Source and, along with the reactivity restriction of pertinent specifications in 3.1, serve as a guide for the review and approval of new and untried experiments by the operations staff as well as the Reactor Safety Subcommittee. Specification 7 assures that no physical or nuclear interferences compromise the safe operation of the reactor by, for example, tilting the flux in a way that could affect the peaking factor used in the Safety Limit calculations. Review of the experiments using the appropriate LCO's and the Administrative Controls of Section 6 assures that the insertion of experiments will not negate the considerations implicit in the Safety Limits.

Specification 8 prohibits experiments using cryogenic materials. (Special NRC permission would be required.) Cryogenic liquids present structural and explosive problems which enhance the potential of an experiment failure. Specification 9 assures that there will be no interference, either instrumental or procedural, between the reactor and the cobalt source during reactor operation.

## 3.7 GASEOUS EFFLUENTS

Applicability

This specification applies to the routine release of gaseous radioactive effluents from the facility.

Objective

The objective is to minimize the release of gaseous radioactive effluents, particularly Argon-41, the effluent most likely to be generated in routine operation.

Specification

The release rate of gaseous radioactive material from the reactor stack shall be limited to 8 microcuries per second averaged over year.

Bases

Calculations based on a very conservative model, allowing for no atmospheric dilution of the gaseous effluent, have predicted an annual dose of 12 mrem to an individual exposed to the effluent on a continual basis for an Argon-41 release rate of 8 microcurie per second. Allowance for even minimal atmospheric turbulence would redu.:; this dose number by about a factor of three.

## 3.8 COOLANT SYSTEM

Applicability

This specification applies to the reactor pool water requirements for operation of the reactor.

Objective

The objectives are to require that the reactor pool water be of high purity in order to retard corrosion and to monitor the integrity of the fuel cladding and the Cobalt-60.

Specification

1. The conductivity of the pool water shall be maintained at a value of 5 micromhos pei centimeter or less averaged over a month.

2. The pool water shall be analyzed for gross activity and for Cobalt-60. Analyses shall be capable of detecting levels of $10^{-7}$ microcuries per milliliter. If a sample analysis reveals a significant increase of activity in the water, with respect to the previous samples, or a contamination level greater than $10^{-6}$ microcuries of Cobalt-60 per milliliter of water, prompt action shall be taken to prevent further contamination of the pool water. If the gross activity of the sample is less than $10^{-7}$ microcuries per milliliter, specific analysis for Cobalt-60 need not be performed. If remedial action is required by this section, notification will be made to the USNRC as required by Section 6.6.2.

Bases

Pool water of high purity minimizes the rate of corrosion. Radionuclide analysis of the pool water allows early determination of any significant buildup of radioactivity from operation of the reactor or the Cobalt-60 source.

## 4.0 SURVEILLANCE REQUIREMENTS

### 4.1 CONTROL AND REGULATING RODS

Applicability

This specification applies to the surveillance requirements for the control and regulating rods.

Objective

To assure the operability of the control and regulating rods.

Specifications

1. The reactivity worth of the regulating rod and each control rod shall be determined annually. The reactivity worth of all rods shall also be determined prior to routine operation of any new fuel configuration in the reactor core.

2. Control rod drop and drive times and regulating rod drive time shall be determined annually, or if maintenance or modification is performed on the mechanism. Nominally, the withdrawl rate of the safety blades is at 3.5 inches per minute and the withdrawl rate of the regulating rod is at 78 inches per minute.

3. The control and regulating rods shall be visually inspected annually.

Bases

The reactivity worth of the control and regulating rods is measured to assure that the required shutdown margin is available, and to provide a means for determining the reactivity worths of experiments inserted in the core. Annual measurement of reactivity worths provides a correction for the slight variations expected because of burnup. The required measurement after any new arrangement of fuel in the core assures that possibly altered rod worths will be known before routine operation. The visual inspection of the regulating and control rods and the measurements of drive and drop times are made to assure that the rods are capable of performing properly and within the considerations used in transient analyses in the FSAR Supplement for Conversion to LEU Fuel. Appropriate inspection data will be recorded and analyzed for trends. Verification of operability after maintenance or modification of the control system will ensure proper reinstallation or reconnection.

4.2 REACTOR SAFETY SYSTEM

Applicability

This specification applies to the surveillance requirements for the Reactor Safety System.

Objective

To assure that the Reactor Safety System (RSS) will remain operable and will prevent the Safety Limits from being exceeded.

Specifications

1. A channel check of each measuring channel in the RSS shal be performed daily when the reactor is in operation.
2. A channel test of each measuring channel in the RSS shall t performed prior to each day's operation, or prior to each operation extending more than one day.
3. A channel calibration (reactor power level) of the Log N an linear safety power level measuring channels shall be mad annually.
4. A channel calibration of the following channels shall be made annually.
   a. Pool water temperature
   b. Primary coolant flow rate
   c. Pool water level
   d. Primary coolant inlet and outlet temperature
5. The manual scram shall be verified to be operable prior to each reactor startup.
6. Any RSS instrument channel replacement must have undeigone a channel check prior to installation, and musi undergo a channel calibration before routine operation of the reactor after channel installation.
7. Any RSS instrument repaired or replaced while the reactor is shutdown must have a channel test prior to reactor operation.
8. Each protective channel in the RSS shall be verified to be operable semi-annually.

Bases

The daily channel tests and checks and periodic verifications will assure that the safety channels are operable. Annual calibrations will assure that long-term drift of the channels is corrected. The calibration of the reactor power level will provide continual reference for the adjustment of the Log N and safety channel detector positions and current alignments.

## 4.3 RADIATION MONITORING EQUIPMENT

Applicability

This specification applies to the surveillance requirements for the area radiation monitoring equipment and systems for monitoring airborne radioactivity.

Objective

To assure that the equipment used for monitoring radioactivity is operable and to verify the appropriate alarm settings.

Specification

1. The operation of the area radiation monitoring equipment and systems for monitoring airborne radioactivity, and their associated alarm set points, shall be verified prior to reactor startup.
2. All radiation monitoring systems shall be calibrated semiannually.

Bases

The area radiation monitoring system, described in the Emergency Plan, includes the stack air monitor, two building constant air monitors, a fission product monitor, 12 GM detectors and two ion chamber detectors at selected sites throughout the building. The detectors used have been chosen for stability and operational reliability. The large number of detectors in the area monitoring system ensures that if a particular monitor should malfunction or drift out of calibration, sufficient backup monitors are available for reliable information. Calibration of the area monitors semi-annually is sufficient to insure the required reliability. Daily checks (during operating days) of the area monitors ensure that any obvious malfunctions will be detected.

## 4.4 CONTAINMENT BUILDING

Applicability

This specification applies to the surveillance requirements for the containment building.

Objective

To assure that the containment system is operable.

Specification

1. Building pressure will be verified at least every eight hours during reactor operation to ensure that it is less than ambient atmospheric pressure.

2. The containment building isolation system including the initiating system shall be tested semi-annually. The test shall verify that valve closure is achieved in <2.5 seconds after the initial signal.
3. An integrated leakage rate of the containment building as-is* shall be performed at a pressure of at least 0.5 psig at intervals of 5 years to verify leakage rate of less than 10% of the building air volume/day at 2 psig.
4. All additions, modifications, or maintenance of the containment building or its penetrations that could affect building containment capability shall be tested to verify containment requirements.
5. The emergency exhaust system including the initiating system shall be verified annually to be operable.
6. At two year intervals, and subsequent to replacement of the facility filters and prior to reactor operation thereafter, the filter trains shall be tested to verify that they are operable.
7. At two year intervals, the air flow rate in the stack exhaust duct shall be measured.

*Non-routine maintenance or repair for the purpose of reducing containment leakage shall not be performed prior to the leak test.

Bases

Maintaining a negative pressure ensures that any leakage in the containment is inward.

Valve closure time was chosen to be 1/2 the time required for a given sample of air to travel from the first to the second valve in series in the exhaust line under regular flow conditions. Semi-annually is considered a reasonable frequency of testing. The containment building was designed to withstand a 2.0 psig internal pressure. An overpressure of less than 0.5 psig would result from an excursion of 538 MWs, which is nearly four times the energy release achieved in the Borax tests. A 0.5 psig test pressure is therefore adequate.

Any additions, modifications or maintenance to the building or its penetrations shall be tested to verify that such work has not adversely affected the integrity of the building.

Surveillance of the emergency exhaust system and the various filters will verify that these are functioning. See Chapters 3 and 7 of the FSAR.

4.5 POOL WATER

Applicability

This specification applies to the surveillance requirement of monitoring the quality and the radioactivity in the pool water.

Objective

To assure high quality pool water and to monitor the radioactivity in the pool water in order to verify the integrity of the fuel cladding.

Specification

1. The conductivity of the pool water shall be measured weekly.
2. The radioactivity in the pool water shall be analyzed weekly.

Bases

Surveillance of water conductivity assures that changes that could accelerate corrosion have not occurred. Radionuclide analysis of the pool water samples will allow early determination of any significant buildup of radioactivity from operation of the reactor or the Co-60 source.

4.6 SCRAM BY PROCESS VARIABLE EFFECT

Applicability

This specification applies to the surveillance requirements applied to process variable scrams.

Objective

To assure that a Safety Limit is not exceeded.

Specification

Following a reactor scram caused by a process variable, the reactor shall not be operated until an evaluation has been made to determine if a Safety Limit was exceeded, the cause of the scram, the effects of operation to the scram point and the appropriate action to be taken.

Bases

This specification assures that if a Safety Limit should be exceeded as a result of a malfunction of a process variable, the fact will be known.

## 4.7 FUEL SURVEILLANCE

Applicability

This specification applies to the surveillance requirements for reactor fuel.

Objective

To assure that reactor fuel is in proper physical condition.

Specification

Visual inspection of a representative sample of reactor fuel elements shall be performed every two years.

Bases

The inspection of reactor fuel assures that fuel elements, when used in the core, will perform as designed.

# 5.0 DESIGN FEATURES

## 5.1 REACTOR FUEL

The reactor fuel shall be as follows:

1. Standard fuel element: the fuel elements shall be flat plate MTR-type elements. The plates shall be fueled with low enrichment (20% U-235) $U_3Si_2$, clad with aluminum. There shall be 18 plates per element with 16 containing fuel and two outside plates of aluminum. There shall be 200$\pm$ 5.6 grams of Uranium-235 per element.
2. Half-element: same as a standard fuel element except each plate has one half the uranium loading.
3. Variable-load element: same as Specification 1 above, but internal plates are removable.

## 5.2 REACTOR CORE

1. The reactor core consists of a 9 x 7 array of 3-inch square modules with the four corners occupied by posts. The reference core for these Technical Specifications consists of 20 standard fuel elements in a 5 x 5 array with corners removed and the central location filled with a graphite-water aluminum clad flux trap element, as shown in Figure 2.6 of the FSAR Supplement for Conversion to LEU Fuel.
2. Cores from 16 standard elements to 28 elements may be used, and cores from 16 elements to 28 elements may contain 2 half-loaded elements.

3. Cores with loadings different from 20 standard elements may be operated under forced convection only after analyses using 1) the methods described in the FSAR Supplement for Conversion to LEU Fuel, or 2) flux measurements in natural convection, establish that no alteration of the LSSS's are required to preclude violation of a SL during the transients anticipated in the FSAR. The analysis results and flux measurements for LEU cores with fewer than 20 standard elements to be operated above 100Kw must be reviewed and approved by the NRC prior to operation with the smaller cores.

5.3 <u>REACTOR BUILDING</u>

The reactor shall be housed in the reactor building, designed for containment.

5.4 <u>FUEL STORAGE</u>

All reactor fuel element storage facilities shall be designed in geometrical configuration so that $k_{eff}$ is less than 0.85 under quiescent flooding with water.

# 6.0 ADMINISTRATIVE CONTROLS

## 6.1 ORGANIZATION AND MANAGEMENT

1. The reactor facility shall be an integral part of the Radiation Laboratory of the University of Massachusetts Lowell. The reactor shall be related to the University structure as shown in Chart 6-1 TS.
2. The reactor facility shall be under the direction of the Director of the Radiation Laboratory, who shall be a member of the graduate faculty, and it shall be supervised by the Reactor Supervisor who shall be an NRC-licensed senior operator for the facility. The Reactor Supervisor shall be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and the provisions of the Reactor Safety Subcommittee.
3. There shall be a Radiation Safety Officer responsible for the safety of operations from the standpoint of radiation protection. He does not report formally to the line organization responsible for reactor operations, but rather to the Vice-Chancellor for Academic Affairs (see Chart 6-1 TS).
4. An Operator or Senior Operator licensed pursuant to 10 CFR 55 shall be present at the controls unless the reactor is secured as defined in these specifications. In addition, a second individual shall be present in the reactor building or Pinanski building whenever the reactor is not secured. This individiual shall be a Licensed Senior Operator, Licensed

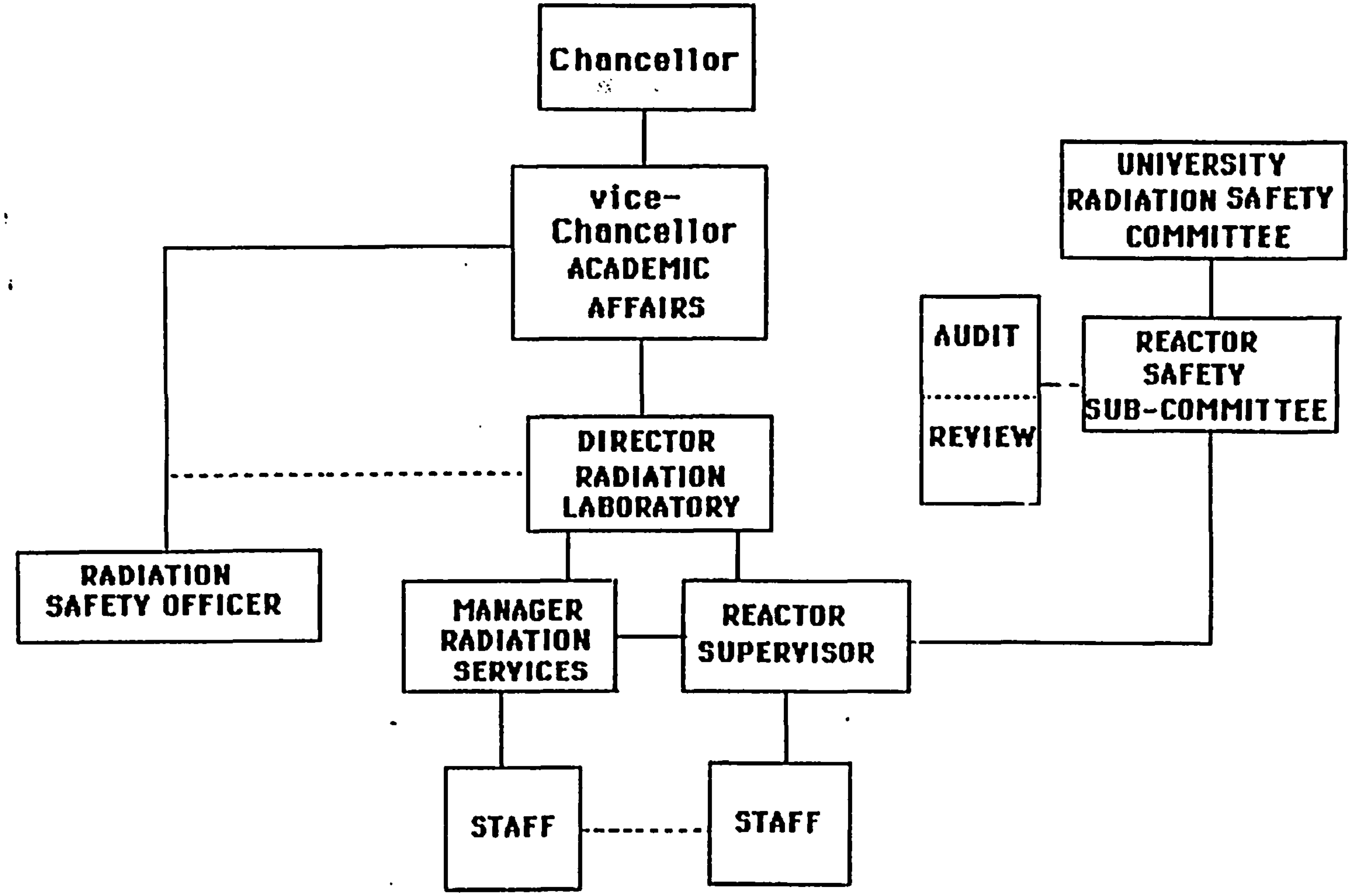


Figure 6.1 TS Organizational Chart For The University of Massachusetts Lowell Radiation Laboratory

Operator or an individual who is capable of shutting the reactor down in case of an emergency.

5. A Licensed Senior Operator shall be on the console or readily available on call whenever the reactor is in operation.

6.2 <u>REVIEW AND AUDIT</u>

1. There shall be a Reactor Safety Subcommittee which shall review reactor operations to assure that the facility is operated in a manner consistent with public safety and within the terms of the facility license. The Subcommittee shall report to the University Radiation Safety Committee which has overall authority in the use of all radiation sources at the University.

2. The responsibilities of the Subcommittee include, but are not limited to, the following:

   a. Review and approval of normal, abnormal and emergency operating and maintenance procedures and records.

   b. Review and approval of proposed tests and experiments utilizing the reactor facilities in accordance with Paragraph 6.8 of these specifications.

   c. Review and approval of proposed changes to the facility systems or equipment, procedures, and operations.

   d. Determination of whether a proposed change, test, or experiment would constitute an unreviewed safety question requiring a change to the Technical Specifications or facility license.

e. Review of all violations of the Technical Specifications and NRC Regulations, and significant violations of internal rules or procedures, with recommendations for corrective action to prevent recurrence.

f. Review of the qualifications and competency of the operating organization to assure retention of staff quality.

3. The Reactor Safety Subcommittee shall be composed of at least five members, one of whom shall be the Radiation Safety Officer or his designee and another of whom shall be the Reactor Supervisor or his designee. The Subcommittee shall be proficient in all areas of reactor operation and reactor safety. The membership of the Subcommittee shall include at least two senior scientific staff members, and the chairman will not have line responsibility for operation of the reactor.

4. The Subcommittee shall have a written charter defining such matters as the authority of the Subcommittee, the subjects within its purview, and other such administrative provisions as are required for effective functioning of the Subcommittee. Minutes of all meetings of the Subcommittee shall be kept.

5. A quorum of the Subcommittee shall consist of not less than a majority of the full Subcommittee and shall include the

Radiation Safety Officer or his designee, and the Reactor Supervisor or his designee.

6. The Subcommittee shall meet at least quarterly.

## 6.3 OPERATING PROCEDURES

Written procedures, reviewed and approved by the Reactor Safety Subcommittee shall be in effect and followed for the following items. The procedures shall be adequate to assure the safe operation of the reactor, but should not preclude the use of independent judgment and action should the situation require such.

1. Startup, operation, and shutdown of the reactor.
2. Installation or removal of fuel elements, control rods, experiments and experimental facilities.
3. Actions to be taken to correct specific and potential malfunctions of systems or components, including responses to alarms, suspected primary coolant system leaks, and abnormal reactivity changes.
4. Emergency conditions involving potential or actual release of radioactivity, including provisions for evacuation, re-entry, recovery, and medical support.
5. Maintenance procedures which could have an effect on reactor safety.
6. Periodic surveillance of reactor instrumentation and safety systems, area monitors and continuous air monitors.
7. Civil disturbance on or near campus.

8. Radiation control, for which procedures shall be maintained and available to all operations personnel.
9. Receipt, inspection, and storage of new fuel elements.
10. Handling and storage of irradiated fuel elements.

Substantive changes to the above procedures shall be made only with the approval of the Reactor Safety Subcommittee. Temporary changes to the procedures that do not change their original intent may be made by the Reactor Supervisor. Temporary changes to procedures shall be documented and subsequently reviewed by the Reactor Safety Subcommittee.

## 6.4 ACTION TO BE TAKEN IN THE EVENT OF AN ABNORMAL OCCURRENCE

In the event of an abnormal occurrence:

1. The Reactor Supervisor or his designee shall be notified promptly and corrective action shall be taken immediately to place the facility in a safe condition until the causes of the abnormal occurrence are determined and corrected.
2. The Reactor Supervisor or his designee shall report the occurrence to the Reactor Safety Subcommittee. The report shall include an analysis of the cause of the occurrence, corrective actions taken, and recommendations for appropriate action to prevent or reduce the probability of a repetition of the occurrence.
3. The Reactor Safety Subcommittee shall review the report and the corrective actions taken.

4. Notification shall be made to the NRC in accordance with Paragraph 6.6 of these specifications.

## 6.5 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED

In the event a Safety Limit has been exceeded:

1. The reactor shall be shut down and reactor operation shall not be resumed until authorization is obtained from the NRC.
2. Immediate notification shall be made to the NRC in accordance with paragraph 6.6 of these specifications and to the Director of the Radiation Laboratory.
3. A prompt report shall be prepared by the Reactor Supervisor or his designee. The report shall include a complete analysis of the causes of the event and the extent of possible damage together with recommendations to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safety Subcommittee for review and appropriate action, and a suitable similar report shall be submitted to the NRC in accordance with Paragraph 6.6 of these specifications and in support of a request for authorization for resumption of operations.

## 6.6 REPORTING REQUIREMENTS

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

1. Within 24 hours, a report by telephone or telegraph to NRC Region I Administrator of:
   a. Any accidental release of radioactivity to unrestricted areas above permissible limits, whether or not the release resulted in property damage, personal injury or exposure.
   b. Any significant variation of measured values from a corresponding predicted or previously measured value of safety related operating characteristics occurring during operation of the reactor.
   c. Any abnormal occurrences as defined in Paragraph 1.1 of these specifications.
   d. Any violation of a Safety Limit.
2. A written report within 14 days in the event of an abnormal occurrence, as defined in Section 1.1. The report shall:
   a. Describe, analyze, and evaluate safety implications;
   b. Outline the measures taken to assure that the cause of the condition is determined;
   c. Indicate the corrective action including any changes made to the procedures and to the quality assurance program taken to prevent repetition of the occurrence and of similar occurrences involving similar components or systems;

d. Evaluate the safety implication of the incident in light of the cumulative experience obtained from the record of previous failure and malfunctions of similar systems and components.

3. Unusual Events.

A written report shall be forwarded within 30 days in the event of:

a. Discovery of any substantial errors in the transient or accident analyses or in the methods used for such analyses, as described in the safety analysis or in the bases for the technical specifications;

b. Discovery of any substantial variance from performance specifications contained in the technical specifications and safety analysis.

c. Discovery of any condition involving a possible single failure which, for a system designed against assumed failures, could result in a loss of the capability of the system to perform its safety function.

4. An annual report shall be submitted in writing within 60 days following the 30th of June of each year. The report shall include the following information:

a. A narrative summary of operating experience (including experiments performed) and of changes in facility design, performance characteristics and operating procedures related to reactor safety occurring during the reporting period, as well as results of surveillance tests and inspections.

b. Tabulation showing the energy generated by the reactor (in megawatt days), the number of hours the reactor was critical, and the cumulative total energy output since initial criticality.

c. The number of emergency shutdowns and inadvertent scrams, including the reasons therefore.

d. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safe operation of the reactor, and the reasons for any corrective maintenance required.

e. A description of each change to the facility or procedures, tests, and experiments carried out under the conditions of Section 50.59 of 10 CFR 50 including a summary of the safety evaluation of each.

f. A description of any environmental surveys performed outside the facility.

g. A summary of radiation exposures received by facility personnel and visitors, including the dates and times of significant exposures, and a summary of the results of radiation and contamination surveys performed within the facility.

h. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge.

Liquid Waste (Summarized on a monthly basis)

(1) Total gross beta radioactivity released (in curies) during the reporting period.

(2) Total radioactivity released (in curies) for specific nuclides, if the gross beta radioactivity exceeds $3 \times 10^{-6}$ $\mu Ci/cm^3$ at point of release, during the reporting period.

(3) Average concentration ($\mu Ci/cm^3$) of release as diluted by sewage system flow of $2.7 \times 10^8$ $cm^3/day$.

Gaseous Waste (Summarized on a monthly basis)

(1) Radioactivity discharged during the reported period (in curies) for: a) gases, b) particulates with half lives greater than eight days.

(2) The MPC used and the estimated activity (in curies) discharged during the reported period,

by nuclide, based on representative isotopic analysis.

Solid Waste (Summarized on a monthly basis)

(1) The total amount of solid waste packaged (in cubic feet).

(2) The total activity and type of activity involved (in curies).

(3) The dates of shipment and disposition (if shipped off-site).

## 6.7 PLANT OPERATING RECORDS

In addition to the requirements of applicable regulations and in no way substituting therefore, records and logs of the following items, as a minimum, shall be kept in a manner convenient for review and shall be retained as indicated:

1. Records to be retained for a period of at least five years:
   a. Reactor operations;
   b. Principal maintenance activities;
   c. Experiments performed including aspects of the experiments which could affect the safety of reactor operation or have radiological safety implications;
   d. Abnormal occurrences; and
   e. Equipment and component surveillance activities.
2. Records to be retained for the life of the facility:
   a. Gaseous and liquid radioactive effluents released to the environs;
   b. Off-site environmental monitoring surveys;
   c. Facility radiation and monitoring surveys;
   d. Personnel radiation exposures;
   e. Fuel inventories and transfers;
   f. Changes to procedures, systems, components, and equipment;
   g. Updated, "as-built" drawings of the facility; and
   h. Minutes of the Reactor Safety Subcommittee meetings.

## 6.8 APPROVAL OF EXPERIMENTS

1. All proposed experiments using the reactor shall be evaluated by the experimenter and a staff member who has been approved by the Reactor Safety Subcommittee. The evaluation shall be reviewed by the Reactor Supervisor and the Radiation Safety Officer to ensure compliance with the provisions of the facility license, these Technical Specifications, and 10 CFR 20. If the experiment complies with the above provisions, it shall be submitted by the Reactor Supervisor to the Reactor Safety Subcommittee for approval if it is a new experiment, as indicated in 4. below. The experimenter evaluation shall include:
   a. The reactivity worth of the experiment;
   b. The integrity of the experiment, including the effect changes in. temperature, pressure, chemical composition, or radiolytic decomposition;
   c. Any physical or chemical interaction that could occur with the reactor components;
   d. Any radiation hazard that may result from the activation of materials or from external beams; and
   e. An estimate of the amount of radioactive materials produced.
2. Prior to performing any new reactor experiment, an evaluation of the experiment shall be made by the Reactor Safety Subcommittee. The Subcommittee evaluation shall consider:

a. The purpose of the experiment;

b. The effect of the experiment on reactor operation and the possibility and consequences of failure of some aspect of the experiment, including, where significant, chemical reactions, physical integrity, design life, proper cooling interaction with core components, and reactivity effects;

c. Whether or not the experiment, by virtue of its nature and/or design, includes an unreviewed safety question or constitutes a significant threat to the integrity of the core, the integrity of the reactor, or to the safety of personnel; and

d. A procedure for the performance of the experiment. A favorable Subcommittee evaluation will not lead to direct failure of any reactor component or of other experiments. An experiment shall not be conducted until a favorable evaluation indicated in writing is rendered by the Reactor Safety Subcommittee.

3. In evaluating experiments, the following assumptions shall be used for the purpose of determining that failure of the experiment would not cause the appropriate limits of 10 CFR 20 to be exceeded:

   a. If the possibility exists that airborne concentrations of radioactive gases or aerosols may be released within the containment building, 100% of the gases or aerosols will escape;

b. If the effluent exhausts through a filter installation designed for greater than 90% efficiency for 0.3 micron particles, at least 10% of gases or aerosols will escape; and

c. For a material whose boiling point is above 130°F and where vapors formed by boiling this material could escape only through a volume of water above the core, at least 10% of these vapors will escape.

4. An experiment that has had prior Subcommittee approval and has been performed safely shall be a routine experiment and requires only the approval of the Reactor Supervisor or his designee and the Radiation Safety Officer or his designee to be repeated. An experiment that represents a minor variation from a routine experiment not involving safety considerations of a different kind nor of a magnitude greater than a routine experiment shall be considered the equivalent of a routine experiment and may be approved for the Subcommittee by agreement of the Reactor Supervisor or his designee and the Radiation Safety Officer or his designee.

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

# 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-125

## UNIVERSITY OF MASSACHUSETTS LOWELL

## DOCKET NO. 50-223

### 1.0 INTRODUCTION

Section 50.64 of Title 10 of the Code of Federal Regulations (10 CFR 50.64) requires licensed research and test non-power reactors to convert from the use of high-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel, unless specifically exempted. The University of Massachusetts Lowell (the licensee) has proposed to convert the fuel in the University of Massachusetts Lowell Reactor (UMLR) from HEU to LEU. In a letter of May 21, 1993, the licensee submitted a supplement to its existing final safety analysis report (FSAR) (September 1973, as amended) describing the changes needed to convert to LEU fuel. A copy of the proposed technical specifications (TSs) needed to operate with LEU fuel was also submitted for NRC's review and approval. Additional information and clarifications to the submittal of May 21, 1993, were submitted by letter dated March 17, 1994, May 16, 1997, and June 6, 1997.

### 2.0 EVALUATION

#### 2.1 General Facility Description

The UMLR is licensed to operate at thermal power levels not to exceed 1 megawatt thermal (MW(t)) when cooled by forced convection at a nominal flow of 1,600 gallons per minute and a pool water level of greater than 24.25 feet above core centerline. The primary coolant is cooled by a heat exchanger and the heat transferred to the secondary coolant system. The secondary coolant system rejects the heat to the atmosphere through a cooling tower.

The UMLR may operate at power levels of 0.1 MW(t) or less when cooled by natural convection with the pool water level greater than 24.25 feet above core centerline. In addition, the UMLR may operate at power levels of 1 kilowatt thermal (KW(t)) or less when cooled by natural convection with a pool water level of greater than 2.25 feet above core centerline.

## 2.2 Fuel Construction and Geometry

The HEU fuel elements used at the UMLR consist of 2 aluminum side plates and 18 equally spaced flat fueled plates of typical materials test reactor design. The uranium in the HEU fuel meat is enriched to about 93 percent uranium-235. Each plate contains approximately 7.5 grams of uranium-235. The outer dimensions of the HEU fuel elements are 7.62 cm by 7.772 cm. Each fuel plate is 7.046-cm wide by 63.5-cm high. The fuel meat is 5.461-cm wide by 60.96-cm high, with a fuel meat thickness of approximately 0.305 mm and a clad thickness of about 0.610 mm.

The LEU fuel elements will be of similar design with essentially the same outer dimensions as the HEU fuel elements, but will contain 16 fuel plates with a dummy aluminum plate at each end and 2 aluminum side plates. Each of these plates will consist of uranium silicide dispersed in aluminum ($U_3Si_2$-Al) and completely clad in aluminum alloy. In the LEU plates, the fuel meat will be about 0.510-mm thick, and the clad will be about 0.380-mm thick. The uranium in the LEU fuel meat is enriched to less than 20 percent uranium-235. Each plate contains about 12.5 grams of uranium-235.

The overall width of the LEU fuel plate will be about 7.14 cm, compared to 7.046 cm for the HEU fuel, and the width of the active fuel will be approximately 6.1 cm maximum, compared to approximately 5.46 cm for HEU. The length of an active LEU fuel plate will be 59.69 cm versus 60.96 cm for HEU.

The Argonne National Laboratory (ANL) developed these fuel element plates for conversion to LEU fuel at the U.S. non-power reactors. Fuel element plates of a design very similar to the UMLR design were tested in the Oak Ridge Research Reactor under extreme operational and hostile environmental conditions for most non-power reactors, including the UMLR, and performed acceptably. The NRC staff reviewed and approved the 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 the LEU conversion at the UMLR are similar to those of the fuel tested and evaluated in NUREG-1313 and are consistent with those previously accepted for other non-power reactors. Therefore, the staff finds the fuel construction and geometry acceptable.

## 2.3 Core Configuration

The current HEU core consists of 25 full fuel assemblies and one partial fuel assembly containing about half the fuel loading of a standard assembly.

The fuel assemblies are in a 5 x 6 array, with the four corners filled with graphite reflectors. The core is moderated and cooled by light water. Reactor control is maintained by four control blades and one regulating rod. Two control blades are located between rows B and C, and two control blades are located between rows E and F. The positions of the control blades will remain the same for the LEU core. The regulating rod for the current HEU core is located in position D9.

The proposed LEU core, or reference core, consists of 20 full fuel assemblies in a 5 x 5 array, with the four corners occupied by radiation baskets and the center core position (D5) occupied by a nonfuel aluminum-graphite neutron flux trap. Actual core configurations can range from 16 to 28 assemblies, including 2 half-loaded partial assemblies. The LEU core is moderated and cooled by light water. Since the core can operate with fewer or more fuel assemblies than the 20-assembly reference core used for all safety analysis calculations, the licensee has agreed to add TS 5.2.3 requiring an analysis to establish that no limiting safety system settings (LSSSs) need to be changed to keep safety limits from being violated during the transients anticipated in the May 1993 FSAR supplement.

In the May 1993 FSAR supplement (Section 2.2, p. 10), the licensee stated that the central aluminum tubes in all but five radiation baskets in the proposed reference core will have to be blocked to ensure adequate flow through the fueled assemblies. The licensee has added this requirement in TS 3.1.3. Since adequate flow through the fuel assemblies for all core configurations is essential, the analysis required by TS 5.2.3 should include flow rate through fuel assemblies.

Reactor control is maintained by the four control blades in the same locations as in the HEU core. Because the LEU core is smaller, the licensee proposes to move the regulating rod from position D9 to position D8 next to the fuel. The licensee estimates that the regulating rod, if left in position D9, would be worth a few hundredths of a percent of reactivity instead of the few tenths of a percent needed to efficiently and safely control reactor power manually and automatically; in all cases, the regulating rod worth is required by TS 3.1.7 to be less than $beta_{eff}$. The design of the regulating rod will not change. However, the licensee will place a 3-inch offset bend in the lift-shaft between the regulating rod and the rod drive and add a support bracket to the suspension bridge gridwork below the offset. The licensee has committed to preoperational testing of the proposed regulating rod design before loading LEU fuel. The staff finds the core configuration proposed by the licensee acceptable.

## 2.4 Fuel Storage

The licensee has analyzed the fresh and spent fuel arrangements in the UMLR pool and has determined that the existing manufactured fuel holding racks will hold all LEU new fuel and all HEU currently on hand. The racks are sectioned into compartments with a 0.5-inch aluminum wall; thus, the stored elements are separated by a minimum of 0.5 inch of aluminum. Calculations performed by ANL[1] indicate that with a 225-gram of uranium-235 fuel element loading (versus 200 grams of uranium-235 in the UMLR fuel) and an element separation

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[1]R. B. Pond and J. E. Matos, "Nuclear Criticality Assessment of LEU and HEU Fuel Elements Storage," Proceedings of the International Meeting on Reduced Enrichment for Research and Test Reactors (JAERI-M,84-073), Tokai, Japan (May 1984).

of 1.766 cm (essentially the same as in the UMLR), the water-reflected infinite array had a $K_{eff} = 0.715$. The method of storing fuel proposed by the licensee is acceptable because the mechanical design of the HEU and the LEU is similar and because the proposed TSs require the licensee to store fuel only in existing storage racks located in the UMLR pool. During the "Low Water" mode of operation the licensee (during a telephone discussion on December 22, 1993) has recognized the need to be particularly mindful of the potential for radiation exposure to personnel from stored spent fuel. The staff finds the proposed simultaneous storage of HEU and LEU fuels acceptable.

## 2.5 Critical Operating Masses of Uranium-235

Each HEU fuel element contains about 135 grams of uranium-235, and each LEU fuel element contains about 200 grams of uranium-235. The current core configuration has a uranium-235 operating mass of approximately 3,510 grams of HEU. For the LEU reference core, the loading would be approximately 4,000 grams of uranium-235. This LEU loading for an operating core is reasonable for the intended purpose of the reactor and is consistent with other LEU conversions when configuration and power level are considered. Therefore, fuel loading for the proposed LEU cores is acceptable.

## 2.6 Basic Nuclear Parameters

Calculated nuclear input parameters for reactivity calculations, such as prompt neutron lifetime and the effective delayed neutron fraction ($B_{eff}$), changed as expected from the HEU to LEU cores. The reference core prompt neutron lifetime decreased from 75.6 microseconds to 64.5 microseconds, primarily because of increased leakage from the proposed smaller core. The $B_{eff}$ for the LEU core is calculated to be $7.8 \times 10^{-3}$ versus $7.69 \times 10^{-3}$ for the HEU core. This slight difference is as expected and is similar to that calculated for other conversions to LEU.

## 2.7 Excess Reactivity

The licensee calculated the amount of excess reactivity needed to operate with an LEU core to be a maximum of 4.7 percent (the current TS limit is 4.7 percent). This excess would compensate for the various operational losses in reactivity from burnup, xenon, temperature, and experiments. Because of safety considerations, the operational excess reactivity is always limited by the requirement to maintain the TS minimum shutdown margin (UMLR TS 3.1.1). 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 and Regulating Rod Worths

The proposed LEU core uses the same four control blades that have been used for the HEU core. The licensee evaluated the worth of these control blades using the ANL standard neutron kinetics models and computer codes and verified that the control blades will acceptably meet the TS requirements for the proposed LEU cores. The licensee will verify the worth of these control blades as a part of the LEU reactor core startup testing program.

The licensee proposes to change the position of the nonscrammable regulating rod for use with the LEU core. Calculations showed that the reactivity worth of the regulating rod in the current position (D9) would decrease from a few tenths percent to a few hundredths percent delta k/k. To ensure an adequate reactivity for the regulating rod, the licensee proposes to move the rod to the D8 position, which is a grid position adjacent to the proposed LEU core. This movement would result in an estimated worth for the regulating rod of several tenths percent delta k/k. This relocation acceptably ensures control for normal plant operations. Since the licensee will determine the worth of the four control rods and the regulating rod during the startup testing of the LEU reactor core and verify that the control rods and regulating rod perform as designed, the staff finds the control rod worth design acceptable.

## 2.9 Shutdown Margin

The staff requires reasonable assurance that the UMLR can be shut down from any operating condition, even if the one control blade of maximum worth and the nonscrammable regulating rod are in their fully withdrawn position. With the calculated control blade worths, proposed LEU core configurations, and ANL standard neutron kinetics methods, the calculated UMLR shutdown margin would not be lower than about 3.4-percent delta k/k with the control and regulating rod worths given in the May 1993 FSAR supplement and the maximum allowable excess reactivity allowed by the TS are assumed. This shutdown margin is considerably greater than the 2.7-percent delta k/k required by TS. Since the shutdown margin will be verified during startup testing with the LEU reactor core to ensure that the TS limit is met, the staff finds the proposed TS shutdown margin for the LEU core acceptable.

## 2.10 Core Power Characteristics

The licensee analyzed the core thermal power characteristics for the proposed LEU core. These analyses used standard computer programs and ANL standard nuclear kinetics models. The analysis calculated a maximum heat flux of $2.75 \times 10^4$ BTU/h - $ft^2$ for the LEU core versus $2.17 \times 10^4$ BTU/hr - $ft^2$ for the HEU core. This heat flux is consistent for the proposed LEU core design.

The core analysis also indicates a peak axial power ratio of approximately 3.91 with control blades fully withdrawn (leading to an assumed chopped cosine flux distribution) and approximately 3.38 with control blades 15 inches withdrawn. The staff reviewed the analysis inputs, methods, and results (including Tables 3.1, 3.2, and 3.4 of the UMLR FSAR supplement of May 21, 1993) and has concluded that the licensee acceptably determined the power conditions to be used in analyzing thermal-hydraulic conditions, as well as transient and accident conditions (discussed in Section 2.14).

## 2.11 Thermal-Hydraulics

The licensee performed a thermal-hydraulic analysis of the LEU elements and core. Using ANL codes, 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. In addition, the licensee incorporated hot channel factors (HCFs) into its calculations to account for fuel and assembly design tolerances and uncertainties. The analyses demonstrated that the LEU fuel elements and core would be cooled and maintained within acceptable limits for forced or natural convection cooling conditions during normal operation. That is, under both forced convection and natural convection coolant flow and associated power conditions, calculated thermal-hydraulic conditions for the LEU fuel, would maintain a substantial margin before the onset of nucleate boiling (ONB).

For operation under natural convection flow, the licensee calculated situations in which the control blades were fully withdrawn (leading to a chopped cosine flux distribution) and situations in which the control blades were withdrawn 15 inches. Comparing the results to establish the limiting situation, the licensee found that the most restrictive case is with the control blades withdrawn to 15 inches. In this case, ONB is calculated to occur at about 429 kW for the HEU core and at about 335 kW for the reference LEU core. Therefore, for the reference LEU core, 335 kW is taken as the safety limit for natural convection operation. This is more than three times the licensed operating limit (100 kW).

At power levels greater than 100 kW, the UMLR TSs require the reactor to be operated in the forced convection mode, in which the flow rate is about 1,600 gallons per minute (gpm) downward through the core. This flow rate is the same for both the HEU and LEU cores. With the nominal flow rate of 1,600 gpm, the UMLR calculated that ONB would occur at greater than 3.5 MW, or more than 3.5 times the licensed operating power (1 MW) for the LEU core. To determine operational conservatism, the licensee also calculated maximum fuel clad temperature for normal forced convection operation with LEU, giving a calculated maximum fuel clad temperature (including HCFs) of approximately 67 °C. The ONB clad temperature at 1 MW is about 118 °C.

The licensee also analyzed the LEU fuel element and core thermal-hydraulic design for off-normal conditions. The analysis for the natural convection conditions demonstrated that the 0.1-MW limit on power operations for natural convection gives significant margin before ONB is reached; that is, under natural convection conditions, ONB would occur at greater than 0.335 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 limiting safety system settings (LSSSs) of 1.25 MW (which is about 125 percent of the normal full-power level) and 1,170 respectively. With this lower flow rate trip, the licensee calculated that ONB would occur at more than 2.5 MW, or more than twice the high-power trip setpoint.

These analyses of the thermal hydraulic performance of the LEU core are acceptable, as are the corresponding TS changes proposed (evaluated in Section 3.0 of this safety evaluation (SE)).

2.12 Reactivity Feedback Coefficients

The licensee asked ANL to calculate the temperature coefficient of reactivity and the void coefficient of reactivity for the HEU and LEU cores. ANL also calculated the coolant density and the Doppler effect in broadening the neutron capture resonances of the more abundant uranium-238 present in the LEU fuel. ANL's results are as follows: (1) the temperature coefficient for the LEU core would be $-0.48 \times 10^{-4}$ delta k/k/°C and for the HEU core $-0.37 \times 10^{-4}$, (2) the coolant density coefficient for the LEU core would be $-0.46 \times 10^{-4}$ delta k/k/°C and for the HEU core $-0.31 \times 10^{-4}$, and (3) the Doppler coefficient for the LEU core would be $-0.15 \times 10^{-4}$ delta k/k/°C and for the HEU core zero. The total would be $-1.09 \times 10^{-4}$ for the LEU core and $-0.68 \times 10^{-4}$ delta k/k °C for the HEU core.

ANL calculated that the average void coefficient was approximately $-2.26 \times 10^{-3}$ delta k/k/percent void for LEU fuel and $-1.59 \times 10^{-3}$ for HEU fuel.

All reactivity feedback coefficients for the LEU core are calculated to be larger than those for the HEU core and are more effective in leading to reactor stability. The licensee will also verify the feedback coefficients to be negative and greater than those of the HEU core. Therefore, the licensee has acceptably addressed the reactivity coefficients for conversion to the LEU fuel.

2.13 Fission Product Containment and Inventory

The cladding is the primary barrier to fission product release for both the HEU and LEU fuels. The cladding and other aspects of fuel construction are described in Section 2.2 of this SE. ANL developed the LEU fuels and extensively tested them under more extreme operational conditions than the UMLR fuels will experience. In these tests, the performance of the proposed LEU fuel was excellent, comparable to that of HEU fuel. Furthermore, use of similar fuel elements and plates in other non-power reactors has demonstrated the excellent fission product retention capability of the LEU fuel.

The total inventory of fission products from operating the proposed LEU core at 1 MW(t) will not differ significantly from that for the HEU core. Therefore, the previously assumed release of fission products remains valid because it conservatively assumed the release of all fission products from a single fuel plate. However, the fission product inventory in each LEU fuel element and plate will be greater than 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. In calculating the release of radioactive materials, the licensee incorporated recent models and assumptions regarding release and dispersal of materials, including dispersion models in Regulatory Guide 1.145. These analyses and evaluations demonstrate that the consequences of the fission product release do not exceed previously established acceptance criteria.

In evaluating the fission product containment and inventory of the LEU fuel, the licensee and the staff have found no new or significant safety considerations. Therefore, the proposed operations with LEU fuel are acceptable for containing the expected fission product inventory.

2.14 Potential Accident Scenarios

For the conversion from HEU to LEU, the staff evaluated the refueling accident, the step increase in reactivity event, the continuous withdrawal of a control blade, and the cold water insertion event. All of the other transients, such as a failed fueled experiment, a partial or total loss of water, binding of control blades, release of coolant header gates during operation, and cross flow during forced convection operation, were reviewed on the basis of the information in the September 1973 FSAR, as amended, and should not be affected by the HEU to LEU conversion.

2.14.1 Refueling Accident

The most severe accident that can be envisioned from the conversion from HEU to LEU is the substitution of a fuel assembly for the central flux trap element. The licensee calculates that this error would add approximately 3 percent in reactivity. Since the control blades have more shutdown reactivity than 3 percent, the UMLR would be subcritical until the control blades were moved to a raised position. The reactor would be critical with the control blades in a much lower position than normal, thereby alerting operations personnel to a potential problem. They would be expected to react in accordance with established procedures.

The licensee calculated that dropping a fuel assembly on top of the core would add less than 0.5 percent reactivity, again well within the shutdown capability of the control blades.

The staff finds the analysis of the refueling accident acceptable.

2.14.2 Step Increase in Reactivity

UMLR is limited by its TSs to experiments that can add no more than 0.5-percent reactivity under any condition. ANL analyzed the instantaneous insertion of 0.5-percent reactivity while the UMLR was operating at 1 MW. ANL concluded that the maximum peak power would be about 2.8 MW, which is well below the safety limit of about 3.8 MW under these operating conditions. The power transient with the LEU core is less than with the HEU core because of the larger Doppler coefficient. Since the consequence of this accident is less with LEU fuel than with HEU fuel, the staff finds this analysis acceptable.

2.14.3 Continuous Withdrawal of a Control Blade

The UMLR TSs limit the control blade reactivity addition rate to 0.025-percent delta k/k/second; however, the licensee has assumed a continuous withdrawal of 0.035-percent delta k/k/second. On the basis of ANL's calculations, the high-power trips would limit the reactor power to about 1.3 MW, which is significantly below the safety limit (3.8 MW). Therefore, the staff finds this accident analysis acceptable.

2.14.4 Cold Water Insertion

The licensee has postulated that the maximum primary temperature decrease would be about 21.7°C if the primary temperature was 43.3°C and the secondary flow of 0°C water went from no flow to 1,600 gpm. The corresponding reactivity change using the calculated negative temperature coefficient for the LEU core of $-1.09 \times 10^{-4}$ delta k/k/°C would be +0.24-percent delta k/k. Since this change is less than the 0.5 percent reactivity step input analyzed in Section 2.14.2, the staff finds the licensee's analysis of this accident acceptable.

2.15 Reactor Startup Testing

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

3.0 CHANGES TO TECHNICAL SPECIFICATIONS

3.1 Administrative Format and Editorial Changes

The licensee proposed to fix typographical errors and misspellings and make other minor editorial and administrative TS changes. The staff has reviewed all of these changes and has determined that they do not alter the meaning or intent of the TSs; therefore, the staff finds them acceptable as included in the revised TSs of this amendment (Amendment No. 12).

3.2 TS 2.1.1.1 "Safety Limits under forced convection flow"

The licensee proposed to change the safety limits for forced flow and add Figure 2.1, which will replace existing TSs 2.1.1.1 and 2.1.1.2. Figure 2.1 represents the relationship between reactor thermal power and reactor coolant flow rate and indicates the flow necessary to prevent ONB for a given power level. Since Figure 2.1 is a more comprehensive way of indicating this safety limit, the staff finds this TS change acceptable.

3.3 TS 2.1.2.1 "Safety Limits in the natural convection mode"

The licensee proposed to reduce the specified safety limit for maximum power while the reactor is in the natural convection mode. The licensee proposed this change for the LEU core after calculating the ONB to be approximately 335 kW when hot channel factors (HCFs) and the 15-inch-control-blade-withdrawn case are considered.

The staff reviewed this requested change, considers it more conservative, and accepts it. This safety limit will be more than three times the licensed power level for the UMLR in the natural convection mode.

3.4 TS 2.2.1 "Limiting Safety System Setting (LSSS) in the forced convection mode"

The licensee proposed to change the LSSS for the coolant flow rate from 1,250 gpm to 1,170 gpm for the LEU core in the forced convection mode. The staff analyzed this change and found it to be appropriate to ensure that automatic protective action would occur to prevent safety limits from being exceeded. With the reactor trip set at a flow rate of 1,170 gpm and a flow coastdown from the nominal operating flow of 1,600 gpm, the control blades would be fully inserted by the time the flow reaches approximately 630 gpm. This 630-gpm flow corresponds to a power level for the ONB of approximately 1.6 MW, which is about 60-percent higher than licensed power; therefore, the safety limit for forced convection would not be violated. In addition, dropping the control blades would reduce the reactor power from 1 MW to approximately 130 kW, well below the safety limit for natural convection (335 kW). Therefore, the staff finds this change to the LSSS for coolant flow to be acceptable.

3.5 TS 3.1.3 "Reactivity"

The licensee proposed to add a TS requirement that all but five of the peripheral radiation baskets contain flow-restricting devices. Further, the licensee proposed to clarify this TS to state that it does not apply in the natural convection mode of operation at low power. The staff finds these additional TS requirements acceptable since they assure that the minimum required quantity of primary coolant flow will be directed to the heat-producing fuel elements in the LEU core.

3.6 TS 3.3 "Coolant Flow Rate"

The licensee proposes to change the reactor automatic scram LSSS coolant flow rate from 1,250 gpm to 1,170 gpm. The staff reviewed this change in Section 3.4 of this safety evaluation and finds it acceptable.

3.7 TS 4.2.6 "Reactor Safety System (RSS) Surveillance"

The licensee proposed to add a surveillance requirement that any RSS instrument channel replacement "must undergo a channel calibration before routine operation of the reactor after channel installation." This addition defines good practice and can only make the replaced channel more reliable. The staff finds this addition to the TSs acceptable.

3.8 TS 5.0 "Design Features"

This specification states the characteristics and physical descriptions of the reactor fuel and reactor core. Changes to reflect the LEU fuel and core are discussed in the following sections.

3.8.1 TS 5.1 "Reactor Fuel"

The licensee proposed to revise this specification to define the design features of the proposed LEU fuel. The changes are as follows:

(1) The LEU fuel matrix will be $U_3Si_2$-Al instead of the HEU fuel matrix alloy Al-$U_3O_8$.

(2) The uranium-235 enrichment will be approximately 19.75 percent (not to exceed 20 percent) for the LEU instead of the approximately 93-percent enrichment for the HEU.

(3) The cold, clean LEU fuel elements will be 18 plates, with 16 containing approximately 200 grams of uranium-235 and 2 outside aluminum plates, instead of the HEU elements, which had consisted of 18 plates containing approximately 135 grams of uranium-235.

These changes are required by the LEU fuel design, which has been previously discussed in this SE and has been demonstrated by ANL to be acceptable; therefore, the staff finds these changes acceptable.

3.8.2 TS 5.2 "Reactor Core"

The licensee proposed to revise this specification to describe the proposed design features of the LEU core. The LEU reference core should consist of 20 standard elements, with the central location filled with a graphite water, aluminum-clad flux trap instead of the HEU reference core, which consists of 26 standard HEU fuel elements. The proposed LEU cores may contain from 16 to

28 LEU elements and may contain 2 half-loaded elements, in contrast to the HEU core, which could contain 23 to 30 HEU elements and 2 half-loaded elements. These proposed changes to the reactor core design features have been discussed in Section 2.3 of this SE. The staff finds these changes acceptable.

3.8.3 TS 5.4 "Fuel Storage"

The licensee proposed to revise this TS to allow reactor fuel element storage of LEU fuel in a geometrical configuration so the $K_{eff}$ is less than 0.85 under quiescent flooding with water instead of a $K_{eff}$ of 0.80 for HEU fuel. This change was discussed in Section 2.4 of this SE and is required by the uranium-235 loading of the LEU fuels. ANL's calculations indicate that $K_{eff}$ of an infinite array of LEU fuel loaded with 225 grams of uranium-235 would be less than 0.80. Because the UMLR fuel is within the envelope of the ANL calculations, the staff finds this change acceptable.

## 4.0 CONCLUSION

The NRC staff reviewed and evaluated all of the operational and safety factors affected by the use of LEU fuel in place of HEU fuel in the UMLR. 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.

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Date: July 31, 1997