NRC PROJECT MANAGER BRIEFING

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OF

NRC-SUPPLIED INCONSISTENCIES SUMMARY

PROJECT I

OF

CONTRACT NO.: NRC-03-85-051

28 JANUARY, 1986

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NRC-SUPPLIED INCONSISTENCIES
SUMMARY

Category A

(1) Concern 9 Technical Specification 3/4.2.5, Table 3.2-1, DNB Parameters

Question (1):

"If they (necessary parameters) are represented under section 2.1.1 (and elsewhere), why are they also represented here?"

Discussion

The DNB parameters are represented under Section 2.1.1 in the context of providing the overall safety limits for the nuclear plant. In Section 3.2.5, the parameters are covered in the context of limiting conditions for operation. Both presentations are essential and having current groupings helps avoid any possible confusion."

Recommendation:

No change to the technical specification is recommended.

Category A

(1) Concern 9 Technical Specification 3/4.2.5, Table 3.2-1, DNB Parameters (continued)

<u>Item b</u>:

<u>Question (1)</u>:

Why is the Reactor Coolant System T_{ave} of Table 3.2-1 given an $\leq$593°F for rated thermal power when the FSAR, Figure 5.3.3-1 uses a value of 588.1°F?

<u>Discussion</u>

The FSAR Figure 5.3.3-1 is designated as "Relationship between Reactor Coolant System Temperature and Power" but paragraph 5.3.3 is not more explicit. The 100% power value is 588.1°F. Based on the thermal and hydraulic data given in Table 4.4.2-1, this appears to be a nominal temperature. The value of $\leq$593°F is a Limiting Condition for Operation for T_{ave}.

<u>Recommendation</u>:

To be determined. Need to verify a value of $\leq$593°F or higher was used in DNB limited safety analyses and that instrument error and lift are consistent with the difference between 588.1°F and 593°F.

drift

Category A

(1) Concern 9 Technical Specification 3/4.2.5, Table 3.2-1, DNB Parameters

Item b: (continued)

Question (2):

"Explain why a related power level has not been ascribed to this temperature."

Discussion

Based on the shape of the constant pressure curves given in Figure 2.1-1 of the Westinghouse Standard Technical specifications, the prescribed limit on T_{avg} would be applicable at any power level for the indicated limiting pressure.

Recommendation:

No change to the Technical Specifications required.

Category A

(1) Concern 9 Technical Specification 3/4.2.5, Table 3.2-1, DNB Parameters (continued)

Item b:

Question (3):

Explain why the Limiting Condition for Operation has not included a value of T_{ave} for zero power.

Discussion

The limiting value for zero power is higher than the prescribed limit of Table 3.2-1.

Recommendation:

No change to the Technical Specifications is requred.

Category A

(1) Concern 9 Technical Specification 3/4.2.5, Table 3.2-1, DNB Parameters (continued)

Item c:

Question (1):

Explain the value of ≥2230 psia in Table 3.2-1 when the FSAR Table 4.1-1 shows a "System Pressure, Nominal" of 2250 psia and FSAR Table 15.1.2-2 makes provisions for a total of 30 psi for steady-state fluctuations and measurement error.

Discussion

Recommendation:

To be determined

Category A

(1) Concern 9 Technical Specification 3/4.2.5, Table 3.2-1, DNB Parameters (continued)

Item c: (continued)

Question (2):

Referring to the LCO of $\geq$2230 psia for Pressurizer Pressure, questions whether this is a setpoint or an allowable value.

Discussion

The quoted value is defined as a limiting condition for operation of the DNB parameters. It is a limiting value and not a setpoint.

Recommendation:

No change to the Technical Specifications is required.

Category A

(1) Concern 9 Technical Specification 3/4.2.5, Table 3.2-1, DNB Parameters (continued)

Item e:

Question:

Why shouldn't pressurizer pressure be included in Table 2.1-1 and in section 3/4.4.3 on the pressurizer?

Discussion

Table 2.1-1 does use pressure as on condition in defining the one safety limits as suggested. The limiting condition for operation is defined in Section 3.2.5. While it could be included in Section 3.4.3, duplication would not be desirable.

Recommendation:

No change to the Technical Specifications is required.

Category A

(2) Concern 10 Table 3-3-1 Nuclear Instruments Modes 3-4-5

Item c:

Question:

Why are there no requirements for intermediate or power range nuclear instrumentation in modes 3-5?

Discussion:

The accidents in the FSAR initiated from the subcritical condition show the Power Range Neutron Flux Trip, Low Setpoint and High Setpoint; the Intermediate Range High Neutron Flux Trip; and the Source Range High Neutron Flux Trip included in the Reactor Trip Correlation Table 7.2.1-4. These accidents are generally considered as initiated from Mode 2 where the Power Range Trip is required by the Technical Specifications (STS). Under those circumstances where rod motion is authorized under higher numbered modes (e.g., for rod testing), provision for a flux trip would appear appropriate.

Recommendation:

Change Technical Specifications to require Power Range Neutron Flux Trip, Low and High Setpoint, in modes 3, 4, and 5 with an asterisk to indicate "applicable whenever the Reactor Trip Breakers are closed."
NOTE: Analysis will be required to determine the effect of boron level, temperature, and other effects on the nuclear instruments and setpoints.

Note:

If it can be shown that with the rod(s) being tested full out, K_{eff} remains <.99, would it really be necessary to have a nuclear trip?
If K_{eff} does not remain <.99, one cannot be in mode 3-5.
In other words, can one stay Boron safe and not need nuclear instruments?

Category A

(3) Concern 14 Technical Specification 3.3.2 Engineered Safety Features Actuation System (ESFAS) Instrumentation, Table 3.3-3 ESFAS Instrumentation

Item 3.b.3:

Question:

Why is there no containment Phase B Isolation signal generated by Containment Pressure - High High when in Mode 4?

Discussion:

The Technical Specifications require Containment Integrity to be maintained in Modes 1-4. (STS 3.6.1.1) Phase B Isolation is actuated by Containment Pressure-High-3 for modes 1-3 only (STS Table 3.3-3).

Notes to File:

(Not for inclusion in final report.)

Possible explanations - (1) Phase B are closed systems only - (FSAR 6.2.4.2) (2) Energy available in Mode 4 of Ice Condenser Containment not high enough to reach high-high trip point. (3) Need to keep operating the Phase B systems e.g., Spray so want essentially manual control.

Recommendation:

To be determined.

Category A

(4) Concern 15 Technical Specification Table 3.3-4 Engineered Safety Features Actuation System (ESFAS) Instrumentation Set Points

Item 11:

Question:

Proposal to add a new Functi... Unit to the Engineered Safety Features Activation System (ESFAS) (e.g., Table 3.3-3). The new Unit would be entitled "Close Feedwater Isolation Valves and Cisoe Feedwater Main and Bypass Modulating Valves."

Discussion:

Currently "Feedwater Isolation" is included in two function Units (numbers 1 and 5).

Worksheet Note:

Proposal has Feedwater Isolation From:

a. Reactor trip and low T_{avg}
b. Reactor trip and steam generator level hi-hi
c. Steam generator Level hi-hi
d. Safety Injection

Item c covered by Function Unit 5
Item d covered by Function Unit 1 (or Unit 5)
Item a and b are presumed covered by item 5 entry: "Automatic Actuation Logic and Actuation Relays" (Would need WCAP-7672/WCAP-7705 to positively verify).

An analogous situation can be found, for example, in the case of initiation of containment isolation, Phase A. The Reactor Trip (P-4) when combined with one of several other possible signals (Low Pressurizer Pressure, Low Steamline Pressure or 2/3 high Containment Pressure) will result in both Feedwater Isolation and Containment Isolation Phase A. In both cases, the sequence is included in Table 3.3-3 as "Automatic Actuation Logic and Actuation Relays." The more direct actuation of this signal is via initiation of Safety Injection. The term "Feedwater Isolation" is considered sufficiently generic to cover the modulating valves and bypasses.

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Category A

(4) Concern 15

Item 11: (Continued)

The proposed change by itself, would lead to additional inconsistency in handling the "Automatic Actuation Logic and Actuation Relay" signals.

Recommendation:

Do not include the proposed addition to the technical specifications as the circuitry in question is adequately covered.

Category A

(5) Concern 19 TS 3/4.4.1.4.1 Cold Shutdown (Modes) with Loops Filled

Querstion 1:

Acceptability of use of secondary side of two steam generators for circumstances in which the residual heat removal loops are isolated from the Reactor Coolant system is questioned.

Discussion:

Acceptability of suitable alternate method for circumstance of loss of both RHR loops was made contingent on resolution of TMI Action Plan Task II.E.3 (NUREG 0660) by the Safety Evaluation Report (NUREG 0422, Supp. 4).

Recommendation:

(Contingent on status of NUREG 0660 item and McGuire implementation).

Question 2:

Proposes that an LCO be included to require a shift to Mode 4 from Mode 5 in the event of failure closed of the RHR isolation valve.

Discussion:

The efficacy of this proposal appears to be a function of a large number of items. While the idea of having additional equipment requirements established is conservative, the idea of going from a cold to a hot plant is non-conservative. The specific action that might be appropiate is contingent on the resolution of question 1 of this concern.

Recommendation:

Do not accept this proposal for an addition to the technical specifications but incorporate the question into the resolution of question 1 above.

Question 3:

The footnote relative to deenergizing the operating RHR pump for one hour is questioned.

Category A

(5) Concern 19 TS 3/4.4.1.4.1 Cold Shutdown (Modes) with Loops Filled

(Continued)

Discussion:

The footnote which allows deenergizing the operating RHR pump for up to one hour includes the following qualifications: "...Provided: (*1) no operations are permitted that would cause dilution of the Reactor coolant System Boron Concentration, and (2) core outlet temperature is maintained at least 10°F below the saturation temperature." With these qualifications, it is considered that plant safety is assured and therefore, allowing some operational flexibility in Mode 5 is desirable.

(Note: RESAR does not support this 1-hour period)

Recommendation:

No change to the Technical Specifications is recommended.

Question 4:

"Safety Related Flow Alarms" with an appropriate surveillance requirement are proposed to provide the assurance that the RHR is in operation and circulating reactor coolant.

Discussion:

Direct measurement of flow is one method to verify actual circulation of coolant in the RHR loop. Such a measurement is available. Other means of determining operation are also available by using a combination of measurements (e.g., pumps discharge pressure, inlet and outlet temperatures).

Recommendation:

(Intuitively a safety grade low flow alarm seems like overkill but need to verify that some item will alarm with a stoppage of flow).

Category A

(5) Concern 19 TS 3/4.4.1.4.1 Cold Shutdown (Modes) with Loops Filled (Continued)

Question 5:

The absence in the surveillance specification (TS 4.4.14.1.2) of specific listing of process conditions which verify the RHR system is capable of performing its Licensing Basis Safety Function is questioned.

Discussion:

The specification in question requires the surveillance that "...one RHR loop shall be determined to be in operation..." Specification 1.18 defines OPERABLE-OPERABILITY to include "...capable of performing its specified functions)..." Thus, the specification in question does require McGuire to do the precise verification suggested.

Recommendation:

No change in the Technical Specifications is required.

Category A

(6) Concern 30 Technical Specification 3/4.7.1.4 Main Steam Isolation Valves

Question:

The proposal is made to require operability of the Main Steam Isolation Valves in Mode 4 in addition to the existing requirement for operability in Modes 1, 2, and 3 (TS 3.7.1.4).

Discussion:

As cited, "Containment Integrity" is required in Modes 1, 2, 3 and 4 (TS 3.6.1). It is contended that the Main Steam Isolation Valves are containment Isolation Valves as defined by 10CFR50 App. A Criterion 57 - 'Closed System isolation' and under FSAR table 6.2.4-1."

These valves are listed with an activation signal of "P" in table 6.2.4-1 (12, 13 of 20). This indicates they are Phase B isolation valves (FSAR Seec. 6.2.4.1 b). Phase B isolation is currently required only in Modes 1, 2, and 3 but is the subject of the question under number 3 of this listing.

The required operability of the Main Steam Isolation Valves in Mode 4 as a nuclear safety protection measure is not obvious. Postulated failure of steam generator tubes does pose a potential need for isolation. Additionally, the desirability of having the valves stay open to represent a possible cooling path in Mode 4 under manual control has merit.

The position that Containment Integrity requirements are violated by not requiring operability of the Main Steam Isolation Valves in Mode 4 also appears to have merit.

Recommendation:

Change the Technical Specifications to require Operability of the Main Steam Line Isolation Valves in Modes 1, 2, 3, and 4 Vice Modes 1, 2, and 3 only.

Category A

(7) Concern 31 Steam Generator Power Operated Relief Valves (SG PORVs)

Question 1:

Propose including the steam generator Power Operated Relief Valves in the Technical Specifications.

Discussion:

To include the Steam Generator PORVs in the Technical Specifications it would appear necessary to demonstrate that the operation of these valves is a necessary condition to accommodate one or more of the accidents in the FSAR licensing basis. It is contended that the loss of non-emergency AC power to the Station Auxiliaries (Station Blackout) FSAR 15.2.6 (1984 Update) requires the use of the steam generator PORVs to provide a flow path for natural circulation cooling.

The accident description in 15.2.6.1 says, "2. As the steam pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. The condenser is assumed not to be available for steam dump. If the steam flow rate through the relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

3. As the no load temperature is approached, the steam operated relief valves (or safety valves, if the relief valves are not available) are used to dissipate the residual decay heat to maintain the plant at the hot shutdown condition."

Analogous words exist for the other accidents that rely on natural circulation for core cooling. Based on these words, it does not appear necessary for the steam generator power operated relief valves to operate to meet the licensing basis accidents.

Recommendation:

No addition to the Technical Specification is necessary to include the steam generator power operated relief valves.

Category A

(7) Concern 31 Steam Generator Power Operated Relief Valves (SG PORVs)

(Continued)

Question 2:

The steam generator Power Operated Relief Valves were postulated as being required to dissipate up to 20% reactor power during natural circulation conditions.

Discussion:

Based on the results from Question 1 above, this question becomes mute.

Recommendation:

No addition to the Technical Specifications regarding sizing of the steam generator Power Operated Relief Valves is considered necessary.

Category A

(8) Concern 32 Technical Specification 3/4.7.3 Component Cooling System

Question:

Proposal to add Modes 5 and 6 to the current TS 3.7.3 requiring operability of at least two dependent component cooling water loops in Modes 1, 2, 3, and 4.

Discussion:

It is contended that the Licensing Basis requirements of FSAR 9.2.4 indicated the need for the Component Cooling loops in Modes 5 and 6. It is agreed that certain components cooled by the Component Coolaing System may be needed at least in Mode 5.

Recommendation:

Change the Technical Specifications to add an appropriate requirement for Component Cooling availability in the higher numbered modes.

Category A

(9) Concern 33 Technical Specification 3/4.7.7 Nuclear Service Water System

Question:

Proposal to add Modes 5 and 6 to the current TS 3.7.4 which requires operability of the Service Water System in Modes 1, 2, 3, and 4.

Discussion:

The Service Water System is needed to support RHR in modes 4 ghrough 6 and to service AFW alternate mode 5 cooling requirements in the event of a fail-closed RHR/RCS isolation valve in the supply to the RHR system.

Recommendation:

Change the Technical Specifications to add an appropriate requirement for Service Water System operability in modes 5 and 6.

Categor: ·

(10) Concern 38 Technical Specification Table 2.2-1, Item 18b - Low Power Reactor Trips Block, P-7. (Item 19.b in STS, Rev. 5)

Question:

Proposal to change the title of P-7 interlock from "low power reactor trips block" to "high power reactor trips block."

Discussion:

Interlock P-7 provides that when reactor power is low (below the setpoint of 10% equivalent power as sensed by P-10 at 10% of rated thermal power and P-13 at 10% of rated thermal power turbine impulse pressure equivalent) certain reactor trips are blocked.

The interlock description could be either high power or low power, depending on the approach taken to define the function.

One description is not considered preferable to the other except for the momentum of present use in the industry and the current level of training and documentation to support it. The anti-thesis is obvious, the effort, confusion and expense to the NRC and licensees in retraining and documentation change makes the desirability of such a change highly questionable. Additionally, there is a certain consistency in the present approach. The title "Low Power Reactor Trips Blocks" is actually blocking trips while the reactor is at low power--although the trips are applicable to high reactor power. The interlock "Intermediate Range Neutron Flux, P-6" has setpoints in the intermediate range while actually blocking or releasing a trip that is applicable to the source range.

Recommendation:

No change to the description of the P-7 interlock in the Westinghouse STS is recommended.

Category B

(11) Concern 3 Technical Specification Table 2.2-1, Reactor Trip Instrumentation Setpoints, Item c (Item b in STS, Rev. 5)

Question 1:

1-c (iii) General Statement that the absence of permissive P-7 introduces new events to evaluate for safety. It is stated that "...design basis events only define the outer envelope of expected severity which is expected to cover a large number of less severe occurrences, undefined."

Discussion:

1-c (iii) The consideration of low power accidents not being explicitly covered by the bounding cases of the FSAR accident analyses is an apparently valid observation of the McGuire Accident Analysis section (Chapter 15 of FSAR) or of the RESAR 35 - Chapter 15 analysis. These chapters discuss the four classes of accidents and groundrules for analysis but do not specifically treat accidents initiated from low power. The accidents covered do assume conditions for the "worst case." For accidents which have a failure consideration of not meeting the DNB criteria, worst case is the full power case. For a limited number of other accidents (e.g., radiological release, loss of coolant inventory) there may be a need for an additional evaluation at low power to ensure that the variation in the low power protection system or some other variant does not lead to a more severe situation.

Recommendation:

It is recommended that a review of accidents at low power (outside the scope of RESAR-35) be investigated to assure no cases exist that are not bounded by the design basis accidents.

Category B

(11) Concern 3 Technical Specification Table 2.2-1, Reactor Trip Instrumentation Setpoints, Item c (Item b in STS, Rev. 5) (Continued)

The high water level trip setpoint provides sufficient margin such that the undesirable condition of discharging liquid coolant through the safety valves is avoided. Even at full power conditions, which produces the worst thermal expansion rates, a failure of the water level control does not lead to any liquid [discharge through the safety valves. This is due to the automatic high--RESAR correction, P 7.2-47] pressurizer pressure reactor trip actuating at a pressure sufficiently below the safety valve setpint, or to the high pressurizer water level reactor trip.

Category B

(11) Concern 3 Technical Specification Table 2.2-1, Reactor Trip Instrumentation Setpoints, Item c (Item b in STS, Rev. 5) (continued)

For control failures which tend to empty the pressurizer, two out of four logic for safety injection action on low pressure ensures that the protection system can withstand an independent failure in another channel. In addition, ample time and alarms exist to alert the operator of the need for appropriate action."

Further, the pressurizer high water level logic of McGuire FSAR figure 7.2.1-1 page 6 of 16 shown the alarm function to be at the controller level and not subject to the 2/3 logic module for the reactor trip. Therefore, the alarm will function if any controller reaches an alarm setpoint.

Recommendation:

No change to the Westinghouse STS is recommended for the block of pressurizer water level-high trip at low reactor power below interlock P-7.

Question:

3-c (ii) It is stated that is is not appropriate to remove the automatic protection of the RCS boundary - referring to blocking of pressurizer water level-high trip below P-7 (10% rated thermal power). Discussion proposes that this trip be effective whenever there is a bubble in the pressurizer.

Discussion:

3-c (ii) The statement that it is not appropriate to remove the automatic protection of the RCS boundary is countered in the explanation given in the McGuire FSAR section 7.2.2.3.4 in discussion of Question 2, above. The automatic protection of the RCS boundary is not solely dependent on the subject reactor trip. The boundary is protected by the pressurizer with bubble, PORVs, and Safety Valves. The reactor trip is applicable only in the 10% and higher power level portion of mode 1, it is not applicable in the other modes as discussed above.

Recommendation:

No change is recommended to the Westinghouse STS with regards to protection of the RCS boundary.

Category B

(12) Concern 10 Technical Specification page 3/4.3-2

Question:

1 - Evaluate the safety consequences of main steam line break below the P-11 interlock where reactor trip will not be initiated by the Negative Steam line pressure rate-high signal.

2 - If the above steam line break is inside containment will containment pressure-high initiate reactor trip within an acceptable time.

3 - What are consequences of a small to intermediate size break inside containment where such containment pressure-high may not occur. Comment in terms of reactor trip system instrumentation requirements to meet these circumstances.

4 - Discussion of application of Safety Analysis Limit using +10% power applied to neutron flux low setpoint of 25% to yield 35% conservative outer limit. States "if this same (total channel error) margin was applicable to both P-10 and P-13, then P-7 would be 20%.

(Remember P-7 is either/or from P-10 and P-13)

Discussion:

1- Section 6.2 - Containment System of SER Supp 4. to McGuire Reference 14 (NRC report reference) addresses small steam line breaks in detail. "In all cases a containment lower compartment pressure high enough to initiate automatic operation of the sprays and fans was calculated in the LOTIC-3 analysis of the event." On page 6-3, after the discussion, the following statement appears:

Category B

(12) Concern 10 Technical Specification page 3/4.3-2

(continued)

"We conclude that the results of the LOTIC-3 analyses for the "generic" ice condenser plant result in containment temperatures equivalent to those which would be calculated specifically for the McGuire Nuclear Station and are therefore acceptable for use on McGuire. We will use the results of the complete spectrum of steamline breaks to assess the equipment qualification tests performed on those instruments and equipment located in the containment lower compartment which are rquired to detect a steamline break, initiate safety system functions, and monitor the course of the accident. This review complies with General Design Criteria 16, 38, 39, 40 and 50. We consider this matter resolved."

Recommendation:

It is recommended that a review of accidents at low power (outside the scope of RESAR-35) be investigated to assure no cases exist that are not bounded by the design basis accidents.

Category B

(13) Concern 12 Technical Specification Table 3.3-3, ESFAS, Item 11
Same as A-15. See item number 4 above.

Category B

(14) Concern 15 Technical Specification 3/4.4.1 Reactor Coolant Loops and Coolant Circulation

Question, discussion and recommendation presented for each item number of G-2

G.2.1

Question:

The proposed increased boron concentration discussed under section 3/4.1.1 of Category A would result in a number of questions as stated in G.2.1.1 through G.2.1.4.

Discussion:

No resolution is required because the basic issue was closed in concern 7 of Category A.

Recommendation:

None required.

G.2.2, G.2.3, G.2.4, G.2.5
Question:

States that the proposed TS are non-conservative with respect to Regulatory requirements and the licensing basis because of removal of operability requirements for all safety-related reactor trips (except SI) in Modes 3, 4 and 5. It is also stated that specifically 5 trips are blocked by P-7 and 2 by P-8, but 9 remain from which automatic protection can potentially be provided and which were removed by unique action of the TS without any safety evaluation.

G.2.2, G.2.3, G.2.4, G.2.5
Discussion:

Operability requirements for reactor trips in modes 3, 4, and 5 are by definition, unncecessary except when withdrawing rods for testing.

Mode definition for modes 3, 4 and 5 includes reactivity conditon of ≤ 0.99 K_{eff}. Except where discussed elsewhere for such testing, reactor trips in higher numbered modes 3, 4, and 5 are not required.

Recommendation:

None required.

Category B

(15) Concern 20 Technical Specification 3/4.7.5 Standby Nuclear Service Water Level (Ultimate Heat Sink)

Question:

Proposal to add modes 5 and 6 to tech spec 3/4.7.5 which requires operability of the ultimate heat sink in modes 1, 2, 3, and 4.

Discussion:

The ultimate heat sink operability requirements should support heat removal in all modes where component cooling water and nuclear service water systems are in use for heat removal.

Recommendation:

Change the STS to add appropriate operability requirements for the ultimate heat sink in modes 5 and 6.

Note: This should include appropriate action statements or modifiers to allow for periodic maintenance.

Category B

(16) Concern 21 Technical Specification 3/4.9-11 Refueling Operations Low Water Level

Question 1:

Proposed to add item to statement of applicability to the effect that: This MODE shall not be used for continuous normal operations, but only as a set of circumstances occurring during the period in which the Reactor Vessel Head is being untensioned and removed and the reactor cavity and refueling canal are being filled, and the same volumes are being drained for replacement and tensioning of the Reactor Vessel Head.

Discussion:

Recommendation: To be determined.

Category B

(16) Concern 21 Technical Specification 3/4.9-11 Refueling Operations Low Water Level
(continued)

Question 2:

Proposed that provision from FSAR 5.5-24 for each of the RHR trains to be powered from different sources.

Discussion: The requirement for redundant power supplies is sufficiently described in the systems description. There is no need to elaborate on this particular aspect of the system design within the tech spec LCO.

Recommendation: No change to the Westinghouse STS is recommended.

Category B

(16) Concern 21 Technical Specification 3/4.9-11 Refueling Operations Low Water Level

(continued)

Question 3:

Proposed that two safety-related RHR low flow alarms be required on each operating system so that the operator can respond within 10 minutes to commence operation of the redundant system.

Discussion: To be determined.

Recommendation:

Category B

(16) Concern 21 Technical Specification 3/4.9-11 Refueling Operations Low Water Level
(continued)

Question 4:

Proposed to evaluate necessity of maintaining two operating RHR systems in mode 6 so boiling will not occur on the loss of one system with a single failure.

Discussion: McGuire and RESAR-35 - Bases support 1 RHR loop in mode 6. The McGuire basis states that in the event of loss of the operating loop adequate time is provided to initiate emergency procedures to cool the core. RESAR-35 states that one system ensures sufficient cooling capacity to remove decay heat.

Recommendation: To be determined.