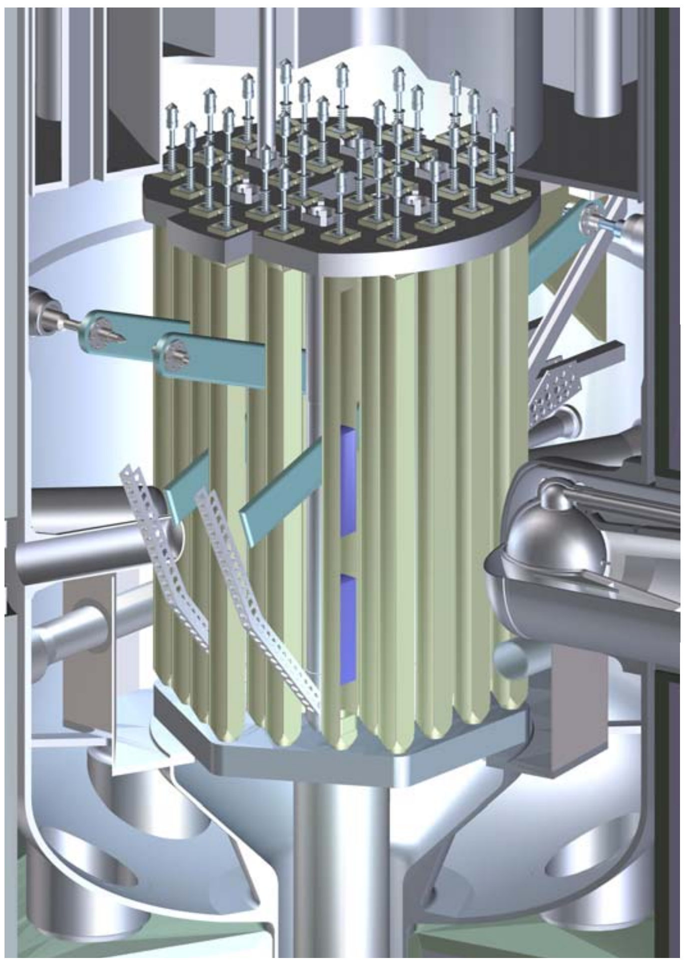
NISTIR 7104

Technical Specifications for License Renewal for the National Institute of Standards and Technology Reactor – NBSR

NBSR 15



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National Institute of Standards and Technology
Technology Administration, U.S. Department of Commerce

On the cover: A 3-D representation of the NBSR reactor core and internals.
Graphic Image by Paul Kopetka

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April 2004

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Appendix A

License No. TR-5

Technical Specifications
For the
National Bureau of Standards
20 Megawatt Test Reactor

April 2004

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1.0 INTRODUCTION

1.1 Scope

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 Test Reactor (NBSR) for operation up to and including 20MWt.

1.2 Application

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.

1.3 Definitions

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

1.3.1 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.

1.3.2 Confinement

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

1.3.3 Confinement Integrity

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

(1) All piping that penetrates the confinement building and is open to the confinement interior is physically intact exterior to the confinement.
(2) All automatic isolation valves in the ventilation, process piping and guide tubes 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 penetrations 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.

1.3.4 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.

1.3.5 Operable

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

1.3.6 Reactor Shutdown

The reactor shall be considered shutdown 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 in their OFF position with their keys removed.

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

1.3.7 Reactor Operating

The reactor is considered to be operating whenever it is not shutdown.

1.3.8 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.

(4) Moderator dump.

2.0 SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTING

2.1 Safety Limit

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:

(1) The reactor power, coolant system flow, and inlet temperature shall not exceed the limits shown in Figures 2.1 and 2.2.

(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 blistering temperature 752 ^{o}F (450 ^{o}C). For all reactor operating conditions that avoid either a departure from nucleate boiling (DNB) or the onset of flow instability (OFI), cladding temperatures remain substantially below the blister temperature. Conservative calculations (SAR, NBSR 14, Chapter 4) have shown that limiting combinations of reactor power and reactor coolant system flow and temperature to values more conservative than the safety limit will prevent cladding failure.

The analysis done in the SAR, NBSR 14, Appendix A, clearly show that the reactor can be operated at 500 kW with reduced or no flow.

2.2 Limiting Safety System Setting

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, ^{o}F (max)	147 (rundown)
Coolant flow, gpm / Mw (min)*	60 inner plenum
	235 outer plenum

Basis

At the values established, the Limiting Safety System Settings provide a significant margin from the Safety Limit. Even in the extremely unlikely event that all three parameters, reactor power, coolant flow, and outlet temperature simultaneously reach their Limiting Safety System Settings; the burnout ratio is at least 1.3. For all other conditions the burnout ratio is considerably higher (SAR, NBSR 14, Chapter 4). 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 the Limiting Safety System Setting.

* 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).

Table 2-1 Reactor Power Limits vs. Inner Plenum Flow.

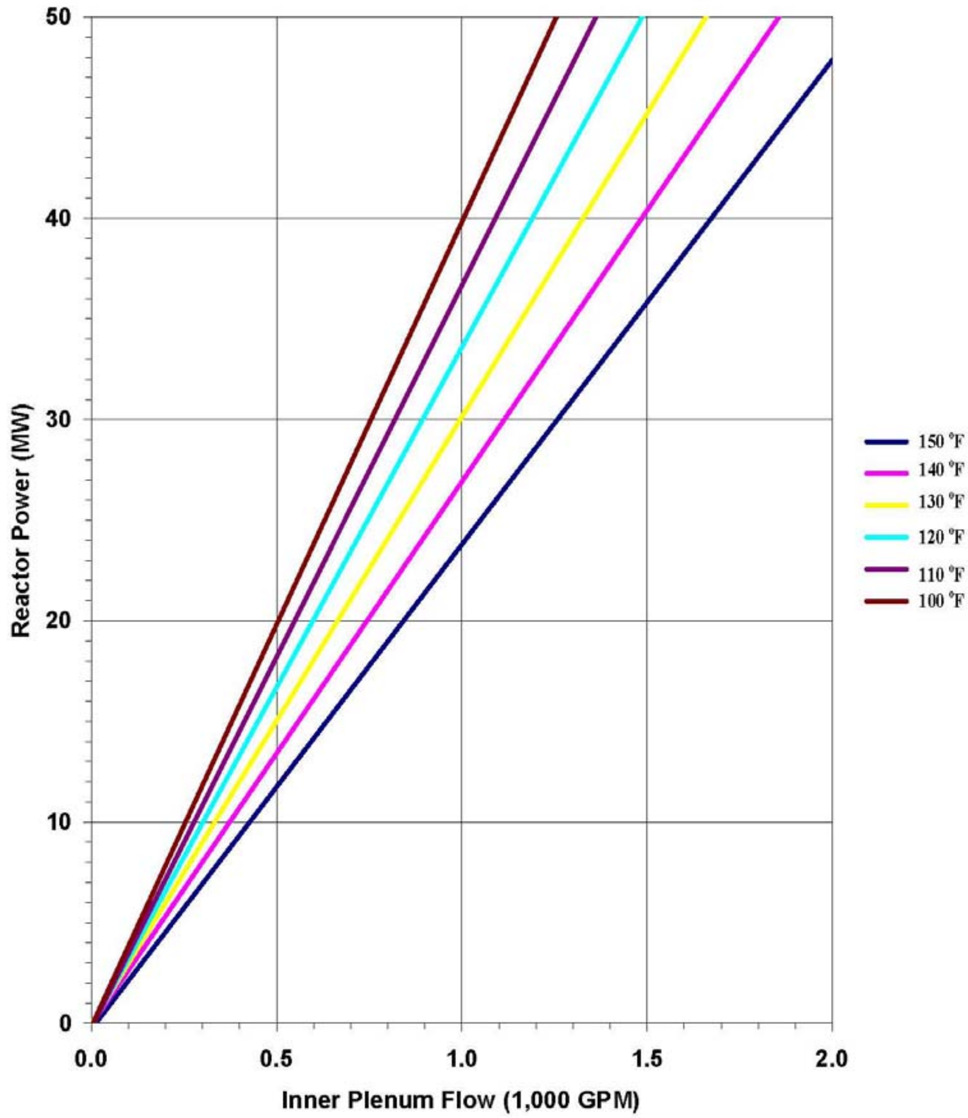
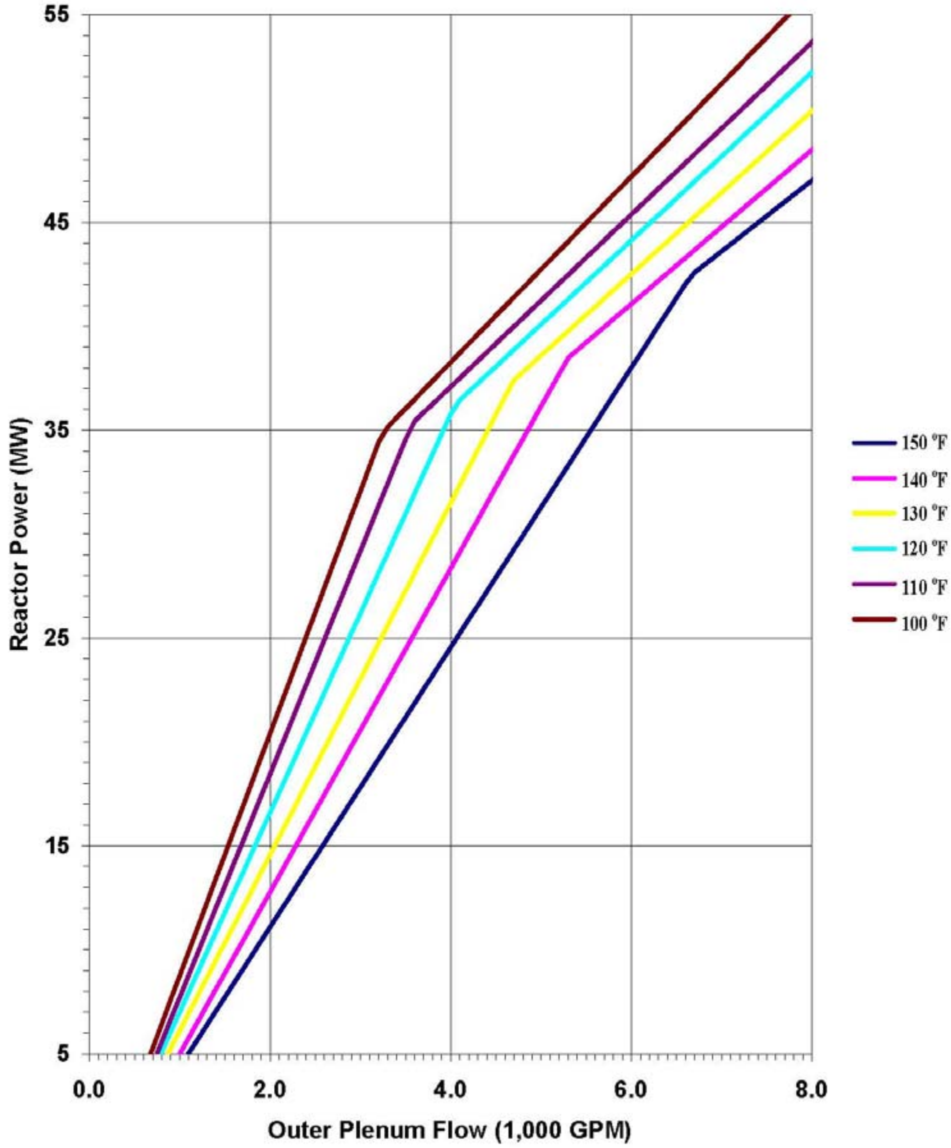


Table 2-2 Reactor Power Limits vs. Outer Plenum Flow.



3.0 LIMITING CONDITIONS FOR OPERATIONS

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.

Specification

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 shutdown for a period equal to or greater than one (1) hour for each megawatt of operating power level

Basis

The confinement system is a major engineered safety feature. It serves as the final physical barrier to mitigate the release of radioactive particles and gasses to the environment following accidents analyzed in the SAR, NBSR 14, and Chapter 13.

Confinement integrity is stringently defined to ensure that the confinement building shall perform in accordance with its design basis (SAR, NBSR 14, and Chapter 3). Piping, penetrations, and conduits that are open to the inside of the confinement building become an extension of confinement and shall be sealed on the exterior to the reactor building to prevent out-leakage. All other piping penetrations that do not have automatic closure devices shall be sealed within the confinement by sealing devices that can withstand the confinement test pressure of 6.0 inches of H_2O over pressure or 2.0 inches of 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 shall 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 one set of building vestibule doors per vestibule 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 shall 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 shall be required when these changes are made because they affect the status of the core.

Confinement integrity is not required when the reactor is shutdown and experiments are to be inserted or removed. The reactor is normally shutdown by a substantial reactivity margin. Experiments are usually inserted and removed one at a time; hence, the total reactivity change in any single operation shall be limited to the specified maximum worth of 0.5 % $\Delta\rho$ for any single experiment (including “fixed” experiments) were postulated, the maximum potential reactivity insertion would not exceed the 2.6 %$\Delta\rho$ (see Section 3.12 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 shutdown, 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, the transfer lock of the refueling system, the sealed shipping cask, and so on, that serve as a secondary barrier of fission product release, confinement integrity is not required. Maintenance that prevents normal rapid closing of the confinement is considered a breach of containment.

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 shall 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, eight hours after shutdown from 10 MW, the maximum temperature was only 550 ^{o}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 ^{o}F without auxiliary coolant. For all other power levels below 20 MW the specified waiting time would result in even lower temperatures. This provides a margin of safety from the lowest temperature at which blistering can occur 850 ^{o}F (450^{o}C). These values are confirmed by fuel temperature tests carried out at the Oak Ridge Research Reactor. Therefore, the waiting times specified will prevent any fuel element damage or fission product release.

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 and to provide the means of containing D_2O to H_2O heat exchanger leakage.

Specifications

The reactor shall not be operated unless :

(1) at least one shutdown cooling pump is operable
(2) the heat exchanger isolation valves are operable
(3) either a secondary cooling water activity monitor or a D_2O storage tank level monitor sensitive to a loss of 300 gallons of D_2O is operable
(4) the reactor vessel coolant level is within than 24 inches from the top of the overflow standpipe level, except during periods when operation at power levels up to 10 kW with no reactor flow are permitted (see section 2.1 of these specifications)
(5) the D_2 concentration in the helium sweep system is less than or equal to 4% by volume

Basis

The NBSR is equipped with shutdown cooling (as described in NBSR 14, Chapter 5), which provides ample cooling for all shutdown conditions. One of the accidents analyzed in Appendix A (SAR, NBSR-14) includes loss of off-site power (and hence main primary pumps), followed by failure of both redundant shutdown pumps. With no shutdown cooling flow, natural circulation in the fuel elements will result in a maximum fuel plate temperature of 107 °C (225 °F). This scenario results in no damage to the fuel, showing that natural convection cooling is adequate to provide cooling of the fuel in the shutdown condition, even immediately following a scram due to loss of all primary pumps. However, 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 (SAR, NBSR-14, Chapter 13). Limits on such leakage have been established in Section 3.6 of these specifications. To minimize the amount of any such leak-age, 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.

*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.

Objectives

The objectives are 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 (6) corner positions in the outer hexagonal ring, are filled with full length core assemblies. The six (6) corner positions shall 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 shutdown 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 (6) corner positions, although not required as part of the shim arm stops, shall 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 shall 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 is adequate
(2) The excess reactivity accident, (SAR, NSBR 14, Chapter 13), 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, (SAR, NBSR 14 Chapter 13), 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.

Objectives

The objectives are to ensure proper operation of reactor control and safety systems.

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 know 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.

A rod withdrawal accident for the NBSR is analyzed (SAR, NBSR 14, Chapter 13 and appendix A) using the maximum reactivity insertion rate, corresponding to the maximum beginning-of-life rod worths with the rods operating at the design speed of their constant speed mechanisms. The analysis showed that the most sever accident, a startup from source level, is bounded by the maximum reactivity insertion accident, and will not result in core damage.

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. The moderator dump provides a shutdown capability for any core configuration. Hence, it is also considered necessary for safe operation. It is shown (FSAR, NBSR 14, chapter 4) that the moderator dump provided sufficient negative reactivity to make the normal Start-Up (SU) core subcritical even with all four shim arms fully withdrawn.

Table 3.1 Reactor Safety System

Function	**Minimum Operable Channels**	
	Scrams	**Major Scrams**
High flux level (two (2) of three (3) or one (1) of two (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 (1) of two (2) channels may be bypassed for tests or during the time maintenance involving the replacement of components and modules or calibrations and repairs are actually being performed.

2.One (1) of these two (2) flow channels may be bypassed during tests, or during the time maintenance involving the replacement of components and modules or calibrations and minor repairs are 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.

3.5 Reactor Emergency Cooling System

Applicability

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

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

Specification
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 (1) sump pump to return spilled coolant to the overhead storage tank. Because only one (1) pump is used, it must be operational whenever the reactor is operational. There is sufficient D_2O available to provide 2.5 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 shutdown and corrective action taken if primary coolant leakage through a heat exchanger to the secondary system exceeds any of the following limits :
(1) 36 gallons in one day
(2) 50 gallons in one week
(3) 180 gallons in one year

Basis
At the end of the term of the NBSR license the maximum tritium concentration in the primary coolant is estimated to be 5 Ci/l. Using this value, the above criteria ensures that tritium concentrations in effluents shall be as low as practicable and below concentrations allowed by 10 CFR, part 20.303 for liquid effluents and 10CFR, part 20.106 for gaseous effluents (SAR, NBSR 14, Chapter 11).

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 fueled experiments outside of the reactor vessel.

Objective

The objective is to prevent fuel element overheating or inadvertent criticality outside 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} shall be 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 shutdown for a period equal to or greater than one (1) 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 exists. This margin is established as a k_{eff} less than 0.9 for the storage and handling of fuel or fueled experiments.

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 shall 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 , eight (8) hours after shutdown from10 MW, the maximum temperature was only 550 ^{o}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 ^{o}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 margin of safety from the lowest temperature at which blistering can occur 850 ^{o}F (450 ^{o}C).These values are confirmed by fuel temperature tests carried out at the Oak Ridge Research Reactor. Therefore, the waiting times specified will prevent any fuel element damage or fission product release.

3.8 Fuel Handling Within the Reactor Vessel:

Applicability

This specification applies to fuel element positioning within the reactor vessel.

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

Specifications
Following handling of fuel within the reactor vessel, the reactor shall not be operated until all fuel elements that have been handled are inspected to determine that they are locked in their proper positions in the core grid structure.

Basis
Each NBSR fuel element employs a latching bar, which shall be rotated to lock the fuel element in the upper grid plate (SAR, NBSR 14, Chapter 4). Following fuel handling, it is necessary to ensure that this bar is properly positioned so that each element that has been handled can not "wash out" when flow is initiated. Three methods may be used to verify bar position: with main pump flow, a single pickup tool may be lowered over the core and then positioned at or near the top of each element; with no flow, a tool designed to rotate the head and bar only in the locking direction may be used to confirm element position; or a visual inspection of the element heads or latching bars may be done.

3.9 Ventilation System

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

Objectives
The objectives are to ensure that the normal and emergency ventilation equipment is operational.

Specification
The reactor shall not be operated unless :

(1) the emergency ventilation system is operable including both fans, each with at least one operable motor and both the absolute and charcoal filters
(2) the reactor building ventilation 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 (SAR, NBSR 14, Chapter 13). These calculations are based on the proper operation of the emergency ventilation 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 ventilation 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 μm or greater with an efficiency of at least 99 % and discharge the effluents 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 the 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.

Objectives

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 is 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 four (4) hours. By allowing this amount of time and by requiring operability of at least one (1) diesel and the station battery, assurance is provided that adequate emergency power sources shall always be available.

3.11 Radiation Monitoring Systems

Applicability

This specification applies to those radiation monitoring 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 one day)
Particulates and halogens with half lives greater than eight (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 will give an 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 shall 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 post incident evaluation would include a helium sweep gas sample, so that the existence of an actual failure would be detected before continuing operation.

3.12 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 doubly contained

Basis

The individual experiment reactivity limit is chosen so that the failure of an experimental installation or component shall 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 semi-permanent 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 semi-permanent 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 seconds, analysis shows that this ramp insertion into the NBSR operating at 20 MW would not result in any fuel failure leading to the release of fission products (SAR, NBSR 14, Chapter 13). 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 shall also be designed to be compatible with their environment in the reactor. Specifically, their failures shall not lead to failures of the core structure or fuel, or to the failure of other experiments. Also, reactor experiments shall 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 shall 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 a least a factor of two (2) 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 three (3) of this specification, provides the potential for reducing the integrity of the fuel elements. For this reason, an added margin of safety shall be 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 materials to be irradiated.

4.0 SURVEILLANCE REQUIREMENTS

Introduction

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

(1) five (5) years : intervals not to exceed six (6) years
(2) biennially : interval not to exceed thirty (30) months
(3) annually : intervals not to exceed fifteen (15) months
(4) semiannually : intervals not to exceed seven and one half (7.5) months
(5) quarterly : intervals not to exceed four (4) months
(6) monthly : intervals not to exceed one and one half (1.5) months
(7) weekly : intervals not to exceed ten (10) days
(8) Daily : Shall 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 4.8(1) and environmental monitoring requirements of Section 4.9 may be deferred; all deferred surveillance tests shall be performed before resuming reactor operation, except when required for testing.

4.1 Confinement System

Applicability

This specification applies to the confinement building.

Objective

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

Specification

(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 inches of water and a vacuum of at least 2.0 inches of water at least annually. If the maximum allowable leakage rate of 24 cfm/inch of water is exceeded in any test, the test frequency shall be increased to twice the previous frequency. It shall not be decreased until two (2) successive tests are completed satisfactorily; whereupon, it may be decreased in steps by a factor of two (2) until an annually 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 inches of water and – 2.5 inches of water. (SAR, NBSR 14, Chapter 3) 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 shall be verified to withstand specified test pressures; therefore, tests shall be performed before the building Confinement System can be considered to be operable.

4.2 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.

Specification
(1) when aluminum heat exchangers are used, sacrificial spools and closed 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
Aluminum heat exchangers have been replaced by stainless steel heat exchangers. Stainless steel is less susceptible to corrosion and can withstand high pressures. Specification (1) has been retained to allow the uses of aluminum heat exchangers in the future. 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 penetrants or similar methods.

4.3 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 control mechanisms and safety system instrumentation.

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 multiplied by ΔT 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 shall ensure adequate reactivity margins.

A channel calibration of the reactor safety channels has been shown to be adequate for the present operating cycle.

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 shall be considered operable if they drop the top five (5^{o}) 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 (SAR, NBSR 14, Chapter 13).

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 times Δ T). Because of the small Δ T in the NBSR (about 15 ^{o}F 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 the flow ΔT indicators occur, comparisons (but not necessarily calibrations) are made above 5 MW.

4.4 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 shall be ensured. Because the equipment in this system is not used in the course of normal operation, its operability shall 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.

4.5 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. The NBSR employs two such N-16 systems for redundancy. However, only one is required to be operable and tested.

Assuming operation of the N-16 monitor and no detectable loss of primary coolant (less than 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).

4.6 Ventilation System

Applicability

This specification applies to the emergency exhaust system and the normal exhaust systems.

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 shall remove 99% of particles with diameters of 0.3μm and greater.

(4) Charcoal absorber banks in the emergency exhaust system shall be in-place tested with Freon or other halogens at least biennially to detect leakage paths

caused by settling of the media or deterioration of the filters seals. Leaks greater than 1% of the total flow shall be unacceptable and shall require that the affected units be repaired or replaced.

Basis

The emergency ventilation 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 should be low. But, since they are not being used continuously, their condition in standby shall be checked sufficiently often to ensure that they shall function properly when needed. An operability test of the active components of the emergency exhaust system shall be performed quarterly to ensure that each component shall be operable if an emergency condition should arise. The quarterly frequency is considered adequate since this system receives very little wear and since the automatic controls are backed up by manual controls.

The absolute filter efficiency shall be tested biennially. This frequency is appropriate for filters subject to continuous air flow. Because the NBSR absolute filters in the emergency exhaust system will be idle except during testing, deterioration should be much less 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.

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, an aerosol in-place leakage test shall be required biennially to detect leakage paths resulting from charcoal settling and deterioration of the filter seals. Experience has shown that the use of an aerosol gas to be an acceptable means for determining the leakage characteristics of charcoal filter installation.

4.7 Emergency Power Systems

Applicability

This specification applies to the emergency electrical power equipment.

Objective

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

Specification

(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 five (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 shall be required. The monthly test frequencies are consistent with industry practice and are considered adequate to ensure continued reliable emergency power for emergency equipment. In addition, an annual test of emergency power equipment under a simulated complete loss of outside power shall be done.

Specific gravity 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 that repeating this test at a five (5) year interval is adequate to detect deterioration of the cells.

4.8 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 the fission product monitors.

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 are simple radiation detection devices whose operability has been shown to be very good over many years. Therefore, 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 has 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 fuel element cladding leak. This monitor is a simple radiation detection device whose operability has been shown to be very good. Therefore, 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 settings does not occur.

4.9 Environmental Monitoring Program

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 study made by the U.S. Geological Survey, a periodic sampling program of area wells and streams has been conducted since November 1962 (SAR, NBSR 9, Chapter 2). To ensure more complete sampling of the area surrounding the NBSR, this program was expanded to include area vegetation and soil samples (grass samples are collected during the growing season, April through September and soil samples during the nongrowing season, October through March).

Based on a database search of wells currently permitted by Montgomery County (Montgomery County, 2003), there is currently only one potable well within a one-mile radius of the site. Therefore, there are no major users of groundwater within a one (1) mile radius of the reactor site. As WSSC supplies more water for this area and development continues, there are no anticipated future uses of groundwater within a one (1) mile radius of the site. Ground water is sampled when found to be available and accessible. Sampling of area streams and /or surface water, however, is continuing and shall be required.

Thermoluminescent dosimeters or other devices also are placed around the perimeter of the NBSR site to monitor direct radiation. The continuation of this environmental monitoring program will verify that the operation of the NBSR presents no significant risk 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 had 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.

A report published in March 2003 (URS, 2003 Geology, Seismology, Geotechnical Engineering, and Hydrology of the NIST Research Reactor Site, Gaithersburg, Maryland, Sections 2.4 and 2.5, March 28, 2003), supports the findings of previous studies conducted on the hydrology and geology of the NIST
site and vicinity. No significant changes in the hydrogeologic systems or ground water use were identified. This report further verifies the assumptions and techniques developed in 1962.

5.0 DESIGN FEATURES

5.1 Site Description

Applicability

This specification applies to the reactor site location.

Objective

The objective is to assure that features of the site location, if altered, would not significantly affect safety.

Specification

(1) The NBSR complex is located within the National Institute for Standards and Technology grounds and access to the reactor shall be controlled.

(2) The reactor shall have a minimum exclusion radius to the nearest site boundary of 400 meters.

Basis

The location and government ownership of the NBSR site ensures auxiliary services including fire and security shall be available. The exclusion radius of 400 m is the distance on which all unrestricted doses are calculated (SAR, NBSR 14, Chapter 13). Should this value decrease for any reason, a recalculation of the unrestricted doses would be necessary. Access to the reactor complex is controlled either by the facility staff or by NIST Police. In addition access to the entire NIST campus is restricted.

5.2 Reactor Coolant System

Applicability

This specification applies to the reactor coolant system.

Objective

The objective is to ensure compatibility of the primary coolant system with the design features in the Safety Analysis Report.

Specifications

(1) The reactor coolant system shall consist of a reactor vessel, a single cooling loop, containing heat exchangers, and appropriate pumps and valves

(2) all materials, including those of the reactor vessel, in contact with the primary coolant (D_2O), shall be aluminum alloys or stainless steel, except gaskets and valve diaphragms.

(3) 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^{o} F. The heat exchangers shall be designed for 100 psig and a temperature of 150^{o} F. The connecting piping shall be designed for 125 psig and a temperature of 150^{o} F.

Basis

The reactor coolant system has been described and analyzed (SAR, NBSR 14, Chapter 5) as a single loop system containing heat exchangers, pumps and valves. 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. A failure of the gaskets or valve bellows would not result in catastrophic failure of the primary system. The design temperature and pressure of the reactor vessel and other primary system components provide adequate margins over operating temperatures and pressures. The reactor vessel was designed to Section VIII, 1959 Edition of the ASME Code for Unfired Pressure Vessels. Any subsequent changes to the vessel should be made in accordance with the most recent edition of this Code. The safety analysis is based on the reactor coolant system as described in FSAR, NBSR 14, chapter 5. Therefore, it is considered necessary to retain this design and these margins.

5.3 Reactor Core and Fuel

Applicability

This specification applies to the design of the reactor core .

Objective

The assure compatibility of the reactor core with the safety analysis.

Specifications

(1) The reactor core may consist of up to 30 (3.0 x 3.3 inch) MTR curved plate type fuel elements. The NBSR MTR-type fuel elements shall be such that the central seven (7) inches of the fuel element contains no fuel. The middle six (6) inches of the aluminum in the unfueled region may be removed.

(2) the side plates, unfueled outer plates, and end adaptor castings of the fuel element shall be aluminum alloy .

(3) the fuel plates shall be uranium-aluminum alloy ; aluminum-uranium oxide or uranium-aluminide clad with aluminum.

Basis

The neutronic and thermal hydraulic analysis (SAR, NBSR 14, Chapter 4) was based on the use of the NBSR MTR-type thirty-four (34) plate fuel element. The NBSR fuel element has a seven (7) inch centrally located unfueled area, in the open lattice array. The middle six (6) inches of aluminum in the unfueled region has been removed. The analysis requires that the fuel be loaded in a specific pattern (SAR, NBSR 14, Chapters 4 and 13). Significant changes in core loading patterns require a recalculation of the power distribution to ensure that burnout ratios shall be within acceptable limits.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 Structure

The organization for the management and operation of the NBSR facility shall be as indicated in Figure 6.1.The Director, NIST Center for Neutron Research and the Chief Nuclear Engineer, 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 requirements of the Operating License and Technical Specifications.

6.1.2 Responsibility

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 (Chief Reactor Operations and Engineering) : The Chief Nuclear Engineer shall have a college degree or equivalent in specialized training and applicable experience, and at least five (5) years experience in a responsible position in reactor operations or related fields, including at least one (1) year experience in reactor facility management or supervision.

(2) Deputy Chief Nuclear Engineer (Chief Reactor Operations): The Deputy Chief Nuclear Engineer shall have a combined total of at least seven (7) years of college level education and/or nuclear reactor experience with at least three (3) years experience in reactor operations or related fields. The person shall also be qualified to hold a senior operator's license.

(3) Reactor Supervisor: The Reactor Supervisor shall have:
   - (a) At least four (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 two (2) years of experience may be substituted for education and formal training)
   - (c) be qualified to hold a senior operator's license

6.1.3 Staffing

For operation, the normal crew complement for a shift shall be three (3) persons. The minimum crew complement for a shift shall be two (2) persons, including at least one licensed senior operator.

6.2 Review and Audit

6.2.1 Safety Evaluation Committee (SEC)

The Safety Evaluation Committee (SEC) shall be composed of at least four (4) senior technical personnel who collectively provide a broad spectrum of expertise in reactor technology. The Committee members shall be appointed by the Director, NIST Center for Neutron Research. At least two (2) members shall be from the NIST Center for Neutron Research and one (1) shall be from Health Physics. The quorum requirement of the Committee shall be three (3) members. Alternates appointed by the Director, NIST Center for Neutron Research, may serve during the absence of regular members.

The SEC shall meet semiannually or more frequently if required. Written records of the proceedings, including any recommendations or concurrences, shall be maintained. The Committee shall report directly to the Director, NIST Center for Neutron Research.
The SEC shall:

a) review proposed tests or experiments significantly different from any previously reviewed or which involve any questions pursuant to 10 CFR, part 50.59
b) determine whether proposed changes or reactor tests or experiments have been adequately evaluated and documented and provide recommendations for action.
c) 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.
d) 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 questions pursuant to 10 CFR 50.59.

6.2.2 Safety Audit Committee (SAC)

The Safety Audit Committee (SAC) shall be composed of three (3) senior technical personnel who collectively provide a broad spectrum of expertise in reactor technology. The Committee members shall be appointed by the Director, NIST Center for Neutron Research. Members of the SAC shall not be regular employees of NIST. At least two (2) members shall pass on any report or recommendation of the Committee. The SAC shall meet annually and as required. The Committee shall audit the NBSR facility operations and the performance of the SEC. The SAC shall report in writing to the Director, NIST Center for Neutron Research.

6.3 Procedures

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

Written procedures shall be provided and utilized for the following :

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

6.4 Required Actions

6.4.1 Actions to Be Taken in the Event a Safety Limit is Exceeded

If a safety limit is exceeded, the reactor shall be shutdown and reactor operation shall not be resumed without authorization by the Nuclear Regulatory Commission (NRC) pursuant to 10 CFR, part 50.36 (c) (1). The NRC shall be notified in accordance with Section 6.5 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.

6.4.2 Action To Be Taken in the Event 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 shutdown. If it is necessary to shutdown the reactor to correct the occurrence, operations shall not be resumed unless authorized by the Chief Nuclear Engineer, or his Deputy.

(2) the event shall be reported to the Chief Nuclear Engineer, or his Deputy. The NRC shall be notified in accordance with Section 6.5 of these specifications.

(3) the event shall be reviewed by the SEC at their next scheduled meeting.

6.4.3 Action to Be Taken in the Event of a Reportable Occurrence

(1) all reportable occurrences shall be promptly reported to the Chief Nuclear Engineer, or his Deputy.

(2) all reportable occurrences shall be reported to the NRC in accordance with Section 6.5 of these specifications.

(3) all reportable occurrences shall be reviewed by the SEC.

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

  a) operation with actual safety system settings less conservative than limiting safety system settings specified in this document
  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

6.5 Reporting Requirements

In addition to the requirements of applicable regulations, reports shall be made to the NRC as follows :

(1) a report not later than the following working day to the NBSR, NRC Project Manager (telephone 301-415-1128 or email at name.nrc.gov) and a report within two (2) weeks (in writing to the USNRC, Document Control Desk, Washington DC, 20555) for any,

  a) violations of safety limits
  b) reportable occurrences as defined in Section 6.4.3
  c) releases of radioactivity from the site above the permissible limits
  d) specified in 10 CFR, part 20, Appendix B, Table II

(2) a report within thirty (30) days (in writing to the USNRC, Document Control Desk, Washington DC, 20555) for any,

  a) significant changes in the facility organization
  b) significant changes in the transient or accident analyses described in the SAR, NBSR 14, Chapter 13.

(3) an annual operating report (in writing to the USNRC, Document Control Desk, 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, procedures, and the tests and experiments, carried out under 10 CFR, part 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

6.6 Records

In addition to the records required by applicable regulations, the following records for a period of at least one (1) year shall be maintained:

(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 SEC, including actions taken to respond 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 4.0 of these Technical Specifications

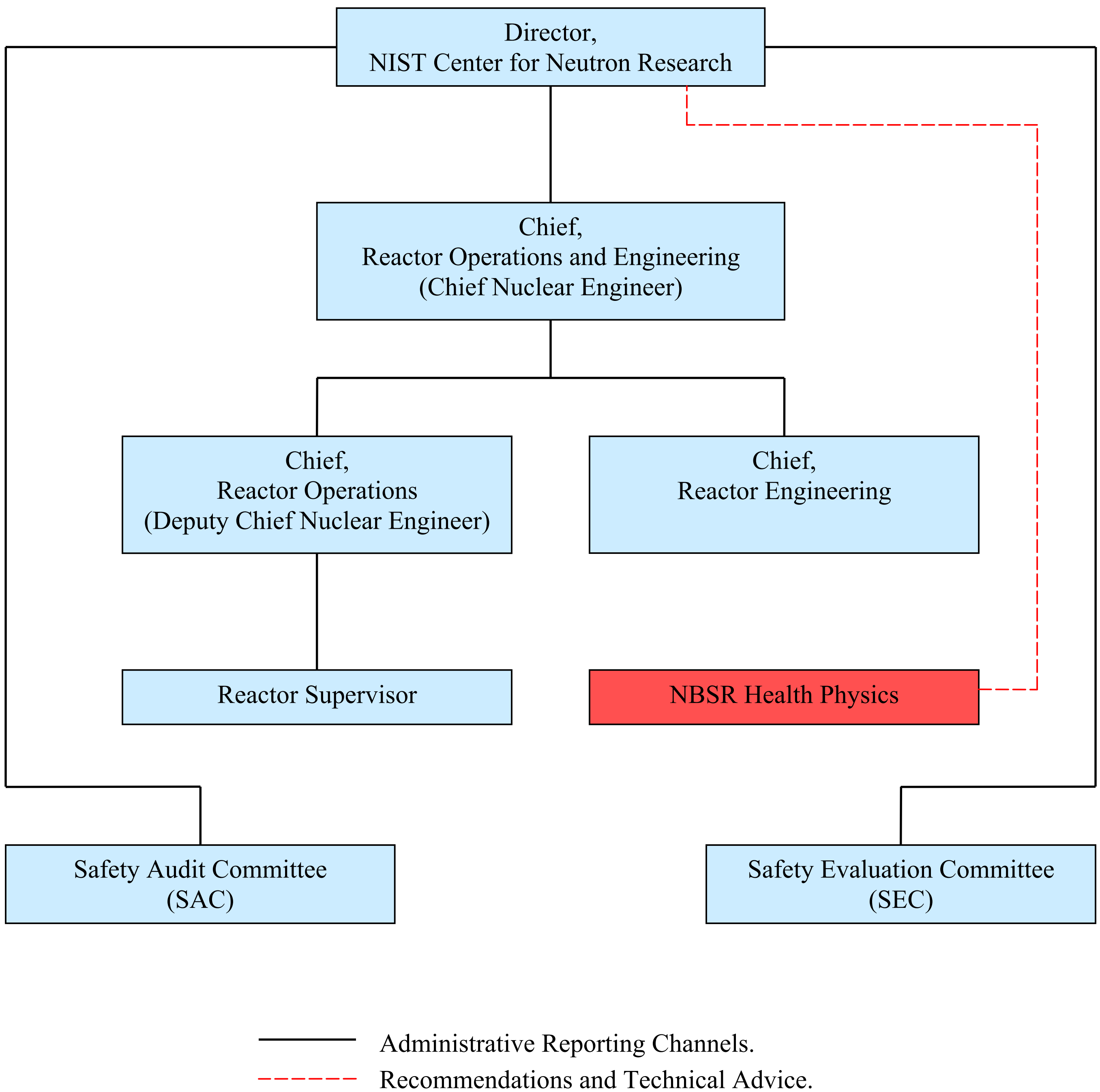


Figure 6.1 NBSR Operational Organization