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

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

May 30, 2007

Subject: NRC Request for Additional Information (RAI's) dated

Docket No. 50-184

Gentlemen,

Please find the attached response to NRC Request for Additional Information dated February 13, 2007. The revised NBSR Technical Specifications will be sent by the end of June 2007.

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

Sincerely,

Wade J Richards

Wade J. Richards
Chief, Reactor Operations and Reactor Engineering

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

Executed on: 5/30/07 by: Wade J Richards

cc.
Marvin Mendonca
NRC Project Manager

A020

NIST

Response to NRC RAI's
May 30, 2007

Part I: Chapter 2 Technical Questions and Comments:

2.1 The SAR text indicates that the 100-year return wind speed of 102.5 mph is within the uncertainty limits of the 100 mph design of the Confinement Building. The 102.5 mph value is calculated based on the 90 mph 50-year return gust taken from ASCE 7-98. However, virtually the entire country away from the coastline is rated with a 90 mph gust level. In all likelihood, a more appropriate value for the 50-year return wind speed is somewhat lower, and as a result, the 100-year return wind speed would be lower as well. In this discussion (Section 2.3.1.5, Table 2.13, and Figure 2.7) provide a more refined estimate of the 100-year return wind speed, which should be less than the design value.

Response: This question was previously asked. Our previous response remains valid and the final draft of the SAR will incorporate new language to address the issue contained in this question.

2.2 The discussion in SAR Section 2.3.1.6 regarding snow density references a publication of the American Meteorological Society with a range of densities of 0.07 to 0.15. Clarify the text to reflect that this range of densities is for freshly fallen snow. Verify that the correct date of the reference is 1989, and make any necessary corrections to the text and references section.

Response: This question was previously asked. Our previous response remains valid and the final draft of the SAR will incorporate new language to address the issue contained in this question.

fallen snow. Verify that the correct date of the reference is 1989, and make any necessary corrections to the text and references section.

Response: This question was previously asked. Our previous response remains valid and the final draft of the SAR will incorporate new language to address the issue contained in this question.

Chapter 4 Technical Questions and Comments:

4.1 Section 4.2.1.1, p. 4-3. In the "Fuel Composition" section, it is stated that the "fuel core is a slug type design." Provide clarification of the term "slug type" or use more descriptive language to describe the fuel core design.

Agreed, will revise as appropriate.. The phrase "Slug type" adds nothing to the description of the fuel, which is an Al – U_3O_8 dispersion, described later in Section 4.2.1.1.

4.2 Section 4.2.1.2, p. 4-3. Provide sufficient overall fuel element dimensions for comparison with the unit cell dimensions provided in this section and TS 5.3.

Agreed, will revise as appropriate.. Revised Figures 4.2.3 and 4.2.4 are attached. These figures show all dimensions mentioned on pages 4-3 and 4-4, as well as the details of the fuel plates and the fuel element.

4.3 Section 4.2.1.3, p. 4-5. Provide clarification that the fabrication of NBSR fuel elements is consistent with ANS 15.2.

Fabrication of NBSR fuel elements is in accordance with ANSI standards (ANS 15.2) for the manufacture of MTR plate type fuel elements, and the NIST specification for aluminum clad fuel elements (NIST, 2004a).

4.4 Section 4.2.1.4, p. 4-6. The second paragraph states "Flow rates of 30 ft/sec which are over two times those seen in operation, (9.1 m/sec) were employed to measure flow conditions in each channel..." Provide clarification of whether the 9.1 m/sec is the flow rate seen in operation or the test flow rate. Also, provide discussion that justifies the use of test flow rates that are over two times the operational flow rates for both the inner and outer plenums.

Flow tests were conducted over a wide range of velocities to observe the behavior under low-flow, normal operations, and well beyond expected conditions. A second series of tests were performed when the fuel plates were modified, eliminating most of the unfueled portion of the plates.

4.5 Section 4.2.2.1, pp. 4-9,10. The description of the "operational travel of 41° and a maximum travel of 50°" appears inconsistent with the statement "To prevent over travel during normal operation of the shim arm, installed upper and lower limit switches are set to approximately 41° and 2°, respectively." Clarify the operational shim arm travel ranges, limits, and corresponding angular positions.

If the shim arms could travel until they struck the top grid plate above and the shim arm guide extensions (catchers) below, they would travel from + 5° to – 45°, a total of 50°. Instead the travel is limited to 40° to 41° by switches that prevent the tips of the shim arms from impact with core internal components that could disturb the calibration of the position measurement channel, or damage the shim arm drive motor. The shim positions indicated on the console are measured from the down position.

4.6 Section 4.2.2.2, p. 4-11. This section states that the regulating rod is "$2^{1}/_{2}$ inches in diameter" and the last SER (NUREG-1007) says the regulating rod is "a 2.25 inch diameter solid aluminum rod." Clarify if the regulating rod design has been changed and describe any impact on the safety analysis.

NUREG-1007 was written before the regulating rod was changed as part of the power increase to 20 MW. It is a 2.5-in diameter solid Al rod. The safety analysis is presented in ECN #293, proposed in October, 1984, and implemented in April, 1985.

4.7 Section 4.2.2.2, p. 4-11. This section states "The regulating rod acts as a poison designed with a reactivity worth approximately 0.58 ." The reactivity worth is inconsistent with the 0.58% stated elsewhere. Confirm the magnitude of this value and clarify if the reactivity worth is derived primarily from absorption (poison) or moderator displacement.

The regulating rod is worth 0.5 – 0.6 %$\Delta\rho$, its value depending on the location of the shim arms. Its worth is measured at least once a year. Although the rod displaces D_2O as it moves through a thermal neutron flux trap, the reactivity is due largely to absorption in the Al rather than the loss of the moderator. Al is not a poison, but the macroscopic absorption cross section of Al is more than 400 times that of heavy water. MCNP calculations indicate that displacement of D_2O contributes 10-20% of the negative reactivity inserted by the regulating rod.

4.8 Section 4.2.4, p. 4-16. This section states that the source is placed into one of the existing experimental thimbles and does not contact the coolant. In the following section, Core Support Structure (p. 4-17), it is stated that coolant passes up through the experimental thimbles. Clarify how the source does not contact the coolant and justify why no cooling is required. Describe the source encapsulation material of construction (MOC) and the design and testing requirements.

The startup neutron source is a 1.9-Ci encapsulated Am-Be source. Its integrity is monitored by a Health Physics surveillance program. The ***source is cooled directly*** *by the D_2O in the experimental thimble. It is checked for leaks before it is loaded into the core, and is used only at very low reactor power.*

4.9 Section 4.2.5, p. 4-17. This section states that the experimental thimbles are held down by poison tubes from the top plug. Describe the design of the poison tubes, including materials of construction and any age-related issues. Describe any other purpose(s) of the poison tubes.

The seven poison tubes are 4-in (10.16 cm) OD Al tubes, approximately 0.25 in (0.64 cm) thick, extending from the bottom of the refueling plug to the 3.5-in (8.89 cm) thimbles in the core. The tubes are latched to the plug preventing any upward motion of the thimbles due to primary flow. The center, 36-in (91.4 cm) portion of each tube contains a 40-mil (1 mm) thick concentric layer of cadmium within the Al wall (refer to drawing D-01-035). The Cd was included to lower the neutron flux above the top grid plate in order to minimize activation of the bottom of the refuel plug. As a result, the thermal neutron flux decreases from about $5x10^{12}$ n/cm^2-s to 10^9 n/cm^2-s, so there are no life-time issues regarding the strength of the Al or the burnup of the Cd.

4.10 Section 4.3.1, p. 4-18. The description of the reactor vessel design discussed the use of two stainless steel O-ring gaskets at the reactor vessel flange. Describe any periodic inspection, leak testing, and replacement requirements or justify why these are not necessary.

There is no periodic inspection as these both seals are located beneath massive shields that are rarely, if ever, moved. The helium and CO_2 systems are at very low positive pressures, and these seals are just two of many inaccessible components of the system boundaries. Helium and CO_2 leak rates have been measured, however, and the performance of the system boundaries is monitored via tritium and ^{41}Ar monitoring, respectively, and the consumption rates of the gases.

4.11 Section 4.3.1, pp. 4-19 & 4-20. This section discusses "grazing tubes" as a separate vessel attachment. Relate the "grazing tubes" in the nomenclature terms used in the experimental facility descriptions in Chapter 10, e.g., radial beam tubes, through tubes, etc. Ensure nomenclature is consistent.

The grazing tubes are tangent to the core adjacent to the north and south rows of fuel elements. They are referred to as through tubes in Chapter 10. Since they completely penetrate the reactor vessel, their walls are part of the vessel, as are all of the radial beam tubes, the cryogenic beam port, and the four thimbles for pneumatic tubes.

4.12 Section 4.3.1, p. 4-20. The fourth paragraph states "Since the vessel is entirely closed, there is no credible mechanism of exerting such a tensile stress, or impact, on the beam tube tips during reactor operation." Describe how all credible mechanisms for stresses resulting from pressures or impacts on the outside (non reactor side) of the beam tubes have been eliminated. Justify that the change in material properties (reduced ductility and Charpy energy) due to irradiation from past and future operations (20 years) will not reduce the design margins of safety to unacceptable levels. Describe the effect of the change in material properties (reduced ductility and Charpy energy) on the reactor vessel design rating and relief valve set pressure.

The only mechanism identified for exerting pressures on the beam tube tips is from an experiment inserted into a beam tube. For all beam tubes, there is an aluminum diaphragm at the inner face of the thermal shield that prevents inadvertent insertion of any experiment into the thimble. All experiments are reviewed for safety, and no experiment or part of an experiment is allowed inside the biological shield unless it is surrounded by a container that can contain the maximum possible internal pressure or force that could be generated by any credible accident. In the case of cold sources, which actually are inserted beyond the inner boundary of the thermal shield into the thimble, the design basis requires a container that can withstand a maximum hypothetical accident. This accident is defined as a detonation of a stoichiometric mixture of hydrogen and air at atmospheric pressure.

The NBSR Vessel operates at very low pressure, and has a design operating pressure of 343 kPa (50 psig), with a relief valve set for 50 psig. The criterion chosen for the vessel was a leak before break analysis of the vessel under design pressure at the position of highest stress in the most irradiated state after 40 more years of operation at design pressure, assuming an irradiation level corresponding to the beam tube tip. A large margin of safety was found, with a crack propagation stress 100 times the design stress. This analysis incorporates the reduction in Charpy energy. This analysis is contained in a memorandum to the Chief, Reactor Operations and Engineering form J. M. Rowe and R. E. Williams dated April 4, 2002. The reduction in elongation (loss of ductility) is not sufficient to cause any reductions in safety margins for the vessel design.

4.13 Section 4.3.1, p. 4-21. The third paragraph states "The shim safety-arm drive and shock absorbing systems are mounted on the biological shield so that only the extremely small reaction between the outer faces and the balls is transmitted to the vessel." Describe what is meant by the "outer faces and balls."

This phrase is confusing and far removed from its context, Section 4.2.2.1. The shim safety-arm drive and shock absorbing systems are mounted on the biological shield so that the impact of a scram is not transmitted to the vessel.

4.14 Section 4.3. Describe any surveillance or inspection programs for the periodic assessment of corrosion or radiation damage or why it is not needed.

While there is no vessel formal surveillance program, the vessel was visually inspected in 2003. The vessel has been filled with D_2O for 40 years. There have never been any signs of corrosion on any vessel components or on the fuel elements removed each cycle. Rigorous attention to the primary water chemistry (pH and conductivity, described in Section 5.4) assures that there will be no corrosion of the reactor vessel. Radiation damage is discussed in the response to RAI 4.12.

4.15 Section 4.4.2, p. 4-23. The second paragraph states "The results yield a fast neutron flux 2.8x10-3 n/cm2-sec and a gamma flux of 2x10-7 mW/cm2 at the outside face of the biological shield." Describe how these results were calculated, and how the subsequent 25% concrete, 75% thermal shield neutron capture gamma fractions were determined.

The flux estimates at the face of the biological shield are taken directly from NBSR-9 but increased by a factor of 2 for the increase to 20 MW. The source terms in Table 4.4.1 were multiplied by the attenuation factors in Table 4.4.2, assuming the minimum thickness of concrete is 74 inches (188 cm). The methodology used to obtain the source terms is outlined in NBSR-7, dated January, 1961. Clearly, detailed shielding calculations could be done with today's codes and computers to refine these 40-year old estimates.. This analysis has not been done, however, because the biological shield has proven to be adequate over the operating history of the facility. Neutron and gamma radiation fields on the experiment level of C-100 are dominated by unshielded reactor hardware (i.e., the thermal shield coolant "ring" header) and apparatus installed in neutron beams (collimators, beam stops, monochromators, etc.), and not by core radiation penetrating the biological shield.

4.16 Section 4.4.3, p. 4-24. The fourth paragraph states "The radiation near the top of the center plug constitutes no health risk since it is in the well in the top floor that is covered with a 6-inch (15.2 cm) steel plate. This plate, an integral part of the transfer system, is always in place when fuel elements are being moved. The plate over each pick-up tool is penetrated by openings up to 6 inches (15.2 cm) in diameter that normally are plugged." It appears the dose rate of 0.5 mrem/hr stated in this section applies to an inaccessible area. Clarify what the radiation field would be in the area above the top shield plug where personnel may be located during transfer operations.

The first sentence of the 4th paragraph is confusing. The radiation near the top of the center plug constitutes no health risk because the plug is covered with a 6-inch (15.2-cm) steel plate.

The measured value in the immediate vicinity of a pick-up tool being manipulated by one of the operators is about 0.5 mrem/hr.

4.17 Section 4.5.1.2.2, p. 4-29. The fourth paragraph states "This 'loss' of material was dealt with by adding elemental Zr and Sn, and 138Ba, to mock up those fission products." Provide the justification for this substitution.

Zr, Sn, and ^{138}Ba were added to keep the density of the fuel constant, as MONTEBURNS rejected fission products from ORIGEN for which there are not MCNP cross section data. These particular isotopes were chosen because they have small absorption cross sections and have masses characteristic of fission products. It is assumed that all of the major neutron absorbers among the fission products were in the MCNP data libraries.

4.18 Section 4.5.1.3.1, p. 4-29. The reactivity change, , is defined and the method for calculating presented. Elsewhere in the chapter, the values of reactivity are presented as k/k. Provide consistent terminology or additional definitions and methodology.

SAR Section 4.5.1.3.1 (page 4-29):

The reactivity change, $\Delta\rho_X$, between a reference case with reactivity ρ_{ref} and some other configuration, ρ_X, is calculated as follows:

$$\Delta\rho_X = \rho_X - \rho_{ref} = (k_X - 1)/k_X - (k_{ref} - 1)/k_{ref}$$

$$= 1/k_{ref} - 1/k_X.$$

We used the units of $\Delta\rho$ and $\Delta k/k$ interchangeably, and were not indicating with $\Delta k/k$ a different quantity. The calculations in Chapter 4 should have units of $\Delta\rho$, because they were calculated as indicated on page 4-29. Chapter 4 is not consistent with Appendix A, however, in which $\Delta k/k$ was sometimes calculated explicitly (See response to 13.21 and 13.22, below).

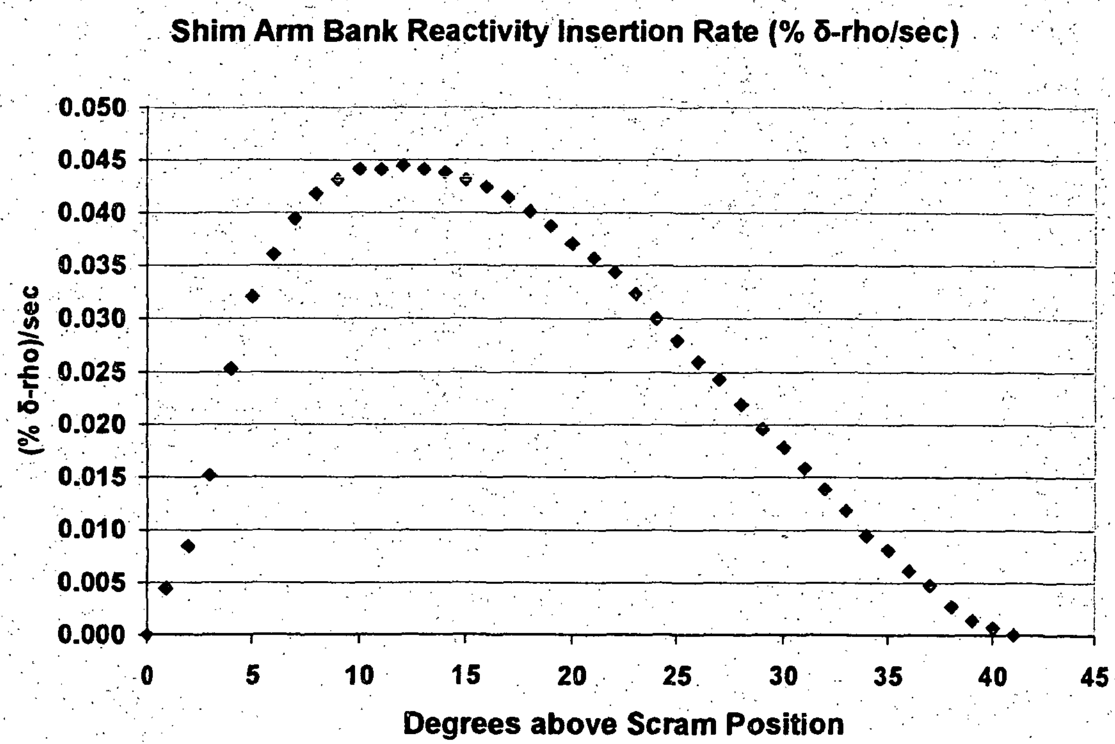
4.19 Section 4.5.1.3.2, p. 4-30. Explain how the reactivity change of 0.34 % from "su183" to "sucold" is consistent with the reactivity temperature coefficients, e.g., the calculated moderator reactivity temperature coefficient.

The apparent moderator temperature coefficient, MTC, estimated from the k-eff values for the "su183" and "sucold" input files ($\Delta\rho$=+0.339 %$\Delta\rho$, ΔT = -28 K, MTC = -.012 %$\Delta\rho$/K) appears to be less than half of the value for the MTC (-.031 %$\Delta\rho$/K) for the SU core in Table 4.5.6.

A calculation of MTC using MCNP has to include three contributions: (1) the mass density for the cells filled with moderator, (2) the input temperature of the moderator material from the TMP card, and (3) the fixed temperature of the S(α,β) thermal neutron scattering kernel for the material. The contribution from (3) was not evaluated above, however, and Table 3-1 in Appendix A shows that (3) dominates the MTC calculation. Adding (-.0222 %$\Delta\rho$/K)(28 K) to $\Delta\rho$ gives $\Delta\rho$ = 0.961 %$\Delta\rho$ and MTC = -.034 %$\Delta\rho$/K.*

4.20 Section 4.5.1.5.1, p. 4-34. The second paragraph states "Multiplying the differential shim bank reactivity worth by the speed of the shim arm drives, 0.0445 °/s, one obtains the reactivity insertion rate vs. position, shown in Fig. 4.5.19." This does not appear to be what is shown in Figure 4.5.19. Clarify the statement or modify the figure to be consistent with the statement.

The correct Figure 4.5.19 is below:



4.21 Section 4.5.1.5.3, p. 4-34. The first paragraph states "its average reactivity insertion rate is 3.8 x 10-4 /sec." Provide the maximum differential rod worth and insertion rate, and provide a comparison with the TS 3.4 limit.

The maximum measured reactivity insertion rate for the regulating rod is 4.5 x 10^{-4} $\Delta\rho$/s, which is below the TS limit of 5 x 10^{-4} $\Delta\rho$/s. (Strictly speaking, the maximum reactivity insertion rate limit in TS 3.4 is for the bank of shim arms; no limit for the regulating rod is specified.)

4.22 Section 4.5.1.6.1, p. 4-35. The second paragraph states "The fuel mass in F-5 is just 138 g, so the normalized worth is 7.6 % /kg." In Figure 4.5.2A, p. 4-86, the F-5 mass is given as 125g. Clarify the apparent difference.

Figure 4.5.2A shows the estimated masses used in the MCNP model before the MONTEBURNS analysis. The calculation described in Section 4.5.1.6.1 uses the masses from the BNL model with the burnup analysis. The calculation is self-consistent, but the masses are not consistent with the earlier model.

4.23 Section 4.5.1.7, p. 4-36. The second paragraph states "There are only three means of adding positive reactivity to the reactor while it is critical: (1) withdrawing the shim safety arms, (2) lowering the inlet D2O temperature, and (3) rapidly removing experiments." Justify not including the regulating rod in this list.

The regulating rod can certainly add positive reactivity, but not very much. Change (1) above to read: "(1) withdrawing the shim safety arms or the regulating rod."

4.24 Section 4.5.3.2, p. 4-47. The 0.2 % limit for the pneumatic irradiation system and the 1.3 % limit for movable experiments are not included in the criteria section of TS 3.12. Provide justification for why these limitations are not criteria in TS 3.12 or modify the criteria accordingly.

We are in the process of revising the 1.3 %$\Delta\rho$ maximum reactivity insertion in 0.5 sec because it is an incredible accident, requiring operators to pull three different experiments from the core inadvertently in 0.5 second. Likewise, the operators control the use of the pneumatic irradiation systems, or operate the systems themselves.

4.25 Table 4.2.3, p. 4-61. Provide operating conditions and calculations for the 3.66 m/sec channel flow velocity under the "NBSR" column in the table.

Table 4.2.3 was prepared in the 1960's and is out of date. Flow conditions are presented in Sections 4.6 and Appendix A.

4.26 Table 4.2.3, p. 4-61. The units for Max. Heat Flux in the first column appear inconsistent with standard heat flux units, e.g., "BTU/hr-ft2 (W/m2)." Also, the max. heat flux given for NBSR as 1.54 x 105 W/m2 appears inconsistent with the hot spot heat flux given on p. 4-54 for element H-1 and the conversion between heat flux units appears incorrect. Clarify or correct the differences, as appropriate.

See RAI 4.25.

4.27 Section 4.2.1.4, p. 4-6. This section indicates that the bypass flow was measured at substantially higher flow rates than the flow rates typically found during normal operation. As the dimensions of the gap for the bypass flow result from hydraulic drag, justify that the measured bypass flow rate is correct for normal operating conditions.

The gaps for bypass flow are indeed opened by hydraulic drag from the primary flow through the fuel elements. The gap size, however, does not continue to increase with increasing flow as there is a mechanical stop; the fuel element makes contact with the latch pin (see Figure 4.2.5). Once this occurs, the bypass flow area is fixed, and the bypass flow will remain proportional, about 4%, to the total flow. See also the response to RAI 13.16.

4.28 Section 4.2.5, p. 4-17. Provide clarification regarding the potential for the poison tubes to buckle due to upward coolant forces on the experimental thimbles. If buckling of the poison tubes is credible, provide analysis that shows it could not cause an accident not bounded by the maximum hypothetical accident.

The poison tubes have adequate mechanical strength to hold the 3.5-inch (8.89 cm) thimbles in place, and they are located above the top grid plate where there will be no radiation induced damage to the aluminum (see the response to RAI 4.9).

If a tube buckled, the thimble would be pushed upward, until most of the flow through the 2.375-in (6.03 cm) ID opening in the bottom grid plate bypassed the thimble. The worst case would be the central thimble, for which the opening is 3.5 in (8.89 cm) ID. The flow area would twice that of the plated sections of the fuel elements. The flow through each of the six elements would be reduced by about 25%. There would be no fuel damage.

4.29 Section 4.5.2.1.1, p. 4-37. The delayed neutron fraction is presented for steady reactor power conditions. Describe and quantify any variation that may occur in this parameter during transient conditions.

The point kinetics model used in the RELAP code assumes that the delayed neutron precursors are always produced in the same ratios (constant β_{eff}) for fission of ^{235}U, and they are indeed constant for steady state conditions. The code does, however, calculate the varying concentrations of the precursors as a function of time during a power transient, and introduces the delayed neutrons accordingly, to correctly calculate the power as a function of time.

4.30 Section 4.2.2.2. The regulating rod withdrawal rate has been changed since NBSR-9 from 30" per minute to 120" per minute. The design of the regulating rod has also been changed. Describe how these changes affect the reactivity insertion rate of the regulating rod. Provide the evaluation that was performed to determine that the change did not impose any unreviewed safety questions.

ECN-002, approved in October 1965 (two years before the startup of the NBSR) recognized that the original hollow rod would have insufficient reactivity to offset the insertion of the maximum allowed moveable experiment. The rod was changed from a hollow rod to a 2.25-in (5.72-cm) OD solid rod. A solid Al rod also has the advantage of precluding an unanticipated reactivity insertion due to flooding the hollow rod. The increase in rod velocity also occurred before the December 1967 startup. The speed was increased to the point that the insertion of the regulating rod could offset the withdrawal of the shim arm bank, a maneuver required several times per cycle. The reactivity insertion rates of the regulating rod and the shim arm bank are nearly equal; both are less than the maximum reactivity insertion rate limit (see response to RAI 4.21).

4.31 Section 4.5.2.3.3. The analyses use 30 fuel elements instead of 24 fuel elements allowed by TS 3.3 when the corner positions of the hexagonal lower grid plate are filled with plugs. Provide analyses to show that the use of 30 fuel elements in these analyses represent the limiting case. Explain how the hot channel factors account for the uncertainties in instrumentation and fuel fabrication tolerances. Describe how the uncertainties are treated (statistical vs. deterministic).

TS 3.3 was changed. In the revised Tech Specs, TS 3.1.3 requires a 30 element core. Any other configuration would require an analysis for reactivity and thermal-hydraulic safety.

The hot channel factors account for the uncertainties in instrumentation and fuel fabrication tolerances in a statistical manner. See response to RAI 4.32, below.

4.32 Section 4.6.3. Justify the assumption that the coolant within a single channel mixes completely. Justify the assumption that the coolant mixes completely in the unfueled gap between the upper and lower core. Justify treating the uncertainties in a statistical manner. Describe the conservatism built into the correlations for DNB and OFI and quantitatively estimate the conservatism provided by these correlations for the NBSR analyses.

Flow through the coolant channels of the NBSR fuel elements is extremely turbulent, having a Reynolds number of about 36,000. When the D_2O enters and exits the unfueled gap, there are abrupt velocity changes. Small, local variations in density and velocity owing to hot spots promote further mixing. The assumption of complete mixing under these flow conditions is more realistic than any other assumption regarding possible flow patterns.

We use a statistical treatment of uncertainties in our calculations of CHF and OFI because it is not credible that every physical parameter and instrumental uncertainty entering the calculation will simultaneously be at their most unfavorable value at a given location in a given instant. Our calculations use a statistical factor of 1.3 which is approximately the 95% confidence level. The statistical treatment is detailed on pages D-1 to D-16 in Appendix A.

Chapter 4 Editorial Questions and Comments:

4.34 Section 4.2.1.1, Fuel Composition, p. 4-3. It is stated that the aluminum powder used is ATA 101 (or equivalent). Clarify the "ATA" abbreviation and add to the "Acronyms" list.

ATA-101 is a commercial product name from Toyal America, Inc. The designation implies a specification for the particle size and impurity concentrations. ATA is a brand name, not an acronym.

4.35 Section 4.2.1.2, Fuel Element Description, p. 4-4. It is stated that "the fuel plate core frames and cladding are aluminum Alloy 6061-T0 (ASMT B209)." This is inconsistent with Table 4.2.2, which has "aluminum clad" as 6061-T6.

Agreed, will revise as appropriate..

4.36 Section 4.2.1.2, Fuel Element Description, p. 4-3. It is stated that "fuel is contained in fuel plates approximately 13 inches in length by 2.793 inches in width...." The width dimension is inconsistent with that in Table 4.2.3, p. 4-61 (2.415 in).

Agreed, will revise as appropriate.. Table 4.2.3 is out of date.

4.37 Section 4.2.1.2, Fuel Element Description, p. 4-4. The first line states "curvature is 5 .5 inches (13.97 cm)." There is an extra space in "5 .5 inches."

Agreed, will revise as appropriate..

4.38 Section 4.2.1.3, Fabrication, p. 4-5. It is stated that "Dents greater than 0.250 inch (0.06 cm) in diameter..." These dimensions are inconsistent.

Agreed, will revise as appropriate.. The sentence should read: "Dents greater than 0.250 inch (0.635 cm) in diameter and/or greater than 0.006 inch (0.015 cm) deep shall result in the rejection of the plate".

4.39 Section 4.2.2.1, Shim Safety Arm, p. 4-9. It is stated that "Helium at just slightly above atmospheric pressure (15 psig) is left in the void." Is the pressure approximately twice atmospheric pressure, or 15 psia?

Agreed, will revise as appropriate.. The helium pressure over the reactor vessel is nominally 4 inches of H_2O (1 kPa or 0.001 bar), close to 15 psia.

4.40 Section 4.2.2.6, Technical Specifications, p. 4-14. TS 4.3, item no. 5, states "a comparison of power range indication with flow time's delta T...." The apostrophe in "time's" appears unnecessary.

Agreed, will revise as appropriate..

4.41 Section 4.2.2.6, Technical Specifications, p. 4-15. TS 4.3, the basis section states "The shim arms shall be considered operable if they drop the top five (5°) within 220 msec." The "top five (5°)" apparently should read "top five degrees (5°)."

Agreed, will revise as appropriate..

4.42 Section 4.2.5, p. 4-16. This is a general comment about units formatting, but it occurs here because this section switches from using English units with SI units in parentheses previously, and in this section that convention is intermittently swapped. ANS-15.21-1996 states "SI units shall be used, with English units posted in parentheses, except where the regulations require a different presentation."

Generally our SAR used notation that is opposite the convention cited above in ANS-15.21-1996. As NIST employees we certainly want to use the correct SI units. But, the NBSR was designed 40 years ago, and its dimensions, tolerances, and specifications were presented and approved in English units. The 2nd paragraph of Section 4.2.5 should be consistent with our convention of citing original English units, with SI or CGS units in parentheses.

4.43 Section 4.3.1, Design, p. 4-20. The third paragraph has "2x1023 n-cm-2-s-1." From the context, it appears the units should be "2x1023 n-cm-2."

Agreed, will revise as appropriate..

4.44 Section 4.4, p. 4-22. In the first paragraph of this section, the sentence "Chapter 10 of NBSR-9 (NBS, 1966a) contains a thorough description the design considerations and shielding calculations for the construction of the biological shield," is apparently missing an "of" in the phrase "description of the design...."

Agreed, will revise as appropriate..

4.45 Section 4.5.1.3.4, Fission Product Poisons and the Equilibrium Core, p. 4-31. The second paragraph states "The reactivity difference between the SU benchmark, "su183," and the BOC equilibrium core, "eqlib," is keff = 0.97911, and = - 2.86 %k/k, or –$3.78." Clarify the reactivity units.

Agreed, will revise as appropriate..

4.46 Section 4.5.1.5.1, The Shim Safety Arms, p. 4-33. The first paragraph states "After the initial shim arm movement, there is a gradual withdrawal until the shim safety arms are above the core and larger withdrawal steps are needed to achieve the same negative reactivity insertion." In this context, it would appear the word "negative" should be "positive."

Agreed, will revise as appropriate..

4.47 Section 4.5.1.5.1, The Shim Safety Arms, p. 4-34. The second paragraph states "The maximum calculated rate is 4.5x10-4 (% k/k)/s. The technical specifications limit the rate to 5.0x10-4 (% k/k)/s." The Technical specifications use the reactivity units /s. Clarify the difference between these units and those used in the TS.

Agreed, will revise as appropriate.. The TS limit is $5.0x10^{-4}$ $\Delta\rho$/s, not %$\Delta\rho$/s. See the response to RAI 4.18.

4.48 Section 4.5.2.3.3, Hot Channels and Hot Spots from the Updated MCNP Model, p. 4-42. The last paragraph states "The rate of consumption of 235U is 1.17 times the fission rate, or 7.1 x 1018 fis/cm3/day." Clarify if this value and the appropriate units represent the average fission rate (fis/cm3/day) or absorption rate (abs/cm3/day).

Agreed, will revise as appropriate.. The reference is to the absorption rate, not the fission rate.

449 Section 4.5.3.1.2, Moderator Dump, p. 4-46. In the first paragraph under "Basis" the phrase "with one shim arm know to be inoperable," is apparently missing an "n" in "known" as in "with one shim arm known to be inoperable."

Agreed, will revise as appropriate..

4.50 Section 4.5.3.1.2, Moderator Dump, p. 4-46. In the second paragraph under "Basis" the sentence beginning "The analysis showed that the most sever accident...," is apparently missing an "e" in "severe" as in "The analysis showed that the most severe accident...."

Agreed, will revise as appropriate..

4.51 Section 4.5.3.3, Safety Limits and Limiting Safety System Settings, p. 4-49. The "Basis" section for TS 2.2 uses the term "burnout ratio" whereas the term "Critical Heat Flux Ratio" is used on the previous page under section 4.5.3.3.1. When practical, use consistent terminology between the SAR and TS.

Agreed, will revise as appropriate..

4.52 Section 4.6.1.2, Power Distribution in the Core, p. 4-51. In this section, the terms "horizontal strips" and "vertical strips" are used. Clarify the use of these terms as compared to the terms "slices" and "stripes" defined previously.

Agreed, will revise as appropriate..

4.53 Section 4.6.2.2 & 4.6.2.3, Departure from Nucleate Boiling & Onset of Flow Instability, p. 4-53. The definition of the term "Ts" (both sections) is given as "saturation pressure." It would appear from the context this term should be "saturation temperature."

Agreed, will revise as appropriate..

4.54 Section 4.6.3, Determination of Limiting Conditions, p. 4-54. The pressure at the hot spot is estimated as "3.34m D2O, or 138.5 kPa, or 1.37 bar." The conversion from kPa to bar is 1 bar = 100 kPa, so these numbers appear inconsistent.

Agreed, will revise as appropriate..

4.55 Section 4.7, References, p. 4-58. Correct the date in the reference for "NIST Center for Neutron Research (s004b)."

Agreed, will revise as appropriate..

4.56 Table 4.5.5, p. 4-67. In the second column, i, the values appear to be in percentage units, i.e. i (%).

Agreed, will revise as appropriate..

4.57 Table 4.6.1, p. 4-72. Check the grammar in the statement "These are the minimum flows to assure that there be no nucleate boiling at any point in the core."

Agreed, will revise as appropriate.

Chapter 5: Technical Questions

5.2 Section 5.2.14.1, p. 5-17. The SAR states "Maintaining the integrity of the fuel cladding requires that it should remain below its melting temperature." The limiting criteria appears to be "blistering" temperature, as is stated in the next sentence. Provide clarification on the use of "melting temperature" vice "blistering."

The blistering temperature is the criterion.

5.3 Section 5.2.14.2, p. 5-17. This section states that if "all three parameters simultaneously reach their safety-system settings, the burnout ratio is at least 1.3." Provide reference to where in the SAR or elsewhere this analysis is performed or provide an analysis that demonstrates a burnout ratio of 1.3 given those conditions.

The analysis of safety limits is given in SAR 4.6.3, where the value of 1.3 is listed under the conditions used in the analysis. The analysis is based upon maintaining the CHFR (burnout ratio) greater than 1.3.

5.4 Section 5.2.14.3, p. 5-18. The second paragraph states "Under this condition, the hot spot of the hottest plate remains below 160 °F (70 °C) (Chapter 13, Accident Analyses)." Provide reference to where in the SAR or elsewhere the corresponding analysis and results are presented supporting this temperature and explain if this temperature is consistent with values in Table 5-5, p. 5-18 of Chp. 13.

The analysis was performed in Appendix A, and does not support the statement in Chapter 5, Section 5.2.14.3. The data presented in Table 5-10 on page 5-18 of Appendix A shows that the fuel centerline temperature remains below 135 °C following loss of off-site power. The analysis in Appendix A is correct, and the reference in Chapter 5 is incorrect.

5.5 Section 5.2.14.3, p. 5-18. The second paragraph states "Further, analyzing the case of no-shutdown cooling flow (Chapter 13, Accident Analyses), the maximum temperature of the fuel plate would be less than 500 °F (260 °C), well below the temperature that would cause any damage." Provide reference to where in the SAR or elsewhere the corresponding analysis and results are presented supporting this temperature. Explain if this temperature is consistent with values in Table 5-10, p. 5-23 of Chp. 13, and with the temperature cited in TS 3.2 as 107 °C (225 °F).

This accident starts from the loss of off-site power scenario discussed in question 5.4. The maximum temperature occurs at very short times as shown in Table 5-5, Appendix A, at the time of the reactor scram. Table 5-10 shows that the fuel plate temperature remains below 110 °C at longer times when shutdown cooling is also lost. Therefore, this accident is bounded by the simple loss of shutdown cooling. However, the statement in Section 5.2.14.3 that the temperature remains below 500 °F is consistent with the fact that the temperature remains below 275 °F (135 °C) for this scenario.

5.6 Section 5.3.2.1.2, p. 5-21. This paragraph states "At flows of 65 gpm (250 lpm) on the primary side...," while Section 5.4.2.3, p. 5-35, states "At flows of 35 gpm (132 lpm) on the primary side...." Both are apparently referring to the D2O Purification Heat Exchanger (HE-2). Clarify the difference between these flow rates.

Flow through the primary side of HE-2 is 35 gpm (132 lpm).

5.7 Section 5.3.2.5, p. 5-24. The first paragraph states "The 150 psi (1 MPa) air to operate the pneumatic control valves...." Similar wording appears in Section 5.4.2.6, p. 5-36. Chapter 9, p. 9-12, states "The NBSR is supplied with a source of 100 psig (680 kPa) air from the main NIST compressed air facility." Clarify the difference between these air pressures.

Air is supplied to pneumatic control valves at a nominal pressure of 100 psig (690kPa).

5.8 Section 5.3.8.1, p. 5-32. This paragraph states "Using this value, the limits ensure that tritium concentrations in effluents will be as low as practicable, and below concentrations allowed by 10 CFR 20.303 for liquid effluents and 10 CFR 20.106 for gaseous effluents (Chapter 11, Radiation Protection and Waste Management)." Explain the applicability of references to 10 CFR 20.303 and 10 CFR 20.106 in both the SAR and the TS, or update these references to current regulatory requirements, as applicable.

Section as written:

5.3.8.1 Technical Specification 3.6, Secondary Cooling System

This Technical Specification applies to the Main Heat Exchangers in the Primary Coolant System. Its objective is to maintain tritium releases as low as practicable. The reactor is required to be shut down and corrective action taken if the leakage of primary coolant through a heat exchanger to the secondary system exceeds the daily, weekly, and yearly limits established in the specification. Using this value, the limits ensure that tritium concentrations in effluents will be as low as practicable, and below concentrations allowed by 10 CFR 20.303 for liquid effluents and 10 CFR 20.106 for gaseous effluents (Chapter 11, Radiation Protection and Waste Management). The specified daily and weekly leakage rates represent the lowest limits of positive detection of D_2O losses under both operating and shutdown conditions. The specified yearly leak rate represents an estimate of the smallest sized leak that can be positively located and repaired.

Response:

The citations referenced here, 10 CFR 20.303 for liquid effluents and 10 CFR 20.106 for gaseous effluents, are outdated and not applicable. The sentence should read, *"Using this value, the limits ensure that tritium concentrations in effluents will be as low as practicable, and below concentrations allowed by 10 CFR 20.2003 and 10 CFR 20.1302 for liquid effluents and 10 CFR 20.1302 for gaseous effluents (Chapter 11, Radiation Protection and Waste Management)."*

5.9 Section 5.3.8.2, p. 5-33. The 2nd and 3rd paragraphs mention a "36 gallon/day" value regarding primary to secondary leakage. The TS uses 40 gpd for minimum sensitivity in surveillance TS 4.5. Clarify the difference between the leakage rate sensitivity values.

NBSR Technical Specifications, revised version , replaces these leakage rates with a primary tritium concentration limit.

5.10 **Section 5.4.2, p. 5-34. In the 3rd paragraph, the last sentence states "Consequently, the minimum time to treat all of the primary coolant is approximately 21 ½ hours." Provide analysis to support the treatment time.**

The treatment time is insignificant. The reference to treatment time will be deleted.

5.11 **Sections 5.7.2.1 & 5.7.2.2, p. 5-42. The heat load is specified as "1.54 x 105 Btu/hr" and the heat sink is specified as "60 x 103 Btu/hr." Explain how these two values relate to one another.**

HE-2 has a heat removal capacity of 3.2 x 10^6 Btu/hr (1000 kW). The steady state, total
system heat load on HE-2 is approximately 420 x 10^3 Btu/hr (120 kW).

5.12 **Section 5.2.2.6.2, p. 5-8. The temperature ranges for TR-2, TR-3, TR-4 and TR-5 have inconsistent temperature ranges listed as the values for Fahrenheit and Celsius. Provide clarification as to which are the correct values and the appropriate temperature range conversions.**

All ranges are nominal. The range of TRA-2 is 50-200 °F (10 to 93 °C). The range for TRCA-3 is 50-130 °F (10 to 55 °C). The range for TRA-4 is 50-150 °F (10 to 66 °C). The range for TR-5 is 50-150 °F (10 to 66 °C) TR-5. The range for TR-1 is 0-20°F (-18 to -7 °C).

5.13 **Section 5.2.2.7.1, p. 5-10. Provide clarification describing methods used to preclude the introduction of objects into the primary coolant system during maintenance associated with removal of the strainer.**

Good engineering practices are used for the strainer inspection and any other opening of the main coolant system.

5.14 **Section 5.3.2.5, p. 5-24. Provide clarification on the response of the pneumatically positioned secondary valves to a loss of instrument air.**

No failure in the secondary system can credibly cause a reactor accident due to loss of secondary cooling. Therefore, the failure of any pneumatically positioned secondary valve, including a failure due to loss of air, has no affect upon reactor safety.

5.15 **Section 5.2.4.3, p. 5-11. In the 2nd paragraph, the phrase "and a reactor scram occur due to...," is apparently missing an "s" at the end of "occur."**

Agreed, will revise as appropriate.

5.16 **Section 5.2.14.3, p.5-18. The third paragraph states "Calculations show that tritium releases offsite are below concentrations allowed by 10 CFR 20 (Chapter**

11, Radiation Protection and Waste Management)." TS 3.2 references Chapter 13 for these calculations. Clarify the difference between the locations of the supporting calculations.

The supporting calculations are in Chapter 13.

5.17 Section 5.3.2, p. 5-20. In the 3rd paragraph, the word "Deminerizer" appears incorrect.

The correct spelling is "… Demineralizer …"

5.18 Section 5.3.2.8, p. 5-29. In the 2nd paragraph, in the phrase "on room D100" it appears the word "on" should be "in."

Agreed, will revise as appropriate..

5.19 Section 5.3.8.2, p. 5-32. This paragraph states "It also requires that, when the N-16 monitor is inoperable, the secondary cooling water is sampled and analyzed for tritium at least monthly." The word "inoperable" should apparently be "operable" to agree with the TS.

Agreed, will revise as appropriate..

5.20 Section 5.4.2.5, p. 5-36. In the 1st paragraph, should "cellulose, acetate cartridges" be hyphenated as in "cellulose-acetate cartridges"?

The reference to the type of cartridge will be deleted.

5.21 Section 5.7.2.1, p. 5-42 & Section 5.7.2.6.1, p. 5-43. Two uses of nomenclature appear inconsistent with the "Cold Neutron Source" terminology used elsewhere.

Heavy water cooling flow is for the cryostat assembly.

5.22 Section 5.7.2.6.2, p. 5-43. In the 1st paragraph, the phrase "thermowell located the 1 ½-inch (3.8 cm) piping" appears to be missing an "in."

Agreed, will revise as appropriate.

<u>Chapter 6: Technical Questions</u>

6.1 Section 6.1.1, p. 6-1. The first paragraph states "a minimum of 28 minutes of coolant flow is always available to the core from the Inner Reserve Tank…." In Chapter 13, Appendix A, p. 5-8, the last paragraph states "For at least 20 minutes after shutdown the tank flow is more than adequate to cool the fuel elements by boiloff." Clarify these statements regarding the amount of cooling time that would be provided by the IRT.

In Figure 5-6 of Appendix A, the flow rate is shown as a function of time, along with the flow required to remove decay heat. The flow remains above that required until 1700 seconds, which is more than 28 minutes. Both statements are accurate (28 minutes is longer than 20 minutes), but the statement on page 5-8 of appendix A is overly conservative.

6.2 Section 6.2.1.2.1, p. 6-9. The last sentence in the second paragraph states "The water makeup capacity must be in excess of 25 gpm (95 lpm), which was calculated as adequate to prevent fuel damage." Provide an analysis and discussion of how this value was determined and compare with the flow from the D2O Storage Tank and the Emergency Sump Pump during a loss of coolant accident.

In figure 5-6 of Appendix A, the flow required for adequate cooling is shown to be approximately 0.5 kg/s or less than 8 gpm. The value given in Section 6.2 was based on an earlier overly conservative analysis, and is incorrect. The correct analysis is based upon the results in Figure 5-6 of Appendix A, which shows that the flow from the storage tank is adequate for more than 28 minutes. The sump pump flow is sized to provide 40 gpm flow, and is adequate to provide cooling at all times.

6.3 Section 6.2.3, p. 6-13. Explain why the flowrates on Figure 6.4 are different from those on Figure 6.5 and the description on pp. 6-13 & 6-14.

All references to re-circulated process room air and winter/summer rates are deleted, as these modes of operation no longer exist. There are now configurations for maximum air flow and normal air flow for the basement ventilation system. Ventilation drawings now depict design flow rates and descriptions match the existing equipment.

6.4 Section 6.2.3.3.4, p. 6-18. The second paragraph states "The height of approximately 100 feet (30 meters) above grade level was chosen to meet the criteria of dilution and reduced potential exposure." Describe how the stack height compares to the guidance in Regulatory Guide 1.111 and GEP stack height criteria for elevated releases. If corrections are required also apply the corrections to all affected analyses.

The stack is 100 feet (30 meters) tall. The previous details were superfluous.

6.5 Section 6.2.3.3.5, p. 6-19. The third and fourth paragraphs state that the Emergency Exhaust Fan motors (AC and DC) for EF-5 & EF-6 are powered from MCC DC. It appears from Chapter 8 that the power source for the AC motors is the A5 emergency bus. Explain and differentiate the power source and switchgear locations for these motors.

The AC motors' power source is Miscellaneous Power Panel on MCCA5 and the DC motors' power supply is the MCCDC. All of the controllers are in the MCCDC cabinet because of space limitations at the time of installation.

6.7 **Section 6.1.1, p. 6-1. The first sentence appears to contain a typo in "Figures 6.1."**

Agreed, will revise as appropriate..

6.8 **Section 6.1.1, p. 6-2. The third paragraph apparently contains a typo in the phrase "this tank will start draining though the two nozzles."**

Clarification will be provided so it is understood that *this tank* is the inner reserve tank.

6.9 **Section 6.1.2, p. 6-2. The first paragraph apparently contains a typo in the phrase "power distribution gears."**

Agreed, will revise as appropriate..

6.10 **Section 6.2.3.2.2, p. 6-17. The first sentence in the second paragraph lists "filters F-26, F-27, F-59 in subsystem A." Figure 6.4 shows F-26, F-27, and F-57. Clarify the apparent mismatch.**

The correct filter designations are F-26, F-27, and F-57.

6.11 **Section 6.2.3.2.2, p. 6-17. The sentence "Since one of the two trains is in operational during an emergency...," apparently contains a typo.**

Agreed, will revise as appropriate..

6.12 **Section 6.2.3.3.4, p. 6-18. The first paragraph states "discharge from Reactor Basement Exhaust System fan EF-27 through ACF-3." The "ACF-3" is apparently a typo for "ACV-3."**

Agreed, will revise as appropriate..

6.13 **Section 6.2.3.4.4, p. 6-21. The last sentence uses the acronym "WSSC." Spell out the abbreviation on first use and add to the "acronyms" list.**

Agreed, will revise as appropriate.

Chapter 7: Technical Questions

Part I: Technical Questions and Comments:

7.1 Section 7.2.1, p. 7-5. Explain why primary coolant temperature is absent from the list of main parameters which are monitored and provide inputs to the logic chains.

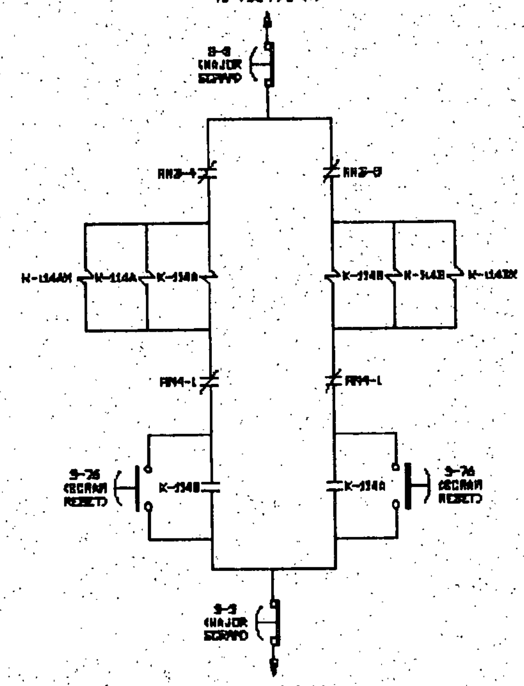
Existing Language: 1. Primary coolant flow, level, and pressure;
Revised Language: 1. Primary coolant flow, level, and differential temperature;
Explanation: This is a typographical error. The second bullet under Item 1 on page 7-4 lists Reactor ΔT as an input to the RPS that initiates a reactor scram. We do not monitor primary coolant pressure.

7.2 Section 7.2.3, p. 7-10. Provide a schematic of the control logic for confinement building isolation, i.e., door scram relays, fan scram relays, ventilation system alignment, etc.

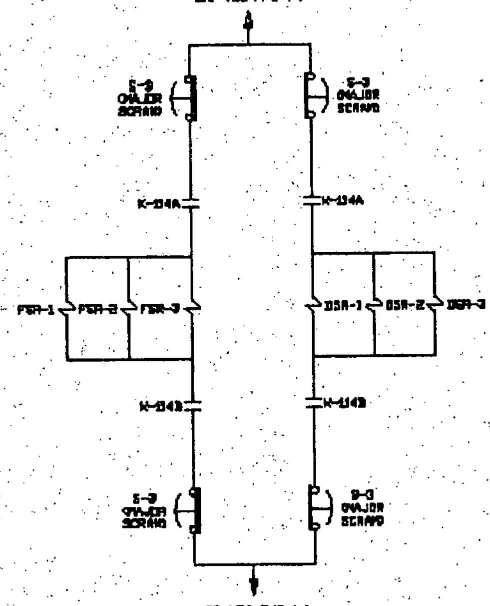
Response: The text from Section 7.2.3 on page 7-10 is repeated here: The Major Scram function is part of the RPS. The Normal Air Monitor Channel, Irradiated Air Monitor Channel and the Stack Monitor Channel control relays in the Major Scram circuit. Upon the detection of an excessive activity level by any of the three channels, the Major Scram relays scram the reactor and initiate Confinement Building isolation. The Major Scram relays open contacts in the Scram logic string, thereby initiating a reactor scram. The relays also shut the doors at the entrances to the Confinement Building by tripping the Door Scram Relays (DSR), shift the ventilation lineup to recirculation mode by tripping the Fan Scram Relays (FSR), and close the Neutron Guide Isolation Valves. A new simplified schematic of the control logic is found below.

7.3 Section 7.3.1.2, p. 7-16. Provide an explanation of the "all rods seated" contacts and the purpose of this interlock.

Existing Language: Contacts associated with the Emergency Cooling Tank Level Indicator Alarm Channel LIA-2, Reactor Vessel Overflow Indicator Channel FIA-2, Shim Safety Arm Clutch Current, Nuclear Instrument Test Fault, and Process Instrument Test Faults are wired in series with the Startup Prohibit relays. This ensures that the level in the Emergency Tank is in its normal operating range, that there is flow of primary coolant through the overflow line and that the Shim Safety Arm clutches are energized before any reactivity control devices can be withdrawn.
Revised Language: Contacts associated with the Rods Not Seated Alarm AN4-42, Emergency Cooling Tank Level Indicator Alarm Channel LIA-2, Reactor Vessel Overflow Indicator Channel FIA-2, Shim Safety Arm Clutch Current, Nuclear Instrument Test Fault, and Process Instrument Test Faults are wired in series with the Startup Prohibit relays. This ensures that all reactivity control devices are fully inserted, the level in the Emergency Tank is in its normal operating range, that there is flow of primary coolant through the overflow line and that the Shim Safety Arm clutches are energized before any reactivity control devices can be withdrawn.
Explanation: This was a typographical error and the missing words have been added. The Startup Prohibit Circuit requires that the reactor be in a specific known state prior to the operator being able to reset the scrams and commence reactor startup. Before being able to reset the scram relays the circuit ensures that the four shim arms and the regulating rod be fully inserted.



MAJOR SCRAM CIRCUIT



DOOR AND FAN SCRAM CIRCUIT

FSR AND DSR RELAY SCHEDULE

FSR-1
- SF-11 N.O.
- EF-27 N.O.
- EF-28 N.O.
- EF-4 N.O.
- SF-3 N.O.
- SF-10 N.O.

FSR-2
- EF-8 N.O.
- EF-4 N.O.
- SF-3 N.O.
- SF-1 N.O.
- SF-2 N.O.
- SF-12 N.O.
- EF-6 DC N.C.
- EF-8 AC N.C.
- TRUCK DOOR N.C.
- 1801/8 DOOR N.C.

FSR-3
- SF-1 N.O.
- SF-2 N.O.
- SF-19 N.C.
- EF-6 AC N.C.

DSR-1
- 049/8 DOOR N.C.
- EF-8 N.O.
- SF-11 N.O.
- EF-27 N.O.
- EF-3 N.O.
- SF-19 N.C.
- 1806/8 DOOR N.C.
- 12122 DOOR N.C.
- 1 SPARE N.O.

DSR-2
- RVV-15 N.O.
- RVV-18 N.O.
- EF-8 DC N.C.
- 1212/2 DOOR N.C.

DSR-3
- EF-23 N.O.
- RUV-1, RUV-2 RUV-3 N.O.
- RVV-14 N.O.
- RVV-16 N.O.

Part II: Editorial Questions and Comments:

7.4 Section 7.2.3, p. 7-9. In the 4th paragraph, the 1st sentence refers to Figure 7.7 and the "relay logic ladder." It appears that this paragraph is referring to the logic diagram in Figure 7.8. If this is true, check and correct the subsequent references to Figure 7.7 in this chapter, as appropriate.

Existing Language: The Reactor Safety System, shown in Figure 7.7, is a hardwired relay logic ladder with multiple inputs and multiple functions.
Revised Language: The Reactor Safety System, shown in Figure 7.8, is a hardwired relay logic ladder with multiple inputs and multiple functions.
Explanation: The text referenced the wrong figure.

7.5 Section 7.3.3.1, p. 7-19, Item 6. (2) and the definition of Reactor Shutdown in the TS are not the same. Clarify the difference between the wording in the two locations.

Existing Language: (2) The reactor control power and the rod drive power key switch are locked in their "off" position.
Revised Language: (2) The reactor control power and the rod drive power key switches in their OFF position with their keys removed.
Explanation: The wording of the two passages cited is essentially identical but for clarity's sake, the two will read the same.

7.6 Section 7.3.3.1, item 8, top of page 20. TS definition 1.3 includes an item (4) "Moderator Dump." Clarify the difference between the wording in the two locations.

Existing Language: No item (4) currently included in Section 7.3.3.1, item 8.
Revised Language: (4) Moderator dump.
Explanation: This was inadvertently omitted from the original draft and will be added.

7.7 Section 7.3.3.2, p. 7-20. In the 1st paragraph, clarify that the 3rd item is intended to be operable 'in accordance with' Table 3.1 of the TS.

Existing Language: 3 The Scrams and Major Scrams are operable Table 3.1 of the Technical Specifications (NBSR 15); and
Revised Language: 3 The Scrams and Major Scrams are operable in accordance with Table 3.1 of the Technical Specifications (NBSR 15); and
Explanation: This phrase was inadvertently omitted from the original draft and will be added.

7.8 Section 7.3.3.2, p. 7-20. In the 3rd paragraph, check and correct the wording and grammar in the 1st sentence "A rod withdrawal accident for the NBSR has been analyzed and are discussed Chapter 13 and Appendix A of this SAR...."

Existing Language: A rod withdrawal accident for the NCSR has been analyzed and are discussed Chapter 13 and Appendix A of this SAR using the maximum insertion rate, corresponding to the maximum beginning-of-life rod worths with the rods operating at the design speed of their constant speed mechanisms.

Revised Language: A rod withdrawal accident for the NCSR has been analyzed and is discussed in Chapter 13 and Appendix A of this SAR using the maximum insertion rate, corresponding to the maximum beginning-of-life rod worths with the rods operating at the design speed of their constant speed mechanisms.
Explanation: These were typographical errors and the wording and grammar have been corrected.

7.9 Section 7.4.1, p. 7-23. In the 2nd paragraph, check and correct the wording and capitalization in the sentence "A minimum of one decade of overlap is designed into the transition between the Source Range and Intermediate Range Nuclear Instrumentation and between Intermediated Range and Power range Nuclear Instrumentation." In the following sentence, check and correct the use of the word "form" in "channels form the source range."

Existing Language: A minimum of one decade of overlap is designed into the transition between Source Range and Intermediate Range Nuclear Instrumentation and between Intermediated Range and Power range Nuclear Instrumentation. The degree of overlap between the channels form the source range to full power operation is shown in Figure 7.5, Flux Coverage of the NBSR.
Revised Language: A minimum of one decade of overlap is designed into the transition between Source Range and Intermediate Range Nuclear Instrumentation and between Intermediate Range and Power Range Nuclear Instrumentation. The degree of overlap between the channels from the source range to full power operation is shown in Figure 7.5, Flux Coverage of the NBSR.
Explanation: These were typographical errors and the wording and grammar have been corrected.

7.10 Section 7.4.1, p. 7-24. In the 1st sentence on p. 7-24, check and correct the usage of "from" and "the" in the second line, "power from directly from the the +/- 10Vdc."

Existing Language: The Source Range and Power Range channels as well as the Nuclear Safety System receive their power from directly from the the ±10 Vdc Nuclear Instrument Power Bus.
Revised Language: The Source Range and Power Range channels as well as the Nuclear Safety System receive their power directly from the ±10 Vdc Nuclear Instrument Power Bus.
Explanation: These were typographical errors and the wording and grammar have been corrected.

7.11 Section 7.6.1, p. 7-26. Check and correct the word "inn" in the sentence beginning, "The instrument panels inn the control room display...."

Existing Language: The instrument panels inn the control room display the nuclear and process variables required by the operator for reactor operation.
Revised Language: The instrument panels in the control room display the nuclear and process variables required by the operator for reactor operation.
Explanation: This was a typographical error and the spelling has been corrected.

7.12 Section 7.6.3, p. 7-27. In the 5th paragraph, the last sentence appears to be missing a "the" before "reactor operator."

Existing Language: Alarm Panel AN-2, located on Panel F, presents individual alarm windows that alert reactor operator to the status of conditions in the Auxiliary Systems.
Revised Language: Alarm Panel AN-2, located on Panel F, presents individual alarm windows that alert the reactor operator to the status of conditions in the Auxiliary Systems.
Explanation: This was a typographical error and the wording and grammar have been corrected.

7.13 Sections 7.7.1 through 7.7.5, pp. 7-30 & 7-31. There are multiple references to Appendix 8 (8A, 8H, 8I, 8J, 8E, 8G, 8F). Explain or correct the use of these reference numbers.

Existing Language:

In Section 7.7.1:	(see Appendix 7A, Appendix 8A and Table 7.B.2)
In Section 7.7.2:	(see Appendix 7A and Appendix 8A)
In Section 7.7.3:	The instrument channels are described in detail in Appendix 7A-8H, 8I, and 8J.
In Section 7.7.4:	The instrument channels are described in detail in Appendix 7A-8E and 8G.
In Section 7.7.5:	(see in Appendix 7A-8F)

Revised Language:

In Section 7.7.1:	The instrument channels are described in detail in Appendix 7A Section 8, Radiation.
In Section 7.7.2:	The instrument channels are described in detail in Appendix 7A Section 8, Radiation.
In Section 7.7.3:	The instrument channels are described in detail in Appendix 7A Section 8, Radiation.
In Section 7.7.4:	The instrument channels are described in detail in Appendix 7A Section 8, Radiation.
In Section 7.7.5:	The instrument channels are described in detail in Appendix 7A Section 8, Radiation.

Explanation: These references were inaccurate and the proper references have been added.

7.14 Section 7.7.3, p. 7-30. In the last line, check "AN47" for correctness.

Existing Language: Remote indication and alarms (AN47, AN4-6, and AN4-8, respectively) are provided on the Main Control Panel in the Control Room.
Revised Language: Remote indication and alarms (AN4-7, AN4-6, and AN4-8, respectively) are provided on the Main Control Panel in the Control Room.
Explanation: This was a typographical error and the wording has been corrected.

7.15 Section 7.8, p. 7-32. Explain the use and applicability of the ANSI/ANS 15.20 standard for the NBSR I&C system design.

ANSI/ANS 15.20 implementation is no different than any of the other ANS 15 standards. The standard is not met to be used as a demand model for back fitting purposes but should be helpful for the facility undergoing changes or modifications; and it use should ease the burden of regulatory agencies.

7.16 Section 7.8, p. 7-32. The IEEE Standard 7-4.3.2 title appears to contain an extra "Systems" after "Computers."

Existing Language: Institute of Electrical and Electronics Engineers, IEEE Standard 7-4.3.2, "IEEE Standard Criteria for Digital Computers Systems in Safety Systems of Nuclear Power Generating Stations," Piscataway, New Jersey, 1993.
Revised Language: Institute of Electrical and Electronics Engineers, IEEE Standard 7-4.3.2, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," Piscataway, New Jersey, 1993.
Explanation: This was a typographical error and the wording has been corrected.

7.17 Table 7.5B, p. 7-35. Check and correct the 1st column heading in the table.

Existing Language: MCP Panel A
Revised Language: MCP Panel B
Explanation: This was a typographical error and the column heading has been corrected.

7.18 Table 7.5G, p. 7-41. Check and correct the range on the D2O IX Inlet/Outlet Conductivity Recorder.

Existing Language: 0-μS
Revised Language: 0-2μS
Explanation: This was a typographical error and the recorder range has been corrected.

7.19 Table 7.5G, p. 7-41. It appears there are several instances of "HE" that should be "He", i.e., to represent helium instead of heat exchanger.

Explanation: There is no error. HE is used for both helium and for heat exchanger.

7.20 Table 7.5I, p. 7-43. The 1st column lists "Storage Pool IX Inlet/Outlet Conductivity" and "Thermal Shield Inlet/Outlet Conductivity." It appears that there are only "Outlet" instruments.

Existing Language: Storage Pool IX Inlet/Outlet Conductivity; Thermal Shield Inlet/Outlet Conductivity.
Revised Language: Storage Pool IX Outlet Conductivity; Thermal Shield Outlet Conductivity.
Explanation: This was an error and the instrument names have been corrected. Engineering Change Notice (ECN) number 444 replaced obsolete temperature and conductivity channels with modern equipment. During this replacement, these two ion exchanger inlet conductivity channels were deleted as they were found to be redundant with the system conductivity channels.

7.21 Table 7.7B, p. 7-46. The nomenclature of the 1st column header "Cubicle" appears to be incorrect.

Existing Language: Cubicle
Revised Language: Breaker
Explanation: This was a typographical error and the nomenclature of the first column header has been corrected.

7.22 Figures 7.4B & 7.4C, p. 7-54 & 7-55. The figure titles appear to be backwards for these two figures, i.e. "Intermediate Range Channel" should go with Figure 7.4c and "Power Range Channel" with Figure 7.4b.

Existing Language: n/a
Revised Language: n/a
Explanation: The titles are correct for Figure 7.4b and 7.4c but the figures associated with them were inadvertently swapped. This was done in the original draft and will be corrected in the final version.

7.23 Appendix 7A, Section 5, p. 7-75. In the second paragraph, the source range channels in the last line are referred to as "ND-1 and ND-2." These appear to be typos for "NC-1 and NC-2."

Existing Language: Triaxial signal cables are used for the source range channels ND-1 and ND-2.
Revised Language: Triaxial signal cables are used for the source range channels NC-1 and NC-2.
Explanation: This was a typographical error and the source range channel designators have been corrected.

7.24 Appendix 7A, Section 5, p. 7-77. In the 3rd paragraph, the 1st sentence appears to be missing an "of" after "rate of change."

Existing Language: The Period portion of the drawer calculates the rate of change reactor power.
Revised Language: The Period portion of the drawer calculates the rate of change of reactor power.
Explanation: This was a typographical error and the wording has been corrected.

7.25 Appendix 7A, Section 7, p. 7-83. In number 13, check and correct the units "pisg" in the last sentence.

Existing Language: The range of the channel is 0-15 pisg.
Revised Language: The range of the channel is 0-15 psig.
Explanation: This was a typographical error and the unit has been corrected.

7.26 Appendix 7A, Section 8, p. 7-88. In number 11, the last sentence is apparently missing a "than" after "rather."

Existing Language: This channel is primarily used for qualitative rather quantitative analysis.
Revised Language: This channel is primarily used for qualitative rather than quantitative analysis.
Explanation: This was a typographical error and the missing word has been added.

Chapter 9: Technical Questions

9.1 During the orientation tour, it was noted that neutron shielding for the cold neutron source and neutron guides consists of lead shot mixed in paraffin. The quantity of shielding material was significant. The paraffin is both a large transient combustible Part I: Technical Questions and Comments: load, but also can melt and pool resulting in more dangerous fires. The SAR does not mention the paraffin as a flammable material that is present even though it is most likely the largest single combustible source in the confinement building. Provide a description of the paraffin in the shielding blocks and the design features that prevent or mitigate its involvement in a fire.

Response: The shielding for the cold neutron source and neutron guides consists of paraffin and shot filled steel assemblies. Various shaped shields are assembled to form the walls and roofs over the cold neutron source and the neutron guides. Each individual shielding assembly is formed from ¼ in. steel plates welded together to form a leak tight volume. During the fabrication of each shield, hot, liquid paraffin and shot are poured into each assembly through a limited number of 7 in. diameter fill holes located on the top of each shield. Once cooled, the paraffin solidifies and holds the shot in place.

While the quantity of shielding material may be significant, the design and placement of each piece around the cold neutron source or the neutron guides ensure that the paraffin cannot leak from a shield assembly and pool on the floor, acting as a source for any fire that might occur. The design of each piece of shielding not only ensures that any paraffin that does melt is contained by the steel plates of the assembly but also that only a small surface areas is exposed to any flame. Early in the process of developing the structure of the shields used in either the Confinement Building or the Guide Hall, the director of the NCNR consulted with the NIST Fire Chief on the use of paraffin in the shield assemblies. It was determined that the design minimized the chances of the paraffin acting as a source in any fire.

9.2 Section 9.2.4.1, p. 9-7. Provide justification for the extrapolation used to determine the minimum time a fuel element must remain submerged in the primary coolant prior to transfer. Include discussion/analyses of power distribution for both the 10 MW core and the 20 MW core, decay heat for worst-case fission density and irradiation time for fuel elements in both the 10 MW core and the 20 MW core, and any assumptions made and all uncertainties (measurement, instrumentation, fabrication, etc.) in all relevant analyses. The discussion/analyses should clearly show that nowhere will the local clad temperature of a worst-case-irradiated fuel element immersed in helium reach 450 C.

Response: In an ORNL report[1] on fuel element handling at the 20 MW (at that time) ORR, one element, ORR #164 was extensively analyzed and reported. This element operated at a fission power of 770 kW, and was studied 19.25 hours after reactor

[1] "Surface temperatures of Irradiated ORR Fuel Elements Cooled in Stagnant Air", J. F. Wett, Jr., ORNL-2892 (1960)

shutdown, following a total irradiation time of 536 hr. The highest temperature measured with the element hanging in stagnant air inside a dry box was 640 °F or 338 °C, well below 450 °C. ORNL used the Way-Wigner result to calculate element powers:

$$\frac{P}{P_0} = 6.22x10^{-2}\left[t^{-0.2} - (T+t)^{-0.2}\right]$$

For element 164, this implies a power of 2.5 kW, while if the irradiation time were infinite, then the power 19.25 hours after shutdown would be 5.2 kW.

Chapter 10: Technical Questions

10.1 Section 10.3 of the SAR references TS 6.2(2) and 6.2(3) regarding the requirement of the SEC to review experimental proposals. Verify that these are the correct references and change the references if appropriate.

The correct reference is TS 6.5 of NBSR Technical Specifications.

Chapter 11: Technical Questions

11.1 Section 11.1.1.2, p. 11-3. The dose limit to members of the general public due to airborne effluents is 10 mrem/yr (10 CFR 20.1101(d)). Revise this section to reflect the appropriate dose limit.

Section as written:

11.1.1.2 Airborne Radiation Sources

The principal airborne sources of radioactivity associated with the operation of the NBSR are ^{41}Ar and tritium (^3H). The only release path for air from the various confinement building ventilation systems is via the building stack exhaust, which has a nominal flow rate of 30,000cfm (850 m^3/sec). Annual emissions of ^{41}Ar typically ranges from 800 to1200 Ci and ^3H ranges from 400 to 800 Ci. This constitutes a dose of less than 2 mrem of exposure to the closest member of the public, which is less than 2% of the NRC dose limit to the public. This analysis was performed with the EPA COMPLY computer code using local wind rose data and computing the dose based on the closest resident in each wind sector, which constitutes conservative analytical boundary conditions. Monitoring in both the stack and in the building ventilation systems utilizes both installed and periodic sampling. This provides redundant methods for assessing both occupational and public exposure. Occupational exposure is discussed below.

Response:
The dose limit for individual members of the public is stipulated in 10 CFR 20.1301. The referenced citation, 10 CFR 20.1101(d), establishes an ALARA constraint for air emissions of radioactive material to the environment such that the highest exposed member of the public will not be expected to receive more than 10 mrem per year from these emissions. This is an ALARA goal and not a dose limit. The dose limit referenced in section 11.1.1.2 is correct.

Also, 850 m^3/sec should be corrected to 850 m^3/min.

11.2 Section 11.1.1.4.2, p. 11-9. Provide more detail in this section to clarify actions related to the disposal of the shim arms. Briefly describe the processes used to remove the shim arms from the reactor vessel (mechanical detachment and physical transfer), including discussions of ALARA practices, and the location where the reactor shims decay for three months.

Section as written:
11.1.1.4.2 Reactor Shims

Control shims are the only other high activity component routinely removed from the reactor. This occurs usually every 4 to 5 full-power years. After a minimum decay period of 3 months, the stainless steel hubs are separated from the Cd-Al body and shipped with the other radioactive non-fuel element metal pieces. The Cd-Al shim body is stored in the storage pool or in shielded dry storage wall cavities. Radioactivity for the hub is typically about 20-25 Ci, and the shim body is typically less than 1 Ci. Personnel exposure when performing preparatory operations for shipping is controlled through the use of shielding and the delay between when the shims are removed from the reactor and the preparation of the offsite shipment begins.

Response:
The control shims discussed in this section, as stated, are routinely removed from the reactor typically every 4 to 5 full-power years. This process is accomplished by procedure under the control of a radiation work permit with strict adherence to ALARA policy. All disassembly work is conducted from the reactor top area through access ports to the reactor vessel, using extended

tools. Transfer operations are done remotely using cameras and remote crane controls from a shielded location when practical and possible. All unnecessary personnel are removed from affected areas during this work. When performing preparatory operations for shipping, personnel exposure is controlled through the use of shielding and the delay between when the shims are removed from the reactor and the preparation of the offsite shipment begins.

After allowing for a minimum decay period, while stored under water in the spent fuel storage pool, the stainless steel hubs are separated from the Cd-Al body and shipped with the other radioactive non-fuel element metal pieces. The Cd-Al shim body is stored in the storage pool or in shielded dry storage wall cavities.

After holding for seven years' decay, the blades qualify for macroencapsulation and disposal at the Envirocare facility in Utah. We have constructed special aluminum storage boxes for these blades. The boxes have compact dimensions that make them easy to store. We currently have some in dry storage, with existing storage capacity for an additional forty operating-years' worth of these spent blades.

11.3 Section 11.1.1.4.4, p. 11-10. Provide more detail as to the type of "materials designated as radioactive waste" that are transferred to H wing. Describe what methods are used to control access to the H wing, or justify not requiring access control.

Section as written:
11.1.1.4.4 Solid Radioactive Waste Disposition

All radioactive waste is disposed of in accordance with 10CFR20, Subpart K. Solid waste is transferred to organizations specifically authorized or licensed to receive the material, such as the Department of Energy. Materials designated as radioactive waste are transferred to the H wing of the facility for characterization, packaging, and preparation for transfer to authorized recipients. Annual radioactive waste volumes and activities are typically in the range of 126 to 423 ft^3 (11 to 36 m^3) and are less than 1 Ci. In years when un-fueled element shipments occur, or major facility modifications performed, larger quantities of radioactive material will be involved. Based on past experience, these are infrequent occurrences on the order of once every 5 or more years.

Response:
As stated, all radioactive waste is disposed of in accordance with 10CFR20, Subpart K. Solid waste is transferred to organizations specifically authorized or licensed to receive the material, such as the Department of Energy. Materials designated as radioactive waste are transferred to the H wing of the facility for characterization, packaging, and preparation for transfer to authorized recipients. All of this waste is 10CFR61.55 Class A waste, unless otherwise noted.

Routine waste collection and screening:

All reactor support systems that flow either light water or heavy water circulate through cartridge-type particulate filters and through H-OH bead-type ion exchange resin beds. Reactor operations personnel periodically replace these filters and resins with new material as part of the routine maintenance of those systems. We collect about one 55-gal. drum of filters per year, and change one or two resin beds per year. Each resin bed change produces one 55-gal. drum that is about 3/4 full of de-watered resin beads. These filters and resin are stored in a shielded area of H100 until shipment for disposal.

- Any reactor support maintenance that requires a Radiation Work Permit results in non-routine waste that is specific to that job, and which is monitored by Health Physics and transferred to H100 as radwaste.

-For reactor maintenance, if a specific component that is activated will be removed for replacement, the activities and procedures for transfer to H100 as radwaste are reviewed beforehand on a case-by-case basis. A component may require shielded transfer, and may have specific activities that require disposal as 10CFR61.55 Class B waste, but this is seldom and comprises less than 0.001 per cent of the volume of waste transferred to H100.

- All ventilation systems have particulate filters that are replaced annually. Those filters that are collected from systems that service radioactive material use areas are collected and transferred to H100 as radwaste.

- Aqueous radioactive liquid wastes that are collected from radioactive systems are transferred to H100 in 55-gal. drums for disposal, if the water cannot be evaporated.
If evaporated, the residues are subsequently collected and transferred to H100 as radwaste.

- Waste solutions from laboratory analyses and separations are neutralized by the researcher prior to Health Physics' acceptance for transfer to H100 as aqueous radwaste.

- Discrete radioactive sources that have decayed or no longer are needed are only picked up by prior arrangement with Health Physics, as the specific activities usually require special handling and disposal procedures, and in many cases the specific activity results in Class B or Class C 10CFR61.55 classification. These are not generally transferred to the H100 annex, but are kept in secure storage until they can be disposed of as part of a turn-key project.

- The continual redesign and upgrading of reactor experimental facilities results in the reconfiguration of instrument shielding. The old shielding generally has slight long-lived activity that is not removable and does not contribute to personnel doses. That shielding is bulky and heavy and is generally kept in a designated storage area other than H100 until it can be disposed of. Occasionally, some component or block of obsolete shielding will have a relatively high induced activity, and will be transferred directly to the H100 annex.

- All contents of vacuum cleaners that are used in access-controlled work areas where the use of radioactive material is authorized, are emptied by Health Physics personnel and transferred to H100 as radwaste.

- Smoke detectors and static charge devices are accepted and transferred to H100.

- Health Physics will respond to any special needs or requests for non-routine waste collection not otherwise mentioned above.

Access to the H wing annex is strictly controlled. Only authorized persons, with proper training and clearance are permitted to gain unescorted access.

Direct access controlled is accomplished by strict adherence to facility security procedures and policy.

Chapter 13 – Responses to Technical Questions and Comments:

13.1 Section 13.2.1, p. 13-5. Provide a discussion/analysis of potential metal-water reactions and associated potential consequences.

No detailed event progression has been carried out since the MHA assumes that a complete fuel element blockage occurs, and that this results in the instantaneous release of all fission products. As pointed out in the SAR, this is a very conservative assumption, as there is a screen upstream of the fuel with a 0.25 inch square mesh. Simple estimates indicate that in the single element, the fuel and clad and matrix would completely melt in approximately 3 seconds, and that if one assumes that the fuel remains in a high flux region until the reactor scram, the temperature could reach or exceed 1000 °C. If the fuel plates melt, however, the fuel will not stay in the high flux region, but will either drop down to the bottom of the element or slump and end up in a clump of U/Al. Either of these scenarios would result in a drastic reduction in heating rate and in reactivity. Within two seconds, the sweep gas will reach the stack and initiate an immediate major scram, after which the power would rapidly drop down to levels at which the fuel would be adequately cooled and the possibility of water-metal interaction would be ended. Thus, the temperature of the metal would not remain above 1000 °C for any significant length of time, and any water-metal reactions would be of minor significance.

13.2 Section 13.2.1, p. 13-6. The 1st paragraph states "The inventory of noble gases and iodine fission products in the most heavily irradiated element is given below in Table 13.1, as determined by the computer code ORIGEN2 (Croff, 1980)." Describe or reference the assumptions on irradiation times, power levels, peaking factors, etc. to verify that this element has the maximum iodine and noble gas concentration.

This element was chosen because it would maximize total fission product releases, rather than gaseous fission product releases. Since the maximum power factor in the NBSR at any stage of prolonged full power operation (at startup, the gaseous inventory will be relatively low) is 1.16, then it is possible that in the worst case, the gaseous releases could be 16 % higher. Given the results shown in Chapter 13, this would not materially change the conclusions.

13.3 Section 13.2.1, p. 13-6. The section states "...consideration of these effects leads to the conclusion that less than 3% of the total iodine release will be present as I2." Provide the analyses on which this conclusion is based. Include your analyses related to the effects of temperature, pH and the presence of other fission products and chemical forms on iodine release fractions. Evaluate the effect of differences in fuel material design and configuration. Specifically, the type of fuel used at NIST (U3O8) is different than the type of fuel for the NUREG 1465 analysis (UO2), on which it is understood the 3% is partially based. Consider reviews such as presented in "The Technology of Nuclear Reactor Safety," Volume 2, Copyright 1973 by the Massachusetts Institute of Technology, Chapter 3, "Fission Product Release" by G. W. Parker and C. J. Barton of ORNL, Section 3.3.2, "Uranium Oxide, U3O8." Also, since some of isotopes have relatively short half-lives relative to the accident duration, the daughter products may be released from solution. Describe how these parent and daughter products are accounted for in the source term and dose estimates. Provide a description of how the iodine daughter products were considered.

As stated in the SAR, the assumption is that all fission products are released at once, a very conservative assumption. In this case, the real and interesting difference in the behavior of different fuels as discussed in the given reference (and in a more recent RERTR paper[2]) does not come into play. As stated in the response to RAI 13.1 (above), no detailed scenario was calculated for this accident; rather, drawing from the experience of other calculations, including the references given in the SAR, and at MITR, the value of 3 % was chosen as a reasonably conservative estimate of iodine releases. The only detailed calculation for a research reactor was done at HFIR[3], and that analysis was for a full-fledged LOCA. Reference 1 contains a discussion of reactor accidents and states that the iodine releases seen are very small (e.g. < 0.001 % released into the air during melting of one element during a 30 MJ excursion of the ISIS reactor). NUREG 1465, which supersedes earlier guidance used in NBSR-9, states:

"The chemical form of iodine entering containment was investigated in Reference 18. On the basis of this work, the NRC staff concludes that iodine entering containment from the reactor coolant system is composed of at least 95% cesium iodide (CsI), with no more than 5% I plus HI. Once within containment, highly soluble cesium iodide will readily dissolve in water pools and plate out on wet surfaces in ionic form. Radiation-induced conversion of the ionic form to elemental iodine will potentially be an important mechanism. If the pH is controlled to a level of 7 or greater, such conversion to elemental iodine will be minimal. If the pH is not controlled, however, a relatively large fraction (greater for PWRs than BWRs) of the iodine dissolved in containment pools in ionic form will be converted to elemental iodine".

For the NBSR pH (pD), radiation-induced conversion of CsI to elemental I would equilibrate at approximately 3 %. Given these results, and others cited in Reference 1, the present assumptions are conservative. In fact, a publication[4] on calculations for the HFIR (the only detailed calculations performed for a test or research reactor) indicate that the release fractions are as low as 10^{-8} from the stack. This represents a reduction is elemental iodine present of 10^{-5} from that which would be estimated by older methods.

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[2] "Fuels for Research and Test Reactors, Status Review", D. Stahl, ANL -83-5, July, 1982

[3] Weber, C. F. and Beahm, E. C. (January 1993). *Iodine Transport During a Large Pipe Break LOCA in the Pipe Tunnel With Drainage Outside Confinement*. Research Reactors Division CHFIR-92-032, Oak Ridge National Laboratory: Oak Ridge, Tennessee.

[4] "Iodine Transport in a Severe Accident at the High-Flux Reactor", Charles F. Weber and Edward C. Beahm, Transactions of the American Nuclear Society **68** I1993) 275.

Thus, the present calculation is conservative when compared to the best calculations for a research reactor.

The concentrations of radioisotopes at various times after the release were based upon ORIGEN generated isotopic masses appropriate to the time period. ORIGEN tracks buildup and decay of fission products and their daughters as a function of time.

13.4 Section 13.2.2.2.2, p. 13-9 & the new calculation provided via email [Mendonca 9/29/2006] following the site orientation visit. The new calculation is for a ramp insertion of 0.5% in 0.5s, whereas the previous accident scenario is for a ramp insertion of 1.3% in 0.5s. SAR section 13.1.2.2.2 provides technical justification for the change in the accident scenario from the existing SAR, however this is not consistent with at least one of the bases in the TS. Specifically, the basis for TS 3.12 refers to the 1.3% insertion transient. Correct this reference and verify that any other renewal submissions are consistent with the revised analyses.

Agreed, will revise as appropriate.. Corrected in revised Tech Specs.

13.5 Section 13.2.3, p. 13-10. Under the assumptions for this accident, it states "The tritium concentration in the primary coolant is at the maximum level permitted by the TS (5,000 μCi/ml)." The statement regarding the estimated concentration in the Basis of TS 3.6 is not a TS limit. Provide a description of how and where this limit is protected in the TS. If there is no limitation established on this parameter in the TS, provide such.

The revised Tech Specs have an explicit limit of 5000 μCi/ml.

If the reactor is operated at full capacity without interruption for non-routine maintenance, the maximum possible tritium concentration would be 5200 μCi/ml. Since the D_2O is routinely changed before reaching 2000 μCi/ml, the assumption is conservative.

13.6 Table 13.1, p. 13-16. Several of the isotopes in the fission product inventory are not in the HOTSPOT library. Provide a description of how these were modeled in the offsite dose projections.

Only the noble gas and iodine isotopes that made significant contribution to doses were considered. All of these isotopes are included in the Hotspot library except ^{137}Xe, which has a short half life. This isotope would only contribute a small amount to any dose, and was ignored.

13.7 Tables 13.3 & 13.4, p. 13-17. Provide the assumptions regarding iodine removal rates in confinement from deposition and filtration for public and staff dose estimates. What DCFs were used for submersion, inhalation, and thyroid doses for staff doses presented in Table 13.4?

No credit was taken in this analysis for iodine deposition.

As stated in the SAR, the circulation system is assumed to be in its emergency configuration, as described in the SAR Section 6.2.3.2, following a major reactor scram. Two different components are directly relevant here – the Emergency Recirculation System (6.3.2.1) and the Emergency Exhaust System (6.3.2.2). Both of these systems incorporate filters which are assumed to be 99 % efficient for iodine removal and completely ineffective for noble gas removal. The Emergency Recirculation System draws air from the building at a flow rate of 5,000 cfm (140 meter^3 per minute), containing 2,000 cfm (60 meter^3 per minute) from the second floor, 2,000 cfm (60 meter^3 per minute) from the first floor, and 1,000 cfm (28 meter^3 per minute) from the reactor basement area controlled by ACV-11. This air passes through the particulate filters F-19 and F-20 and a carbon filter F-21 which is 99 % effective for iodine removal, before being returned to the building. The Emergency Exhaust System pulls 100 cfm (3 m^3 per second) through a similar filter chain (see 6.2.3.2) and discharges it directly to the atmosphere. The use and efficiency of the filters will be referred to the SAR, Section 6.2.3.2.

The dose conversion factors used were taken from Federal Guidance Reports 11 and 12, issued by the EPA. References are given in the SAR.

13.8 Table 13.2, p. 13-16. The values for removal rates from C-200 are not consistent. Determine the appropriate values and ensure that they are correct in both sets of units.

We agree. The C-200 removal rate in Table 13.2 is 19. x 10^{-3} m^3/s. The difference was due to rounding; the values were originally in cfm.

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.

We will comply.

Editorial (Chapter 13):

13.10 Section 13.1.4, p. 13-3. This section states "Five different scenarios for loss of primary coolant flow have been analyzed," and in Section 13.2.4, p. 13-11, it states "Four scenarios have been given for an accident of this type [Loss of Primary Coolant Flow]...." Clarify the apparent discrepancy.

Five scenarios were analyzed.

13.11 Section 13.2.1, p. 13-7. In the second paragraph, the phrase "for estimation of long-term (>1 day)" seems to be related to dose. Should it be "for estimation of long-term doses (>1 day)"?

We agree.

13.12 Section 13.2.2.2.1, p. 13-8. In the first paragraph, the reactivity insertion rate, "5x10-4 k/s" appears to be inconsistent. Should the units be "5x10-4 /s"?

The value is $5x10^{-4}$ $\Delta\rho/s$.

13.13 Section 13.2.3, p. 13-10. The 1st paragraph states "Thus, with only one operator action (which can be accomplished at any time in the first 20 minutes), the core is fully protected for several hours." In Chapter 6, p. 6-2, the time the IRT and D2O Emergency Cooling Tank provide cooling is 2 ½ hours. The term "several" used in the statement from section 13.2.3 appears to be subjective.

We agree; text should say "... the core is fully protected for 2 ½ hours without further operator action."

13.14 Section 13.2.3, p. 13-11. The last paragraph states "For the conditions analyzed, this will result in a concentration approaching 1.25x10-4 DAC." Shouldn't this be 1.25x104 DAC?

We agree.

13.15 Figures 13.2, 13.3 & 13.4, p. 13-21, 13-22. Provide clarification if these are plots of MCHFR versus time, or CFHR versus time.

The figures are correct. The results shown are the MCHFR in the core as a function of time. The lowest value is then the MCHFR for the transient.

Appendix A - Technical Questions and Comments:

13.16 Section 2.2, p. 2-4. The 1st paragraph states "About 4% of the total flow in each plenum bypasses the fuel elements and cools the in-core thimbles." Chapter 4 (SAR), p. 4-4, states "A small amount of coolant, 4%, bypasses the external surface of the lower nozzle...preventing bulk stagnation in the moderator." In Chp. 4 (SAR), p. 4-12, the description of the regulating rod states "A fixed orifice in the nozzle of the shroud delivers a coolant water flow of 8 gpm from the outer plenum." In Chp. 4 (SAR), p. 4-50, the description of the core flow distribution states "Approximately 4% of the flow bypasses the core; this is treated conservatively in the next sections [T-H Analysis] by reducing both flows to 95% when calculating the flow through any element." In the "Core Bypass Flow" section of Appendix A, p. 4-5, the RELAP model description states "About 4% of the total primary flow bypasses the fuel elements. In RELAP5 the areas of the bypass flow junctions have been adjusted so that 4% of flow to the inner and outer plenums is bypassed." In Chp. 10, Section 10.2.6.1, p. 10-6, the description of the seven 3 ½ in. thimbles states "The end fitting largely blocks the normal flow, but contains a small opening that allows approximately 8 gpm (0.5 liter/sec) to flow upwards through the tube to cool it, and any experiment that may be in it." In Chapter 10, Section 10.2.6.2, the description of the 2 ½ in. thimbles states "These smaller sockets have a small hole at the bottom that allows approximately 10 gpm (0.6 liter/sec) of plenum cooling water to flow up through the experimental thimble."

There appear to be some discrepancies in the above statements regarding bypass flow. Some specific considerations are:

a. The 4% bypass flow is not predominately for in-core thimble cooling, since these have individual orifices for coolant flow.
b. Chp. 4 (SAR) indicates that fuel element flow is treated as 95% full flow while Appendix A indicates the RELAP model uses 96%.
c. If six of seven 3 ½ in. thimbles at 8 gpm and four 2 ½ in. thimbles at 10 gpm are fed separately from the outer plenum, then this accounts for approximately 88/6400 = 1.4% of outer plenum flow not accounted for in the RELAP model.

Specific consideration a is correct; the 4% bypass flow serves to cool thimbles and to remove the heat deposited directly into the moderator. The actual flows are based upon NBSR-9, Table 4.7-2, page 4-25. The relevant parts of this table are reproduced below. Note that this table is for a 24 element core at 10 MW with different flows than the current 30 element core at 20 MW.

Flow area	Flow (gpm)
Inner plenum fuel elements	1165
Outer plenum fuel elements	3745
Bypass flows	

Regulating rod	8
4 - 2 inch thimbles	40
6 – 3 inch thimbles	48
Around fuel elements and other core components	94
TOTAL FLOW	5100

From this table, the total bypass flow is 190 gpm, or 3.73% of the total flow of 5100 gpm, as stated in the SAR. The relative magnitudes of these flows will be maintained as the total flow is changed, so that the same proportions will be expected for the present flows (see the response to RAI 4.27). Of these flows, the central 3 inch thimble, two 2 inch thimbles and 6 fuel elements are fed from the inner plenum, for a total inner plenum flow of 1216 gpm, with bypass accounting for 51/1216 = 4.2 %. The remainder of the bypass flow, 139 gpm, is fed from the outer plenum, for a total outer plenum flow of 3884 gpm, with bypass accounting for 139/3884 = 3.6 %. Thus, the bypass flow is approximately 4 % of the flow, and this ratio will continue to apply at other flows, such as the present 20 MW flow. Accordingly, within the errors of the data, bypass flow is properly accounted for in the RELAP flows, in response to consideration c.

Consideration b requires more discussion. The 95 % refers to the calculation of the limits of safe operation only, and represents an added element of conservatism in these limits. This was a deliberate choice for this calculation, even though it is more conservative than the RELAP calculations of Appendix A.

Provide clarification of the following comments (13.17-13.25).

13.17 Figure 3-5, p. 3-12. The 235U content in this figure differs from that in Figure 4.5.2A, p. 4-86, in Chapter 4 of the SAR. Are these "BNL" versus "updated model" differences?

No. The ^{235}U mass values in Figure 4.5.2A predate the BNL model. See response to RAI 4.22.

13.18 Figures 3-13 through 3-18, pp. 3-16 to 3-18. The orientation of the plates in these figures is north-south which differs from the east-west orientation in Figure 4.5.4 through 4.5.9, pp. 4-88 to 4-90. Is the orientation different in the two MCNP models? If so, provide clarification of the effect this has on the peaking factors.

The observation is correct. The plate orientation was in the original model NIST provided BNL, and was used in the analysis in their report (Appendix A). Subsequent calculations have shown that the analysis in Appendix A is conservative with respect to more recent models. The plate orientation does not change the fact that the hot spots are at the corners of the fuel sections. The plate-to-plate power distributions and the transverse power distributions across the plates are almost interchangeable.

13.19 Figure 3-28, p. 3-23. Provide analyses which demonstrate that the regulating rod maximum reactivity differential worth and withdrawal rates will not exceed the startup accident maximum reactivity insertion rate. Alternatively, propose limits on regulating rod reactivity insertion rates to limit them to the same rate as specified for the shim rods. Additionally, provide justification as to why the regulating rod worth should not be considered in conjunction with shim arm worth in the startup accident.

The maximum measured reactivity rate is 4.5 x 10^{-4} $\Delta\rho$/s (see response to RAI 4.21), but can only be sustained for a few seconds. Although the operator is not physically prevented from withdrawing the regulating rod and the shim arm bank simultaneously, such an action would not occur inadvertently when the power is increasing with a short period. The regulating rod is not part of the startup accident because it is not a credible accident.

13.20 Figure 3-30, p. 3-24. The caption for the figure includes the description "Equilibrium Core at Startup" and the title includes the description "SU Core." In previous nomenclature, the "SU Core" is defined as the startup core prior to equilibrium fission product poison concentrations, and the "BOC Core" as the startup core with equilibrium fission product poison concentrations. For which core was this figure developed? Provide consistent references in the renewal application documents.

We agree with the statement. The same phrase is used in Figures 19, 20, 21, 26, 28, 30 and 32. They all refer to the SU core, not the BOC core. Thus, the word "equilibrium" should not appear those captions. The last paragraph of Section 3.1 in Appendix A defines the SU and BOC cores in a way that is consistent with Chapter 4 of the SAR.

13.21 Table 3-2, p.3-28. As previously mentioned, the description for the voided thimbles indicates 5 thimbles voided whereas p. 3-6, App. A and Chapter 4 (SAR), p. 4-39 indicates 6 thimbles voided. The values of k/k appear to be calculated as (kvoid – kbase case)/ kbase case instead of (kvoid – kbase case)/ kvoid. What thimble volume was used for the void coefficients calculated for the voided thimbles case? These numbers appear to be inconsistent with those in Table 4.5.7, p. 4-67 of Chapter 4 of the SAR. What case or analysis supports the statement in Section 4.5.2.2.2, p. 4-39 of Chapter 4 of the SAR that "Finally, from the BNL analysis, if somehow only the unfueled regions between the upper and lower fuel sections were to be voided, the coefficient would be -0.025% /l..."?

*A review of the input files for the void coefficient calculations shows that there indeed **six** thimbles voided. Both Table 3-2 and 4.5.7 contained errors. The k_{eff} values in Table 3-2 are correct, however, so the void coefficients have been recalculated. Table 4.5.7 contained incorrect void volumes. The corrected version of Table 4.5.7 is:*

Table 4.5.7A Calculated Moderator Void Coefficients (6 Thimbles).

Core Model	%Δρ	Volume of Void (liters)	Void Coefficient (%Δρ/liter)
SU	-2.05 ± 0.06	47.78	-0.043 ± 0.001
EOC	-1.45 ± 0.06	47.78	-0.030 ± 0.001

Table 4.5.7B Calculated Moderator Void Coefficients (Coolant).

Core Model	%Δρ	Volume of Void (liters)	Void Coefficient (%Δρ/liter)
SU	-6.00 ± 0.06	163.7	-0.0367 ± 0.0004
EOC	-4.87 ± 0.06	163.7	-0.0298 ± 0.0004

The final entries in Table 3-2 are not recomputed because coolant flow cannot physically leave only the gap voided. The void coefficient, however, is indeed negative everywhere.

13.22 Tables 3-3 & 3-4, p.3-28. The values of Δk/k appear to be calculated as (kflooded – kbase case)/ kbase case instead of (kflooded – kbase case)/ kflooded. Provide clarification as to which is the correct method for determining the values of k/k.

The correct way to calculate reactivity is $\Delta\rho = (k_2-1)/k_2 - (k_1-1)/k_1$. However, the differences resulting from the different representations are quite small – well below the level of accuracy of the results.

The values computed in Section 4.5.1.6.2 are the correct values for the reactivity insertions that would be caused by the flooding of the cold source, one radial beam tube, or one grazing tube.

13.23 Section 4.2.2.4, p. 4-3. The fuel plate width is given as 2.3734 in. in this section, and 2.436 in. on p. 4-3 of Chapter 4 of the SAR. The 2.436 in. appears consistent with the peak heat flux given in Chapter 4 on p. 4-54, element H-1.

The nominal dimension is 2.436; BNL used the smaller value in the RELAP calculations, which is within allowed errors of fabrication, and is conservative.

13.24 Section 5.3, p. 5-3. This section states "The minimum CHFR is 1.28 and 1.18 for BOC and EOC, respectively." These values are both below the 99.9% limit values determined for CHFR on p. 4-10 of Appendix A. Provide justification to demonstrate that these provide acceptable margins.

The MCHFR quoted above are for a reactivity insertion arising from withdrawing three experiments worth 1.3 %$\Delta\rho$ in 0.5 second, an incredible scenario. The maximum reactivity accident has been modified to an insertion of 0.5 %$\Delta\rho$ in 0.5 sec, and the SAR and the Tech Specs are being changed to reflect the modification.

13.25 Table 5-13, p. 5-26. This table presents CHFRs as determined by the Mirshak and Costa correlations for 500 kW operation under natural circulation. Provide justification that these correlations are applicable for natural circulation flow. Describe the flow velocity ranges and conditions where the correlations are valid.

While the Mirshak data were taken down to 1.5 m/s and the Costa data were taken down to 3 m/s, the calculated velocities (RELAP) for natural circulation at 500 kW are only about 0.06 m/s. We have recalculated the CHF and OFI ratios using appropriate low-flow correlations, as shown in the table below.

Table 5-13. Thermal Margins for 500 kW Operation under Natural Convection

	Inner Plenum		Outer Plenum	
	Top of Lower Core	Top of Upper Core	Top of Lower Core	Top of Upper Core
Coolant Temperature, K	334.1	342.5	336.2	344.7
Coolant Velocity, m/s	0.0585	0.0610	0.0643	0.0675
Wall Heat Flux, W/m^2	3.417×10^4	1.760×10^4	4.192×10^4	2.160×10^4
CHF (Sudo/Kaminaga)[1], W/m^2	2.326×10^5	2.387×10^5	2.463×10^5	2.538×10^5
Minimum CHFR	6.8	13.6	5.9	11.8
OFI Heat Flux (Oh/Chapman)[2], W/m^2	7.008×10^5	5.654×10^5	6.706×10^5	5.338×10^5
OFI Ratio	20.5	32.1	16.0	24.7

1 Sudo, Y. and Kaminaga, M., "A New CHF Correlation Scheme Proposed for Vertical Rectangular Channels Heated from Both Sides in Nuclear Research Reactors," Journal of Heat Transfer, Vol. 115, May 1993.

2 Oh, Chang H. and Chapman, John C., "Two-Phase Flow Instability for Low-Flow Boiling in Vertical Uniformly Heated Thin Rectangular Channels," Nuclear Technology, Vol. 113, March 1996. See also: Saha, P. and Zuber, N., "Point of Net Vapor Generation and Vapor Void Fraction in Subcooled Boiling," Proc. 5th International Heat Transfer Conference, Tokyo, Japan, Vol. IV, September 3-7, 1974

Editorial (Appendix A):

13.26 Section 2.1, p.2-1. The 2nd paragraph states "The fuel elements are located on 0.177m (7 in) centers in a hexagonal array." Chapter 4, p. 4-4 indicates 0.175m and p. 4-17 indicates a 17.6 cm pitch for exp. thimbles.

The correct value is 6.928 inches, or 0.176 m (see Fig. 4.2.11 in SAR).

13.27 Section 2.1, p. 2-2. Paragraphs 7 & 8 (next to last & last) indicate reactivity worths of 26%, 6 ½%, and 0.6%. Should the units be %   ?

Yes. The units of reactivity should be %$\Delta\rho$.

13.28 Section 2.1, p.2-3. The 2nd paragraph states "The uranium content is about 1 gm/cm3." Data from Chapter 4, p. 4-3 indicates 1.23 gm/cm3.

Agreed, will revise as appropriate..

13.29 Section 2.2, p. 2-3. The 1st paragraph indicates a nominal core flow of 9000 gpm. Chapter 4, p. 4-50, Table 4.1.1, p. 4-59, indicates 8700 gpm as nominal flow.

BNL: 9000 gpm refers to the nominal operating condition. 8700 gpm is the conservative primary flow used in the T/H analysis. Table 4-5 on p. 4-26 of Appendix A provides a comparison of operating versus design basis values for process flow parameters.

13.30 Section 2.2, p. 2-4. The 1st paragraph indicates an outer plenum flow of 6700 gpm. Chapter 4, p. 4-50, Table 4.1.1, p. 4-59, indicates 6400 gpm as outer plenum flow.

BNL: 6700 gpm refers to the nominal operating condition for the outer plenum flow. 6400 gpm is the conservative outer plenum flow used in T/H analysis.

13.31 Section 3.3, p. 3-4. The 1st paragraph states "Also included in this figure is the percent decrease in the 235U content for each fuel element during a single 38-day cycle." Figure 3-5, p. 3-12 shows "Decrease in 235U (grams)."

Agreed, will revise as appropriate..

13.32 Section 3.4.3, p. 3-5. The 2nd paragraph states "The D-4 element is separated from the shim arm by one row of elements…." Should the element described "D-1"?

Yes.

13.33 Section 3.5.2, p. 3-6. The 1st paragraph states "In the first case, the six vacant irradiation thimbles …are voided." In Table 3-2, p. 3-28, this case is described as "SU with 5 thimbles voided."

Agreed, will revise as appropriate.. See response to RAI 13.21.

13.34 Section 3.5.2, p. 3-6. The 1st paragraph states "The calculations were performed for the SU and EOC cores for two different void cases." Table 3-2, p. 3-28 shows three cases.

Agreed, will revise as appropriate.; third case is with void in gap only, which is impossible to achieve.

13.35 Section 3.5.8, p. 3-8. The 1st paragraph states "In the present work, the maximum relative power peaking was 1.16. In the updated model, the maximum value was 1.11." In the SAR, Chapter 4, Figure 4.5.3, p. 4-87, the maximum peaking factor is 1.15 calculated with the updated model.

Agreed, will revise as appropriate., Chapter 4 is correct, but Appendix A model is still conservative.

13.36 Figures 3-29 & 3-32, p. 3-24 & 3-25. The y-axis labels appears to be missing the units "(%)."

The axes are indeed mislabeled.

13.37 Figures 3-26 through 3-33, pp. 3-22 to 3-26. The y-axis labels are not discernable on provided copy.

The figures are correct on our copies; we can supply better copies upon request.

13.38 Section 4.2.3.8, p.4-6. In the 2nd paragraph, the sentence beginning states "A set of power factor is determined...." Should this be "A set of power factors is determined...."

Agreed, will revise as appropriate..

13.39 Table 4-5, p.4-26. "Normal" primary flow in the table is given as 8800 gpm and 9000 gpm in the footnote.

Agreed, will revise as appropriate.. The nominal value is 9000 gpm.

13.40 Section 5.2, p. 5-2. In the 1st paragraph, the shim arm withdrawal reactivity rate is given as "5 x 10-4Δk per second." Use consistent reactivity units.

Agreed, will revise as appropriate..

13.41 Section 5.4, p. 5-3. In the 2nd paragraph it states "After a 0.4s delay a reactor scram is initiated at 1.286 s." If the flow trip is initiated at 0.896 s, shouldn't the reactor scram be initiated at 1.296 s?

BNL: Agreed, will revise as appropriate.. The correct scram time is 1.296.

13.42 Tables 5-1 through 5-4, pp. 5-14 to 5-17. Shouldn't the column headings be CHFR instead of MCHFR?

No. The values are the MCHFR at each step in the transient. See response to 13.15.