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

NATIONAL BUREAU OF STANDARDS

DOCKET NO. 50-184

RENEWAL OF FACILITY OPERATING LICENSE

Amendment No. 5
License No. TR-5

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for facility license renewal by the National Bureau of Standards (the licensee) dated December 2, 1980, as supplemented, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR, Chapter I;

   B. The construction of the facility was completed in substantial conformance with Construction Permit No. CPTR-5, dated April 22, 1963, the provisions of the Act, and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   D. There is reasonable assurance (i) that the activities authorized by this amended license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   E. The licensee is technically and financially qualified to engage in the activities authorized by this amended license in accordance with the rules and regulations of the Commission;

   F. The licensee is a Federal agency which, in accordance with 10 CFR Part 140, is not required to furnish proof of financial protection. The licensee has executed an indemnity agreement which satisfies the requirements of 10 CFR Part 140;

   G. The issuance of this amended license will not be inimical to the common defense and security or to the health and safety of the public;

   H. The issuance of this amended license is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession and use of the byproduct and special nuclear materials, as authorized by this amended license, will be in accordance with the Commission's regulations in 10 CFR Parts 30 and 70, including Sections 30.33, 70.23 and 70.31.

2. Facility Operating License No. TR-5 is hereby amended in its entirety to read as follows:

A. This amended license applies to the high-flux, heavy water-moderated and cooled, tank-type nuclear reactor (hereinafter referred to as the reactor) which is owned by the National Bureau of Standards (hereinafter referred to as the licensee or NBS) and is located on the NBS site, one mile southwest of Gaithersburg, in Montgomery County, Maryland.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses NBS:

(1) Pursuant to Section 104c of the Act and 10 CFR Part 50 - "Domestic Licensing of Production and Utilization Facilities," to possess, use and operate the reactor as a utilization facility at the designated location in accordance with the procedures and limitations described in the application and in this amended license;

(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess and use up to 45.0 kilograms of contained uranium-235 in connection with operation of the reactor.

(3) Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," to receive, possess and use a two-curie americium-beryllium source for reactor startup, and to possess, use and transfer but not to separate,* such byproduct material as may be produced by operation of the facility.

C. This amended license shall be deemed to contain and is subject to the conditions specified in Parts 20, 30, 50, 51, 55, 70 and 73 of 10 CFR, Chapter I, to all applicable provisions of the Act, and to the rules, regulations and orders of the Commission now or hereafter in effect and to the additional conditions specified below:

(1) <u>Maximum Power Level</u>

NBS may operate the reactor at steady-state power levels not in excess of 20 megawatts (thermal).

*Byproduct material produced in reactor experiments may be separated.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 5, are hereby incorporated in the amended license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Physical Security Plan

The licensee shall maintain and fully implement all of the provisions of the Commission-approved physical security plan, including changes made pursuant to the authority of 10 CFR 50.54(p). The approved physical security plan consists of a National Bureau of Standards document, withheld from public disclosure pursuant to 10 CFR 73.21, entitled, "NBSR Safeguards Plan," dated May 1983, transmitted by letter dated May 5, 1983.

3. This amended license is effective as of the date of issuance and shall expire at midnight twenty years from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut, Director
Division of Licensing

Enclosure:
Appendix A Technical
Specifications

Date of Issuance: MAY 1 6 1984

APPENDIX A

Technical Specifications
for the
Nuclear Test Reactor
Facility License TR-5

National Bureau of Standards
Gaithersburg, Maryland

May 1984

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INTRODUCTION

These technical specifications incorporate the significant safety limits, functional performance requirements, operating limits, administrative requirements, and surveillance schedules applicable to the National Bureau of Standards reactor for operation up to and including 20 MWt.

The dimensions, measurements, and other numerical values given in these specifications may differ slightly from actual values as a result of the normal construction and manufacturing tolerances, or normal accuracy of instrumentation.

INTRODUCTION

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1.0 DEFINITIONS

The following terms are sufficiently important to be separately defined. Where further discussion of the definition is useful, it follows the definition.

1.1 Channel

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

The following are the functions performed on a channel.

1.1.1 Channel Check

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

1.1.2 Channel Test

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

1.1.3 Channel Calibration

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

1.2 Confinement

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

1.3 Confinement Integrity

Confinement integrity shall mean that all of the following conditions are satisfied:

(1) All piping that pentrates the confinement building and is open to the confinement interior is physically intact exterior to the confinement.

(2) All automatic isolation valves in ventilation and process piping are either operable or closed.

(3) All automatic personnel access doors are capable of being closed and sealed or are closed and sealed.

(4) Except during passage, one set of the reactor building vestibule doors at the northeast and southeast personnel entrances and the elevator entrance from the laboratory basement are closed or attended.

(5) The reactor building truck door is closed and sealed.

(6) All other piping pentrations are sealed within the reactor building and are capable of withstanding the confinement test pressure.

(7) The building has passed its most recent leakage test.

1.4 Experiment

Experiment shall mean any installed apparatus, device, or material that is not rigidly installed within the confines of the thermal shield and that is intended to be used for irradiations or other measurements.

1.5 Operable

Operable shall mean that the system or component is capable of performing its intended function, as determined by functional testing or indication.

1.6 Reactor Shutdown

The reactor shall be considered shut down if any of the following conditions exists:

(1) The reactor contains less than 2.2 kg U-235.

(2) The reactor control power and the rod drive power key switches are locked in their "Off" position.

(3) The reactor is in the rod drop test mode, and a senior licensed operator is in direct charge of the operation.

1.7 Reactor Operating

The reactor is considered to be operating whenever it is not shut down.

1.8 Reactor Shutdown Mechanisms

Reactor shutdown mechanisms are those mechanisms involved in reactor shutdown and include:

(1) Rundown is the electrically driven insertion of all shim safety arms and the regulating rod at their normal operating speed.

(2) Scram is the spring assisted gravity insertion of all shim safety arms.

(3) Major scram is the spring assisted gravity insertion of all shim safety arms and automatic isolation of the confinement building.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Applicability: This specification applies to reactor power and reactor coolant system flow and temperature.

Objective: The objective is to maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products.

Specifications: Reactor power, coolant system flow, and inlet temperature shall not exceed the limits shown in Figures 2.1 and 2.2.

The reactor may be operated at power levels of up to 10 kW with reduced flow (including no flow) if decay heat is insufficient to cause significant heating of the reactor coolant.

Basis: Maintaining the integrity of the fuel cladding requires that the cladding remain below its melting temperature. For all plant operating conditions that avoid a departure from nucleate boiling, cladding temperatures remain substantially below the melting temperature. Conservative calculations (FSAR, NBSR 9, Addendum 1, Section 3.2.2, Nov. 1980) have shown that limiting combinations of reactor power and reactor coolant system flow and temperature to values more conservative than the safety limits will prevent cladding failure.

2.2 Limiting Safety System Settings

Applicability: This specification applies to limiting settings for instruments monitoring safety limit parameters.

Objective: The objective is to ensure protective action if any of the principal process variables should approach a safety limit.

Specification: The limiting safety system trip settings shall be

Reactor power, % (max)	130
Reactor outlet temperature, °F (max)	147 (rundown)
Coolant flow, gpm/MW (min)*	60 inner plenum
	235 outer plenum

Basis: At the values established, the safety system settings provide a significant margin from the safety limits. Even in the extremely unlikely event that all three parameters, reactor power, coolant flow, and outlet temperature simultaneously reach their safety system settings, the burnout ratio is at least 1.3. For all other conditions the burnout ratio is considerably higher

*May be bypassed during periods of reactor operation (up to 10 kW) when a reduction in safety limit values is permitted (Section 2.1 of these specifications).

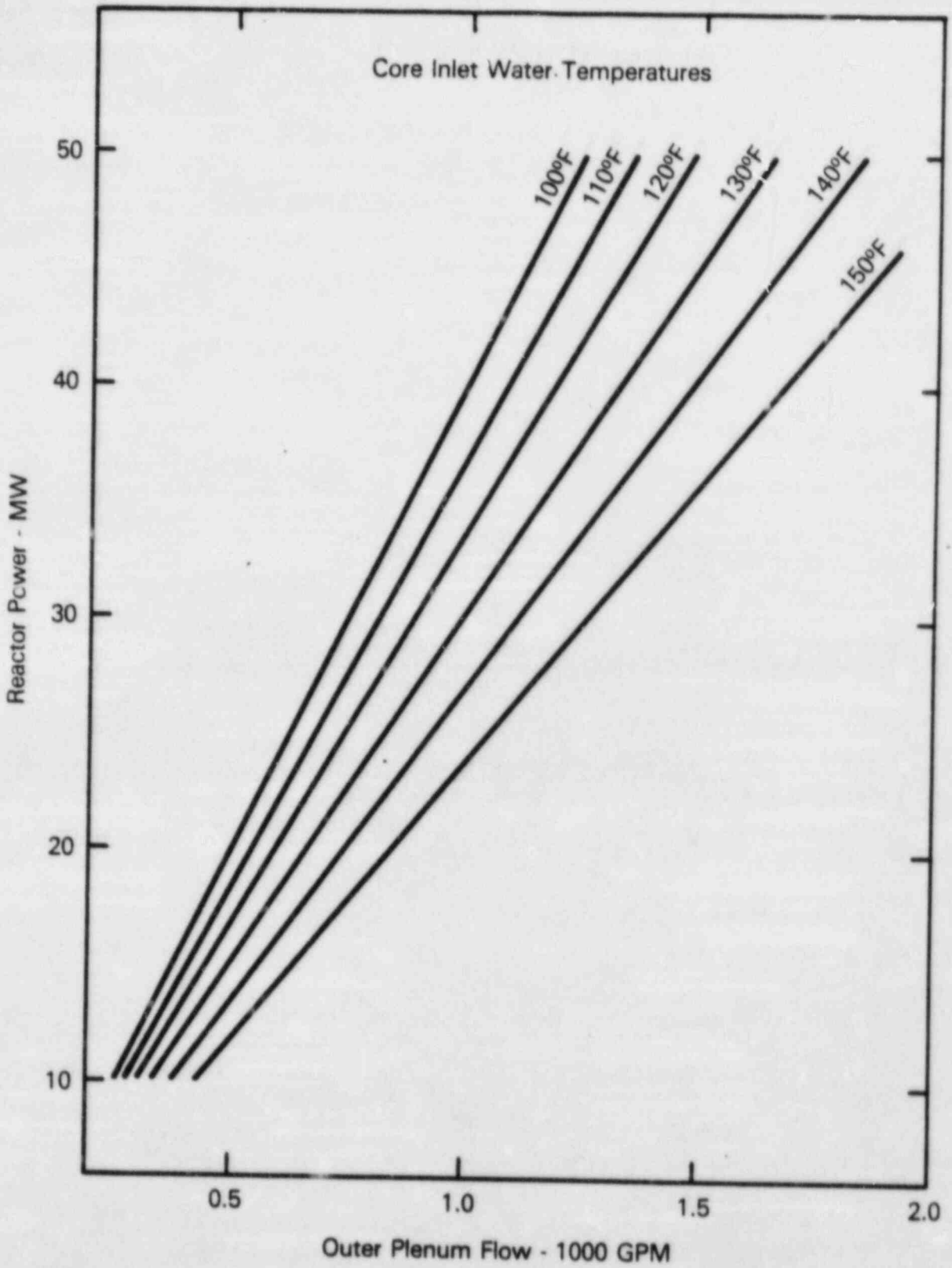

Figure 2.1 Safety limits - inner plenum

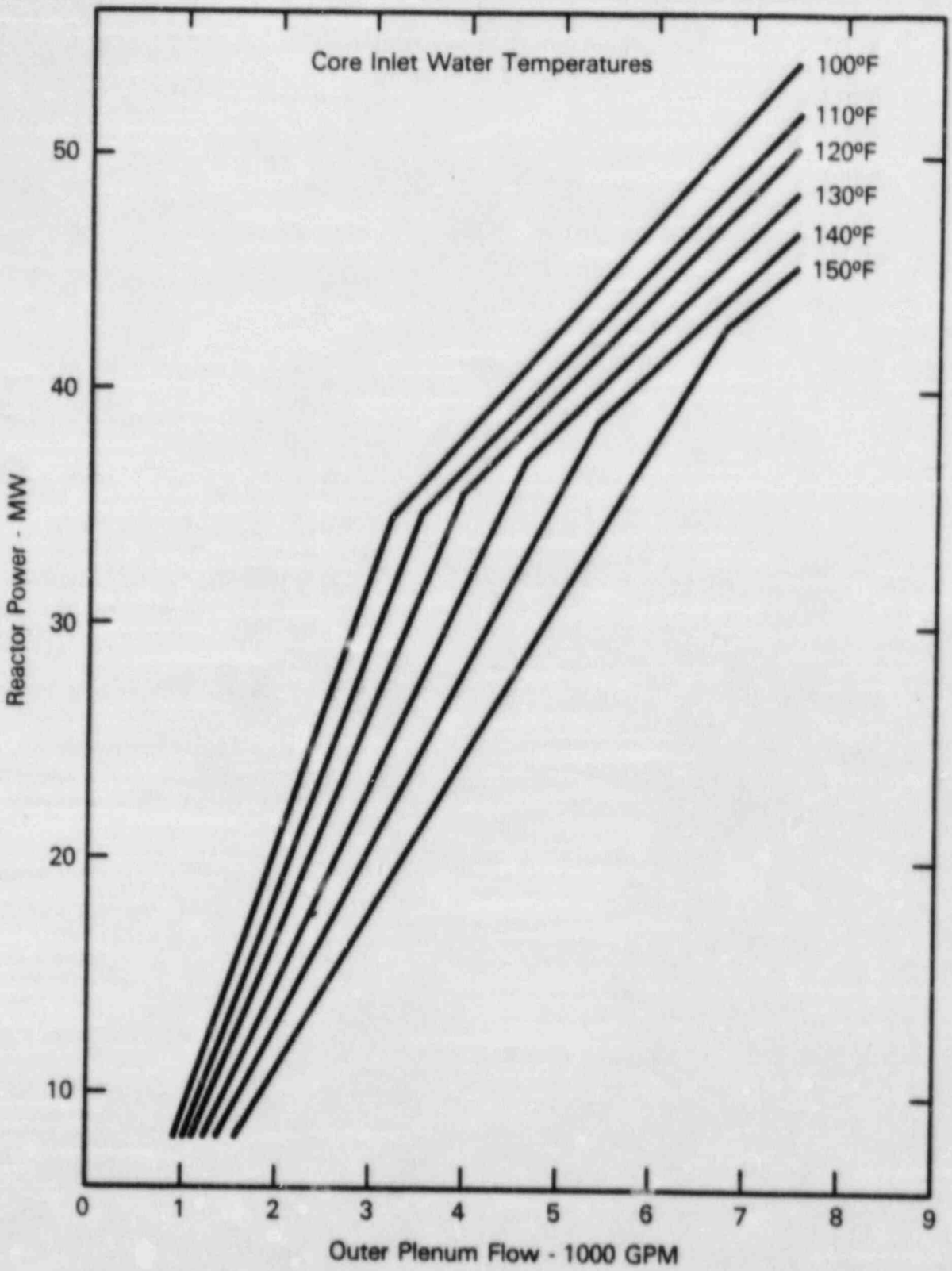

Figure 2.2 Safety limits - outer plenum

(FSAR, NBSR 9, Addendum 1, Section 3.2.2, Nov. 1980). This will ensure that any reactor transient caused by equipment malfunction or operator error will be terminated well before the safety limits are reached. Overall uncertainties in process instrumentation have been incorporated in limiting safety system setting values.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Confinement System

Applicability: This specification applies to the operating status of the confinement building.

Objective: The objective is to ensure confinement integrity when it is required.

Specifications: Confinement integrity shall be maintained when any of the following conditions exist:

(1) The reactor is operating.

(2) Changes of components or equipment within the confines of the thermal shield, other than rod drop tests or movement of experiments, are being made which could cause a change in reactivity.

(3) Movement of irradiated fuel, which contains significant fission product inventories outside a sealed container or system, is being conducted.

(4) No maintenance that causes a breach in confinement shall be performed unless the reactor has been shut down for a period equal to or greater than 1 hour for each megawatt of operating power level.

Basis: The confinement system is a major engineering safeguard. It serves as the final physical barrier to mitigate the release of radioactive particles and gasses to the environment following accidents (FSAR, NBSR 9, Section 13.6, p. 13-15, Apr. 1966, and Addendum 1, Sections 2 and 3, Nov. 1980).

Confinement integrity is stringently defined to ensure that the confinement building will perform in accordance with its design basis (FSAR, NBSR 9, Section 3.7.1, p. 3-16, April 1966).

Piping, penetrations, and conduits that are open to the inside of the confinement building become an extension of confinement and must be sealed on the exterior to the reactor building to prevent out-leakage. All other piping penetrations that do not have automatic closure devices must be sealed within the confinement by sealing devices that can withstand the confinement test pressure of 6.0 in. H_2O over pressure or 2.0 in. H_2O vacuum within specified leakage limits.

The confinement building is designed to be automatically sealed upon indication of high activity. All automatically operated valves and doors that affect this sealing must either be operable or already sealed. To attempt to operate the reactor with any of these devices inoperable is a violation of the confinement design basis. Although tests have shown that the confinement building can continue to operate with one or more of these closures failed, its margin of effectiveness is reduced. If the closure is placed in its closed or sealed condition, then operability of the automatic closure devices is not required.

Tests performed on the confinement building have shown that even if one of the automatically closing personnel doors fails to operate properly, confinement design capability can be met if the building vestibule doors are closed. By specifying that these doors remain closed except when they are being used or attended, a backup to the normal confinement closure is provided.

The reactor building truck door is not provided with automatic closure devices and must be in the closed position for confinement integrity to exist. Tests have shown that the confinement building can continue to operate properly, although at reduced efficiency, if the truck door seal were to fail. It will not operate properly if the truck door itself is open.

Changes in the core involving such operations as irradiated fuel handling or control rod repairs affect the reactivity of the core and could reduce the shutdown margin of the reactor. Confinement integrity is required when these changes are made because they affect the status of the core.

Confinement integrity is not required when the reactor is shut down and experiments are to be inserted or removed. The reactor is normally shut down by a substantial reactivity margin (calculated to be at least 14% $\Delta\rho$). Experiments are usually inserted and removed one at a time; hence, the total reactivity change in any single operation will be limited to the specified maximum worth of 0.5% $\Delta\rho$ for any single experiment. Even if the sequential movement of all experiments (including "fixed" experiments) were postulated, the maximum potential reactivity insertion would not exceed the 2.6% $\Delta\rho$ (see Section 4.0 of these specifications) worth of all experiments permitted in the reactor at any time. Under this circumstance, the shutdown margin would still be substantial.

Even when the reactor is shut down, irradiated fuel, which contains significant fission product inventories (sufficient to allow specification 3.11 to be exceeded should the element fail), poses a potential hazard should its cladding be violated when it is not otherwise contained (e.g., during transit or during sawing of aluminum end pieces). When irradiated fuel is contained within a closed system, such as the reactor vessel, transfer lock of the refueling system, sealed shipping cask, and so on, these serve as a secondary barrier to fission product release; therefore, confinement integrity is not required.

All possibilities of fission product release from fuel melting must be precluded to perform maintenance that prevents normal rapid closing of the confinement. For this reason, no such maintenance should be permitted unless the reactor has been shut down a specified time.

3.2 Reactor Coolant System

Applicability: This specification applies to the capability of the primary coolant system emergency cooling and heat exchanger isolation.

Objectives: The objectives are to ensure adequate reactor cooling capability and to provide the means of containing D_2O to H_2O heat exchanger leakage.

Specifications: The reactor shall not be operated

(1) unless at least one shutdown cooling pump is operable

(2) unless the heat exchanger isolation valves are operable

(3) unless either a secondary cooling water activity monitor or a D_2O storage tank level monitor sensitive to a loss of 300 gal of D_2O is operable

(4) with a reactor vessel coolant level more than 24 in. below the overflow standpipe level, except during periods when operations at power levels up to 10 kW with no reactor flow are permitted (see Section 2.1 of these specifications)

(5) with a D_2 concentration in the helium sweep system of greater than 4% by volume

Basis: Loss of flow accidents have been analyzed for the NBSR, assuming a single shutdown cooling pump is operable. Under this condition, the hot spot of the hottest plate remains below 160°F (FSAR, NBSR 9, Addendum 1, Section 3.3.3.1, Nov. 1980). The NBSR also has been analyzed assuming a no-shutdown cooling flow (FSAR, NBSR 9, Addendum 1, Section 3.3.3.2, Nov. 1980). For this case the maximum fuel plate temperature would be less than 500°F, well below the temperature that would cause any damage. To ensure that fuel plate temperatures following loss of flow will be near or below normal operating temperatures, a shutdown pump will be required.

The effect of leakage through the heat exchangers from the primary to the secondary system was analyzed. Calculations show that tritium releases offsite are below the concentrations allowed by 10 CFR 20 (FSAR, NBSR 9, Addendum 1, Section 2.6.4, Nov. 1980). Limits on such leakage have been established in Section 3.6 of these specifications. To minimize the amount of any such leakage, the heat exchanger D_2O isolation valves must be operable and means for detecting the leakage must be provided.

The limiting value for reactor vessel coolant level is somewhat arbitrary because the core is in no danger so long as it is covered with water. However, a drop of vessel level indicates a malfunction of the reactor cooling system and possible approach to uncovering the core. Thus, a measurable value well above the minimum level is chosen in order to provide a generous margin (i.e., about 7 ft) above the fuel elements. To permit periodic surveillance of the effectiveness of the moderator dump, it is necessary to operate the reactor without restrictions on reactor vessel level. This is permissible under conditions when forced reactor cooling is not required, such as is permitted in Section 2.1 of these specifications.

Deuterium gas will collect in the helium cover gas system because of radiolytic disassociation of D_2O. Damage to the primary system could occur if this gas were to reach an explosive concentration (about 7.8% by volume at 25°C in helium*). To ensure a substantial margin below the lowest potentially explosive value, a 4% limit is imposed.

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*The U.S. Atomic Energy Commission Report No. TID-20898, "Flammability of Deuterium in Oxygen-Helium Mixtures," Explosives Research Center, Bureau of Mines, June 15, 1964.

3.3 Reactor Core Parameters

Applicability: This specification applies to the core grid positions and core loading.

Objective: The objective is to ensure that the core grid positions are correctly filled and the core is properly loaded.

Specifications:

(1) The reactor shall not be operated unless all grid positions, except the six corner positions in the outer hexagonal ring, are filled with full length core assemblies. The six corner positions must be plugged in the lower grid, if not filled with such assemblies.

(2) The core shall not be loaded so that

(a) it cannot be shut down with the highest-worth shim arm withdrawn at ambient temperatures

(b) the excess reactivity at normal operating temperatures exceeds 15% $\Delta\rho$

Basis: The NBSR employs shim safety arm stops to prevent a broken shim arm from dropping from the reactor core. The proper operation of these stops depends on adjacent fuel elements or experimental thimbles being in place to prevent the broken arm from falling from the core lattice. The six corner positions, although not required as part of the shim arm stops, must be plugged at the bottom to prevent cooling flow from bypassing the fuel elements.

To allow control rod testing and to provide for the possibility of a stuck rod, the reactor must be subcritical with the highest-worth shim arm fully withdrawn.

The excess reactivity limit was established to ensure a substantial shutdown margin and to accommodate postulated reactivity accidents. The selected value of 15% $\Delta\rho$ is based on the following:

(1) The shutdown margin with the most reactive rod fully withdrawn (14% $\Delta\rho$) is adequate.

(2) The design-basis reactivity accident, which assumes the insertion of 1.3% $\Delta\rho$ into a just critical core, is not affected by the total core excess reactivity.

(3) The startup accident, which assumes constant withdrawal of all control rods until a scram occurs, is terminated by scram action after an insertion of reactivity, which is small compared to the total core excess reactivity.

3.4 Reactor Control and Safety Systems

Applicability: This specification applies to reactor control and safety system operation.

Objective: The objective is to ensure proper operation of reactor control and safety system.

Specifications: The reactor shall not be operated unless

(1) all four shim safety arms are operable

(2) the reactivity insertion rate, using all four shim safety arms, does not exceed 5.0×10^{-4} $\Delta\rho$/sec

(3) the SCRAMS and MAJOR SCRAMS are operable in accordance with Table 3.1

(4) the moderator dump system is operable

Basis: Although the NBSR could operate and could maintain a substantial shutdown margin with less than the four installed shim safety arms, flux and rod worth distortions could occur by operating in this manner. Furthermore, operation of the reactor with one shim arm known to be inoperable would further reduce the shutdown margin that would be available if one of the remaining three shim arms were to suffer a mechanical failure that prevented its insertion.

Rod withdrawal accidents for the NBSR were analyzed (FSAR, NBSR 9, Addendum 1, Section 3.3.1, Nov. 1980) using a maximum withdrawal rate of 5×10^{-4} $\Delta\rho$/sec. This rate corresponds to the maximum beginning-of-life rod worths with the rods operating at the design speed of their constant speed mechanisms. These analyses showed that for the most severe accident (startup from source level), the resultant energy of 4.8 MW-sec is significantly below the 34 MW-sec required to adiabatically heat the core to the point of fuel cladding failure.

The parameters listed in Table 3.1 are monitored by the reactor safety system. This system automatically initiates action to ensure that appropriate safety limits and minimum conditions for operation are not violated. With the channels operable as required by Table 3.1, the safety system meets the reliability requirements (including testing and maintenance provisions as suggested by the appropriate Institute of Electrical and Electronics Engineers (IEEE) standard for these systems*). Where only single channels are required in Table 3.1, parameters measured by other channels combine to provide the necessary redundancy.

In the unlikely event that the shim safety arms cannot be inserted, an alternate means of shutting down the reactor is provided by the moderator dump (FSAR, NBSR 9, Section 4.6.9, p. 4-19, Apr. 1966). The moderator dump provides a shutdown capability calculated to be at least 4% $\Delta\rho$. Hence, it is also considered necessary for safe operation.

3.5 Reactor Emergency Cooling System

Applicability: This specification applies to the availability of the emergency cooling system.

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*IEEE/NSG/Reactor Instrumentation and Control, "Standards for Nuclear Power Plant Protection Systems," 8th Rev., September 13, 1966.

Table 3.1 Reactor safety system

Function	Minimum Operable Channels	
	Scram	Major Scram
High flux level (2 of 3 or 1 or 2 logic only)	2	
Short period below 5% rated power	2	
High activity, effluent air exhaust RD 3-4, 3-5, and 4-1		2
Low reactor vessel D_2O level[1,3]	2	
Low flow reactor outlet[2,3]	1	
Low flow reactor inlet, inner or outer plenum[2,3]	1	
Manual in control room	1	1

[1]One of two channels may be bypassed for tests or during the time maintenance involving the replacement of components and modules or calibrations and minor repairs is actually being performed.

[2]One of these two flow channels may be bypassed during tests, or during the time maintenance involving the replacement of components and modules or calibrations and minor repairs is actually being performed. However, outlet low flow may not be bypassed unless both inner and outer low-flow reactor inlet safety systems are operating.

[3]All channels may be bypassed when not required by the exceptions of Sections 2.2 or 3.2 of these specifications.

Objective: The objective is to ensure an adequate supply of emergency coolant.

Specifications: The reactor shall not be operated unless

(1) the D_2O emergency cooling system is operable

(2) a source of makeup water to the D_2O emergency cooling tank is available

Basis: In the event of a loss of core coolant, the emergency cooling system provides adequate protection against melting of the reactor core and associated release of fission products. Thus, operability of this system is a prerequisite to reactor operation.

The emergency cooling system employs one sump pump to return spilled coolant to the overhead storage tank. Because only one pump is used, it must be operational. There is sufficient D_2O available to provide 2-1/2 hours of cooling on a once-through basis. In the event that the sump pump fails and the D_2O supply in the emergency cooling tank is exhausted, domestic water or a suitable alternate would be used to furnish water for once-through cooling. The water makeup capacity must be in excess of the 25 gpm, which was found adequate in cooling calculations to prevent fuel damage.

3.6 Secondary Cooling System

Applicability: This specification applies to the primary system heat exchangers.

Objective: The objective is to maintain tritium releases as low as practicable.

Specification: The reactor shall be shut down and corrective action taken if primary coolant leakage through a heat exchanger to the secondary system exceeds any of the following limits:

(1) 36 gal in one day
(2) 50 gal in one week
(3) 180 gal in one year

Basis: At the end of the term of the NBSR license (2004), the tritium concentration in the primary coolant is calculated to be 5 mCi/mL. Using this value, the above criteria ensure that tritium concentrations in effluents will be as low as practicable and below concentrations allowed by 10 CFR 20.303 for liquid effluents and 10 CFR 20.106 for gaseous effluents (FSAR, NBSR 9, Addendum 1, Section 2.6.4, Nov. 1980).

The specified daily and weekly leakage rates represent the lowest limits of positive detection of D_2O losses under both reactor-operating and shutdown conditions. The specified yearly leak rate represents an estimate of the smallest size leak that can be positively located and repaired.

3.7 Fuel Handling and Storage

Applicability: This specification applies to the handling and storage of fuel elements or fuel experiments outside of the reactor vessel.

Objective: The objective is to prevent fuel element overheating or inadvertent criticality outside of the reactor vessel.

Specifications:

(1) All fuel elements or fueled experiments shall be stored and handled in a geometry such that the calculated k_{eff} is less than 0.9 under optimum conditions of water moderation and reflection.

(2) A fuel element shall not be placed in the fuel transfer chute or be otherwise removed from the reactor vessel unless the reactor has been shut down for a period equal to or greater than one hour for each megawatt of operating power level.

Basis: To ensure that no inadvertent criticality of stored or handled fuel elements occurs, they shall be maintained in a geometry that ensures an adequate margin below criticality. This margin is established as a k_{eff} of 0.9 for normal storage facilities or for handling outside of normal storage facilities.

To ensure that a fuel element, which may become stuck in the fuel transfer chute, does not melt and release radioactive material, a time limit is specified before a fuel element may be removed from the vessel following reactor shutdown.

Measurements carried out during reactor startup showed that for the hottest element placed dry in the transfer chute, 8 hours after shutdown from 10 MW, the maximum temperature is only 550°F without auxiliary cooling. Extrapolation of these measurements shows that 20 hours after shutdown from 20 MW, the maximum temperature for the hottest element would be less than 800°F without auxiliary cooling. For all other power levels below 20 MW, the specified waiting time would result in even lower temperatures. This provides a significant margin from the melting temperature of 1,200°F. These values are confirmed by fuel temperature tests carried out at the Oak Ridge Research Reactor. Therefore, the waiting times specified will preclude any fuel element damage or fission product release.

3.8 Fuel Handling Within Reactor Vessel

Applicability: This specification applies to fuel elements positioning in the reactor vessel.

Objective: The objective is to ensure that all fuel elements are latched between the reactor grid plates.

Specification: Following fuel handling within the reactor vessel, the reactor shall not be operated until all fuel elements that have been moved are inspected to determine that they are locked in their proper position in the core grid structure.

Basis: Each NBSR fuel element employs a latching bar, which must be rotated to lock the fuel element in the upper grid (FSAR, NBSR 9, Section 7.2.1.2, p. 7-3, Apr. 1966). Following fuel handling, it is necessary to ensure that this bar is properly positioned so that each element that has been moved cannot "wash out" when flow is initiated. Either of two inspection methods may be employed. A periscope can be used for visual inspection. Alternatively, a pickup tool can be positioned at or near the top of each element, one at a time. Tests have shown that flow from a primary main pump will raise an unlatched element above its normal position and thus will be detected by the pickup tool.

3.9 Normal and Postincident Exhaust Systems

Applicability: This specification applies to the normal ventilation and emergency exhaust system.

Objective: The objective is to ensure that normal and emergency ventilation equipment is operational.

Specifications: The reactor shall not be operated unless

(1) the emergency exhaust system is operable including both fans, each with at least one operable motor, and both the absolute and charcoal filters

(2) the reactor building exhaust system is capable of filtering exhaust air and discharging this air above the building roof level

Basis: The potential radiation exposure to persons at the site boundary and beyond has been calculated following an accidental release of fission product activity (FSAR, NBSR 9, Addendum 1, Section 3.4, Nov. 1980). These calculations are based on the proper operation of the emergency exhaust system to maintain the confinement building at a negative pressure and to direct all effluents through filters and up through the reactor building stack. The emergency exhaust system has been made redundant to ensure its operation. Because of its importance, this redundancy should be available at all times so that any single failure would not preclude system operation when required.

The normal reactor building exhaust is designed to pass reactor building effluents through high-efficiency particulate filters at least capable of removing particles of 0.3 microns or greater with an efficiency of at least 99% and discharge them above the reactor building roof level. This system ensures filtering and dilution of gaseous effluents before these effluents reach personnel either onsite or offsite. The system can properly perform this function using various combinations of its installed fans and building stack. Gaseous effluent monitors are required by Section 3.4(3) of these specifications.

3.10 Emergency Power Systems

Applicability: This specification applies to the emergency electrical power supplies.

Objective: The objective is to ensure emergency power for vital equipment.

Specifications: The reactor shall not be operated unless

(1) at least one of the diesel-powered generators, including the associated distribution equipment, is operable

(2) the station battery, including its associated distribution equipment, is operable

Basis: One diesel-powered generator is capable of supplying emergency power to all necessary emergency equipment. The second diesel-powered generator has been provided to permit outages for maintenance and repairs.

The station battery provides an additional source of emergency power for the nuclear instruments, the emergency exhaust fans, and the shutdown cooling pumps. These fans and pumps are provided with dc as well as ac motors. The battery is capable of supplying this emergency load for a minimum of 4 hours. By allowing this amount of time and by requiring operability of at least one diesel and the station battery, assurance is provided that adequate emergency power sources will always be available.

3.11 Miscellaneous Systems

Applicability: This specification applies to those miscellaneous systems necessary to ensure operation of the facility.

Objectives: The objectives are to monitor the helium sweep gas for possible fission products and to ensure that gaseous effluent releases are within acceptable limits.

Specifications: The reactor shall not be operated unless

(1) a continuous fission products monitor is operable or sample analysis for fission product activity is conducted at least daily

(2) the gaseous release from the confinement building is below the following activity levels of maximum permissible concentration (MPC):

Type of Radioactivity	Average Yearly Concentration	Maximum Concentration (Averaged Over 1 Day)
Particulates and halogens with half lives greater than 8 days	1.4	14
All other radioisotopes (including H^3)	10^3	10^4

Basis: A fission products monitor located in the helium sweep gas or the primary coolant will give indication of a "pin-hole" breach in the cladding so that early preventive measures can be taken. Because this monitor is not redundant, periodic sampling and analysis of the helium sweep gas must be substituted for periods when it is undergoing maintenance. The frequency chosen (daily) is adequate to ensure early detection of any small failures before they would be expected to grow significantly. Larger failures would occur only after an accidental reactor transient, which would be followed by a reactor shutdown. Part of the postincident evaluation would include a helium sweep gas sample, so that the existence of an actual failure would be detected before continuing operation.

The concentration limits specified ensure that 10 CFR 20 limits (MPC) are not exceeded at the site boundary. An allowance for dilution from the reactor building stack to the nearest site boundary of 1,000 (as justified in the FSAR, page 2-7) is given. This value of 1,000 from a diffusion view point is the minimum expected at the nearest site boundary under the least favorable meteorological conditions. This number could be increased by one or two orders of magnitude if normal variations in wind speed and direction were considered. Because these variations are not considered, a one or two order of magnitude margin is inherent in this limit.

The instantaneous release limit ensures that the average release is not obtained by a small number of very large releases with the attendent possibility of high local concentrations of released effluents. This specification, although more restrictive than 10 CFR 20, provides additional assurance that releases to offsite personnel are minimized.

In specifying the limits on particulates and long half-lived (greater than 8 days) halogens, consideration was given to the possibility of biological reconcentration in food crops or dairy products. Using available information (Soldat, J.D., Health Physics 9, p. 1170, 1963), a conservative reconcentration factor of 700 is applied. Thus, the limit for those isotopes is the Maximum Permissible Concentration as specified in Appendix B, Table II of 10 CFR 20 multiplied by the 1000 dilution factor divided by the 700 reconcentration factor (i.e., 1.4 MPC).

4.0 EXPERIMENTS

Applicability: This specification applies to any experiments to be installed within the NBSR.

Objectives: The objectives are to establish criteria for placing experiments in the NBSR and to establish limits on these experiments.

Specifications: Any experiment installed in the reactor shall meet the following criteria:

(1) The absolute reactivity of any experiment shall not exceed 0.5% $\Delta\rho$.

(2) The sum of the absolute values of reactivity of all experiments in the reactor and experimental facilities shall not exceed 2.6% $\Delta\rho$.

(3) No experiment malfunction shall affect any other experiment so as to cause its failure. Similarly, no reactor transient shall cause an experiment to fail in such a way as to contribute to an accident.

(4) Explosive or metastable materials capable of significant energy releases shall be irradiated in double walled containers that have been satisfactorily prototype tested with at least twice the amount of the material to be irradiated.

(5) Each experiment containing materials corrosive to reactor components or highly reactive with reactor or experimental coolants shall be double contained.

Basis: The individual experiment reactivity limit is chosen so that the failure of an experimental installation or component will not cause a reactivity increase greater than can be controlled by the regulating rod. Because the failure of individual experiments cannot be discounted during the operating life of the NBSR, failure should be within the control capability of the reactor. This limit does not include such semipermanent structural materials as brackets, supports, and tubes that are occasionally removed or modified, but which are positively attached to reactor structures. When these components are installed, they are considered structural members rather than part of an experiment.

The combined reactivity allowance for experiments was chosen to allow sufficient reactivity for contemplated experiments while limiting neutron flux depressions to less than 10%. Included within the specified 2.6% $\Delta\rho$ is a 0.2% $\Delta\rho$ allowance for the pneumatic irradiation system, 1.3% $\Delta\rho$ for experiments that can be removed during reactor operation, and the remainder for semipermanent experiments that can only be removed during reactor shutdown. Even if it were assumed that all of the 1.3% $\Delta\rho$ for removable experiments moved in 0.5 sec, analysis has shown that this ramp insertion into the NBSR operating at 20 MW would not result in any core damage (FSAR, NBSR 9, Addendum 1, Section 3.3.2, Nov. 1980). The 0.2% $\Delta\rho$ for the combined pneumatic irradiation systems is well below this referenced accident as well as being within the 0.5% $\Delta\rho$ capability of the regulating rod.

In addition to all reactor experiments being designed not to fail from internal overheating or gas buildup, they must also be designed to be compatible with their environment in the reactor. Specifically, their failures must not lead to failures of the core structure or fuel, or to the failure of other experiments. Also, reactor experiments must be able to withstand, without failure, the same transients that the reactor itself can withstand without failure (i.e., loss of reactor cooling flows, startup accident, and others where the reactor's safety system provides the ultimate protection.

The detonation of explosive or metastable materials within the reactor is not an intended part of the experimental procedure for the NBSR; however, the possibility of a rapid energy release must be considered when these materials are present. Because the analytical methods used for designing containers for very rapid energy releases are not well developed, full prototype testing of the containment design is specified. The requirement for testing twice the amount of material to actually be irradiated provides a safety margin of at least a factor of two to allow for possible experimental uncertainties.

Experiments containing materials corrosive to reactor components or highly reactive with reactor or experimental coolants, although limited by item (3) of this specification, provide the potential for reducing the integrity of the fuel elements. For this reason, an added margin of safety is required to prevent the release of these materials to the reactor coolant system. This margin of safety is provided by the double encapsulation, each container being capable of containing the material to be irradiated.

5.0 SURVEILLANCE STANDARDS

In the following sections, deviations from the specified performance frequencies for surveillance tests shall be permitted as follows:

(1) 5-year: intervals not to exceed 6 years
(2) biennially: interval not to exceed 30 months
(3) annually: intervals not to exceed 15 months
(4) semiannually: intervals not to exceed 7 1/2 months
(5) quarterly: intervals not to exceed 4 months
(6) monthly: intervals not to exceed 1 1/2 months
(7) weekly: intervals not to exceed 10 days
(8) daily: must be done during the calendar day

A surveillance requirement with a due date occurring during a reactor shutdown period, except area radiation monitoring requirements of Section 5.7(1) and environmental monitoring requirements of Section 5.9, may be deferred; all deferred surveillance tests shall be performed before resuming reactor operation, except when required for testing.

5.1 Confinement System

Applicability: This specification applies to the confinement building.

Objective: The objective is to ensure the continued reliability of the confinement building.

Specifications:

(1) A test of the operability of the confinement closure system shall be performed quarterly. The trip feature shall be initiated by each of the radiation monitors that provides a signal for confinement closure as well as by the manual major scram switch. A radiation source shall be used to test the trip feature of each of the radiation monitors at least annually.

(2) An integrated leakage test of the confinement building shall be performed at a pressure of at least 6.0 in. H_2O and a vacuum of at least 2.0 in. H_2O at least annually. If the maximum allowable leakage rate of 24 cfm/in. H_2O is exceeded in any test, the test frequency shall be increased to twice the previous frequency. It shall not be decreased until two successive tests are completed satisfactorily; whereupon, it may be decreased in steps by a factor of two until an annual frequency is reached.

(3) Any additions, modifications, or maintenance to the confinement building or its penetrations shall be tested to verify that the building can maintain its required leak tightness.

Basis: The confinement closure system is initiated either by a signal from the confinement building exhaust radiation detectors or manually by the major scram button. To ensure complete surveillance, the system is tested by using these same devices to initiate the test. In addition, checks of both the trip

features and the ability of the radiation detectors to respond to ionizing radiation are made.

A preoperational test program was conducted to measure the representative leakage characteristics at values of +7.5 in. H_2O and -2.5 in. H_2O (FSAR, NBSR 9, Section 3.7.2, p. 3-16, Apr. 1966). The specified test pressures and vacuums are acceptable because past tests have shown leakage rates to be linear with applied pressures and vacuums.

Changes in the building or its penetrations must be verified to withstand specified test pressures; therefore, tests must be performed before the building can be considered to be operable.

5.2 Reactor Coolant System

Applicability: This specification applies to the primary coolant system.

Objective: The objective is to ensure continued integrity of the primary coolant system.

Specifications:

(1) When aluminum heat exchangers are used, sacrificial spools and those tubes visible from the secondary inlet and outlet nozzles of the primary and purification system heat exchangers shall be visually inspected annually for evidence of pitting and corrosion.

(2) The reactor primary coolant system relief valve shall be lifted annually.

(3) Major additions, modifications, or repairs of the reactor coolant system or its connected auxiliaries shall be tested before use.

Basis: The most probable failure mechanisms for the primary coolant system are overpressure and corrosion. The only area where significant corrosion is possible is the secondary side of the main heat exchanger. To protect this all-aluminum heat exchanger from corrosion by secondary water, anodic sacrificial spool pieces are inserted in the secondary cooling system on either side of the heat exchanger. To protect against overpressure, a relief valve is installed on the primary system. To be effective, the condition of each of these protective devices must be verified periodically.

The corrosion rate of the aluminum spool pieces is expected to be only a few mils per year, hence, a yearly inspection should be adequate to detect any impending heat exchanger damage. To be sure that these sacrificial spools are performing properly, an inspection of accessible heat exchanger tubes and nozzles also is made. Excessive corrosion in these areas would be a strong indication that the sacrificial spools are not performing properly; conversely, lack of corrosion in these areas is indicative of proper performances of the sacrificial spools. The frequency for lifting the relief valve is consistent with industry practices on this type of valve for clear water service conditions.

Major additions, modifications, or repairs of the primary system shall be either pressure tested or checked by X-ray, ultrasonic, gas-leak test, dye-penetrant or similar methods.

5.3 Reactor Control and Safety System

Applicability: This specification applies to reactor control and safety systems.

Objective: The objective is to ensure the continued operability of reactor safety system instrumentation and control mechanisms.

Specifications:

(1) Reactivity worth of each shim and regulating rod shall be determined at least annually.

(2) The withdrawal and insertion speeds of each shim arm and the regulating rod shall be determined at least semiannually.

(3) Scram times of each shim arm drive shall be measured at least semiannually.

(4) Reactor safety system channels shall be tested for operability before each reactor startup following a shutdown in excess of 24 hours, or at least quarterly. This test shall include a verification of proper safety system channel trip settings. The safety channels shall be calibrated annually.

(5) A comparison of power range indication with flow-ΔT product shall be performed weekly when the reactor is operating above 5 MWt.

(6) Following maintenance on any portion of the reactor control or reactor safety systems, the repaired portion of the system shall be satisfactorily tested before the system is considered operable.

Basis: Measurements of reactivity worths of the shim arms have shown (over many years of operation) to vary slowly as a result of absorber burnup and only slightly with respect to operational core loading and experimental changes. An annual check will ensure adequate reactivity margins.

The shim arm drives are constant speed mechanical devices. Scram is aided by a spring that opposes drive motion during arm withdrawal. Withdrawal and insertion speeds or scram time should not vary except as a result of mechanical wear. The surveillance frequency is chosen to provide a significant margin over the expected failure or wear rates of these devices. The shim arms are considered operable for scram if they drop the top 5 degrees within 220 msec. This value is consistent with the amount and rate of reactivity insertion assumed in analyzing the accident requiring the most rapid scram (FSAR, NBSR 9, Addendum 1, Section 3.3.2, Nov. 1980).

Because redundancy of all important safety channels is provided, random failures should not jeopardize the ability of these systems to perform their required functions. However, to ensure that failures do not go undetected, frequent surveillance is required and specified.

Because various experiments require precise operating conditions, the NBSR has been designed to ensure that accurate recalibration of power level channels

can be easily and frequently achieved. The calibration is performed by comparison of nuclear channels with the thermal power measurement channel (flow-ΔT product). Because of the small ΔT in the NBSR (about 15° at 20 MW) these calibrations will not be performed below 5 MW for 10 MW operation or below 10 MW for 20 MW operation. However, to ensure that no gross discrepancies between nuclear instruments and flow-ΔT indicators occur, comparisons (but not necessarily calibrations) are made above 5 MW.

5.4 Reactor Emergency Cooling System

Applicability: This specification applies to the emergency cooling system.

Objective: The objective is to ensure proper operation of the emergency cooling system.

Specifications:

(1) Control valves in the reactor emergency cooling system shall be exercised quarterly.

(2) The starting function of the emergency cooling sump pump shall be checked quarterly. The operability of the pump using either heavy or light water shall be tested annually.

(3) The light water injection valves shall be exercised semiannually.

Basis: The proper operation, and, hence, the continued reliability of the emergency cooling system must be ensured. Because the equipment in this system is not used in the course of normal operation, its operability must be verified periodically. The frequencies are chosen so that deterioration or wear would not be expected to be an important consideration. Moreover, the frequency should be sufficient to ensure that the pumps and valves will not fail because of extended periods of standby operation. Possible failure resulting from corrosion buildup or other slow acting effects should become apparent with these surveillance schedules. Control and injection valves specified are those leading to or from the D_2O emergency cooling tank.

5.5 Secondary Cooling System

Applicability: This specification applies to secondary coolant activity.

Objective: The objective is to ensure adequate monitoring for radioactivity in the secondary cooling system.

Specifications:

(1) The N-16 monitor shall be tested for operability at least monthly and calibrated at least annually.

(2) When the N-16 monitor is operable, sampling of the secondary cooling water and analysis for tritium shall be conducted at least monthly. Should the N-16 monitor be inoperable, sampling of the secondary cooling water and analysis for tritium shall be performed at least daily.

Basis: Section 3.6 of these specifications places a limitation on leakage from the primary to the secondary coolant system. This limit can be maintained by monitoring for N-16 carryover, indicating a leak of water recently irradiated in the reactor, or by doing laboratory analysis for the presence of tritium. Both of these methods are employed.

The N-16 monitor is a simple radiation detection device sensitive to as little as 40 gal/day leakage. Its operability is expected for many years without repair; nevertheless, its failure at any time cannot be discounted or predicted. A determination of its operability at least monthly is considered a reasonable frequency for a device of this type. The annual calibration frequency is considered adequate to ensure that significant deterioration in accuracy does not occur.

Assuming operation of the N-16 monitor and no detectable loss of primary coolant (less than the 40 gal/day sensitivity), a monthly sampling for tritium should be adequate to detect small tritium leaks. If, however, the N-16 monitor is out of service, then sampling is the primary means of leak detection and more frequent sampling is required. A daily frequency is judged adequate since large leaks would still be detected by the level instruments that indicate a loss from the D_2O storage tank (sensitive to at least 300 gal).

5.6 Postincident and Gaseous Waste Systems

Applicability: This specification applies to the emergency exhaust system and the normal gaseous waste system.

Objective: The objective is to ensure the operability of the emergency and normal exhaust systems.

Specifications:

(1) An operability test of the emergency exhaust system, including the building static pressure controller and the vacuum relief valve, shall be performed at least quarterly.

(2) An operability test of the controls in the emergency control station shall be performed at least monthly. An inspection to determine that all instruments in the emergency control station are indicating normally shall be made at least daily.

(3) Absolute filters in both normal and emergency exhaust systems shall be tested for particulate removal efficiency at least biennially. The tests shall be designed to demonstrate that the absolute filters will remove 99% of particles with diameters of 0.3 microns and greater.

(4) Charcoal absorber banks in the emergency exhaust system shall be in-place tested with Freon or other halogen at least biennially to detect leakage paths caused by settling of the media or deterioration of the filter seals. Leaks greater than 1% of the total flow will be unacceptable and will require that the affected units be repaired or replaced.

Basis: The postincident gaseous waste system depends on the proper operation of the emergency exhaust system fans, valves, and filters, which are not routinely in service. Because they are not continuously used, their failure rate as a result of wear or loading should be low. On the other hand, since they are not being used, their condition in standby must be checked sufficiently often to ensure that they will function properly when needed. An operability test of the active components of the emergency exhaust system is performed quarterly to ensure that each component will be operable if an emergency condition requires use of the system. The quarterly frequency is considered adequate since this system receives very little wear and since the automatic controls are backed up by manual control provisions.

The test frequency for the efficiency of the absolute filters has been established as at least biennially. This frequency is consistent for filters subject to continuous air flow. Because the NBS absolute filters in the emergency exhaust system will be idle except during testing, deterioration should be much less critical than for filters subjected to continuous air flow where dust overloading and air breakthrough are possible after long periods of use. Therefore, a biennial testing frequency should be adequate in detecting filter deterioration. Absolute filter efficiency is checked by testing in-place, using a polydispersed aerosol of DOP or other suitable substitute.

The test requirement for the charcoal filters in the emergency exhaust system is basically a physical integrity test. It is prudent to verify that the NBSR filters are not installed or operated in such a way as to be damaged or bypassed. Therefore, a Freon, or other halogen gas, in-place leakage test is required annually to detect leakage paths resulting from charcoal settling and deterioration of the filter seals. Experience has demonstrated Freon, or other halogen gas, to be an acceptable means for determining the leakage characteristics of charcoal filter installations. The 1% acceptability limit is specified to give assurance that a high overall iodine filter efficiency significantly above the 95% used in the DBA will be maintained (FSAR, NBSR 9, Addendum 1, Section 3.4.2, Nov. 1980).

5.7 Radiation Monitoring Systems

Applicability: This specification applies to area and fission product monitors.

Objective: The objective is to ensure continued proper operation and calibration of area and fission products monitoring systems.

Specifications:

(1) Area monitors shall be tested for operability at least monthly and calibrated at least annually.

(2) The fission products monitor shall be tested for operability at least monthly and calibrated at least annually.

Basis: The area radiation monitors quite often give the first indication of a radioactive release resulting from an experiment or reactor malfunction. These monitors, similar to other plant radiation monitors, are simple radiation detection devices whose operability is expected for many years; nevertheless,

their failure at any time cannot be discounted or predicted. A determination of their operability monthly is considered reasonable for devices of this type. Because these devices are primarily used to detect an increase in activity over that which had previously existed, they are normally set at some reasonable value above background and their absolute accuracy is not critical. Hence, the annual calibration frequency is considered adequate to assure that a significant deterioration in accuracy does not occur.

The fission product monitor usually gives the first indication of a leak in the fuel cladding. This monitor, similar to other plant radiation monitors, is a simple radiation detection device whose operability is expected for many years; nevertheless, its failure at any time cannot be discounted or predicted. A monthly determination of its operability is considered reasonable for a device of this type. An annual calibration frequency is considered adequate to ensure that a significant deterioration in accuracy from its normal setting does not occur.

5.8 Emergency Power System

Applicability: This specification applies to the emergency electrical power equipment.

Objective: The objective is to ensure the availability of emergency power equipment.

Specifications:

(1) Each diesel generator shall be tested for automatic starting and operation at least monthly.

(2) Should one of the diesel generators become inoperative, the operable generator shall be tested for starting at least weekly.

(3) All emergency power equipment shall be tested under a simulated complete loss of outside power at least annually.

(4) The voltage and specific gravity of each cell of the station battery shall be tested semiannually. A discharge test of the entire battery shall be performed once every 5 years.

Basis: The NBSR is equipped with two diesel power generators, each capable of supplying full emergency load; therefore, only one of the generators is required. The monthly test frequencies are consistent with industry practice and are considered adequate to ensure continued reliable emergency power for necessary emergency equipment. In addition, an annual test of neccessary emergency power equipment under a simulated complete loss of outside power also is specified.

Hydrometer and voltage checks of individual cells are the accepted method of ensuring that all cells are in satisfactory condition. The semiannual frequency for these detailed checks is considered adequate to detect any significant changes in the ability of the battery to retain its charge.

During initial installation, the station battery was discharge tested to measure its capacity. Experience has shown this test should be repeated at 5-year intervals to detect deterioration of cells.

5.9 Environmental Monitoring

Applicability: This specification applies to the environmental monitoring program.

Objective: The objective is to determine the levels of radioactivity in the environment in the vicinity of the facility.

Specification: An environmental monitoring program shall be carried out and shall include as a minimum the quarterly analysis of samples from area streams, vegetation or soil, and air monitoring.

Basis: Consistent with the recommendation of the U.S. Geological Survey, a periodic sampling program of area wells and streams has been conducted since November 1962 (FSAR, NBSR 9, Section 2.4.3.2, p. 2-6, Apr. 1966). To ensure more complete sampling of the area surrounding the NBSR, this program has been expanded to include area vegetation or soil samples (soil samples being more meaningful during the nongrowing season). By 1982, most of the wells in the vicinity of the facility had been closed and only one well remained active, making further analysis of well water no longer meaningful. Sampling of area streams, however, is continuing and is required. Thermoluminescent dosimeters or other devices also are placed around the perimeter of the NBSR site to monitor the air. The continuation of this environmental monitoring program will ensure that the operation of the NBSR presents no significant hazard to the public health and safety. Since 1969 when the NBSR began routine power operation, the environmental monitoring program revealed nothing of significance thereby confirming that operation of the NBSR has little or no effect on the environment. The quarterly frequency is considered adequate to detect any long-term changes in the activity levels in the vicinity of the NBSR. Shorter term changes would require a significant release which would be detected by the exhaust system radiation monitors.

6.0 DESIGN FEATURES

6.1 Site Description

The reactor shall have a minimum exclusion radius to the nearest site boundary of 400 m. The reactor facility complex shall be located within NBS grounds and access to the reactor shall be controlled.

Basis: The location and ownership of the reactor site ensures necessary auxiliary services such as fire and security protection are available. The exclusion radius of 400 m is the distance on which all upper limit dose calculations are based (FSAR, NBSR 9, Addendum 1, Sections 2 and 3, Nov. 1980). Should this value decrease for any reason, a recalculation of upper limit doses would be necessary. Access to the reactor facility complex is controlled either by the facility staff or by a guard. In addition, access to the entire NBS campus is restricted at other than normal working hours.

6.2 Reactor Coolant System

The reactor coolant system shall consist of a reactor vessel, a single cooling loop, containing one or two, shell and tube, heat exchangers, and appropriate pumps and valves. All materials, including those of the reactor vessel, in contact with primary coolant (D_2O), shall be aluminum alloys or stainless steel, except gaskets and valve diaphragms. The reactor vessel shall be designed in accordance with the American Society of Mechanical Engineers (ASME) Code for Unfired Pressure Vessels. It shall be designed for 50 psig and 250°F. Heat exchanger tubes shall be designed for 100 psig and a temprature of 150°F. The connecting piping shall be designed for 125 psig and a temperature of 150°F.

Basis: The reactor coolant system has been described and analyzed in the FSAR as a single loop system containing two heat exchangers. Materials of construction, being primarily aluminum alloys and stainless steel, are chemically compatible with the D_2O coolant. The stainless steel pumps are heavy-walled members and are in areas of low stress, so they should not be susceptible to chemical attack or stress corrosion failures. The failure of the gaskets and valve bellows, although undesirable, would not result in catastrophic failure of the primary system; hence, strict material limitations are not required for technical specifications. The design, temperature, and pressure of the reactor vessel and other primary system components provide adequate margins over operating temperatures and pressures. It is believed prudent to retain these margins to further reduce the probability of a primary system failure. The reactor vessel was designed to Section VIII, 1959 Edition, of the ASME Code for Unfired Pressure Vessels. Subsequent changes should be made in accordance with the most recent edition of this Code.

Because the safety analysis is based on the reactor coolant system as presently designed and with the present margins, it is considered necessary to retain this design and these margins or to redo the analysis.

6.3 Reactor Core

The reactor core may consist of up to 30 (3.0 x 3.3 in.) MTR curved-plate-type fuel elements, except that the central 7-in. region of each fuel element shall contain no fuel. The middle 6 in. of aluminum in the unfueled region may be removed. The side plates, unfueled outer plates, and end adaptor castings of the fuel elements shall be aluminum alloy; the fuel plates shall be uranium-aluminum alloy; aluminum-uranium oxide or uranium-aluminide clad with aluminum.

Basis: The thermal design analysis and power distributions on which the analysis was based assumed an MTR-type fuel element with a 7-in. unfueled central region in the open lattice array and specific core loading patterns (FSAR, NBSR 9, Addendum 1, Sections 3.1 and 3.2, Nov. 1980). Significant changes in core loading patterns require a recalculation of the power distribution to ensure that burnout ratios are within acceptable limits.

7.0 ADMINISTRATIVE CONTROLS

7.1 Organization

The organization for the management and operation of the reactor facility shall be as indicated in Figure 7.1. The Chief, Reactor Radiation Division, and the Chief Nuclear Engineer, Reactor Operations, shall have line responsibility for direction and operation of the reactor facility, including safeguarding the general public and facility personnel from radiation exposure and adhering to all requirments of the Operating License and Technical Specifications.

The minimum qualifications with regard to education and experience backgrounds of key supervisory personnel in Reactor Operations shall be as follows:

(1) Chief Nuclear Engineer

The Chief Nuclear Engineer must have a college degree or equivalent in specialized training and applicable experience, and at least 5 years experience in a responsible position in reactor operations or related fields, including at least 1 year experience in reactor facility management or supervision.

(2) Deputy Chief Nuclear Engineer

The Deputy Chief Nuclear Engineer must have a combined total of at least 7 years of college level education and/or nucelar reactor experience with at least 3 years experience in reactor operations or related fields. The person must also be qualified to hold a senior operator's license.

(3) Reactor Supervisor

(a) At least 4 years experience in reactor operations, including experience in the operation and maintenance of equipment and in the supervision of technicians and/or reactor operators.

(b) A high school education or equivalent and formal training in reactor technology and reactor operations. (An additional 2 years of experience may be substituted for education and formal training.)

(c) Qualified to hold a senior operator's license.

For operation, the normal crew complement for a shift shall be three persons. The minimum crew complement for a shift shall be two persons, including at least one licensed senior operator.

7.2 Safety Evaluation Committee

The Safety Evaluation Committee shall be composed of at least four senior technical personnel who collectively provide a broad spectrum of expertise in reactor technology. The Committee members shall be appointed by the Chief, Reactor Radiation Division. At least two members shall be from the Reactor Radiation

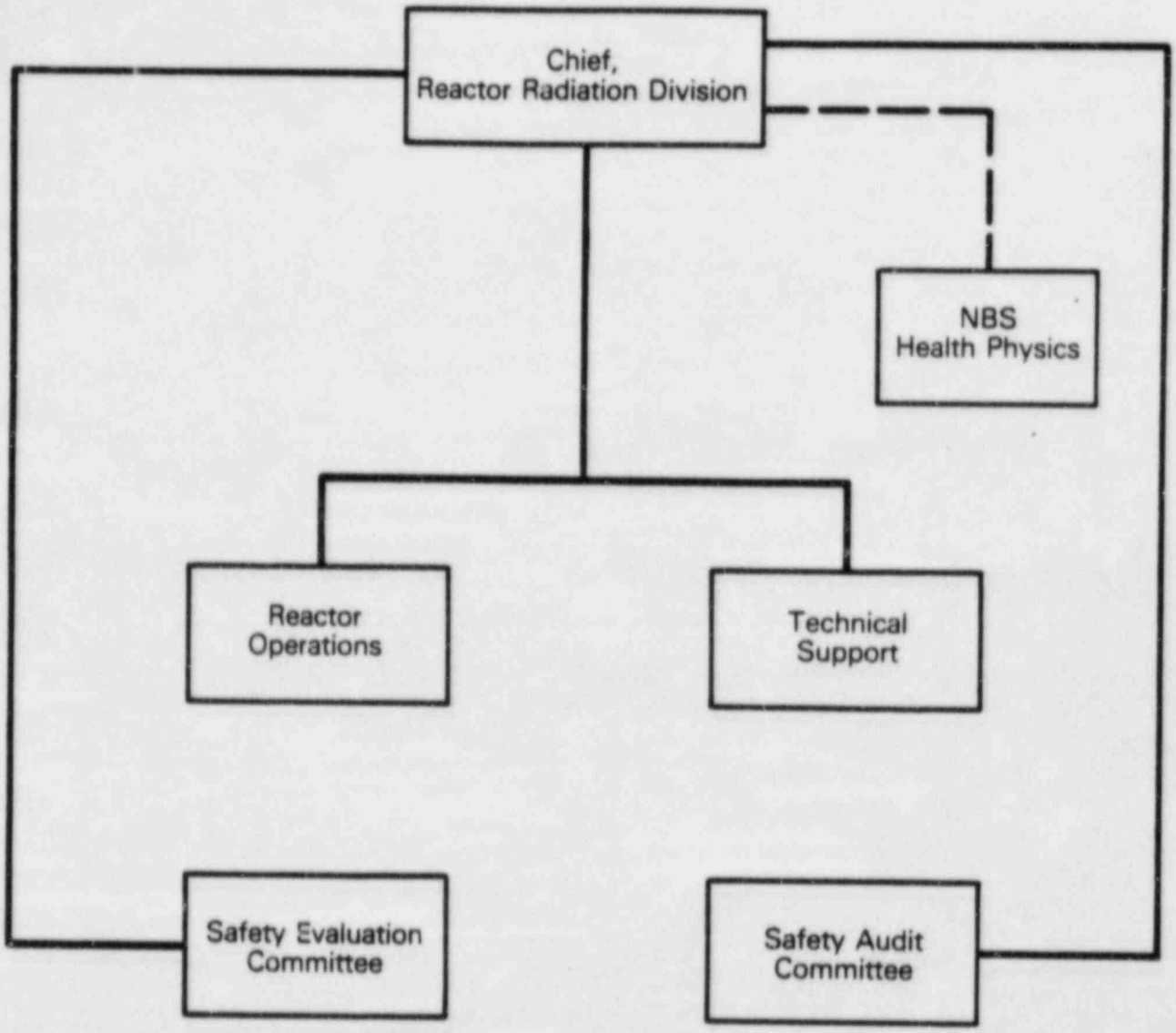

Figure 7.1 Organization for the management and operation of the reactor facility

Division and one shall be from Health Physics. The quorum requirement of the Committee shall be three members. Alternates appointed by the Chief, Reactor Radiation Division, may serve during the absence of regular members.

The Safety Evaluation Committee shall meet semiannually during reactor operations and as circumstances warrant. Written records of the proceedings, including any recommendations or concurrences, shall be maintained. The Committee shall report directly to the Chief, Reactor Radiation Division.

The Safety Evaluation Committee shall

(1) Review proposed changes to the NBSR facility equipment or procedures when such changes have safety significance, or involve an amendment to the facility license, a change in the Technical Specifications incorporated in the facility license, or an unreviewed safety question pursuant to 10 CFR 50.59.

(2) Review proposed tests or experiments significantly different from any previously reviewed or which involve any unreviewed safety question pursuant to 10 CFR 50.59.

(3) Determine whether proposed changes or reactor tests or experiments have been adequately evaluated and documented and provide recommendations for action.

(4) Review the circumstances of all reportable occurrences and violations of Technical Specifications and the measures taken to preclude a recurrence and provide recommendations for action.

7.3 Safety Audit Committee

The Safety Audit Committee shall be composed of three senior technical personnel who collectively provide a broad spectrum of expertise in reactor technology. The Committee members shall be appointed by the Chief, Reactor Radiation Division. Members of the Safety Audit Committee shall not be regular employees of the National Bureau of Standards. At least two members must pass on any report or recommendation of the Committee.

The Safety Audit Committee shall meet annually and as required. The Committee shall audit NBSR facility operations and the performance of the Safety Evaluation Committee. The Safety Audit Committee shall report in writing to the Chief, Reactor Radiation Division.

7.4 Procedures

All procedures and major changes thereto, before being effective, shall be reviewed by the Safety Evaluation Committee and approved in writing by the Chief, Reactor Operations, or his Deputy. Changes, which do not significantly change the original intent of a procedure, may be approved by the Chief, Reactor Operations, or his Deputy.

Written procedures shall be provided and utilized for the following:

(1) normal startup, operation, and shutdown of major components and systems

(These procedures shall include applicable check-off lists and instructions as required.)

(2) operator action necessary to correct specific equipment malfunctions and emergencies

(3) emergency conditions involving the potential or actual release of radioactivity

(4) radiation and radioactive contamination control

(5) a site emergency plan delineating the action to be taken in the event of emergency conditions and accidents that result in, or could lead to, the release of radioactive materials in quantities that could endanger the health and safety of employees or the public (Periodic evacuation drills for facility personnel shall be conducted to ensure that facility personnel are familiar with the emergency plan.)

(6) handling of irradiated and unirradiated fuel elements

7.5 Action To Be Taken in the Event a Safety Limit is Exceeded

If a safety limit is exceeded, the reactor shall be shut down and reactor operation shall not be resumed without authorization by the NRC pursuant to 10 CFR 50.36(c)(1). The NRC shall be notified in accordance with Section 7.8(1) of these specifications. A complete analysis of the circumstances leading to and resulting from the situation together with recommendations to prevent recurrence shall be submitted to the NRC.

7.6 Action To Be Taken if a Limiting Safety System Setting is Exceeded or a Limiting Condition of Operation is Violated

(1) Reactor conditions shall be returned to normal or the reactor shall be shut down. If it is necessary to shut down the reactor to correct the occurence, operations shall not be resumed unless authorized by the Chief, Reactor Operations, or his Deputy.

(2) Occurrence shall be reported to the Chief, Reactor Operations, or his Deputy and to the NRC, if required by Section 7.7 of these specifications.

(3) Occurrence shall be reviewed by the Safety Evaluation Committee at their next scheduled meeting.

7.7 Action To Be Taken in the Event of a Reportable Occurrence

(1) All reportable occurrences shall be promptly reported to the Chief, Reactor Operations, or his Deputy.

(2) All reportable occurrences shall be reported to the NRC in accordance with Section 7.8(1) of these specifications.

(3) All reportable occurrences shall be reviewed by the Safety Evaluation Committee.

(4) Reportable occurrences shall include, but not necessarily be limited to, the following:

(a) operation with actual safety system settings less conservative than limiting safety system settings specified in the Technical Specifications

(b) operation in violation of limiting conditions for operation, unless prompt remedial action is taken

(c) an uncontrolled or unanticipated significant reactivity change

(d) an uncontrolled or unanticipated significant release of radioactivity from the site

(e) an engineered safety system component malfunction or other component or system malfunction which could or threatens to render the affected system incapable of performing its intended safety function

(f) major degradation of one of the several boundaries which are designed to contain the radioactive materials resulting from the fission process

(g) an observed inadequacy in the implementation of major administrative or major procedural controls, such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operation

7.8 Reporting Requirements

In addition to the requirements of applicable regulations, and in no way substituting therefor, reports shall be made to the NRC as follows:

(1) A report not later than the following working day (by telephone or telegraph to the Director, NRC Region I) and a report within 2 weeks (in writing to the Director, Division of Reactor Licensing, NRC, Washington, D.C. 20555) of

(a) violations of safety limits
(b) reportable occurrences as defined in Section 7.7
(c) releases of radioactivity from the site above the permissible limits specified in Appendix B, Table II, 10 CFR 20

(2) A report within 30 days (in writing to the Director, Division of Reactor Licensing, NRC, Washington, D.C. 20555) of

(a) significant changes in the facility organization
(b) significant changes in the transient or accident analyses described in the Safety Analysis Report, as amended

(3) An annual operating report (in writing to the NRC Region I Office with a copy to the Director, Division of Reactor Licensing, NRC, Washington, D.C. 20555) providing the following information:

(a) a narrative summary of reactor operating experience, including the energy produced by the reactor (in megawatt-hours)

(b) the unscheduled shutdowns, including corrective action, if any, taken to preclude recurrence

(c) tabulation of major preventive and corrective maintenance operations performed having safety significance

(d) tabulation of major changes in the facility and procedures, and the tests and experiments, carried out without prior approval by the NRC pursuant to 10 CFR 50.59

(e) a summary of the nature and amount of radioactive material released into the sewer system and radioactive effluents discharged and the results of environmental surveys performed

(f) a summary of significant exposures received by facility personnel and visitors

7.9 Records

In addition to the records required by applicable regulations, the licensee shall maintain the following records for a period of at least 1 year.

(1) records of all safety or safety-related equipment maintenance activities, violations of Technical Specifications, reportable occurrences and those technical and safety considerations supporting the recommendations of the Safety Evaluation Committee, including action taken responsive to such recommendations

(2) records and logs of reactor operations

(3) records of principal maintenance activities

(4) records of surveillance activities performed in accordance with Section 5 of these Technical Specifications