TABLE OF CONTENTS

1.0 DEFINITIONS

1.1 Certified Operator

An individual authorized by the U. S. Nuclear Regulatory Commission to carry out the responsibilities associated with the position requiring the certification.

1.1.1 Senior Reactor Operator

An individual who is licensed to direct the activities of reactor operators. Such an individual may be referred to as a class A operator.

1.1.2 Reactor Operator

An individual who is licensed to manipulate the controls of a reactor. Such an individual may be referred to as a class B operator.

1.2 Confinement

Confinement means an enclosure on the overall facility which controls the movement of air into it and out through a controlled path.

1.3 Experiment

Any operation, component, or target (excluding devices such as detectors, foils, etc.), which is designed to investigate non-routine reactor characteristics or which is intended for irradiation within the pool, on or in a beam tube or irradiation facility and which is not rigidly secured to a core or shield structure so as to be part of their design.

1.3.1 Experiment, Moveable

A moveable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

1.3.2 Experiment, Secured

A secured experiment is any experiment, experiment facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining force must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible conditions.

1.3.3 Experimental Facilities

An experimental facility is any structure or device which is intended to guide, orient, position, manipulate, or otherwise facilitate a multiplicity of experiments of similar character.

1.4 Explosive Material

Explosive material is any solid or liquid which is categorized as a severe, dangerous, or very dangerous explosion hazard in DANGEROUS PROPERTIES OF INDUSTRIAL MATERIALS by N.I. Sax, third Ed. (1968), or is given an identification of Reactivity (Stability) Index of 2, 3, or 4 by the National Fire Protection Association in its publication 704-M, 1966.

1.5 Instrumentation Channel

A channel is the combination of sensor, line, amplifier, and output device which are connected for the purpose of measuring the value of a parameter.

1.5.1 Channel Test

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

1.5.2 Channel Check

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

1.5.3 Channel Calibration

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

1.6 Limiting Conditions of Operation (LCO)

Lowest functional capability or performance levels of equipment required for safe operation of the reactor (10CFR50.36).

1.7 Limiting Safety System Setting (LSSS)

Settings for automatic protective devices related to those variables having significant safety functions, and chosen so that automatic protective action will correct an abnormal situation before a safety limit is exceeded (10CFR50.36).

1.8 Measured Channel

A measured channel is the combination of sensor, amplifiers, and output devices which are used for the purpose of measuring the value of a parameter.

1.9 Measured Value

The measured value of a parameter is the value of the variable as indicated by a measuring channel.

1.10 Operable

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

1.11 Operating

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

1.12 Operational Reactor Core

An operational core is a standard core for which the core parameters of excess reactivity, shutdown margin, fuel temperature, power calibration, and reactivity worths of control rods and experiments have been determined to satisfy the requirements set forth in the Technical Specifications.

1.13 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 facility having reached a specified limit.

1.14 Reactivity Excess

Excess reactivity is that amount of reactivity that would exist if all the control rods were moved to the maximum reactive condition from the point where the reactor is exactly critical.

1.15 Reactivity Limits

The reactivity limits are those limits imposed on the reactor core excess reactivity. Quantities are referenced to a reference core condition.

1.16 Reactivity Worth of an Experiment

The reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter equipment position or configuration.

1.17 Reactor Operating

The reactor is operating whenever it is not secured or shut down.

1.18 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.

1.19 Reactor Secure

The reactor is secure when:

1.19.1 Subcritical:

There is insufficient fissile material or moderator present in the reactor, control rods or adjacent experiments, to attain criticality under optimum available conditions of moderation and reflection, or the following conditions exist:

a. The minimum number of neutron absorbing control rods are fully inserted in shutdown position, as required by technical specifications.

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

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.

d. No experiments are being moved or serviced.

1.20 Reactor Shutdown

The reactor is shut down if it is subcritical by at least the shutdown margin in the reference core condition with the reactivity of all installed experiments included.

1.21 Readily Available on Call

Readily available on call shall mean a licensed senior operator shall insure that he can be contacted within ten minutes and is within a 30 minute driving time from the reactor building when the reactor is being operated by a licensed operator.

1.22 Reference Core Condition

The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible $<.05\%$ $\Delta k/\kappa$.

1.23 Regulating Blade

The regulating blade is a control blade of low reactivity worth fabricated from stainless steel and used to control reactor power. The blade may be controlled by the operator with a manual switch or by an automatic controller.

1.24 Removable Experiment

A removable experiment is any experiment, experimental facility, or component of an experiment, other than a permanently attached appurtenance to the reactor system, which can reasonably be anticipated to be moved one or more times during the life of the reactor.

1.25 Reportable Occurrence

A reportable occurrence is any of the following:

1. A safety system setting less conservative than the limiting setting established in the Technical Specifications;

2. Operation in violation of a limiting condition for operation established in the Technical Specifications;

3. A safety system component malfunction or other component or system malfunction which could, or threaten to, render the safety system incapable of performing its intended safety functions;

4. Release of fission products from a failed fuel element;

5. An uncontrolled or unplanned release of radioactive material from the restricted area of the facility;

6. An uncontrolled or unplanned release of radioactive material which results in concentrations of radioactive materials within the restricted area in excess of the limits specified in Appendix B, Table 1 of 10 CFR20;

7. An uncontrolled or unanticipated change in reactivity in excess of 0.5 %ΔK/K;

8. Conditions arising from natural or man-made events that affect or threaten to affect the safe operation of the facility;

9. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or threatens to cause the existence or development of an unsafe condition in connection with the operation of the facility.

1.26 Research Reactor

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

1.27 Rundown

A rundown is the automatic insertion of the shim safety blades.

1.28 Safety Channel

A safety channel is a measuring channel in the reactor safety system.

1.29 Safety Limits

Safety limits are limits on important process variables which are found to be necessary to reasonably protect the integrity of the principal barriers which guard against the uncontrolled release of radioactivity. The principal barrier is the fuel element cladding.

1.30 Scram Time

Scram time is the elapsed time between reaching a limiting safety system set point and specified control rod movement.

1.31 Shim Safety Blade

A shim safety blade is a control blade fabricated from borated aluminum which is used to compensate for fuel burnup, temperature, and poison effects. A shim safety blade is magnetically coupled to its drive unit allowing it to perform the function of a safety blade when the magnet is deenergized.

1.32 Shall, Should and May

The word "shall" is used to denote a requirement. The word "should" is used to denote a recommendation. The word "may" is used to denote permission, neither a requirement nor a recommendation.

1.33 Shutdown Margin

Shutdown margin shall mean 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 most reactive rod in its most reactive position, and that the reactor will remain subcritical without further operator action.

1.34 Secured Experiment

Any experiment, experimental facility, or component of an experiment is deemed to be secured, or in a secured position, if it is held in a stationary position relative to the reactor by mechanical means. The restraint shall exert sufficient force on the experiment to overcome the expected effects of hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or of forces which might arise as a result of credible malfunctions.

1.35 Static Reactivity Worth

The static reactivity worth of an experiment is the absolute value of the reactivity change which is measurable by calibrated control rod comparison methods.

1.36 Standard Reactor Core

A standard core is an arrangement of (14) 22-plate LEU fuel elements in the reactor grid plate and may include installed experiments.

1.37 Surveillance Activities

Surveillance activities (except those specifically required for safety when the reactor is shutdown), may be deferred during reactor shutdown, however, they must be completed prior to reactor startup unless reactor operation is necessary for performance of the activity. Surveillance activities scheduled to occur during an operating cycle which cannot be performed with the reactor operating may be deferred to the end of the cycle.

1.38 Surveillance Intervals

Maximum intervals are to provide operational flexibility and to reduce frequency. Established frequencies shall be maintained over the long term. Allowable surveillance intervals shall not exceed the following:

1. 5 years (interval not to exceed 6 years).

2. 2 years (interval not to exceed 2 1/2 years).

3. Annual (interval not to exceed 15 months).

4. Semiannual (interval not to exceed 7 1/2 months).

5. Quarterly (interval not to exceed 4 months).

6. Monthly (interval not to exceed 6 weeks).

7. Weekly (interval not to exceed 10 days).

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

1.39 True Value

The true value is the actual value of a parameter.

1.40 Unscheduled Shutdown

An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation, not including shutdowns which occur during testing or check-out operations.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

2.1.1 Safety Limits in the Forced Convection Mode

Applicability:

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

Reactor Thermal Power, P
Reactor Coolant Flow through the Core, m
Reactor Coolant inlet Temperature, T_i
Height of Water Above the Top of the Core, H

Objective:

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

Specifications:

1. The true value of reactor power (P) shall not exceed 2.4 MW and the true value of flow (m) shall not be less than 1580 gpm.

2. The true value of reactor coolant inlet temperature (T_i) at power levels up to 2 MW shall not exceed 115°F.

3. The true value of the reactor coolant outlet temperature shall not exceed 125°F.

4. The true value of water height above the active core (H) shall not be less than 23.54 feet while the reactor is operating at any power level.

Bases:

The basis for forced convection safety limits is that the calculated maximum cladding temperature in the hot channel of the most compact core will not be exceeded. The inlet temperature shall apply to all power levels up to the 2 MW licensed power level. The thermal hydraulic analysis (Part B, of the SAR) shows that with this inlet temperature, the outlet temperature will not exceed the outlet temperature scram setting technical specification even at the safety limit of 2.4 MW.

2.1.2 Safety Limits in the Natural Convection Mode

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:

Reactor Thermal Power, P
Height of Water Above the Top of the Core, H
Pool Temperature, T_p

Objective:

1. To assure that the integrity of the fuel clad is maintained.

2. To assure consistency with other defined safety system parameters.

Specification:

1. The true value of the reactor thermal power shall not exceed 217 kw.

2. The height of pool water above the core shall not be less than 23.54 feet.

3. The pool temperature does not exceed 130°F.

Bases:

The basis for natural convection safety limits is that the calculated maximum cladding temperature in the hot channel of the most compact core will not reach nucleate boiling of the water coolant at a pool depth of 23.54 feet.

2.2 Limiting Safety System Settings (LSSS)

2.2.1 Limiting Safety System Setting in the Forced Convection Mode

Applicability:

LEU Fuel Temperature - Forced Convection Mode

Objective:

This specification applies to the setpoint for the safety channels monitoring reactor power, primary coolant flow, pool level and core inlet and outlet temperatures to assure that the maximum fuel temperature permitted is such that no damage to the fuel cladding will result in the forced convection mode.

Specification:

The limiting safety system settings for reactor thermal power (P), primary coolant flow through the core (m), height of water above the top of the core (H), and reactor coolant inlet (T_i) and exit temperatures (Te) shall be as follows:

Parameter	LSSS
P (Max)	2.30 MW Max
m (Min)	1600.00 gpm
H (Min)	23.70 ft
T_i (Max)	111.0°F
T_e (Max)	121.0°F

Bases:

These specifications were determined to prevent fuel temperatures from exceeding NRC temperature limits of 530°C. This temperature was found acceptable as a result of NUREG-1313, "Safety Evaluation Report related to the Evaluation of Low-Enriched Uranium Silicide-aluminum Dispersion Fuel for Use in Nonpower Reactors". The SAR (Part B) provides the analyses showing fuel cladding temperatures well below the NUREG limit at normal operation. Flow and temperature limits were chosen to prevent incipient boiling even if transient power rises to the 2 MW trip limit of 2.3 MW. Variables used in the SAR were analyzed using uncertainties in flow measurement (3%) and temperature measurement (3%). These uncertainties were incorporated in the hot channel factors (1) used in the SAR thermal hydraulic studies. These same uncertainties were applied to the inlet and outlet temperature measurements.

The LSSS for the pool level is set for a scram upon a 2" drop in water level. The reference height of 23.7 is the depth of water above the top of the active fuel setting in the existing reactor grid box. A 2" drop in water level causes a scram (H = 23.54). This depth was used in the SAR Loss of Coolant Analysis (Part B of the SAR). The safety limit settings chosen provide acceptable safety margins to the maximum fuel cladding temperature. The startup accident transient analysis (Part A, Section XI of the SAR) also provides results showing that the cladding temperature limit is not exceeded. The LOCA analysis (Design Basis Accident, Part A, Section IX and Part B, Section X and Appendix D of the SAR) shows that the fuel cladding limit is not exceeded.

The LSSS for the pool level results in a higher number since the pool level scrams upon a 2" drop in water level.

(1)Reference: Report on the Determination of Hot Spot Factors for the RINSC Research Reactor, August, 1989

2.2.2 Limiting Safety System Settings in the Natural Convection Flow Mode

Applicability:

These specifications apply to the setpoint for the safety channels monitoring reactor thermal power level (P), monitors for pool level(H), and pool water temperature (T_p) in the natural convection mode.

Objective:

To assure that automatic protective action is initiated to prevent a safety limit from being exceeded.

Specification:

1. The limiting safety system setting for reactor thermal power (P), height of water above the top of the core (H), and pool water temperature (T_p) shall be as follows:

Parameter	LSSS
P (Max)	115.0 kw
H (Min)	23.7 ft. above top of the active core
T_p (Max)	126.0°F

Bases:

The SAR has determined that up to 217 kw can be removed by natural convection, however, the existing license requirement of 100 kw operation will be maintained and with a 15% overpower trip, 115 kw will be the LSSS. The pool level scram (2" drop) is the same as the forced convection mode. The pool temperature 130°F safety limit, having a 3% error, results in a LSSS of 126°F. The LSSS for natural convection assures that automatic protective action will prevent a safety limit from being exceeded.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity Limits

Applicability:

This specification applies to the reactivity of the reactor core and to the reactivity worths of control rods and experiments.

Objective:

To assure that the reactor can be controlled and shut down at all times and that the safety limits will not be exceeded.

Specification:

1. The shutdown margin relative to the reference core condition shall be at least 1.0 %ΔK/K with the most reactive shim safety rod and the regulating rod fully withdrawn.

2. The overall core excess reactivity including movable experiments shall not exceed 4.7 %ΔK/K.

3. The total reactivity worth of all experiments shall not exceed 0.6 %ΔK/K.

4. The reactivity worth of each experiment shall be limited as follows:

Experiment	Maximum Reactivity Worth
Moveable	0.08 %ΔK/K
Secured	0.60 %ΔK/K

5. The reactor shall be subcritical by at least 3.0 %ΔK/K during fuel loading changes.

6. The reactivity worth of the regulating rod shall not exceed 0.6 %ΔK/K.

7. Experiments which could increase reactivity by flooding, shall not remain in or adjacent to the core unless the shutdown margin required in Specification 3.1.(1) would be satisfied after flooding.

8. The temperature coefficient will be negative and surveillance will be conducted at initial startup and change in fuel type.

9. For operation at power levels in excess of 0.1 MW in the forced convection mode, all grid positions shall contain fuel elements, baskets, reflector elements, grid plugs or experimental facilities.

10. For operation at powers in excess of 0.1 MW, the pool gate must be in its storage location.

Bases:

The shutdown margin required by Specification 3.1.1 assures that the reactor can be shutdown from any operating condition and will remain subcritical after cool down and xenon decay even if the rod of the highest reactivity worth should be in the fully withdrawn position. The SAR (Part A, Section V) demonstrates that the shutdown margin conservatively exceeds the 1% in Specification 3.1.1.

Specification 3.1.2 limits the allowable excess reactivity to the value necessary to overcome the combined negative reactivity effects of: (1) an increase in primary coolant temperature; (2) fission product xenon and samarium buildup in a clean core; (3) power defect due to increasing from a zero power, cold core to a 2 MW, hot core; (4) fuel burnup during sustained operation; and (5) moveable experiments.

Specification 3.1.3 limits the reactivity worth of experiments to values of reactivity which, if introduced as positive step changes, will not cause fuel melting.

Specification 3.1.4 limits the individual reactivity worth of an experiment to a value that will not produce a stable period of less than 30 seconds and which can be compensated for by the action of the control and safety system without exceeding any safety limits.

Specifications 3.1.5 provide assurance that the core will remain subcritical during fuel loading changes.

Specification 3.1.6 assures that failure of the automatic control system will not introduce sufficient excess reactivity to produce a prompt critical condition.

Specification 3.1.7 assures that the shutdown margin required by Specification 3.1.1 will be met in the event of a positive reactivity insertion caused by the flooding of an experiment.

Specification 3.1.8 assures that the power increase is self limiting.

Specification 3.1.9 assures that all grid positions be occupied will prevent the degradation of flow rates due to flow bypassing the active fueled region through an unoccupied grid plate position.

Specification 3.1.10 assures that the gate be stored assures that the full volume of the pool water is available to provide cooling of the core during normal operation and in the event of a loss of coolant accident.

3.2 Reactor Safety System

Applicability:

These specifications apply to the reactor safety system and other safety related instrumentation.

Objective:

To specify the lowest acceptable level of performance or the minimum number of acceptable components for the reactor safety system and other safety related instrumentation.

Specification:

The reactor shall not be made critical unless:

1. The reactor safety systems and safety related instrumentation are operable in accordance with Tables 3.1 and 3.2 including the minimum number of channels and the indicated maximum or minimum setpoint;

2. All shim safety blades are operable in accordance with Technical Specification 4.1.1 and 4.1.2.

3. The time from initiation of a scram condition until the control element is fully inserted shall not exceed 1 second in accordance with Technical Specification 4.2.5 and 4.2.6.

4. The reactivity insertion rates of individual control and regulating blades will not exceed 0.02 $\%\Delta K/K$ per second.

Bases:

Neutron flux level scrams provide redundant automatic protective action to prevent exceeding the safety limit on reactor power. The period scram limits the rate of rise of the reactor power to periods which are manually controllable without reaching excessive power levels or fuel temperatures.

The loss of flow scram assures that an automatic loss of flow scram will occur in the event of a loss of flow when the reactor is operating at power levels above 0.1 MW.

The reactivity insertion rate limit was determined in the SAR, Section XI and predicts a safe fuel clad temperature.

TABLE 3.1

REQUIRED SAFETY CHANNELS

Reactor Safety System Component/Channel	Minimum Required	Function	Operating Mode in Which Required
Reactor Power Level	2	Automatic scram when ≥ 115% of range scale with 2.3 MW max	All Modes
Coolant Flow Rate	1	Automatic scram at ≤1600 gpm	Forced Convection above 0.1 MW
Seismic Disturbance	1	Automatic scram at Modified Mercalli Scale IV	All modes
Bridge Misalignment	1	Automatic scram	Forced Convection above 0.1 MW
Pool Water Level	1	Automatic scram at 16" below suspension frame base plate elevation	All modes
Coolant Outlet Temperature	1	Automatic scram 121°F	Forced Convection above 0.1 MW

Reactor Safety System Component/Channel	Minimum Required	Function	Operating Mode in Which Required
Bridge Movement	1	Automatic scram	All Modes
Coolant Gates Open	1	Automatic scram if either the coolant riser or coolant downcomer gates open	Forced Convection above 0.1 MW
Detector High Voltage Failure	3	Automatic scram if Voltage decreases 50V max	All Modes
Log N Period	1	Automatic scram if period ≤4 sec	All Modes
No Flow Thermal Column	1	Automatic scram	Forced Convection above 0.1 MW
Manual Scram Switch (console, bridge)	2	Manual scram	All Modes

TABLE 3.2

Required Safety Related Instrumentation

	Instrumentation	Setpoint	Minimum Number Required	Function	Operating Mode
1.	Reactor Coolant Inlet Temperature	≤ 111°F	1	Alarm	FC≥0.1MW
2.	Reactor Coolant Outlet Temperature	≤ 119°F	1	Alarm	FC≥0.1MW
3.	Log Count Rate	<3 cps	1	Rod withdrawal interlock	All
4.	Servo Control Interlock	≥30 sec (fullout)	1	Auto Control Interlock	All
Facility Radiation (a) Monitoring System					
5.	Building Air Exhaust (Stack)	Gaseous 2.5 x normal particulate 2 x normal	1	Alarm	All
6.	Reactor Bridge	2 x normal	1	Alarm	All
7.	Fuel Safe	2 x normal or 5mR/hr, which ever is higher	1	Alarm	All

	Instrumentation	Setpoint	Minimum Number Required	Function	Operating Mode
8.	Thermal Column	2 x normal or 2mR/hr, which ever is higher	1	Alarm	All
9.	Heat Exchanger	2 x normal	1	Alarm	All
10.	Primary Demineralizer (Hot DI)	2 x normal	1	Alarm	All
11.	Continuous Air Monitoring Unit	2 x normal	1	Alarm	All

NOTES

(a) The facility radiation monitoring system consists of 8 radiation detectors which alarm and readout in the control room except for #11 which has a local alarm and readout only. The normal setpoints for this system are shown in Table 3.2. Use of higher than normal setpoints will require approval of the Director or the Assistant Director. Any senior operator member may adjust a setpoint lower than the normal value.

3.3 Coolant Water

(a) Primary Coolant Water

Applicability:

This specification applies to the limiting conditions for primary coolant pH, resistivity, available pool water volume and radioactivity.

Objective:

To maintain the primary coolant in a condition to minimize the corrosion of the primary coolant system, fuel cladding, and other reactor components, and to assure proper conditions of coolant for normal and emergency requirements.

Specification:

1. The primary coolant pH shall be maintained between 5.5 and 7.5.

2. The primary coolant resistivity shall be maintained at a value greater than 500Kohms-cm (conductivity 2mmhos/cm).

3. The concentration of radioactive materials in the primary coolant is maintained at a level such that the reading on the reactor bridge radiation monitor does not exceed 10 mrem/hr.

Bases:

Experience at this and other facilities has shown that the maintenance of primary coolant system water quality in the ranges specified in specification 3.3.1 and 3.3.2 will control the corrosion of the aluminum components of the primary coolant system and the fuel element cladding. Conductivity Specification 3.3.2 also insures adequate

water purity to control activation of coolant water impurities.

The requirement in specification 3.3.3 ensures that the presence of unusual impurities or corrosion products is detected.

(b) Secondary Coolant Water

Applicability:

This specification applies to the limiting conditions for secondary coolant pH, cycles of chloride, resistivity and radioactivity.

Objective:

To maintain the secondary coolant in such a condition as to minimize corrosion and/or scale buildup on the heat exchanger tubes and to detect a primary to secondary system leak.

Specification:

1. The secondary coolant water pH shall be maintained between 5.5 and 9.0.

2. The sample will be analyzed for the presence of sodium-24.

Bases:

The facility has maintained the above coolant water conditions for many years based on consultant recommendations and have good results in maintaining heat exchanger tube and shell cleanliness.

Radioactivity in the secondary system would indicate a leak and therefore samples are analyzed for detectable concentrations of sodium-24.

3.4, 3.5, 3.6 Confinement and Emergency Exhaust System and Emergency Power

Applicability:

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

Objective:

To assure that the confinement and emergency exhaust system is in operation to mitigate the consequences of possible release of radioactive materials resulting from reactor operation, fuel movement and handling of radioactive material.

Specification:

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

Equipment/Condition	Function
Personnel access doors to reactor closed (except for entrance and egress). Roof hatch closed. Truck door closed;	To maintain confinement system integrity
Reactor Room fresh air intake valve and exhaust ventilation valve to the stack are open;	To maintain confinement system integrity
Initiation system for confinement isolation, i.e. evacuation buttons and alarm horns;	To initiate system operation and alert personnel

Equipment/Condition	Function
Emergency cleanup exhaust system;	To maintain a negative building pressure without unloading any large fraction of possible airborne activity
Emergency generator.	To insure power source to clean up system and other designated systems

Bases:

In the unlikely event of a release of fission products, or other airborne radioactivity, the confinement isolation initiation system will secure the normal ventilation exhaust fan, will bypass the normal ventilation supply up the stack, and will close the normal inlet and exhaust valves. In confinement, the emergency exhaust system will tend to maintain a negative building pressure with a combination of controls intended to prevent unloading any large fraction of airborne activity. The emergency exhaust purges the building air through charcoal and absolute filters and controls the discharge which is diluted by supply air through a 115 foot stack.

3.7 Radiation Monitoring Systems and Effluents

3.7.1 Radiation Monitoring Systems

Applicability:

This specification applies to the availability of radiation monitoring equipment which must be operable during reactor operation, fuel movement and handling of radioactive materials in the reactor building.

Objective:

To assure that radiation monitoring equipment is available for evaluation of radiation conditions and that the release of airborne radioactive material is maintained below the limits established in 10CFR20.

Specification:

1. When the reactor is operating, gaseous and particulate sampling of the stack effluent shall be monitored by a stack monitor with a readout in the control room.

   The particulate activity monitor and the gaseous activity monitor for the facility exhaust stack shall be operating. If either unit is to be out of service, either the reactor shall be shut down or the unit shall be replaced by one of comparable monitoring capability;

2. When the reactor is operating, at least one constant air monitoring unit (Table 3.2.10) located in the confinement building shall be operating. Temporary shutdown of this unit shall be limited as in 3.7.1 above.

3. The reactor shall not be continuously* operated without a minimum of one area radiation monitor (Table 3.2.7) on the "ground floor level" of the reactor building and one area monitor (Table 3.2.5) over the reactor pool (reactor bridge) operating and capable of warning personnel of high radiation levels.

   *In order to continue operation of the reactor, replacement of an inoperative monitor must be made within 15 minutes of recognition of failure, except that the reactor may be operated in a steady-state power mode if the

monitor is replaced with portable gamma-sensitive instruments having their own alarm.

Bases:

A continuing evaluation of the radiation levels within the reactor building will be made to assure the safety of personnel. This is accomplished by the monitoring systems described in Table 3.2.

3.7.2 Effluents

a. Airborne Effluents

Applicability:

This specification applies to the monitoring or airborne effluents from the Rhode Island Nuclear Science Center (RINSC).

Objective:

To assure that containment integrity is maintained during reactor operation and that the release of airborne radioactive material from the RINSC is maintained below the limits established in 10CFR20.

Specification:

1. The concentration of radioactive materials in the effluent released from the facility exhaust stacks shall not exceed 10^5 times the concentrations specified in 10CFR20, Appendix B, Table II, when averaged over time periods permitted by 10CFR20.

Bases:

The limits established in specification 3.7.2 incorporate a dilution factor of 10^5 for effluents released through the exhaust stacks. This dilution factor is more conservative than that calculated from actual meteorological data which represents the lowest dispersion factor determined and the highest frequency of wind in any sector. Because of the use of the most conservative measured values of wind directional frequency and dispersion factors, this dilution factor will assure that concentrations of radioactive material in unrestricted areas around the Rhode Island Nuclear Science Center will be far below the limits of 10CFR20. (Refer to letter dated April 16, 1963 sent to the NRC in connection with license questions.)

This dilution factor is used for calculating maximum ground concentration down wind for noble gases. The SAR contains calculations for doses from the iodine at the 48 meter distance.

a. Liquid Effluents

Applicability:

This specification applies to the monitoring of radioactive liquid effluents from the Rhode Island Nuclear Science Center.

Objectives:

The objective is to assure that exposure to the public resulting from the release of liquid effluents will be within the regulatory limits and consistent with as low as reasonably achievable requirements.

Specification:

The liquid waste retention tank discharge shall be batch sampled and the gross activity per unit volume determined before release. All off-site releases shall be directed into the municipal sewer system.

Bases:

All radioactive liquid and solid wastes disposed of off-site shall be within the limits established by 10CFR20 or shall be removed from the site by a commercial licensed organization.

3.8 Limitations on Experiments

Applicability:

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

Objectives:

To prevent damage to the reactor or release of radioactive materials in excess of 10CFR20.

Specification:

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

1. All materials to be irradiated shall be either corrosion resistant or encapsulated within corrosion resistant containers to prevent interaction with reactor components or pool water. Corrosive materials shall be doubly encapsulated.

2. Irradiation containers to be used in the reactor, in which a static pressure will exist or in which a pressure buildup is predicted, shall be designed and

tested for a pressure exceeding the maximum expected by a factor of 2.

3. Fissionable materials shall have total iodine and strontium inventory less than that allowed by the facility by-product license.

4. Explosive materials, in any quantity, shall not be allowed in the reactor pool or experimental facilities.

5. All experiments shall be designed against failure from internal and external heating at the true values associated with the LSSS for reactor power level and other process parameters.

6. Experimental apparatus, material or equipment to be irradiated shall be positioned so as not to cause shadowing of the nuclear instrumentation, interference with control blades, or other perturbations which may interfere with safe operation of the reactor.

7. Cryogenic liquids shall not be used in any experiment with the reactor pool without approval from the Nuclear Regulatory Commission.

8. No highly water reactive materials shall be used in an experiment in the reactor pool.

9. No experiment should be performed unless the material content (with the exception of trace constituents) is known.

10. If a capsule fails and releases material which could damage the reactor fuel or structure by corrosion or other means, removal and physical inspection shall be performed to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the Director, or his designated

alternate, and determined to be satisfactory before operation of the reactor is resumed.

Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under: (1) normal operating conditions of the experiment or reactor, (2) credible accident conditions in the reactor, and (3) possible accident conditions in the experiment shall be limited in activity such that: if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor room or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the occupational limits for maximum permissible concentration.

In calculations pursuant to the above, the following assumptions shall be used: (1) If the effluent from an experimental facility exhausts through ductwork which closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape. (2) If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of these vapors can escape. (3) For materials whose boiling point is above 55°C and where vapors formed by boiling this material can escape only through an undisturbed column of water above the core, at least 10% of these vapors can escape. (4) Limits for maximum permissible concentrations are specified in the appropriate section of 10CFR20.

Bases:

Specifications 1 through 5, 8 and 9 are intended to reduce the likelihood of damage to reactor components and/or radioactivity releases resulting from experiment failure and, along with the reactivity restriction of pertinent specification in 3.1, serve as a guide for the review and approval of new and untried experiments by the operations staff as well as the Reactor Utilization Subcommittee.

Specifications 3 and 4 are self explanatory.

Specification 6 assures that no physical or nuclear interferences compromise the safe operation of the reactor by, for example, tilting the flux in a way that could effect the peaking factor used in the Safety Analysis.

Specification 7 insures NRC review of experiments containing or using cryogenic materials. Cryogenic liquids present structural and explosive problems which enhance the potential of an experiment failure.

Specification 10 is self explanatory.

3.9. Reactor Core Components

a. LEU Fuel-Fission Density

Applicability:

This specification applies to fission density limits of the RINSC fuel.

Objective:

To prevent fuel plate swelling which could result in cold rupture and release of radioactive fission products.

Specification:

None required.

Bases:

The fission density is below operational fission densities reached in other operating reactors using the same kind of fuel without failure attributed to the fuel.

An experimental data base which supports the safe use of UAl_x and U_3O_8 fuel in the Rhode Island Nuclear Science Center (RINSC) reactor was derived from irradiation tests performed in the Materials Test Reactor (MTR), the Engineering Test Reactor (ETR), and the Advanced Test Reactor (ATR) at the Idaho National Engineering Laboratory, the High Flux Isotope Reactor (HFIR) at the Oak Ridge National Laboratory, and the German Karlsruhe FR2 reactor. The SAR (Part A, VI) defines the fact that a calculated limit of $1.5x10^{21}$ fissions/cm^3 will not be reached. Therefore no limit is specified.

b. Beryllium Reflectors

Applicability:

This specification applies to neutron flux damage to the standard and plug type beryllium reflectors.

Objective:

To prevent physical damage to the beryllium reflectors in the core from accumulated neutron flux exposure.

Specification:

1. The maximum accumulated neutron flux shall be $1x10^{22}$ neutrons/cm^2.

Bases:

The RINSC SAR (Part A Section VIII) has addressed this limit as a conservative limit.

c. LEU Fuel

Applicability:

This specification applies to the physical condition of the fuel elements.

Objective:

To prevent operation with damaged fuel elements.

Specification:

Fuel elements to be inspected for physical defects and reactor core box fit in accordance with manufactured specifications.

Bases:

The RINSC inspects and tests each fuel element for reactor core box fit in accordance with written procedures to assure operation with fuel elements that are not damaged and meet specifications.

4.0 SURVEILLANCE REQUIREMENTS

Surveillance tests may be deferred for periods of reactor shutdown providing they are performed prior to restart (ANS 15.1, 4.1)

4.1 Reactivity Limits

Applicability:

This specification applies to the surveillance requirements for reactivity limits.

Objective:

To assure that the reactivity limits of Specification 3.1 are not exceeded.

Specification:

1. Shim safety blade reactivity worths and insertion rates shall be measured:

   a. annually;

   b. whenever the core is changed from the startup core to the three other cores as analyzed and specified in the SAR (Part A, Section V).

2. Shim safety blades shall be visually inspected and checked for swelling at least annually.

3. The reactivity worth of those experiments, whose safety review indicates a need for such a determination, shall be measured prior to the experiment's initial use.

Bases:

Specification 4.1.1 will assure that shim safety blade reactivity worths are not degraded or changed by core arrangements.

Shim safety rod inspections are the single, largest source of radiation exposure to facility personnel. In order to minimize personnel radiation exposure and provide an inspection frequency that will detect early evidence of swelling and cracking, an annual inspection interval was selected for Specification 4.1.2.

The specified surveillance relating to the reactivity worth of experiments will assure that the reactor is not operated for extended periods before determining the reactivity worth of experiments. This specification also provides assurance that experiment reactivity worths do not increase beyond the established limits due to core configuration changes.

4.2 Reactor Safety System

Applicability:

This specification applies to the surveillance of the reactor safety system.

Objective:

To assure that the reactor safety system is operable as required by Specification 3.2.

Specification:

1. A channel test of the neutron flux level safety channels and period safety channel shall be performed:

   a. Prior to each reactor startup following a period when the reactor was secured;

b. After a channel has been repaired or deenergized.

2. A channel calibration of the safety channels listed in Table 3.1, which can be calibrated, shall be performed annually.

3. The radiation monitoring system required in Table 3.2 shall be operable prior to every reactor startup for which safety system channel tests are required as in 4.2.1. If the system has been repaired, the system shall be operable prior to use.

4. Shim safety blade release-drop time shall be measured annually.

5. Shim safety rod release-drop time shall be measured whenever the shim safety rod's core location is changed or whenever maintenance is performed which could effect the rod's drop time. (Specification 3.2.3)

6. Shutdown Margin (Specification 3.1.1)

   The shutdown margin shall be determined annually. It shall be determined when a new core is configured as described in the SAR (Part A, Section V). The determination will be made in accordance with operating procedures.

7. Excess Reactivity (Specification 3.1.2)

   The excess reactivity shall be determined annually. It shall be determined when a new core is configured as described in the SAR (Part A, Section V). The determination will be made in accordance with operating procedure.

8. Reactivity Insertion Rate (Specification 3.2.4)

The reactivity insertion rate shall be measured annually. It shall be determined when a new core is configured as described in the SAR (Part A, Section V). The determination will be made in accordance with written procedures.

Bases:

Prestartup tests of the safety system channels assure their operability. Annual calibration detects any long term drift that is not detected by normal intercomparison of channels. The channel operability check of the neutron flux level channels assures that the detectors are properly adjusted to accurately monitor the parameter they are measuring.

Radiation monitors are checked for proper operation in Specification 4.2.3. Calibration and setpoint verification involve use of a calibration source and significant personnel radiation exposure. It is determined that annual calibration of radiation monitors is adequate since they displayed excellent stability over many years of operation.

The measured release-drop times of the shim safety blades have been consistent over many years. Annual check of these parameters is considered adequate to detect any deterioration which could change the release-drop time. Binding or rubbing caused by rod misalignment could result from maintenance; therefore, release drop times will be checked after such maintenance.

4.3 Water Coolant System

a. Primary Coolant System

Applicability:

This specification applies to the surveillance of the primary coolant system.

Objective:

To assure high quality pool water and to detect the deterioration of components in the primary coolant loop.

Specification:

1. The pH of the primary coolant shall be measured weekly.

2. The resistivity of the primary coolant shall be measured weekly.

3. The radioactivity of the primary coolant shall be analyzed weekly for gross activity and quarterly for isotopic activity.

4. Pool water level scram switch shall be checked for operation monthly.

5. Pool inspections shall be made annually in accordance with operating procedures.

6. Pool level shall be visually inspected daily in accordance with operating procedures.

Bases:

Regular surveillance of pool water quality and radioactivity provides assurance that pH and resistivity changes that could accelerate the corrosion of the primary system components would be detected before significant damage would occur, and that the presence of leaking fuel elements in the reactor is detected.

The low pool level switch is checked for operation monthly. Upon a one inch pool level drop, the automatic fill begins; upon a two inch drop, the reactor scrams (if operating) and a local and remote alarm sounds. The remote alarm is continuously monitored offsite.

Annual pool system inspections are made to provide assurance that other cooling system components (eg. gate valves, gasketing etc.) are functioning properly.

b. Secondary Coolant System

Applicability:

This specification applies to the surveillance of the secondary coolant water.

Objective:

To assure the conditions of the coolant meet specification 3.3.(b) and to detect a primary to secondary water leak.

Specification:

1. The pH shall be measured weekly during reactor operation.

2. A sample shall be drawn and analyzed weekly for sodium-24 activity.

3. The gross radioactivity shall be measured daily.

Bases:

Proper secondary coolant conditions are obtained by blowdown and makeup water systems which maintain the proper water quality pH. Radioactive concentrations are measured in accordance with written procedures.

A radiation detector, mounted in the cooling tower basin, displays gross radioactivity in the control room. The system has an alarm to notify the operator who will take action to secure blowdown water from entering the sewer system. If this system fails, sampling or another detection method can be employed.

4.4, 4.5, 4.6 RINSC Confinement Building and Emergency Exhaust System

Applicability:

This specification applies to the surveillance of the facility openings and dampers.

Objective:

To assure that the condition of the closure devices for the building openings are in satisfactory condition and to assure their ability to provide adequate confinement of any airborne radioactivity released into the building.

Specification:

1. The confinement and emergency exhaust system described in Specification 3.4 shall be tested weekly for operability and after any maintenance that could affect system operability. The system operation is as described in the operating procedures and as herein discussed. The building cleanup system shall be activated by pressing an evacuation button, then automatically:

    a. the evacuation horn sounds

    b. the building ventilation blowers deenergize (air conditioner, exhaust blower, off gas blower, rabbit system blower, heating system blowers);

    c. the building ventilation dampers close (air intake and exhaust system);

    d. the cleanup system blower (through the scrubber filter) and air dilution blower (chem lab) are energized;

e. the negative differential pressure between the inside and outside of the building is at least 0.5 inches of water. This is determined by reading the pressure gauge in the control room;

f. the exhaust rate through the cleanup system shall not exceed 4500 CFM with not more than 1500 CFM coming from the reactor building and passing through the scrubber filter. The remaining air will be provided by a separate blower from an uncontaminated source.

2. The condition of the following equipment shall be inspected in accordance with written operating procedures every 6 months.

   a. Building ventilation blowers and dampers (including solenoid valves, pressure switches, piping, etc.);

   b. Personnel access and reactor room overhead doors.

3. The testing and maintenance of the emergency generator will be performed in accordance with the RINSC operating procedures and manufacturer recommendation.

4. The efficiency test for the charcoal filter shall be tested annually as specified in the operating procedures.

Bases:

The weekly check of the confinement system provides assurance that the automatic function will be actuated when confinement isolation is required. The semiannual inspection of valves and doors will provide assurance that the closures will perform their function of limiting leakage through these

openings in the event of a release of airborne activity into the building.

The weekly testing of the emergency generator (operating procedure) assures reliable response and operation. The monthly load testing (operating procedure) assures proper handling of expected system loads. The emergency generator system has annual maintenance performed in accordance with manufacturer recommendations.

4.7 Radiation Monitoring Systems and Effluents

a. Airborne Effluents

Applicability:

This specification applies to the surveillance of the monitoring equipment used to measure airborne radioactivity.

Objective:

The objective is to assure that accurate assessment of airborne effluents can be made.

Specification:

1. The particulate air monitors shall be calibrated annually.

2. The gaseous activity monitors shall be calibrated annually.

3. A channel check of the stack monitor and the main floor monitor shall be performed daily when the reactor is in operation.

Bases:

Experience with the electronic reliability and calibration stability of the units used by the Rhode Island Nuclear Science Center Reactor demonstrates that the above periods are reasonable surveillance frequencies.

b. Liquid Effluents

Applicability:

This specification applies to the surveillance of the monitoring equipment used to measure the radioactivity in liquid effluents.

Objective:

The objective is to assure that accurate assessment of liquid effluents can be made.

Specification:

1. The monitoring equipment used to measure the radioactive concentrations in the waste retention tanks shall be calibrated annually.

2. The contents of every tank released shall be sampled and evaluated for radioactive concentrations and pH prior to its release.

Bases:

Experience with the electronic reliability and calibration stability of the units used by the Rhode Island Nuclear Science Center Reactor demonstrates that the above periods are reasonable surveillance frequencies.

4.8 Surveillance of Experiments

Applicability:

This specification applies to the surveillance of experiments and the limitations on experiments as described in Technical Specification 3.8.

Objective:

To assure that the experiments and their limitations are reviewed with respect to 10CFR50.59 for reactor operation and personnel safety and prevent release of radioactive materials in excess of 10CFR20.

Specification:

Experiments shall be reviewed, approved and properly installed and operational in accordance with written operating procedures.

Experiments in progress shall undergo a review annually.

Bases:

Review of the experiments using the appropriate LCO's and the Administrative Controls assures that the insertion of experiments will not negate the consideration implicit in the Safety Limits.

4.9 Reactor Core Components

Applicability:

This specification applies to the surveillance requirements for reactor core components affecting reactor power.

a. LEU Fission Density
   Not Required - See T.S. 3.9.a

b. Beryllium Reflectors

Applicability:

This specification applies to the surveillance of beryllium lifetime for the standard and plug type beryllium reflectors.

Objective:

To prevent physical damage to the beryllium reflectors in the core from accumulated neutron flux exposure.

Specification:

The maximum accumulated neutron flux shall be 1×10^{22} neutrons/cm^2. The element fluence shall be determined annually in accordance with the operating procedures. Inspections and core fit shall be conducted annually.

Bases:

The RINSC SAR (Part A Section VIII) has addressed this limit as a conservative limit. (Annual inspections and core box fit as well as calculated total exposure serve as a ways to monitor the beryllium lifetime.)

c. LEU Fuel Elements

Applicability:

This specification applies to surveillance of LEU fuel elements.

Objective:

To prevent operation with damaged fuel elements and verify the physical condition of the fuel element.

Specification:

The fuel elements shall be visually examined and functionally fit into the core grid box annually.

Bases:

Fuel elements are initially inspected for manufactured specifications and then inserted into the grid box in accordance with QA/QC program requirements for functional fit. Core reloading is performed in accordance with operating procedures. Routine fuel movements are logged and visual inspections are conducted during fuel movements. Pool sampling also is used to detect a ruptured element (Tech. Spec. 4.3.3). The fission density limit for this reactor cannot be exceeded (reference SAR, Part A, Section VI). Burnup calculations are made quarterly (4.9.1).

5.0 DESIGN FEATURES

The basic design features of the facility are described in "General Electric's Operation and Maintenance Manual-GEI-77793", Oct. 1962, also in the "Safety Analysis Report for the Low Enriched Fuel Conversion of the Rhode Island Nuclear Science Center Research Reactor", Rev. 1, 1992. These documents are on file at the Science Center. A general description of the important components is included in the following sections.

5.1 Description

The reactor is located at the Rhode Island Nuclear Science Center on 3 acres of a 27-acre former military reservation, originally called Fort Kearney and now called the Narragansett Bay Campus of the University of Rhode Island. The 27-acre reservation is controlled by the State of Rhode Island through the University of Rhode Island. The reservation is in the Town of Narragansett, Rhode Island, on the west shore of Narragansett Bay, approximately 22 miles south of Providence, Rhode Island, approximately six miles north of the entrance of the Bay from the Atlantic Ocean. The Rhode Island Nuclear Science Center and various buildings used for research, education and training purposes are located on this 27-acre campus.

5.2 Reactor Fuel

The fuel assemblies shall be of the MTR type, consisting of plates containing uranium silicide fuel enriched to less than 20% in the isotope U-235 clad with aluminum. Each fuel element will contain 22 plates for a total of 275 grams of U-235 per element.

5.3 Reactor Core

The reactor core consists of a 9 x 7 array of 3" square modules with the 4 corners occupied by posts. The reference core for these technical specifications consists of 14 standard LEU fuel elements arranged symmetrically within 4 safety control blades as shown in Figure 4 of the SAR (Revision 1, Section V, Dec. 1992) as approved by the NRC in the conversion order (letter of March 17, 1992).

5.4 Reactor Building

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

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

The reactor building exhaust blower operates in conjunction with additional exhaust blower(s) which provide dilution air from non-reactor building sources.

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

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

5.5 Fuel Storage

All reactor fuel element storage facilities shall be designed in geometrical configuration where k_{eff} is less than 0.8 under flooding with water. A maximum of four fuel elements will be stored in the fuel safe with no two elements in adjacent positions in the storage box. The adjacent row will be an empty box. Irradiated fuel is stored in the underwater storage racks as described in the SAR (Part A, Section XII).

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization and Management

1. The Rhode Island Atomic Energy Commission (RIAEC) shall have the responsibility for the safe operation of the reactor. The organization of RIAEC is shown in Figure 6-1. The RIAEC shall appoint a Director and a Reactor Utilization Committee consisting of a minimum of five members, as follows:

    a. The Director

    b. The Radiation Safety Officer

    c. A qualified representative from the faculty of Brown University

    d. A qualified representative from the faculty of Providence College

    e. A qualified representative from the faculty of the University of Rhode Island

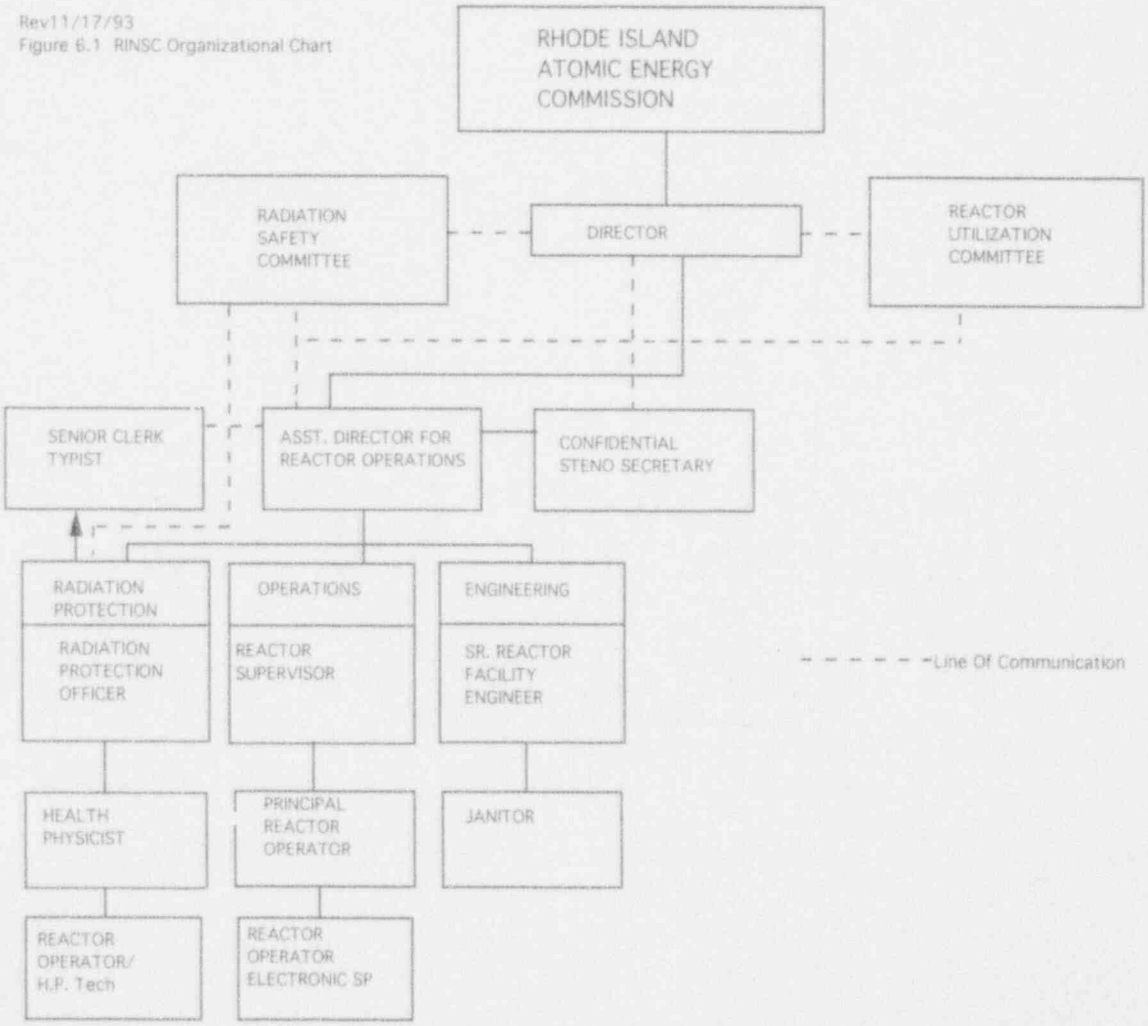
    A qualified alternate may serve in lieu of one of the above. The Director and Radiation Safety Officer are not eligible for chairmanship of the Committee.

2. An operator or senior operator licensed pursuant to 10CFR55 shall be present in the control room unless the reactor is secured as defined in these specifications. The minimum operating crew shall be two individuals.

3. A licensed senior operator shall be on duty or readily available on call whenever the reactor is in operation.

FIGURE 6.1 - ORGANIZATIONAL CHART

Rev11/17/93
Figure 6.1 RINSC Organizational Chart



4. In accordance with the emergency plan, a list of emergency personnel, management and offsite agencies is posted in the control room.

6.2 Qualifications of Personnel

1. At the time of appointment to the position, the Director shall have a minimum of six years of nuclear experience. The Director shall have an advanced degree in one of the physical sciences or engineering, and be a licensed senior operator. The degree will fulfill four years of the six-year requirement.

2. The Radiation Safety Officer shall have a master's degree in health physics or radiological health and three years of applied health physics experience in a program with radiation safety problems similar to those in the program to be managed.

3. The reactor operators and senior operators shall be licensed in accordance with the provisions of 10CFR55.

4. In the event of temporary vacancy in the position of Director or the Radiation Safety Officer, the functions of that position shall be assumed by qualified alternates appointed by the RIAEC.

6.3 Responsibilities of Personnel

1. Director

   a. The Director shall have responsibility for all activities in the reactor facility which may affect reactor operations or involve radiation hazards, including controlling the admission of personnel to the building.

This responsibility shall encompass administrative control of all experiments being performed in the facility including those of outside agencies.

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

2. Senior Reactor Operators

a. A licensed senior reactor operator pursuant to 10CFR55 shall be assigred each shift and be responsible for all activities during his shift which may affect reactor operation or involve radiation hazards. The reactor operators on duty shall be responsible directly to the senior operator.

b. The identity of and method for rapidly contacting the on-call senior reactor operator shall be known to the reactor operator on duty. The on-call senior reactor operator must be capable of being contacted by the duty reactor operator within ten minutes. The senior reactor operator shall be present at the facility during initial startup and approach to power, recovery from an unplanned or unscheduled shutdown or significant reduction in power, and refueling. The name of the person serving as senior reactor operator as well as the time he assumes the duty shall be entered in the reactor log. When the senior operator is relieved, he shall turn the

operation duties over to another licensed senior operator.

In such instances, the change of duty shall be logged and shall be definite, clear, and explicit. The senior reactor operator being relieved of his duty shall insure that all pertinent information is logged. The senior reactor operator assuming duty shall check the log for information or instructions.

3. Reactor Operators

a. The responsible senior reactor operator shall pursuant to 10CFR55 designate for his shift a licensed operator (hereafter called "operator") who shall have primary responsibility under the senior reactor operator for the operation of the reactor and all associated control and safety devices, the proper functioning of which is essential to the safety of the reactor or personnel in the facility. The operator shall be responsible directly to the senior reactor operator.

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

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

entry on duty by proper notation in the reactor log.

d. The operator, under the senior reactor operator on duty, shall be responsible for the operation of the reactor according to the approved operating procedures.

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

4. Radiation Safety Officer

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

6.4 Review and Audit

1. The Reactor Utilization Committee shall review reactor operations to assure that the facility is operated in a manner consistent with public safety and within the terms of the facility license.

2. The responsibilities of the Reactor Utilization Committee include, but are not limited to, the following:

a. Audit of operating, and emergency procedures and records.

b. Review and audit of proposed tests and experiments utilizing the reactor facilities.

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

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

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

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

g. Review changes to the Utilization Committee charter.

h. Review, at least annually, the radiation safety aspects of the facility.

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

4. A quorum of the Reactor Utilization Committee shall consist of not less than a majority of the Full Utilization Committee and shall include the Radiation Safety Officer or designee, and the chairman or designee.

5. The Reactor Utilization Committee shall meet at least annually.

6.5 Operating Procedures

Written procedures, reviewed and approved by the Reactor Utilization Committee, shall be used for items 1-9 listed below. The procedures shall be adequate to assure the safe operation of the reactor, but should not preclude the use of independent judgment and action should the situation require such.

1. Startup, operation and shutdown of the reactor;

2. Installation and removal of fuel elements, control blades and incore devices where necessary;

3 Maintenance procedures which could have an effect on reactor safety;

4. Periodic surveillance of reactor instrumentation and safety systems, area monitors, and continuous air monitors;

5. Implementation of the physical Security Plan and Emergency Plan;

6. Radiation control procedures;

7. Receipt, inspection, and storage of new fuel elements;

8. Storage and shipment of irradiated fuel elements.

9. Experiment review on a case-by-case basis assuring that section 3.8.3(2) of ANSI/ANS 15.1 is satisfied. Operational approval shall be by written approval by a licensed senior operator. Written procedures should be established and supervision of the installation of such experiments shall be defined and exercised.

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

6.6 Action to be Taken in the Event of a Reportable Occurrence

In the event of a reportable occurrence:

1. The Senior Reactor Operator shall be notified promptly and corrective action shall be taken immediately to place the facility in a safe condition until the cause of the reportable occurrence is determined and corrected.

2. The Director shall report the occurrence to the Reactor Utilization Committee. The report shall include an analysis of the cause of the occurrence, corrective actions taken, and recommendations for appropriate action to prevent or reduce the probability of a repetition of the occurrence.

3. The Reactor Utilization Committee shall review the report and the corrective actions taken.

4. Notification shall be made to the NRC in accordance with Paragraph 6.8 of these specifications.

6.7 Action to be Taken in the Event a Safety Limit is Exceeded

In the event a Safety Limit has been exceeded:

1. The reactor will be shut down and reactor operations will not be resumed until authorization is obtained from the NRC.

2. Immediate notification shall be made to the NRC in accordance with paragraph 6.8 of these specifications and to the Director.

3. A prompt report shall be prepared by the Senior Reactor Operator. The report shall include a complete analysis of the causes of the event and the extent of possible damage together with recommendations to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Utilization Committee for review and appropriate action, and a suitable similar report shall be submitted to the NRC in accordance with Paragraph 6.8 of these specifications and in support of a request for authorization for resumption of operations.

6.8 Reporting Requirements

In addition to the requirements of applicable regulations, all written reports shall be sent to the U. S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC 20555, with a copy to the Region I Administrator. The written reports include the following:

1. Within 24 hours, a report by telephone through the NRC Operation Center, 301-951-0550 and the NRC Region 1:

a. Any accidental release of radioactivity to unrestricted areas above permissible limits, whether or not the release resulted in property damage, personal injury or exposure.

b. Any significant variation of measured values from a corresponding predicted or previously measured value of safety related operating characteristics occurring during operation of the reactor.

c. Any reportable occurrences as defined in Paragraph 1.35 of these specifications.

d. Any violation of a Safety Limit.

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

2. A written report within 14 days in the event of a reportable occurrence, as defined in Section 1.35, a through e. The report shall:

a. Describe, analyze, and evaluate safety implications;

b. Outline the measures taken to assure that the cause of the condition is determined;

c. Indicate the corrective action taken, including any changes made to the procedures and to the quality assurance program, to prevent repetition of the occurrence and of similar occurrences involving similar components or systems;

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

3. Unusual Events

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

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

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

b. Permanent changes in the facility organization involving the Director or Assistant Director.

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

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

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

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

d. A description of each change to the facility or procedures, tests, and experiments carried out under the conditions of Section 50.59 of 10CFR50 including a summary of the safety evaluation of each.

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

f. A summary of annual radiation exposures in excess of 500 mrem received by facility personnel, including the dates and times of significant exposures.

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

6.9 Plant Operating Records

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

1. Records to be retained for a period of at least five years:

   a. Reactor operations;

   b. Principal maintenance activities;

c. Experiments performed including aspects of the experiments which could affect the safety of reactor operation or have radiological safety implications;

d. Reportable occurrences;

e. Equipment and component surveillance activities;

f. Facility radiation and monitoring surveys;

g. Fuel inventories and transfers; and

h. Changes to procedures systems, components, and equipment.

2. Records to be retained for the life of the facility:

a. Gaseous and liquid radioactive effluents released to the environs;

b. Off-site environmental monitoring surveys;

c. Personnel radiation exposures;

d. Updated, "as-built" drawings of the facility; and

e. Minutes of Reactor Utilization Committee meetings.