UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of OHIO STATE UNIVERSITY ENGINEERING EXPERIMENT STATION Columbus, Ohio 43210	Docket No. 50-150 Facility Operating License No. R-75 Amendment No. 12

ORDER MODIFYING LICENSE

I

Ohio State University (licensee or OSU) is the holder of Facility Operating License No. R-75 (License) issued on October 24, 1961, by the U.S. Nuclear Regulatory Commission (Commission). The license authorizes operation of the OSU Training and Research Reactor (facility) at a power level of up to 10 kilowatts (thermal). The facility is located in Columbus, Ohio, on property owned by the OSU, approximately two miles west of the main campus. The mailing address is Ohio State University, Engineering Experiment Station, 142 Hitchcock Hall, Columbus, Ohio 43210.

II

On February 25, 1986, the Commission promulgated a final rule in 10 CFR 50.64 of its regulations limiting the use of high-enriched uranium (HEU) fuel in domestic research and test reactors (non-power reactors) (see 51 FR 6514). The rule, which became effective on March 27, 1986 requires that a licensee of an existing non-power reactor replace HEU fuel at its facility with low-enriched uranium (LEU) fuel acceptable to the Commission: (1) unless the Commission has determined that the reactor has a unique purpose and (2) contingent upon Federal Government funding for conversion-related costs. The rule is intended to promote the common defense and security by reducing the risk of theft and diversion of HEU fuel used in non-power reactors and the adverse consequences to public health and safety and the environment from such theft or diversion.

10 CFR 50.64(b)(2)(i) and (ii) require that a licensee of a non-power reactor: (1) r t initiate acquisition of additional HEU fuel, if LEU fuel acceptable to the Commission for that reactor is available when it proposes that acquisition, and (2) replace all HEU fuel in its possession with available LEU fuel acceptable to the Commission for that reactor, in accordance with a schedule determined pursuant to 10 CFR 50.64(c)(2).

10 CFR 50.64(c)(2)(i) of the rule, among other things, requires each licensee of a non-power reactor, authorized to possess and to use HEU fuel, to develop and to submit to the Director of the Office of Nuclear Reactor Regulation (Director) by March 27, 1987, and at 12-month intervals thereafter, a written proposal (proposal) for meeting the rule's requirements.

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

10 CFR 50.64(c)(2)(iii) requires the licensee to include in its proposal, to the extent required to effect conversion, all necessary changes to the license, to the facility, and to the licensee's procedures (all three types of changes hereafter called modifications). This paragraph also requires the licensee to provide supporting safety analyses so as to meet the schedule established for conversion.

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

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

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

10 CFR 2.714 sets out the requirements for a person whose interest may be affected by any proceeding to initiate a hearing or to participate as a party.

III

On October 7, 1987, the Director received the licensee's proposal, including its proposed modifications, supporting safety analyses and schedule for conversion. The conversion consists of replacement of high-enriched with low-enriched uranium fuel elements. The fuel elements contain MTR-type fuel plates with the fuel meat in the form of uranium silicides dispersed in an aluminum matrix. The enrichment is less than 20% in the U-235 isotope. The Licensing Conditions and Technical Specification charges needed to amend the facility license are included in the attachment to this Order. On the bases of the licensee's submittals and the requirements of 10 CFR 50.64, I have made a determination that the public health and safety and the common defense and security require the licensee to convert from the use of HEU to LEU fuel pursuant to the modifications set forth in the attachment in accordance with the schedule set out below.

IV

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

On the later date of either receipt of low-enriched uranium fuel elements by the licensee, or 30 days following the date of publication of this Order in the Federal Register, Facility Operating License No. R-75 is modified by amending the License Conditions and Technical Specifications as stated in the Attachment to this Order.

V

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

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland
this 27 day of September 1988

Enclosure:
As stated

ATTACHMENT TO ORDER

MODIFYING FACILITY OPERATING LICENSE NO. R-75

A. License Conditions Revised and Added By This Order

No. 2.B. Pursuant to the Act and 10 CFR Part 70, "Special Nuclear Material," to receive, possess and use in connection with operation of the reactor 80 grams of plutonium contained in encapsulated plutonium-beryllium sources, up to 10 grams of contained Uranium-235 enriched to 93% in the form of fission chamber linings, foil targets and other research applications and up to 5.2 kilograms of contained Uranium-235 at enrichments equal to or less than 20%.

No. 2.D. Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to possess, but not to use, a maximum of 4.6 kilograms of contained uranium at greater than 20% enrichment until the existing inventory of high enriched uranium is removed from the facility.

No. 3.B. The Technical Specifications contained in Appendix A, as revised through Amendment 12, are hereby incorporated in the license. The licensee shall operate the reactor in accordance with these Technical Specifications.

No. 3.F. Physical Security Plan

The licensee shall fully implement and maintain in effect all provisions of the physical security plan currently approved by the Commission and all amendments and revisions made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). This plan, which contains information withheld from public disclosure under 10 CFR 2.790, is entitled "Ohio State University Nuclear Reactor Laboratory Physical Security Plan for Protection of Special Nuclear Material of Moderate or Low Strategic Significance," submitted by letter dated March 14, 1988, as amended by letter dated April 13, 1988.

B. Technical Specifications Revised by This Order

The Technical Specifications (TS) have been revised to conform with the American National Standards Institute (ANSI), American National Standard 15.1-1982, for the Development of Technical Specifications for Research Reactors. The paragraph numbers in the TS have, therefore, been completely renumbered to follow the ANSI Standard. The changes made to accommodate the LEU fuel appear in the following revised paragraphs.

2.0 SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTINGS (LSSS)

2.1 Safety Limit

Applicability

This specification applies to the melting temperature of the aluminum fuel cladding.

Objective

The objective is to assure that the integrity of the fuel cladding is maintained.

Specification

The reactor fuel temperature shall be less than 550°C.

Bases

The melting temperature of aluminum is 660°C (1220°F). The blister threshold temperature for U_3Si_2 dispersion fuel has been measured as approximately 550°C (ANL/RERTR/TM-10, October 1987). Because the objective of this specification is to prevent release of fission products, any fuel whose maximum temperature reaches 550°C is to be treated as though the safety limit has been reached until shown otherwise.

2.2 Limiting Safety System Settings

Applicability

This specification applies to the following items associated with core thermodynamics:

(2) Reactor Coolant Inlet Temperature

Objective

To assure that the fuel cladding integrity is maintained.

Specification

(2) Reactor safety systems settings shall initiate automatic protective action so that core inlet water temperature shall not exceed 35°C.

Bases

The criterion for this safety limit is established as the fuel integrity. If the temperature of the clad is maintained below that for ONB then cladding integrity is maintained. This is the case for a power level of 15 kw and a core inlet temperature of 35°C (normal inlet temperature is ~ 20-25°C. The maximum credible accident analysis is provided in Section 8.4.3 of the Safety Analysis Report. Because any operation above 10kw is not authorized, possible increase to 15kw would necessarily be transient, and would add only small additional energy to the fuel.

5.0 DESIGN FEATURES

5.3 Reactor Core and Fuel

Up to 30 positions on the core grid plate are available for use as fuel element positions. Control rod fuel elements occupy four of these positions and one is reserved for the Central Irradiation Facility flux trap. Several arrangements for the cold, clean, critical core have been investigated. Approximately 16 standard fuel elements in addition to the control rod fuel elements will be required. Partial elements, core plugs, and graphite elements may be utilized in various combinations to achieve the proper K excess.

The reactor fuel is the DOE Standard uranium-silicide (U_3Si_2) with a U-235 enrichment of less than 20%. It is flat plate fuel with a "meat" thickness of 0.020" and aluminum cladding of 0.015". Standard fuel elements have a total of 16 fueled plates and two outer pure aluminum plates. The control rod fuel elements have eight of the inner fuel plates removed to allow the control rods to enter. Pure aluminum guide plates are on the inside of this gap. The outer two plates for each control rod assembly are fueled. Partial elements are also available with 25%, 40%, 50%, and 60% of the nominal loading of a standard element. These partial fuel elements are prefabricated by the vendor with fixed numbers of plates.

(1) References: NRC NUREG 1313
ANL/RERTR/TM-10
ANL/RERTR/TM-11

5.4 Fuel Storage

The fuel storage pit, located below the floor of the reactor pool and at the end opposite from the core, shall be flooded with water whenever fuel is present and shall be capable of storing a complete core loading. When fully loaded with fuel and filled with water, K_{eff} shall not exceed 0.90, and natural convective cooling shall ensure that no fuel temperatures reach a point at which ONB is possible.

The two fuel storage racks located in the Bulk Shielding Facility storage pool shall each:

(a) Contain no more than 16 fuel elements spaced on a pitch of at least 6 inches in a 2 by 8 matrix.

(b) Be placed no closer than 24 inches in any direction from each other or any other fuel storage facility.

(c) Have a K_{eff} less than 0.90 when fully loaded with fuel and flooded with water.

ATTACHMENT TO LICENSE AMENDMENT NO. 12

FACILITY OPERATING LICENSE NO. R-75

DOCKET NO. 50-150

The Technical Specifications have been revised to conform with the American National Standards Institute (ANSI) American National Standard 15.1-1982, for the Development of Technical Specifications for Research Reactors. The Technical Specifications have therefore, been replaced in their entirety.

APPENDIX A

TO

FACILITY OPERATING LICENSE NO. R-75

Technical Specifications

And Bases For The

Ohio State University

Pool-Type Nuclear Reactor

Columbus, Ohio

Docket No. 50-150

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* These sections do not apply to the operation of the facility at 10Kw. The licensee is authorized to operate the reactor at steady state power levels up to a maximum of 10 kilowatts thermal.

1. INTRODUCTION

1.1 Scope

This document constitutes the Technical Specifications for Facility License No. R-75 and supersedes all prior Technical Specifications. Included are the "Specifications" and the "Bases" for the Technical Specifications. These bases, which provide the technical support for the individual technical specifications, are included for information purposes only. They are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

This document was written to be in conformance with ANSI/ANS-15.1-1982. The content of the Technical Specifications includes: Definitions, Safety Limits, Limiting Safety System Settings, Limiting Conditions for Operation, Surveillance Requirements, Design Features, and Administrative Controls.

1.2 Application

1.2.1 Purpose

These Technical Specifications have been written specifically for The Ohio State University Research Reactor (OSURR).

The Technical Specifications represent the agreement between the licensee and the U.S. Nuclear Regulatory Commission on administrative controls, equipment availability, and operational parameters.

Specifications are limits and equipment requirements for safe reactor operation and for dealing with abnormal situations. They are typically derived from the Safety Analysis Report (SAR). These specifications represent a comprehensive envelope for safe operation. Only those operational parameters and equipment requirements directly related to preserving that safe envelope are listed.

1.2.2 Format

The format of this document is in general accordance with ANSI/ANS-15.1-1982.

1.3 **Definitions**

Administrative Controls - those organizational and procedural requirements established by the Commission and/or the facility management.

ALARA - as low as is reasonably achievable.

Channel - the combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter.

Channel Calibration - an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the measured parameter. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip settings, and shall be deemed to include a channel test.

Channel Check - 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.

Channel Test - the introduction of a signal into the channel for verification that it is operable.

Cold Clean Core - when the core is at ambient temperature and the reactivity worth of xenon is negligible.

Commission - the U.S. Nuclear Regulatory Commission (or NRC).

Confinement - a closure on the overall facility which controls the movement of air into it and out of it through a controlled path.

Containment - a testable enclosure which can support a defined pressure differential and which is normally closed.

Control Rod - a device fabricated from neutron absorbing material which is used to establish neutron flux changes.

Control Rod Fuel Element - a fuel element capable of holding a control rod.

Controls - mechanisms used to regulate the operation of the reactor

Core - the general arrangement of fuel elements and control rods.

Critical - when the effective multiplication factor (k_{eff}) of the reactor is equal to unity.

Direct Supervision - in visual and audible contact.

Excess Reactivity - that amount of reactivity that would exist if all control rods were removed from the core.

Exclusion Area - that area around the reactor building in which the licensee has the authority to determine all activities as per 10CFR100.3.

Experiment - any operation, or any apparatus, device, or material installed in or near the core or which could conceivably have a reactivity effect on the core and which itself is not a core component or experimental facility, intended to investigate non-routine reactor parameters or radiation interaction parameters of materials.

Experimental Facility - any structure or device associated with the reactor that is intended to guide, orient, position, manipulate, or otherwise facilitate completion of experiments.

Explosive Material - any material that is given an Identification of Reactivity (Stability) index of 2, 3, or 4 by the National Fire Protection Association in its publication 704-M, Identification System for Fire Hazards of Materials, or is enumerated in the Handbook for Laboratory Safety published by the Chemical Rubber Company (1967).

Facility - the Reactor Building including offices and laboratories.

Fueled Experiment - any experiment that contains U-235 or U-233 or Pu-239, not including the normal reactor fuel elements.

Licensee - The Ohio State University.

Limiting Conditions for Operation (LCO) - the lowest functional capability or performance levels of equipment required for safe operation of the facility. LCO are administratively established constraints on equipment and operational characteristics.

Limiting Safety System Settings (LSSS) - settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

Measured Value - the value of a parameter as it appears on the output of a channel.

Movable Experiment - one for which it is intended that all or part of the experiment may be moved in relation to the core while the reactor is operating.

Nuclear Regulatory Commission - (NRC).

Onset of Nucleate Boiling - (ONB).

Operable - a component or system which is capable of performing its intended functions in a normal manner.

Operating - a component or system which is performing its intended function.

Protective Action - the initiation of a signal or the operation of equipment within the reactor safety system in response to a variable or condition of the reactor facility having reached a specified limit.

Reactivity Limits - those limits imposed on reactor core excess reactivity based upon a reference core condition.

Reactivity Worth of an Experiment - the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter an experiment's position or configuration.

Reactor - the combination of core, permanently installed experimental facilities, control rods, and connected control instrumentation.

Reactor Operating - whenever the reactor is not secured or shutdown.

Reactor Operations Committee - (ROC).

Reactor Operator (RO) - an individual who is licensed to manipulate the controls of the reactor in accordance with 10CFR55.

Reactor Safety Systems - 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.

Reactor Secured - whenever (1) all shim/safety rods are fully inserted, (2) the console key is in the OFF position and is removed from the lock, and (3) no in-core work is in progress involving fuel or experiments or maintenance of the core structure, control rods, or control rod drive mechanisms.

Reactor Shutdown - when the reactor is subcritical by at least 1% delta k/k in the cold clean core condition.

Regulating Rod - a low reactivity-worth control rod used primarily to maintain an intended power level. Its position may be varied either by manual control or by the automatic servo-controller.

Reportable Occurrence - any of the conditions described in Section 6.5.2 of these specifications.

Restricted Area - the Reactor Building to which access is controlled for purposes of protection of individuals from exposure to radiation and radioactive materials.

Safety Analysis Report - (SAR), October 7, 1987.

Safety Channel - a measuring or protective channel in the reactor safety system.

Safety Limits (SL) - limits on important process variables which are found to be necessary to reasonably protect the integrity of certain physical barriers which guard against the uncontrolled release of radioactivity.

Scram - the rapid insertion of the shim/safety rods into the reactor for the purpose of quickly shutting down the reactor.

Scram Time - the elapsed time between reaching a limiting safety system setting and the time when a control rod is fully inserted.

Secured Experiment - any experiment, experimental facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected from the normal environment of the experiment or by forces which can result from credible malfunctions.

Senior Reactor Operator (SRO) - an individual who is licensed to direct the activities of reactor operators. Such an individual may also operate the controls of the reactor pursuant to 10CFR55.

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, which is neither a requirement nor a recommendation.

Shim/Safety Rods - high-reactivity worth control rods used primarily to provide coarse reactor control. They are connected electromagnetically to their drive mechanisms and have scram capabilities.

Shutdown Margin - the shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems with the most reactive shim/safety rod and the regulating rod in the most reactive position (fully withdrawn) and that the reactor will remain subcritical without further operator action.

Standard Fuel Element - an element to be used or stored in the core, fuel storage pit or other approved area, but not a control rod element.

Startup Source - a spontaneous source of neutrons which is used to provide a channel check of the startup (fission chamber) channel, and provide neutrons for subcritical multiplication during reactor startup.

Surveillance Time Intervals - The average over any extended period for each surveillance time interval shall be the normal surveillance time, e.g. for the two year interval the average shall be two years.

two-year (interval not to exceed 30 months).
annually (interval not to exceed 15 months).
semiannually (interval not to exceed 7-1/2 months).
quarterly (interval not to exceed 4 months).
monthly (interval not to exceed 6 weeks).
weekly (interval not to exceed 10 days).
daily (shall be done during the same working day).

Any extension of these intervals shall be occasional and for a valid reason and shall not affect the average as defined.

True Value - the actual value of a parameter.

Unscheduled Shutdowns - any unplanned shutdown of the reactor caused by actuation of the reactor safety systems, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation. They do not include those shutdowns resulting from expected testing operations, or planned shutdowns, whether initiated by controlled insertion of control rods or planned manual scrams.

2.0 SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTINGS (LSSS)

2.1 Safety Limit

Applicability: This specification applies to the melting temperature of the aluminum fuel cladding.

Objective: The objective is to assure that the integrity of the fuel cladding is maintained.

Specification: The reactor fuel temperature shall be less than $550^{o}C$.

Bases: The melting temperature of aluminum is $660^{o}C$ ($1220^{o}F$). The blister threshold temperature for U_3Si_2 dispersion fuel has been measured as approximately $550^{o}C$. (ANL/RERTR/TM-10, October 1987). Because the objective of this specification is to prevent release of fission products, any fuel whose maximum temperature reaches $550^{o}C$. is to be treated as though the safety limit has been reached until shown otherwise.

2.2 Limiting Safety System Settings

Applicability: This specification applies to the following items associated with core thermodynamics:

(1) Reactor Thermal Power Level and

(2) Reactor Coolant Inlet Temperature.

Objective: To assure that the fuel cladding integrity is maintained.

Specification:

(1) Reactor safety systems settings shall initiate automatic protective action so that reactor thermal power level shall not exceed 15 kw (150% of full power).

(2) Reactor safety systems settings shall initiate automatic protective action so that core inlet water temperature shall not exceed $35^{o}C$.

Bases: The criterion for this safety limit is established as the fuel integrity. If the temperature of the clad is maintained below that for ONB then cladding integrity is maintained. This is the case for a power level of 15 kw and a core inlet temperature of $35^{o}C$ (normal inlet temperature is ~ $20\text{-}25^{o}C$). The maximum credible accident analysis is provided in Section 8.4.3 of the Safety Analysis Report. Because any operation above 10KW is not authorized, possible increase to 15kw would necessarily be transient, and would add only small additional energy to the fuel.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 **Reactor Core Parameters**

3.1.1 Reactivity

Applicability: These specifications apply to the reactivity condition of the reactor and the reactivity worths of the shim/safety rods and regulating rod under any operating conditions.

Objective: To ensure safe shutdown of the reactor and that the safety limits are not exceeded.

Specification: The reactor shall be operated only if the following conditions exist:

(1) The reactor core shall be loaded so that the excess reactivity, including the effects of installed experiments does not exceed 1.5% delta k/k under any operating condition.

(2) The minimum shutdown margin under any operating condition with the maximum worth shim/safety rod and the regulating rod full out shall be no less than 1.0% delta k/k.

(3) The total reactivity worth of the regulating rod shall be less than 0.7% delta k/k.

(4) All core grid positions internal to the active fuel boundary shall be occupied by a standard, control, regulating rod, instrumented, or blank fuel element; or by an experimental facility.

(5) The moderator temperature coefficient shall be negative and shall have a minimum absolute reactivity value of at least $2 \times 10^{-5}/^{o}C$ across the active core at all normal operating temperatures.

(6) The moderator void coefficient of reactivity shall have a minimum value of at least $2.8 \times 10^{-3}/1\%$ void across the active core.

Bases:

(1) The maximum allowed excess reactivity of 1.5% delta k/k provides sufficient reactivity to accommodate fuel burnup, xenon buildup, experiments, control requirements, and fuel and moderator temperature feedback (Section 4.2 of the SAR). Also, calculations show that this excess reactivity assures that the maximum temperature of the surface of the cladding will be well below the melting temperature of aluminum (SAR August 4, 1965, Amendment 5 and SAR May 6, 1988, Attachment B).

(2) The minimum shutdown margin ensures that the reactor can be shutdown from any operating condition and remain shutdown after cooling and xenon decay even with the highest worth rod and the regulating rod fully withdrawn.

(3) 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 cannot result in a prompt critical condition.

(4) The requirement that all grid positions be filled during reactor operation assures that the volume flow rate of primary coolant which bypasses the heat producing elements will be within the range specified in Section 4.8 of the SAR. Furthermore, the possibility of accidentally dropping an object into a grid position and causing increase of reactivity is precluded.

(5) A negative moderator temperature coefficient of reactivity assures that any moderator temperature rise will cause a decrease in reactivity. The U_3Si_2 fuel also has a significant negative temperature coefficient of reactivity due to the Doppler broadening of neutron capture resonances in ^{238}U, but no credit is taken for this effect in our Safety analyses.

(6) A negative void coefficient of reactivity helps provide reactor stability in the event of moderator displacement by experimental devices or other means.

3.2 **Reactor Control and Safety System**

3.2.1 Control Rod Drop Times

Applicability: This specification applies to the time from the receipt of a safety signal to the time it takes for a shim/safety rod to drop from fully withdrawn to fully inserted.

Objective: To ensure that the reactor can be shutdown within a specified period of time.

Specification: The reactor will not be operated unless the drop time of each of the three shim/safety rods is less than 600 msec.

Bases: Control rod drop times as specified ensure that the safety limit will not be exceeded in a short period transient.

The analysis for this is given in Section 4.3.3 of the SAR.

3.2.2 Maximum Reactivity Insertion Rate

Applicability: This applies to the maximum positive reactivity insertion rate by the most reactive shim/safety rod and the regulating rod simultaneously.

Objective: To ensure the reactor is operated safely and the safety limit is not exceeded due to a short period.

Specification: The reactor will not be operated unless the maximum reactivity insertion rate is less than 0.02% delta k/k per second.

Basis: This maximum reactivity insertion rate assures that the Safety Limit will not be exceeded during a startup accident due to a short period generated by a continuous linear reactivity insertion.

3.2.3 Minimum Number of Scram Channels

Applicability: This specification applies to the reactor safety system channels.
Objective: To stipulate the minimum number of reactor safety system channels that shall be operable to ensure the Safety Limits are not exceeded by ensuring the reactor can be shutdown at all times.

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

Reactor Safety System Component	Minimum Required	Function
1. Core H_2O Inlet Temp.	1	Slow scram if temp. $\geq 35^{\circ}C$
2. Reactor Thermal power level (Safety Channels)	2	Fast scram if thermal power = 15 kw, as indicated on calibrated ionization chamber channels.
3. Reactor Period	1	Fast scram if period $\leq$ 1 sec
4. Pool Water Level	1	Slow scram if pool level $\leq$ 20 feet (15 feet above core)
5. Switches a. Magnet Power Key "On" b. Startup Cal-Use in "Use" c. Period Generator Switch "Off" d. LOG-N Amp Calibrate Switch "Norm" e. LOG-Period Amp Calibrate Switch "Norm" f. Effluent Monitor Compressor "On"	6	Slow scram if any one switch is not properly set at the position indicated in quotes. (Also prohibits startup)
6. Recorders a. LOG-N b. Linear Level c. Start-Up Channel d. Period e. Effluent Monitor	5	Slow scram if power is lost to any one of the listed recorders
7. Manual Scrams a. Control Room Console b. Pool Top Catwalk c. BSF Catwalk d. Rabbit/BP Area e. Thermal Column/BP Area	5	Slow scram upon activation of any one manual scram switch
8. Compensated Ion Chambers	2	Slow scram if voltage drops below operational specifications

Reactor Safety System Component/Channel	Minimum Required	Function
9. Safety Set Points On Recorders		Slow scram if associated recorder values are exceeded
a. Period		$\leq$ 5 sec
b. Linear Level		$\geq$ 120%
c. Linear Level		Servo deviation $\geq$ Set point (nominal 10%)
d. Start-Up Channel		$\leq$ 2 cts/sec (may be bypassed if $K_{eff} < 0.9$)
10. Safety System	2	Slow scram in case of a safety amp fault or if system is discontinuous
11. Backup Shutdown Mechanisms	3	Rod drop will occur for any control rod which has excess magnet current $\geq$ 60 ma

Bases:

1. Assures safety limit is not exceeded
2. Assures safety limit is not exceeded
3. Assures safety limit is not exceeded
4. Assures there is enough primary coolant for natural convection cooling
5. Assures nuclear instrumentation is in proper mode for operation
6. Assures information is available for observation by the reactor operator during operation, and is recorded if required as a record of reactor operations
7. Assures that the reactor can be shut down by the reactor operator in the control room or at other locations near experimental facilities if deemed necessary by other reactor staff
8. Assures shutdown if nuclear instrumentation fails
9. Assures backup shutdown capability from short period or high power level. Assures shutdown if servo operation varies too greatly. Assures shutdown if count rate is too low to provide meaningful startup information. The startup interlock may be bypassed if K_{eff} is < .9
10. Assures all components of the safety system are installed and operational
11. Assures that any control rod exhibiting excess magnet current will be released and fall to the bottom due to gravity

3.3 Coolant System

3.3.1 Pump Requirements

(This section deleted for 10KW operation)

Applicability: This specification applies to the operation of pumps for both the primary and secondary coolant loops.

Objective: To ensure that both pumps are functioning whenever the reactor is operated above 120 kw.

Specification: The reactor will not be operated above 120 kw unless both the primary and secondary coolant pumps are activated and there is flow in the primary coolant loop.

Bases: Having both pumps operating and flow in the primary loop will ensure there is adequate cooling of the primary coolant so the Safety Limit is not exceeded.

3.3.2 Coolant Level

Applicability: This specification applies to the height of the water in the Reactor Pool above the core.

Objective: To ensure there is adequate primary coolant in the Reactor Pool and sufficient biological shielding above the core.

Specification: The reactor shall not be operated unless there is 20 feet of water in the reactor pool and 15 feet of water above the core.

Bases: With the pool full of water to a level of 20 feet there is adequate primary coolant for natural convection cooling. With 15 feet of water above the core there is sufficient shielding at the licensed power level. Section 7.1.1.4 of the SAR discusses this shielding.

3.3.3 Water Chemistry Requirements

Applicability: This specification applies to the purity of the primary coolant water.

Objective: To minimize corrosion of the cladding on the fuel elements, and to reduce the probability of neutron activation of ions in the water.

Specification:

(1) The conductivity of the pool water shall not exceed the limit of 2.0 µmho/cm.

(2) The pH of the pool water shall not exceed 8.0.

Bases: Operation in accordance with these specification ensures aluminum corrosion is within acceptable limits, and that the concentration of dissolved impurities that could be activated by neutron irradiation remains within acceptable limits.

3.3.4 <u>Leak or Loss of Coolant Detection</u>

Applicability: This specification applies to the capability of detecting and preventing the loss of primary coolant.

Objective: To ensure there is adequate primary coolant in the Reactor Pool and sufficient biological shielding above the core when the reactor is operating.

Specification: There shall be a system to detect if pool water level drops below 20 feet (15 feet above the core).

Bases: The same system that functions to scram the reactor on low pool level will also be used as the detection system for this specification. Design criteria of the cooling system to prevent large losses of pool water due to siphoning are discussed in Section 3.2.2.1 of the SAR.

3.3.5 <u>Secondary Coolant Activity Limits</u>

(This section deleted for 10KW operation)

Applicability: This specification applies to the buildup of radioactive materials in the secondary coolant system.

Objective: To ensure there is a level low enough so as not to exceed 10CFR20 limits if secondary coolant is released to the sanitary sewer system.

Specification: The secondary coolant system shall be monitored for the buildup of radioactive materials.

Basis: The basis for this specification is to ensure releases are legal and consistent with the ALARA principal.

3.4 Containment Isolation

Applicability: This specification applies to the capability of isolating the reactor building from the unrestricted area outside.

Objective: To prevent the exposure of the public to airborne radioactivity exceeding the limits of 10CFR20, and the ALARA principle.

Specification: The reactor shall not be operated unless the following are operable:

(1) Ventilation fan

(2) Reactor Building bay door

(3) Reactor Building front and rear personnel doors
(4) Office doors and windows

Bases: By having the capability to isolate the Reactor Building, the release of airborne radioactive material may be contained.

3.5 Ventilation Systems

Applicability: This specification applies to all heating, ventilating, and air conditioning systems that exhaust building air to the outside environment.

Objective: To provide for normal ventilation and the reduction of airborne radioactivity within the reactor building during normal reactor operation and to provide a way to turn off all vent systems quickly in order to isolate the building for emergencies.

Specification:

(1) An exhaust fan with a capacity of at least 1000 cfm shall be on whenever the reactor is operating.

(2) This fan, as well as all other heating, ventilating, and air conditioning systems shall have the capability to be shut off from a single switch in the control room.

Bases: In the unlikely event of a release of fission products or other airborne radioactivity, the ventilation system will reduce radioactivity inside the reactor building or be able to be isolated. An analysis of fission product release is found in section 8.4.4 of the SAR.

3.6 Radiation Monitoring Systems and Radioactive Effluents

3.6.1 Radiation Monitoring

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

Objective: To assure that monitoring equipment is available to evaluate radiation levels in restricted and unrestricted areas and to be consistent with ALARA.

Specification:

(1) When the reactor is operating, the building gaseous effluent monitor shall be on and have a readout and alarm in the control room. It may be used in either the "normal" mode or "sniffer" mode.

(2) (This Section Deleted for 10KW Operation)

When the reactor is operating and the rabbit experimental facility is used, the rabbit monitoring system shall be on and have a readout and alarm in the control room.

(3) When the reactor is operating, the following Area Radiation Monitors (ARMs) shall be on and have both local and control room readouts and alarms.

a. Pool Top
b. Primary Cooling System
c. Beam Port/Rabbit Area
d. Thermal Column Area

(4) Portable survey instrumentation shall be available whenever the reactor is operating to measure beta-gamma exposure rates and neutron dose rates.

(5) Portable instruments, surveys, or analyses may be substituted for the instruments in the above sections (3.6.1.1, 3.6.1.2, or 3.6.1.3) for periods up to 48 hours. Read-out and alarms from these temporary instruments shall be reported to the reactor operator on duty at least once per hour.

Bases:

(1) The gaseous effluent monitor will detect Ar-41 levels in the reactor building. During "normal" mode operation it will sample and monitor air just before it is released from the reactor building. (SAR 6.3.1) During "sniffer" mode of operation it may be used for short periods to monitor in and around experimental facilities to determine local Ar-41 levels.

(2) (This Section Deleted for 10KW Operation)

The rabbit stack monitor is used with the rabbit since the rabbit system uses air as its transport mechanism and Ar-41 production takes place. This monitor will provide warning if Ar-41 levels being released in the building are too high (SAR 6.3.2 and 6.3.4.3)

(3) The ARMs provide a continuing evaluation of the radiation levels within the Reactor Building (SAR 3.7) and provide a warning if levels are higher than anticipated.

(4) The availability of survey meters enables the Reactor Staff to independently confirm radiation levels throughout the building.

(5) In the event of instrument failure short term substitutions will enable the safe continued operation of the Reactor.

3.6.2 <u>Radioactive Effluents</u>

Applicability: This specification applies to the monitoring of radioactive effluents from the facility.

Objectives:

(1) To ensure that liquid radioactive releases are safe and legal.

(2) To assure that the release of Ar-41 beyond the site boundary does not result in exposures above MPC.

Specifications:

(1) The release rate for radioactive liquids beyond the site boundary shall not exceed the limits as specified in 10CFR20.

(2) The release of Ar-41 on the North side of the facility shall not exceed MPC when averaged over one year or 10 x MPC when averaged over one day.

Bases:

(1) The basis for this specification is found in Section 6.2 of the Safety Analysis Report.

(2) The basis for this specification is found in Section 6.3 of the Safety Analysis Report.

3.7 **Experiments**

3.7.1 Reactivity Limits

Applicability: This specification applies to experiments to be installed in or near 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:

(1) The value of the reactivity worth of any single secured experiment shall not exceed 0.7% delta k/k.

(2) The value of the reactivity worth of any single movable experiment shall not exceed 0.4% delta k/k.

(3) The value of the reactivity worth of all movable experiments shall not exceed 0.6% delta k/k.

(4) The value of the reactivity worth of experiments having moving parts shall be designed to have an insertion rate less than 0.05% delta k/k per second.

(5) The value of the reactivity worth of any movable experiment that may be oscillated shall have a reactivity change of less than 0.05% delta k/k.

(6) The total reactivity worth of all experiments shall not be greater than 0.7% delta k/k.

Bases:

(1) The bases for specifications 1, 2, 3, and 6 are found in Section 8.4.3.2 of the SAR which evaluates a step insertion of reactivity from an experiment.

(2) The bases for specifications 4 and 5 allows for certain reactor kinetics experiments to be performed but still limits the rate of change of reactivity insertions to safe levels.

3.7.2 Design and Materials

Specification:

(1) No experiment shall be installed that could shadow the nuclear instrumentation, interfere with the insertion of a control rod, or credibly result in fuel element damage.

(2) All materials to be irradiated in the reactor shall be either corrosion resistant or doubly encapsulated within corrosion resistant containers.

(3) Explosive materials shall not be allowed in experiments, except for neutron radiographic exposures of items performed outside of the core and experimental facilities. The amount of explosive material contained in capsules used for radiographic exposures shall not exceed 5 grains of gunpowder.

Bases:

(1) Specification 1 assures no physical interference with the operation of the reactor detectors, control rods, or physical damage to fuel element will take place.

(2) Limiting corrosive materials in Specification 2, and explosives in Specification 3 reduces the likelihood of damage to reactor components and/or releases of radioactivity resulting from experiment failure.

(3) Limiting explosive materials to neutron radiographic exposures done outside of the core and experimental facilities reduces the likelihood of damage resulting for this experimental failure.

4.0 SURVEILLANCE REQUIREMENTS

4.1 Reactor Core Parameters

4.1.1 Excess Reactivity and Shutdown Margin

Applicability: This specification applies to surveillance requirements for determining the excess reactivity of the reactor core and its shutdown margin.

Objective: To assure that the excess reactivity and shutdown margin limits of the reactor are not exceeded.

Specifications:

(1) Whenever a net change in core configuration, for which the predicted change in reactivity is >.2% delta k/k, involving grid position is made, both excess reactivity and shutdown margin shall be determined.

(2) Both shutdown margin and excess reactivity shall be determined annually.

Bases: A determination of excess reactivity is needed to preclude operating without adequate shutdown margin. Moving a component out of the core and returning it to its same location is not a change in the core configuration and does not require a determination of excess reactivity.

4.2 Reactor Control and Safety Systems

4.2.1 Control Rods

Applicability: This specification applies to the surveillance requirements for the shim safety rods and the regulating rod.

Objective: To assure that all rods are operable.

Specifications:

(1) The reactivity worth of the shim safety rods and regulating rod shall be determined annually and prior to the routine operation of any new core configuration.

(2) Shim safety rod drop and drive times and regulating rod drive time shall be determined annually or after maintenance or modification is completed on a mechanism.

(3) The shim safety rods and regulating rod shall be visually inspected annually, for indication of corrosion, and indication of excessive friction with guides.

Bases: The reactivity worth of the rods is measured to assure the required shutdown margin and reactivity insertion rates are maintained. It also provides a means for determining the reactivity of experiments. Measuring annually will provide corrections for burnup and after core changes assures that altered rod worths will be known prior to continued operations.

The visual inspection of the rods and measurements of drive and drop times are made to assure the rods are capable of performing properly. Verification of operability after maintenance or modification of the control system will ensure proper reinstallation.

4.2.2 Reactor Safety System

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

Objective: To assure the reactor safety system channels will remain operable and prevent safety limits from being exceeded.

Specification:

(1) A channel check of each measuring channel shall be performed daily when the reactor is operating.

(2) A channel test of each measuring channel shall be performed prior to each day's operation, or prior to each operation extending more than one day.

(3) A channel calibration of the reactor power level measuring channels shall be made annually. (Linear Level and LOG-N.)

(4) A channel calibration of the Level and Period Safety Channels shall be made annually. Channel tests are done on these before each day's operation.

(5) A channel calibration of the following shall be made annually

a. Core inlet temperature measuring system
b. Pool water level measuring system
c. Coolant system pumps measuring system
d. Primary coolant flow measuring system

(6) The control room manual scram shall be verified to be operable prior to each day's operation. All other manual scram switches shall be tested annually.

(7) Other scram channels shall be tested/calibrated annually.

(8) Any instrument channel replacement shall be calibrated after installation and before utilization.

(9) Any instrument repair or replacement shall have a channel test prior to reactor operation.

Bases: The daily channel tests and checks will assure that the scram channels are operable. Appropriate annual tests or calibrations will assure that long term functions not tested before daily operation are operable.

4.3 Coolant System

4.3.1 Primary Coolant Water Purity

Applicability: This specification applies to the conductivity of the primary coolant water.

Objective: To assure high quality pool water.

Specification: The conductivity of the pool water shall be measured weekly.

Bases: This assures that changes that might increase the corrosion rate do not occur.

4.3.2 Coolant System Radioactivity.

Applicability: This specification applies to the radioactive material in the primary coolant.

Objective: To identify radionuclides as potential sources of release to the sanitary sewer system.

Specification: Primary coolant shall be analyzed for radioactivity quarterly or before release.

Bases: Radionuclide analysis of the pool water allows for determination of any significant buildup of fission or activiation products.

4.4 Containment

Applicability: This specification applies to the surveillance requirements for building confinement.

Objective: To assure that the building closure capability exists.

Specification: A monthly test shall be made to assure that the building exhaust fan, bay door, front and rear personnel doors, and office doors and windows are operable.

Bases: Monthly surveillance of this equipment will verify that the confinement of the reactor bay can be maintained if needed.

4.5 Ventilation System

Applicability: This specification applies to the surveillance requirements for the building ventilation system.

Objective: To assure that the ventilation system functions satisfactorily.

Specification:

(1) Ventilation fans and closures shall be checked for proper operation on a quarterly basis.

(2) The shutoff switch for all fans and air conditioning systems shall be tested on a quarterly basis.

Bases: This surveillance will assure that during normal operations the airborne radioactivity will be minimized inside the building and that the building can be isolated quickly if necessary to prevent uncontrolled escape of air-borne radioactivity to the unrestricted environment.

4.6 Radiation Monitoring Systems and Radioactive Effluents

4.6.1 Effluent Monitor

Applicability: This specification applies to the surveillance requirement of the effluent monitor.

Objective: To assure the effluent monitor is operational and providing accurate effluent readings.

Specification: The effluent monitor shall have a channel calibration annually and a channel test before each days operation.

Bases: The calibration will assure effluent release estimates are accurate and the test will assure the monitor is operable whenever the reactor is operating.

4.6.2 Rabbit Vent Monitor

(This Section deleted for 10KW Operation)

Applicability: This specification applies to the surveillance requirements of the rabbit vent monitor.

Objective: To assure the monitor is operational and providing meaningful information about effluent releases from the rabbit into the reactor building.

Specification: The monitor shall have a channel calibration annually and a channel test before each day's reactor operation.

Bases: The calibration will assure effluent releases inside the building are accurately estimated and the test will assure the monitor is operable before the rabbit is used.

4.6.3 Area Radiation Monitors (ARMs)

Applicability: This specification applies to the area radiation monitoring equipment.

Objective: To assure that radiation monitoring equipment is operable whenever the reactor is operating.

Specification: A channel test of the ARMs shall be completed before each day's operation and a channel calibration shall be completed annually.

Bases: Calibration annually will insure the required reliability and a check on days when the reactor is operated will detect obvious malfunctions in the system.

4.6.4 Portable Survey Instrumentation

Applicability: This specification applies to the portable survey instrumentation available to measure beta-gamma exposure rates and neutron dose rates.

Objective: To assure that radiation survey instrumentation is operable whenever the reactor is operating.

Specification: Beta-gamma and neutron survey meters shall be tested for operability each day the reactor is to be operated and shall be calibrated annually.

Bases: Tests on days when the reactor is operated will detect obvious detector deficiencies and an annual calibration will assure reliability.

5.0 DESIGN FEATURES

5.1 Site and Facility Description

5.1.1 Facility Location

The reactor and associated equipment is housed in a building at 1298 Kinnear Road, Columbus, Ohio. It is in the area of The Ohio State University Research Center.

5.1.2 Exclusion and Restricted Area

The fence surrounding the Research Center shall describe the exclusion area. The restricted area as defined in 10CFR20 shall consist of the Reactor Building.

5.2 Reactor Coolant System

5.2.1 <u>Primary Coolant Loop</u>

Natural convective cooling is the primary means of heat removal from the core. Water enters the core at the bottom and flows upward through the flow channels in the fuel elements.

5.2.2 <u>Secondary and Tertiary Coolant Loops</u>

(This Section Deleted for 10KW Operation)

The secondary coolant loop removes heat from the primary coolant. The secondary coolant (ethylene glycol and water) passes through two separate heat exchangers to remove heat if necessary. If the outside air temperature is $\leq 78^{\circ}F$ then an outside fan-forced drycooler is sufficient to remove all heat generated at 500 kw. City water flow through the secondary side of an additional heat exchanger makes up the tertiary loop. It provides additional cooling for the secondary coolant.

5.3 Reactor Core and Fuel

Up to 30 positions on the core grid plate are available for use as fuel element positions. Control rod fuel elements occupy four of these positions and one is reserved for the Central Irradiation Facility flux trap. Several arrangements for the cold, clean, critical core have been investigated. Approximately sixteen standard fuel elements in addition to the control rod fuel elements will be required. Partial elements, core plugs, and graphite elements may be utilized in various combinations to achieve the proper K excess.

The reactor fuel is The DOE Standard uranium-silicide (U_3Si_2) with a U-235 enrichment of less than 20%. It is flat plate fuel with a "meat" thickness of 0.020" and aluminum cladding of 0.015". Standard fuel elements have a total of 16 fueled plates and 2 outer pure aluminum plates. The control rod fuel elements have eight of the inner fuel plates removed to allow the control rods to enter. Pure aluminum guide plates are on the inside of this gap. The outer two plates for each control rod assembly are fueled. Partial elements are also available with 25, 40, 50, and 60 percent of the nominal loading of a standard element. These partial fuel elements are prefabricated by the vendor with fixed numbers of plates.

(1) References: NRC NUREG 1313
ANL/RERTR/TM-10
ANL/RERTR/TM-11

5.4 Fuel Storage

The fuel storage pit, located below the floor of the reactor pool and at the end opposite from the core, shall be flooded with water whenever fuel is present and shall be capable of storing a complete core loading. When fully loaded with fuel and filled with water, K_{eff} shall not exceed 0.90, and natural convective cooling shall ensure that no fuel temperatures reach a point at which ONB is possible.

The two fuel storage racks located in the Bulk Shielding Facility storage pool shall each:

(a) Contain no more than 16 fuel elements spaced on a pitch of at least 6 inches in a two by eight matrix.

(b) Be placed no closer than 24 inches in any direction from each other or any other fuel storage facility.

(c) Have a K_{eff} less than 0.90 when fully loaded with fuel and flooded with water.

5.5 Fuel Handling Tools

All tools designed for or capable of removing fuel from core positions or storage rack positions shall be secured when not in use by a system controlled by the supervisor of reactor operations, or the senior reactor operator on duty.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 Structure

The Ohio State University Research Reactor is a part of the College of Engineering administered by the Engineering Experiment Station. The organizational structure is shown in Figure 6.1.

6.1.2 Responsibility

The Director of the Engineering Experiment Station (Level 1) is the contact person for communications between the U.S. Nuclear Regulatory Commission and The Ohio State University.

The Director of the Nuclear Reactor Laboratory (Level 2) will have overall responsibility for the management of the facility.

The Associate Director (or Manager of Reactor Operations) (Level 3) shall be responsible for the day-to-day operation and for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and Technical Specifications. During periods when the Associate Director is absent, his responsibilities are delegated to a Senior Reactor Operator (Level 4).

6.1.3 Staffing

During Reactor Operations:

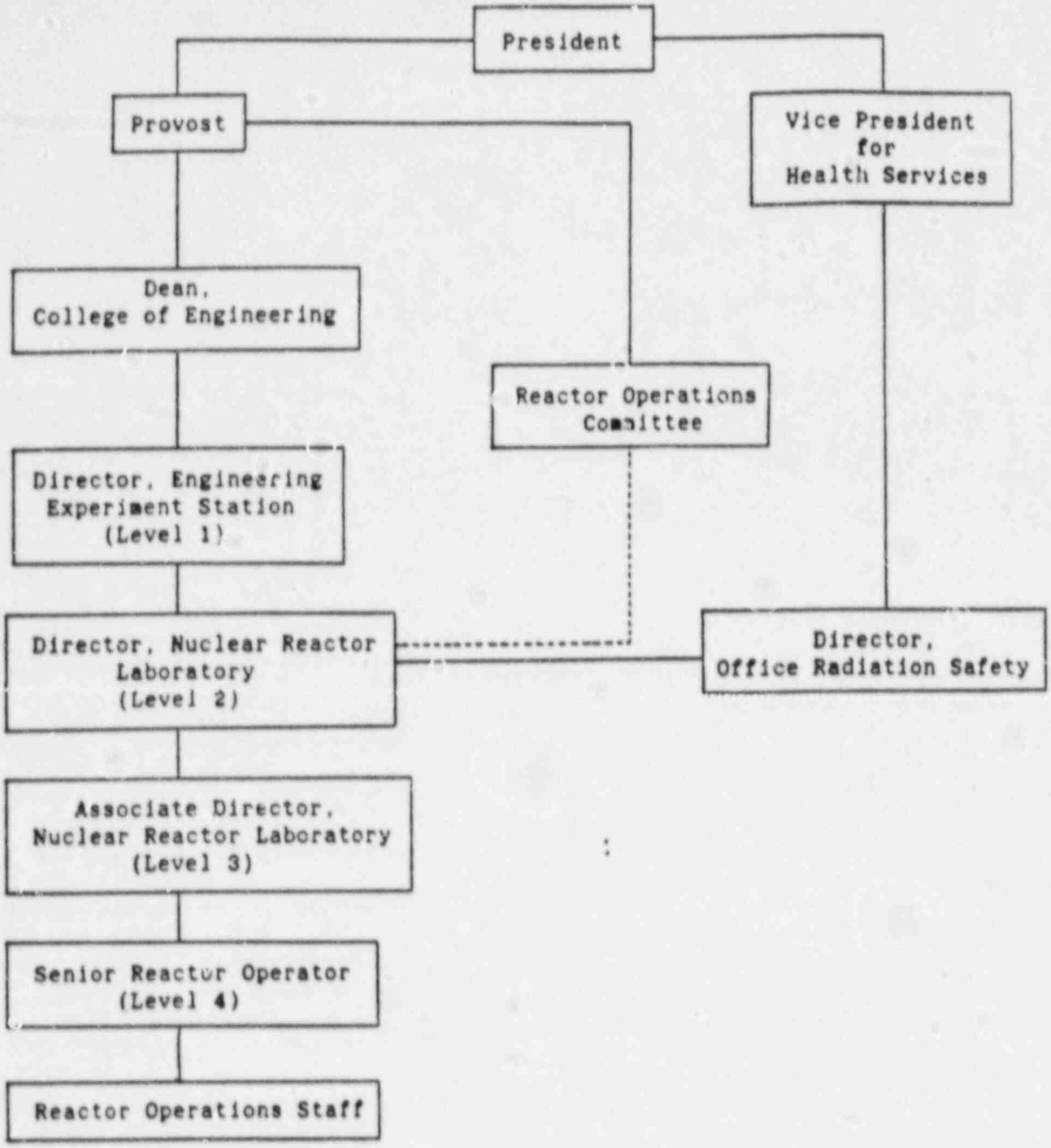
(1) Two or more personnel, at least one of whom is a licensed reactor operator, shall be in the building during all reactor operations. The second shall be capable of following simple written instructions for shutting down the reactor.

(2) At least two licensed operators should be in the building during any extended operations (longer than 60 minutes).

(3) Two persons, one of whom shall be a licensed senior reactor operator, shall be in the building for the first start-up of the day.

(4) Two persons, one of whom shall be a licensed senior reactor operator, shall be in the building during start-up after an unplanned shutdown.

(5) During all operations, a licensed operator shall be in the control room either as console operator or directing the activities of a student operator or trainee.



Solid Lines __________ Paths of Direct Responsibility
Dashed Lines ---------- Paths of Information

Figure 6.1 'Administrative Organization'

(6) A minimum of three people shall be present during fuel handling. One shall be a licensed senior reactor operator, and one shall be at least a licensed reactor operator.

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1977, Sections 4-6.

6.2 Review and Audit

There shall be a Reactor Operations Committee (ROC) which shall review and audit reactor operations to assure the facility is operating in a manner consistent with public safety and within the terms of the facility license. The Committee advises the Director of the NRL, and is responsible to the Provost of The Ohio State University.

6.2.1 Composition and Qualifications of the ROC

Committee members shall be appointed annually by the Provost of The Ohio State University. The Committee shall be composed of at least nine members including ex-officio members. The Director and Associate Director of the Nuclear Reactor Laboratory, and the Director of the Office of Radiation Safety shall be ex-officio voting members of the Committee. The remaining Committee members shall be faculty, staff, and student representatives of The Ohio State University, having professional backgrounds in engineering, physical, biological, or medical sciences, as well as knowledge of and interest in applications of nuclear technology and ionizing radiation.

6.2.2 ROC Meetings

The Committee shall meet at least once each quarter. It should meet during the first two weeks of each calendar quarter. A quorum shall consist of at least 50 percent of the members who are not directly involved in or responsible for facility operations. Ex-officio members shall be counted in the quorum as follows:

(1) The Provost is an ex-officio member. Since the Provost is not appointed as a member of the ROC, the Provost is not required to act as a member, is not counted as a member when counting a quorum, but does have the right to vote.

(2) Ex-officio members who are under the authority of the Provost serve in the same capacity as those who are appointed by the Provost, i.e., they have the right to vote but they are not counted as members when counting a quorum if they are directly involved in or responsible for facility operations.

(3) Ex-officio members, if any, who are not under the authority of the Provost, have the right to vote, but have no obligation to participate. Accordingly, they are not counted as members when counting a quorum.

(4) All ex-officio members hold membership by virtue of their office. They cease to be members when they cease to hold office.

6.2.3 Sub-Committees

The chairperson may appoint a Subcommittee from within the Committee membership to act on behalf of the full committee on those matters which cannot await the regular quarterly meeting. The full Committee shall review the actions taken by the Subcommittee at the next regular meeting.

6.2.4 ROC Review and Approval Function

The responsibilities of the ROC include, but are not limited to the following:

(1) Review and approval of experiments utilizing the reactor facilities

(2) Review and approval of procedures

(3) Review and approval of all proposed changes to the license and technical specifications

(4) Determination of whether a proposed change, new test, or experiment would constitute an unreviewed safety question or require a change in the technical specifications per 10CFR50.59

(5) Review of audit reports

(6) Review of abnormal performance of plant equipment and operating abnormalities having safety significance

(7) Review of unusual occurrences and incidents which are reportable under 10CFR19, 20, 21, and 50, or Section 6.6.4 of this document, and

(8) Review of violations of technical specifications, license, or procedures having safety significance.

Relative to item (1), responsibility for review of experiments on a day-to-day basis shall lie with the Director of the Nuclear Reactor Laboratory or his designee. This day-to-day review shall determine whether a specific experiment has previously been approved in the generic sense by the ROC. A quarterly report of performed experiments shall be provided for ROC review.

Relative to item (2), the NRL Director or his designee shall be responsible for approval of procedures or changes to procedures on a day-to-day basis. He shall provide a summary of all procedure changes to the ROC for their review and approval.

A complete set of minutes of all Committee and Subcommittee meetings, including copies of all documentary material reviewed, and all approvals, disapprovals, and recommendations shall be kept. Minutes or reports of all Committee meetings or Subcommittee meeting should be disseminated to the Committee members prior to the next regularly scheduled meeting, and should be read for approval as the first item on each Agenda. A copy of the minutes, or any reports reviewed, should also be forwarded to the Director of the Engineering Experiment Station in a timely manner.

6.2.5 ROC Audit Function

A three member Subcommittee shall meet annually to perform an audit of NRL operations and records or review the results of an independent audit completed by another qualified agency. At least two individuals on the Audit Subcommittee shall be ROC members. The third may be a staff member from the Reactor Laboratory or another individual appointed by the ROC chairperson. No member shall audit a function that he is responsible for performing. Each person should serve for three consecutive audits, at which time he or she should be replaced by a new member. In this way, each Subcommittee should consist of two holdovers and one new member. The member serving for his or her second audit should be the Audit Subcommittee Chairperson. The following items shall be audited:

(1) Reactor operations for adherence to facility procedures, Technical Specifications, and license requirements

(2) The requalification program for the operating staff,

(3) The facility Emergency Plan and implementing procedures,

(4) The facility Security Plan and implementing procedures, and

(5) The results of actions taken to correct any deficiencies that affect reactor safety, and

(6) Conformance with the ALARA Policy and the effectiveness of radiologic control.

Deficiencies found by the Audit Subcommittee that affect Reactor Safety, shall be reported immediately to the Director of the Engineering Experiment Station. A written report of audit findings should be submitted to the Director of the Engineering Experiment Station and the full Reactor Operations Committee within three months of the audit's completion.

6.3 Procedures

6.3.1 Reactor Operating Procedures

Written procedures, reviewed and approved by the Director and the ROC, shall be in effect and followed. The procedures shall be adequate to assure the safety of the reactor, but should not preclude the use of independent judgement and action should the situation require such. All new procedures and changes to existing procedures shall be documented by the NRL staff and subsequently reviewed by the ROC. At least the following items shall be covered:

(1) Startup, operation, and shutdown of the reactor,

(2) Installation, removal, or movement of fuel elements, control rods, experiments, and experimental facilities,

(3) Actions to be taken to correct specific and foreseen potential malfunctions of systems or components; including responses to alarms, suspected cooling system leaks, and abnormal reactivity changes,

(4) Emergency conditions invovling potential or actual release of radioactivity including provisions for evacuation, re-entry, recovery, and medical support,

(5) Preventive and corrective maintenance procedures for systems which could have an effect on reactor safety,

(6) Periodic surveillance of reactor instrumentation and safety systems, area monitors, and radiation safety equipment,

(7) Implementation of Security, Emergency and Operator training and requalification plans, and

(8) Personnel radiation protection.

6.3.2 Administrative Procedures

Procedures shall also be written and maintained to assure compliance with Federal regulations, the facilitiy license, and committments made to the ROC or other advisory or governing bodies. As a minimum, these procedures shall include:

(1) Audits,

(2) Special Nuclear Material accounting,

(3) Operator requalification,

(4) Record keeping, and

(5) Procedure writing and approval.

6.4 Experiment Review and Approval

6.4.1 Definitions of Experiments

Approved experiments are those which have previously been reviewed and approved by the ROC. They shall be documented and may be included as part of the Procedures Manual. New experiments are those which have not previously been reviewed, approved, and performed. Routine tests and maintenance activities are not experiments.

6.4.2 Approved Experiments

All proposed experiments utilizing the reactor shall be evaluated by the experimenter and a licensed Senior Reactor Operator to assure compliance with the provisions of the utilization license, the Technical Specifications, and 10CFR Parts 20 and 50. If, in the judgement of the Senior Reactor Opertor, the experiment meets with the above provisions, is an approved experiment, and does not constitute a threat to the integrity of the reactor, it may be approved for performance. When pertinent, the evaluation shall include considerations of:

(1) The reactivity worth of the experiemnt

(2) The integrity of the experiment, including the effects of changes in temperature, pressure, or chemical composition

(3) Any physical or chemical interaction that could occur with the reactor components, and

(4) Any radiation hazard that may result from the activation of materials or from external beams.

6.4.3 New Experiments

Prior to performing an experiment not previously approved for the reactor, the experiment shall be reviewed and approved by the Reactor Operations Committee. Committee review shall consider the following information:

(1) The purpose of the experiment,

(2) The procedure for the performance of the experiment, and

(3) The safety evaluation previously reviewed by a licensed Senior Reactor Operator.

6.5 Required Actions

6.5.1 Action To Be Taken In the Event A Safety Limit Is Exceeded

(1) The reactor shall be shut down, and reactor operations shall not be resumed until authorized by the NRC.

(2) The safety limit violation shall be promptly reported to the Director of the Reactor Laboratory.

(3) The safety limit violation shall be reported by telephone to the NRC within 24 hours..

(4) A safety limit violation report shall be prepared. The report shall describe the following:

a. Applicable circumstances leading to the violation including, when known, the cause and contributing factors,
b. Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public, and
c. Corrective action to be taken to prevent recurrence.

(5) The report shall be reviewed by the Reactor Operations Committee and shall be submitted to the NRC within 14 working days when authorization is sought to resume operation of the reactor.

6.5.2 Action To Be Taken In The Event Of A Reportable Occurrence

A reportable occurrence is any of the following conditions:

(1) Operating with any safety system setting less conservative than stated in these specifications.

(2) Operating in violation of a Limiting Condition for Operation established in Section 3 of these specifications.

(3) Safety system component malfunctions or other component or system malfunctions during reactor operation that could, or threaten to, render the safety system incapable of performing its intended function.

(4) An uncontrolled or unanticipated increase in reactivity in excess of .4% delta k/k.

(5) 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, and

(6) Abnormal and significant degradation in reactor fuel and/or cladding, coolant boundary, or confinement boundary (excluding minor leaks) where applicable that could result in exceeding prescribed radiation exposure limits of personnel and/or the environment.

(7) Any uncontrolled or unauthorized release of radioactivity to the unrestricted environment.

In the event of a reportable occurrence, the following action shall be taken:

(1) The reactor conditions shall be returned to normal, or the reactor shall be shutdown, to correct the occurrence.

(2) The Director of the Reactor Laboratory shall be notified as soon as possible and corrective action shall be taken before resuming the operation involved.

(3) A written report of the occurrence shall be made which shall include an analysis of the cause of the occurrence, the corrective action taken, and the recommendations for measures to preclude or reduce the probability of recurrence. This report shall be submitted to the Director and the Reactor Operations Committee for review and approval.

(4) A report shall be submitted to the Nuclear Regulatory Commission in accordance with Section 6.6.2 of these specifications.

6.6 Reports

Reports shall be made to the Nuclear Regulatory Commission as follows:

6.6.1 Operating Reports

An annual report shall be made by September 30 of each year to the Director, Office of Nuclear Reactor Regulation, NRC, Washington, DC 20555, with a copy to the NRC, Region III, in accordance with 10CFR 50.4, providing the following information:

(1) 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.

(2) A tabulation showing the energy generated by the reactor (in kilowatt hours) and the number of hours the reactor was in use.

(3) The results of safety-related maintenance and inspections. The reasons for corrective maintenance of safety related items shall be included.

(4) A table of unscheduled shutdowns and inadvertent scrams, including their reasons and the corrective actions taken.

(5) A summary of the Safety Analyses performed in connection with changes to the facility or procedures, which affect reactor safety, and performance of tests or experiments carried out under the conditions of Section 50.59 of 10CRF50.

(6) A summary of the nature and amount of radioactive gaseous, liquid, and solid effluents released or discharged to the environs beyond the effective control of the licensee as measured or calculated at or prior to the point of such release or discharge.

(7) A summary of radiation exposures received by facility personnel and visitors, including the dates and times of significant exposures.

6.6.2 Special Reports

(1) A telephone or telegraph report of the following shall be submitted as soon as possible, but no later than the next working day, to the NRC Region III Office:

(a) Any accidental offsite release of radioactivity above authorized limits, whether or not the release resulted in property damage, personal injury, or known exposure.

(b) Any exceeding of the safety limit as defined in Section 2.1 of these specifications.

(c) Any reportable occurrences as defined in Section 6.5.2 of these specifications.

(2) A written report shall be submitted within 14 days to the Director, Office of Nuclear Reactor Regulation, US NRC, Washington, DC 20555 with a copy to the NRC Region III, in accordance with 10CFR 50.4, of the following:

(a) Any accidental offsite release of radioactivity above permissible limits, whether or not the release resulted in property damage, personal injury, or known exposure.

(b) Any exceeding of the safety limit as defined in Section 2.1.

(c) Any reportable occurrence as defined in Section 6.5.2 of these specifications.

(3) A written report shall be submitted within 30 days to the Director, Office of Nuclear Reactor Regulation, US NRC, Washington, DC 20555, with a copy to the NRC, Region III, Office in accordance with 10CFR 50.4, of the following:

(a) Any substantial variance from performance specifications contained in these specifications or in the SAR,

(b) Any significant change in the transient or accident analyses as described in the SAR, and

(c) Changes in personnel serving as Director, Engineering Experiment Station, Reactor Director, or Reactor Associate Director.

(4) A report shall be submitted within nine months after initial criticality of the reactor or within 90 days of completion of the startup test program, whichever is earlier, to the Director, Office of Nuclear Reactor Regulation, U.S. NRC, Washington, DC 20555, with a copy to the NRC, Region III upon receipt of a new facility license, an amendment to license authorizing an increase in power level or the installation of a new core of a different fuel element type or design than previously used.

The report shall include the measured values of the operating conditions or characteristics of the reactor under the new conditions, and comparisons with predicted values, including the following:

(a) Total control rod reactivity worth,

(b) Reactivity worth of the single control rod of highest reactivity worth, and

(c) Minimum shutdown margin both at ambient and operating temperatures.

(d) Excess reactivity

(e) Calibration of operating power levels

(f) Radiation leakage outside the biological shielding

(g) Release of radioactive effluents to the unrestricted environment.

6.7 Records

Records or logs of the items listed below shall be kept in a manner convenient for review, and shall be retained for as long as indicated.

6.7.1 Records to be Retained for a Period of at Least Five Years

(1) normal plant operation,

(2) principal maintenance activities,

(3) experiments performed with the reactor,

(4) reportable occurrences,

(5) equipment and component surveillance activity,

(6) facility radiation and contamination surveys,

(7) transfer of radioactive material,

(8) changes to operating procedures, and

(9) minutes of Reactor Operations Committee meetings.

6.7.2 Records to be Retained for at Least One Requalification Cycle

Regarding retraining and requalification of licensed operations personnel, the records of the most recent complete requalification cycle shall be maintained at all times the individual is employed.

6.7.3 Records to be Retained for the Life of the Facility

(1) gaseous and liquid radioactive effluents released to the environment,

(2) fuel inventories and transfers,

(3) radiation exposures for all personnel,

(4) changes to reactor systems, components, or equipment that may affect reactor safety,

(5) updated, corrected, and as-built drawings of the facility.

(6) records of significant spills of radioactivity, and status,

(7) annual operating reports provided to the NRC,

(8) copies of NRC inspection reports, and related correspondence.