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

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

September 16, 2008

Subject: NRC Request for Additional Information (TAC No. MD3410) dated July 1, 2008.

Docket No. 50-184

Gentlemen,

Please find the attached response to NRC Request for Additional Information (TAC No. MD3410), the revised NBSR Technical Specifications and the NBSR Emergency Plan.

Questions concerning these responses should be directed to Dr. Wade J. Richards, Chief, Reactor Operations and Engineering, (301-975-6260) or wade.richards@nist.gov.

Sincerely,

Wade J Richards

Wade J. Richards
Chief, Reactor Operations and Engineering
NIST Center for Neutron Research

I certify under penalty of perjury that the following is true and correct.

Executed on: Sept 16, 2008 by: Wade J Richards

cc.
William B. Kennedy
U.S. Nuclear Regulatory Commission
MS 012-G15
Washington, D.C. 20555-0001

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NRR

NIST

NIST Response to NRC Request for Information (TAC No. MD3410), August 19, 2008

<u>4.1 Section 4.6, Thermal hydraulic Design & TS 2.2, LSSS</u>. Provide justification for 500 kW power operations under natural convection flow by demonstrating that no credible accidents would result in exceeding the safety limit. (See TS question 2.2)

<u>Response</u>
The only credible accidents for this case where there is no forced flow are those initiated by reactivity insertion. Therefore, we have analyzed both the maximum reactivity insertion and startup accidents for natural circulation. In this case, the best measure of the safety margin is the clad temperature, rather than the CHFR, because flow can oscillate during the excursion. These excursions are initiated exactly as in the forced flow case, and are terminated by the period scram, which is tripped when the period decreases below 5 s. For conservatism, a 3-s delay between reaching the trip point and actual trip initiation is assumed. In Table 1 below, the results for both initiation sequences are shown for both Startup (SU) and End of Cycle (EOC) cores.

Table 1. Summary of natural circulation accident scenarios.

Accident Scenario	Maximum Power (MW)	Time (sec)	Maximum Clad Temperature (K)
Maximum Reactivity Insertion – SU	3.89	3.02	400
Maximum Reactivity Insertion – EOC	4.02	3.08	397
Startup Accident - SU	2.03	8.14	393
Startup Accident - EOC	2.09	8.20	390

These results show that the maximum fuel clad temperature is 400 K for the maximum reactivity insertion accident with the startup core, which is much less than the safety limit of 723 K (450 °C). The evolution of the maximum reactivity insertion accident is shown in Figures 1 and 2 below. Note that the value of the maximum temperature is dominated by the 3-s delay.

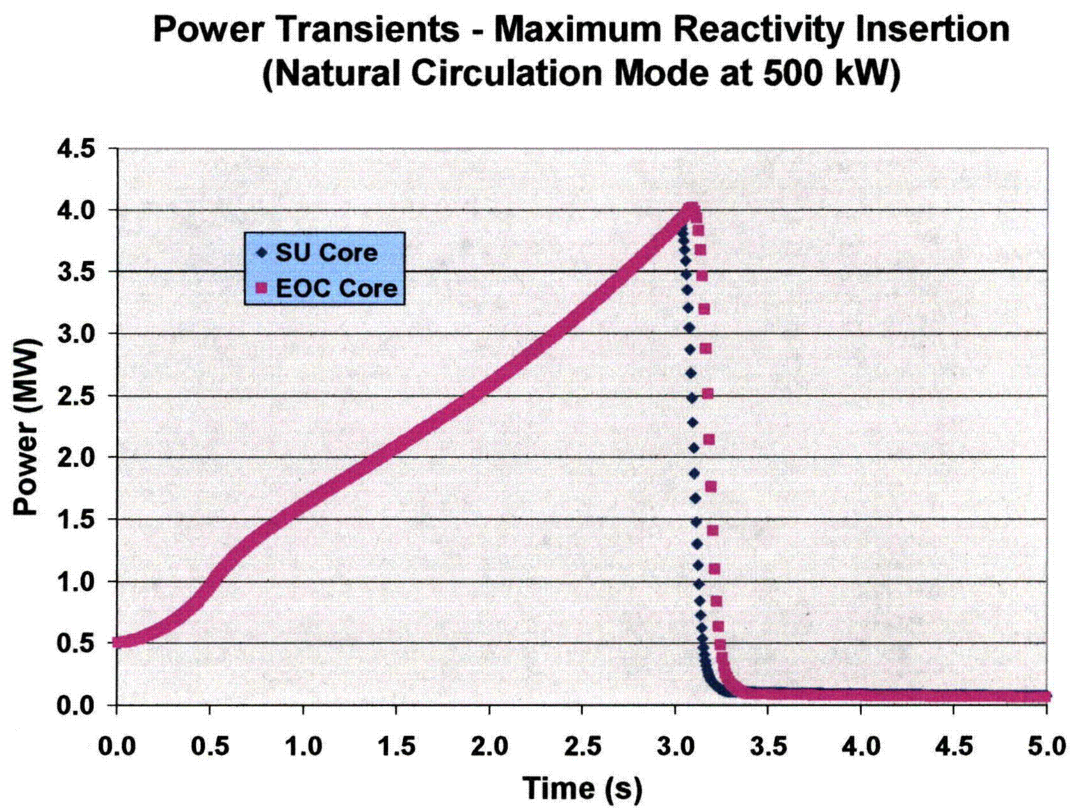

Figure 1. Power transients for the maximum reactivity insertion accident (0.5 %$\Delta\rho$ in 0.5 s) starting from 500 kW with natural circulation for the SU and EOC cores. Note that the scram level is reached at approximately 0.2 s, but the actual scram is delayed by 3 s for conservatism to account for any delay in scram signal propagation.

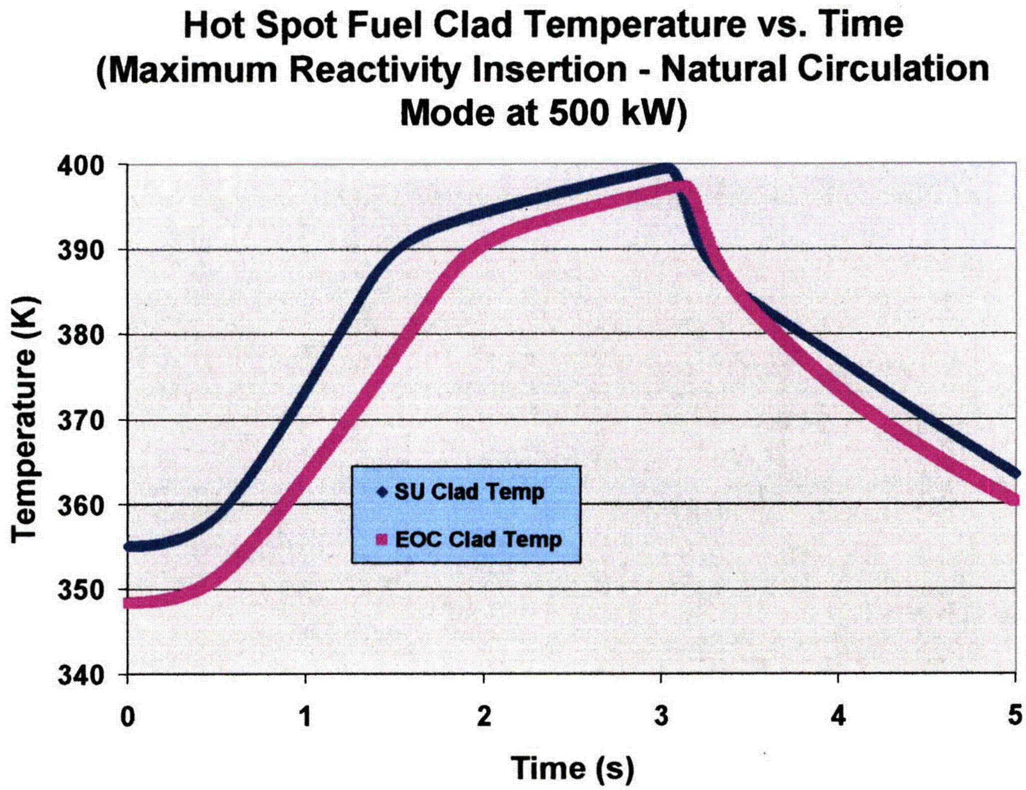

Figure 2. Fuel plate clad temperatures for the maximum reactivity insertion accident (0.5 %$\Delta\rho$ in 0.5 s) starting from 500 kW with natural circulation. Note that the scram level is reached at approximately 0.2 s, but the actual scram is delayed by 3 s for conservatism to account for any delay in scram signal propagation.

This analysis shows that there is ample margin between the maximum clad temperature in any credible accident and the safety limit of 450 °C.

....

4.2. Section 4.6, Thermal Hydraulic Design, p.4-49. Provide justification that the Costa correlation is the appropriate and limiting method for calculating the critical heat flux given the coolant pressure found in the NBSR fuel channels.

Response

The Costa correlation for the Onset of Flow Instability (OFI) was chosen for the analysis because it had been used in both the ILL reactor and the HFIR reactor. Although the minimum pressure studied in the derivation of the correlation was 1.75 bar, while the pressure in the NBSR is of order 1.3 bar, there is no reason to expect a large change in physical behavior at the lower pressure. The pressure enters the correlation through the pressure dependence of the saturation temperature, and this quantity varies smoothly between 1.3 and 1.75 bars. A factor of 1.3 was used with the Costa correlation to account for possible errors. In order to check this correlation, we have redone the analysis using the Saha-Zuber correlation[1], which relates to the onset of net vapor generation (ONVG), which is considered a precursor of instability. This was also the point used by Costa to identify the onset of flow instability. Both correlations were developed using circular and rectangular flow channels, but the subsequent analysis was quite different for the two cases.

In order to study and compare the predictions of the two correlations, we have studied the predictions for the case where the outlet temperature of the water is fixed at the LSSS of 147 °F, and the flow is varied up to approximately 8000 gpm total, or 6000 gpm for the outer plenum and 1600 gpm for the inner plenum. At each flow, the inlet temperature for the power corresponding to the LSSS for flow is calculated, and the predictions of the two models are calculated, along with the critical heat flux, calculated using the Mirshak correlation. At each flow, the actual limit occurs in the outer plenum, and therefore, this is the result plotted in Figure 3 below.

The first point that should be noted is that a safety factor of 1.3 is included in all plotted results to allow for the uncertainties in the correlations. The Saha-Zuber correlation predicts a lower power limit over almost the entire range; however, the difference between the two predictions is within the estimated combined uncertainty of the correlations. This result shows that the LSSS proposed are very conservative relative to either correlation. It should also be noted that the Onset of Net Vapor Generation, which is taken as the basis for both correlations, is an ill-defined quantity, determined in quite different ways in the derivation of the two correlations. ONVG is a *precursor* of flow instability, giving added conservatism to the settings.

[1] "Point of net vapor generation and vapor fraction in sub-cooled boiling", proceeding of Fifth International Heat Transfer Conference, Vol. 4, pp175-179 (1974).

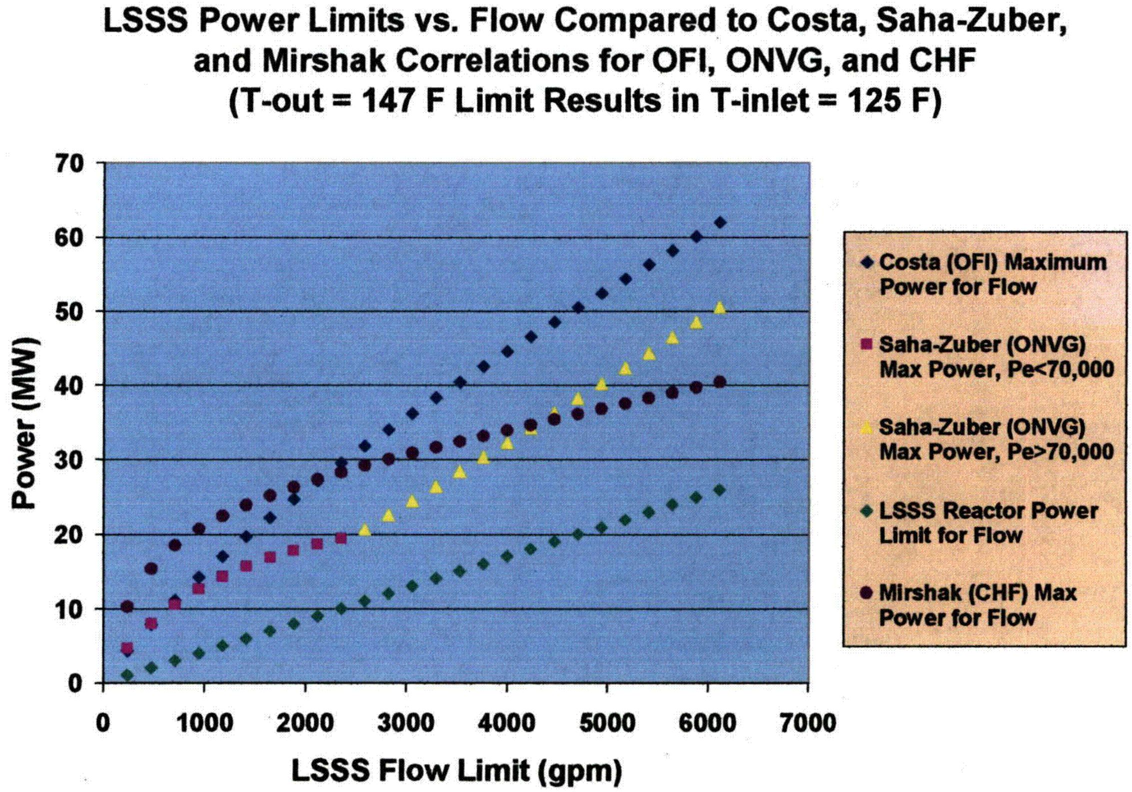

Figure 3. Comparison of the LSSS power limit as a function of flow with the maximum allowed power before either OFI, ONVG, or CHF.

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13.1 Section 13.2.2.2.2, Rapid Removal of Experiments, p. 13.9. Please explain the result of minimum CHFR at BOC in the new analysis for the ramp reactivity insertion versus the EOC minimum CHFR for the previous analysis. Please explain the differences in the initial CHFRs for the two analyses, original analysis for FSAR submitted with the renewal application dated April 9, 2004, and new analysis submitted by letter dated October 2, 2006.

Response

The minimum CHFR is determined by both the maximum reactor power during the excursion and the maximum peaking factor within the core. The former is highest for the EOC because of the lower differential reactivity worth of the shim arms, while the latter is highest for the SU core, when the power is concentrated in the bottom half of the core. In the case of the excursion resulting from removal of experiments, the former dominates for an excursion at 2.6 %Δρ/s, so that the EOC case is limiting. However, when the reactivity insertion is slower, as in the modified experiment removal (1 %Δρ/s), the latter is determining, and the limiting case occurs for the startup core.

When the new analysis for the changed maximum reactivity accident was performed, the problems with RELAP had just been determined. As a result, the input files had to be rewritten to accommodate the work-around using MATLAB calculations of transient power. During this process, the starting power for the transient was inadvertently changed from 20.4 MW to 20 MW, resulting in the 2% increase in minimum CHFR shown in the October 2 submittal. This discrepancy has only a minor effect on the outcome, but the analysis has been repeated using exactly the same parameters as were used for the original SAR, as shown below. The corrected figures and tables will be inserted into the revised SAR.

Table 2. Response to Maximum Reactivity Insertion (SU)

Time (s)	Power (MW)	MCHFR Inner Core	MCHFR Outer Core	Excursion Energy (MJ)
0.00	20.40	4.02	2.67	0.00
0.01	20.41	4.02	2.67	0.00
0.02	20.45	4.02	2.67	0.00
0.04	20.60	4.01	2.66	0.00
0.06	20.82	3.99	2.65	0.01
0.08	21.11	3.96	2.64	0.02
0.10	21.46	3.92	2.61	0.04
0.12	21.85	3.87	2.58	0.06
0.14	22.30	3.82	2.55	0.10
0.16	22.79	3.75	2.50	0.14
0.18	23.32	3.68	2.46	0.19
0.20	23.89	3.60	2.40	0.26
0.22	24.50	3.52	2.35	0.33
0.24	25.16	3.44	2.29	0.42
0.26	25.87	3.35	2.22	0.52
0.28	26.63	3.26	2.16	0.64
0.30	27.43	3.17	2.09	0.77
0.32	28.29	3.07	2.03	0.92
0.34	29.22	2.98	1.96	1.09

0.36	30.20	2.89	1.89	1.27
0.38	31.16	2.79	1.82	1.48
0.40	31.22	2.71	1.76	1.70
0.42	29.37	2.69	1.75	1.90
0.44	25.18	2.77	1.82	2.04
0.46	19.26	3.03	2.02	2.08
0.48	13.17	3.52	2.35	2.08
0.50	8.41	4.32	2.80	2.08
0.52	5.51	5.49	3.44	2.08
0.54	4.01	7.05	4.26	2.08
0.56	3.24	8.99	5.25	2.08
0.58	2.78	11.31	6.40	2.08
0.60	2.48	13.99	7.71	2.08
0.62	2.29	17.00	9.18	2.08
0.64	2.17	20.25	10.79	2.08
0.66	2.13	23.56	12.50	2.08
0.68	2.13	26.66	14.22	2.08
0.70	2.12	29.43	15.91	2.08
0.72	2.12	31.82	17.53	2.08
0.74	2.11	33.82	19.05	2.08
0.76	2.10	35.43	20.43	2.08
0.78	2.09	36.72	21.67	2.08
0.80	2.09	37.74	22.76	2.08

Table 3. Response to Maximum Reactivity Insertion (EOC)

Time (s)	**Power (MW)**	**MCHFR Inner Core**	**MCHFR Outer Core**	**Excursion Energy (MJ)**
0.00	20.40	4.49	3.32	0.00
0.02	20.45	4.48	3.32	0.01
0.04	20.60	4.48	3.31	0.02
0.06	20.82	4.46	3.30	0.03
0.08	21.11	4.42	3.28	0.05
0.10	21.46	4.38	3.25	0.08
0.12	21.85	4.32	3.21	0.11
0.14	22.30	4.26	3.17	0.15

0.16	22.79	4.19	3.12	0.20
0.18	23.32	4.11	3.06	0.26
0.20	23.89	4.02	3.00	0.34
0.22	24.50	3.93	2.94	0.42
0.24	25.16	3.84	2.87	0.52
0.26	25.87	3.75	2.80	0.63
0.28	26.63	3.65	2.73	0.75
0.30	27.43	3.55	2.66	0.89
0.32	28.29	3.44	2.58	1.05
0.34	29.22	3.34	2.51	1.22
0.36	30.20	3.24	2.43	1.42
0.38	31.23	3.13	2.34	1.63
0.40	32.03	3.03	2.26	1.87
0.42	32.24	2.95	2.19	2.11
0.44	31.80	2.91	2.15	2.35
0.46	31.37	2.89	2.14	2.58
0.48	29.87	2.91	2.15	2.80
0.50	27.63	2.99	2.20	2.97
0.52	24.42	3.14	2.31	3.10
0.54	19.40	3.43	2.49	3.14
0.56	14.14	3.94	2.80	3.14
0.58	9.64	4.75	3.29	3.14
0.60	6.38	5.95	3.99	3.14
0.62	4.54	7.58	4.94	3.14
0.64	3.50	9.66	6.11	3.14
0.66	2.91	12.14	7.51	3.14
0.68	2.53	15.01	9.10	3.14
0.70	2.26	18.23	10.87	3.14
0.72	2.07	21.77	12.80	3.14
0.74	1.93	25.58	14.89	3.14
0.76	1.87	29.52	17.09	3.14
0.78	1.86	33.27	19.31	3.14
0.80	1.85	36.64	21.49	3.14

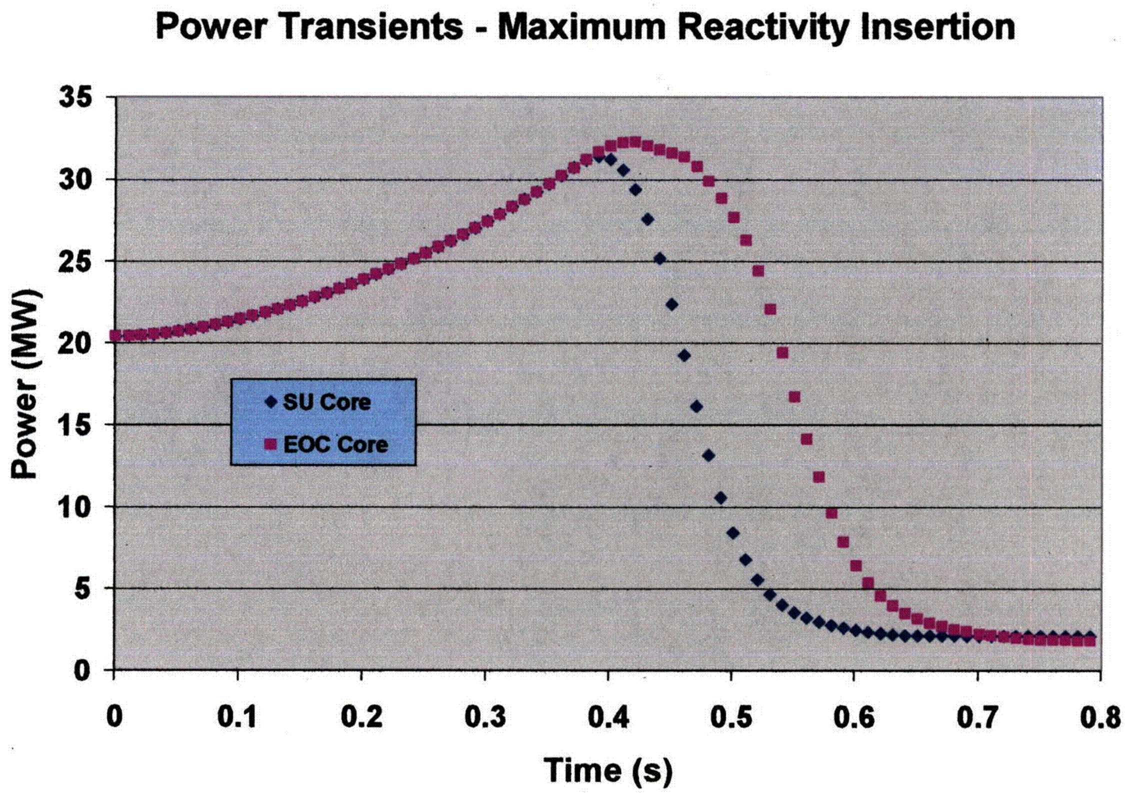

Figure 4. Maximum reactivity insertion excursion.

13.2 Section 13.2.2.2.1, Startup Accident, p. 13.2. The new ramp reactivity insertion analysis states that the RELAP5 MOD3.3 point kinetics model was found to incorrectly predict the power excursion during the transient. Justify the use of this model for the other accident analyses using this model.

Response

Prior to discovery of the error, all RELAP calculations had been compared to a local transient code written in MATLAB. In fact, this was how the error was discovered. In all cases, the transient powers predicted for other accident analyses were verified as correct. When the new analyses were being done, a very small anomaly at the beginning of the power transients was noted, and traced to step size. In accord with normal practice, it was assumed that smaller step size was better, and indeed, the anomaly disappeared as the step size was reduced, but the power excursion was incorrect. After a great deal of detective work, the problem in RELAP was traced to an incorrect program branch designed to avoid round off error, and by decreasing the step size, that branch became the only one taken, thus removing the anomaly when the program changed branches. At that point, analysis was changed to input reactor powers as a function of time, and the original calculations were spot checked. From this work, the correctness of the original analysis of the MCHFR was confirmed. The RELAP code has since been rewritten, and now produces the correct excursion results.

....

<u>13.3 Appendix A, Section 5.4.3 & 5.4.4 Throttling of Coolant Flow to the Outer Plenum p. 5-4.</u> What is the stroke time for valve DWV-1?

<u>Response</u>
The stroke time for valve DWV-1 is 30 seconds.

....

1.0 Introduction

<u>1.3.16</u> Revise the definition to make it specific to the NBSR.

<u>Response</u>
An individual licensed by the U.S. Nuclear Regulatory Commission to manipulate the controls of the NBSR.

....

<u>1.3.18c</u> Define the phrase, "rod drop test mode"

<u>Response</u>
Any combination of control systems and mechanical systems that allows for the movement of only a single shim arm and ensures the reactor remains shutdown, when sufficient fissile material for criticality is present.

....

<u>1.3.19</u> Clarify the portion of the definition that states," when the reactor is if it subcritical...." Also, clarify whether or not the statement "with the reactivity worth of all installed experiments included," means that all experiments are in their most reactive positions.

Response
The definition should read as follows: When the reactor is subcritical by at least one dollar ($1.00) in the Reference Core Condition and with all installed experiments in their most reactive condition.

....

1.3.29 Explain treatment of the regulating rod (e.g., assumed to be in the most reactive position) in the calculation of the shutdown margin.

Response
The definition of shutdown margin will be revised to read: The minimum shutdown reactivity necessary to provide confidence that the reactor can be shutdown by means of the control and safely systems starting from any permissible operating condition, with the most reactive shim arm in the most reactive position and the regulating rod fully withdrawn, and that the reactor will remain shutdown without further operator action.

....

2.0 Safety Limit and Limiting Safety System Setting

2.2 Define "nominal reactor power. "(See TS question 3.1.1)

Response
The NBSR "nominal reactor power" is 20 MW.

....

2.2 The proposed TS allows for operation of the reactor with natural convection cooling at power levels up to 500kW. Provide analysis that demonstrates the fuel temperature will remain below the safety limit during all credible accidents imitated during operation with natural convection cooling. (See SAR question 4.1)

Response
See response to **4.1 Section 4.6**, page 1.

....

3.0 Limiting Conditions for Operations

3.1.1 Define "nominal reactor power" (See TS question 2.2)

Response
The NBSR "nominal reactor power" is 20 Mw.

....

3.1.2(2) Proposed TS 3.1.2(2) is unclear with regard to the assumed positions of the highest-worth shim arm and the regulating rod. Confirm that the shutdown margin will be met with the highest-worth shim arm and the regulating rod in their most reactive positions and revise TS 3.1.2(2) accordingly. Otherwise, provide additional justification for proposed TS 3.1.2(2). (See TS question 1.3.29)

Response
The reactor shall remain subcritical with the highest-worth shim arm and regulating rod fully withdrawn.

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3.1.3 Explain what is meant by "the reactor shall not normally operate," in proposed TS 3.1.3 and revise proposed TS 3.1.3 accordingly. If it is intended that proposed TS allow operation without all grid positions, filled with full length fuel elements or thimbles, provide justification for such operation and analyses that show that such operation would not lead to accidents with consequences greater than those accidents analyzed in the SAR.

Response
The revised TS 3.1.3 shall read "... the reactor shall not operate ..."

....

3.3 Section 5.4.1
Section 5.4.1 of the SAR states, "the chemistry of the primary coolant must be properly controlled to ensure that the components in contact with the primary coolant are not degraded over the life of the plant". The guidance contained in ANSI/ANS-15.1 Section 3.3(9) includes requirements for water chemistry. Provide justification for not including primary coolant chemistry requirements in the proposed TS, or propose primary coolant chemistry

requirements that ensure that the components in contact with the primary coolant will not be degraded over the life of the plant. (See TS question 4.3.1.)

Response

TS 5.2 (2) states, "All materials, including those of the reactor vessel, in contact with the primary coolant shall be compatible with the D_2O environment." This specification goes beyond pD and conductivity limits. Further, NIST has demonstrated that normal methods of adding heavy water to the system will cause the water to be in equilibrium with the CO_2 in the building atmosphere. The pD of that water, and so the entire system inventory, will remain between 5 and 7. The introduction of a large volume of contaminant to the closed primary system is not credible. Therefore, no action is required to maintain water chemistry. TS 5.2 (2) will be deleted from section 5 and inserted into section 3.3.

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3.6 The guidance contained in ANSI/ANS-15.1 Section 3.6 states that "minimum equipment required to be connected to emergency power..." should be listed in the specification. Such equipment is listed in the basis of proposed TS 3.6, but not in the specification. Provide justification for not specifying the equipment required to be connected to emergency power, or specify the equipment required to be connected to emergency power and the minimum operating time for the equipment.

Response

Technical Specification 3.6 will be revised to read as follows:

Specification

The reactor shall not be operated unless at least one (1) of the diesel-powered generators and the station battery are operable, including associated distribution equipment, and the nuclear instrumentation and emergency exhaust fans can be supplied with electrical power from the diesel generator or the battery.

Exception: In order to provide time for prompt remedial action, the Emergency Power may be inoperable for a period of no greater than 15 minutes when the specification is not met or does not exist.

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 and the emergency exhaust fans. These fans and pumps may be powered from AC or DC power supplies. The battery is capable of supplying this emergency load for a minimum of 4 hours. By allowing this amount of time and by requiring operability of at least one diesel and the station battery, adequate emergency power sources shall always be available.

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3.7(1) Provide a basis for the statement, "this value may be increased in an emergency situation up to 500 mrem per calendar year if authorized by the Emergency Director." Include in the basis reference to discussion and/or analysis in the SAR of emergency situations that would warrant the Emergency Director authorizing an increase in the dose to a person outside the site boundary. Provide a discussion of the authorization process, including acceptable reasons for authorization, ALARA considerations, consultation with experts, and any other important considerations.

Response
The statement will be deleted from TS 3.7.2(1). TS 3.7.1 and T.S. 3.7.2 were revised. See **Response** to Question **3.7, Part 1.**

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3.7, Part 1 Proposed TS 3.7 does not include specifications for equipment required to monitor the routine release of effluents. Provide a specification for such equipment and a related specification for surveillance of the equipment in Section 4 of the proposed TSs. Include a basis that references discussion or analysis in the SAR that provides reasonable assurance that effluent releases will be monitored and recorded as required by 10 CFR 20.2103(b)(4), and that doses from airborne effluents will be within the constraint of 10 mrem given in 10 CFR20.1101(d). Otherwise, provide a justification based on calculation and analysis for not requiring effluent monitoring equipment. (See TS question 3.7 below.)

Response
TS 3.7 will be revised as follows (TS 3.7.2.has been combined with 3.7.1):

3.7 Radiation Monitoring Systems and Effluents

3.7.1 Monitoring Systems and Effluent Limits

Applicability: Radiation monitoring systems

Objective: To detect abnormal levels or locations of radioactivity.

Specifications

The reactor shall not be operated unless:

(1) Two of three gaseous effluent monitors are operable for normal air, irradiated air, and stack air.

(2) One fission products 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.

(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) An environmental monitoring program shall be carried out and shall include as a minimum the analysis of samples from surface waters from the surrounding areas, vegetation or soil and air sampling.

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.

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 G-M detectors sensitive to β and γ emissions and the combined air is exhausted through the stack. The stack release is monitored with a G-M detector.

(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.

(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 concentration in the primary coolant is estimated to be 5 Ci/l. This value and reliable leak detection ensures that tritium concentrations in effluents shall be as low as is practicable.

(6) Area vegetation and soil samples are collected for analysis. Grass samples are collected during the growing season, April through September, and soil samples during the non-growing season, October through March. 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 has revealed nothing of significance, thereby confirming that operation of the NBSR has had little or no effect on the environment.

A report published in March 2003 supports the findings of previous studies conducted on the hydrology and geology of the NIST site and vicinity. No significant changes in the hydro-geologic systems or ground water use were identified. This report further verifies the assumptions and techniques developed in 1964.

3.7.2 Effluents

Applicability: Annual releases

Objective: To minimize exposures to the public.

Specification

The reactor shall not be operated unless:

The total exposure from effluents from the reactor facility to a person at the site boundary shall not exceed 100 mrem per calendar year, less any external dose from the facility. The limit shall be established at the point of release or measurement using accepted diffusion factors to the boundary. For halogens and particulates with half-lives longer than 8 days, a reconcentration factor shall be included where appropriate.

Basis

The criteria for determination of concentration limits specified above ensure that 10 CFR 20 (2007) limits are not exceeded at the site boundary. The allowance for dilution from the reactor building stack to the nearest site boundary is 1,000. This value of 1,000 from the diffusion view point is the minimum expected at the nearest site boundary under the least favorable meteorological conditions. This number could be increased by one or two orders of magnitude if normal variations in wind speed and direction were considered. Because these variations are not considered, a one or two order of magnitude margin is inherent in this limit.

In specifying the limits on particulates and long lived (longer than 8 days) halogens, consideration was given to the possibility of biological reconcentration in food crops or dairy products. Using available information (Soldat, J.D., Health Physics 9, p. 1170, 1963), a conservative (both the COMPLY and CAP88 codes indicate that 700 is at least an order of magnitude higher than needed) reconcentration factor of 700 is applied. Thus, the limits for those isotopes are the Effluent Concentration Limits as specified in Appendix B, Table II of 10 CFR 20 (2007) multiplied by the 1,000 dilution factor divided by the 700 reconcentration factor; that is, 1.4 times the Effluent Concentration Limit.

For the purpose of converting concentrations to dose, the values of 10 CFR 20, Appendix B, Table 2 (2007), represent an annual dose of 50 mrem, except for submersion gases where they represent an annual dose of 100 mrem. It should be taken into consideration that the values for submersion gases are based on an infinite hemisphere geometry which is rarely achievable and therefore tends to overestimate the dose.

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3.7, Part 2 ANSI/ANS-15.1 guidance on radiation monitoring and effluents is divided into two sections. Section 3.7.1 provides guidance on the TS requirements for the minimum number of radiation monitors. Section 3.7.2 recommends including effluent release limits for different categories of radionuclides. Contrary to this, effluent release limits are not included in proposed TS 3.7. Further, the scope of proposed TS 3.7.1 is listed as covering only the ARM system, although fission product monitors are mentioned in proposed TS 3.7.1(2). Sections 11.1.4.2 and 11.1.4.3 of the SAR discuss a continuous tritium monitor for building and effluent purposes that is not included in the TS. Also, the basis for proposed TS 3.7.1 contains a discussion of effluent release limits that are not included in the TS and are more appropriately contained in proposed TS 3.7.2 to meet ANSI/ANS-15.1 guidance. Restructure both proposed TS 3.7.1 and 3.7.2 to meet ANSI/ANS-15.1 guidance or justify this departure.

__Response__
TS 3.7 has been completely revised to remedy this concern. See response to 3.7.1 Part 1.

....

3.8.1(1) Basis (2) of proposed TS 3.8.1 states that the maximum allowed reactivity for the pneumatic irradiation system is 0.2% $\Delta\rho$. Provide discussion and/or analysis that demonstrate(s) that the maximum rate of reactivity addition possible with the pneumatic irradiation system is bounded by the analysis of a ramp insertion of 0.5% $\Delta\rho$ in 0.5 seconds.

__Response__
The effect of a step reactivity insertion of 0.2 %$\Delta\rho$ (which is the limiting rate of reactivity insertion for a rabbit) has been analyzed using RELAP, and shown to be bounded by the reactivity insertion accident discussed in RAI 13.1 above. The results are shown below, and the maximum power of 27

MW should be compared to the maximum reactivity insertion accident, where the maximum power is greater than 30 MW.

Table 4. Step Insertion of 0.2% vs. Maximum Reactivity Insertion, 20.4 MW

Accident Scenario	Maximum Power (MW)	Time (msec)	Minimum CHFR	Excursion Energy (MJ)
0.2% Step Insertion – BOC	27.0	295	2.00	1.76
0.2% Step Insertion – EOC	27.1	305	2.55	2.04
Max. Reactivity Insertion - BOC	31.4	391	1.75	2.08
Max. Reactivity Insertion - EOC	32.2	421	2.14	3.14

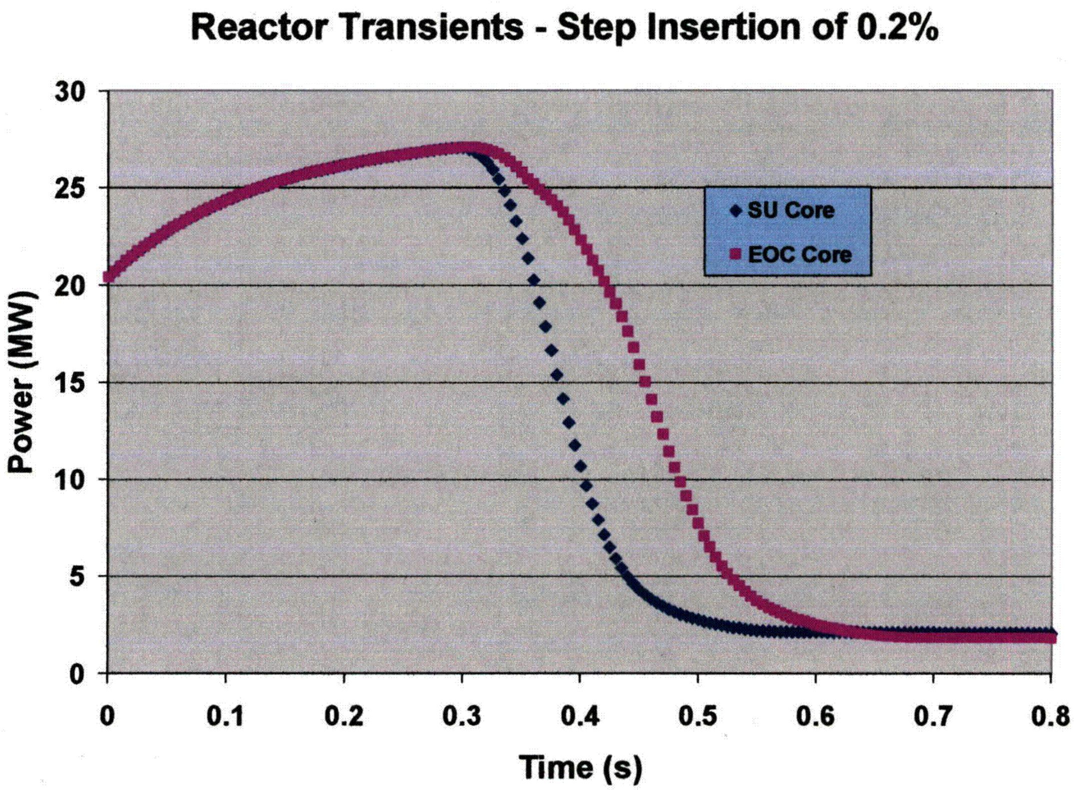

Figure 5. Step reactivity insertion of 0.2 %Δρ (maximum reactivity of any allowed sample in a rabbit), which should be compared to maximum reactivity insertion excursion which is limiting.

3.8.1(3) ANSI/ANS-15.1 Section 3.8.3 guidance includes conditions related to failure and malfunctions of experiments. Specifically, the ANSI guidance states, "experiments shall be designed such that they will not contribute to the failure of other experiments, core components, or principal physical barriers to uncontrolled release of radioactivity." Contrary to this, proposed TS 3.8.1(3) does not include comparable requirements. Revise proposed TS 3.8.1 to include the additional conditions related to failure and malfunctions of experiments or provide justification for not including the additional conditions.

<u>Response</u>
These conditions are covered in TS 3.8.2 Materials. TS 6.5 ensures a review/approval process that is consistent with the intent of ANSI/ANS 15.1, section 3.8.3.

3.9.1(1) Current TS 3.7(1) specifies an optimal keff of 0.9 for fuel elements or fueled experiments being stored or handled. The proposed TS 3.9.1(1) specifies an optimal keff of 0.95 for fuel elements or fueled experiments being stored or handled. ANSI/ANS-15.1, Section 5.4, specifies an optimal keff of 0.9. Provide justification for the reduction in the safety margin for fuel elements or fueled experiments being stored or handled.

<u>Response</u>
TS 3.9.1(1) will be modified to be in accord with ANSI/ANS-15.1, Section 5.4

3.9.1(2) Proposed TS 3.9.1(2) regarding fuel storage indicates that the fuel storage pool is "a stable environment, where water chemistry, temperature, and level are easily monitored..." However, no requirements are included regarding these parameters. Include acceptable ranges of values for these parameters or justify why these parameters should not be included in the proposed TS.

Response

Evaporative losses from the storage pool are small and so the volume of makeup water to the pool is small. Other than the introduction of a large volume of contaminant, there is no reasonable expectation of an undetected change in water chemistry from that of the makeup water supply. The undetected introduction of a large volume of caustic or acid solution is not credible given that access to the pool is restricted to authorized personnel, none of whom would move such solutions through the pool area without knowledge of their supervisors. There is no routine need for such solutions in the pool area or the adjoining room. As to the water chemistry of the makeup water, it is produced by a standard pure water generation machine, which meets industry standards for water purity output, i.e. a resistivity setpoint equivalent to a pH of 7. Once the water comes into equilibrium with the CO_2 in the building atmosphere, the pH will remain between 5 and 7. The temperature of the pool has been demonstrated not to exceed approximately 100 degrees Fahrenheit with a typical fuel storage inventory and with no pool cooling flow. The level of the pool cannot be changed through a failure of the system piping because the pumps do not take a direct suction from the pool; the pool overflows to the pump suction sump. Any loss of water would be only from the sump.

The water chemistry is stable, the temperature will not exceed a known temperature, and the level cannot decrease to less than approximately 16 feet.

....

3.9.2.2(2) Proposed TS 3.9.2.2(2) appears to be applicable at all times, and not only during "all other fuel handling conditions." Provide justification that proposed TS 3.9.2.2(2) is only applicable during "all other fuel handling conditions," or consider incorporating this requirement into proposed TS 3.4.1

Response

TS 3.9.2.2(2) has become 3.4.1(4). TS 3.4.1(4) has become 3.9.2.2.

....

4.0 Surveillance Requirements

4.2.1 ANSI/ANS-15.1 Section 4.2(4) includes scram time surveillance after any work on the rods or drive system. Verify that the testing required by proposed TS 4.1.1(4) includes the surveillances required by proposed TS 4.2.1, and thus is in conformance with the ANSI guidance.

Response
See 1.3.2 and 1.3.12, definitions for channel and operable, respectively. TS 3.2.1(1) specifies operable shim arms for reactor operation. Appropriate tests, which could include TS 4.2.1, would be performed following any maintenance that affects TS 3.2.1(1).

....

4.2.1(1) Provide justification for not determining the withdrawal and insertion speed for the regulating rod.

Response
No credit is taken for regulating rod motion in mitigation of any accident.

....

4.2.2 The guidance contained in ANSI/ANS-15.1 Section 4.2(9) includes interlocks in the operability checks which are not present in the proposed TSs. Include interlocks or justify this departure from ANSI guidance.

Response
There are no design interlocks for the reactor safety system.

....

4.3.1 ANSI/ANS-15.1 guidance specifies quarterly checks of the starting function of emergency shutdown and sump pumps. Contrary to this, proposed TS 4.3.1 specifies annual checks for shutdown cooling pumps. Provide justification for the non-conservative departure from ANSI guidance

Response

There are no emergency shutdown pumps. The specification for shutdown pumps will be removed. See Chapter 13 of the SAR.
The emergency sump pump has been checked for over 25 years without a failure. An annual frequency in 4.3.2 far exceeds the necessary interval.

....

4.3.1 Operability checks for the secondary cooling water activity monitor are included in both proposed TS 4.3.1 and TS 4.7.1. Correct or justify this duplication.

Response
This surveillance requirement shall be moved to TS 4.7.1. TS 3.3.1 and 3.7.1 shall be changed to reflect changes to Section 4 surveillances.

....

4.3.1 Provide justification for not including a surveillance requirement on the concentration of D2 in the helium sweep system (TS 3.3.1(4)), or include such a surveillance requirement.

Response
TS 4.3.1(3) The D_2 concentration in the helium sweep gas shall be verified every five (5) years.

Basis:
The helium sweep gas was sampled approximately every two years between 1984 and 2002, was significantly less than 4%, and showed no appreciable change in D_2 concentration between samples. Based on these results, a period of five (5) years is determined to be sufficient to meet this requirement.

....

4.3.1 ANSI/ANS-15.1 Section 4.3(6) guidance includes a surveillance requirement for coolant system conductivity, pH or both. Contrary to this, proposed TS 4.3.1 does not include any such provisions. Include surveillance requirements for coolant system water quality or justify not including surveillance requirements for coolant system water quality. (See TS question 3.3.)

Response
See 3.3 response.

....

4.6 The ANSI guidance specifies quarterly to semiannual checks for battery voltage and specific gravity. Contrary to this, proposed TS 4.6 specifies annual checks while current TS 4.7 specifies semiannual checks. Change the surveillance period, or justify this non-conservative departure from ANSI guidance and prior practice.

Response
The battery and associated equipment have been checked for over 25 years without a failure. An annual frequency far exceeds the necessary interval.

....

4.7.1 Surveillance requirements for fission product monitors are not included in the proposed TSs. Add surveillance requirements for the fission product monitors or justify this departure from ANSI/ANS-15.1 guidance. (See TS question 3.7.).

Response
The monitor shall be calibrated annually and its operability checked monthly. See Section 4.7.

....

4.9 Proposed TS 3.9.1(2) regarding fuel storage indicates that the fuel storage pool is "a stable environment, where water chemistry, temperature, and level are easily monitored…" However, no surveillance requirements are included regarding these parameters. Include surveillance requirements or justify why these parameters should not be included in the Proposed TS. (See TS question 3.9.1(2).)

Response
See response to 3.9.1(2)

....

5.0 Design Features

5.3(3) Proposed TS 5.3(3) states the fuel plates shall be "uranium-aluminum alloy, either aluminum-uranium oxide or uranium-aluminide, clad with aluminum." Only aluminum uranium oxide is addressed in the SAR (Section 4.2.1.1, 4.2.1.4). Remove mention of the nondescribed fuel types or justify the inclusion of fuel types that have not been described in the SAR.

Response
TS 5.3(3) shall read as follows:
The fuel plates shall be U_3O_8 dispersed in a matrix of aluminum, clad in aluminum alloy.

....

6.0 Administrative Controls

6.1.3 Proposed TS 6.1.3.a.3 specifies the events when an SRO is required to be present. Contrary to ANSI/ANS-15.1 guidance, the proposed TS does not include the condition of "recovery from unplanned or unscheduled shutdown or significant power reduction." Include this condition or justify this departure from ANSI guidance.

Response
ANSI-15.4-2007, section 5.2 reads, "A training program shall be established at each reactor facility based on the knowledge and skill required for reactor operators and senior reactor operators to perform their functions safely and effectively. The amount and depth of training should be commensurate with the level of responsibility and should take into account previous experience and training. A performance based–type training program is preferred." The proposed NCNR organization chart includes SROs and RO as level 4 personnel, which is consistent with Figure 1 of 15.4. The licensed staff is made up of experienced Navy veterans (there are no student operators at the NBSR). An unplanned shutdown or power reduction are events for which they are prepared by their experience and their on-the-job training. Any future operator who did not have such experience would be expected (trained) to respond to these two events in a similar fashion.

....

6.2.2 Proposed TS 6.2.2 specifies that the Safety Evaluation Committee (SEC) shall operate with a written charter that includes provisions for meeting frequency, quorums, use of subcommittees, and treatment of minutes, but provides no details regarding these provisions. ANSI/ANS-15.1 Section 6.2.2 provides related guidance. Include specifics regarding meeting frequency, quorums, use of subcommittees, and treatment of minutes or justify not including such specifics.

Response
The Safety Evaluation Committee Charter is over three pages long and covers all aspects of ANSI/ANS 15.1 requirements. We state that the committee will operate within the charter and the charter is available to the NRC inspector during his visits. The ANSI/ANS 15.1 guidance states that their will be a charter covering the listed topics. We have such a charter.

....

6.2.3 There are two TS 6.2.3 listed; one for the SEC and one for the Safety Audit Committee (SAC). Correct this error and indicate the appropriate numbers for the two proposed TS provisions.

Response
TS 6.2.3 Safety Audit Committee (SAC) shall be changed to TS 6.2.5 and shall read as follows:

6.2.5 <u>Safety Assessment Committee (SAC)</u>

The Safety Assessment Committee (SAC) shall be composed of at least three 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 members shall pass on any report or recommendation of the Committee. The SAC shall meet annually and as required. The Committee shall review or audit the NCNR reactor operations and the performance of the SEC. The SAC shall report in writing to the Director, NIST Center for Neutron Research.

....

6.2.3 The review scope of the SEC does not include operating abnormalities having safety significance or audit reports as suggested by ANSI/ANS-15.1. Add these requirements or justify this departure from ANSI guidance

Response
TS 6.2.3 (2) states "... review the circumstances of all events described in Specification 6.7.2". Specification 6.7.2 covers operating abnormalities.

....

6.2.3 The scope of the audit function of the SAC is stated as "the committee shall audit the NBSR reactor operations and the performance of the SEC." Provide more detail about the specific areas of reactor operation audited and how the SAC audits the performance of the SEC. ANSI/ANS-15.1 Section 6.2.4 contains additional guidance.

Response
TS 6.2.4 will be changed to assign responsibility for audit functions, per the standard, to the SEC. The SAC may also perform the audit per 6.2.5, in addition to their review function.

....

6.4 Proposed TS 6.4 does not have a requirement for procedures for "maintenance that could have an impact on safety...", as recommended in ANSI/ANS-15.1 guidance. Include maintenance that could have an impact on safety in the scope of procedures or justify this departure from ANSI guidance.

Response
6.4 (3) lists procedures that may have an affect on reactor safety. These procedures ensure that the design function of a component or system is unaltered after maintenance by verifying performance before placing the equipment into service.

....

6.5 Proposed TS 6.5.b indicates that minor changes to experiments that do not significantly alter the experiment safety envelope shall be reported to the

SEC at their next meeting. However, proposed TS 6.5.b does not indicate at what level of authority determinations are made regarding what constitutes a "minor change" to a previously approved safety envelope contrary to ANSI/ANS-15.1 guidance. Include the approval level for minor changes to experiment safety envelopes or justify this departure from ANSI guidance.

Response
TS 6.5 (b) shall be changed to 6.5 (2) and read as follows:
Substantive changes to previously approved experiments shall be made only after review by the SEC and approved in writing by the Director, NCNR. Minor changes that do not significantly alter the experiment safety envelope may be made in accordance with the SEC charter.

....

6.8.1 Proposed TS 6.8.1 provides a list of record categories to be retained for five years or the life of a component if less than five years. ANSI/ANS-15.1 guidance includes four categories that are missing from proposed TS 6.8.1: (1) radiation surveys, (2) experiments performed, (3) operating procedure changes, (4) audit reports & meetings. Add these record retention requirements or justify this departure from ANSI guidance.

Response
TS 6.7.1 specifies an annual report. Included in the Annual Report are: Tabulation of Major Items of Plant Maintenance; Tabulation of Major Changes in the Facility and Procedures, Test and Experiments, Carried out without Prior Approval by the NRC Pursuant to 10 CFR 50.59 (2007); Summary of Radioactive Material Released and Results of Environmental Surveys Performed; and Summary of Significant Exposures Received by Facility Personnel and Visitors. Items (1) through (3) are met through the annual report. For item (4), see response to 6.2.3.

Appendix A

License No. TR-5

Technical Specifications
for the
NIST Test Reactor (NBSR)

Table of Contents

1.0 Introduction

These technical specifications apply to the National Institute of Standards and Technology (NIST) Test Reactor (NBSR) license TR-5.

1.1 Scope

The following areas are addressed: Definitions, Safety Limits (SL) and Limiting Safety System Settings (LSSS), Limiting Conditions for Operation (LCO), Surveillance Requirements, Design Features, and Administrative Controls.

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.

1.2.1 Purpose

These specifications are derived from NISTIR 7102 (NBSR 14 Safety Analysis Report). They consist of specific limitations and equipment requirements for the safe operation of the reactor and for dealing with abnormal situations. These specifications represent a comprehensive envelope of safe operation. Only those operational parameters and equipment requirements directly related to verifying and preserving this safety envelope are listed.

1.2.2 Format

The format of these specifications is as described in ANSI/ANS 15.1- 2007.

1.3 Definitions

The following terms are sufficiently important to be separately defined:

1.3.1 ALARA

As Low As is Reasonably Achievable. The practice of making every reasonable effort to maintain exposures to radiation as far below dose limits as is practicable, consistent with the purpose and benefits of licensed activities and the mission of the NBSR.

1.3.2 Channel

The combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter.

1.3.2.1 Channel Calibration

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

1.3.2.2 Channel Check

A qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same variable.

1.3.2.3 Channel Test

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

1.3.3 Confinement

An enclosure of the C wing of the NCNR that is designed to limit the release of effluents between the enclosure and its external environment through controlled or defined pathways.

1.3.4 Core Configuration

The number, type, or arrangement of fuel elements, reflector elements and regulating or control rods occupying the core grid.

1.3.5 Excess Reactivity

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

1.3.6 Emergency Director

The functions of the Emergency Director are defined in the NBSR Emergency Plan.

1.3.7 Experiment

1.3.7.1 In-Reactor Vessel

Any operation, hardware, or target (excluding devices such as detectors and foils), that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the reactor vessel.

1.3.7.2 Beam Tubes

Any sample or hardware placed in a beam tube that has an unobstructed view of the reactor vessel or any materials placed in a beam tube, such as filters and shields for which accident mitigation credit is taken.

1.3.7.3 Movable Experiment
Any experiment in which all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating

1.3.7.4 Secured Experiment
Any experiment, experimental apparatus, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining force must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.

1.3.8 License
The written authorization, by the Nuclear Regulatory Commission, for an individual or organization to carry out the duties and responsibilities associated with a facility requiring licensing.

1.3.9 Measured Value
The value of a parameter as it appears on the output of a channel.

1.3.10 Moderator Dump
An action which drops the water level to approximately one inch (2.5 cm) above the reactor core, thereby ensuring a subcritical state for an emergency shutdown under all reactor operating conditions.

1.3.11 Natural Convection Cooling
That flow of primary water between the reactor core and a heat exchanger with no pumps operating.

1.3.12 Operable
The condition of a system or component when it is capable of performing its intended function, as determined by testing or indication.

1.3.13 Operating
The condition of a component or system when it is performing its intended function.

1.3.14 Protective Action
The initiation of a signal or the operation of equipment within the reactor safety system in response to a variable or condition of the reactor facility having reached a specified limit.

1.3.15 Reactor Operating
The condition of the reactor when it is not secured or shutdown.

1.3.16 Reactor Operator
An individual licensed by the U.S. Nuclear Regulatory Commission to manipulate the controls of the NBSR.

1.3.17 Reactor Safety System
Those systems designated in these technical specifications, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

1.3.18 Reactor Secured
The condition of the reactor when (a), (b), or (c) is true.

(a) (1) The Control Power key switch or the Rod Drive Power key switch is in the off position with the key removed and under the control of a licensed operator; and
(2) The condition of the shim arms is per the specification of Section 3.1.2(3); and
(3) No work is in progress involving core fuel, core structure, installed shim arms, or shim arm drives, unless the shim arm drive shafts are mechanically fixed; and
(4) No experiments in any reactor experiment facility, or in any other way near the reactor, are being moved or serviced if the experiments have, on movement, reactivity worth exceeding the maximum value allowed for a single experiment or $1.00, whichever is smaller.

(b) There is insufficient fissile material in the reactor core or adjacent experiments to attain criticality under optimum available conditions of moderation and reflection.

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

1.3.19 Reactor Shutdown
When the reactor is subcritical by at least one dollar ($1.00) in the Reference Core Condition with all installed experiments in their most reactive condition.

1.3.20 Reactor Shutdown Mechanisms
Mechanisms that can place the reactor in a shutdown condition, and include:

(a) Rundown
(b) Scram
(c) Major Scram
(d) Moderator Dump

1.3.21 Reference Core Condition
The condition of the core when it is at ambient temperature and the reactivity worth of xenon is negligible.

1.3.22 Reactor Rundown
The electrically driven insertion of all shim arms and the regulating rod at their normal operating speed.

1.3.23 Rod, Control
A device, also known as a shim arm, fabricated from neutron absorbing material that is used to establish neutron flux changes and to compensate for routine reactivity losses. The shim arms, when coupled to their drives, provide reactivity control and therefore flux control. When the shim arm becomes decoupled from its drive mechanism it provides a safety function by rapidly introducing negative reactivity into the reactor core.

1.3.24 Rod Drop Mode
Any combination of control systems and mechanical systems that allows for the movement of only a single shim arm and ensures the reactor remains shutdown, when sufficient fissile material for criticality is present.

1.3.25 Rod, Regulating
A low worth control rod used primarily to maintain an intended power level that need not have scram capability. Its position may be varied manually or automatically.

1.3.26 Scram
The spring assisted gravity insertion of all shim arms.

1.3.26.1 Major Scram
A scram accompanied by the immediate activation of the confinement isolation system.

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

1.3.28 Senior Reactor Operator
An individual licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

1.3.29 Shall, Should and May
The word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation.

1.3.30 Shutdown Margin

The minimum shutdown reactivity necessary to provide confidence that the reactor can be shutdown by means of the control and safely systems starting from any permissible operating condition, with the most reactive shim arm in the most reactive position and the regulating rod fully withdrawn, and that the reactor will remain shutdown without further operator action.

1.3.31 Surveillance Activities

Those tests, checks and calibrations done to predict the operability of the equipment described in Section 4.0.

1.3.32 Surveillance Intervals

Maximum intervals are established to provide operational flexibility and not to reduce frequency. Established frequencies shall be maintained over the long term. The surveillance interval is the time between a check, test or calibration, whichever is appropriate to the item being subjected to the surveillance, and is measured from the date of the last surveillance. Surveillance intervals are:

(a) Five Year
Interval not to exceed six years.

(b) Biennial
Interval not to exceed two and half years.

(c) Annual
Interval not to exceed 15 months.

(d) Semi-annual
Interval not to exceed seven and a half months.

(e) Quarterly
Interval not to exceed four months.

(f) Monthly
Interval not to exceed six weeks.

(g) Weekly
Interval not to exceed ten days.

1.3.33 Unscheduled Shutdown

Any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation, not including shutdowns that occur during testing or equipment operability checks.

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 500 kW.

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 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.

3.0 Limiting Conditions for Operations

3.1 Reactor Core Parameters

3.1.1 Reactor Power

Applicability: Reactor power

Objective: To ensure that licensed power is not exceeded and the safety limit is not exceeded through initiation of protective action at a specified power.

Specification

The nominal reactor power shall not exceed 20 MW thermal. The reactor scram set point for a reactor power level safety channel shall not exceed 125% of full power.

Basis

Operational experience and thermal-hydraulic calculations demonstrate that the fuel elements may be safely operated at these power levels. The operating limits developed here are based upon well tested correlations, are conservative, and provide ample margin to ensure that there will be no damage to fuel during normal operation. In addition, the operating conditions provide ample margin for all credible accident scenarios to ensure that there will be no fuel damage.

3.1.2 Reactivity Limitations

Applicability: Core reactivity and shim arm worth

Objective: To ensure that the reactor can be placed in a shutdown condition at all times and that the safety limit shall not be exceeded.

Specifications

(1) The maximum available excess reactivity for reference core conditions shall not exceed 15% $\Delta\rho$ (approximately $20).

(2) The reactor shall not be operated unless the shutdown margin provided by the shim arms is greater than 0.757% $\Delta\rho$ ($1.00) with:

(a) The reactor in any core condition, and
(b) All movable experiments in their most reactive condition.

(3) The reactor shall remain subcritical with the highest-worth shim arm and regulating rod fully withdrawn.

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) and (3) 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 a failed shim arm does not adversely affect core reactivity and cooling flow is maintained.

Specification

The reactor shall not operate unless all grid positions are filled with full length fuel elements or thimbles.

Basis

The NBSR employs shim 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 shim arm from falling from the core lattice. Furthermore, core grid positions shall be filled to prevent coolant flow from bypassing the fuel elements.

3.1.4 Fuel Burnup

Applicability: Fuel

Objective: To remain within allowable limits of burnup

Specification

The fission density shall not exceed 2.6 x 10^{27} fissions/m^3.

Basis

The U_3O_8 – Al dispersion fuels have been in widespread use for over 40 years. Extensive testing of fuel plates has been performed to determine the limits on fission density as a function of fuel loading. Fuel elements in the NBSR are burned for 7 or 8 cycles. Several measurements of swelling in fuel plates show that NBSR fuel, which is moderately loaded at 18%, and for an 8-cycle fuel element with an average fission density of approximately 1.9 x 10^{27} fissions/m^3, is well below the curve that represents the allowable limit of burnup.

3.2 Reactor Control and Safety Systems

3.2.1 Shim Arms

Applicability: Shim arms and shim arm worth

Objective: To ensure proper shim arm reactivity insertion.

Specifications

The reactor shall not be operated unless:

(1) All four shim arms are operable.

(2) The scram time shall not exceed 240 msec for a shim arm insertion of 5 degrees.

(3) The reactivity insertion rate for the four shim arms shall not exceed 5 x 10^{-4} $\Delta\rho$/sec.

Basis

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

(2) and (3) A shim arm withdrawal accident for the NBSR was analyzed using the maximum reactivity insertion rate, corresponding to the maximum 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			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 (up to 500 kW) when a reduction in Limiting Safety System Setting values is permitted per the specifications of Sections 2.2 and 3.3.1.

[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.

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 permit periodic surveillance of the effectiveness of the moderator dump, it is necessary to operate the reactor without restriction on reactor vessel level.

(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 surveillance of the effectiveness of the moderator dump, 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 2.0.

(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 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.

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 tests 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.

Basis

(1) The confinement system is a major engineered safety feature. It is the final physical barrier to mitigate the release of radioactive particles and gasses to the environment following accidents. Confinement is stringently defined to ensure that the confinement building shall perform in accordance with its design basis. Confinement is not required when the reactor is shutdown and experiments are to be inserted or removed.

(2) Changes in the core involving such operations as irradiated fuel handling or shim arm repairs affect the reactivity of the core and could reduce the shutdown margin of the reactor. Confinement shall be required when these changes are made because they affect the status of the core.

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). Under this circumstance, the shutdown margin would be substantial.

(3) Even when the reactor is shutdown, irradiated fuel contains fission product inventories sufficient to allow the specification of Section 3.7.2 to be exceeded should the element fail. This fuel poses a potential hazard in that its cladding could be damaged when it is not contained in a closed system, such as during transit or during sawing of aluminum end pieces. Confinement integrity is not required when irradiated fuel is contained within a closed system, such as the reactor vessel, the transfer lock of the refueling system, or a sealed shipping cask, that serves as a secondary barrier of fission product release.

(4) The specification of Section 3.9.2.2 restricts fuel movement for a specified period. Maintenance that would disable the confinement is prohibited during that period. Building doors could be opened, however, provided that confinement can be rapidly re-established. Confinement integrity is no longer required after the waiting period, because a loss of all water to fuel in a sealed container or system will not cause fuel damage.

3.4.2 Equipment to Achieve Confinement

Applicability: Confinement system

Objective: To ensure that TS 3.4.1 can be met.

Specifications

Confinement shall mean that:

(1) All penetrations of the confinement building are either sealed or capable of being isolated. All piping penetrations within the reactor building are capable of withstanding the confinement test pressure.

(2) All automatic isolation valves in the ventilation, process piping and guide tubes are either operable or can be closed.

(3) All automatic personnel access doors can be closed and sealed.

(4) Except during passage, at least one set of the reactor building vestibule doors for each automatic personnel door is closed or attended, or the automatic door is closed and sealed.

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

Exception to (1) - (5): In order to provide for prompt remedial action, reactor confinement effectiveness may be reduced for a period of no longer than 15 minutes when specifications (1) - (5) are not met or do not exist.

Basis

(1) and (2) The confinement building is designed to be automatically sealed upon indication of high activity. To attempt to operate the reactor with any of these conditions unmet 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 a closure device is placed in its closed or sealed condition, then operability of the automatic closure device is not required.

(3) and (4) 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.

(5) The reactor building truck door is not provided with automatic closure devices. Tests have shown that the confinement building can continue to operate properly, although at reduced efficiency, if the truck door seal were to fail. Confinement cannot be established if the truck door is open.

3.5 Ventilation System

Applicability: Emergency and normal ventilation

Objective: To minimize exposures outside of the confinement building

Specifications

The reactor shall not be operated unless:

(1) The building emergency recirculation system and emergency exhaust systems, including both fans, are operable, and both the absolute and charcoal filter efficiencies are at 99% or greater.

(2) The reactor building ventilation system can filter exhaust air and discharge it above the confinement building roof level.

Exception to (1) and (2): In order to provide time for prompt remedial action, reactor ventilation may be inoperable for a period of no longer than 15 minutes when the specifications are not met or do not exist. Minor maintenance which disables a single fan and can be suspended without affecting the operability of the system may be performed during reactor operation.

Basis

The potential radiation exposure to staff personnel and persons at the site boundary and beyond has been calculated following an accidental release of fission product activity. These calculations are based on the proper operation of the building recirculation system and the emergency exhaust system to maintain the confinement building at a negative pressure and to direct all effluents through filters and up through the reactor building stack. The emergency exhaust system is a redundant system 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 emergency exhaust system is designed to pass reactor building effluents through high-efficiency particulate filters capable of removing particles of 0.3 μm or greater with an efficiency of at least 99% and the charcoal filters are capable of removing greater than 99% of the Iodine from the air. All discharge of the effluents is 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.

3.6 Emergency Power System

Applicability: Emergency electrical power supplies
Objective: To ensure emergency power for vital equipment.

Specification

The reactor shall not be operated unless at least one (1) of the diesel-powered generators and the station battery are operable, including associated distribution equipment, and the nuclear instrumentation and emergency exhaust fans can be supplied with electrical power from the diesel generator or the battery.

Exception: In order to provide time for prompt remedial action, the Emergency Power may be inoperable for a period of no longer than 15 minutes when the specification is not met or does not exist.

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 and the emergency exhaust fans. These fans may be powered from AC or DC power supplies. The battery is capable of supplying this emergency load for a minimum of 4 hours. By allowing this amount of time and by requiring operability of at least one diesel and the station battery, adequate emergency power sources shall always be available.

3.7 Radiation Monitoring Systems and Effluents

3.7.1 Monitoring Systems and Effluent Limits

Applicability: Radiation monitoring systems

Objective: To detect abnormal levels or locations of radioactivity.

Specifications

The reactor shall not be operated unless:

(1) Two of three gaseous effluent monitors are operable for normal air, irradiated air, and stack air.

(2) One fission products 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.

(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) An environmental monitoring program shall be carried out and shall include as a minimum the analysis of samples from surface waters from the surrounding areas, vegetation or soil and air sampling.

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.

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

^3H. 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 G-M detectors sensitive to β and γ emissions and the combined air is exhausted through the stack. The stack release is monitored with a G-M detector.

(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.

(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 concentration in the primary coolant is estimated to be 5 Ci/l. This value and reliable leak detection ensures that tritium concentrations in effluents shall be as low as is practicable.

(6) Area vegetation and soil samples are collected for analysis. Grass samples are collected during the growing season, April through September, and soil samples during the non-growing season, October through March. 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 has revealed nothing of significance, thereby confirming that operation of the NBSR has had little or no effect on the environment.

A report published in March 2003 supports the findings of previous studies conducted on the hydrology and geology of the NIST site and vicinity. No significant changes in the hydro-geologic systems or ground water use were identified. This report further verifies the assumptions and techniques developed in 1964.

3.7.2 Effluents

Applicability: Annual releases

Objective: To minimize exposures to the public.

Specification

The reactor shall not be operated unless:

The total exposure from effluents from the reactor facility to a person at the site boundary shall not exceed 100 mrem per calendar year, less any external dose from the facility. The limit shall be established at the point of release or measurement using accepted diffusion factors to the boundary. For halogens and particulates with half-lives longer than 8 days, a reconcentration factor shall be included where appropriate.

Basis

The criteria for determination of concentration limits specified above ensure that 10 CFR 20 (2007) limits are not exceeded at the site boundary. The allowance for dilution from the reactor building stack to the nearest site boundary is 1,000. This value of 1,000 from the diffusion view point is the minimum expected at the nearest site boundary under the least favorable meteorological conditions. This number could be increased by one or two orders of magnitude if normal variations in wind speed and direction were considered. Because these variations are not considered, a one or two order of magnitude margin is inherent in this limit.

In specifying the limits on particulates and long lived (longer than 8 days) halogens, consideration was given to the possibility of biological reconcentration in food crops or dairy products. Using available information (Soldat, J.D., Health Physics 9, p. 1170, 1963), a conservative (both the COMPLY and CAP88 codes indicate that 700 is at least an order of magnitude higher than needed) reconcentration factor of 700 is applied. Thus, the limits for those isotopes are the Effluent Concentration Limits as specified in Appendix B, Table II of 10 CFR 20 (2007) multiplied by the 1,000 dilution factor divided by the 700 reconcentration factor; that is, 1.4 times the Effluent Concentration Limit.

For the purpose of converting concentrations to dose, the values of 10 CFR 20, Appendix B, Table 2 (2007), represent an annual dose of 50 mrem, except for submersion gases where they represent an annual dose of

100 mrem. It should be taken into consideration that the values for submersion gases are based on an infinite hemisphere geometry which is rarely achievable and therefore tends to overestimate the dose.

3.8 Experiments

3.8.1 Reactivity Limits

Applicability: Reactivity of experiments

Objective: To limit reactivity excursions.

Specifications

The reactor shall not be operated unless:

(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.

Basis

(1) 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.

(2) 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 one experiment with the maximum allowable reactivity of 0.5% $\Delta\rho$ for movable

experiments was removed 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. The 0.2% $\Delta\rho$ for the combined pneumatic irradiation systems has been shown to be bounded by the ramp insertion of 0.5% $\Delta\rho$ and is well below this referenced accident as well as being within the $\Delta\rho$ capability of the regulating rod.

(3) In addition to all reactor experiments being designed not to fail from internal gas buildup or overheating, they shall be designed so that their failure does not affect either the reactor or other experiments. They shall also be designed to withstand, without failure, the same transients that the reactor itself can withstand without failure.

3.8.2 Materials

Applicability: All materials used in experiments

Objective: To prevent damage to the reactor or a significant release of radioactivity.

Specifications

(1) Explosive or metastable materials capable of significant energy releases shall be irradiated in double walled containers that have been satisfactorily tested.

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

(3) All experiments performed at the NBSR shall be reviewed and authorized in accordance with the specifications of Section 6.5.

Basis

(1) 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 reactor fuel, or to the failure of other experiments. Also, reactor experiments shall be able to withstand the same transients that the reactor itself can withstand, such as loss of reactor cooling flows and startup accident.

The detonation of explosive or metastable materials within the reactor is not an intended part of the experimental procedure for the NBSR, but the possibility of a rapid energy release shall be considered when these materials are present. Full testing of the container design shall be done.

(2) Experiments containing materials corrosive to reactor components or highly reactive with reactor or experimental coolants shall have an added margin of safety 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.

(3) An independent technical review of experiments ensures the experiment will not reduce the reactor safety margin.

3.9 Facility Specific

3.9.1 Fuel Storage

Applicability: Fuel element storage

Objective: To prevent inadvertent criticality and maintain fuel element cladding integrity.

Specifications

(1) All fuel elements or fueled experiments shall be stored and handled in geometry such that the calculated k_{eff} shall not exceed 0.90 under optimum conditions of water moderation and reflection.

(2) The water chemistry, level, and temperature in the spent fuel storage pool shall be maintained so as to ensure the integrity of the fuel elements.

Basis

(1) To ensure that no inadvertent criticality of stored fuel elements or fueled experiments occurs, they shall be maintained in a geometry that ensures an adequate margin below criticality exists. This margin is established as a k_{eff} of no greater than 0.90 for the storage and handling of fuel or fueled experiments.

(2) The cooling of spent fuel elements in storage at the NBSR depends upon the decay heat of the elements, the volume of water in a storage pool, and any additional cooling, such as the use of pumps and heat exchangers. A storage pool is a stable environment, where water chemistry, temperature and level are easily monitored and the fuel is adequately shielded.

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 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.

3.9.2.2 All Other Conditions

Applicability: Refueling system

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

Specification

A fuel element shall not be removed from water in the reactor vessel unless the reactor has been shutdown for a period equal to or longer than one hour for each megawatt of operating power level.

Basis

To ensure that a fuel element does not melt and release radioactive material, a time limit is specified before a fuel element may be removed from the vessel following reactor shutdown. Measurements carried out during reactor startup showed that for the hottest element placed dry in the transfer chute, 8 hours after shutdown from 10 MW, the maximum temperature was only 550°F without auxiliary cooling. Extrapolation of these measurements shows that 20 hours after shutdown from 20 MW, the maximum temperature for the hottest element would be less than 800°F without auxiliary coolant. For all other power levels below 20 MW the specified waiting time would result in even lower temperatures. This provides a substantial margin of safety from the safety limit.

4.0 Surveillance Requirements

Introduction

The Surveillance frequencies denoted herein are based on continuing operation of the reactor. Surveillance activities scheduled to occur during an operating cycle which can not be performed with the reactor operating may be deferred to the end of that current reactor operating cycle. If the reactor is not operated for a reasonable time, a reactor system or measuring channel surveillance requirement may be waived during the associated time period. Prior to reactor system or measuring channel operation, the surveillance shall be performed for each reactor system or measuring channel for which surveillance was waived. A reactor system or measuring channel shall not be considered operable until it is successfully tested. Surveillance intervals shall not exceed those defined in these Technical Specifications. Discovery of noncompliance with any of the surveillance specifications below shall limit reactor operations to that required to perform the surveillance.

4.1 Reactor Core Parameters

4.1.1 Reactor Power

Applicability: Reactor Safety System channels

Objective: To ensure operability of the safety system channels.

Specifications

(1) The reactor safety system channels shall be channel tested before each reactor startup, following a reactor shutdown that exceeds 24 hours, or quarterly.

(2) The reactor safety system channels shall be channel calibrated annually.

(3) A channel check of power range indication, with flow multiplied by ΔT, shall be performed weekly when the reactor is operating above 5 MW.

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

Basis

The channel tests, calibrations and flow ΔT comparison will ensure that the indicated reactor power level is correct. The power level channel calibration is performed by comparison of nuclear channels with the thermal power measurement channel (flow times ΔT). Because of the small ΔT (about 15°F at 20 MW), these calibrations will not be performed below 5 MW.

4.1.2 Reactivity Limitations

Applicability: Core reactivity and shim arm worth.

Objective: To ensure that the reactor can be placed in a shutdown condition at all times and that the safety limit shall not be exceeded.

Specifications

(1) The excess reactivity (reference core conditions) shall be verified annually or following any significant changes in the core or shim arm configuration.

(2) The total reactivity worth of each shim arm and the regulating rod, and the shutdown margin shall be verified annually as described in these Technical Specifications, or following any significant change in the core or shim arm configuration.

Basis

(1) Determining the core excess reactivity annually will ensure that the critical shim arm positions do not change unexpectedly.

(2) Measurements of reactivity worth of the shim arms and regulating rod over many years of operation have shown rod worths 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 that adequate reactivity margins are maintained.

4.2 Reactor Control and Safety Systems

4.2.1 Shim Arms

Applicability: Shim arm motion

Objective: To ensure proper shim arm reactivity insertion.

Specifications

(1) The withdrawal and insertion speeds of each shim arm shall be verified semiannually.

(2) Scram times of each shim arm shall be measured semi-annually.

Basis

The shim arm drives are constant speed mechanical devices. A reactor scram is aided by a spring that opposes drive motion during shim 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.

4.2.2 Reactor Safety System Channels

Applicability: Required instrument channels

Objective: To ensure reliability of protective action for nuclear and process variables.

Specifications

The Scram and Confinement Channels shall have the surveillance requirements shown in Table 4.2.2.

Table 4.2.2
Surveillance Requirements for the Scram and Confinement Channels

Channel	Action Required	Surveillance Required
(1) High Flux level	Scram	X, A
(2) Short period below 5% rated power	Scram	X, A
(3) Low reactor vessel D_2O level	Scram	X, A
(4) Low flow reactor outlet	Scram	X, A
(5) Low flow reactor inner or outer plenum	Scram	X, A
(6) Manual (outside of the Control Room)	Scram	X, A
(7) Manual	Scram	X, A
(8) Normal Air Exhaust Activity High	Major Scram	X, A
(9) Irradiated Air Activity High	Major Scram	X, A
(10) Stack Air Activity High	Major Scram	X, A
(11) Reactor Coolant Outlet Temperature	Rundown	X, A

X - Channel test before startup after a shutdown of longer than 24 hours, or quarterly.
A - Annual Channel Calibration.

Basis

To ensure that instrument failures do not go undetected, frequent surveillance of the listed channels is required and operating experience has shown these frequencies to be adequate to ensure channel operability.

4.3 Coolant Systems

4.3.1 Primary and Secondary

Applicability: Primary fluid systems

Objective: To prevent degradation of primary system materials.

Specifications

(1) The primary cooling system relief valve shall be tested annually.

(2) Major additions, modifications, or repairs of the primary cooling system or its connected auxiliaries shall be tested before the affected portion of the system is placed into service.

(3) The D_2 concentration in the helium sweep gas shall be verified every five (5) years.

Basis

(1) The frequency for testing the pressure at which the relief valve opens is consistent with industry practices on this type of valve for clean water service conditions.

(2) 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 other methods.

(3) Recombination of deuterium and oxygen is accomplished primarily by the reactor. Operational experience and data suggests that the specified frequency is appropriate for verifying D_2 levels.

4.3.2 Emergency Core Cooling System

Applicability: Emergency Core Cooling System

Objective: To ensure an emergency supply of coolant.

Specifications

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

(2) The operability of the emergency sump pump, using either heavy or light water, shall be tested annually.

(3) The light water injection valves shall be exercised semi-annually.

Basis

The equipment in this system is not used in the course of normal operation, so 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 corrosion buildup or other slow acting effects during extended periods of standby operation. Control and injection valves specified are those leading to or from the D_2O emergency cooling tank.

4.3.3 Moderator Dump System

Applicability: Moderator dump valve

Objective: To provide a backup shutdown mechanism.

Specification

The Moderator Dump valve shall be cycled annually.

Basis

The moderator dump valve is of proven dependable design. Operating the dump valve annually is and has been a reliable predictor of performance.

4.4 Confinement System

Applicability: Confinement building and components

Objective: To ensure the continued integrity and reliability of the confinement building.

Specifications

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

(2) An integrated leakage test of the confinement building shall be performed annually at a gauge pressure of at least 6.0 inches of water and a vacuum of at

least 2.0 inches of water, with a maximum allowable leak rate of 24 cfm/inch of water.

(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

(1) The confinement closure system is initiated either by a signal from the confinement building gaseous effluent radiation detectors or manually by the major scram switch and each of these signal sources is used to initiate the test. In addition, each radiation detector is tested for proper response to ionizing radiation.

(2) A preoperational test program was conducted to measure the representative leakage characteristics at values of a gauge pressure of +7.5 inches of water and –2.5 inches of water. The specified test pressures and vacuums are acceptable because past tests have shown leakage rates to be linear with applied pressures and vacuums.

(3) 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.

4.5 Ventilation System

Applicability: Normal and Emergency ventilation system

Objective: To ensure the operability of the ventilation system.

Specifications

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

(2) An operability test of the controls in the Emergency Control Station and an inspection to determine that all instruments in the Emergency Control Station are indicating normally shall be made monthly.

(3) The efficiency of the absolute filters in both normal and emergency exhaust systems shall be verified biennially. It shall be verified that the absolute filters remove 99% of particles with diameters of 0.3 μm and greater.

(4) It shall be verified biennially that the charcoal filter banks in the emergency exhaust and recirculation systems have a removal efficiency of 99% for Iodine.

Basis

(1) 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. 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 quarterly will ensure that each component will 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.

(2) The Emergency Control Station instrumentation must be operable to monitor the reactor's condition in the event the Control Room becomes uninhabitable. Therefore, monthly checks of the instrumentation have been shown to be adequate to ensure operability.

(3) The biennial verification of the absolute filter efficiency has been shown to be appropriate for filters subject to continuous air flow. Because the absolute filters in the emergency exhaust system will be idle except during brief periods of fan operation, 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 frequency should be adequate to detecting filter deterioration.

(4) Biennial verification of filter banks, which are subjected to flow only during brief periods of fan operation ensures that the filters will perform as analyzed.

4.6 Emergency Power System

Applicability: Emergency electrical power supply equipment

Objective: To ensure emergency power for vital equipment after the reactor is shutdown.

Specifications

(1) Each diesel generator shall be tested for automatic starting and operation quarterly.

(2) Should one of the diesel generators become inoperative, the operable generator shall be started monthly.

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

(4) The voltage and specific gravity of each cell of the station battery shall be tested annually. A discharge test of the entire battery shall be performed once every 5 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) Specific gravity and voltage checks of individual cells are the accepted method of ensuring that all cells 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 battery was discharge tested to measure its capacity. Experience has shown that repeating this test at the specified interval is adequate to detect deterioration of the cells.

4.7 Radiation Monitoring System and Effluents

4.7.1 Monitoring System

Applicability: Radiation monitoring equipment

Objective: To 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.

(3) The secondary coolant activity monitor shall be channel tested monthly and channel calibrated annually. Analysis of the secondary water for tritium shall be conducted monthly. Should the secondary cooling water activity monitor be inoperable, analysis for tritium shall be performed daily.

(4) The Area Radiation Monitors shall be channel tested monthly and channel calibrated annually.

(5) For primary tritium concentrations of less than or equal to 4 Ci/l, the primary water shall be sampled annually. For tritium concentrations of greater than 4 Ci/l, the primary water shall be sampled quarterly.

Basis

(1) A channel test ensures the monitoring systems will respond correctly to an input signal. An annual channel calibration ensures the detection and response capability of the channels.

(2) A channel test monthly is considered reasonable for a device of this type. A channel calibration annually is considered adequate to ensure that a significant deterioration in accuracy from its normal setting does not occur.

(3) The secondary cooling water activity monitor usually gives the first indication of a primary-to-secondary leak. This monitor employs a simple radiation detector, the operability of which has been shown to be very good. Therefore, a monthly channel test is considered reasonable. An annual channel calibration frequency is considered adequate to ensure that a significant deterioration in accuracy from its normal settings does not occur. Assuming operation of the secondary cooling water activity monitor and no detectable loss of primary coolant, a monthly sampling for tritium should be adequate to detect small tritium leaks. If the secondary cooling water activity 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 a decreasing level in the D_2O storage tank.

(4) The area radiation monitors (ARM) may give the first indication of a radioactive release resulting from an experiment or reactor malfunction. A monitor employs a simple radiation detector, the operability of which has been shown to be very good over many years. Therefore, a monthly channel test is considered reasonable. These monitors are primarily used to detect an increase in activity over that which has previously existed, so 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 ensure that a significant deterioration in accuracy does not occur.

(5) The primary tritium concentration can be carefully monitored by annual analysis of the primary water. All new water is tested prior to addition to the system. Operational experience and well established neutron activation principles provide a good basis for predicting tritium buildup in the primary. Increasing the sampling frequency after concentrations exceed 4 Ci/l will ensure that the tritium concentration limit is not exceeded.

4.7.2 Effluents

Applicability: Environmental monitoring sampling program

Objective: To minimize radiation exposures outside of the confinement building.

Specifications

(1) Water, soil and vegetation samples shall be collected quarterly.

(2) Thermoluminescent dosimeters shall be collected quarterly.

(3) Air sampling shall be done quarterly.

Basis

(1) Collecting and analyzing the water, soil and vegetation samples on a quarterly basis will provide information that environmental limits are not being exceeded.

(2) Collecting and analyzing the thermoluminescent dosimeters on a quarterly basis will provide information that radiation limits are not being exceeded.

(3) Sampling the air on a quarterly basis will provide information that release limits are not being exceeded.

4.8 Experiments

Applicability: Irradiation Experiments

Objective: To ensure that experiments conform to the limits of the specifications of Section 3.8.

Specification

The reactivity worth of any experiment installed in a pneumatic transfer tube, or in any other NBSR irradiation facility inside the thermal shield shall be estimated before reactor operation with said experiment.

Basis

Estimation of the reactivity worth based either on calculation or on previous or similar measurements ensures that the experiment is within authorized reactivity limits.

5.0 Design Features

5.1 Site Description

Specifications

(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 of 400 meters, as measured from the reactor stack.

Basis

The location and government ownership of the NBSR site ensures auxiliary services including fire and security are available. The exclusion radius of 400 meters is the distance on which all unrestricted doses are calculated. 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.

5.2 Reactor Coolant System

Specifications

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

(2) The reactor vessel shall be designed in accordance with Section VIII of the American Society of Mechanical Engineers (ASME) Code for Unfired Pressure Vessels. The vessel shall be designed for 50 psig and 250°F. The heat exchangers shall be designed for 100 psig and a temperature of 150°F. The connecting piping shall be designed for 125 psig and a temperature of 150°F.

Basis

(1) The reactor coolant system has been described and analyzed as a single cooling loop system containing heat exchangers, pumps and valves.

(2) 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.

5.3 Reactor Core and Fuel

Specifications

(1) The 20 MW reactor core consists of 30, 3.0 x 3.3 inch (7.6 x 8.4 cm) MTR curved plate-type fuel elements. The NBSR MTR-type fuel element shall be such that the central 7 inches of the fuel element contains no fuel. The middle 6 inches of the aluminum in the unfueled region of each plate shall have been 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 U_3O_8 dispersed in a matrix of aluminum, clad in aluminum alloy

Basis

(1) The neutronic and thermal hydraulic analysis was based on the use of 30 NBSR MTR-type thirty-four (34) plate fuel elements. The NBSR fuel element has a 7 inch centrally located unfueled area, in the open lattice array. The middle 6 inches of aluminum in the unfueled region has been removed. The analysis requires that the fuel be loaded in a specific pattern. Significant changes in core loading patterns would require a recalculation of the power distribution to ensure that the CHFR would be within acceptable limits.

(2) and (3) The aluminum clad dispersion fuels used in the MTR fuel elements have a 50 year record of reliability at many research reactors.

6.0 Administrative Controls

6.1 Organization

The Director, NIST Center for Neutron Research shall be the licensee for the NBSR. The NBSR shall be under the direct control of the Chief, Reactor Operations and Engineering. The Chief, Reactor Operations and Engineering shall be accountable to the Director, NCNR for the safe operation and maintenance of the NBSR.

6.1.1 Structure

The management for operation of the NBSR shall consist of the organizational structure as shown in Figure 6.1.

6.1.2 Responsibility

Responsibility for the safe operation of the NBSR shall be with the chain of command established in Figure 6.1. Individuals at the various management levels shall be responsible for the policies and operation of the NBSR, for safeguarding the public and facility personnel from undue radiation exposures, and for adhering to all requirements of the operating license and technical specifications.

6.1.3 Staffing

(1) The minimum staffing when the reactor is not secured shall be:

- (a) A Reactor Operator in the Control Room.
- (b) A Reactor Supervisor present within the reactor exclusion area.
- (c) An SRO present in the facility whenever a reactor startup is performed, fuel is being moved within the reactor vessel, or experiments are being placed in the reactor vessel.

(2) A list of reactor facility personnel by name and telephone number shall be available to the reactor operator in the Control Room. This list shall be updated annually. The list shall include:

- (a) Management personnel.
- (b) Health Physics personnel.
- (c) Reactor Operations personnel.

6.1.4 Selection and Training of Personnel

The selection, training and requalification of operations personnel shall meet or exceed the requirements of the American National Standard for Selection and Training of Personnel for Research Reactors (ANSI/ANS 15.4-2007). Qualification and requalification of licensed reactor operators shall be performed in accordance with a Nuclear Regulatory Commission (NRC) approved program.

6.1.4.1 Selection of Personnel

Minimum educational and experience requirements for those individuals who have line responsibility and/or authority for the safe operation of the facility are as follows:

(1) Chief, Reactor Operations and Engineering

The Chief, Reactor Operations and Engineering shall have an advanced college degree in engineering or a science related field, or equivalent experience and training. Equivalent experience for this position requires five years experience in a responsible position in reactor operations or reactor engineering, including one year experience in senior reactor facility management or supervision.

(2) Chief, Reactor Operations

The Chief, Reactor Operations shall have a college degree in engineering or a science related fields or a combined seven years of college level education and nuclear reactor experience. Three years of reactor operations experience is required. The individual shall demonstrate the capability to be an SRO at the NBSR.

(3) Reactor Supervisor

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

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

(c) Shall have been a licensed as a Senior Reactor Operator at the NBSR.

(4) Senior Reactor Operator

A Senior Reactor Operator shall have a high school diploma or equivalent and one year experience in reactor operations. The individual shall be licensed as a Senior Reactor Operator.

(5) Reactor Operator

A Reactor Operator shall have a high school diploma or equivalent and six months of technical training. The individual shall be licensed as a Reactor Operator.

(6) Auxiliary Operator

An Auxiliary Operator shall have a high school diploma or equivalent.

6.1.4.2 Training of Personnel

(1) A training program shall be established to maintain the overall proficiency of the Reactor Operations organization. This program shall include components for both initial licensing and requalification, consistent with ANSI/ANS 15.4-2007.

(2) The training program shall be under the direction of the Chief, Reactor Operations and/or the Chief, Reactor Operations and Engineering.

(3) Records of individual reactor operations staff members' qualifications, experience, training, and requalification shall be maintained as described the specification of Section 6.8.2.

6.2 Review and Audit

The NCNR Safety Evaluation Committee (SEC) is established to provide an independent review of NCNR reactor operations to ensure the facility is operated and maintained in such a manner that the general public, facility personnel and property shall not be exposed to undue risk.

The NCNR Safety Assessment Committee (SAC) is established to provide an independent review or audit of NCNR reactor operations. This audit is to ensure that safety reviews and reactor operations are being performed in accordance with regulatory requirements and public safety is being maintained.

6.2.1 Composition and Qualifications

The Director, NCNR, upon recommendation of the Chief, Reactor Operations and Engineering, shall appoint all members and alternates to the SEC. The SEC shall be composed of no less than four members and membership terms are indefinite and at the discretion of the Director. Members and alternates shall be selected on their ability to provide independent judgment and to collectively provide a broad spectrum of expertise in reactor technology and operation. At least two members shall be from the NCNR and one from Health Physics. Unless otherwise designated by the Director, the SEC shall include the following ex officio members: the Chief, Reactor Operations; Chief, Reactor Engineering; and the Senior Supervisory Health Physicist.

6.2.2 Safety Evaluation Committee Charter and Rules

The SEC shall conduct its review functions in accordance with a written charter and the charter shall be consistent with ANSI/ANS 15.1-2007. This charter shall include provisions for:

(1) Meeting frequency.
(2) Voting rules.
(3) Quorums.
(4) Method of submission and content of presentation to the committee.
(5) Use of subcommittees.
(6) Review, approval and dissemination of minutes.

6.2.3 SEC Review Function

The responsibilities of the SEC, or a designated subcommittee thereof, shall include but are not limited to the following:

(1) Review proposed tests or experiments significantly different from any previously reviewed or which involve any questions pursuant to 10 CFR 50.59 and determine whether proposed changes or reactor tests or experiments have been adequately evaluated, documented, approved and recommendations sent to the NCNR director for action.

(2) Review the circumstances of all events described in this section and the measures taken to preclude a recurrence and provide recommendations to the NCNR director for action.

(3) 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 questions pursuant to 10 CFR 50.59 and provide recommendations to the NCNR director for action. Review SAC reports.

(4) The SEC shall on a biennial basis review its charter and recommend to the NCNR director any changes necessary to ensure the continued effectiveness of the charter.

6.2.4 SEC Audit Function

The responsibility of the SEC, or a designated subcommittee thereof, shall include but not be limited to the following audits:

(1) Facility operations at a frequency of once per calendar year, not to exceed fifteen (15) months.

(2) Results of actions taken to correct deficiencies that affect reactor safety at a frequency of once per calendar year, not to exceed fifteen (15) months.

(3) Requalification program at a frequency of once every other calendar year, not to exceed thirty (30) months.

(4) NBSR Emergency Plan at a frequency of once every other calendar year, not to exceed thirty (30) months.

6.2.5 Safety Assessment Committee (SAC)

The Safety Assessment Committee (SAC) shall be composed of at least three 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 members shall pass on any report or recommendation of the Committee. The SAC shall meet annually and as required. The Committee shall review or audit the NCNR reactor operations and the performance of the SEC. The SAC shall report in writing to the Director, NIST Center for Neutron Research.

6.3 Radiation Safety

The NIST Reactor Health Physics Group shall be responsible to support the licensee in the implementation of the radiation protection and ALARA program at the reactor using the guidelines of the American National Standard for Radiation Protection at Research Reactor Facilities, ANSI/ANS 15.11-2004. The NIST Reactor Health Physics Group leader shall report to the Director, NIST Center for Neutron Research for radiological matters concerning the NBSR.

6.4 Procedures

Written procedures shall be prepared, reviewed and approved prior to initiating any of the activities listed in this section. The safety significant changes (determined by the Chief, Reactor Operations and Engineering or the Chief, Reactor Operations) to operating procedures shall be reviewed by the SEC and approved by the Chief, Reactor Operations and Engineering or the Chief, Reactor Operations. Such reviews and approvals shall be documented in a timely manner. Activities requiring written procedures are:

(1) Startup, operation, and shutdown of the reactor.

(2) Fuel loading, unloading, and fuel movement within the reactor vessel.

(3) Surveillance checks, calibrations, and inspections of equipment required by the technical specifications that may have an effect on reactor safety.

(4) Personnel radiation protection, consistent with applicable regulations or guidelines. The procedures shall include management commitment and programs to maintain exposures and releases as low as is reasonably achievable in accordance with the guidelines of ANSI/ANS 15.11-2004.

(5) Conduct of irradiations and experiments that could affect reactor safety or core reactivity.

(6) Implementation of required plans such as emergency or security plans.

(7) Use receipt, and transfer of byproduct material, if appropriate.

Substantive changes to the procedures listed above shall be made effective only after documented review by the SEC and approval by the Chief, Reactor Operations and Engineering or the Chief, Reactor Operations. Minor modifications or temporary deviations to the original procedures which do not effect reactor safety or change their original intent may be made by the Reactor Supervisor in order to deal with special or unusual circumstances or conditions. Such changes shall be documented and reported within 24 hours or the next working day to the Chief, Reactor Operations and Engineering or the Chief, Reactor Operations.

6.5 Experiment Review and Approval

Experiments shall be carried out in accordance with established and approved procedures. The following provisions shall be implemented:

(1) All new experiments or class of experiments shall be reviewed by the SEC and approved in writing by the Director, NCNR.

(2) Substantive changes to previously approved experiments shall be made only after review by the SEC and approved in writing by the Director, NCNR. Minor changes that do not significantly alter the experiment safety envelope may be made in accordance with the SEC charter.

6.6 Required Actions

6.6.1 Actions to Be Taken in the Event the Safety Limit is Exceeded

(1) The reactor shall be shutdown and reactor operations shall not be resumed until authorized by the NRC.

(2) An immediate notification of the occurrence shall be made to the Chief, Reactor Operations and Engineering and the Chief, Reactor Operations. The Chief, Reactor Operations and Engineering shall inform the NCNR director.

(3) Reports shall be made to the NRC in accordance with the specifications of Section 6.7.2. A written report shall include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. The report shall be prepared by the Chief, Reactor Operations and Engineering and submitted to the SEC for review. The SEC shall review the report and submit it to the Director, NIST Center for Neutron Research director for approval. The Director shall then submit the report to the NRC.

6.6.2 Actions to Be Taken in the Event of an Occurrence of the Type Identified in Section 6.7.2 other than a Safety Limit Violation

(1) The reactor shall be secured and the Chief, Reactor Operations and Engineering and the Chief, Reactor Operations notified.

(2) Operations shall not resume unless authorized by the Chief, Reactor Operations and Engineering.

(3) The SEC shall review the occurrence at their next scheduled meeting.

(4) Where appropriate and in addition to the initial notification, a report shall be submitted to the NRC in accordance with the specifications of Section 6.7.2.

6.7 Reports

6.7.1 Annual Operating Report

A report shall be submitted annually to the NRC and include:

(1) A brief summary of operating experience including the energy produced by the reactor and the hours the reactor was critical.

(2) The number of unscheduled shutdowns, including reasons therefore.

(3) A tabulation of major preventative and corrective maintenance operations having safety significance.

(4) A brief description, including a summary of the safety evaluations, of changes in the facility or in procedures and of test and experiments carried out pursuant to 10 CFR 50.59 (2007).

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

(6) A summary of environmental surveys performed outside the facility.

(7) A summary of significant exposures received by facility personnel and visitors.

6.7.2 Special Reports

In addition to the requirements of applicable regulations, and in no way substituting therefore, reports shall be made by the Director, NCNR or the Chief, Reactor Operations and Engineering, to the NRC as follows:

(1) There shall be a report within 24 hours by telephone, facsimile, or other NRC approved method, to the NRC Operations Center and confirmed in writing by facsimile or similar conveyance, to be followed by a written report within 14 days that describes the circumstances associated with any of the following:

(a) Accidental release of radioactivity above applicable limits in unrestricted areas, whether or not the release resulted in property damage, personal injury, or exposure.

(b) Violation of the safety limit.

(c) Operation with a safety system setting for required systems less conservative than the Limiting Safety System Setting values.

(d) Operation in violation of a Limiting Condition for Operation (LCO) established in the technical specifications unless prompt remedial action is taken as permitted by exception statements.

(e) A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function. If the malfunction or condition is caused by maintenance, then no report is required.

Where components or systems are provided in addition to those required by the technical specifications, the failure of the extra components or systems is not considered reportable.

(f) Any change in reactivity greater than one dollar ($1.00) that could adversely affect reactor safety.

(g) An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of conditions which could result in operations of the reactor outside the safety limit.

(h) Abnormal and significant degradation in reactor fuel, cladding, coolant boundary, or confinement boundary (excluding minor leaks) where applicable.

(2) There shall be a report submitted in writing within 30 days to the NRC, Document Control Desk, Washington D.C. 20555, of:

(a) Permanent changes in the facility organization involving the Director, NCNR, or the Chief, Reactor Operations and Engineering.

(b) Significant changes in the accident analyses as described in the Safety Analysis Report.

6.8 <u>Records</u>

6.8.1 <u>Records to be Retained for a Period of at Least Five Years or for the Life of the Component Involved if Less than Five Years</u>

Records of this section may be in the form of logs, data sheets, or other retrievable forms. The required information may be contained in single or multiple records, or a combination thereof. Annual reports as described in the specifications of Section 6.7.1, to the extent the reports contain all of the required information, may be used as a record of the following:

(1) Normal reactor operation logs, not including supporting documents such as checklists and log sheets. (Supporting documents shall be retained for a period of at least one year.)

(2) Principal maintenance activities.

(3) Special Reports.

(4) Surveillance activities required by these Technical Specifications.

(5) Solid radioactive waste shipped off-site.

(6) Fuel inventories and transfers.

6.8.2 <u>Records to be Retained for at Least One Operator Licensing Cycle</u>

Records of retraining and requalification of licensed operations personnel shall be maintained for the period the individual is employed or until the license is renewed.

6.8.3 <u>Records to be Retained for the Life of the Reactor Facility</u>

(1) Gaseous and liquid radioactive effluents released to the environs.

(2) Off-site environmental monitoring surveys required by these Technical Specifications.

(3) Radiation exposure for all personnel monitored.

(4) Drawings of the reactor facility.

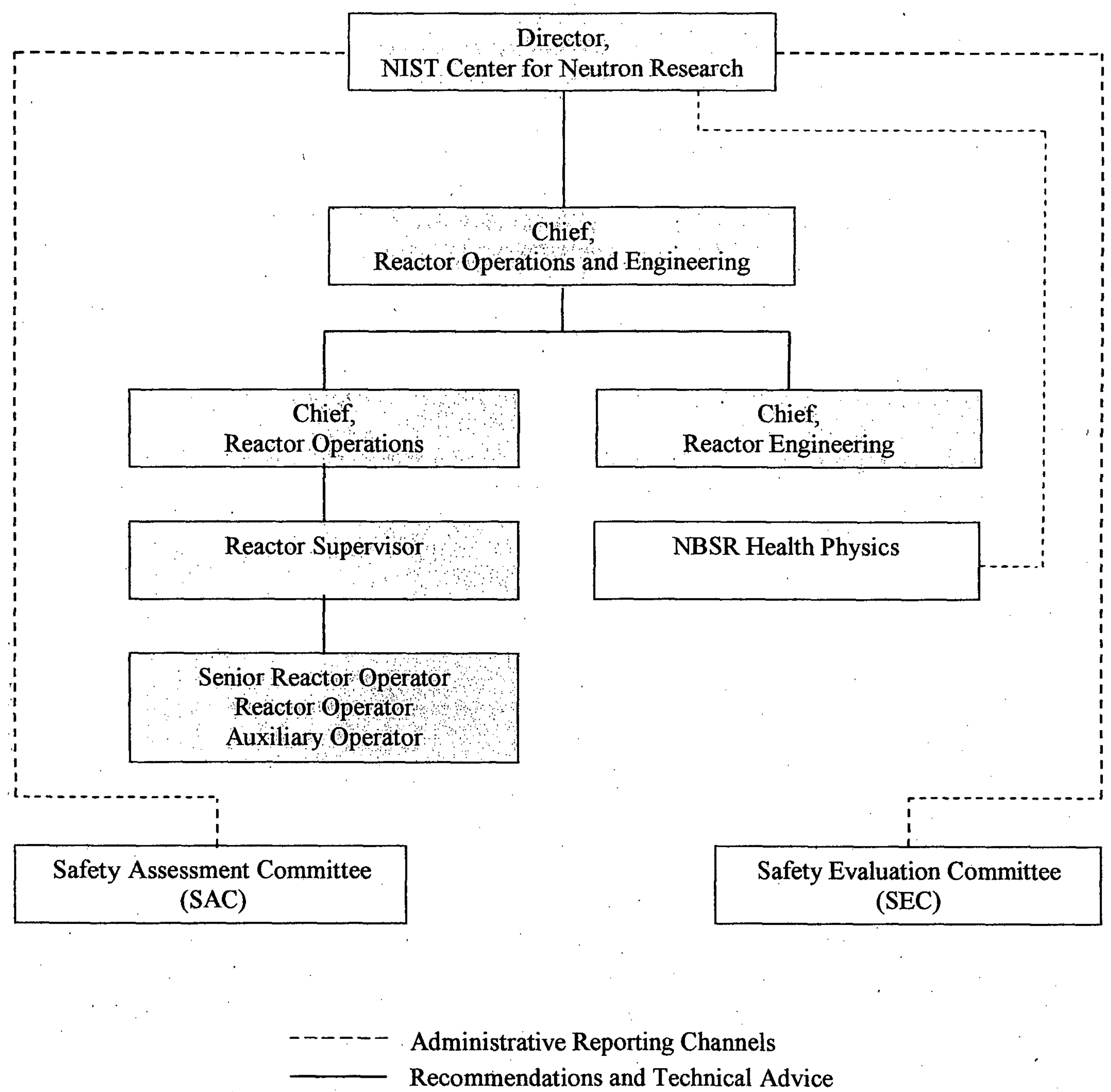
Director,
NIST Center for Neutron Research
Chief,
Reactor Operations and Engineering
Chief,
Reactor Operations
Chief,
Reactor Engineering
Reactor Supervisor
NBSR Health Physics
Senior Reactor Operator
Reactor Operator
Auxiliary Operator
Safety Assessment Committee
(SAC)
Safety Evaluation Committee
(SEC)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

Figure 6.1