UNITED STATES DEPARTMENT OF COMMERCE
National Institute of Standards and Technology
Gaithersburg, Maryland 20899-

May 25, 2017

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Supplement for request for changes to NBSR Technical Specifications to allow low power testing.

Ref: Docket 50-184, TR-5 Facility License

Sirs/Madams:

On May 15, 2017, NRC sent a request for us to supplement our license amendment request of March 2, 2017, to clarify some of the proposed changes. Attached is our response to that request. Also attached is a copy of the Technical Specification pages to be changed, with change bars. It should be noted, that as a result of this request, additional changes were made to technical specifications 2.2, 3.3.1, and 3.7.1.

Please contact Dr. Thomas Newton at (301) 975-6260 if you have any questions.

Respectfully,

Robert M. Dimeo, Director
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100 Bureau Drive, MS 6100
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I declare under penalty of perjury that the foregoing is true and correct.

Executed on May 25, 2017

By: ______________________

cc: Xiaosong Yin, BRR/DPR/PRLB

A020
NRR

NIST

NCNR response to NRC Supplemental Information request of 5/15/17

1) *The proposed amendment refers to an "unknown core" loading and gives examples. Are the neutronics, thermal-hydraulics and compliance of the core with the TSs of "unknown cores" unknown? Discuss the modeling performed to predict core behavior before core loading including compliance with TS requirements. Discuss the attributes of your core loading procedures (hold points, intermediate acceptance criteria) that help ensure that the "unknown core" is within the licensing basis and compliant with the TSs.*

All core configurations are analyzed prior to any fuel loading operation. The term "unknown" was meant to apply to a configuration in which criticality or other reactivity conditions have not yet been measured. The NBSR has a fixed refueling pattern that has been extensively neutronically and thermal-hydraulically modeled and any deviations from that pattern would require further analysis prior to fuel movement to ensure TS compliance from both a reactivity and power peaking standpoint. In order to avoid confusion, we are proposing to change the word "unknown" in TS 2.2 and 3.3.1 to "previously unmeasured." Infrequently, the core is fully unloaded for maintenance purposes. Upon reloading, a strict procedure (RP-19 or RP-20) is followed whereby a small number of fuel elements are loaded in predetermined positions (the number and position depend on whether or not the shim arms are new), along with inverse multiplication measurements and verification of estimated criticality.

2) *The proposed amendment implies that natural convection operation would only occur for "unknown core" loading. If this is true, your proposed TS should include this restriction. If not true, the difference between "unknown core" activities and normal natural convection mode operation needs to be described in your application and reflected in your proposed TSs.*

We are proposing to reword 2.2(4) as follows: " (4) Reactor power, with natural circulation cooling flow, shall not exceed 10 kW. Operation in this mode shall only be made with a core that has been previously analyzed and shown to be within the envelope of conditions described in the SAR."

3) *You proposed that TS 2.2(2) not be applicable to natural convection operation. Provide a safety analysis or a reference to your existing safety analysis that shows that a limit on coolant temperature in natural convection mode operation is not needed.*

Measurement of power is the most accurate indication of reactor thermal-hydraulic conditions during natural convection operation. As analyzed in section 4.6.5 of the NBSR SAR, the reactor can be operated in natural convection mode safely above 500 kW, well above the TS limit of 10 kW. This analysis showed that even with in-core coolant temperatures near saturation, there was ample margin to CHF and OFI.

4) *Some TSs discussed in your application do not appear to have any proposed changes to the specifications (e.g. TS 3.1.2). Confirm what changes you are proposing to the TSs.*

A few TS were included in the change document because repagination was necessary to fit proposed wording in other TSs. The TSs proposed to be changed are: 2.2, 3.1.3, 3.2.2, 3.3.1, 3.3.2, 3.7.1, 3.9.2.1, 4.7.1 (correct a typographical error), and Figure 6.1.

5) *It is not clear if you are requesting that some TSs (e.g., 3.1.2, 3.2.1 (3)) not be applicable during operation with an "unknown core" or if compliance can be shown by calculations or other means until measurements can be taken. Clarify what you are proposing and submit a safety analysis to support your request.*

As was stated in the answer to question 1, we never knowingly load a core in a configuration that has not been analyzed to be in compliance with all relevant TSs. We are not asking for an exemption from the reactivity TSs listed in the question, but it should be understood that verification measurements are not possible until the core is completely loaded and such measurements can take place.

6) *The basis of proposed TS 3.1.3 is changed to remove shim arm failure. Confirm that the shim arm stops will continue to be able to perform their function during forced convection operation. One of the "unknown core" activities in your application is shim replacement. How would operators know that a shim arm had failed in natural convection mode operation with open grid positions that prevented the shim arm stops from performing their function*

Shim arm stops are permanently installed in the core grid. Immediately following shim arm replacement, the first fuel elements to be reinserted are those adjacent to the shim arm positions. These, along with installed vertical thimbles, ensure that the shim arms remain in place. Assuming the raising and lowering mechanisms remained functional, a failed shim arm would be detected by abnormalities in the required 1/M plots. If the mechanisms failed, the operator would immediately notice and halt any further activities until the apparent failure was investigated.

7) *Several proposed TS changes discuss the reflector dump. Clarify the heavy water levels used in the application. Explain the reactivity difference from dumping the reflector from difference starting levels (e.g., 154 inches and 70 inches).*

When fuel is moved to and from the reactor core to the spent fuel pool, it is necessary for the level to remain at the refueling level (an indicated 70 inches), as this prevents D_2O from draining into the spent fuel pool.

As shown in Figures 3-30 and 3-31 of the NBSR SAR Appendix A, virtually all of the reactivity worth of the reflector is within 45 cm (18 inches) of the top of the fuel. Thus, it makes no difference from a reactivity point of view whether the starting level is an indicated 154 inches (~117 inches above the top of the fuel) or 70 inches (24 inches above TOF). Accordingly, we are now proposing to leave TS 3.3.3 as is.

8) *The proposed basis added for TS 3.9.2.1 stated that "[U]ntil main pump flow is used the flow forces are not present to cause the lifting of the elements if they are not latched." Explain how a fuel element is confirmed to be properly seated without latching it and what is the*

mechanism verifying it. What prevents operation of the primary cooling system when the reactor is in natural convection mode? Perform a safety analysis to support this statement.

Upon insertion of a fuel element into the grid, the head is rotated, engaging a latching bar. This verifies that the element is properly seated. Also, immediately after inserting and latching the element, the height of the fuel element head is verified using the refueling tool and a metal bar lying on the refuel index top. This measurement would indicate the whether the element is latched or not. Although the element is measured during refueling to be latched, verification using the elevation check method in TS 3.9.2.1 (1) cannot be performed until primary flow is established, and the rotational check in TS 3.9.2.1 (2) requires a special latch check tool be inserted, which could result in higher personnel tritium doses during multiple loadings of a complete core. In a no-flow condition and operating at a maximum of 10kW, no credible accidents would result in exceeding the safety limit (See SAR 4.6.5).

Both refueling procedures Reference Procedure (RP)-19 (Reloading Plan with known Shims) and RP-20 (Reloading Plan after installation of new shims) have a Limitation and Precaution that states "Do not operate the main or shutdown D_2O pumps until loading is completed."

9) *Your application contains a section discussing change of the basis for TS 3.7.1(1). However, it appears you have not proposed any changes to the TS. Provide your proposed revised wording.*

This was an inadvertent omission. Proposed wording of the basis is as follows (eliminating "G-M"):

(1) The requirements of 10 CFR 20.1502(b) (2007) are met by regular monitoring for airborne radionuclides and bioassay of exposed personnel. The two primary airborne radionuclides present at the NBSR are ^{41}Ar and ^{3}H. The normal air exhaust system draws air from areas supplied by conditioned air, such as the first and second floors of the confinement building. The irradiated air exhaust system draws air from areas most likely to have contaminated air, such as waste sumps and penetrations in the biological shield. Normal and irradiated air are monitored continuously with detectors sensitive to β and γ emissions and the combined air is exhausted through the stack. The stack release is monitored with a detector sensitive to β and γ emissions.

2.0 Safety Limit and Limiting Safety System Settings

2.1 Safety Limit

Applicability: Fuel temperature

Objective: To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products.

Specification

The reactor fuel cladding temperature shall not exceed 842°F (450°C) for any operating conditions of power and flow.

Basis

Maintaining the integrity of the fuel cladding requires that the cladding remain below its blistering temperature of 842°F (450°C). For all reactor operating conditions that avoid either a departure from nucleate boiling (DNB), or exceeding the Critical Heat Flux (CHF)), or the onset of flow instability (OFI), cladding temperatures remain substantially below the fuel blistering temperature. Conservative calculations have shown that limiting combinations of reactor power and reactor coolant system flow and temperature will prevent DNB and thus fuel blistering.

2.2 Limiting Safety System Settings

Applicability: Power, flow, and temperature parameters

Objective: To ensure protective action if any combination of the principal process variables should approach the safety limit.

Specifications

(1) Reactor power shall not exceed 130% of full power.

(2) Reactor outlet temperature shall not exceed 147°F.

(3) Forced coolant flow shall not be less than 60 gpm/MW for the inner plenum and not less than 235 gpm/MW for the outer plenum.

(4) Reactor power, with natural circulation cooling flow, shall not exceed 10 kW. Operation in this mode shall only be made with a core that has been previously analyzed and shown to be within the envelope of conditions described in the SAR.

Basis

At the values established above, the Limiting Safety System Settings provide a significant margin from the Safety Limit. Even in the extremely unlikely event that reactor power, coolant flow, and outlet temperature simultaneously reach their Limiting Safety System Settings, the critical heat flux ratio (CHFR) is at least 2. For all other conditions the CHFR is considerably higher. This will ensure that any reactor transient caused by equipment malfunction or operator error will be terminated well before the safety limit is reached. Overall uncertainties in process instrumentation have been incorporated in the Limiting Safety System Settings.

Steady state thermal hydraulic analysis shows that operation at less than 500 kW with natural circulation results in a CHFR and OFI ratio greater than 2. Transient analysis of reactivity insertion accidents shows that the fuel cladding temperature remains far below the safety limit. The limit of 10 kw was chosen since that was deemed adequate for any operational situation requiring natural circulation operation, such as testing of a previously unmeasured core loading.

Basis

(1) An excess reactivity limit provides adequate excess reactivity to override the xenon buildup and to overcome the temperature change in going from zero power to 20 MW, without affecting the required shutdown margin. In addition, the maximum reactivity insertion accident at startup, which assumes the insertion of 0.5% $\Delta\rho$ into a critical core, is not affected by the total core excess reactivity.

(2) These specifications ensure that the reactor can be put into a shutdown condition from any operating condition and remain shutdown even if the maximum worth shim arm should stick in the fully withdrawn position with the regulating rod also fully withdrawn.

3.1.3 Core Configuration

Applicability: Core grid positions

Objective: To ensure that effective fuel cooling is maintained during forced flow reactor operation.

Specification

The reactor shall not be operated with forced coolant flow unless all grid positions are filled with full length fuel elements or thimbles.

Basis

Core grid positions shall be filled to prevent coolant flow from bypassing the fuel elements for operation of the reactor with forced coolant flow.

3.1.4 Fuel Burnup

Applicability: Fuel

Objective: To remain within allowable limits of burnup beginning-of-life shim arm worths with the shim arms operating at the design speed of their constant speed mechanisms. The analysis shows that the most severe accident, a startup from source level, will not result in core damage.

3.2.2 Reactor Safety System Channels

Applicability: Required instrument channels

Objective: To provide protective action for nuclear and process variables to ensure the LSSS values are not exceeded.

Specifications

The reactor shall not be operated unless the channels described in Table 3.2.2 are operable and the information is displayed in the reactor Control Room.

Table 3.2.2 Reactor Safety System Channels
Minimum Nuclear and Process Channels Required

Channel	Scram	Major Scram	Rundown
(1) High Flux level	2		
(2) Short period below 5% rated power	2		
(3) Low reactor vessel D_2O level [1, 3]	2		
(4) Low flow reactor outlet [2, 3]	1		
(5) Low flow reactor inner or outer plenum [2, 3]	1		
(6) Manual (outside of the Control Room)	1		
(7) Manual	1	1	
(8) Reactor Outlet Temperature[3]			1
(9) Gaseous Effluent Monitors[4]		2	

[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] May be bypassed during periods of reactor operation when a reduction in Limiting Safety System Settings are permitted by the specifications of Sections 2.2(4) and 3.3.1(1).

[4] See specifications of Section 3.7.1

Basis

The nuclear and process channels of Table 3.2.2 initiate protective action to ensure that the safety limit is not exceeded. With these channels operable, the safety system has redundancy.

The exceptions are required to perform surveillance, maintenance or operation permitted by the specification of Sections 2.2(4), 3.3.1(1), and 3.7.1.

3.3 Coolant System

3.3.1 Primary and Secondary

Applicability: Primary fluid systems

Objective: To prevent degradation of primary systems' materials.

Specifications

The reactor shall not be operated unless:

(1) The reactor vessel coolant level is no more than 25 inches below the overflow standpipe.

Exception: To perform periodic surveillance of the effectiveness of the moderator dump or approach to critical testing for a previously unmeasured core loading, it is necessary to operate the reactor as permitted in the specifications of Section 2.2(4) and without restriction on reactor vessel level above the dump tube.

(2) The D_2 concentration in the Helium Sweep System shall not exceed 4% by volume.

(3) All materials, including those of the reactor vessel, in contact with the primary coolant shall be compatible with the D_2O environment.

Basis

(1) 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 of approximately 7 feet (2.13 m) above the fuel elements. To permit periodic testing, such as surveillance of the

effectiveness of the moderator dump or approach to critical for a previously unmeasured core loading, it is necessary to operate the reactor without restriction on reactor vessel level. This is permissible under conditions when forced reactor cooling flow is not required, such as is permitted in the specifications of Section 2.2(4).

(2) 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 77°F (25°C) in helium if mixed with air). To ensure a substantial margin below the lowest potentially explosive value, a 4% limit is imposed.

(3) Materials of construction, being primarily low activation alloys and stainless steel, are chemically compatible with the primary 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. Other materials should be compatible so as not to cause a loss of material and system integrity.

3.3.2 Emergency Core Cooling

Applicability: Emergency Core Cooling System

Objective: To ensure an emergency supply of coolant.

Specifications

The reactor shall not be operated, except under Section 2.2(4), unless:

(1) The D_2O emergency core cooling system is operable.

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

Basis

(1) In the event of a loss of core coolant, the emergency core cooling system provides adequate protection against melting of the reactor core and associated release of fission products.

(2) The emergency core cooling system employs one sump pump to return spilled coolant to the overhead storage tank. Because only one sump pump is used, it must be operational whenever the reactor is operational. There is sufficient D_2O available to provide approximately 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 overhead storage tank is exhausted, domestic water or a suitable alternative would be used to furnish water for once-through cooling. The water makeup capacity must be in excess of 25 gpm, which was found adequate in cooling calculations to prevent fuel damage.

Full operability is not available, nor is it needed, when operating as permitted by the specifications of Sections 2.2(4) and 3.3.1(1). However, the 3000 gallon D_2O emergency cooling tank and a source of makeup water would be available.

3.3.3 Moderator Dump System

Applicability: Moderator dump

Objective: To provide a backup shutdown mechanism.

Specification

The reactor shall not be operated unless the reactor moderator dump system is operable.

Basis

In the unlikely event that the shim 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 considered necessary for safe operation. It has been shown that the moderator dump provides sufficient negative reactivity to make the normal startup (SU) core subcritical even with all four shim arms fully withdrawn.

3.4 Confinement System

3.4.1 Operations that Require Confinement

Applicability: Reactivity changes within the vessel and fuel movements outside of the vessel

Objective: To provide an additional barrier to fission product releases.

Specifications

Confinement shall be maintained when:

(1) The reactor is operating.

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

(3) There is movement of irradiated fuel outside a sealed container or system.

(4) The reactor has been shutdown for shorter than the time specified in the specification of Section 3.9.2.2.

(2) One fission product monitor is operable or sample analysis for fission product activity is conducted daily.

(3) One secondary coolant activity monitor is operable or a D_2O storage tank level monitor is operable. [1]

(4) Two area radiation monitors are operable on floors C-100 and C-200.

(5) The primary tritium concentration is less than or equal to 5 Ci/l.

(6) Removed to 3.7.2.

When required monitors are inoperable, then portable instruments, survey or analysis may be substituted for any of the normally installed monitors in specifications (1) – (4) for periods of one (1) week or for the duration of a reactor run.

1. Operability of the monitors specification (2) and (3) above are not required for operation permitted by the specifications in Section 2.2(4) since these systems are neither operable nor needed without forced primary and secondary coolant flow.

Basis

(1) The requirements of 10 CFR 20.1502(b) (2007) are met by regular monitoring for airborne radionuclides and bioassay of exposed personnel. The two primary airborne radionuclides present at the NBSR are ^{41}Ar and ^{3}H. The normal air exhaust system draws air from areas supplied by conditioned air, such as the first and second floors of the confinement building. The irradiated air exhaust system draws air from areas most likely to have contaminated air, such as waste sumps and penetrations in the biological shield. Normal and irradiated air are monitored continuously with detectors sensitive to β and γ emissions and the combined air is exhausted through the stack. The stack release is monitored with a detector sensitive to β and γ emissions.

(2) 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. When this monitor is not functional, daily testing will ensure that the fuel cladding is intact. These two measures ensure that there are no undetected releases of fission products to the primary coolant. Specification (1) alone is adequate to assure detection of abnormal effluent radioactivity during operation as permitted by Section 2.2(4).

(3) Monitoring for primary water leakage into the secondary coolant is done by a secondary water monitor that is sensitive to radionuclides in the primary water. Leakage of primary to secondary would also be detected by a change in the D_2O storage tank level

(4) Fixed gamma area radiation monitors are positioned at selected locations in the confinement building. Typical alarm setting are less than 5 mrem/hr and adjusted as needed for non-routine activities, generally with the objective of identifying unusual changes in radiation conditions.

(5) At the end of the term of the NBSR license the maximum tritium

3.9.2 Fuel Handling

3.9.2.1 Within the Reactor Vessel

Applicability: Fuel element latching

Objective: 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, except under Section 2.2(4), 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. This shall be accomplished by one of the following methods:

(1) Elevation check of the fuel element with main pump flow.

(2) Rotational check of the element head in the latching direction only.

(3) Visual inspection of the fuel element head or latching bar.

Basis

Each NBSR fuel element employs a latching bar, which shall be rotated to lock the fuel element in the upper grid plate. Following fuel handling, it is necessary to ensure that this bar is properly positioned so that an element cannot be lifted out of the lower grid plate, which could lead to a reduction in flow to the element after pump flow is initiated. Any of the three methods above may be used to verify bar position. Tests have shown that flow from a primary pump will raise an unlatched element above its normal position and thus will be detected by the pickup tool under flow conditions. The efficacy of rotational checks has been confirmed by visual inspections.

Operation of the reactor in natural circulation at $\leq$10 kW is permitted prior to verifying that the elements are latched. Until main pump flow is used the flow forces are not present to cause the lifting of the elements if they are not latched.

3.9.2.2 All Other Conditions

Applicability: Refueling system

Objective: To ensure the integrity of the fuel element cladding.

(4) The voltage and specific gravity of each cell of the Vented Lead Acid (VLA) battery shall be tested annually. A discharge test of the entire battery shall be performed once every 5 years.

(5) A discharge test of the Valve-Regulated Lead Acid (VRLA) batteries shall be performed once every two years.

Basis

(1) 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 diesel generators have proven to be very reliable over decades of service. The quarterly test frequencies are consistent with industry practice and are considered adequate to ensure continued reliable emergency power for emergency equipment.

(2) This testing frequency of the operable generator will ensure that at least one of the required emergency generators will be operable.

(3) An annual test of the emergency power equipment under a simulated complete loss of outside power will ensure the source will be available when needed.

(4) and (5), Specific gravity and voltage checks of individual cells are the accepted method of ensuring that all cells of a VLA battery are in satisfactory condition. The annual 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 batteries were discharge tested to measure their capacity. Experience has shown that repeating these tests at the specified intervals is adequate to detect deterioration of the cells and loss of battery capacity.

4.7 Radiation Monitoring System and Effluents

4.7.1 Monitoring System

Applicability: Radiation monitoring equipment

Objective: To ensure operability of radiation monitors.

Specifications

(1) The gaseous effluent monitors for normal air, irradiated air and stack air shall be channel tested before startup, after a shutdown of longer than twenty-four (24) hours, or quarterly. Each of the above air monitors shall be channel calibrated annually.

(2) The fission products monitor shall be channel tested monthly and channel calibrated annually.

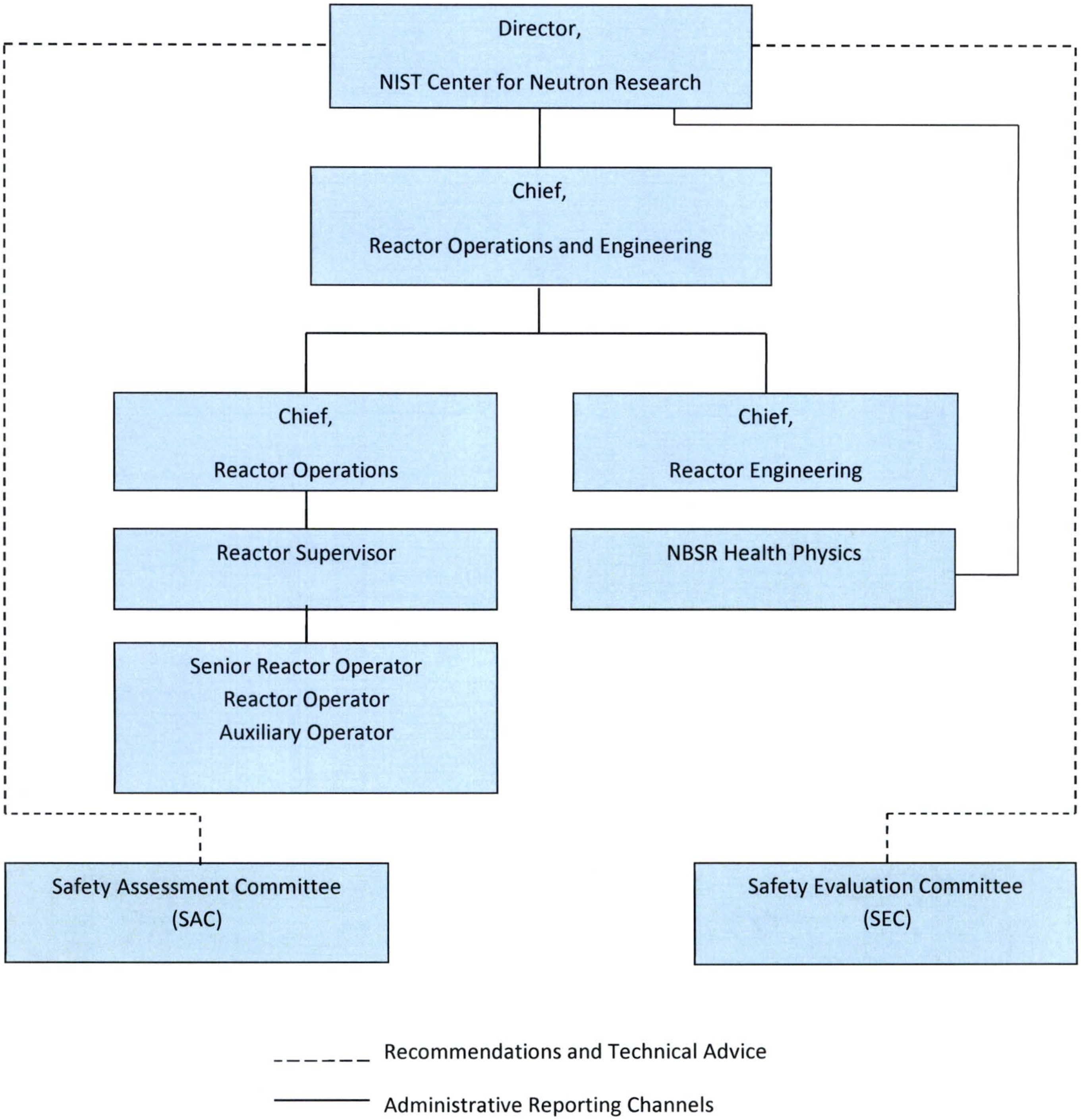
Director,
NIST Center for Neutron Research
Chief,
Reactor Operations and Engineering
Chief,
Reactor Operations
Chief,
Reactor Engineering
Reactor Supervisor
NBSR Health Physics
Senior Reactor Operator
Reactor Operator
Auxiliary Operator
Safety Assessment Committee
(SAC)
Safety Evaluation Committee
(SEC)
Recommendations and Technical Advice
Administrative Reporting Channels

Figure 6.1