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

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

March 19, 2009

Subject: RAI (TAC NO. MD3410)

Docket No. 50-184

Gentlemen:

Please find attached the National Institute of Standards and Technology Test Reactor response to your Request for Information RE: Advisory Committee on Reactor Safeguards Subcommittee Meeting Follow-Up Items (TAC NO. MD3410). Also enclosed are the revised NBSR Technical Specifications. Questions regarding these documents should be addressed to Dr. Wade J. Richards, Chief Reactor Operations and Engineering at 301-975-6260 or wade.richards@nist.gov.

Sincerely,

Wade J Richards

Wade J. Richards, Ph.D.
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: 3/20/09 By: Wade J Richards

Enclosures

cc: U.S. Nuclear Regulatory Commission
ATTN: William B. Kennedy
One White Flint North
11555 Rockville Pike, M/S 012-G13
Rockville, MD 20852-2738

A020
NRR

NIST

Attachment 1
RAI (ACRS Meeting)

RAI (ACRS Meeting)

1. Provide a discussion of the methodology used to monitor the groundwater at the reactor site for contamination by releases of radioactive material from the facility.

Response:

Environmental water samples are collected regularly by the NIST Reactor Health Physics staff. These samples are then analyzed for gamma and beta emitting radionuclides using two methods. A composite sample of all locations is put together and measured by gamma spectroscopy for gamma emitting nuclides and each individual sample is also analyzed by liquid scintillation counting for tritium.

The collected samples are mostly from streams and ponds, both on and off site, in the vicinity of the NBSR. There is one residential well SSW of the site that is sampled when available. The home is usually winterized and not available for sampling during the cold months. There is also one ground water sample collected on the NIST site, from the basement of Building 217, located to the NE of the NBSR. These sample locations surround the NBSR site.

URS Group, Inc. performed an independent study to assess the Adequacy of Downgradient Sampling for NIST Research Reactor Monitoring in April 2006. The URS study conclusions state, "........ URS agrees with the USGS* that it is reasonable to expect that the structural fabric of the bedrock would impose some control on ground water flow. Therefore, groundwater probably moves preferentially in a southwestward direction (parallel to the structural fabric) from the site. Surface water sampling from a point southwest of the site should be adequate to detect potential releases from the reactor site before potential releases were to travel too far. Therefore, since NIST is currently sampling from a surface water point located southwest of the reactor, this appears to be a technically adequate approach to monitoring potential off-site migration of constituents in groundwater".

* US Geological Survey Report on Rate of Movement of Groundwater at the NBSR, by Alfred Clebsch, July 1962.

2. Section 3.4 of National Bureau of Standards Reactor (NBSR) 14, "Safety Analysis Report (SAR) for the National Institute of Standards and Technology Reactor – NBSR" states that the building and reactor systems have been analyzed and shown to be able to withstand the stresses generated

by a 0.1g earthquake loading. This statement references the National Bureau of Standards Reactor Final Safety Analysis Report dated 1966. Please provide a discussion of seismically-induced damage to any safety –related structures and components that have been installed since the analysis in 1966. Please prove a discussion of seismically –induced damage to any structures and components that have been installed since the analysis in 1966 whose failure could impact proper operation of safety-related structures and components.

Response:

The US Geological Survey updated its seismic hazard maps for the conterminous United States based on new seismological, geophysical, and geological information. They employed a probabilistic methodology that uses a combination of gridded spatially smoothed seismicity, large background zones, and specific fault sources to calculated hazard curves for a grid of sites throughout the country (Frankel, et.al. 2002). The documentation for these hazard maps indicates that a maximum moment magnitude (Mmax) of 7.5 is applicable for an area that includes the Wabash Valley, New Madrid, Charleston, the aerial seismic source zones in New England, and the ECC in which the Site is located (Frankel,et.al.,2002). The USGS probabilistic analysis still results in relatively low ground-motion risk for a broad area surrounding the site.

There is a single structure installed since 1966 which could fail in a way that would affect proper operation of the non-vessel portion of the emergency cooling system for the core. The structure is a 6' tall by 17' long irregular block wall erected near emergency cooling water supply components. The wall, commonly known as the shadow shield, serves as a radiation shield to minimize radiation dose to the reactor operators who occupy a nearby office. The radiation source is activation products in the normal heavy water flow through pipes and valves from the emergency cooling water tank to the reactor vessel during reactor operation. Those components and other emergency cooling water components (which provide emergency cooling water in an emergency) are located to the south of the shadow shield. If the shield were to completely collapse to the south at the same time the core required makeup water to the in-vessel emergency cooling water tank, emergency cooling water makeup flow could be impeded. Therefore, this wall is presently being analyzed and the wall will be modified such that it cannot impact the emergency coolant flow.

3. The most recent revision of the proposed Technical Specifications(TSs) submitted September 16, 2008, contains some surveillance requirements that are less conservative that those specified in the current TSs and/or those recommended by the guidance contained in American National Standard ANSI/ANS -15.1-2007. These surveillance requirements include that starting function of the emergency sump pump, start testing of the diesel generators, and voltage and specific gravity testing of the station battery. For each deviation from the standard, please proved a discussion that explains why the recommended surveillance requirement contained in ANSI/ANS - 15.1-2007 is inappropriate or overly –conservative for the NBSR. For each decrease in conservatism in the surveillance requirements in the proposed TSs for the surveillance requirements in the current TSs proved a discussion that explains how the decrease in conservatism maintains the current level of safety. Alternately, revise the proposed TSs to be in conformance with the standard and/or the current TSs.

Response:

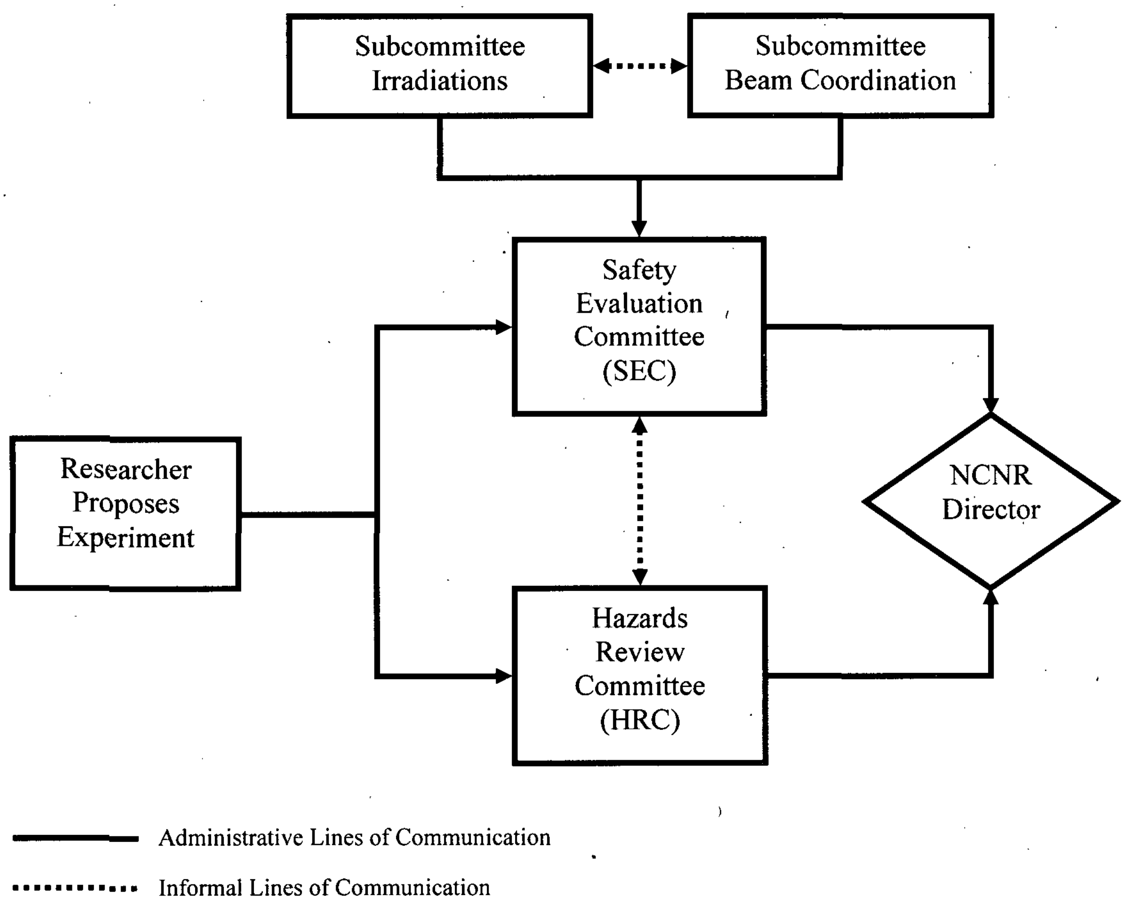
An annual frequency for emergency coolant supply testing is stipulated by section 4.3(2) of ANSI/ANS-15.1-2007. Section 4.3(1) stipulates a quarterly frequency for start testing of an emergency sump pump. The NBSR has three independent emergency coolant supplies: a 3000 gallon tank; the sump pump; and a potable water header. Given the number of redundant sources, ALARA considerations, demonstrated reliability and the manufacturer's recommendation of a 1500 hour maintenance interval, an annual test of the motor/pump is the most appropriate surveillance frequency. A quarterly frequency for diesel testing is permitted by section 4.6.1(1) of ANSI/ANS-15.1-2007.A semi-annual frequency for voltage and specific gravity testing is permitted by section 4.6.2(1) of ANSI/ANS-15.1-2007. An annual frequency is justified because: the battery manufacturer permits maintenance frequencies other than those recommended based on the user's policies and if checked with adequate records; ; data collected over 40 years demonstrates that individual battery cells fail in capacity and in numbers at a rate which will ensure that the entire battery is operable for at least a five year period and because the entire battery is twice the size than that needed to be considered operable; the NBSR has three emergency power supplies; and chemical and electrical hazards associated with the maintenance can be reduced with a reduced maintenance frequency.

4. Please provide a detailed, step-by-step explanation of the experiment review, approval, and implementation process. Include discussions of the applicable requirements specified in the proposed Tss and administrative requirements that ensure no experiment will have an adverse impact on reactor safety or the health and safety of the public and personnel.

Response:

All experiments proposed by researchers at the NIST Center for Neutron Research are reviewed by the Safety Evaluation Committee and the Hazards Review Committee. These reviews ensure that the proposed experiment meets the requirements of the NBSR Technical Specifications sections 3.8, 4.8 and 6.5.

These Technical Specifications prescribe the limits imposed on the experimental reactivity effects and types of materials that can or are excluded from use (i.e. explosives, highly reactive materials and corrosive materials). Technical Specification Section 6.5 requires the process shown in the figure below to be followed, that is, the SEC and HRC review and recommends actions to be taken by the NCNR Director, who will approve or reject the experiment.



Committee Responsibilities

Safety Evaluation Committee

- Review all Reactor Safety Issues
- Establish operational and radiation envelopes
- Review reactor operations and technical specification compliance
- Recommend actions to the NCNR Director

Irradiation Subcommittee

- Review all proposed irradiations (i.e. radiation effects, ALARA compliance
- Ensures proposed irradiations are within SEC approved envelope
- Recommend actions to the SEC Chairman

Beam Coordination Subcommittee

- Review all Beam Tube experiment proposals
- Ensure all Experiments are within SEC approved envelope
- Recommend actions to the SEC Chairman

Hazards Review Committee

- Review all experiments for industrial hazards
- Recommend actions to the NCNR Director

NCNR Director

- Reviews all recommendations and approves or rejects the experiment proposal

5. Please provide a discussion and analysis of the adequacy of natural circulation cooling in the event that DWV-19 is isolated during extended full-power operation. Include discussion of heat sinks, flow paths, peak vessel temperature, and ability of the pressure relief valve to perform its intended function. Also please discuss any actions or preventative measures needed to mitigate the consequences of such a occurrence or prevent it altogether.

Response:

Valve DWV-19 is a motorized 18 in. butterfly valve mounted in the outlet line from the NBSR, with a measured stroke time, fully open to fully closed, of 21 seconds. Although DWV-19 is used only during maintenance, it is conceivable that it could receive a spurious signal while operating, causing a loss-of-flow accident of a type different from those already analyzed. The characteristics of this valve have been used to generate a transient analysis of the system using RELAP, with the results shown below. The starting power and flow are chosen (as for all accident analyses) as being at the alarm setpoints for power, primary inlet power and flow. The reactor is assumed to have been operating at full power for a full cycle when the spurious signal to close DWV-19 is received. The valve is assumed to go from fully open to fully closed in the measured stroke time of 21 seconds.

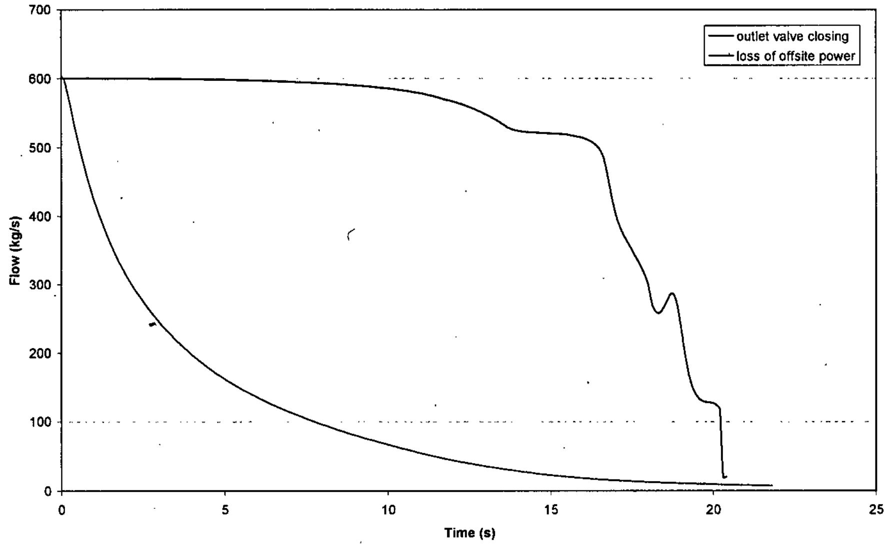


Figure 1. RELAP simulation of closing of DWV-19 at full power.

Table 1.

	MCHFR		OFI	
	Mirshak[i]	Sudo-Kaminaga[ii]	Costa[iii]	Saha - Zuber[iv]
Start up Core	2.18	2.63	3.79	1.67 (lf)(*)
End of Cycle Core	2.81	3.44	5.32	2.35

(*) lf = Saha-Zuber for low flows (Peclet is close but < 70000)

Trip times:

For SU: 17.24 s (low outer plenum flow)
For EOC: 17.24 s (low outer plenum flow)

As can be seen from Figure 1, Table1 and Appendix A there will be no departure from nucleate boiling during the forced flow reduction, given the CHFR and OFI ratio found for this accident. Further, the model used is the same as was used for the SAR, which has been shown to be very conservative, since it ignores thermal conductivity of the fuel and clad, non-local deposition of energy at the hot spot, and non-uniform burnup. The factors mentioned decrease the peak heat flux by a factor of 1.6, giving a very large margin to fuel damage. Previous calculations reported in the SAR show that natural circulation is adequate to maintain the fuel clad temperature well below the Safety Limit following a scram. However, it is necessary to address the heating of the heavy water moderator during removal of the decay heat from the fuel after primary circulation is stopped. The decay power from the core has been estimated using the methods described in ANS-5.1, and assuming that DWV-19 closes at the worst possible point of the cycle – the end of cycle, where the decay heat will be largest, with the results shown below.

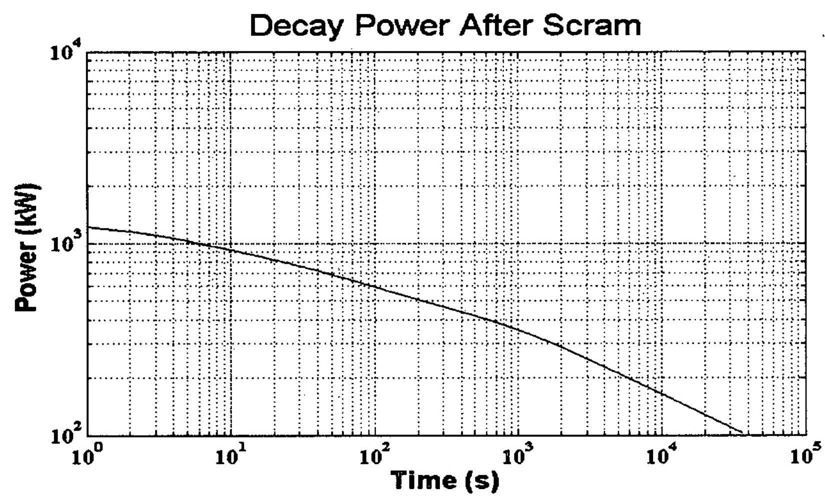


Figure 2.Decay power from NBSR following scram.

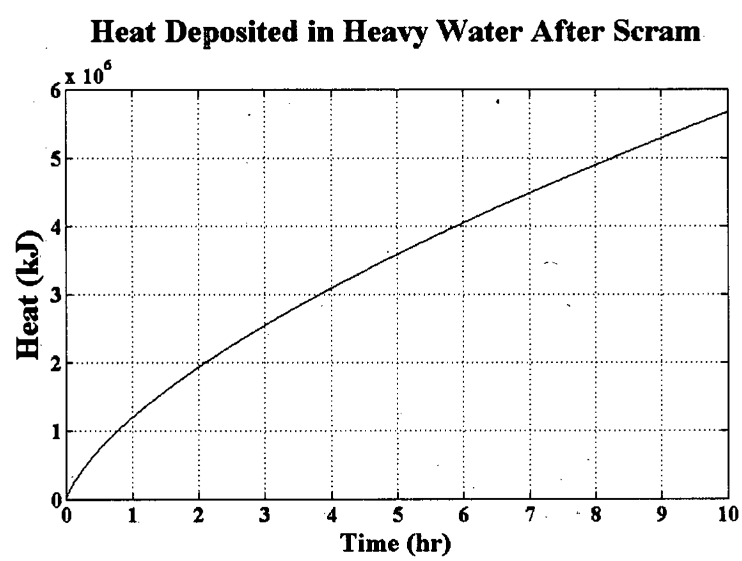


Figure 3.Total heat deposited in heavy water, assuming no heat sinking to the vessel or biological shield.

The results given in Figure 3 can be used to estimate the temperature rise of the moderator with DWV-19 closed, again assuming no heat transfer to the reactor vessel or biological shield. The total volume of heavy water in the vessel can be calculated from the data given in the SAR. In particular, the diameter of the vessel is 2 m, and the height of the heavy water is at least 3.81 m above the lower grid plate. This gives a volume of heavy water of 12 m^3, to which must be added the volume below the lower grid plate. This volume corresponds closely to the elliptical shell that forms the bottom of the vessel, which has a volume of 1.57 m^3, for a total volume of 13.5 m^3. The density of heavy water at saturation temperature and 1 bar pressure is 1062 kg/ m^3 and it is conservative to use this value for our estimates of water heating. The specific heat of heavy water is approximately 4.18 kJ/kg/K at representative temperatures and pressures, so that for the total volume of heavy water, a temperature rise of 1K will require a heat addition of:

$$\Delta q = 13.5 * 1062 * 4.18 = 6 \times 10^4 \text{ kJ}$$

Thus, using the results shown in Figure 3, after 2 hours in which less than 2×10^6 kJ will added to the water, the temperature will have risen by only 33 K, thus remaining well below the saturation temperature, and this provides more than ample time to open DWV-19 manually, and restore normal circulation.

Thus, the spurious closing of DWV-19 while operating the reactor at 20 MW will not result in any fuel damage.

Referring to the system P&ID drawing E-60-035, the motor operated control valve DWV-19, is located in the 18 inch diameter reactor outlet pipe, which in turn is connected to the suction of the D_2O main pumps. If this valve is inadvertently closed when the reactor is operating at 20MW, the relief valve, which is downstream of DWV-19 is isolated from the reactor vessel. However, in such an event, the vessel is not isolated, and cannot "go solid". As discussed in the preceding analysis, it will take more than two hours before any steam can be generated, and this allows ample time to open DWV-19, manually if necessary. In addition, there are several vent paths that will remain open while DWV-19 is closed.

1. The reactor vessel is maintained under helium blanket system posing a positive pressure of 3 inches of D_2O and has a loop seal with 10 inches of D_2O. In the presence of any excess pressure, the loop seal will be broken, and the vessel will be vented through the 3 inch line.

2. The vessel is also vented to the D_2O storage tank, emergency D_2O storage tank and helium system pipe lines via helium pressure relief and the loop seal mentioned above. The loop seal vents into the process room atmosphere. The vessel can be vented to the irradiated air system, if needed.

3. An overflow line is connected to the D_2O storage tank via a 3 inch diameter pipe, which will also serve as a vent for the vessel.

4. The overflow system dry leg is equipped with a level control such that when it sees water it will yield a zero level indication and the main pumps will shutoff before any substantial pressurization of the vessel can occur. The overflow will be directed to the D_2O storage tank.

Thus, the reactor vessel remains vented when DWV-19 is closed, with many relief paths. There is no threat to vessel integrity through such an accident.

[1] S. Mirshak, W. D. Durant and R. H. Towell, "Heat Flux at Burnout", DP-355, DuPont (1959)
[1] Y. Sudo and M. Kaminaga, "A New Critical Heat Flux Correlation Scheme Proposed for Verticular Rectangular Channels heated from Both Sides in Nuclear Research Reactors", Transactions of the ASME **115** (426) 1993
[1] J. Costa, "Measurement of the Momentum Pressure Drop and Study of the Appearance of Vapor and Change in the Void Fraction in Subcooled Boiling at Low Pressure", ORNL/TR-90/21, Oak Ridge National Laboratory
[1] P. Saha and N. Zuber, "Point Of Net Vapor Generation And Vapor Void Fraction In Subcooled Boiling", Fifth International Heat Transfer Conference, Tokyo, Japan

6. The moderator temperature coefficient and moderator void coefficient presented during the ACRS subcommittee meeting appeared to be inconsistent with each other. Please discuss how these coefficients were determined including methods of calculation. Please discuss whether these coefficients necessarily need to be consistent given the methods of calculation. Please discuss how these coefficients represent the most limiting conditions in the coolant and/or how the values chosen for these coefficients provide adequate conservatism.

Response:

The MTC and the void coefficient appear to be inconsistent with each other because they were not computed for the same volumes of water. The MTC was calculated* over the entire volume of D_2O in the NBSR vessel: the coolant inside the fuel elements (FE), the moderator between the FE, and the reflector region. The range of calculated values is -0.031 to -0.025 %$\Delta\rho$/°C, from SU to EOC. The MTC was also measured by heating the entire vessel, and determined to be -0.029 %$\Delta\rho$/°C. Thus, the calculation of the MTC using MCNP is certainly adequate.

The void coefficients presented in the SAR, and modified in the response to Question 13.21 of the first set of RAI, were for very specific volumes of D_2O. Coefficients were calculated for the coolant, by voiding only those cells inside the fuel elements (-0.037 to -0.030 %$\Delta\rho$/liter), and the moderator between the FE, by voiding the cells inside 6 vertical thimbles (-0.043 to -0.030 %$\Delta\rho$/liter). The former were calculated for use in RELAP analyses of transients (though ***we later omitted all the reactivity feedback in the accident analyses for conservatism***), while the later coefficients were calculated to verify that the void coefficient is indeed negative everywhere.

Recently, a core-wide void coefficient was calculated by reducing the D_2O density everywhere by about 9% (750 liters). The reactivity change was $\Delta\rho$ = -3.5 %$\Delta\rho$. Thus the core-wide void coefficient would be -0.005 %$\Delta\rho$/liter, considerably smaller than those listed above. It really does matter where the void is created.

If one wishes to check for consistency between the MTC and the void coefficient, one can compute the moderator density coefficient. For the case above, the density change was -100 kg/m^3, so the density coefficient would be +0.035 %$\Delta\rho$/kg/m^3. This value is roughly consistent with the density decrease associated with a temperature increase, reflected in the MTC*.

* Three MCNP input parameters affect the MTC, namely the water density, the temperature assigned to the cells, and the temperature for the thermal neutron scattering kernel. The calculations of the MTC are presented in Table 3-1 of Appendix A in the NBSR Safety Analysis Report submitted in 2004 (NBSR-14). From Table 3-1, at EOC, the density change associated with a temperature increase from 319 K to 373 K is -33.1 kg/m^3, and the reactivity change is -0.968 %$\Delta\rho$. Thus the density coefficient for this change is +0.029 %$\Delta\rho$/kg/m^3, which is not so different with that above.

7. Please discuss the storage and disposal of Class B and Class C waste that may be generated at the NBSR during the period of the renewed license. Note that your response should not contain any sensitive or security –related information.

Response:

The NBSR does generate some Class B and C wastes as defined by 10CFR61.55. These waste items are typically reactor materials and components, such as the non-fuel sections of the NBSR spent fuel elements. Over the last twenty years or so our records indicate that we have averaged about one shipment of approximately 80 to 90 cubic feet, every four years. These have been shipped in Type B shipping casks, for burial at the disposal site in Barnwell, SC. The Barnwell facility closed its doors to outside compact waste shipments effective July 1st, 2008. The last shipment from the NCNR was received at Barnwell in June 2008. Hence, the year 2012, or so, will present us with a need for disposal or storage. We currently have space in the reactor fuel storage pool for at least this much material, and probably more if space is used efficiently. We have also incorporated new, shielded storage space in the NCNR expansion project. The new construction component of the NCNR expansion project which is currently underway and scheduled for completion sometime in 2010. The provisions incorporated in this phase of the project should give us adequate storage for forty years or more of facility operation or at least until another disposal option is presented to us. The proposed shielded storage areas have been designed to provide adequate shielding to keep all accessible radiation levels to ALARA, and are expected to be less than 0.5 mr/hr at one foot from any accessible surface. This is based on design elements and historical data. These storage areas are located inside a restricted access area of the facility with secure access controls. All persons having access to these areas are required to be trained and monitored.

8. Proposed TS 3.9.1, "Fuel Storage", specified that "all fuel elements 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". Please provide a summary and discussion of this calculation for storage in the spent fuel storage pool. Note that your response should not contain any sensitive or security-related information.

Response:

A series of MCNP criticality calculations was performed to estimate k_{eff} for each of the four types of fuel storage racks in use in the NBSR Spent Fuel

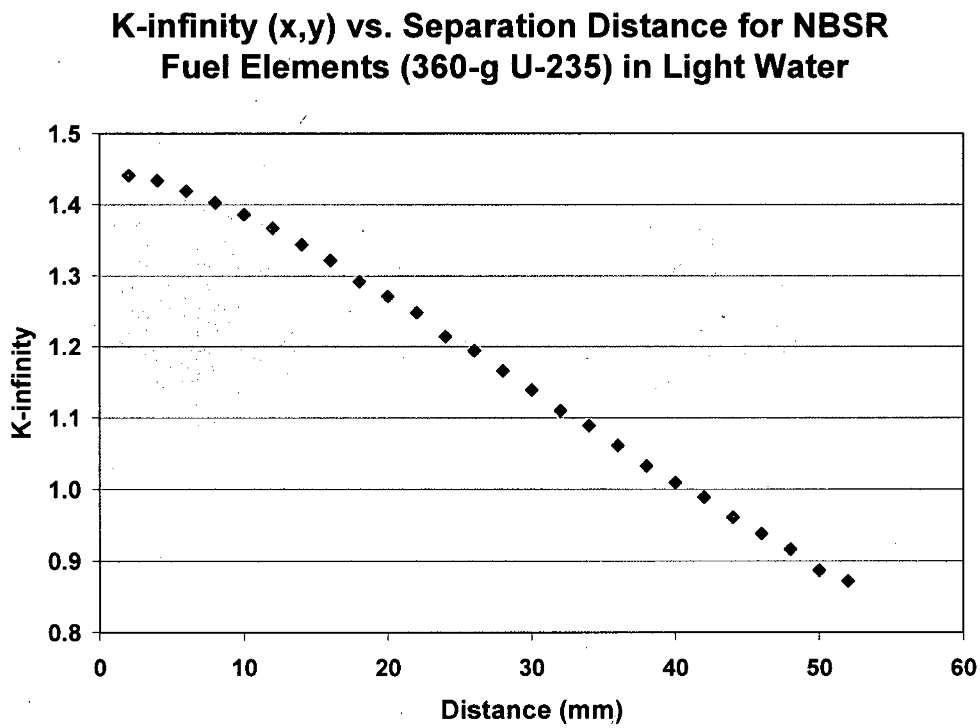
Pool. In each rack, it is possible to model a "unit cell" of the rack, and, by the use of parallel reflecting boundaries, determine k_∞ for an infinite line of unit cells. All of the allowed configurations are far subcritical, 0.5 to 0.62, as can be seen in the table below. The calculations were very conservative in that the elements were loaded with 360 grams of ^{235}U rather than the nominal value of 350 g, and there was no allowance for burnup, nor were there any fission products present. Details of the calculations are in the report "Criticality Safety Evaluation of the Fuel Storage Racks in the NBSR Spent Fuel Pool" by R. E. Williams, dated March 15, 2009.

Criticality safety is assured in the pool by the separation distance between the FEs when they are secured in position in each type of rack, and by the fact that only one FE is being moved at any time. An infinite plane of FEs was also modeled to show the dependence of k_∞ on the thickness of light water between the elements. In the figure below, it can be seen that an infinite array is safely subcritical if there is just a 5-cm layer (2 inches) of H_2O in each direction ($k_\infty = 0.88$) between the FEs. The racks are a minimum of 18 inches apart. The maximum k_{eff} for 6 FEs in H_2O is 0.78, when they are packed into a corner of the pool (procedures allow 6 FEs to be placed temporarily outside of any rack). Finally, the racks are securely mounted on the walls or bolted to the floor so they cannot be moved.

There is no possibility of an inadvertent criticality in the NBSR Fuel Storage Pool if the elements and fuel pieces are secured properly in the four types of storage racks currently in use, and the procedures for moving fuel are strictly followed.

Fuel Storage Rack Type	K-infinity
Two-Row Fuel Element Rack*	0.5061 ± 0.0011
Single-Row FE Wall Rack	0.5773 ± 0.0007
Fuel Pieces in Stainless Steel Cells	0.5079 ± 0.0007
Combination Rack* (Whole FEs and Fuel Pieces)	0.6210 ± 0.0003

* Rack contains Boral omitted in the MCNP model.



9. Please clarify whether the confinement building design is based on loading from 100-mph sustained winds or a 100-mph wind gust.

Response:

At the time of original construction (planning 1958; construction start 1963; construction complete 1968) the governing code in Gaithersburg, Montgomery County, Maryland was BOCA (Building officials Code Administrators), 1955 edition. The wind load provisions of this version of BOCA were based on ANSI A58.1-1955, Minimum Design Loads in Buildings and Other Structures (Note: the name shared with the modern ASCE 7 is no coincidence; ANSI A58.1 is the predecessor document). In ANSI A58.1-1955 wind speeds were not given on any basis; the design process was based solely on "Minimum Allowable Resultant Wind Pressures" as shown graphically in Figure 1 (copy attached) and provided for selected locations in Table A7.

However, as noted in the original document, early codes typically used fastest-mile wind speeds. This is because the typical equipment of the time could not record short duration (gust) speeds. Thus, we presume that the background data used to develop the method in ANSI A58.1-1955 was recorded on a fastest-mile basis but no documentation could be found to support this.

"Sustained Wind" is defined today by NOAA as "The wind speed obtained in the U.S. by averaging observed values over a period of at least 1 minute." Because the fastest-mile method is effectively an averaging function over variable time – varying according to the wind speed (see original text for explanation) – it would be categorized as "sustained" for winds speeds less than 60 mph. Higher wind speeds, such as the extreme event speeds under consideration here, would not be categorized as "sustained" (for example, the 100mph speed referenced in the question would be measured by the fastest-mile method in 36 seconds).

Question 10 and 11 are answered together.

10. Please provide a discussion of the derivation of the 50-year and 100 year numerical scaling factors for wind speeds. If Caribbean hurricane wind data was used to estimate the 100-year maximum wind gust speed at the NBSR site, please provide a discussion of the relevance of that date to winds at the NBSR site. Otherwise, please provide a discussion of the wind data used to estimate the 100-year maximum wind just speed for the NBSR site.

Response:

The referenced ratios were obtained by dividing the 100-year Mean Recurrence Interval (MRI) wind speeds by the 50-year MRI wind speeds from typical probability curves (or charted data from same) in each of the referenced documents. The relevant charts (Figure 1 and Table 1) from the second reference (*Return Period of Hurricane Perils...*, original document footnote #7) are attached here for reference. Figure 1 shows a standard reverse Weibull distribution curve that has been fit to a set of data for the Kingston, Jamaica Metropolitan Area (KMA). Table 1 provides the same information in a different format, with the first column corresponding directly to curve in Figure 1 (MLE = Maximum Likelihood Estimate). From Table 1 we take the 100-year and 50-year numbers and divide to derive the MRI multiplier: 102/89=1.146.

For this to be functionally applicable two things must hold true. First, the Weibull distribution must be reliable. This has been borne out by research; both Simiu and Heckert (1996) and Simiu and Whalen (1998) demonstrate that regions characterized by both extratropical storms and hurricanes are best represented by reverse Weibull distributions. Second, the shape a variety of Weibull curves must be sufficiently regular for the ratio to be nearly constant. An examination of a limited number of curves reveals some variation with higher ratios found in regions with more frequent and higher-intensity storm, as characterized by the KMA data discussed above.

For the United States a simple conversion is offered by ASCE-7-2005 in Table C6-7 (attached for reference). This table gives the ratio as 1.07 for all regions except Alaska, which is given as 1.06. From this you can see that the MRI multiplier derived in the original document is somewhat conservative, as would be predicted by the use of Caribbean hurricane data.

12. In March 2008, storms in the Washington DC metro area generated wind gust speeds of 74 mph and 60 mph measured at Reagan National Airport and 66 mph measured at Dulles International Airport. Discuss how this storm data affects the projection of the 100-year maximum wind gust speed.

Response:

These data points are relevant only to the extent that they might be incorporated into future reassessments of the wind speed curves in the model codes. If, as was traditionally done, the MRI is based on "epochal" data (the single, maximum wind speed data point per epoch – 1 year) these reading might not be included at all if higher readings exist for the year.

However, today the data sets used are most commonly based on the "peaks over threshold" method whereby every extreme wind speed record (a peak) that exceeds a certain threshold is recorded. If the referenced wind speeds exceed the threshold established for the data set, which they likely would given the relatively high speeds recorded, then they would be included, although they may have little or no impact on the final wind speed curves displayed in the model building code(s). Setting of appropriate thresholds to ensure quality models is an ongoing area of study.

13. Please discuss the technical basis for the applied additional rain-on-snow loading including historical data to justify the assumed 50% rain fraction.

Response:

ASCE 7-05, Article 7.10 states:

> For locations where p_g is 20lb/ft^2 or less, but not zero, all roofs with slopes (in degrees) less than W/50 with W in ft shall have a 5 lb/ft^2 rain-on-snow surcharge. This rain-on-snow augmented design load

applies only to the balanced load case and need not be used in combination with drift, sliding, unbalanced, or partial loads.

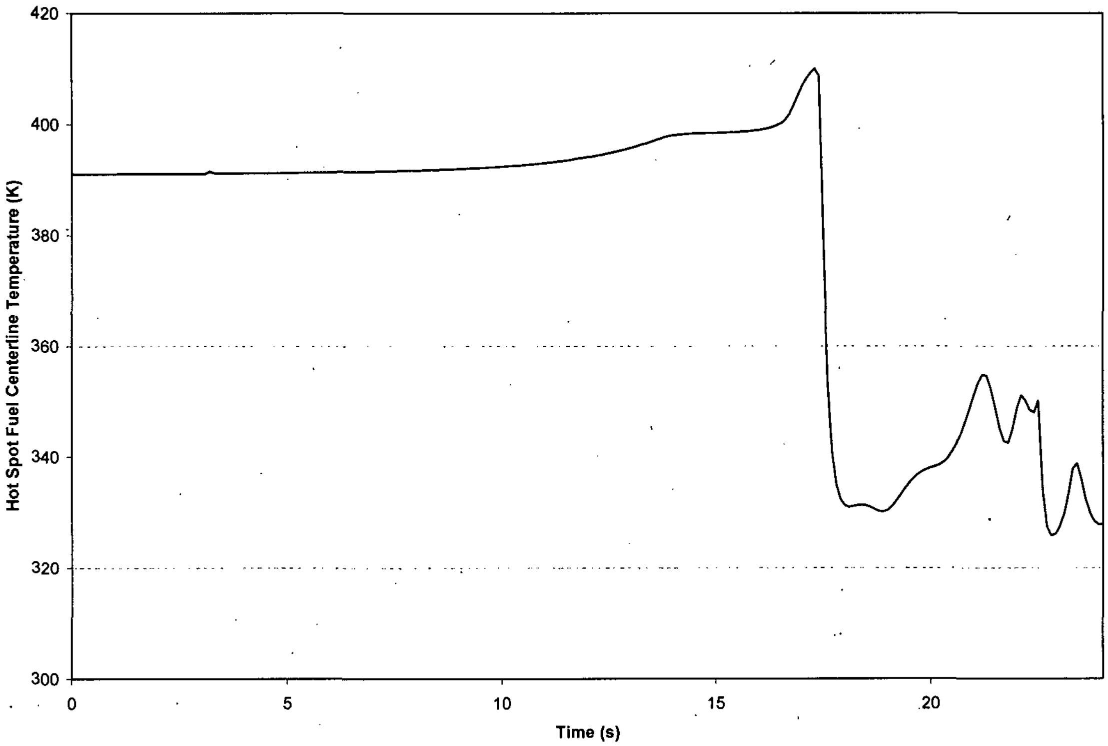
Based on this it is not strictly necessary to apply the rain-on-snow surcharge because the ground snow load (p_g) is 25 psf in Montgomery County, Maryland. However, the application of a surcharge is conservative given the p_g in this location.

The literature expresses no consensus on the amount of rain that might be prudently planned for in the event of resting snowpack on a roof. References suggesting use of part or all of 10-year to 50-year MRI rain events can be found.

Appendix A: Details of excursion

Transient Conditions at Hot Spot Following Outlet Valve Closing (SU)

Time (s)	Primary Flow (kg/s)	Inner Plenum Flow (kg/s)	Power (MW)	MCHFR (Mirshak)	MCHFR (Sudo-Kaminaga)	OFI ratio (Costa)	OFI ratio (Saha-Zuber)	Fuel temp (K)
0.0	600.0	157.9	20.4	2.67	3.14	5.58	1.93	391.1
2.0	599.6	157.8	20.4	2.66	3.14	5.58	1.93	391.1
4.0	598.7	157.6	20.4	2.66	3.14	5.57	1.93	391.2
6.0	596.8	157.0	20.4	2.66	3.14	5.55	1.93	391.4
8.0	593.0	156.0	20.4	2.65	3.13	5.53	1.92	391.8
10.0	585.6	154.0	20.4	2.63	3.11	5.47	1.92	392.4
12.0	566.8	148.9	20.4	2.60	3.07	5.34	1.90	394.1
14.0	524.2	137.4	20.4	2.50	2.98	5.02	1.86	398.0
15.0	520.6	136.8	20.4	2.49	2.97	4.98	1.85	398.5
15.2	520.0	136.6	20.4	2.49	2.97	4.98	1.85	398.5
15.4	519.3	136.4	20.4	2.49	2.97	4.97	1.85	398.6
15.6	518.0	136.0	20.4	2.49	2.97	4.96	1.85	398.6
15.8	516.0	135.4	20.4	2.49	2.96	4.95	1.85	398.8
16.0	513.7	134.8	20.4	2.48	2.96	4.94	1.85	399.0
16.1	512.0	134.3	20.4	2.48	2.96	4.92	1.84	399.1
16.2	509.8	133.6	20.4	2.47	2.95	4.91	1.84	399.3
16.3	506.8	132.7	20.4	2.47	2.95	4.89	1.84	399.5
16.4	502.9	131.6	20.4	2.46	2.94	4.87	1.84	399.7
16.5	497.2	129.8	20.4	2.45	2.93	4.83	1.83	400.1
16.6	487.2	126.9	20.4	2.44	2.92	4.77	1.83	400.7
16.7	469.1	120.9	20.4	2.42	2.89	4.67	1.82	401.7
16.8	445.1	113.3	20.4	2.38	2.85	4.51	1.80	403.3
16.9	421.4	106.1	20.4	2.33	2.80	4.33	1.78	405.1
17.0	401.9	100.6	20.4	2.28	2.75	4.16	1.75	406.8
17.1	386.7	96.8	20.4	2.24	2.70	4.01	1.72	408.2
17.2	375.6	94.0	20.4	2.21	2.66	3.88	1.69	409.2
17.3	367.0	92.3	20.4	2.18	2.63	3.79	1.67	410.0
17.4	358.8	90.5	16.9	2.25	2.72	3.87	1.73	408.6
17.5	350.6	88.6	4.1	4.17	5.10	7.45	3.36	380.0
17.6	341.8	86.3	2.2	7.91	9.87	15.12	6.87	353.8
17.7	332.7	83.9	2.0	12.62	15.98	24.91	11.42	340.8
17.8	323.0	81.3	2.0	16.86	21.53	33.54	15.59	335.0
17.9	312.5	78.4	2.0	19.97	25.67	39.62	18.70	332.4
18.0	297.9	74.3	2.0	21.95	28.39	43.13	20.79	331.2
18.2	261.6	63.0	1.9	23.82	31.44	45.09	23.15	331.1
18.4	260.2	64.5	1.9	23.83	31.60	44.87	23.22	331.4
18.6	278.1	72.0	1.8	23.83	31.38	45.67	23.01	330.9
18.8	283.7	74.3	1.8	24.64	32.24	47.58	23.69	330.2
19.0	238.7	58.4	1.8	26.25	35.03	48.27	25.93	330.5
19.2	179.5	38.8	1.7	27.37	38.16	46.09	28.09	332.1
19.4	145.9	29.0	1.7	27.17	39.37	42.51	28.50	334.3
19.6	132.5	26.1	1.7	26.31	38.88	39.49	27.82	336.2
19.8	129.0	26.6	1.7	25.50	37.95	37.58	26.99	337.4
20.0	127.4	27.4	1.6	25.14	37.54	36.68	26.63	338.0



Hot Spot Fuel Centerline Temperature (SU)

Transient Conditions at Hot Spot Following Outlet Valve Closing (EOC)

Time (s)	Primary Flow (kg/s)	Outer Plenum Flow (kg/s)	Power (MW)	MCHFR (Mirshak)	MCHFR (Sudo-Kaminaga)	OFI ratio (Costa)	OFI ratio (Saha-Zuber)	Fuel temp (K)
0.0	600.0	442.1	20.4	3.31	3.90	7.29	2.51	376.1
2.0	599.6	441.8	20.4	3.31	3.90	7.28	2.51	376.2
4.0	598.7	441.1	20.4	3.31	3.90	7.28	2.51	376.2
6.0	596.8	439.7	20.4	3.31	3.90	7.26	2.51	376.4
8.0	592.9	436.9	20.4	3.30	3.89	7.23	2.50	376.7
10.0	585.6	431.6	20.4	3.28	3.87	7.17	2.50	377.2
12.0	566.8	417.9	20.4	3.24	3.83	7.02	2.49	378.6
14.0	524.2	386.7	20.4	3.14	3.74	6.67	2.46	382.1
15.0	520.6	383.7	20.4	3.13	3.73	6.63	2.45	382.6
15.2	520.0	383.3	20.4	3.12	3.73	6.63	2.45	382.6
15.4	519.3	382.8	20.4	3.12	3.73	6.62	2.45	382.7
15.6	518.1	381.9	20.4	3.12	3.73	6.61	2.45	382.8
15.8	516.0	380.5	20.4	3.12	3.72	6.60	2.45	382.9
16.0	513.8	378.9	20.4	3.11	3.72	6.58	2.45	383.1
16.1	512.0	377.6	20.4	3.11	3.72	6.57	2.45	383.2
16.2	509.8	376.1	20.4	3.11	3.71	6.56	2.45	383.4
16.3	506.9	374.1	20.4	3.10	3.71	6.54	2.45	383.6
16.4	503.0	371.4	20.4	3.10	3.70	6.51	2.45	383.9
16.5	497.3	367.5	20.4	3.09	3.70	6.48	2.45	384.3
16.6	487.5	361.1	20.4	3.08	3.69	6.42	2.45	384.9
16.7	469.3	348.6	20.4	3.06	3.68	6.33	2.46	386.0
16.8	445.2	331.9	20.4	3.03	3.66	6.19	2.46	387.9
16.9	421.5	315.2	20.4	2.99	3.62	6.01	2.45	390.3
17.0	401.9	301.0	20.4	2.94	3.57	5.82	2.43	392.7
17.1	386.8	289.7	20.4	2.88	3.52	5.64	2.40	394.7
17.2	375.5	281.0	20.4	2.84	3.47	5.49	2.37	396.3
17.3	367.0	274.4	20.4	2.81	3.44	5.37	2.35	397.4
17.4	358.9	268.1	19.3	2.81	3.45	5.32	2.36	397.9
17.5	350.6	261.8	12.2	3.39	4.17	6.40	2.87	388.6
17.6	341.9	255.3	3.6	5.85	7.25	11.13	5.05	363.0
17.7	332.7	248.5	2.0	11.01	13.79	21.34	9.80	344.4
17.8	323.1	241.5	1.8	17.29	21.88	34.04	15.83	335.2
17.9	312.6	234.0	1.8	22.57	28.81	44.70	21.10	330.8
18.0	298.2	224.2	1.7	26.65	34.28	52.53	25.30	328.5
18.2	261.2	197.7	1.7	31.37	41.27	59.84	30.71	327.3
18.4	260.4	195.2	1.7	32.28	42.67	61.38	31.72	327.0
18.6	278.7	206.1	1.6	32.63	42.79	63.11	31.74	326.6
18.8	284.5	210.3	1.6	33.80	44.06	65.81	32.72	326.1
19.0	238.7	180.2	1.6	35.89	47.77	66.61	35.73	326.3
19.2	179.4	140.0	1.6	37.30	51.95	63.48	38.64	327.5
19.4	145.8	116.2	1.5	36.96	53.59	58.59	39.24	329.0
19.6	132.3	105.6	1.5	35.82	53.07	54.61	38.41	330.3
19.8	128.9	101.7	1.5	34.78	51.96	52.17	37.40	331.1
20.0	127.6	99.6	1.5	34.37	51.53	51.07	37.00	331.6

Appendix B

Figure 1 & Table A7 (question 9)

1. Introduction

Return periods provide a conceptual basis for quantifying the uncertainties associated with wind, wave and storm surge due to tropical cyclones. Return period is technically defined (Simiu and Scanlan, 1996, p. 603) as the reciprocal of the annual probability of observing a specific hurricane effect (or more extreme effect). Suppose the probability distribution corresponding to the annual extreme wind at a given site were known (for example, the smooth curve in Figure 1). The ninetieth percentile of this distribution reflects the 10 year return period wind, since there is a one in ten chance of observing such a wind (or greater) in a given year. Similarly, the ninety-sixth percentile gives the 25 year return period wind; the ninetyeighth percentile gives the 50 year return period wind and the ninety-ninth percentile gives the 100 year return period wind. The key to determining return periods is to determine the probability distribution of annual extremes. If the distribution were "known," then there would be no uncertainty associated with the return period values. Of course, these distributions are not known and must be estimated from available data and modeling efforts.

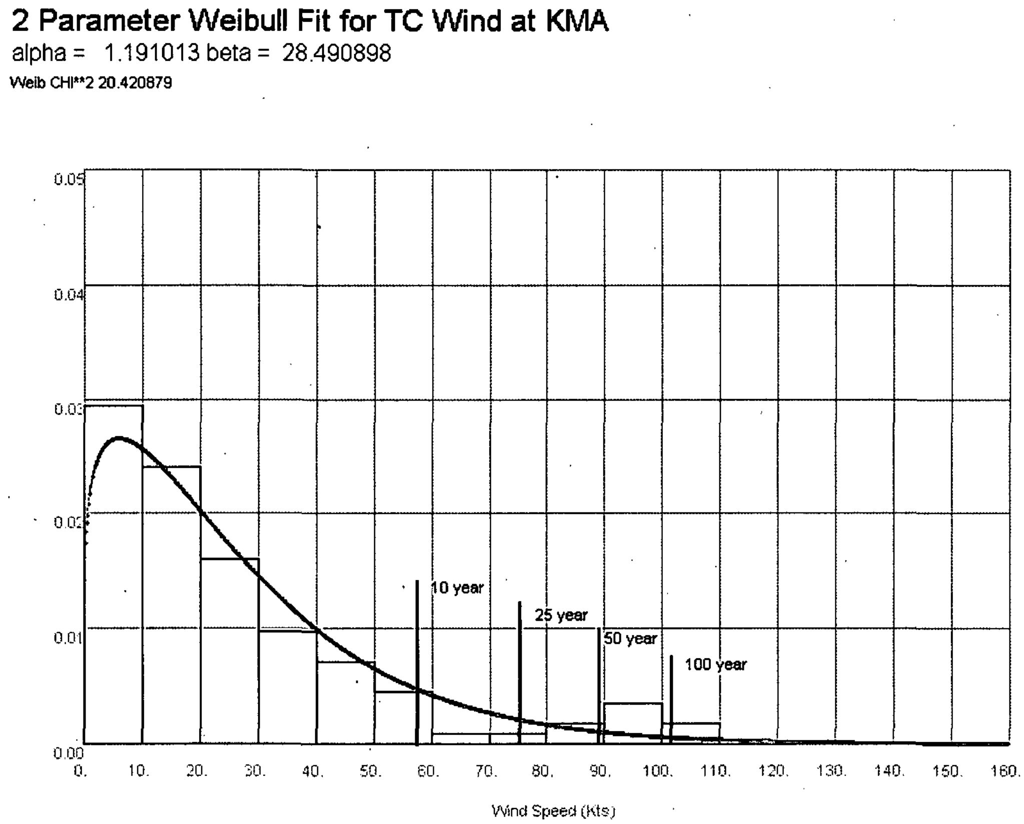


Figure 1. Relationship of Return Period to Annual Probability Distribution of Extreme Wind (1 minute sustained at 10m, in knots).

The approach applied here for estimation of the annual probability distribution of extremes has been reported by Johnson and Watson (1998) and is
given in abbreviated form in Appendix A. The historical Atlantic tropical cyclone storm set HURDAT (Neumann, Jarvinen, McAdie and Elms, 1993 with updates through 1997) provides a set of 959 storms for the period 1886-1997. For each storm, TAOS/R (Watson, 1995) simulates the hurricane using the current topography, land use/land cover and bathymetry and collects the maximum wind, wave and storm surge height over the duration of the
storm at each study site location (i.e., for each grid cell). The annual maximum of each peril is recorded and provides the basic "data" used in the statistical fitting procedure outlined in Appendix A. The Weibull distribution is used as the probability distribution of choice for each peril, with the distributions estimated separately for each location. From the fitted Weibull distributions, both maximum likelihood estimates (best estimates) and

attendant prediction limits (plausible upper limits) are determined. Specific examples of the use and interpretation of these calculations are given in the next section.

Previous approaches to determining return period phenomena in the Caribbean have been rather subjective, focusing on the most severe event or events
recorded this century followed by some intelligent perturbations of these storms to identify 25-, 50-, and 100- year basic events. The extent of selection and modification of the events could lead to debate that is driven by the results of the analysis rather than by the intrinsic scientific merit of the approach. Another classical approach involves selecting all storms within a given distance of a site and then performing an analysis presuming that
the storms were direct hits (Chu and Wang, 1998). By using the complete set of storms in this study, issues with an arbitrary distance cutoff are circumvented.

The purpose of this report is to present the results of the return period estimation methodology as it pertains to the Caribbean, especially the Kingston
area of Jamaica. These results are intended to address the question, "What sort of extreme hurricane phenomena should be expected at a given site in the next n years?" Values of n of interest here are 10, 25, 50 and 100 years. Complete statistical details can be found in Johnson and Watson (1998), beyond those available in Appendix A. An earlier version of the methodology can be found in Johnson (1997).

Section 2 introduces the return period estimation calculations and interpretations, using the center of the port of Kingston as illustration. Section 3
expands consideration to the entire Caribbean at low resolution (1 arc minute and 30 arc seconds). An electronic data base of return periods (maximum likelihood estimates and 90% prediction limits) for wind, wave and storm surge is described. Section 4 addresses the issue of extrapolating low resolution results to higher resolution grids, using multiple resolution results obtained in Jamaica and Florida. Situations are characterized where the low resolution results may suffice and indicate other possibilities where high resolution results could be different and quantify the discrepancies.

Finally, in Section 5 the key findings for the Organization of American States (OAS) are reviewed and a discussion of potential future efforts are described.

2. Return Period Estimation
To illustrate return period concepts applied to a specific example, consider a central location in the port of Kingston, Jamaica (specifically at 76.676W, 17.976N). All storms in the full set of 959 storms were simulated using TAOS/R and the extreme effects of wind, wave and storm surge were recorded at each site for the storms' duration. TAOS/R is a meteorological hazard model having an object oriented design with state of the art
modules for wind, wave and storm surge components of the model (Watson, 1995; Watson and Johnson, 1998). TAOS/R integrates a geographic information system technology allowing the seamless input and output of calculated results for graphical display. The statistical analysis (TAOS/SAP) proceeds using the annual maxima (112 values) for each peril. Focussing on wind for the Kingston port, the Weibull fit gave the estimate (1.191013, 28.490898) for the parameters (a , b). This fit was given previously in Figure 1 in Section 1. For many applications (structural design, land use planning and emergency management), the associated return periods are of primary interest, while the Weibull fits provide assurance that the methodology is in order. The results for the central port (units in knots) are, as follows:

	MLE	50%	75%	90%	95%	99%
10 year	57	58.2	61.2	63.9	66	70.4
25 year	76	77	81.6	86.7	90.6	104.4
50 year	89	90.5	97	105	111.4	130.4
100 year	102	103.1	112.8	124	133.1	157.8

Table 1. Kingston Central Port Wind Results (knots): Maximum Likelihood Estimates and Upper Prediction Limits for Various Return Periods (1 minute sustained wind at 10 meters above ground).

The MLE (maximum likelihood estimate) column provides the best estimate as to the mostly likely extreme one minute-ten meter sustained wind for the various time frames. The MLEs are approximately equal to the "median" or 50% values (in knots). This implies that roughly speaking, the MLE is

too low about half the time and is too high the other half of the time. From a planning standpoint, one might wish to "hedge one's bets" to protect against worse than expected phenomena. The 75%, 90%, 95% and 99% columns are useful for this purpose. For example, 66.0 knots is given for the 95% upper prediction limit for the 10 year return period. Although the best guess of extreme wind in the next ten years is 57 knots, there is only a 1 in 20 chance (corresponding to the 95% level) that the extreme wind will exceed 66 knots. The difference between the 57 and 66 knot wind values reflect
the inherent uncertainty in predicting return period values.

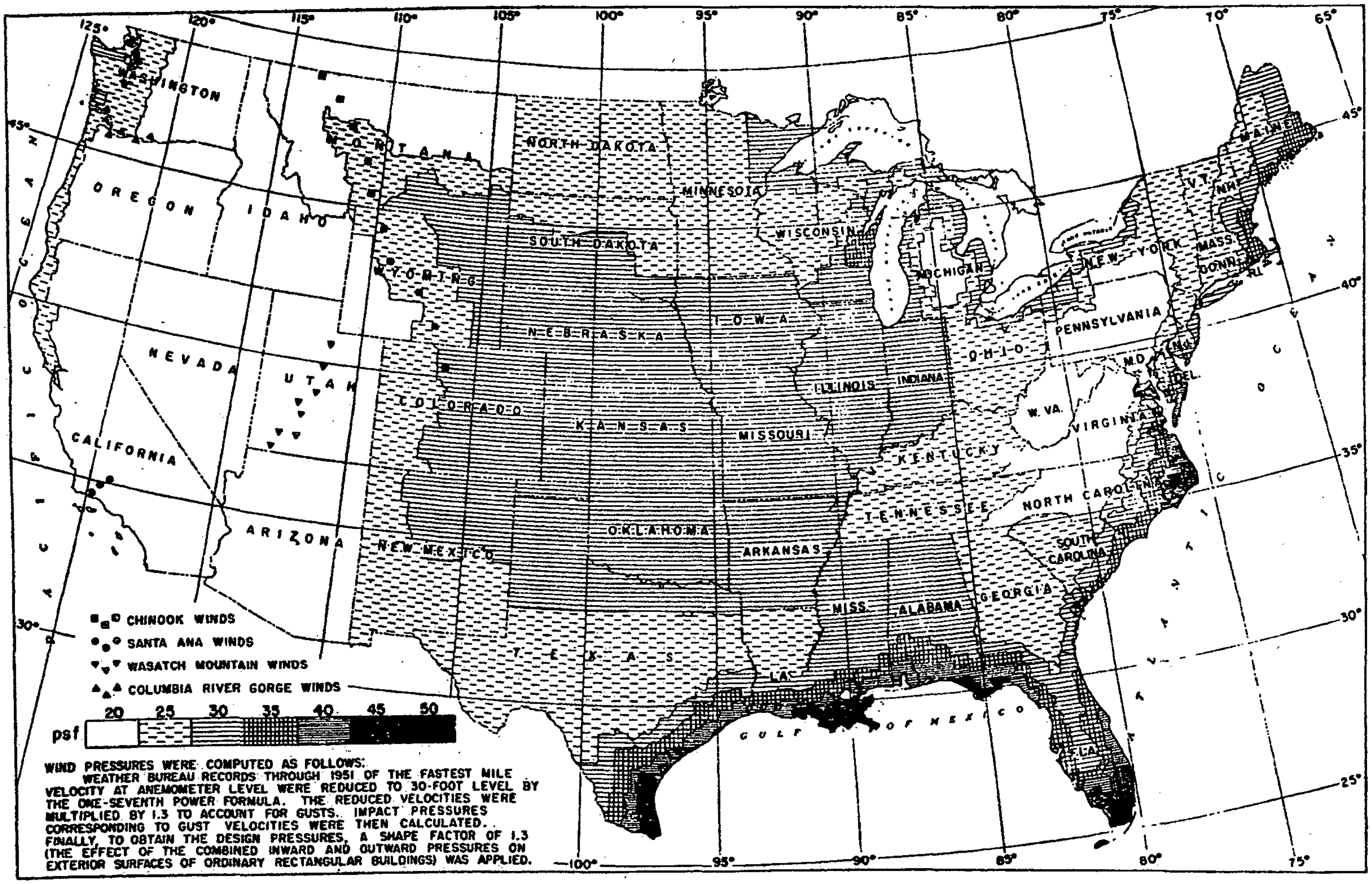


FIG. 1

Minimum Allowable Resultant Wind Pressures

See Table A7 on page 30

Appendix C
Wind Data (Question 11)

TABLE C6-6 PROBABILITY OF EXCEEDING DESIGN WIND SPEED DURING REFERENCE PERIOD

Annual	Reference (Exposure) Period, *n* (years)					
Probability Pa	**1**	**5**	**10**	**25**	**50**	**100**
0.04 (1/25)	0.04	0.18	0.34	0.64	0.87	0.98
0.02 (1/50)	0.02	0.10	0.18	0.40	0.64	0.87
0.01 (1/100)	0.01	0.05	0.10	0.22	0.40	0.64
0.005 (1/200)	0.005	0.02	0.05	0.10	0.22	0.39

TABLE C6-7 CONVERSION FACTORS FOR OTHER MEAN RECURRENCE INTERVALS

MRI	Peak Gust Wind Speed, V (mph) m/s		
(years)	Continental U.S.		
	V = 85–100 (38–45 m/s)	V > 100 (hurricane) (44.7 m/s)	Alaska
500	1.23	1.23	1.18
200	1.14	1.14	1.12
100	1.07	1.07	1.06
50	1.00	1.00	1.00
25	0.93	0.88	0.94
10	0.84	0.74 (76 mph min.) (33.9 m/s)	0.87
5	0.78	0.66 (70 mph min.) (31.3 m/s)	0.81

Note: Conversion factors for the column "V > 100 (hurricane)" are approximate. For the MRI = 50 as shown, the actual return period, as represented by the design wind speed map in Fig. 6-1, varies from 50 to approximately 90 years. For an MRI = 500, the conversion factor is theoretically "exact" as shown.

Attachment 2
NBSR Technical Specifications

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 10 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 10 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.

Basis

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

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

3.1.3 Core Configuration

Applicability: Core grid positions

Objective: To ensure that 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 average fission density shall not exceed 2 x 10^{27} fissions/m^3.

Basis

Fuel elements in the NBSR are burned for seven (7) or eight (8) cycles. An eight (8) cycle fuel element has an average fission density of approximately 1.9 x 10^{27} fissions/m^3. The U_3O_8 – Al dispersion MTR 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. Several measurements of swelling in fuel plates show that NBSR fuel, which is moderately loaded at 18% 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 10 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 product monitor is operable or sample analysis for fission product activity is conducted daily.

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

(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.

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, experiments are being placed in the reactor vessel or a recovery from an unplanned or unscheduled shutdown or a significant power reduction.

(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 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 Section 6.7.2 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, inspections and maintenance 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 Records

6.8.1 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

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.

(7) reactor facility radiation and contamination surveys where required by applicable regulations.

6.8.2 Records to be Retained for at Least One Operator Licensing Cycle

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 Records to be Retained for the Life of the Reactor Facility

(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.

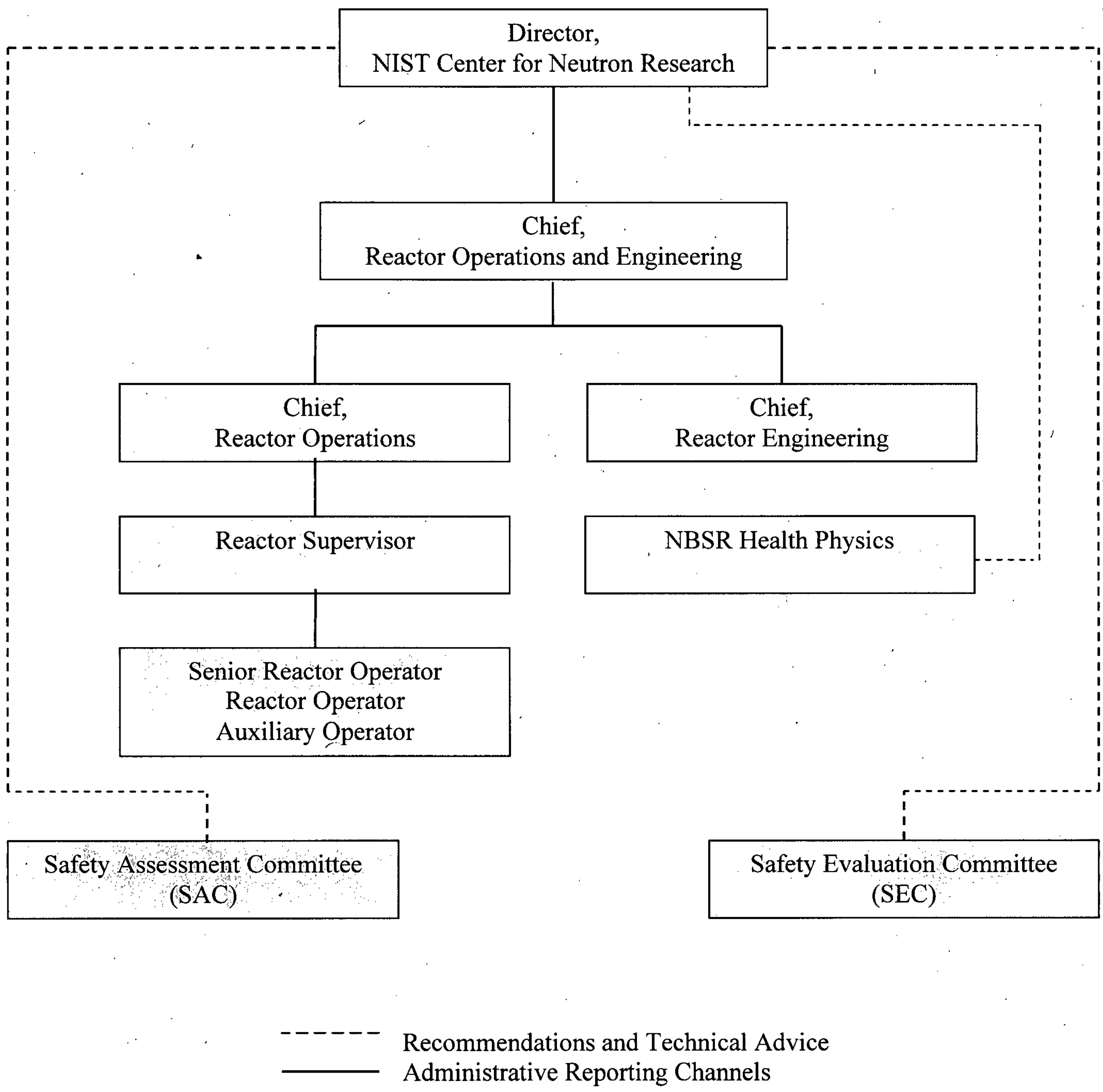

Director,
NIST Center for Neutron Research
Chief,
Reactor Operations and Engineering
Chief,
Reactor Operations
Chief,
Reactor Engineering
Reactor Supervisor
NBSR Health Physics
Senior Reactor Operator
Reactor Operator
Auxiliary Operator
Safety Assessment Committee
(SAC)
Safety Evaluation Committee
(SEC)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

Figure 6.1