2.0. SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTINGS

2.1. Safety Limits

2.1.1. Safety Limits in Forced Convection Mode of Operation

Applicability: This specification applies to the interrelated variables associated with core thermal and hydraulic performance in the forced convection mode of operation. These variables are:

P = Reactor thermal power
W = Reactor coolant flow rate
T_1 = Reactor coolant inlet temperature
L = Height of water above the core

Objective: The objective is to ensure that the integrity of the fuel clad is maintained.

Specification: In the forced convection mode of operation:

(1) The pool water level shall not be less than 19 ft above the top of the core.

(2) The reactor coolant inlet temperature shall not be greater than 111°F.

These changes are Based on LEU Study results.

(3) The true value of reactor coolant flow shall not be below 575 gpm.

(4) The combination of true values of reactor core power and reactor coolant flow shall be below the line defined by:

$P = 0.24 + (4.5 \times 10^{-3} * W)$

$P = 0$ for $W < 575$; P in MW, W in gpm

The allowed region of operation is shown by the unshaded region of Figure 2.1.

Basis: Above 575 gpm in the region of full power operation, the criterion used to establish the safety limit was a burnout ratio of 1.49 including the worst variation in the manufacturer's tolerance and specification, hot channel factors and other appropriate uncertainties. The analysis is given in the LEU SAR.

Below 575 gpm buoyancy forces competing with forced convection may lead to flow instabilities in some of the channels and is therefore not allowed. The analysis of the loss of flow transient shows that during the transition from forced convection to natural convection following a loss of flow and reactor scram that the fuel temperature is well below the temperature at which fuel clad damage could occur.

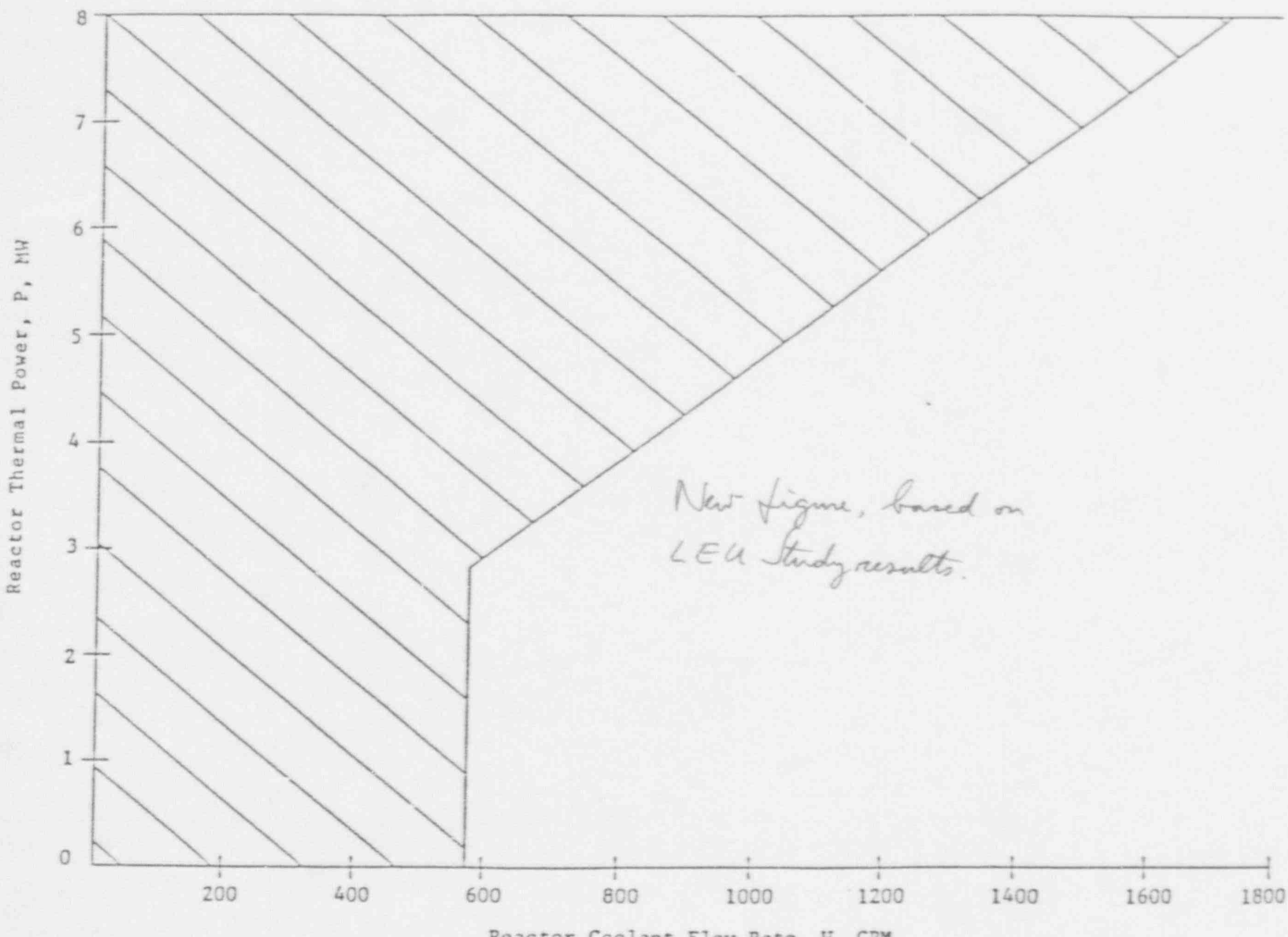

Figure 2.1 Safety Limits with Forced Convection Flow

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 in the natural convection mode of operation. These variables are:

P = Reactor thermal power
T_I = Reactor coolant inlet temperature

Objective: The objective is to ensure that the integrity of the fuel clad is maintained.

Format change only.

Specification: In the natural convection mode of operation:

(1) The true value of reactor power shall not exceed 750 kW.

(2) The reactor coolant inlet temperature shall not be greater than 111°F.

Basis reworded to take into account premises for LEU Study of natural convection mode.

Basis: The criterion for establishing a safety limit with natural convection flow is established as a fuel plate temperature. The analysis for natural convection flow shows that at 750 kW, the maximum fuel plate temperature is well below the temperature at which fuel clad damage could occur.

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New, based on considerations related to the LEU study.

2.1.3. Safety Limit for the Transition from Forced to Natural Convection Mode of Operation

Applicability: This specification applies to the condition when the reactor is in transition from forced convection flow to natural convection flow.

Objective: The objective is to ensure that the integrity of the fuel clad is maintained.

Specification: The current to the control rod magnets must be off when the reactor is making a transition from forced to natural convection.

Basis: The safety analysis of the loss of coolant transient demonstrates that the fuel plate temperature is maintained well below the temperature at which fuel clad damage could occur during the transition from forced downflow through flow reversal to the establishment of natural convection provided that the loss of flow transient is accompanied by a scram.

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2.2. Limiting Safety System Settings

Applicability: These specifications apply to the set points for the safety channels monitoring reactor thermal power, coolant flow rate, reactor coolant inlet temperature, and the height of water above the core.

Objective: The objective is to ensure that automatic protective action is initiated to prevent the safety limit from being exceeded.

Specifications:

New format

2.2.1. Forced Convection Mode

For operation in the forced convection mode, the limiting safety system settings shall be:

Reactor Thermal Power	=	3.0 MWt	(max)	
Reactor Coolant Flow Rate	=	900 gpm	(min)	Revised value based on LEU Study.
Reactor Coolant Inlet Temperature	=	108°F	(max)	
Height of Water above Core	=	19'2"	(min)	
Reactor Period	=	3.3 sec	(min)	New, based on LEU Study

New format

2.2.2. Natural Convection Mode

For operation in the natural convection mode, the limiting safety system settings shall be:

Reactor Power	=	300 kWt	(max)	
Reactor Coolant Inlet Temperature	=	108°F	(max)	
Reactor Period	=	3.3 sec	(min)	New, based on LEU Study

Re-written with basis on the LEU Study results.

Bases: The analysis in the LEU SAR shows there is sufficient margin between these settings and the safety limit under the most adverse conditions of operation:

(2.2.1.) For the forced convection mode, the LEU SAR considers accidents with reactor power at 3.45 MW, a period of 3 seconds, pool inlet temperature of 111°F and a coolant flow of 837 gpm. The maximum fuel plate temperature calculated was considerably below the aluminum clad melting point. The LSSS specified above for this mode of operation are more conservative than the parameters used in the LEU SAR analysis.

(2.2.2.) With natural convection flow, there is no minimum coolant flow rate and no minimum height of water above the core so long as there is a path for flow (see Section 3.8 of these specifications). The LEU SAR shows that the maximum fuel plate temperature under natural convection with initial power of 750 kW and pool inlet temperature of 111°F was well below the aluminum clad melting point. The LSSS specified above for this mode of operation are below the analyzed condition.

3.2. Reactor Safety System

Applicability: This specification applies to the reactor safety system channels.

Objective: The objective is to stipulate the minimum number of reactor safety system channels that must be operable to ensure that the safety limit is not exceeded during normal operation.

Table specifically identified and named.

Specification: The reactor shall not be operated unless the safety system channels described in Table 3.1 Safety System Channels are operable. (typo correction)

Bases: The startup interlock, which requires a neutron count rate of at least 2 counts per second (CPS) before the reactor is operated, ensures that sufficient neutrons are available for proper operation of the startup channel.

The pool-water temperature scram provides protection to ensure that if the limiting safety system setting is exceeded an immediate shutdown will occur to keep the fuel temperature below the safety limit. Power level scrams are provided to ensure that the reactor power is maintained within the licensed limits and to protect against abnormally high fuel temperatures. The manual scram allows the operator to shut down the reactor if an unsafe or abnormal condition arises. The period scram is provided to ensure that the power level does not increase above that described in the SAR.

Since SARs are updatable, it is best not to specify specific sections in the SAR.

Specifications on the pool-water level are included as safety measures in the event of a serious loss of primary water. Reactor operations are terminated if a major leak occurs in the primary system. The analysis in the SAR shows the consequences resulting from loss of coolant.

The bridge radiation monitor gives warning of a high radiation level in the reactor room from failure of an experiment or from a significant drop in pool-water level.

A scram from loss of primary coolant flow, loss of power to the pump, or application of power to the pump when operating in the natural convection mode, protects the reactor from overheating.

Additional factual information added.

Air pressure to the header above ambient results in a scram to:

1) Ensure that the header falls with loss of primary pump power when the reactor is operating in the forced convection mode.
2) Prevent raising the header when the reactor is in the natural convection mode.
3) Avoid producing additional Ar-41 by activating air introduced into the header.

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TABLE 3.1 SAFETY SYSTEM CHANNELS

Previously, none was specified.

Measuring Channel	Minimum No. Operable	Set Point*	Function	Operating Mode Required
Pool water level monitor	2	19'2" (min)	Scram	Forced convection
Bridge radiation monitor	1	30 mr/hr	Scram	All modes
Pool water temperature	1	108°F (max)	Scram	All modes
Power to primary pump	1	loss of power	Scram	Forced convection
		application of power	Scram	Natural convection
Primary coolant flow	1	900 gpm (min)	Scram	Forced convection
Startup count rate	1	2 cps (min)	Prevents withdrawal of any shim rod	Reactor startup
Manual button	1		Scram	All modes
Reactor power level	2	3 MWt (max)	Scram	Forced convection
		0.3 MWt (max)	Scram	Natural convection
Reactor period	1	3.3 sec (min)	Scram	All modes
Air pressure to header	1	above ambient	Scram	All modes

LEU Study result

* Values listed are limiting set points. For operational convenience, set points may be changed to more conservative values.

See Bases to TS 3.2

4.8. Reactor HEU Fuel Dose Measurements

Phrase required by LEU conversion process, here and below.

Applicability: This specification applies to the highly enriched uranium (HEU) UVAR fuel possessed under the Reactor Facility license. These specifications are applicable until all HEU UVAR fuel elements have been removed from the Reactor Facility.

Objective: The objective of this specification is to ensure that the maximum quantity of special nuclear material does not exceed the limits specified in the Reactor Facility license.

Specifications:

Headings introduced for legibility & clarity.

4.8.1. Schedule

The amount of special nuclear material (SNM) possessed at the Reactor Facility will be determined, as necessary, to ensure that limits specified by the Reactor Facility licenses are not exceeded. As a minimum, an evaluation will be completed and documented every 6 months.

4.8.2. Quantity Limits

HEU UVAR fuel elements possessed following the conversion of the UVAR to LEU fuel will be shipped away from the Reactor Facility, as necessary, to ensure that the quantity of nonexempt SNM (as defined in 10 CFR 73) does not exceed that allowed by the Reactor Facility licenses. If the amount of nonexempt SNM exceeds 5 kg the Reactor Safety Committee will be informed and the actions specified in the Physical Security Plan implemented.

4.8.3. "Self-Protection" Determinations

If HEU UVAR fuel elements have not been irradiated as a part of the UVAR core for at least one month, dose rate measurements of these HEU fuel elements will be made, as necessary, to determine which elements have dose rates higher than specified by 10 CFR 73.67(b).

Bases: The specifications provide a high degree of assurance that the amount of SNM and nonexempt SNM will not exceed the license limits. The amount of nonexempt SNM will normally be maintained at less than 5 kg, if necessary by shipping spent-fuel off-site. In the event that the 5 kg nonexempt SNM quantity is exceeded, the Reactor Safety Committee will be informed of this and the actions specified in the Physical Security Plan will be taken.

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5.0. DESIGN FEATURES

New!

5.1. Reactor Fuel Specifications

Adopted use of these terms throughout TS, for consistency of style.

Applicability: These specifications apply to UVAR low enriched uranium (LEU) fuel.

Objective: The objective is to describe LEU fuel approved by the U.S. NRC for use in the UVAR.

Specifications:

5.1.1. Fuel Material

Re-written in new format w/ information updated for the LEU fuel to be used following conversion.

UVAR LEU fuel is of a type described for use at U.S. research reactors by the U.S. Nuclear Regulatory Commission (NUREG-1313 "Safety Evaluation Report Related to the Evaluation of LEU Silicide Aluminum-Dispersed Fuel for Use in Non-Power Reactors"). The fuel meat is U_3Si_2 dispersed in an aluminum matrix and enriched to less than 20% U-235.

5.1.2. Element Description

(1) Plate-type elements of the MTR type are used. The fuel "meat" is clad with aluminum alloy to form flat fuel plates. The active length of the fuel region in the fuel plates is approximately 24 inches and the width is approximately 2.5 inches. The LEU fuel plates are joined at their long-side edges to two side plates. The entire fuel plate assembly is joined at the bottom to a cylindrical nose piece that fits into the UVAR core gridplate. The overall fuel element dimensions are approximately 3 inches by 3 inches by 36 inches. Each fuel plate contains 12.5 grams of U-235.

(2) "Standard" LEU fuel elements are composed of 22 parallel flat fuel plates each, and contain 275 grams of U-235.

(3) "Control-rod" LEU elements are similar to the standard elements, with the exception that they have half as many fuel plates (the 11 center plates being removed to form a channel which is bounded by 0.125 inch thick aluminum plates). Control-rod elements accommodate the control rods in the central channel. Their U-235 content is 137.5 grams.

(4) "Partial" LEU fuel elements are half-fueled elements composed of 11 LEU fuel plates and 11 unfuelled (dummy) plates. The U-235 content in these elements is 137.5 grams.

(5) "Special" LEU fuel elements have 22 fuel plates, of which 20 are removable. The maximum U-235 content in these elements is 275 grams and the minimum is 25 grams.

5.1.3. Core Configurations

A variety of UVAR core configurations may be used to accommodate experiments, but the loadings shall always be such that the minimum shutdown margin and excess reactivity specified in the UVAR Technical Specifications are not exceeded.

Bases: The NRC has described LEU silicide-fuel suitable for use in U.S. research reactors in NUREG-1313 "Safety Evaluation Report Related to the Evaluation of LEU Silicide Aluminum-Dispersed Fuel for Use in Non-Power Reactors," [$36.00, from NTIS, Springfield Va. (703-487-4650)]. Also, Bretscher and Snelgrove from the Argonne National Laboratory documented LEU fuel test results in ANL/RERTR/TM-14, "The Whole-Core LEU U_3Si_2-Al Fuel Demonstration in the 30-MW Oak Ridge Research Reactor." The LEU-SAR for the UVAR contains the safety analysis performed for the 22 flat-plate University of Virginia fuel elements. The LEU elements were designed by EG&G, Idaho, and are manufactured by the Babcock and Wilcox Company of Lynchburg, Virginia.

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New! Adopted for consistency of style and format.

5.3. Fuel Use and Storage

Applicability: The specifications below apply to University of Virginia Reactor fuel used and/or stored at the University of Virginia Reactor Facility following conversion of the UVAR to low enriched uranium (LEU) reactor fuel.

Objective: The objective is to describe reactor fuel which may be used, possessed and/or stored at the University of Virginia Reactor Facility as well as measures that avoid nuclear criticality or fuel-related accidents.

Specifications:

These TS have been updated to reflect LEU conversion requirements. The values quoted are justified by necessity, federal regulations or calculated needs for operational flexibility. These "needs" are justified on a separate sheet.

5.3.1. HEU Possession Limit

Following UVAR conversion to LEU:

(a) A maximum of 4.5 kilograms of contained uranium-235 at 20% or greater enrichment (which is defined as highly-enriched uranium, HEU) in the form of MTR-type reactor fuel elements may be possessed at the Reactor Facility but not used in the UVAR core, until this existing inventory of fuel is removed from the Reactor Facility.

(b) Until all HEU reactor fuel elements have been shipped offsite, HEU not in the form of reactor fuel may be possessed and used in amounts such that the total amount of HEU present at the Reactor Facility at any time, in any form, is less than 5 kilograms of uranium-235.

(c) After all HEU reactor fuel elements have been shipped offsite, the uranium-235 limit for HEU present in fission chambers, flux foils, powder form, and any other form excluding reactor fuel elements, shall be 1 kilogram. Also, the limit for uranium-235 for all enrichments shall be 12 kilograms from that moment forward.

5.3.2. LEU Possession Limit

A maximum of 11 kilograms of contained uranium-235 at less than 20% enrichment (which is defined as low enriched uranium, LEU) may be possessed and used at the Reactor Facility.

5.3.3. Plutonium Possession Limit

All plutonium generated or present in UVAR LEU reactor fuel, start-up sources, irradiation targets, flux foils and fission chambers may be possessed and used. Following conversion of the UVAR to LEU fuel, all plutonium present in HEU reactor fuel elements may be possessed until the HEU reactor fuel elements are shipped offsite.

Note: ANS 15.1 was basis for this wording, w/ slight modification underlined.

5.3.4. Storage Reactivity Limitation

All reactor fuel elements, including fueled experiments and fuel devices not in the reactor, shall be stored in a geometric array where calculated k_{eff} is no greater than 0.9, for all conditions of moderation and reflection using light water, except in cases where an approved fuel shipping container is used, in which case the calculated k_{eff} for the container shall apply.

5.3.5. Storage Cooling Requirement

Irradiated fuel elements and fueled devices shall be stored in an array that will permit sufficient natural convection cooling by water or air so that the fuel element or fueled device surface temperature will not exceed the boiling point of water.

Bases: Section 5.4 of the American National Standard ANSI/ANS-15.1-1990, "The Development of Technical Specifications for Research Reactors," was used as the overall basis for the above specifications. The specification 5.3.1 dealing with HEU fuel is made pursuant to the Atomic Energy Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." The limit given in specification 5.3.2 is based on an estimated reasonable need for reactor fuel for use in the core and a spare fuel requirement determined by DOE's expected spare fuel manufacturing schedule. The specification in 5.3.3 is based on the unavoidable production of small amounts of plutonium in reactor fuel, sources, irradiation targets, flux foils and fission chambers, as a consequence of normal reactor operation. Precise amounts of plutonium produced, decayed or burned during reactor operation can't be quantified, and this is not necessary for the small amounts of plutonium produced and contained in these aforementioned devices pose no undue reactor or radiation safety risks.

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PROPOSED SPECIAL NUCLEAR MATERIAL POSSESSION LIMITS

The following limits are proposed for the UVAR license when it is amended by the LEU Conversion Order:

1) Amount of U-235 of enrichment less than 20%: 11 000 grams

Justification: The first LEU Core and spares will be composed of

26 Standard Elements (@ 275 g U-235/element)	7 150
5 Control Elements (@ 138 g U-235/element)	690
2 Partial Elements (@ 138 g U-235/element)	276
1 Special Element (@ 275 g U-235/element)	275

Every 2 years, 8 spare elements will be constructed by B&W for use in the UVAR. All 8 elements might be sent to the Reactor Facility for storage in the Facility's Fuel Storage Room, because the fuel is LEU (hence not subject to the NRC Show Cause Order on Spare Reactor HEU Fuel) and it is likely that B&W would not act as a storage facility for U.VA.. The entire first LEU core with first spares could be at the Reactor Facility when the second set of spare elements are received, so these elements should be added to the above total.

8 Standard Elements (@ 275 g U-235/element)	2 200
TOTAL	10 591 g

This number is rounded off to 11 kg.

2) Amount of U-235 of enrichment including and above 20%: 1 000 g

Justification: At present there are 427 g of U-235 present in fission chambers, flux foils, powder form, etc.. at the Reactor Facility. It is reasonable to have a margin above this value to allow for additional chambers which may be purchased by or donated to U.VA., and for additional flux foils. The requested 1 Kg HEU limit would not require an increase in security level.

3) Proposed total U-235 limit, for all enrichments, is the sum of 1) and 2) above, that is, 12 000 g.

4) Proposed possession limit for plutonium:
All Pu generated in the HEU fuel.
All Pu in UVAR start-up sources, flux foils and fission chambers.
All Pu generated in the LEU fuel.

Justification: The production of Pu in fuel, sources, foils and fission chambers is an unavoidable consequence of normal reactor operation and the precise amounts produced, decayed and burned up can't be quantified. The Pu contained in these devices poses no reactor or radiation safety risks.

Justification for LEU Conversion Related TS Amendments

Coolant Flow

Changes to TS 2.1 involving the reactor power and coolant flow relationship are based on the LEU UVAR safety study. Note: Please look for annotations in the margins of the "marked" copy for further explanatory details about this and other proposed TS amendments.

Limiting Safety System Settings

The revisions made to TS 2.2 values for reactor coolant flow rate and reactor period are based on the analysis and results of the LEU UVAR safety study.

UVAR TS 4.8, HEU Fuel Dose Measurements

This TS has been updated to reflect a temporary transition period where both LEU and HEU fuel would be possessed at the Reactor Facility, although only LEU would be licensed for use in the UVAR core. This transition period is made necessary because restrictions on use of the spent-fuel cask state that fuel run in the core must be allowed to cool for a 3-month period prior to shipment.

Reactor Fuel Specifications

The LEU fuel which U.Va. proposes for use in the UVAR core has been specified with the detail necessary for NRC approval.

SNM Possession Limits

LEU, HEU and Pu licensed possession limits are proposed that meet regulatory and operational requirements.

APPENDIX A

TECHNICAL SPECIFICATIONS

FOR THE

UNIVERSITY OF VIRGINIA REACTOR

FACILITY LICENSE No. R-66
DOCKET No. 50-62

December 10, 1992

TABLE OF CONTENTS

1.0. DEFINITIONS

Administrative Controls: Administrative controls are those organizational and procedural requirements that are established by the reactor licensee management.

Applicability: As regards use of this term in the Technical Specifications, it is a statement that indicates which components are involved.

Basis: As regards use of this term in the Technical Specifications, it is a statement that provides the background or reason for the choice of specification(s), or references a particular portion of the Safety Analysis Report (SAR) that does.

Beamports: The beamports are the two 8-inch diameter neutron beamports that penetrate the shield on the south side of the UVAR pool.

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. (Also, see definition for measuring channel).

Channel Calibration: A channel calibration is an adjustment of the channel such that its output corresponds with acceptable range and accuracy to known input 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.

Channel Check: A channel check is a qualitative verification of acceptable performance by observation of channel behavior, or comparison of the channel with other independent channels or systems measuring the same variable, where this capability exists.

Channel Test: A channel test is the introduction of a signal into a channel to verify that it is operable.

Confinement: Confinement means a closure on the overall facility that controls the movement of air into it and out through a controlled path.

Design Features: The definition for design features is as defined in 10 CFR 50.36.

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

Experiment: Any operation, hardware, or target (excluding devices such as detectors, foils, activation samples in an irradiation facility, etc...) that is designed to investigate non-routine reactor characteristics or that is intended for reactor irradiation within the UVAR pool, on or in the beamport or irradiation facility, and that is not rigidly secured to a core or shield structure so as to be a part of their design.

Experimental Facility: An experimental facility is a structure or device associated with the reactor that is intended to guide, orient, position, manipulate, or otherwise facilitate a multiplicity of experiments of similar character.

Experimental Methods: Experimental Methods are written and approved instructions which provide guidance to the reactor staff or experimenters for the completion of tasks specified in Experimental Procedures (EPs). While EPs, and changes thereto, are reviewed and approved by the Reactor Safety Committee (RSC), experimental methods are written and reviewed by reactor staff and/or experimenters and approved by a reactor supervisor or administrator. Newly developed experimental methods or changes to existing experimental methods should be sent to the RSC as information items.

Experimental Procedures: Written procedures reviewed and approved by the Reactor Safety Committee which describe the manner in which experiments are run in conjunction with the UVAR, to assure reactor and radiological safety. Operational limits peculiar to the experiment are included in these procedures. Detailed implementation of experimental procedures may be made through the use of experimental methods.

Explosive Material: Explosive material is a solid or liquid that is categorized as a Severe, Dangerous, or Very Dangerous Explosion Hazard in "Dangerous Properties of Industrial Materials" by N.I. Sax, or is given an Identification of Reactivity (Stability) index 2, 3, or 4 by the National Fire Protection Association in its publication 704-M, "Identification System for Fire Hazards of Materials," also enumerated in the "Handbook for Laboratory Safety" published by the Chemical Rubber Co.

Forced Convection Mode: The reactor is in the Forced Convection Mode when the flow header is up and the primary pump is operating.

Fueled Experiment: A fueled experiment is an experiment that contains U-235, U-233 or Pu-239 in levels exceeding trace quantities. Reactor fuel elements are not included in this definition. (Also, see the definition for trace quantities and TS 3.7.).

Important Process Variables: Important process variables are measurable parameters that individually or in combination reflect the basic physical condition of physical barriers. They may include fuel temperature, reactor power, reactor coolant flow rate, reactor coolant inlet or outlet temperature, pool level, or coolant pressure. (Also, see definition for safety limits).

Large Access Facilities: The large access facilities are the two large openings approximately 5 ft wide by 6 ft high that penetrate the shield on the south side of the UVAR pool.

Licensed Operator: A licensed operator is an individual authorized by the U.S. Nuclear Regulatory Commission to carry out the duties and responsibilities associated with operation of the UVAR. (Also, see definitions for Senior Reactor Operator and Reactor Operator).

Limiting Conditions for Operations: Limiting Conditions of Operation (LCOs) are those administratively established constraints on equipment and operational characteristics that shall be adhered to during operation of the facility. The LCOs are the lowest functional capability or performance level required for safe operation of the reactor.

Limiting Safety System Settings: Limiting Safety System Settings (LSSS) are those limiting values for settings of the safety channels by which point protective action must be initiated. The LSSS are chosen so that automatic protective action will terminate the abnormal situation before a safety limit is reached. The calculation of the LSSS shall include the process uncertainty, the overall measurement uncertainty, and transient phenomena of the process instrumentation. To achieve operational flexibility, it is recommended that actual trip points, where possible, be set more conservatively than specification values.

Measured Value: The measured value of a parameter is the value of the variable as it appears on the output of a measuring channel.

Measuring Channel: A measuring channel is the combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter. (Also, see definition for channel).

Methods: Methods are written and approved instructions which provide guidance to the reactor staff for the completion of tasks specified in Standard Operating Procedures (SOP's). While SOP's, and changes thereto, are reviewed and approved by the Reactor Safety Committee (RSC), methods are written and reviewed by reactor staff and approved by a reactor supervisor or administrator. Newly developed methods or changes to existing methods should be sent to the RSC as information items.

Movable Experiment: A movable experiment is one where it is intended that all or part of the experiment may be inserted, removed, or manipulated in or near the core while the reactor is critical.

Natural Convection Mode: The reactor is in the Natural Convection Mode when the flow through the core is maintained by the buoyancy forces associated with the water being heated by the reactor.

Objective: As regards use of this term in the Technical Specifications, it is a statement that indicates the purpose of the specifications.

On Call: To be on call refers to an individual who (1) has been specifically designated and the designation is known to the operator on duty, (2) keeps the operator on duty informed of where he may be contacted and the phone number, and (3) is capable of getting to the Reactor Facility within a reasonable time under normal conditions (e.g., approximately 30 min).

Operable: A component or system is operable when it is capable of performing its intended function in a normal manner.

Operating: A component or system is operating when it is performing its intended function in a normal manner.

Protective Action: Protective action is 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 having reached a specific limit.

(1) channel level. At the protective instrument channel level, protective action is the generation and transmission of a trip signal indicating that a reactor variable has reached a specified limit.

(2) subsystem level. At the protective instrument subsystem level, protective action is the generation and transmission of a trip signal indicating that a specified limit has been reached.

NOTE: Protective action at this level would lead to the operation of the safety shutdown equipment to immediately shut down the reactor.

(3) instrument system level. At the protective instrument system level, protective action is the generation and transmission of the command signal for the safety shutdown equipment to operate.

(4) safety system level. At the reactor safety system level, protective action is the operation of sufficient equipment to immediately shut down the reactor.

Reactivity Limits: Reactivity limits for experiments are quantities referenced to an average pool temperature of $<90°$ F with the effect of xenon poisoning on core reactivity accounted for if greater than or equal to 0.07\$. The reactivity worth of samarium in the core will not be included in reactivity limits. The reference core condition will be known as the cold, xenon-free critical condition.

Reactivity Worth of an Experiment: The reactivity worth of an experiment is the value of the reactivity change that results from the experiment being inserted into or removed from its intended position.

Reactor Operating: The reactor is operating whenever it is not secured or shutdown.

Reactor Operation: The reactor is in operation when not all of the shim rods are fully inserted and six or more fuel elements are loaded in the grid plate.

Reactor Operator: An NRC-licensed reactor operator is an individual who is certified by the NRC and the reactor administration to manipulate the controls of the UVAR reactor.

Reactor Safety Systems: Reactor safety systems are 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: The reactor is secured when:

(1) Either there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection, or

(2) The following conditions exits:
   a. All shim rods are fully inserted,
   b. The console key is in the OFF position and 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 are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum reactivity value allowed for a single experiment, or one dollar, whichever is smaller.

Reactor Shutdown: The reactor is shut down if it is subcritical by at least one dollar in the reference core condition with the reactivity worth of all installed experiments included.

Reactor Staff: The Reactor Director and all personnel administratively reporting to him.

Reference Core Condition: The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible ($<0.30\$$).

Regulating Rod: The regulating rod is a control rod of low reactivity worth fabricated from stainless steel and used primarily to maintain an intended power level. The regulating rod need not have scram capability. The rod may be controlled by the operator with a manual switch or by the automatic servo-controller.

Reportable Occurrence: A reportable occurrence is any of the conditions described in Section 6.6.2 of these specifications.

Research Reactor: A research reactor is defined as a device designed to support a self-sustaining neutron chain reaction for research, development, education, training, or experimental purposes, and that may have provisions for the production of radioisotopes.

Safety Limits: Safety Limits are limits on important process variables that are found to be necessary to reasonably protect the integrity of the principal physical barriers that guard against the uncontrolled release of radioactivity. The principal physical barrier is often the fuel cladding. (Also, see the definition for important process variables).

Scram Time: Scram time is the elapsed time between the initiation of a scram signal and a specified movement of a control or safety device.

Secured Experiment: A secured experiment is an experiment, experiment 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 by hydraulic, pneumatic, buoyant, or other forces that are normal to the operating environment of the experiment or by forces that can arise as a result of credible malfunctions.

Senior Reactor Operator: An NRC-licensed senior reactor operator is an individual who is certified by the NRC and the reactor administration to manipulate the controls of the UVAR reactor and to direct the activities of reactor operators.

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.

Shim Rod: A shim rod is a control rod fabricated from borated stainless steel, which is used to compensate for fuel burnup, temperature, and poison effects. A shim rod is magnetically coupled to its drive unit allowing it to perform the function of a safety rod when the magnet is de-energized. (Also, see definition for regulating rod).

Shutdown Margin: Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition and with the regulating rod and the most reactive shim rod in their most reactive position, and that the reactor will remain subcritical without further operator action.

Specification(s): As regards use of this term in the Technical Specifications, it is a statement that provides specific data, conditions, or limitations that bound a system or operation. This statement is the most important statement in the technical specifications agreement. Only the specifications statements are governing.

Standard Operating Procedures: Written procedures reviewed and approved by the Reactor Safety Committee to assure reactor safety and compliance with federal regulations, which describe the manner by which the reactor staff will operate and maintain the UVAR. (Also, see TS 6.3).

Surveillance Requirements: The definition for surveillance requirements is as defined in 10 CFR 50.36.

Surveillance Time Intervals:

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 (must be done during the calendar day)

Trace Quantities: As related to fissionable or fissile nuclides such as U-235, U-233 or Th-232 potentially present in environmental samples on which neutron activation analysis may be attempted, trace quantities are taken to be negligibly-small concentration levels below 100 parts-per-million (ppm). (Also, see the definition for Fueled Experiment).

Tried Experiment: A tried experiment is (1) an experiment previously performed in the UVAR or (2) an experiment for which the size, shape, composition, and location does not differ significantly enough from an experiment previously performed in the UVAR to affect reactor safety.

True Value: The true value is the actual value of a parameter.

Unscheduled Shutdown: An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by the actuation of the reactor safety system, operator error, equipment malfunction, or manual shutdown in response to conditions that could adversely affect safe operation, not including shutdowns that occur during testing nor check-out operations.

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2.0. SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTINGS

2.1. Safety Limits

2.1.1. Safety Limits in Forced Convection Mode of Operation

Applicability: This specification applies to the interrelated variables associated with core thermal and hydraulic performance in the forced convection mode of operation. These variables are:

P = Reactor thermal power
W = Reactor coolant flow rate
T_I = Reactor coolant inlet temperature
L = Height of water above the core

Objective: The objective is to ensure that the integrity of the fuel clad is maintained.

Specification: In the forced convection mode of operation:

(1) The pool water level shall not be less than 19 ft above the top of the core.

(2) The reactor coolant inlet temperature shall not be greater than 111°F.

(3) The true value of reactor coolant flow shall not be below 575 gpm.

(4) The combination of true values of reactor core power and reactor coolant flow shall be below the line defined by:

$P = 0.24 + (4.5 \times 10^{-3} * W)$

$P = 0$ for $W < 575$; P in MW, W in gpm

The allowed region of operation is shown by the unshaded region of Figure 2.1.

Basis: Above 575 gpm in the region of full power operation, the criterion used to establish the safety limit was a burnout ratio of 1.49 including the worst variation in the manufacturer's tolerance and specification, hot channel factors and other appropriate uncertainties. The analysis is given in the LEU SAR.

Below 575 gpm buoyancy forces competing with forced convection may lead to flow instabilities in some of the channels and is therefore not allowed. The analysis of the loss of flow transient shows that during the transition from forced convection to natural convection following a loss of flow and reactor scram that the fuel temperature is well below the temperature at which fuel clad damage could occur.

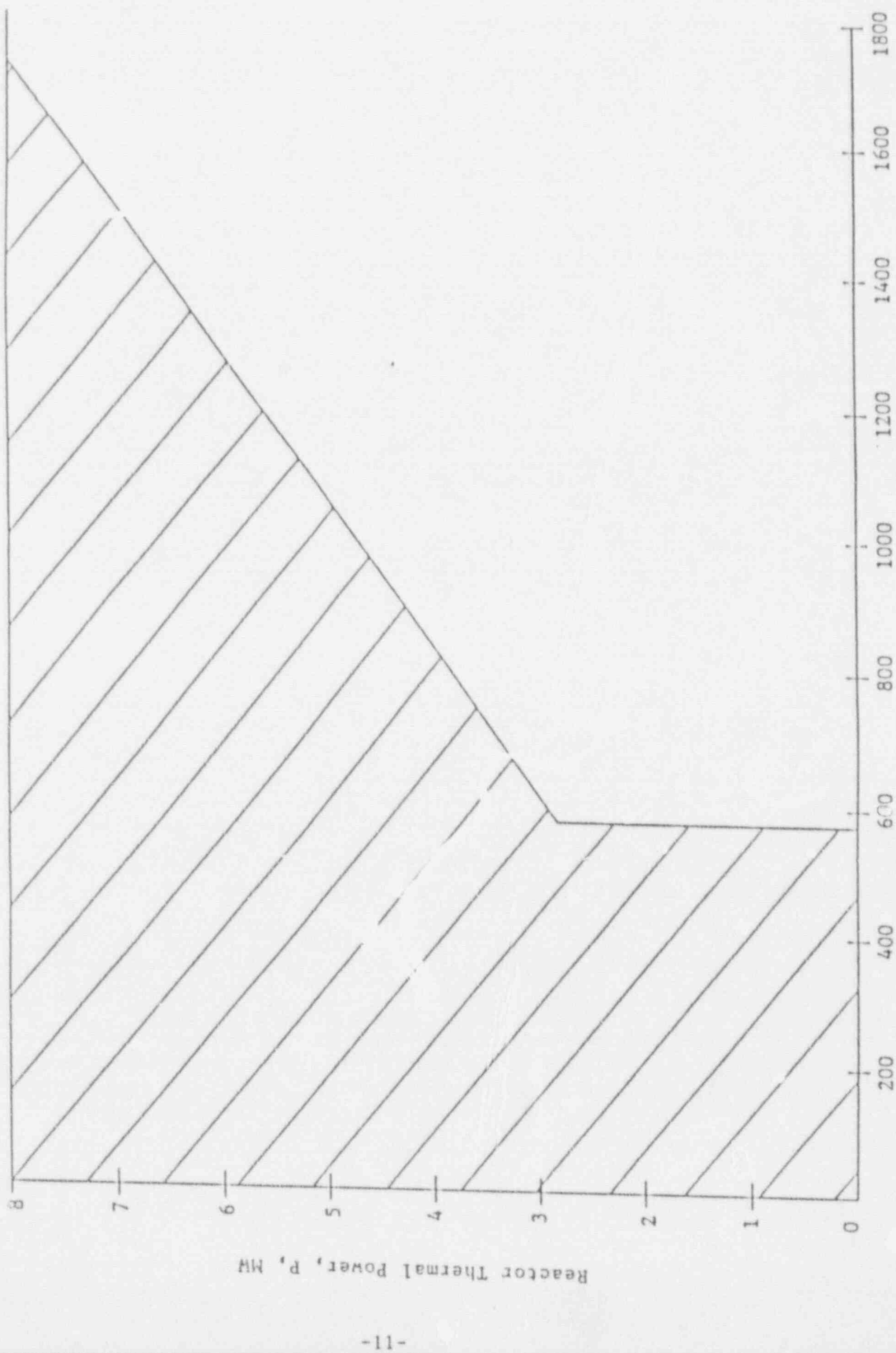

Figure 2.1 Safety Limits with Forced Convection Flow

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 in the natural convection mode of operation. These variables are:

P = Reactor thermal power
T_I = Reactor coolant inlet temperature

Objective: The objective is to ensure that the integrity of the fuel clad is maintained.

Specification: In the natural convection mode of operation:

(1) The true value of reactor power shall not exceed 750 kW.

(2) The reactor coolant inlet temperature shall not be greater than 111°F.

Basis: The criterion for establishing a safety limit with natural convection flow is established as a fuel plate temperature. The analysis for natural convection flow shows that at 750 kW, the maximum fuel plate temperature is well below the temperature at which fuel clad damage could occur.

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2.1.3. Safety Limit for the Transition from Forced to Natural Convection Mode of Operation

Applicability: This specification applies to the condition when the reactor is in transition from forced convection flow to natural convection flow.

Objective: The objective is to ensure that the integrity of the fuel clad is maintained.

Specification: The current to the control rod magnets must be off when the reactor is making a transition from forced to natural convection.

Basis: The safety analysis of the loss of coolant transient demonstrates that the fuel plate temperature is maintained well below the temperature at which fuel clad damage could occur during the transition from forced downflow through flow reversal to the establishment of natural convection provided that the loss of flow transient is accompanied by a scram.

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2.2. Limiting Safety System Settings

Applicability: These specifications apply to the set points for the safety channels monitoring reactor thermal power, coolant flow rate, reactor coolant inlet temperature, and the height of water above the core.

Objective: The objective is to ensure that automatic protective action is initiated to prevent the safety limit from being exceeded.

Specifications:

2.2.1. Forced Convection Mode

For operation in the forced convection mode, the limiting safety system settings shall be:

Reactor Thermal Power	=	3.0 MWt	(max)
Reactor Coolant Flow Rate	=	900 gpm	(min)
Reactor Coolant Inlet Temperature	=	108°F	(max)
Height of Water above Core	=	19'2"	(min)
Reactor Period	=	3.3 sec	(min)

2.2.2. Natural Convection Mode

For operation in the natural convection mode, the limiting safety system settings shall be:

Reactor Power	=	300 kWt	(max)
Reactor Coolant Inlet Temperature	=	108°F	(max)
Reactor Period	=	3.3 sec	(min)

Bases: The analysis in the LEU SAR shows there is sufficient margin between these settings and the safety limit under the most adverse conditions of operation:

(2.2.1.) For the forced convection mode, the LEU SAR considers accidents with reactor power at 3.45 MW, a period of 3 seconds, pool inlet temperature of 111°F and a coolant flow of 837 gpm. The maximum fuel plate temperature calculated was considerably below the aluminum clad melting point. The LSSS specified above for this mode of operation are more conservative than the parameters used in the LEU SAR analysis.

(2.2.2.) With natural convection flow, there is no minimum coolant flow rate and no minimum height of water above the core so long as there is a path for flow (see Section 3.8 of these specifications). The LEU SAR shows that the maximum fuel plate temperature under natural convection with initial power of 750 kW and pool inlet temperature of 111°F was well below the aluminum clad melting point. The LSSS specified above for this mode of operation are below the analyzed condition.

3.0. LIMITING CONDITIONS FOR OPERATION

3.1. Reactivity

Applicability: This specification applies to the reactivity condition of the reactor and the reactivity worth of control rods and experiments.

Objectives: The objectives are to ensure that the reactor can be shut down at all times and that the safety limit will not be exceeded.

Specification: The reactor shall not be operated at powers in excess of 1 kW unless the following conditions exist:

(1) The minimum shutdown margin provided by shim rods, with secured experiments (see Section 1.0) in place and referred to the cold, xenon-free condition with the highest-worth shim rod and the regulating rod fully withdrawn, is greater than 0.55$.

(2) An experiment with a reactivity worth greater than 0.60$ must be a secured experiment.

(3) The total reactivity worth of the two experiments having the highest reactivity worth is less than 2.00$.

(4) The total reactivity worth of all experiments is less than 2.50$.

(5) The maximum excess reactivity with fixed experiments in place and referred to cold, xenon-free condition shall be limited to 6.50$.

Bases: Operation of the reactor at power levels below 1 kW to measure the reactivity worth of untried experiments, and to measure the excess reactivity of new core loadings, is allowed with procedures approved by the Reactor Safety Committee. Reactivity is measured in dollars from the reactor period, and as such is the quantity of safety significance. In the past, an "effective beta-bar" multiplicative conversion factor of 0.75 was used for the conversion of reactivity expressed in $ units to reactivity in % delta k/k units. However, reactivity limits expressed in $ are more appropriate for the Technical Specifications, since they are not dependent on the type of fuel used in the reactor or on the geometry of a particular core loading.

(1) The shutdown margin required by Specification 3.1(1) is necessary so that the reactor can be shut down from any operating condition and remain shut down after cooldown and xenon decay, even if the highest worth shim rod should stick in the fully withdrawn position, and with no credit taken for the non-scrammable regulating rod.

(2) The reactivity of 0.60$ in Specification 3.1(2) corresponds to an asymptotic 3-sec period. If this period were sustained without scramming the reactor until the reactor power reaches the maximum true value for the Limiting Safety System Setting (LSSS) for the High Power Scram (at which time the reactor scrams on high power), the resulting power overshoot would not exceed the safety limit for power vs. flow.

(3) The reactivity of 2.00$ in Specification 3.1(3) is less than 2.16$ which corresponds to a 6.9-msec period. Reactor Core DU-12/25 of the SPERT-1 series of tests had MTR plate type elements (Reference: Thompson and Beckerly, "Technology of Nuclear Reactor Safety," Volume I, page 683 (1964)). A 6.9-msec period was nondestructive. The simultaneous failure of more than two experiments is considered unlikely.

(4) The total reactivity of 2.50$ in Specification 3.1(4) places a reasonable upper limit on the worth of all experiments.

(5) The limit of 6.50$ on excess reactivity is to allow for xenon override and operational flexibility and to ensure that the operational reactor is reasonably similar in configuration to the reactor core analyzed in the SAR. In general, the excess reactivity is limited by the shutdown margin requirement.

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3.2. Reactor Safety System

Applicability: This specification applies to the reactor safety system channels.

Objective: The objective is to stipulate the minimum number of reactor safety system channels that must be operable to ensure that the safety limit is not exceeded during normal operation.

Specification: The reactor shall not be operated unless the safety system channels described in Table 3.1 Safety System Channels are operable.

Bases: The startup interlock, which requires a neutron count rate of at least 2 counts per second (CPS) before the reactor is operated, ensures that sufficient neutrons are available for proper operation of the startup channel.

The pool-water temperature scram provides protection to ensure that if the limiting safety system setting is exceeded an immediate shutdown will occur to keep the fuel temperature below the safety limit. Power level scrams are provided to ensure that the reactor power is maintained within the licensed limits and to protect against abnormally high fuel temperatures. The manual scram allows the operator to shut down the reactor if an unsafe or abnormal condition arises. The period scram is provided to ensure that the power level does not increase above that described in the SAR.

Specifications on the pool-water level are included as safety measures in the event of a serious loss of primary water. Reactor operations are terminated if a major leak occurs in the primary system. The analysis in the SAR shows the consequences resulting from loss of coolant.

The bridge radiation monitor gives warning of a high radiation level in the reactor room from failure of an experiment or from a significant drop in pool-water level.

A scram from loss of primary coolant flow, loss of power to the pump, or application of power to the pump when operating in the natural convection mode, protects the reactor from overheating.

Air pressure to the header above ambient results in a scram to:

1) Ensure that the header falls with loss of primary pump power when the reactor is operating in the forced convection mode.
2) Prevent raising the header when the reactor is in the natural convection mode.
3) Avoid producing additional Ar-41 by activating air introduced into the header.

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TABLE 3.1 SAFETY SYSTEM CHANNELS				
Measuring Channel	Minimum No. Operable	Set Point*	Function	Operating Mode Required
Pool water level monitor	2	19'2" (min)	Scram	Forced convection
Bridge radiation monitor	1	30 mr/hr	Scram	All modes
Pool water temperature	1	108°F (max)	Scram	All modes
Power to primary pump	1	loss of power	Scram	Forced convection
		application of power	Scram	Natural convection
Primary coolant flow	1	900 gpm (min)	Scram	Forced convection
Startup count rate	1	2 cps (min)	Prevents withdrawal of any shim rod	Reactor startup
Manual button	1		Scram	All modes
Reactor power level	2	3 MWt (max)	Scram	Forced convection
		0.3 MWt (max)	Scram	Natural convection
Reactor period	1	3.3 sec (min)	Scram	All modes
Air pressure to header	1	above ambient	Scram	All modes

* Values listed are limiting set points. For operational convenience, set points may be changed to more conservative values.

3.3. Reactor Instrumentation

Applicability: This specification applies to the instrumentation that must be operable for safe operation of the reactor.

Objective: The objective is to require that sufficient information is available to the operator to ensure safe operation of the reactor.

Specification: The reactor shall not be operated unless the measuring channels described in Section 3.2 Reactor Safety Systems and in Table 3.2 Measuring Channels are operable.

Bases: The neutron detectors and the core gamma monitor provide assurance that measurements of the reactor power level are adequately covered at both low and high power ranges.

The radiation monitors provide information to operating personnel of a decrease in pool-water level and of an impending or existing danger from radiation contamination or streaming, allowing ample time to take necessary precautions to initiate safety action.

The reactor room constant air monitor and reactor face monitor provide redundant measures of abnormal high radiation levels. Because the other measuring channels for determining the radiation levels are required for reactor operation, the reactor can be operated safely if these monitors are not functioning for short periods of time.

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Table 3.2 Measuring Channels

Measuring Channel	Minimum No. Operable	Operating Mode in Which Required
Linear power	1	All modes
Intermediate power (Log N) and period	1	All modes
Core gamma monitor *	1	Forced convection mode *
Reactor room constant air monitor *	1	All modes *
Bridge radiation monitor	1	All modes
Reactor face monitor *	1	All modes*
Pool-water level monitor	2	Forced convection mode
Pool-water temperature	1	All modes
Primary coolant flow	1	Forced convection mode
Startup count rate	1	Reactor startup
Reactor power level	2	All modes

* The reactor room constant air monitor, reactor face monitor, and core gamma monitor may be out of service for a period not to exceed 7 days without requiring reactor shutdown. If the reactor face monitor cannot be repaired within 7 days, it may be replaced by a locally alarming monitor of similar range for up to 30 days without requiring a reactor shutdown.

3.4. Radioactive Effluents

Applicability: This specification applies to the monitoring of radioactive effluents from the Reactor Facility. Airborne and liquid effluents are discussed separately in the following sections.

3.4.1. Airborne Effluents

Objective: The objective is to ensure that exposure to the public resulting from the release of Ar-41 and other airborne effluents to the environment will be well below the limits of 10 CFR 20 for unrestricted areas.

Specification: The activity of gases released beyond the Reactor Facility's site boundary shall not exceed 10 CFR 20 limits. When a neutron beamport vented to the atmosphere is drained of water during reactor operations, the effluent shall be monitored by an instrument located in the effluent vent and the effluent vent will have sufficient flow to maintain releases within 10 CFR 20 limits.

Bases: A basis for this specification is given by the analysis in the SAR. Compliance with federal regulation is another basis.

3.4.2. Liquid Effluents

Objective: The objective is to ensure that exposure to the public resulting from the release of radioactive effluents will be well below the limits of 10 CFR 20 for unrestricted areas.

Specification: The activity of liquids released beyond the Reactor Facility's site boundary shall not exceed 10 CFR 20 limits.

Basis: The basis for this specification is compliance with federal regulations.

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3.5. Confinement

Applicability: This specification applies to the capability of isolating the UVAR's reactor room, when necessary.

Objective: The objective is to prevent exposure to the public from exceeding the limits of 10 CFR 20 for unrestricted areas, resulting from airborne activity released into the UVAR's reactor room, by providing confinement.

Specification: The reactor shall not be operated unless the following equipment is operable.

Equipment	Function
Truck door closed switch	Scram reactor when truck door is not fully closed
Ventilation duct doors	Close and seal when Bridge Radiation Monitor alarms
Personnel door	Close and seal when Bridge Radiation Monitor alarms
Emergency exit manhole water level	Water level is high enough to form a water seal at least 6 in. in depth

Basis: The basis for this specification is compliance with federal regulations.

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3.6. Limitation on Experiments

Applicability: These specifications apply to experiments installed in the reactor and its experimental facilities.

Objective: The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications:

3.6.1. Reactivity

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

(1) The reactivity worth of all experiments shall be in conformance with specifications in Section 3.1.

(2) Movable experiments must be worth less than 0.13$.

(3) Experiments worth more than 0.13$ must be inserted or removed with the reactor shut down except as noted in Item (4).

(4) Previously tried experiments with measured worth less than 0.50$ may be inserted or removed with the reactor 2.70$ or more subcritical.

(5) If an experiment worth more than 0.50$ is inserted in the reactor, a procedure approved by the Reactor Safety Committee shall be followed.

3.6.2. Containers

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

(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.

3.6.3. Dangerous Materials

Explosive material shall not be allowed in the reactor unless specifically approved by the Reactor Safety Committee. Experiments reviewed by the Reactor Safety Committee in which the material is potentially explosive, either while contained or if it leaks from the container, shall be designed to prevent damage to the reactor core or to the control rods or instrumentation, and to prevent any changes in reactivity.

3.6.4. Cooling

Cooling shall be provided to prevent the surface temperature of an experiment to be irradiated from exceeding the boiling point of the reactor pool water.

3.6.5. Precautions

Experimental apparatus, material, or equipment to be inserted in the reactor, shall not be positioned so as to cause shadowing of the nuclear instrumentation, interference with the control rods, or other perturbations that may interfere with the safe operation of the reactor.

3.6.6. Cobalt Facility

The Co-60 rods possessed under the UVAR Operating License shall be used and stored in the UVAR pool at distances greater than 5 feet from the operating UVAR reactor. Gamma irradiation facilities utilizing the Co-60 rods shall be designed to prevent physical damage to the Co-60 rods. UVAR pool water samples shall be subjected to gamma spectroscopy for the presence of Co-60 on a monthly frequency, (interval not to exceed six weeks) to assure that substantial leakage of Co-60 from the rods to reactor pool water does not occur.

Bases: (TS 3.6.1 - 3.6.5) The limitations on experiments specified in TS 3.6.1 through TS 3.6.5 are based on the irradiation program authorized by Amendment No. 3 to License No. R-66 dated August 13, 1962. The reactivity of less than 0.13$ that can be inserted or removed with the reactor in operation is to accommodate experiments in the rabbits.

(Co-60 Facility) The Co-60 rods are to be kept a safe distance away from the UVAR reactor when it is operated, to avoid neutron activation and possible failure of the rod cladding, which may result in leakage of Co-60 to the reactor pool water. The Co-60 rods and the gamma irradiation facilities in which they are used will not be used in conjunction with the UVAR.

The monthly reactor pool water sampling frequency, adopted to monitor possible Co-60 leakage from the rods, is the same as that used in the U.S. AEC Safety Evaluation that was performed for these rods by the Division of Reactor Licensing on August 4, 1971. This is a reasonable frequency, for the most likely damage to the rods would be caused by cladding corrosion leading to pin holes. Co-60 leakage under these circumstances would proceed very slowly, into a large pool of water. Therefore, a monthly water sampling and analysis frequency should be adequate to indicate contamination levels before they become significant.

3.7. Operation with Fueled Experiments

Applicability: This specification applies to the operation of the reactor with a fueled experiment within the reactor building.

Objective: The objective is to ensure that the confinement leak rate and fission product inventory in fueled experiments are within limits used in the safety analysis.

Specifications:

3.7.1. Fueled Experiments Generating Power Above or Equal to 1 W

For fueled experiments in which the thermal power generated is greater than or equal to 1 watt (W), the reactor shall not be operated unless the following conditions are satisfied:

(1) The experiment must be in the reactor pool and under at least 15 ft of water.

(2) The thermal power (or fission rate) generated in the experiment is not greater than 100 W (3.2×10^{12} fissions/sec).

(3) The calculated total energy produced by the experiment shall not exceed 600 W-years.

(4) The leak rate from the reactor room is not greater than 50% of containment of volume in 20 hours as measured within the previous 12 months.

3.7.2. Fueled Experiments Generating Power Below 1 W

Fueled experiments in which the thermal power generated is less than 1 W (3.2×10^{10} fissions/sec):

(1) May be located anywhere in the reactor building.

(2) The calculated total energy produced by the experiment shall not exceed 600 W-years.

Bases: In the event of the failure of a fueled experiment, with the subsequent release of fission products (100% noble gas, 50% iodine, 1% solids), the 2-hour inhalation exposures to iodine and strontium 90 isotopes at the facility exclusion distance, 70 meters, are less than the limits set by 10 CFR 20, using an averaging period of 1 year.

The safety analyses for which results are used here are found in the SAR. The analysis supporting Specification 3.7.2 assumes 100% exfiltration of fission products from the reactor building in 2 hours. The analysis supporting Specification 3.7.1 for

for the fueled experiments within the reactor pool assumes a fission product retention in the reactor room equivalent to 100% fission product exfiltration in 20 hours. The specification provides suitable allowance for degradation between tests. The measurement of the exfiltration value is described in the SAR.

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3.8. Height of Water Above the Core in Natural Convection Mode of Operation

Applicability: This specification applies to the height of water above the reactor core when the reactor is operating with natural convection cooling.

Objective: The objective is to ensure that there is a continuous path for circulation of water when the reactor is operated in the natural convection mode.

Specification: The reactor shall not be operated in the natural convection mode unless there is at least 1 ft of water above the core.

Bases: One foot of water above the core is sufficient to provide a continuous path for natural convection cooling. For other than zero power operation, the radiation levels may require a greater depth for shielding, in which case the regulations in 10 CFR 20 will govern.

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3.9. Rod Drop Times

Applicability: This specification applies to the time from the initiation of a scram to the time a rod starts to drop (magnet release time) as well as to the time it takes for a rod to drop from the fully withdrawn to the fully inserted position (free-drop time).

Objective: The objective is to ensure that the reactor can be shut down within a specified period of time.

Specification: The reactor will not be operated unless (1) the magnet release time for each of the three shim rods is less than 50 msec and (2) the free-drop time for each of the three shim rods is less than 700 msec.

Bases: Rod drop times as specified will ensure that the safety limit will not be exceeded in a short period transient. The analysis is given in the SAR.

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3.10. Emergency Removal of Decay Heat

Applicability: This specification applies to the emergency removal of decay heat.

Objective: The objective is to ensure that the flow rate from this system is sufficient to prevent overheating of the fuel elements subsequent to a total loss of primary water from the core.

Specification: There shall be two separate emergency core spray systems, each capable of maintaining a flow rate of at least 10 gpm over the 64 fuel element positions for the first 30 min, and at least 7 1/2 gpm over the 64 fuel element positions for the next 60 min following a total loss of coolant.

Basis: Either of the two spray systems, as specified, will provide sufficient cooling to maintain the fuel temperature below its melting point as demonstrated by the evaluation in the SAR.

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3.11. Primary Coolant Condition

Applicability: This specification applies to the quality of the primary coolant in contact with the fuel cladding.

Objectives: The objectives are (1) to minimize the possibility for corrosion of the cladding on the fuel elements and (2) to minimize neutron activation of dissolved materials.

Specifications:

3.11.1. Conductivity

Conductivity of the poolwater shall be no higher than 5×10^{-6} mhos/cm.

3.11.2. Water pH

The pH of the poolwater shall be between 5.0 and 7.5.

Bases: A small rate of corrosion continuously occurs in a water-metal system. To limit this rate, and thereby extend the longevity and integrity of the fuel cladding, a water cleanup system is required. Experience with water quality control at many reactor facilities has shown that maintenance within the specified limits provides acceptable control.

By limiting the concentrations of dissolved materials in the water, the radioactivity of neutron activation products is limited. This is consistent with the as low as is reasonably achievable (ALARA) principle, and tends to decrease the inventory of radionuclides in the entire coolant system, which will decrease personnel exposures during maintenance and operations.

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4.0. SURVEILLANCE REQUIREMENTS

4.1. Shim Rods

Applicability: These specifications apply to the surveillance requirements for the shim rods.

Objectives: The objectives are to ensure that the shim rods are capable of performing their function and to establish that no significant physical degradation in the rods has occurred.

Specifications:

4.1.1. Drop-time Measurements

Shim rod drop times shall be measured semiannually. Shim rod drop times shall also be measured if the control assembly is moved to a new position in the core or if maintenance is performed on the mechanism.

4.1.2. Reactivity Measurements

The shim rod reactivity worth shall be measured whenever the rods are installed in a new core configuration.

4.1.3. Visual Inspections

The shim rods shall be visually inspected annually and when rod drop times exceed the limiting conditions for operation (Section 3.9 of these specifications).

Bases: The reactivity worth of the shim rods is measured to assure that the required shutdown margin is available and to provide means for determining the reactivity worth of experiments inserted in the core. The visual inspection of the shim rods and measurement of their drop times are made to determine whether the shim rods are capable of performing properly.

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4.2. Reactor Safety System

Applicability: These specifications apply to the surveillance requirements for the reactor safety system of the reactor.

Objective: The objective is to ensure that the reactor safety system is operable as required by Specification 3.2 by specifying frequency of tests, checks and calibrations.

Specifications:

4.2.1. Channel Tests

A channel test of each of the reactor safety system measuring channels shall be performed before each day's operation or before each operation extending more than one day.

4.2.2. Channel Checks

A channel check of each of the reactor safety system measuring channels shall be performed daily when the reactor is in operation.

4.2.3. Channel Calibrations

A channel calibration of the reactor safety measuring channels shall be performed semiannually.

4.2.4. Heat Balance

The power range channels 1 and 2 shall be checked against a primary system heat balance at least once each week the reactor is in operation above 100 kW in the forced convection mode.

4.2.5. Reactor Safety Channels Checks

The above specifications (4.2.1 through 4.2.4) do not apply to the following reactor safety channels: power to primary coolant pump, manual button, header air pressure, and pool water level monitor. Operation of these safety channels will be checked before each day's operation or before each operation extending more than one day.

Bases: The daily channel tests and channel checks will ensure that the safety channels are operable. The semiannual calibration will permit long-term drift of the channels to be corrected. The weekly calibration of the power measuring channels will correct for drift and ensure operation within the requirements of the license.

4.3. Emergency Core Spray System

Applicability: These specifications apply to the emergency core spray system.

Objective: The objective is to ensure that the spray systems are operable and will deliver the specified flow rate of emergency coolant.

Specifications:

4.3.1. Spray System Checks

Whenever the reactor bridge is moved and replaced into position for forced convection operation, the remote coupler for each spray system shall be air-pressure checked to ensure that there is no leakage.

4.3.2. Flow Rate Measurements

Measurements will be made annually to verify that each spray system will deliver at least 10 gpm for 30 min.

Bases: The emergency spray system is an engineered safeguard. At the initial installation, each of the two core spray systems was checked to ensure that it delivered the flow as specified in Section 3.10 of these specifications. Because there are no moving parts and no automatic electronic or mechanical mechanisms subject to failure, a verification that the remote couplers are engaged and not leaking will ensure that the two core spray systems are operable. The annual measurement of the flow rate will verify that each of the two core spray systems will deliver the flow as desired. The pre-operational test of the core spray system demonstrated that water delivery is at least 10 gpm for 30 min and 7 1/2 gpm for the next 60 min. Subsequent annual tests, which verify the 30 min flow rate, are adequate to verify design performance. The core spray system is described and the safety analysis is given in the SAR.

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4.4. Area Radiation Monitoring Equipment

Applicability: These specifications apply to the area radiation monitoring equipment required by Sections 3.2 and 3.3 of these specifications.

Objectives: The objectives are to ensure that the radiation monitoring equipment is operating and to verify appropriate alarm settings.

Specification:

4.4.1 Daily Operability Verification

The operation of the radiation monitoring equipment and the position of their associated alarm set points shall be verified daily during periods when the reactor is in operation.

4.4.2. Semiannual Calibration

Calibration of the radiation monitoring equipment shall be performed semiannually.

Bases: Surveillance of the monitoring equipment will provide assurance that sufficient warning of a potential radiation hazard is available.

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4.5. Maintenance

Applicability: This specification applies to the surveillance requirements following maintenance of control or safety systems.

Objective: The objective is to ensure that a system is operable before being used after maintenance has been performed.

Specification: Following maintenance or modification of a control or safety system or component, it shall be verified that the system is operable before it is returned to service or during its initial operation.

Bases: The intent of the specification is to ensure that work on the system or component has been properly carried out and that the system or component has been properly reinstalled or reconnected. Correct operation of some systems, such as power range monitors, cannot be verified unless the reactor is operating. Operation of these systems will be verified during their initial operation following maintenance or modification.

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4.6. Confinement

Applicability: This specification applies to the surveillance requirements for confinement of the reactor room.

Objective: The objective is to ensure that the closure equipment to the reactor room is operable.

Specifications:

4.6.1. Daily Emergency Exit Manhole Water Level Check

Before each day's operation or before each operation extending more than one day, the water level in the emergency exit manhole shall be verified.

4.6.2. Monthly Operability Tests

At least once each month, a test shall be made to ensure that the following equipment is operable:

truck door closed switch
ventilation exhaust duct door
personnel door

4.6.3. Semiannual Visual Inspections of Seals

Semiannually, a visual inspection of the seal and gaskets of the truck door, the personnel door, and the ventilation exhaust duct door shall be made to verify that they are operable.

4.6.4. UVAR Room Leak-Rate Measurement

Before operation with fueled experiments whose power generation is greater than 1 W, the UVAR room leak rate shall be verified when the interval since the last verification is greater than 12 months.

Bases: Surveillance of this equipment will verify that the confinement of the reactor room is maintained.

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4.7. Airborne Effluents

Applicability: This specification applies to the surveillance of the instrument that monitors the airborne effluents in the ventilation line from the ground floor experimental area.

Objective: The objective is to ensure that the airborne effluent monitor is operating and properly calibrated.

Specifications:

4.7.1. Monitor Channel Check

When the operation of the airborne effluent monitor is required (TS 3.4.1.), a channel check shall be performed on the monitor prior to reactor operation.

4.7.2. Monitor Calibration

A calibration of the airborne effluent monitor will be performed semiannually with a radioactive source.

Bases: The channel check of the monitor will ensure that it is operable. The semiannual calibration with an external source will permit long-term drift to be corrected. It is noted that the use of the airborne effluent monitor is required only if one of the two eight-inch neutron beamports, when in the process of being re-filled with water, is vented past the monitor to the atmosphere.

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4.8. Reactor HEU Fuel Dose Measurements

Applicability: This specification applies to the highly enriched uranium (HEU) UVAR fuel possessed under the Reactor Facility license. These specifications are applicable until all HEU UVAR fuel elements have been removed from the Reactor Facility.

Objective: The objective of this specification is to ensure that the maximum quantity of special nuclear material does not exceed the limits specified in the Reactor Facility license.

Specifications:

4.8.1. Schedule

The amount of special nuclear material (SNM) possessed at the Reactor Facility will be determined, as necessary, to ensure that limits specified by the Reactor Facility licenses are not exceeded. As a minimum, an evaluation will be completed and documented every 6 months.

4.8.2. Quantity Limits

HEU UVAR fuel elements possessed following the conversion of the UVAR to LEU fuel will be shipped away from the Reactor Facility, as necessary, to ensure that the quantity of nonexempt SNM (as defined in 10 CFR 73) does not exceed that allowed by the Reactor Facility licenses. If the amount of nonexempt SNM exceeds 5 kg the Reactor Safety Committee will be informed and the actions specified in the Physical Security Plan implemented.

4.8.3. "Self-Protection" Determinations

If HEU UVAR fuel elements have not been irradiated as a part of the UVAR core for at least one month, dose rate measurements of these HEU fuel elements will be made, as necessary, to determine which elements have dose rates higher than specified by 10 CFR 73.67(b).

Bases: The specifications provide a high degree of assurance that the amount of SNM and nonexempt SNM will not exceed the license limits. The amount of nonexempt SNM will normally be maintained at less than 5 kg, if necessary by shipping spent-fuel off-site. In the event that the 5 kg nonexempt SNM quantity is exceeded, the Reactor Safety Committee will be informed of this and the actions specified in the Physical Security Plan will be taken.

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4.9. Primary Coolant Conditions

Applicability: This specification applies to the surveillance of primary water quality.

Objective: The objective is to ensure that water quality does not deteriorate over extended periods of time if the reactor is not operated.

Specification: The conductivity and pH of the primary coolant water shall be measured at least once every 2 weeks and shall be

Conductivity	:	$\leq 5 \times 10^{-6}$ mhos/cm
pH:		between 5.0 and 7.5

Bases: Section 3.11 of these specifications ensures that the water quality is adequate during reactor operation. This section ensures that water quality is not permitted to deteriorate over extended periods of time even if the reactor does not operate.

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5.0. DESIGN FEATURES

5.1. Reactor Fuel Specifications

Applicability: These specifications apply to UVAR low enriched uranium (LEU) fuel.

Objective: The objective is to describe LEU fuel approved by the U.S. NRC for use in the UVAR.

Specifications:

5.1.1. Fuel Material

UVAR LEU fuel is of a type described for use at U.S. research reactors by the U.S. Nuclear Regulatory Commission (NUREG-1313 "Safety Evaluation Report Related to the Evaluation of LEU Silicide Aluminum-Dispersed Fuel for Use in Non-Power Reactors"). The fuel meat is U_3Si_2 dispersed in an aluminum matrix and enriched to less than 20% U-235.

5.1.2. Element Description

(1) Plate-type elements of the MTR ype are used. The fuel "meat" is clad with aluminum alloy to form flat fuel plates. The active length of the fuel region in the fuel plates is approximately 24 inches and the width is approximately 2.5 inches. The LEU fuel plates are joined at their long-side edges to two side plates. The entire fuel plate assembly is joined at the bottom to a cylindrical nose piece that fits into the UVAR core gridplate. The overall fuel element dimensions are approximately 3 inches by 3 inches by 36 inches. Each fuel plate contains 12.5 grams of U-235.

(2) "Standard" LEU fuel elements are composed of 22 parallel flat fuel plates each, and contain 275 grams of U-235.

(3) "Control-rod" LEU elements are similar to the standard elements, with the exception that they have half as many fuel plates (the 11 center plates being removed to form a channel which is bounded by 0.125 inch thick aluminum plates). Control-rod elements accommodate the control rods in the central channel. Their U-235 content is 137.5 grams.

(4) "Partial" LEU fuel elements are half-fueled elements composed of 11 LEU fuel plates and 11 unfuelled (dummy) plates. The U-235 content in these elements is 137.5 grams.

(5) "Special" LEU fuel elements have 22 fuel plates, of which 20 are removable. The maximum U-235 content in these elements is 275 grams and the minimum is 25 grams.

5.1.3. Core Configurations

A variety of UVAR core configurations may be used to accommodate experiments, but the loadings shall always be such that the minimum shutdown margin and excess reactivity specified in the UVAR Technical Specifications are not exceeded.

Bases: The NRC has described LEU silicide-fuel suitable for use in U.S. research reactors in NUREG-1313 "Safety Evaluation Report Related to the Evaluation of LEU Silicide Aluminum-Dispersed Fuel for Use in Non-Power Reactors," [$36.00, from NTIS, Springfield Va. (703-487-4650)]. Also, Bretscher and Snelgrove from the Argonne National Laboratory documented LEU fuel test results in ANL/RERTR/TM-14, "The Whole-Core LEU U_3Si_2-Al Fuel Demonstration in the 30-MW Oak Ridge Research Reactor." The LEU-SAR for the UVAR contains the safety analysis performed for the 22 flat-plate University of Virginia fuel elements. The LEU elements were designed by EG&G, Idaho, and are manufactured by the Babcock and Wilcox Company of Lynchburg, Virginia.

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5.2. Reactor Building

Applicability: This specification applies to the room containing the reactor pool, the UVAR and the UVAR control room.

Objective: The objective is to provide a description of the UVAR confinement.

Specifications:

5.2.1. Confinement

The reactor shall be housed in a room designed to restrict leakage, as stated in Section 3.7(1)(d) of these specifications.

5.2.2. Ventilation

The reactor room shall be equipped with a ventilation system designed to exhaust air or other gases from the reactor room through a stack at a minimum of 37 ft above ground level.

5.2.3. Free Volume

The minimum free volume of the reactor room shall be 60,000 ft^3.

Bases: The parameters specified were used in the analysis presented in the SAR for the UVAR.

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5.3. Fuel Use and Storage

Applicability: The specifications below apply to University of Virginia Reactor fuel used and/or stored at the University of Virginia Reactor Facility following conversion of the UVAR to low enriched uranium (LEU) reactor fuel.

Objective: The objective is to describe reactor fuel which may be used, possessed and/or stored at the University of Virginia Reactor Facility as well as measures that avoid nuclear criticality or fuel-related accidents.

Specifications:

5.3.1. HEU Possession Limit

Following UVAR conversion to LEU:

(a) A maximum of 4.5 kilograms of contained uranium-235 at 20% or greater enrichment (which is defined as highly-enriched uranium, HEU) in the form of MTR-type reactor fuel elements may be possessed at the Reactor Facility but not used in the UVAR core, until this existing inventory of fuel is removed from the Reactor Facility.

(b) Until all HEU reactor fuel elements have been shipped offsite, HEU not in the form of reactor fuel may be possessed and used in amounts such that the total amount of HEU present at the Reactor Facility at any time, in any form, is less than 5 kilograms of uranium-235.

(c) After all HEU reactor fuel elements have been shipped offsite, the uranium-235 limit for HEU present in fission chambers, flux foils, powder form, and any other form excluding reactor fuel elements, shall be 1 kilogram. Also, the limit for uranium-235 for all enrichments shall be 12 kilograms from that moment forward.

5.3.2. LEU Possession Limit

A maximum of 11 kilograms of contained uranium-235 at less than 20% enrichment (which is defined as low enriched uranium, LEU) may be possessed and used at the Reactor Facility.

5.3.3. Plutonium Possession Limit

All plutonium generated or present in UVAR LEU reactor fuel, start-up sources, irradiation targets, flux foils and fission chambers may be possessed and used. Following conversion of the UVAR to LEU fuel, all plutonium present in HEU reactor fuel elements may be possessed until the HEU reactor fuel elements are shipped offsite.

5.3.4. <u>Storage Reactivity Limitation</u>

All reactor fuel elements, including fueled experiments and fuel devices <u>not in the reactor</u>, shall be stored in a geometric array where calculated k_{eff} is no greater than 0.9, for all conditions of moderation and reflection using light water, except in cases where an approved fuel shipping container is used, in which case the calculated k_{eff} for the container shall apply.

5.3.5. <u>Storage Cooling Requirement</u>

Irradiated fuel elements and fueled devices shall be stored in an array that will permit sufficient natural convection cooling by water or air so that the fuel element or fueled device surface temperature will not exceed the boiling point of water.

<u>Bases</u>: Section 5.4 of the American National Standard ANSI/ANS-15.1-1990, "The Development of Technical Specifications for Research Reactors," was used as the overall basis for the above specifications. The specification 5.3.1 dealing with HEU fuel is made pursuant to the Atomic Energy Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." The limit given in specification 5.3.2 is based on an estimated reasonable need for reactor fuel for use in the core and a spare fuel requirement determined by DOE's expected spare fuel manufacturing schedule. The specification in 5.3.3 is based on the unavoidable production of small amounts of plutonium in reactor fuel, sources, irradiation targets, flux foils and fission chambers, as a consequence of normal reactor operation. Precise amounts of plutonium produced, decayed or burned during reactor operation can't be quantified, and this is not necessary for the small amounts of plutonium produced and contained in these aforementioned devices pose no undue reactor or radiation safety risks.

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6.0. ADMINISTRATIVE CONTROLS

Applicability: The specifications listed below in TS 6.1.1 through TS 6.1.4 apply to the organizational structure of the University of Virginia as it relates to the operation of the Reactor Facility.

Objective: The objective is to describe the chain of command having responsibility for the safe operation of the Reactor Facility. At the various administrative levels, the functions, assignments, responsibilities and associated professional background, training and requalification requirements are listed, as applicable.

Specifications:

6.1. Organization

6.1.1. Structure

The Reactor Facility shall be an integral part of the School of Engineering and Applied Science of the University of Virginia. The organizational structure of U.VA. relating to the Reactor Facility is shown in Figure 6.1. The Chair of the Department of Mechanical, Aerospace and Nuclear Engineering will have overall responsibility for management of the Facility (Level 1).

6.1.2. Responsibility

The Reactor Facility Director shall be responsible for the overall facility operation (Level 2). During periods when the Reactor Facility Director is absent, his responsibilities are delegated to the Reactor Supervisor (Level 3).

The Reactor Facility Director shall have at least a bachelor degree in science or engineering and have a minimum of 5 years of experience in the nuclear field. A graduate degree may fulfill 4 years of experience on a one-for-one time basis.

The Reactor Supervisor(s) shall be responsible for the day-to-day operation of the UVAR and for ensuring 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 Committee. During periods when the Reactor Supervisor(s) is (are) absent, his responsibilities are delegated to a person holding a Senior Reactor Operator license (Level 4).

A Reactor Supervisor shall have the equivalent of a bachelor degree in science or engineering and have at least 2 years of experience in Reactor Operations at this facility, or an equivalent facility, or at least 6 years of experience in Reactor Operations. Equivalent education or experience may be substituted for a degree. Within nine months after being assigned to the position, the Reactor Supervisor shall obtain and maintain an NRC Senior Operator license.

6.1.3. Staffing

When the reactor is operating the following conditions will be met:

(1) A licensed Senior Reactor Operator or a licensed Reactor Operator shall be present at the reactor controls.

(2) A licensed Senior Reactor Operator shall be on call, but not necessarily at the Reactor Facility.

(3) At least one other person, not necessarily licensed to operate the reactor, shall be present at the Reactor Facility.

(4) All rearrangements of the core or other non-routine actions shall be supervised by a licensed Senior Reactor Operator.

(5) One or more health physicists, organizationally independent of the Reactor Staff as shown in Figure 6.1, shall be responsible for radiological safety at the Reactor Facility.

6.1.4. Selection and Training of Personnel

The selection, training and requalification of operations personnel shall meet the requirements of the American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1988, Sections 4-6.

Bases: Sections 6.1, 6.1.1, 6.1.2, 6.1.3 and 6.1.4 of the American National Standard ANSI/ANS 15.1-1990 "The Development of Technical Specifications for Research Reactors," describe a generic and generally acceptable organizational structure for U.S. research reactors. They provide the bases for TS 6.1 above.

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6.2. Reactor Safety Committee

Applicability: The specifications 6.2.1 through 6.2.3 apply to the expert group who will provide specific reviews and audits of Reactor Facility operations.

Objective: There shall be a Reactor Safety Committee (RSC) to review and audit reactor operations and ensure that the Reactor Facility is operated in a safe manner within the terms of the reactor license. Collectively, the committee members shall represent a broad spectrum of expertise in the research-reactor field. The members may be drawn from within or outside the operating organization. The RSC reviews and audits are designed to uncover deficiencies that affect reactor safety. The Reactor Safety Committee is part of the Radiation Safety Committee and shall report to the Chair of the Radiation Safety Committee, who is the coordinator for all licenses involving the use of radioactive materials and radiation producing equipment. The RSC shall advise the Chair of the Department of Mechanical, Aerospace and Nuclear Engineering, and the Director of the Reactor Facility, on those areas of responsibility specified below.

Specifications:

6.2.1. Composition and Qualification

The Committee shall be composed of at least five members, and shall include the Radiation Safety Officer of the University and the Director of the Reactor Facility. The Reactor Director shall be the sole reactor staff representative on the Committee. The membership of the Committee shall be such as to maintain a degree of technical proficiency in areas relating to reactor operation and reactor safety.

6.2.2. Charter and Rules

(1) A quorum of the Committee shall consist of not less than the majority of the full committee. The Chair can designate another member from the Committee to preside in his absence.

(2) The Committee shall meet at least semiannually and shall be on call by the Chair. Minutes of all meetings shall be disseminated as designated by the Chair.

(3) The Committee shall have a written charter defining such matters as the authority of the Committee, the subjects within its purview, and other administrative provisions as are required for effective functioning of the Committee.

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6.2.3. Review and Audit Functions

As a minimum the responsibilities of the Reactor Safety Committee include:

(1) Review and approval of untried experiments and tests that are significantly different from those previously used or tested in the reactor, as determined by the Facility Director.

(2) Review and approval of changes to the reactor core, reactor systems or design features that may affect the safety of the reactor.

(3) Review and approve all proposed amendments to the reactor license, Technical Specifications, and changes to the standard operating procedures (Note: SOPs are discussed in Section 6.3 of these specifications).

(4) Review reportable occurrences and the actions taken to identify and correct the cause of the occurrences.

(5) Review significant operating abnormalities or deviations from normal performance of facility equipment that affect reactor safety.

(6) Review reactor operation and audit the operational records for compliance with reactor procedures, Technical Specifications, and license provisions. Audits consist of spot checks of reactor staff compliance with SOP's, Technical Specifications and licenses.

Bases: American National Standard ANSI/ANS-15.1-1990, "The Development of Technical Specifications for Research Reactors," describes in Section 6.2 acceptable composition and qualification criteria for reactor safety committees and their review and audit functions. Section 6.3 of the standard describes the organizational relationship of the group responsible for radiation safety to the reactor operations group. These sections of the standard are used as bases for the specifications listed above.

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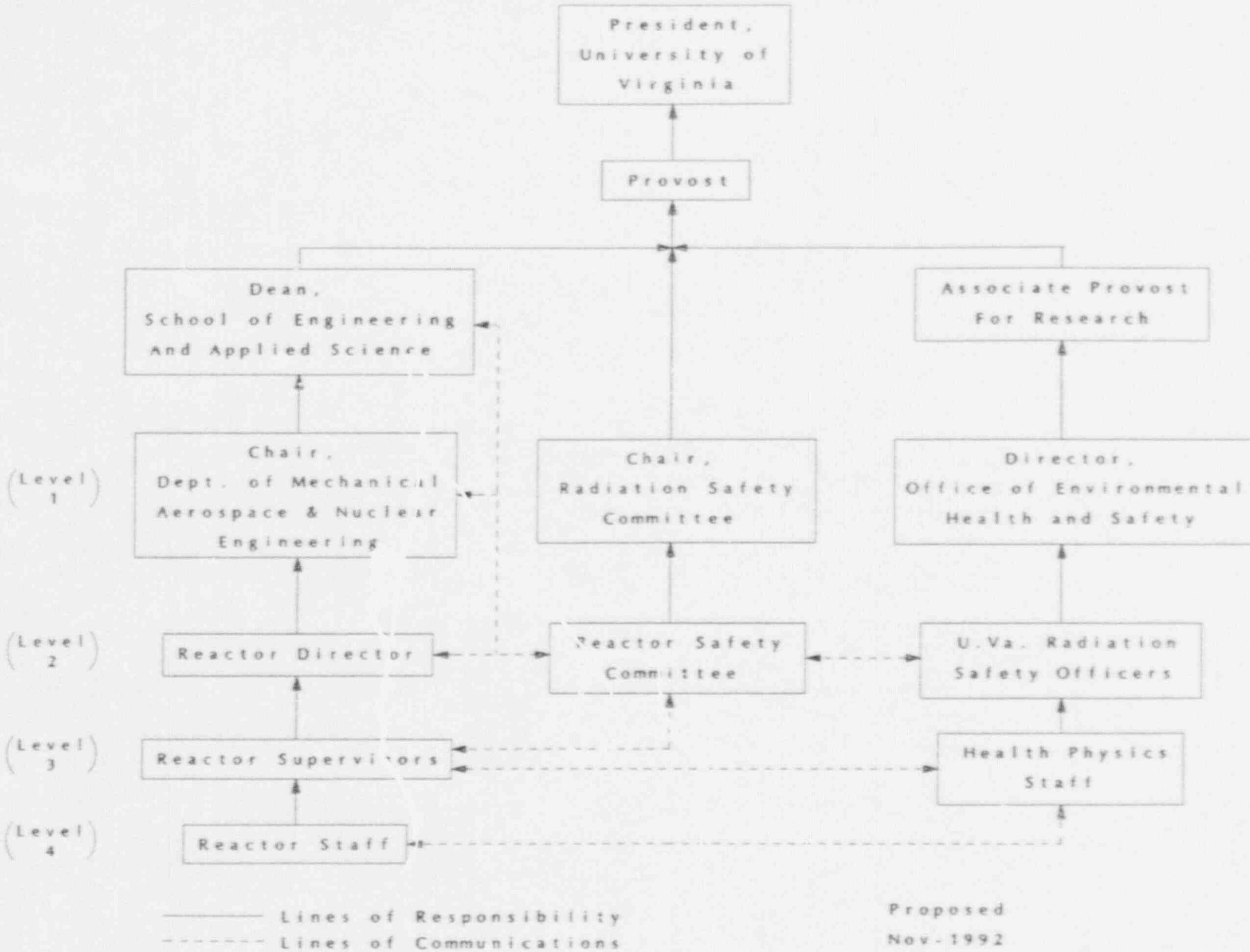

Figure 6.1 Organizational Structure of the U.Va. Research Reactor Facility

6.3. Standard Operating Procedures

Applicability: The specification below concerns the procedural controls used to operate the University of Virginia Reactor (UVAR) and conduct experiments.

Objective: The objective is the safe operation of the reactor in compliance with license conditions and federal regulations.

Specifications:

6.3.1. Items Covered by SOPs

Written procedures, reviewed and approved by the Reactor Safety Committee, shall be in effect and followed for the items listed below. These procedures shall be adequate to ensure 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 foreseen potential malfunctions of systems or components, including responses to alarms, suspected primary coolant system leaks, abnormal reactivity changes.

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

(5) Preventative and corrective maintenance operations that could have an effect on reactor safety.

(6) Periodic surveillance (including test and calibration) of reactor instrumentation and safety systems.

(7) Radiation control.

6.3.2. Changes to SOPs

Substantive changes to approved procedures shall be made only with the approval of the Reactor Safety Committee. Changes that do not change the original intent of the procedures may be made with the approval of the Facility Director. All such minor changes to procedure shall be documented and subsequently reviewed by the Reactor Safety Committee.

Basis: Section 6.4 of American National Standard ANSI/ANS 15.1-1990, "The Development of Technical Specifications for Research Reactors," suggests acceptable procedural controls to be applied to U.S. research reactors. This section of that standard is the basis for the above specification.

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6.4. Review and Approval of Experiments

Applicability: Specifications 6.4.1 through 6.4.7 listed below apply to classes of experiments run in the UVAR core, in the UVAR pool, or which use UVAR-generated neutron and/or gamma-radiation beams. However, a partial listing of examples of experimental work covered under experiment classes for which broad approval may have been obtained and, therefore, for which individualized experimental procedures would not be required follows below:

(a) Samples to be irradiated in approved irradiation facilities, such as the neutron activation facilities, where the samples meet the criteria in TS 3.6 and TS 6.4.

(b) Samples to be irradiated in the neutron radiography facility beamport which are known not to be hazardous to reactor safety.

Objective: The objective is the safe operation of the reactor and experiments, in accordance with license conditions and federal regulations. Experiments run in conjunction with the reactor should not adversely affect reactor and radiation safety. Notwithstanding the regard for safety, the requirement for review and approval of experiments shall not limit the flexibility of experimenters performing work covered under general written procedures, or for which unanalyzed safety issues do not exist, as determined by the Reactor Director.

Specifications:

6.4.1. Experimental Procedures and Methods

(1) Classes of experiments involving the UVAR, the UVAR pool or UVAR radiation beam facilities shall be carried out with established and approved written experimental procedures. The Reactor Safety Committee shall review all new classes of experiments prior to their initiation and approve written experimental procedures governing their operation.

(2) Written experimental methods that implement Reactor Safety Committee approved experimental procedures may be developed by the staff and/or experimenters, as needed. Such experimental methods shall be approved by a Reactor Supervisor or the Reactor Director prior to use.

(3) The Reactor Director or the Reactor Safety Committee shall decide whether an experimental procedure is required. Usually, an experimental procedure will not be required if the work in question is already covered under an existing approved general experimental procedure or by a Standard Operating Procedure.

6.4.2. Reactivity limits

As applicable, reactivity limits for experiments given in experimental procedures shall be based on analyses of maximum reactivity insertions that can be handled by the reactor or its control and safety systems without exceeding safety limits. Reactivity limits have been established in TS 3.6 Limitations on Experiments for maximum absolute reactivity worth of individual experiments and the sum of the absolute values of the worth of all experiments.

6.4.3. Materials

As applicable, special requirements shall be established in the experimental procedures for significant amounts of special materials such as fissionable materials, explosives or metastable materials capable of significant energy release, or materials that are corrosive to reactor components or highly reactive with coolants. Requirements listed in experimental procedures may range from detailed analyses to double encapsulation and prototype testing.

6.4.5. Failure and Malfunctions

(1) Credible failures of any experiments shall not result in the release or exposures in excess of established limits nor in excess of the annual limits established in Title 10, Code of Federal Regulations, Part 20.

(2) Experiments shall be designed such that they will not contribute to the failure of other experiments, core components, or principal physical barriers to uncontrolled release of radioactivity. Similarly, no reactor transient shall cause an experiment to fail in such a way as to contribute to an accident.

6.4.6. Experimental Facility Specific LCO

Limiting Conditions of Operation limits unique to an experiment shall be specified, as necessary, in the written experimental procedures. Specific surveillance activities which may be required for experiments will also be addressed in the experimental procedures.

6.4.7. Deviations from Experimental Procedures

(1) Changes to previously approved experiments and experimental procedures, determined by the Reactor Director to be substantive, shall be made only after review and approval by the Reactor Safety Committee.

(2) Minor changes to experimental procedures may be made with the approval of the Reactor Director, who will determine that no new reactor safety concerns exist, and with the approval of the Reactor Health Physicist, who will assure that radiological safety requirements can be met.

Bases: National Standard ANSI/ANS-15.1-1990, "The Development of Technical Specifications for Research Reactors," suggests acceptable provisions governing reactor-based experiments in sections 3.8, 4.8, and 6.5. These sections served as bases for the above specification. In addition, examples are presented of activities involving the reactor which typically do not require individualized written procedures, because they are covered under a general procedure for an approved class of experiments, or covered by SOPs. It is unreasonable to require procedures with undue specificity when this would limit reasonable experimental flexibility and no unanalyzed safety issues exist. The Reactor Director has the resources and authority to determine when experimental procedures are required.

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6.5. Plant Operating Records

Applicability: The specifications below apply to UVAR operating records.

Objective: The objective is to maintain and keep on file reactor operating records necessary for future reference and for demonstration of compliance with license conditions and federal regulations.

Specifications:

6.5.1. Records To Be Retained for a Period of at Least Five Years

In addition to the requirements of applicable regulations, records of the items listed below shall be kept in a manner convenient for review and shall be retained as indicated:

(1) Normal reactor facility operation (for example, reactor logbooks, reactor checklists and irradiation request forms).

(2) Principal reactor systems maintenance records.

(3) Reportable occurrences.

(4) Equipment and component surveillance activity required by Technical Specifications.

(5) Reactor Facility radiation and contamination surveys.

(6) Experiments performed with the UVAR.

(7) Fuel inventories, transfers of radioactive material to and from the R-66 license.

(8) Approved changes to operating procedures.

(9) Records of meetings and audit reports of the Reactor Safety Committee.

6.5.2. Records To Be Retained for One Certification Cycle

Records of retraining and requalification of licensed operators shall be maintained at all times the individual is employed or until licensing is renewed.

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6.5.3. Records To Be Retained for the Life of the Facility

In addition to the requirements of applicable regulations, records (or logs) of the items listed below shall be kept in a manner convenient for review and shall be retained as indicated:

(1) Gaseous and liquid radioactive effluents released from the Reactor Facility.

(2) Off-site (radiological) environmental monitoring surveys.

(3) Radiation exposures for all personnel monitored at the Reactor Facility.

(4) Updated, corrected and as-built drawings of the Reactor Facility.

Basis: American National Standard ANSI/ANS-15.1-1990, "The Development of Technical Specifications for Research Reactors," provides record-keeping guidance in Section 6.8. This is the basis for the above specifications.

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6.6. Required Actions

Applicability: The specifications below apply to instances where reactor safety limits have been exceeded, or radiologically unsafe situations have been, or were likely to have been, generated.

Objective: The objective is to report safety limit violations or unsafe conditions, study their causes and consequences, determine their effect on the health and safety of personnel and the public, and take corrective action to prevent recurrence.

Specifications:

6.6.1. Actions To Be Taken in the Event the Safety Limit is Exceeded

In the event the safety limit is violated, the following actions shall be taken:

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

(2) The occurrence shall be reported to the Reactor Facility Director and the Chair of the Reactor Safety Committee, or their designees, as soon as possible, but not later than the next work day. Reports shall be made to the Commission in accordance with Section 6.7 of these specifications.

(3) A written safety limit violation report shall be made that shall include an analysis of the causes of the violation and extent of resulting damage to reactor components, systems, or structures; corrective actions taken; and recommendations for measures to preclude reoccurrence. This report shall be submitted to the Reactor Safety Committee for review.

6.6.2. Action To Be Taken in the Event of a Reportable Occurrence

A reportable occurrence is any of the following conditions:

(1) Safety system setting less conservative than specified in Section 2.2 of these specifications.

(2) Operating in violation of a Limiting Condition of Operation (LCO) established in these specifications, unless prompt remedial action is taken.

(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 safety function, unless immediate shutdown of the reactor is initiated.

(4) An uncontrolled or unanticipated increase in reactivity in excess of 0.70$.

(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.

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

(7) Major damage to the Co-60 rods resulting in Co-60 concentrations in reactor pool water in excess of 1×10^{3} micro-curies/ml.

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

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

(b) 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 recommendations for measures to preclude or reduce the probability of reoccurrence. This report shall be submitted to the Director and the Reactor Safety Committee for review.

(c) A report shall be submitted to the Nuclear Regulatory Commission in accordance with Section 6.7 of these specifications.

<u>Bases</u>: National Standard ANSI/ANS-15.1-1990, "The Development of Technical Specifications for Research Reactors," describes in sections 6.6 and 6.7 acceptable specifications for required actions related to safety limits violations, actions to be taken upon their discovery, and reporting requirements. These form the bases for the above specifications.

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6.7. Reporting Requirements

Applicability: The specifications 6.7.1 and 6.7.2 listed below apply to routine and special reports made by the University ˙Virginia Reactor Facility to the U.S. Nuclear Regulatory Commission.

Objective: The objective is to provide the licensing agency (NRC) with relevant information concerning normal and abnormal reactor operations which are necessary for the fulfillment of its mission to protect the public health and safety. A secondary objective is to comply with reporting requirements as given in the federal regulations.

Specifications: In addition to federal regulatory requirements (for example, follow 10 CFR 20, 30.50, 40.60, and 70.50, as applicable), reports should be made to the U.S. Nuclear Regulatory Commission as follows:

6.7.1. Reporting of Incidents

(1) Immediate notification should be made by telephone, to the U.S. Nuclear Regulatory Commission, Region II, as well as to the NRC Headquarters Operations Center of:

(a) Personnel exposures or releases of radioactive material greater than the limits in 10 CFR 20.

(2) A special report should be made by telephone as soon as possible, but no later than the next working day, to the U.S. Nuclear Regulatory Commission, Region II as well as to the NRC Headquarters Operations Center of:

(a) Personnel exposures or releases of radioactive material greater than the limits in 10 CFR 20.

(b) Reportable occurrences as defined in Section 6.6.2 of these specifications.

(c) Violation of a safety limit.

(3) A special written report should be sent by mail within 14 days to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555 with a copy to the NRC Region II Regional Administrator of:

(a) Accidental off-site release of radioactivity above permissible limits, whether or not the release result d in property damage, personal injury, or exposure.

(b) Reportable occurrence as defined in Section 6.6.2 of these specifications.

(c) Violation of a safety limit.

(4) A special written report should be sent by mail within 30 days to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the NRC Region II Regional Administrator of:

(a) Substantial variance from performance specifications contained in these specifications or in the UVAR SAR.

(b) Significant change in the transient or accident analyses as described in the UVAR SAR.

(c) Changes in personnel serving as Chair of the Department of Mechanical, Aerospace and Nuclear Engineering, Reactor Facility Director, or Reactor Supervisor.

(5) A written report should be sent within nine months after initial criticality of the reactor or within 90 days of completion of the startup test programs, whichever is earlier, to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the NRC Region II Regional Administrator, upon receipt of a new facility license, an amendment to the license authorizing an increase in power level or the installation of a new core with fuel elements of a design different design than previously used. The report will include the measured values of the operating conditions or characteristics of the reactor under the new conditions, including:

(a) Total control rod reactivity worth.

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

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

6.7.2. <u>Routine Reports</u>

A routine report will be made by March 31 of each year to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the NRC Region II Regional Administrator, providing the following information:

(1) A narrative summary of operating experience (including experiments performed), and of changes in reactor design, performance characteristics, and operating procedures related to the reactor safety occurring during the reporting period.

(2) A tabulation showing the energy generated by the reactor (in megawatt hours) and the number of hours the reactor was critical each quarter during the year.

(3) A report of the results of the safety-related maintenance and inspections. The reasons for corrective maintenance of safety-related items will be included.

(4) A report of the number of emergency shutdowns and inadvertent scrams, including their reasons and the corrective actions taken.

(5) A summary of 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 10 CFR 50.

(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 report with a description of environmental surveys performed outside the Reactor Facility.

(8) A summary of radiation exposures received by Reactor Facility personnel and visitors, including the dates and time of significant exposures (greater than 500 mrem for adults and 50 mrem for persons under 18 years of age) and a summary of the results of radiation and contamination surveys performed within the Reactor Facility.

Bases: National Standard ANSI/ANS-15.1-1990, "The Development of Technical Specifications for Research Reactors," describes in sections 6.6 and 6.7 acceptable specifications for required actions related to violations of safety limits, actions to be taken upon their discovery, and reporting requirements. These, and applicable federal regulations, form the bases for the above specifications.

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