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

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

August 14, 2007

Subject: Response to NRC, Request for Additional Information (RAI), dated 6/27/07
Docket Number: 50-184

Gentlemen,

Attached please find the response to the NRC RAI's dated 6/27/07. In addition to the RAI's the revised NBSR Technical Specifications are also included. If you have any further questions concerning this submittal, please contact Dr. Wade J. Richards at 301-975-6260 or wade.richards@nist.gov.

Sincerely,

Patrick D. Gallagher
Director, NIST Center for Neutron Research

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

Executed on: August 14, 2007 by: ____________

cc:
U.S. Nuclear Regulatory Commission
ATTN: Mr. Marvin Mendonca
One White Flint North, MS 012-D3
11555 Rockville Pike
Rockville MD. 20852-2738

NIST

A020
NRR

NIST Response to Second Round of NRC
Request for Additional Information
Dated 6/27/07

2.1 Provide a reference to the mentioned previous response or a copy of the response.

2.2 Provide a reference to the mentioned previous response or a copy of the response.

defined by the "fastest mile" method. This method was based on the early 20th century technology of mechanical rotating anemometers that measured the number of miles passing the anemometer. In the 1980's the National Weather Service modernized its airport weather systems with microprocessor based sensors and systems that could be programmed to measure and display wind in more conventional and universally acceptable ways. This has led the American Society of Civil Engineers (ASCE) and US Army Corps of Engineers to revise their published wind load standards and methods using the 3-second gust values.

**ASCE 7-93** and earlier versions used the fastest mile speed. The "fastest mile" wind speed, which is the average speed obtained during the passage of one mile of wind. At an average speed of 60 mph, one mile of wind passes in one minute, so in this case it represents a 60 second wind speed average.

**ASCE 7-95** and later versions use the 3-second gust method as the basis for its 50-year return maximum wind zones. The 3-second peak gust is the highest sustained wind speed averaged over a three second period of time.

What is the difference between fastest mile wind speed and three-second peak gust wind speed?
A 3-second peak gust is typically 20%-25% higher than fastest mile method. Building codes and standards have converted to three-second peak gust wind speeds because that means of measurement is now commonly used at wind speed reporting stations across the U.S. As an example of the difference, using the ASCE 7-93 standard, a 70 mph fastest mile would be converted to ~88 mph using the 3-second gust method. This correlates well with the published standards of previous 70 mph and new 90 mph 50-year return winds for central Maryland.

Historically, multi year return period wind speeds are calculated on a 50-year return period. The most recent standard ASCE 7-98, states that the maximum 50-year return period 3-second wind speed gust for Montgomery County, Maryland is 90 miles per hour (mph). The old standard [Out of Print] ASCE 7-93, states that the maximum 50-year return period fastest mile wind velocity for Montgomery County is 70 mph. [Out of Print] For consistency with the climate study submitted by NIST with the original NRC application, this new study will use the "fastest mile" method.

For reference, Figure 2.3.1-3a illustrates the ASCE fastest mile and 50-year return wind zones for Central Maryland as 70 mph. Figure 2.3.1-3b illustrates the ASCE 7-98 3-second peak gust and 50-year return wind zones Central Maryland as 90 mph.

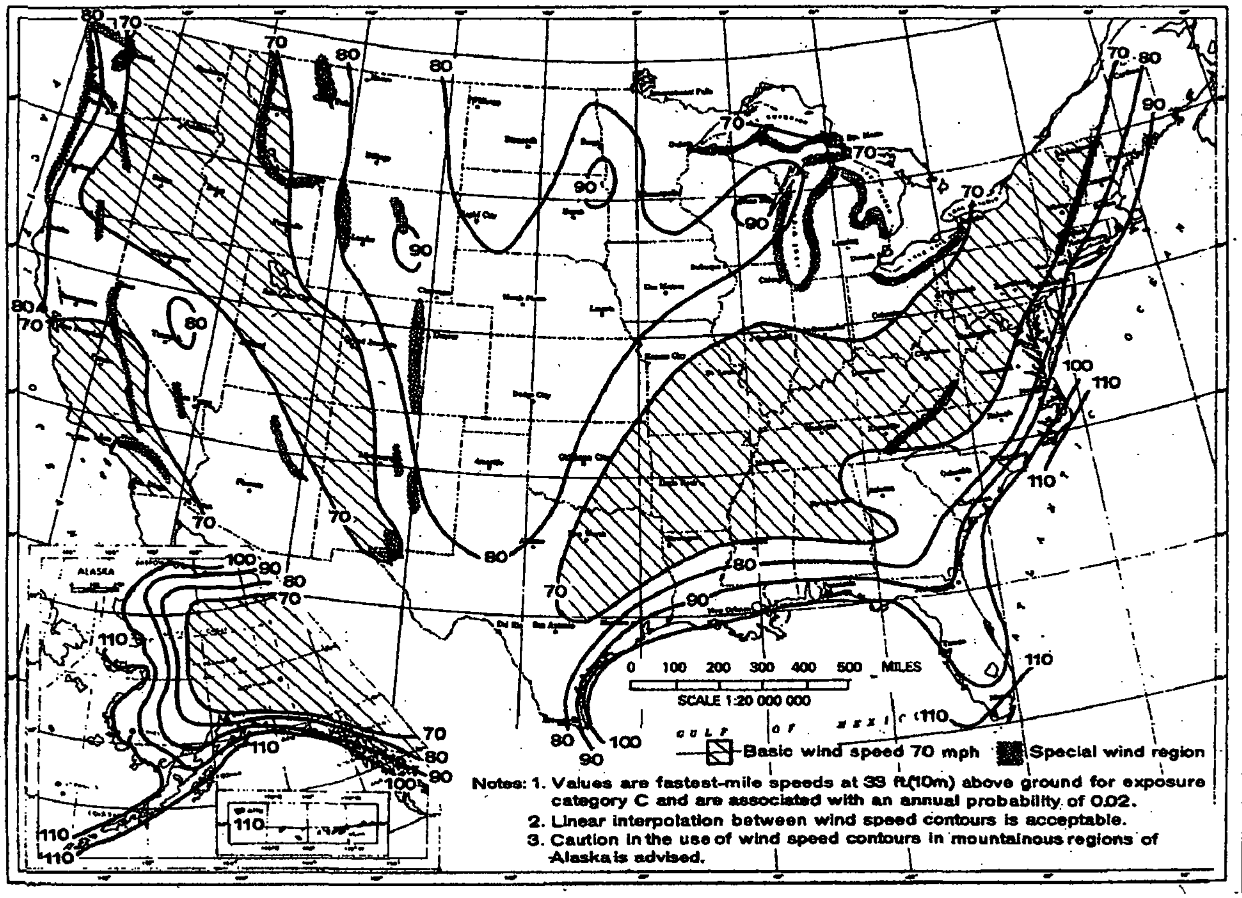
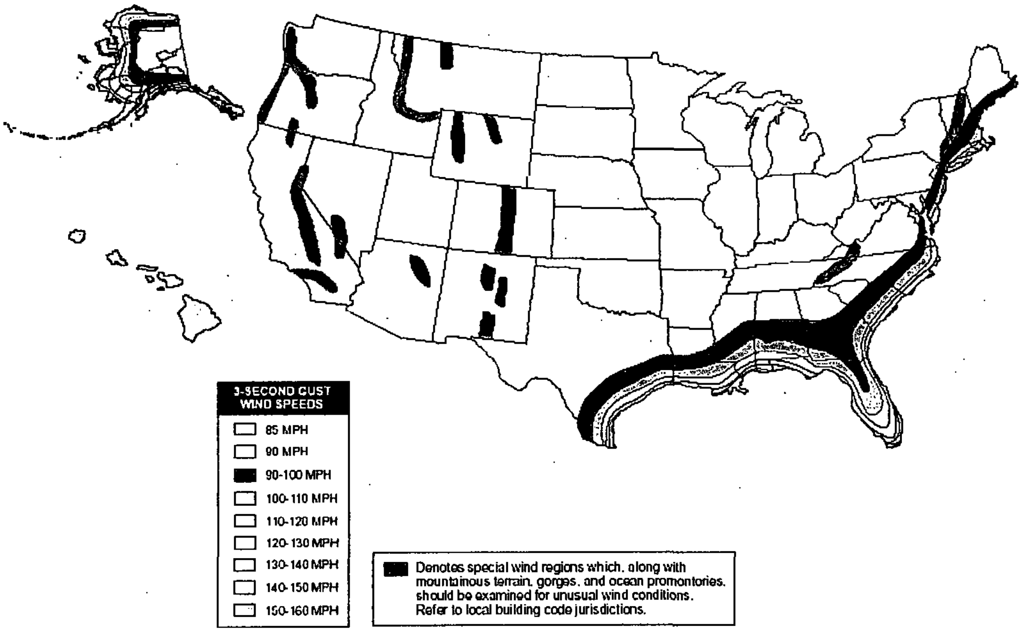



Figure 2.3.1-3a ASCE 7-93 50-Year Wind Zones




*Reproduced from ASCE 7-98, by the American Society of Civil Engineers, copyright 2000, with the permission of the publisher.*

Figure 2.3.1-3b ASCE 7-98 50-Year Wind Zones

This climate study requires a term estimation of the 100-year return wind speed so additional calculations and estimations are required. Conversion of the 50-year peak return wind to the 100-year peak return wind as required by NRC is made using several published studies identified below:

| Published Source | 50-Yr to 100-Yr Return Period Multiplier | x | ASCE 7-93 50-YR Fastest Mile Return Period Wind | = | Estimated 100-YR Return Period Peak Wind |
|---|---|---|---|---|---|
| Hurricane Hazard Information for Coastal Construction [6] | 1.134 | x | 70 mph | = | 79.4 mph |
| Return Period of Hurricane Perils...[7] | 1.146 | x | 70 mph | = | 80.2 mph |
| | | | Average... | | 79.8 mph |
| **Published Source** | **50-Yr to 100-Yr Return Period Multiplier** | **x** | **ASCE 7-98 50-YR 3-Second Return Period Wind** | **=** | **Estimated 100-YR Return Period Peak Wind** |
| Hurricane Hazard Information for Coastal Construction | 1.134 | x | 90 mph | = | 102 mph |

| | | | | | |
|---|---|---|---|---|---|
| [6] | | | | | |
| Return Period of Hurricane Perils…[7] | 1.146 | x | 90 mph | = | 103 mph |
| | | | | Average… | 102.5 mph |

*Based on the results of these calculations, we estimate that the 100-year return peak wind for Montgomery County is 79.8 mph. The use of the lower fastest mile value is justified for the calculation because the original NRC application also used the fastest mile method.*

## 100-Year Return Period Snowpack

Ground snow loads are generally calculated on a 50-year return period. The most recent American Society of Civil Engineers (ASCE) standard, ASCE 7-98, states that the ground snow load for Montgomery Count is 25 pounds per square foot (psf). [5] This climate study requires a more complete estimation, namely the weight of the 100-year return period snowpack and the 48-hour probable maximum precipitation at the site. First, the weight of the 100-year snow pack will be determined. The following tables show the 1-day, 2-day, and 3-day 100-year return estimate snowfalls for IAD, DCA, and Rockville.

**Rockville - Based on data from 1948-2000**

| Time Frame | Snowfall Amount (Inches) Return Period 10-Yr | 25-Yr | 50-Yr | 100-Yr | Observed Max |
|---|---|---|---|---|---|
| 1-day | 11.3 | 15.3 | 18.5 | 22.0 | 19.3 |
| 2-day | 13.1 | 18.0 | 22.3 | 27.0 | 25.7 |
| 3-day | 13.6 | 18.5 | 22.5 | 26.9 | 28.3 |

**IAD (Dulles AP) - Based on data from 1963-2000**

| Time Frame | Snowfall Amount (Inches) Return Period 10-Yr | 25-Yr | 50-Yr | 100-Yr | Observed Max |
|---|---|---|---|---|---|
| 1-day | 13.0 | 17.0 | 20.3 | 23.7 | 22.5 |
| 2-day | 15.0 | 19.3 | 22.6 | 26.1 | 23.2 |
| 3-day | 15.3 | 19.4 | 22.6 | 25.9 | 24.6 |

**DCA (Reagan National AP) - Based on data from 1949-2000**

| Time Frame | Snowfall Amount (Inches) Return Period 10-Yr | 25-Yr | 50-Yr | 100-Yr | Observed Max |
|---|---|---|---|---|---|
| 1-day | 10.2 | 13.5 | 16.4 | 19.4 | 16.4 |
| 2-day | 11.5 | 15.4 | 18.6 | 22.1 | 18.7 |
| 3-day | 12.0 | 15.6 | 18.4 | 21.4 | 18.7 |

It must be noted that snowfall and snow pack are not the same thing. Snowfall is the amount of snow that accumulates during an event and is measured in inches. Snowpack is the weight of the snow as it lies on the ground. As shown in the Rockville table above, the 2-day total 100-year snowfall data for Rockville is 27 inches. To convert the snow accumulation to snowpack, we must make an estimation of the snow water equivalent (SWE) and then convert SWE to snow load. SWE is the amount of water contained within the snow and is related to the snow depth and density. Snow density varies significantly depending on air temperature and wind velocity. According to the American Meteorological Society, the density of freshly fallen snow can range from 0.07 – 0.15. [8]

Applying the known 2-day 100-year return snowfall and typical snow densities to the formulas below, we can calculate the snow load in pounds per square foot.

SWE = Snow Depth * Snow Density

Snow Load = SWE * 5.2

| 2-Day 100-Yr Snow (inches) | * | Density | = | SWE (inches) | x | 5.2 (Conversion Factor) | = | Snow Load (psf) |
|---|---|---|---|---|---|---|---|---|
| 27 | * | 0.05 | = | 1.35 | x | 5.2 | = | 7.0 |
| 27 | * | 0.1 | = | 2.7 | x | 5.2 | = | 14.0 |
| 27 | * | 0.15 | = | 4.1 | x | 5.2 | = | 21.1 |

As shown in the table, the worst-case snow load occurs when the snow is very dense. Based on the results of these calculations, we estimate the 100-year return period ground snow load for Montgomery County is 21.1 psf, less than the published ASCE 7-98 50-year return estimate of 25 psf.

NRC calculations of snow load must also account for the "rain-on-snow" surcharge that occurs when a significant rain event occurs on the existing snowpack. The maximum 24-hour precipitation that has occurred during the winter months in Rockville is shown in the table below.

**<u>Rockville - </u>Maximum Daily Precipitation from 1948 through 1998.**

| Month | Maximum Daily Precipitation | Most Recent Date of Occurrence |
|---|---|---|
| January | 2.42 | 1/1/76 |
| February | 1.92 | 2/12/85 |
| March | 2.75 | 3/23/91 |
| April | 2.20 | 4/14/70 |
| May | 3.15 | 5/5/89 |
| June | 7.90 | 6/22/72 |
| July | 4.32 | 7/9/58 |
| August | 4.50 | 8/1/78 |
| September | 4.46 | 9/26/75 |
| October | 4.36 | 10/23/90 |
| November | 3.50 | 11/27/93 |
| December | 2.28 | 12/24/86 |
| Winter Average | 2.34 | Calculated |

For the months of December through March, the average maximum daily precipitation is 2.34 inches SWE. Note that the table of precipitation values includes the SWE of the accumulation of rain, snow, and sleet and therefore overestimates what just fell as rain. To compensate for this, we have averaged the 4 monthly maximum precipitations and make the estimate that 50% of the precipitation is in the form of snow and 50% is in the form of rain. Because we have already calculated the 100-year snow load, we therefore consider the rain load to be 2.34 / 2 = 1.17 inches SWE. Converting the SWE to water load uses the same formula as used in the snow load calculations above.

$$
\begin{aligned}
\text{Water Load} &= \text{SWE} * 5.2 \\
&= 1.17 \text{ inches} * 5.2 \\
&= 6.1 \text{ psf}
\end{aligned}
$$

Based on the results of these calculations, we estimate the worst-case 100-year return period ground snow load with "rain-on-snow" surcharge for Montgomery County is 21.1 psf + 6.1 psf = 27.2 psf.

Calculation of roof load for a flat roofed building depends on several factors as described below. ASCE 7-1998 [5] provides the following formula to convert ground snow load to roof snow load:

$$pf = 0.7\, C_e\, C_t\, I\, p_g$$

where: $C_e$ = Exposure Factor
$C_t$ = Thermal Factor
I = Importance Factor
$p_g$ = Ground Snow Load

$C_e$ = Exposure Factor = 1.0 Rationale – based on "partially exposed terrain with urban and suburban areas".

$C_t$ = Thermal Factor = 1.0 Rationale - based on "All structures except structures kept just above freezing, unheated structures, and continuously heated greenhouses…"

I = Importance Factor = 1.2 Rationale - based on Category III, "Structures containing highly toxic materials… where the quantity of the material exceeds the exempt."

$p_g$ = Ground Snow Load = 27.2 psf Rationale - From calculations above.

$$pf = 0.7*1.0*1.0*1.2*27.2 = 22.8 \text{ psf}$$

*Based on the results of these calculations, we estimate the NIST Containment Building roof snow load with 100-year return period rain-on-snow surcharge is 22.8 psf.*

## Appendix A - Bibliography and Data Sources with Applicable URL's

[8] Glossary of Meteorology, Ralph E. Huschke, Ed, American Meteorological Society, Boston, MA, 1959
http://amsglossary.allenpress.com/glossary/

4.2 Provide revised Figures 4.2.3 and 4.2.4, which were not attached to the response.

NIST Response to 4.2

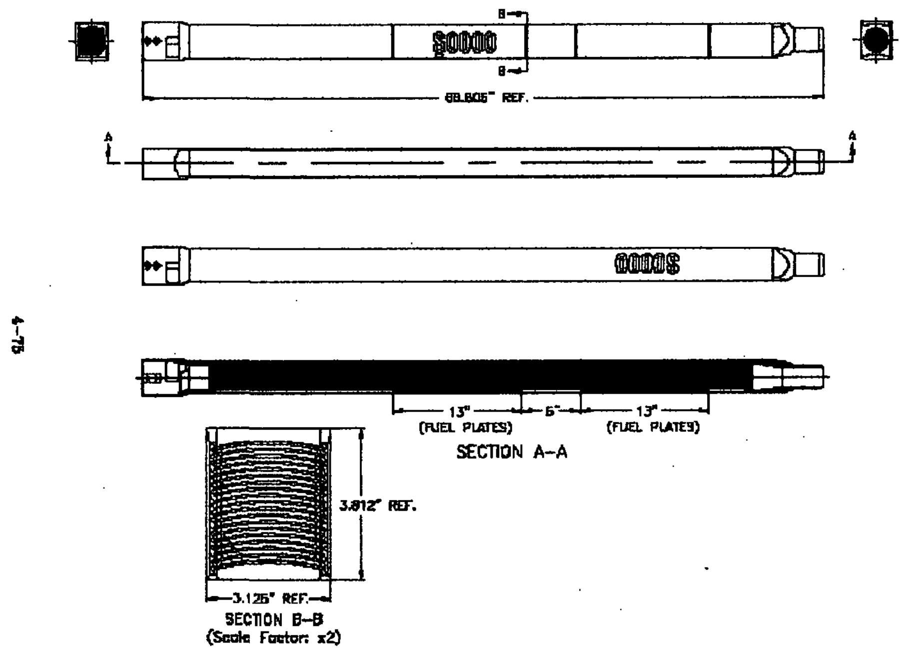



Figure 4.2.3: Fuel Element Assembly (see Figure 4.2.4 for Fuel Plate Detail)

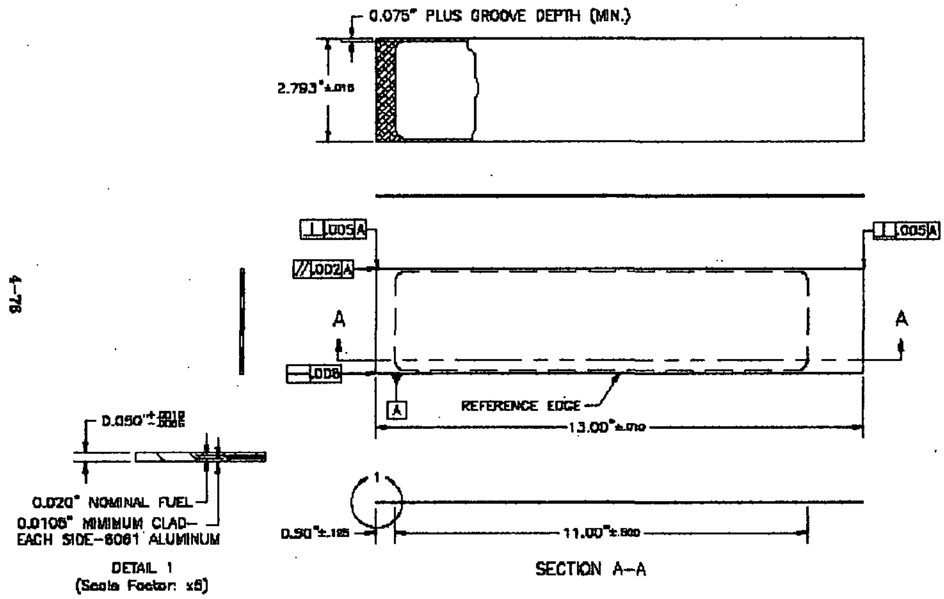



**Figure 4.2.4: Typical Top and Bottom Flat Fuel Plate**

4.9 Provide justification for the conclusion that there are no lifetime issues regarding the strength of the Al or the burnup of the Cd used in the poison tubes. The neutron flux and duration of exposure should be compared to design limits for the tubes to ensure the tubes will continue to function as designed under the most limiting conditions for the license renewal period.

<u>NIST Response to 4.9</u>

An analysis of the strength of the poisoned hold down tubes against buckling (see response to RAI 4.28, below) shows that failure of a hold-down tube is not a credible accident. In addition, neither the Compressive Yield Strength nor Young's Modulus decrease upon neutron irradiation, so there is no weakening of the tubes under irradiation.

Our earlier response did not make it clear that the poisoned hold-down tubes mate with the vertical thimbles at the top grid plate, 21 in (54 cm) above the top of the core. In 1994, MCNP calculations were performed to assess the impact of removing the layer of boral from the bottom of the refueling plug. In the simulations, the thermal neutron flux was tallied in 10-cm thick cells, from the top of the top grid plate to the top of the $D_2O$ level, and beyond, into the plug. The flux drops about a factor of $10^4$ through this 2.5 m of $D_2O$. The maximum thermal neutron flux just above the top grid plate is about $8x10^{12}$ n/cm$^2$-s, but still a factor of 20 less than the flux at the beam tube tips. The flux is irrelevant, however, as the tubes are not weakened by neutron irradiation.

The bottom of the Cd layer in the poisoned tubes is 2 feet (61 cm) above the top grid plate. The thermal neutron flux at the bottom of the Cd is about $1x10^{12}$ n/cm$^2$-s, more than 2 orders of magnitude less than the flux in the core. Since the Cd in the shim arms has about a 4 year life span, we conclude the Cd in the poisoned tubes, which has the same thickness, is sufficient for the life of the facility.

4.25 Provide an update to Table 4.2.3. Alternatively, remove or commit to remove Table 4.2.3 from the SAR, if the SAR contains all necessary information elsewhere.

NIST Response to 4.25
Table 4.2.3 is indeed out of date. It will be removed from Chapter 4 along with all references to it.

4.28 Provide or reference analysis to support the conclusion that there would be no fuel damage from the decreased flow through the fuel elements cooled by the inner plenum. Also, the response should discuss how poison tube buckling would be detected and what actions would be required as these considerations may affect the consequences of such an accident.

NIST Response to 4.28
A 33% reduction in flow was analyzed in Appendix A (loss of a primary pump) and it was shown that the MCHFR in the inner plenum was 3.6. Thus, there would be no fuel damage if there was a 25% reduction. A hold-down tube buckling accident, however, is shown below to not be credible.

A buckling analysis has recently been performed for the Cd containing section of a poisoned hold-down tube (J. Michael Rowe, Memo, "Failure of the Vertical Thimble Hold-Down Tubes", July 7, 2007). The conservative analysis considered three possible failure modes: Euler Buckling (bending), Local Buckling (crumpling), and Compressive Yield failure (deformation). The forces required for failure were calculated for each case for the thinnest part of a hold-down tube, namely the top section above the Cd layer, which has a 4.485 in OD, and a 4.326 in ID. The force required for compressive deformation was lower than the force necessary for either Euler or Local Buckling. The compressive force is the product of the Compressive Yield Strength, $2x10^4$ psi, and the cross sectional area, 1.1 $in^2$, or $2.2x10^4$ lb. For Euler Buckling the force required is $3.5x10^4$ lb, assuming the weakest section extends the entire length of the tube, while the force needed for Local Buckling is $1.1x10^5$ lb. Neither the Compressive Yield Strength nor Young's Modulus, used in the Buckling calculations, decrease as a result of neutron irradiation.

The maximum possible upward force on a thimble is given conservatively as the primary pump discharge pressure times the flow area blocked, F = (65 psi)(4.4 $in^2$)* = 290 lb, nearly 80 times less than the force required for compressive deformation of the tube at its weakest point. Since neutron irradiation cannot weaken the tubes, we conclude that a poisoned hold-down tube buckling accident is not credible.

Since failure of a poisoned hold-down tube is not credible, it is not necessary to speculate further as to how it would be detected or how the flow would be affected.

*The statement in the previous response to RAI 4.28 that the central thimble has a larger diameter is incorrect. All of the thimbles have nozzles of 2.375 in OD.

4.35 Provide the alloy or alloys, which compose the core frames and cladding.

NIST Response to 4.35
The alloy for the frames and covers is Alloy 6061 - Temper "O". Ambiguities in the SAR as to the exact temper of the frames and covers will be eliminated.

4.36 Provide an update to Table 4.2.3. Alternatively, remove or commit to remove Table 4.2.3 from the SAR, if the SAR contains all necessary information elsewhere.

NIST Response to 4.36
4.36 Table 4.2.3 is indeed out of date. It will be removed from Chapter 4 along with all references to it.

4.39 Provide the pressure of the helium left in the voids in the shim arms.

NIST Response to 4.39
Fabrication specifications for the shim arms state that the voids are purged of air and then helium leak tested at 5 psig. The voids are then sealed "maintaining one atmosphere of helium inside the shim arm", and tested again.

5.7 Provide verification that the nominal air supply pressure in the response is capable of operating the valves, and that the larger pressure is not required to operate the valves.

NIST Response to 5.7
Air at a nominal pressure of 100 psig (690 kPa) meets the design requirements for operation of these pneumatically operated valves or has been empirically determined to be sufficient for operation of these pneumatically operated valves. References to air pressures for these valves will be changed to 100 psig (690 kPA).

5.12 Provide clarification on the temperature range for TR-1 in degrees C. Verify that TR-1 is a temperature difference instrument. Verify that the SAR gives the correct temperature range for TR-1. Note TR-1 was not included in the RAI.

NIST Response to 5.12

TR-1 is a differential temperature channel. The nominal range for TR-1 is 0-20°F (0-11°C). This is the range that will be used for TR-1 where applicable.

13.9 For each accident analysis, provide the limiting assumptions, conditions and safety system settings and where these limiting assumptions, conditions and safety system settings are required by Technical Specifications as required by 10CFR50.36. Compare the assumptions, conditions and safety system settings to those in ANSI 15.1 and NUREG 1537, which are applicable to test reactors. The response should provide a distinct set of information regarding accident scenarios so that a determination can be made that each accident scenario assumed the most conservative conditions allowed by the Technical Specifications. The information should show that the TS limits are supported by the appropriate accident analyses.

NIST Response to 13.9

In Chapter 13, several accidents are analyzed and all except the MHA, which was expressly postulated to cause fuel damage with no credible initiating event, are shown to cause no fuel damage. This is the basis for **Technical Specification (TS) 2.1 which establishes as the safety limit** that the fuel cladding temperature shall not exceed 450 °C, which is the minimum temperature at which fuel blistering has been observed. In each case, it is sufficient to ensure that neither a Departure from Nucleate Boiling (DNB) or Onset of Flow Instability (OFI) occurs. This TS is at the heart of all analyses except for the MHA. In what follows, the TS relevant to each accident analyzed is presented and analyzed for relevance to the bases of the Technical Specifications[1] and to the conditions assumed in any mitigating action taken. The initial conditions for each accident analyzed (except for the MHA for which the initial reactor conditions are irrelevant) are given below.

[1] The numbering scheme used is that for the revised Technical Specifications submitted in July, 2007.

In every case, RELAP analyses were conducted for very conservative conditions, with the reactor power, coolant level, inlet temperature and flow at the limit of their normal operating range. These conditions are[2]:

| Parameter | Limit | Value |
|---|---|---|
| Reactor Power | 102 % of Nominal Rating | 20.4 MW |
| Reactor $D_2O$ Level | Low | 3.81 m (150 in.) |
| Core Inlet Temperature | High | 43.3 ° C (110 °F) |
| Main Primary Coolant Flow | Low | 549 l/s (8700 gpm) |

Also, the hot channel (the one with limiting gap, and hence flow area) was used to derive the CHFR, rather than the normal channel.

### 13.2.1 Maximum Hypothetical Accident

Limiting Assumptions:

A complete flow blockage of one element is assumed, with no credible initiating event.
The entire fission product inventory from this element is assumed to be released to the primary water immediately.

Technical Specifications

TS 3.2.2 specifies that the reactor will scram on high effluent air, and this is used to terminate reactor operation.
TS 3.4.1 specifies that confinement must be present when the reactor is operating, and this is assumed for calculation of consequences.
TS 3.5 specifies that ventilation (normal and emergency) must be operational, including both effluent and recirculation filters, and this is assumed for calculation of consequences.

[2] See NBSR-14, Appendix A, Table 4.5

TS 5.1 establishes a 400 m exclusion zone around the reactor, and this provides the boundary that excludes the public, and therefore the distance at which public doses are calculated.

For this accident the Technical Specifications above provide assurance that the mitigating factors assumed are present. The analysis shows that that the doses are less than those specified in 10CFR part 100.

### 13.2.2 Insertion of Excess Reactivity

Technical Specifications

TS 2.2 specifies the Limiting Safety System Setting at which reactor power scrams are calculated as the terminating event for all accidents of this type.
TS 3.1.3 specifies the core configuration.
TS 3.2.1 specifies the shim insertion times used in the RELAP model, and also specifies the maximum allowed rate of reactivity insertion.

These three Technical Specifications are common to the reactivity insertion accidents, and the results show that the TS provide assurance that the safety limit will not be exceeded.

#### 13.2.2.1 Step reactivity Insertion

Limiting Assumptions

None (No credible initiating scenario)

Technical Specifications

TS 3.1.3 provides assurance that there are no empty core positions into which an element could fall accidentally.

### 13.3.2.2 Ramp Reactivity Insertion

Limiting Assumptions

The startup accident assumes that reactivity is inserted at the limit specified in TS 3.2.1, starting from 100 W and continuing until the reactor scrams at the LSSS of 26 MW.
The period scram active below 2 MW is assumed not to operate.
The rod withdraw prohibit function is assumed not to function.
The rapid Removal of Experiments accident assumes that an experiment having the maximum reactivity of 0.5 % is withdrawn in 0.5 s.

Technical Specifications

TS 3.2.1 provides assurance that the shim arms will operate as designed after a scram is received, limits the rate of reactivity insertion by shim arm withdrawal, and provides assurance that the shim insertion time will no longer than that assumed in the RELAP analysis.
TS 3.8.1 limits the worth of any single experiment to 0.5 %, which provides assurance that the insertion considered is bounding.
The analysis shows that the limits specified provide assurance that the Safety Limit will not be exceeded. This analysis also provides the basis for the Technical Specification limits.

### 13.2.3 Loss of Primary Coolant

Limiting Assumptions

A major rupture in the cold leg of the primary system is assumed, which leads to draining the reactor core.

Technical Specifications

TS 3.3.3 provides assurance that emergency cooling is available as assumed.
TS 3.3.2 establishes a limit of 5 Ci/liter on tritium concentrations in the primary.

TS 3.4.1 ensures that confinement is established when the reactor is operating as assumed.
TS 3.5 ensures that emergency ventilation is available when the reactor is operating as assumed.
The analysis provides assurance that the limits established will protect the public and ensure that the accident is bounded by the MHA.

### 13.2.4 Loss of Primary Coolant Flow

Limiting Assumptions

The off-site power loss scenario assumes that all three primary pumps coast down, and that a low flow trip occurs 400 ms after the low flow condition is reached.
The seizure of one pump scenario assumes that the flow through one pump stops instantaneously and that a scram occurs at the normal setpoint (note that the flow will not drop to the LSSS, and thus that the CHFR will exceed 2 even if no scram occurs).
The throttling of flow to either the inner or outer plenums assumes that valves are fully closed, and that the reactor scrams on low flow 400 ms after the low flow condition is met.
The loss of both shutdown cooling pumps assumes the off-site power loss scenario, plus failure of both shutdown pumps. It is further assumed that shutdown cooling can be restored in the several hours before the entire vessel warms up to the boiling point.

Technical Specifications

TS 3.3.1 provides assurance that at least one shutdown pump is operable, as is assumed for all analyses except for the fifth scenario, in which the shutdown cooling is assumed to fail.
TS 2.2 establishes the LSSS for reactor flow scrams, and the RELAP analysis uses these levels, which are shown to result in no fuel damage.

TS 3.6 assures that emergency power is available to provide shutdown cooling following a loss of off-Site power.
All accidents are shown to result in no fuel damage, providing the basis for the Technical Specifications.

### 13.2.5 Mishandling or Malfunction of Fuel

Limiting Assumptions

It is assumed that a fresh fuel element is placed sequentially in every possible position in the core, in spite of all precautions to prevent this occurrence.

Technical Specifications

TS 3.1.3 controls core configuration.
TS 3.9.1 provides assurance that fuel shall not be stored so that $k_{eff}$ can exceed and will be not be handled in air without adequate cool down time.
TS 3.9.3 provides assurance that all fuel is properly latched into the grid plate before operation.

### 13.2.6 Experiment Malfunction

Limiting Assumptions

None beyond the Technical Specifications.

Technical Specifications

TS 3.8.1 requires that the failure of single experiment not affect any other experiment, and that no reactor excursion shall cause an experiment to fail in a manner that could affect an accident.
TS 3.8.2 specifies the requirements on materials used in experiments to provide assurance that they will not damage reactor systems.
TS 4.8 specifies the review needed before an experiment can be installed.

These specifications provide assurance that no experiment can cause or contribute to an accident that could cause a violation of the Safety Limit.

### 13.2.7 Loss of Normal Power

Limiting Assumptions

None; this accident is bounded by the loss-of-flow caused by loss of off-site power accident analyzed in the loss of primary coolant flow accidents.

Technical Specifications

This accident is bounded by the loss-of-flow caused by loss of off-site power accident analyzed in the loss of primary coolant flow accidents, and the Technical Specifications given there apply here.

### 13.2.8 External Events

Limiting Assumptions

It is assumed that the plant will be shut down upon sighting of a tornado, as specified in the Emergency Plan.

Technical Specifications

There are no Technical Specifications relating to external events other than the safety limit. Actions are specified in the Emergency Plan in the event of a tornado.

### Conclusions

In summary, the analyses given in this chapter show that adherence to the Technical Specifications provide assurance that no reactor accident will lead to fuel damage, except for the MHA, which is postulated to do so. These analyses support the bases for the

Technical Specifications, as do the thermal hydraulic limits specified in Chapter 4 which established the LSSS. The Chapter 13 results show that the LSSS determined for routine operation are also adequate to provide assurance that the safety limit will not be exceeded during any credible accident.

# Appendix A

# License No. TR-5

# Technical Specifications
# For the
# NIST 20 Megawatt 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 the 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- 2004.

### 1.3 Definitions

The following terms are sufficiently important to be separately defined:

1.3.1 ALARA
As defined in 10 CFR, 20.1003, As Low As Reasonably Achievable.

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 <u>Channel Calibration</u>

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 <u>Channel Check</u>

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 <u>Channel Test</u>

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

1.3.3 <u>Confinement</u>

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

1.3.4 <u>Core Configuration</u>

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

1.3.5 <u>Excess Reactivity</u>

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 exactly critical (i.e. $k_{eff} = 1$).

1.3.6 <u>Emergency Director</u>

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

1.3.7 <u>Experiment</u>

1.3.7.1 <u>In-Reactor Vessel</u>

Any operation, hardware, or target (excluding devices such as detectors, foils, etc.), 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.

1.3.7.1 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

1.3.7.2 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 position, material or 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
The Moderator Dump 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
Natural Convection Cooling is that flow of primary water from the reactor vessel to the heat exchanger with no pumps operating.

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

1.3.13 Operating
A component or system 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 reactor is operating whenever it is not secured or shutdown.

1.3.16 Reactor Operator

Individual who is licensed to manipulate the controls of a reactor.

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** are true.

The Console Power key switch or the Shim Arm Drive Power key switch is in the off position with the key removed from the lock and under the control of a licensed operator.

a. (1) The minimum number of shim arms are fully inserted to ensure the reactor is shutdown, as required by technical specifications; and
(2) 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
(3) 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 if it is subcritical by at least one dollar ($1.00) both in the Reference Core Condition and for all allowed ambient conditions with the reactivity worth of all installed experiments included.

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 (cold T<20 °C) and the reactivity worth of xenon is negligible (<$0.30) (i.e., cold and clean).

1.3.22 Reactor Rundown

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

1.3.23 Rod-Control

A device, also know 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 Regulating Rod

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.

1.3.25 Scram

A scram is the spring assisted gravity insertion of all shim arms.

1.3.24.1 Major Scram

A reactor scram accompanied by the immediate activation of the confinement isolation system.

1.3.26 Scram Time

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

1.3.27 Senior Reactor Operator

An individual who is licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

1.3.28 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.

1.3.29 Shutdown Margin

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

1.3.30 Surveillance Activities

Those tests, checks and calibrations done to predict the operability of the equipment in section 4.0 of this Technical Specification.

1.3.31 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 interval 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 shall not exceed the following:

1.3.31.1 Five Year

Interval not to exceed six years.

1.3.31.2 Biennial

Interval not to exceed two and half years.

1.3.31.3 Annual

Interval not to exceed 15 months.

1.3.31.4 Semiannual

Interval not to exceed seven and a half months.

1.3.31.5 Quarterly

Interval not to exceed four months.

1.3.31.6 Monthly

Interval not to exceed six weeks.

1.3.31.7 Weekly

Interval not to exceed ten days.

1.3.32 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 Setting

### 2.1 Safety Limit

Applicability:

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

Objective:

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

Specification:

The reactor fuel 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 (SAR, NBSR 14, Chapter 4, Section 4.6 ) 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:

This specification applies to the protective action for the reactor fuel element temperature.

Objective:

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

Specifications:

The Limiting Safety System Settings shall be:

| | |
|---|---|
| Reactor power* | 130 % of full power |
| Reactor outlet temperature | 147 °F (maximum) |
| Forced coolant flow | 60 gpm/MW inner plenum<br>235 gpm/MW outer plenum |
| Reactor power with natural convection cooling flow | 500 kW |

Basis:

At the values established above, the Limiting Safety System Settings provide a significant margin from the Safety Limit. Even in the extremely unlikely event that all three parameters, reactor power, coolant flow, and outlet temperature simultaneously reach their Limiting Safety System Settings, the burnout ratio is at least 2. For all other conditions the burnout ratio is considerably higher (SAR, NBSR 14, Chapter 4, Section 4.6). This will ensure that any reactor transient caused by equipment malfunction or operator error will be terminated well before the safety limits are reached. Overall uncertainties in process instrumentation have been incorporated in the Limiting Safety System Setting.

The analysis in the SAR, NBSR 14, Appendix A, Section 5.7, clearly shows that the reactor can be operated at 500 kW with reduced or no flow. In the natural circulation mode at 500 kW, the minimum critical heat flux ratio is 5.9, and the minimum OFI ratio is 16. This analysis assumes that the primary coolant is able to flow freely through the primary coolant loop (reactor, heat exchangers and pumps).

*Nominal reactor power < 20 MW for forced flow.

## 3.0 Limiting Conditions for Operations

### 3.1 Reactor Core Parameters

#### 3.1.1 Reactor Power

Applicability:

This specification applies to the reactor power.

Objective:

The objective is to assure that the reactor fuel temperature safety limit is not exceeded, and to provide for a set point for the high flux limiting safety systems, so that automatic protective action will prevent the safety limit from being reached during operation.

Specification:

The nominal reactor power shall not exceed 20 MW thermal. The reactor scram set point for the reactor power level safety channels shall be set at 125% of full power or less.

Basis:

Operational experience and thermal-hydraulic calculations (NBSR-14, Chapter 4, Section 4.6) 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, as shown in (SAR, NBSR-14, Chapter 13 and Appendix A) the operating conditions provide ample margin for all credible accident scenarios to assure that there will be no fuel damage.

#### 3.1.2 Reactivity Limitations

Applicability:

These specifications apply to the reactivity conditions of the reactor core and reactivity worth of the shim arms.

Objective:

The objective is to assure that the reactor can be placed in a shutdown condition at all times and to assure that the fuel temperature safety limit shall not be exceeded.

Specifications:

(1) Excess Reactivity – The maximum available excess reactivity (reference core conditions) shall not exceed 15% $\Delta\rho$ (~ \$20).

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

a. the reactor in any core condition, and

b. all movable experiments in their most reactive condition.

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

Basis:

(1) This specification sets an overall reactivity limit which provides adequate excess reactivity to override the xenon buildup, and to overcome the temperature change in going from zero power to 20 MW. In addition, the maximum reactivity insertion accident at startup, which assumes the insertion of 0.5% $\Delta\rho$ into a just critical core, is not affected by the total core excess reactivity.

(2) & (3) These specifications assure 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. By maintaining a shutdown margin of \$1.00, with all experiments in their most reactive condition the reactor will remain subcritical with no further operator action.

3.1.3 Core Configuration

Applicability:

The specification applies to the core grid positions.

Objective:

The objective is to ensure that the core grid positions are correctly filled.

Specification:

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

Basis:

The NBSR employs shim safety arm stops to prevent a broken shim arm from dropping from the reactor core. The proper operation of these stops depends on adjacent fuel elements or experimental thimbles being in place to prevent the broken 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:

This specification applies to fuel burnup.

Objective:

The objective is to establish the maximum allowable fission density in the NBSR fuel.

Specification:

The average fission density shall not exceed $2 \times 10^{27}$ fissions/m$^3$ which will result in a fuel burnup of 73%.

Basis:

The $U_3O_8$ – Al dispersion fuels have been in widespread use for over 40 years. Extensive testing of fuel plates has been performed to determine the limits on fission density as a function of fuel loading. Fuel elements in the NBSR are burned for 7 or 8 cycles. With a burnup of 73% in the 8-cycle fuel elements, the typical fission density is $1.9 \times 10^{27}$ fissions/m$^3$. Figure 4.2.6, Chapter 4 of NBSR-14, shows the results of several measurements of swelling in fuel plates. The NBSR fuel is moderately loaded at 18% (i.e. volume of $U_3O_8$), and the 8-cycle average fission density (i.e. $2 \times 10^{27}$ fissions/m$^3$) is well below the curve that represents the allowable limit of burnup.

3.2 Reactor Control and Safety Systems

3.2.1 Shim Arms

Applicability:

This specification applies to the reactor control and safety system operation.

Objectives:

The objective is to ensure proper operation of reactor control and safety systems.

Specifications:

The reactor shall not be operated unless:

(1) All four shim arms are operable.

(2) The scram time as measured from the instant a signal reaches the value of a safety system setting to the instant the shim arm reaches 5 degrees insertion shall not exceed 240 msec.

(3) The reactivity insertion rate, using all four shim arms, is less than or equal to $5 \times 10^{-4}\ \Delta\rho$/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) & (3) A shim arm withdrawal accident for the NBSR was analyzed (memo from A. Cuadra and L. Cheng, April, 2007) 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. The scram times shown in (SAR, NBSR-14, Appendix A, Figure 4-12) ensure that this drop time results in a reactivity insertion of sufficient magnitude to shutdown the reactor.

3.2.2 Reactor Safety System Channels

Applicability:

This specification applies to the instrumentation which shall be available to the reactor operator during reactor operation.

Objective:

The objective is to specify the minimum number of reactor safety system channels that shall be available to the operator to assure safe operation of the reactor.

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 |
|---|---|---|
| a) High Flux level | 2 | |
| b) Short period below 5% rated power | 2 | |
| c) Low reactor vessel $D_2O$ level [1, 3] | 2 | |
| d) Low flow reactor outlet [2, 3] | 1 | |
| e) Low flow reactor inner or outer plenum [2, 3] | 1 | |
| f) Manual (outside of the Control Room) | 1 | |
| g) Manual | 1 | 1 |
| h) Normal Building Exhaust Activity High[4] | | 1 |
| i) Irradiated Air Activity High[4] | | 1 |
| j) Stack Air Activity High[4] | | 1 |

[1] One (1) of two (2) channels may be bypassed for tests or during the time maintenance involving the replacement of components and modules or calibrations and repairs are actually being performed.

[2] One (1) of these two (2) flow channels may be bypassed during tests, or during the time maintenance involving the replacement of components and modules or calibrations and minor repairs are actually being performed. However, outlet low flow may not be bypassed unless both inner and outer low-flow reactor inlet safety systems are operating.

[3] May be bypassed during periods of reactor operation (up to 500 kW) when a reduction in Limiting Safety System Setting values is permitted (Specifications 2.2 and 3.3.1).

[4] May be inoperable only if the other two air monitor channels are operable.

Basis:

The nuclear and process channels of table 3.2.2 are monitored by the reactor safety system which automatically initiates protective action to ensure that

appropriate safety limit and minimum conditions for operation are not violated. With these channels operable, the safety system has redundancy.

## 3.3 Coolant System

### 3.3.1 Primary and Secondary

Applicability:

This specification applies to the reactor primary and secondary cooling systems.

Objective:

The objective is to assure that adequate cooling is provided to maintain fuel temperatures below the safety limit and to provide the means of containing $D_2O$ to $H_2O$ heat exchanger leakage.

Specifications:

The reactor shall not be operated unless the following conditions are met:

(1) At least one primary shutdown cooling pump is operable.

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

(3) Either a secondary cooling water activity monitor (i.e: $^{16}N$ monitor) or a $D_2O$ storage tank level monitor is operable.

(4) The $D_2$ concentration in the Helium Sweep System is less than 4% by volume.

Exception for specification #2: To permit periodic surveillance of the effectiveness of the moderator dump, it is necessary to operate the reactor without restriction on reactor vessel level.

Basis:

(1) The NBSR is equipped with shutdown cooling (as described in NBSR-14, Chapter 5), which provides ample cooling for all shutdown conditions. One of the accidents analyzed in (SAR, NBSR-14, Appendix A, Section 5.4) includes loss of off-site power (and hence the loss of main primary pumps), followed by failure of both redundant primary cooling shutdown pumps. With no forced shutdown flow, natural circulation in the fuel elements will result in a maximum fuel plate temperature of 279 °F (137 °C). This scenario results in

no damage to the fuel, showing that natural convection cooling is adequate to provide cooling to the fuel in the shutdown condition, even immediately following a scram due to loss of all main primary pumps. However, to ensure that fuel plate temperatures following loss of flow will be near or below normal operating temperatures, a shutdown pump shall be required.

(2) 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 about 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 is not required, such as is permitted in Section 2.2 of these specifications.

(3) In order to minimize tritium concentration in the secondary, it is essential to detect any primary to secondary system leakage immediately. A secondary cooling water activity monitor, such as a $^{16}N$ detector, would quickly detect any primary leakage. The $D_2O$ storage tank level indicator will also detect such leaks, and allow corrective action to be taken.

(4) 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.

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

3.3.2 Primary Coolant

Applicability:

This specification applies to the primary coolant.

Objective:

The objective is to maintain tritium releases as low as reasonably achievable.

Specification:

The reactor shall not be operated unless the primary coolant activity level is less than or equal to 5 Ci/l.

Basis:

At the end of the term of the NBSR license the maximum tritium concentration in the primary coolant is estimated to be 5 Ci/l. This value and reliable leak detection ensures that tritium concentrations in effluents shall be as low as practicable and below concentrations allowed by 10 CFR, Part 20.2003 for liquid effluents and 10 CFR, Part 20.1302 for gaseous effluents (SAR, NBSR 9, Addendum 1, Section 2.6.4).

### 3.3.3 Emergency Core Cooling

Applicability:

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

Objective:

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

Specification:

The reactor shall not be operated unless:

(1) The $D_2O$ emergency 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. Thus, operability of this system is a prerequisite to reactor operation.

(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

emergency core cooling 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.

#### 3.3.4 Moderator Dump System

Applicability:

This specification applies to the reactor moderator dump system.

Objective:

The objective is to ensure the reactor can be made subcritical in any core configuration.

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 is shown (NBSR-14, Chapter 4) 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:

This specification applies to the operating conditions that require the confinement system to be operable.

Objective:

The objective is to establish when confinement is required.

Specification:

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

(1) The reactor is operating.

(2) Changes of components or equipment within the confines of the thermal shield, other than shim arm 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.

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 analyzed (SAR, NBSR-14, Chapter 13). Confinement is stringently defined to ensure that the confinement building shall perform in accordance with its design basis (SAR, NBSR-14, Chapter 3). 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, which contains fission product inventories sufficient to allow Specification 3.7 to be exceeded, should the element fail, poses a potential hazard in that its cladding could be damaged when it is not contained in a closed system (e.g., during transit or during sawing of aluminum end pieces). Confinement integrity is not required, however, when irradiated fuel is contained within a closed system, such as the reactor vessel, the transfer lock of the refueling system, or the sealed shipping cask, that serves as a secondary barrier of fission product release.

3.4.2 Equipment to Achieve Confinement

Applicability:

This specification shall apply to the equipment required to achieve Confinement.

Objective:

The objective is to list the equipment that is required for Confinement to be established.

Specification:

Confinement shall mean that **all** of the following conditions have been met:

(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: In order to provide for prompt and 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) & (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) & (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:

This specification applies to the normal and emergency ventilation system.

Objectives:

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

Specifications:

The reactor shall not be operated unless:

(1) The building emergency recirculation system and emergency exhaust systems are operable, including both fans, each with at least one operable motor 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: In order to provide time for prompt remedial action, reactor ventilation may be inoperable for a period of no greater than 15 minutes when specifications 1-2, 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 (SAR, NBSR 14, Chapter 13). 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 at least 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. Gaseous effluent monitors are required by Specification 3.2.2 of these specifications.

## 3.6 Emergency Power System

Applicability:

This specification applies to the emergency electrical power supplies.

Objectives:

The objective is to ensure emergency power for vital equipment.

Specifications:

The reactor shall not be operated unless:

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

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

Exception: In order to provide time for prompt remedial action, the Emergency Power may be inoperable for a period of no greater than 15 minutes when specifications 1-2 are not met or do not exist.

Basis:

(1) 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.

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

## 3.7 Radiation Monitoring Systems and Effluents

### 3.7.1 Monitoring Systems and Effluent Limits

Applicability:

This specification applies to the facility area radiation monitoring (ARM) systems.

Objectives:

The objective is to specify the criteria for facility area radiation monitoring (ARM).

Specifications:

The reactor shall not be operated unless the conditions below are met:

(1) The total exposure from effluents from the facility to a person outside the site boundary shall not exceed 100 mrem per calendar year, less any external dose from the facility. This value may be increased in an emergency situation up to 500 mrem per calendar year if authorized by the Emergency Director.

a. 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 greater then 8 days, a reconcentration factor shall be included where appropriate.

b. For the purpose of converting concentrations to dose, the values of 10 CFR 20, Appendix B 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.

(2) A continuous fission products monitor is operable or sample analysis for fission products activity is conducted at least daily.

(3) Two area radiation monitors on floors C-100 and C-200 are operable and below the set points.

Basis:

(1) The criteria for determination of concentration limits specified above ensure that 10 CFR 20 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 (as justified in NBSR-7C, pages 22-27, Aug. 1, 1962). 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.

Since limits are determined at the point of release, instantaneous releases are necessarily considered. This ensures that the average release is not obtained by a small number of very large releases with the attendant possibility of high local concentrations of released effluents. This specification, although more restrictive than 10 CFR 20, provides additional assurance that releases to offsite personnel are minimized.

In specifying the limits on particulates and long half-lives (greater than 8 days) halogens, consideration was given to the possibility of biological reconcentration in food crops or dairy products. Using available information (Soldat, J.D., Health Physics 9, p. 1170, 1963), a conservative (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 multiplied by the 1000 dilution factor

divided by the 700 reconcentration factor (i.e., 1.4 times the Effluent Concentration Limit).

(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 provide assurance that the fuel cladding is intact. These two measures ensure that there are no undetected releases of fission products to the primary coolant.

(3) The ARM on C-100 and C-200 provide early indications of possible problems.

### 3.7.2 Environmental Monitoring Program

Applicability:

This specification applies to the environmental monitoring program.

Objective:

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

Specification:

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

Basis:

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 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 (URS, 2003 Geology, Seismology, Geotechnical Engineering, and Hydrology of the NIST Research Reactor Site,

Gaithersburg, Maryland, Sections 2.4 and 2.5, March 28, 2003), supports the findings of previous studies conducted on the hydrology and geology of the NIST site and vicinity. No significant changes in the hydro-geologic systems or ground water use were identified. This report further verifies the assumptions and techniques developed in 1964.

3.8 Experiments

3.8.1 Reactivity Limits

Applicability:

This specification applies to the reactivity limits for any in-reactor experiments to be installed within the NBSR.

Objectives:

The objective is to establish criteria for placing in-reactor experiments in the NBSR and to establish limits on these in-reactor experiments.

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 (SAR, NBSR 14, Chapter 13). The 0.2% $\Delta\rho$ for the combined pneumatic irradiation systems is well below this referenced accident as well as being within the $\Delta\rho$ 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, as discussed in NBSR-14, Chapter 13.

3.8.2 Materials

Applicability:

This specification applies to the materials that are placed into NBSR in-reactor experimental facilities.

Objective:

The objective is to prevent damage to the reactor or a significant release of radioactivity by limiting material quantity and the radioactive material inventory of the experiment.

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 Specification 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, without failure, the same transients that the reactor itself can withstand without failure (i.e., loss of reactor cooling flows, startup accident, and others where the reactor's safety system provides the ultimate protection).

The detonation of explosive or metastable materials within the reactor is not an intended part of the experimental procedure for the NBSR, 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:

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

Objective:

The objective is to prevent inadvertent criticality outside the reactor vessel and fuel element overheating during refueling operations.

Specifications:

(1) All fuel elements or fueled experiments shall be stored and handled in a geometry such that the calculated $k_{eff}$ shall be less than 0.95 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 $k_{eff}$ less than 0.95 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:

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

Objective:

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

Specification:

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 n 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 (SAR, NBSR 14, Chapter 4). Following fuel handling, it is necessary to ensure that this bar is properly positioned so that each element that has been handled can not "wash out" when flow is initiated. 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:

This specification applies to all other fuel handling conditions.

Objective:

The objective is to ensure fuel is handled in a safe manner.

Specifications:

(1) 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 grater than one hour for each megawatt of operating power level.

(2) Maintenance that prevents normal rapid closing of the confinement system is considered a breach of confinement and shall not be performed unless the reactor has been shutdown for a period equal to or grater than one hour for each megawatt of operating power level.

Basis:

(1) 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 margin of safety from the lowest temperature at which blistering can occur 842 °F (450°C) and an even larger margin from the fuel damage temperature of 986° F (530°C) that has been accepted by the Nuclear Regulatory Commission (NUREG 1537, Part 1 Section 2.1). These values are confirmed by fuel temperature tests carried out at the Oak Ridge Research Reactor. Therefore, the waiting times specified will prevent any fuel element damage or fission product release.

(2) During the waiting times in Specification (1), above, the fuel cannot be moved from the core, which is typically cooled by operating the primary cooling system. 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, when the fuel cannot be damaged even if only air cooled,

## 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 section 1.3.26. 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:

This specification applies to the surveillance requirement for the power level monitoring channels.

Objective:

The objective is to verify that the maximum power level of the reactor does not exceed the authorized limit.

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 $\Delta T$, 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 $\Delta T$ comparison will assure 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 $\Delta T$). Because of the small $\Delta T$ (about 15°F at 20 MW) these calibrations will not be performed below 5 MW for 10 MW operations or below 10 MW for 20 MW operations. However, to ensure that no gross discrepancies between nuclear instruments and the flow $\Delta T$ indicators occur, channel checks are made above 5 MW.

4.1.2 Reactivity Limitations

Applicability:

These specifications apply to the surveillance requirements for reactivity control of the reactor core.

Objective:

The objective is to measure and verify the reactivity worth, performance, and operability of these systems affecting the reactivity of the reactor.

Specification:

(1) The excess reactivity (reference core conditions) shall be determined 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 determined (as described in Specification 3.1.2(b)) annually or following any significant change in the core or shim arm configuration.

Basis:

(1) Determining the core excess reactivity annually will assure 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:

These specifications apply to the surveillance of the shim arms.

Objective:

The objective is to establish the operability of the shim arms.

Specifications:

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

(2) Scram times of each shim arm shall be measured semiannually.

Basis:

(1) & (2) 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:

These specifications apply to the surveillance requirements for measurements, tests, calibrations, and the operability of the reactor safety system channels.

Objective:

The objective is to ensure that the reactor safety systems are operable.

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 |
|---|---|---|
| a) High Flux level | Scram | X,A |
| b) Short period below 5% rated power | Scram | X,A |
| c) Low reactor vessel $D_2O$ level | Scram | X,A |
| d) Low flow reactor outlet | Scram | X,A |
| e) Low flow reactor inner or outer plenum | Scram | X,A |
| f) Manual (outside of the Control Room) | Scram | X,A |
| g) Manual | Scram | X,A |
| h) Normal Building Exhaust Activity High | Major Scram | X,A |
| i) Irradiated Air Activity High | Major Scram | X,A |
| j) Stack Air Activity High | Major Scram | X,A |

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

Basis:

The measuring channels listed in Table 4.2.2 ensure that the reactor shall be operated in a safe manner. To ensure that system failures do not go undetected frequent surveillance is required and specified above. Operating experience has shown these surveillance frequencies to be adequate to ensure safe operation.

4.3 Coolant Systems

4.3.1 Primary and Secondary

Applicability:

This specification applies to radioactivity in the primary and secondary coolant systems.

Objective:

The objective is to assure the availability of the shutdown cooling pumps and secondary radioactivity monitoring.

Specifications:

(1) The primary shutdown cooling pumps shall be checked for operability annually.

(2) The secondary cooling water activity monitor shall be channel tested for operability monthly and channel calibrated annually. When the secondary cooling water activity monitor is operable, sampling of the secondary cooling water and analysis for tritium shall be conducted monthly. Should the secondary cooling water activity monitor be inoperable, sampling of the secondary cooling water and analysis for tritium shall be performed daily.

(3) For tritium level less than or equal to 4 Ci/l the primary water shall be sampled annually. For tritium level greater than 4 Ci/l the primary water shall be sampled quarterly.

Basis:

(1) Operability checks annually of the primary AC & DC shutdown pumps have been sufficient to ensure the pump will operate when needed.

(2) The secondary cooling water activity monitor is a simple radiation detection device. A determination of its operability by channel testing monthly is considered a reasonable frequency for a device of this type. The annual channel calibration frequency is considered adequate to ensure that significant deterioration in accuracy 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, however, 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 the level instruments that indicate a loss from the $D_2O$ storage tank.

(3) The 5 Ci/l limit can be maintained by monitoring $^{16}N$ carryover indicating a leak of water recently irradiated in the reactor, or by doing laboratory analysis for the presence tritium. Both of these methods are employed. Experience has shown that sampling for tritium annually ensures regulatory limits are maintained. Since the relationship between activity level and leak rate is not linear above 4Ci/l quarterly sampling in this range is adequate.

4.3.2 Primary Coolant System

Applicability:

This specification applies to the primary coolant system.

Objective:

The objective is to ensure the operability of the primary coolant system.

Specifications:

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

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

Basis:

(1) The frequency for lifting the relief valve 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.

4.3.3 Emergency Core Cooling System

Applicability:

This specification applies to the emergency core cooling system.

Objective:

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

Specifications:

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

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

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

Basis:

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

4.3.4 Moderator Dump System

Applicability:

This specification applies to the reactor moderator dump system.

Objective:

The objective is to ensure the reactor can be made subcritical in any core configuration.

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:

This specification applies to the confinement building.

Objective:

The objective is 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 at a gauge pressure of at least 6.0 inches of water and a vacuum of at least 2.0 inches of water annually with a maximum allowable leak rate of twenty-four (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 exhaust radiation detectors or manually by the major scram button. To ensure complete surveillance, the system is tested by using these same devices to initiate the test. In addition, checks of both the trip features and the ability of the radiation detectors to respond to ionizing radiation are made.

(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:

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

Objective:

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

Specifications:

(1) An operability test of the emergency exhaust system, including the building static pressure controller and the vacuum relief valve, shall be performed 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) Absolute filters in both normal and emergency exhaust systems shall be tested for particulate removal efficiency biennially. The tests shall be designed to demonstrate that the absolute filters shall remove 99% of particles with diameters of 0.3 μm and greater.

(4) The charcoal filter banks in the emergency exhaust and recirculation systems shall be analyzed biennially to verify 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. But, since they are not being used continuously, their condition in standby shall be checked sufficiently often to ensure that they shall function properly when needed. An operability test of the active components of the emergency exhaust system 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 reactors 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 analysis of the absolute filter efficiency has been shown to be appropriate for filters subject to continuous air flow. Because the NBSR 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, the biennial analysis frequency should be adequate to detecting filter deterioration.

(4) Biennial analysis of filter banks, which are subjected to flow only during brief periods of fan operation provides assurance that the filters will perform as assumed in the analysis in NBSR-14, Chapter 13.

4.6 Emergency Power System

Applicability:

This specification applies to the emergency electrical power equipment.

Objective:

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

Specifications:

(1) Each diesel generator shall be tested for automatic starting and operation 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 a 5 year interval is adequate to detect deterioration of the cells.

## 4.7 Radiation Monitoring System and Effluents

### 4.7.1 Monitoring System

Applicability:

This specification applies to the Area Radiation Monitors and the secondary cooling water activity monitor.

Objective:

The objective is to ensure continued proper operation and calibration of the ARM and secondary cooling water activity monitor.

Specifications:

(1) The Area Radiation Monitors shall be tested for operability monthly and calibrated annually.

(2) The secondary cooling water activity monitor shall be tested for operability monthly and calibrated annually.

Basis:

(1) The area radiation monitors (ARM) may give the first indication of a radioactive release resulting from an experiment or reactor malfunction. Monitors are simple radiation detection devices whose operability has been shown to be very good over many years. Therefore, a determination of their operability monthly is considered reasonable for devices of this type. Because these devices are primarily used to detect an increase in activity over that which has previously existed, they are normally set at some reasonable value above background and their absolute accuracy is not critical. Hence, the annual calibration frequency is considered adequate to assure that a significant deterioration in accuracy does not occur.

(2) The secondary cooling water activity monitor usually gives the first indication of a primary to secondary leak. This monitor is a simple radiation detection device whose operability has been shown to be very good. Therefore, a monthly determination of its operability is considered reasonable for a device of this type. An annual calibration frequency is considered adequate to ensure that a significant deterioration in accuracy from its normal settings does not occur.

4.7.2 Environmental Monitoring

Applicability:

This specification applies to the frequency of the environmental monitoring sampling program.

Objective:

The objective of this specification is to ensure the environs are monitored on a timely basis.

Specification:

(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:

This specification applies to the surveillance requirements for experiments to be conducted at the NBSR.

Objective:

The objective is to prevent the conduct of experiments or irradiations which may damage the reactor or release excessive amounts of radioactive material as a result of experiment failure.

Specifications:

(1) A new experiment shall not be installed in any NBSR experiment facility until the requirements of Specification 6.5 have been met.

(2) 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:

(1) Experience has demonstrated the importance of reactor staff and safety committee reviews.

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

## 5.0 Design Features

5.1 Site Description

Applicability:

This specification applies to the reactor site location.

Objective:

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

Specification:

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

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

Basis:

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

5.2 Reactor Coolant System

Applicability:

This specification applies to the reactor coolant system.

Objective:

The objective is to ensure compatibility of the primary coolant system with the design features in NBSR-14.

Specifications:

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

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

(3) The reactor vessel shall be designed in accordance with Section VIII of the American Society of Mechanical Engineers (ASME) Code for Unfired Pressure Vessels. It 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 (NBSR 14, Chapter 5) as a single loop system containing heat exchangers, pumps and valves.

(2) 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) The design temperature and pressure of the reactor vessel and other primary system components provide adequate margins over operating temperatures and pressures. The reactor vessel was designed to Section VIII, 1959 Edition of the ASME Code for Unfired Pressure Vessels. Any subsequent changes to the vessel should be made in accordance with the most recent edition of this Code. The reactor coolant system is described in NBSR-14, Chapter 5. Therefore, it is considered necessary to retain this design and these margins.

5.3 Reactor Core and Fuel

Applicability:

This specification applies to the design of the reactor core.

Objective:

To assure compatibility of the reactor core with NBSR-14.

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 elements 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 uranium-aluminum alloy, either aluminum-uranium oxide or uranium-aluminide, clad with aluminum.

Basis:

(1) The neutronic and thermal hydraulic analysis (NBSR-14, Chapter 4) 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 (NBSR- 14, Chapters 4 and 13). Significant changes in core loading patterns require a recalculation of the power distribution to ensure that burnout ratios shall be within acceptable limits.

(2) & (3) The aluminum clad U-Al 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, NCNR shall be the licensee for the NBSR. The NBSR shall be under the direct control of the Chief Reactor Operations and Reactor Engineering. The Chief Reactor Operations and Reactor 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 NIST Center for Neutron Research, 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 Fig 1. Individuals at the various management levels, in addition to having responsibility for the policies and operation of the NBSR, shall be responsible 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

a. The minimum staffing when the reactor is not secured shall be:

1. A Reactor Operator in the Control Room.

2. A Reactor Supervisor present within the reactor exclusion area..

3. A SRO present is present in the reactor control room whenever a reactor startup is performed, fuel is being moved within the reactor vessel, or experiments are being placed in the reactor vessel.

b. A list of reactor facility personnel by name and telephone number shall be available to the senior reactor operator in the control room. This list shall be updated annually. The list shall include:

1. Management personnel.

2. Health Physics personnel.

3. 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). Qualification and requalification of licensed reactor operators shall be performed in accordance with an approved Nuclear Regulatory Commission (NRC) program.

6.1.4.1 Selection of Personnel

Minimum educational and or 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 Reactor Engineering

The Chief Reactor Operations and Reactor 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 and/or a combined seven years of college level education and nuclear reactor experience. Three years of reactor operations experience is required. The individual shall demonstrate the capability to be an SRO at the NBSR.

(3) Reactor Supervisor

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

(b) A high school diploma or equivalent and formal training in reactor technology and reactor operations. An additional

two years of experience may be substituted for education and formal training.

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

(4) Senior Reactor Operator

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

(5) Reactor Operator

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

(6) Auxiliary Operator

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

6.1.4.2 Training of Personnel

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

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

(3) Records of individual reactor operations staff members', qualifications, experience, training, and requalification shall be maintained as specified in Specification 6.8.2.

## 6.2 Review and Audit

The NIST Center for Neutron Research Safety Evaluation Committee (SEC) is established to provide an independent review of the NIST Center for Neutron Research (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 NIST Center for Neutron Research Safety Audit Committee (SAC) is established to provide an independent audit of the NIST Center for Neutron Research 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 Reactor 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 NCNR 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 NCNR 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 (SEC) Charter and Rules:

The SEC shall conduct its review functions in accordance with a written charter. This charter shall include provisions for:

a. Meeting frequency.
b. Voting rules.
c. Quorums.
d. Method of submission and content of presentation to the committee.
e. Use of subcommittees.
f. Review, approval and dissemination of minutes.

6.2.3 SEC Review Function:

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

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

b. Review the circumstances of all events described in Specification 6.7.2 and the measures taken to preclude a recurrence and provide recommendations to the NCNR Director for action.

c. 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 the SAC annual audit report.

d. 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.

6.2.3 Safety Audit Committee (SAC)

The Safety Audit 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 NIST Center for Neutron Research Director. 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 audit the NBSR reactor operations and the performance of the SEC. The SAC shall report in writing to the NIST Center for Neutron Research Director.

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 (R2004). The NIST Reactor Health Physics Group leader shall report to the NIST Center for Neutron Research Director 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 (i.e. to be determined by the Chief Reactor Operations and Reactor Engineering or the Chief Reactor Operations) to operating procedures shall be reviewed by the SEC and approved by the Chief Reactor Operations and Reactor Engineering or the Chief Reactor Operations. Such reviews and approvals shall be documented in a timely manner. Activities requiring written procedures are:

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

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

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

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

(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 Reactor 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 Reactor 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:

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

b. 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 that was approved by the Director, NCNR shall be reported to the SEC at their next meeting.

6.6 Required Actions

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

In the event a safety limit is exceeded:

a. The reactor shall be shutdown and reactor operations shall not be resumed until authorized by the NRC.

b. An immediate notification of the occurrence shall be made to the Chief Reactor Operations and Reactor Engineering and the Chief Reactor Operations. The Chief Reactor Operations and Reactor Engineering shall inform the NIST Center for Neutron Research Director.

c. Reports shall be made to the NRC in accordance with Section 6.7.2 of these Technical Specifications. A written report (required within 14 days) 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 Reactor 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 NIST Center for Neutron Research 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

For all events which are required by regulations or Technical Specifications to be reported to the NRC within 24 hours under Section 6.7.2, except a Safety Limit Violation, the following actions shall be taken:

a. The reactor shall be secured and the Chief Reactor Operations and Reactor Engineering and the Chief Reactor Operations notified.

b. Operations shall not resume unless authorized by the Chief Reactor Operations and Reactor Engineering.

c. The SEC shall review the occurrence at their next scheduled meeting.

d. Where appropriate and in addition to the initial notification a report shall be submitted to the NRC in accordance with Section 6.7.2 of these Technical Specifications.

## 6.7 Reports

### 6.7.1 Annual Operating Report

An annual report shall be created and submitted by the Director, NIST Center for Neutron Research to the NRC by March 31st of each year consisting of:

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

b. The number of unscheduled shutdowns, including reasons therefore.

c. A tabulation of major preventative and corrective maintenance operations having safety significance.

d. 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.

e. 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.

f. A summarized result of environmental surveys performed outside the facility.

g. 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 Reactor Engineering to the NRC as follows:

a. There shall be a report within 24 hours by telephone or fax 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:

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

2. Violation of a safety limit.

3. Operation with a safety system setting for required systems less conservative than the limiting safety system settings specified in the technical specifications.

4. Operation in violation of a Limiting Condition for Operation (LCO) established in the technical specifications unless prompt remedial action is taken as permitted by the exception statements in Section 3.

5. 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 no report is required.

**NOTE:** 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.

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

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

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

b. A report within 30 days in writing to the NRC, Document Control Desk, Washington, D. C. 20555 of:

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

2. 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 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. Records and logs shall be prepared for the following items and retained for a period of at least 5 years (Annual reports, to the extent they contain all of the required information, may be used as records for any of the items below.):

a. Normal reactor operation logs (but not including supporting document such as checklists, log sheets, etc., which shall be retained for a period of at least one year).

b. Principal maintenance activities.

c. Special Reports.

d. Surveillance activities required by the Technical Specifications.

e. Solid radioactive waste shipped off-site.

f. Fuel inventories and transfers

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

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

### 6.8.3 Records to be Retained for the Life of the Reactor Facility

a. Gaseous and liquid radioactive effluents released to the environs,

b. Off-site environmental-monitoring surveys required by the Technical Specifications,

c. Radiation exposure for all personnel monitored,

d. Drawings of the reactor facility.

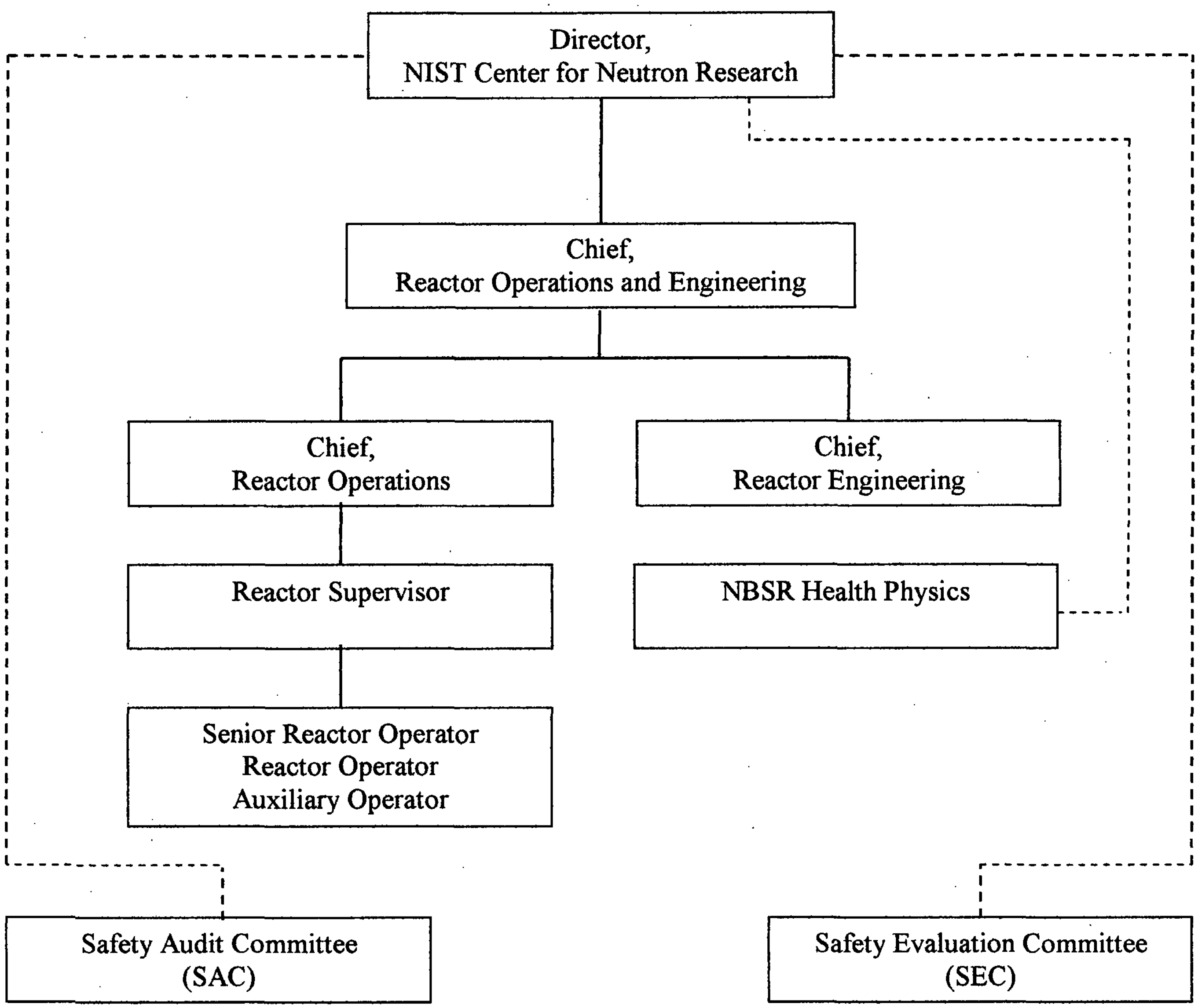



**Figure 1.**

——— Administrative Reporting Channels

- - - - - Recommendations and Technical Advice