JUN 25 1984

MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

THRU: R. Wayne Houston, Assistant Director
for Reactor Safety
Division of Systems Integration

FROM: Brian W. Sheron, Chief
Reactor Systems Branch
Division of Systems Integration

SUBJECT: REVIEW STATUS OF TECHNICAL ISSUES
ON MCGUIRE TECH SPECS

Reference:

1. Memorandum, Mattson to Denton, "Status Report on the Review Plan for Technical Issues on McGuire Tech Specs" dated May 17, 1984

2. Memorandum, Licciardo to Sheron, "Review of McGuire Technical Specifications," dated June 11, 1984

3. Memorandum, Mattson to Denton, "Status Report on the Review Plan for Technical Issues on McGuire Tech Specs," dated May 17, 1984.

In the Reference (1) memorandum we advised we would provide you with an updated status report on the progress of resolution of the technical issues on the McGuire Technical Specifications.

Mr. Licciardo has now formally completed documentation of the clarification of his technical issues in his Reference (2) memorandum.

As you were advised in Reference (3), the clarifications provided with Reference (2) contain additional new concerns identified by Mr. Licciardo since his original DPO, and his clarification document is 111 pages long with approximately 5 to 6 items identified per page. Thus I estimate he has identified some 500 to 600 items.

I am initiating RSB management review of Reference (2). I estimate it will involve approximately .5 PSM of RSB management time to review and categorize his issues.

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We will then forward to DL the results of our review. I plan to complete the RSB management review by July 13, 1984.

I plan to provide you with a status report on the results of the RSB management review shortly after it is completed.

Unless I hear from you, I will assume our approach and review plan is acceptable to you.

Original signed by:
Brian W. Sheron

Brian W. Sheron, Chief
Reactor Systems Branch
Division of Systems Integration

cc: E. Case
D. Eisenhut
D. Muller
L. Rubenstein
C. Thomas
F. Miraglia
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 3 0 1984

MEMORANDUM FOR: Darrell G. Eisenhut, Director Division of Licensing

FROM: Robert M. Bernero, Director Division of Systems Integration

SUBJECT: CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

Reference: Memorandum, Sheron to Denton, "Review Status of Technical Issues on McGuire Tech Specs" dated June 25, 1984

In the reference memorandum, Mr. Denton was advised that the RSB management would review the concerns of R. Licciardo on the McGuire technical specifications as he clarified them in his June 11, 1984, memorandum and forward the results to DL. RSB has completed its review and categorization of the concerns, and this memorandum forwards the results to your office for disposition.

In summary, we identified no concerns of safety significance that required immediate action, and all concerns could be addressed as part of the process described later on in this memo.

Our categorization process eliminated those concerns that RSB management felt were either not appropriate for technical specifications or still did not clearly specify the issue. The remaining concerns were categorized as either category A, those concerns that were plant specific within the scope of the standard Technical Specifications and were appropriate to ask an applicant, and category B, concerns that were felt to be philosophic in nature, questioning the scope and content of the technical specifications.

The category A concerns are provided in enclosure (1) and the category B concerns are provided in enclosure (2).

With regard to the category A items, these are questions which the RSB management felt were appropriate to be asked of an applicant, but not necessarily considered to be final "positions." Based on the response, the staff would have to decide whether it was acceptable or if changes to the McGuire and standard technical specifications were warranted. If it were the latter, we would follow the Office Letter 38 guidance.

We also note that the categorization process was done by 5 managers. Different judgments could result in some differences in categorization. You should therefore feel free to recategorize those items you believe are miscategorized.

We have worked with Cecil Thomas of your staff and have agreed on the following approach to final resolution:

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1. DL will review the category A and B items and identify those for which they believe acceptable answers already exist for the Technical Specifications. These concerns and the answers will be documented by DL.

2. Of the remaining concerns, DL will review the categorization and revise them as necessary into items which are plant specific to McGuire, items which are generic, and items which are applicable to both.

3. For those items that are generic, they will be returned to DSI by DL for consideration by DSI for incorporation in the next periodic update of the standard technical specifications in accordance with the provisions of Office Letter 38.

4. For those items that are plant specific, DL will determine how to address them with the McGuire licensee.

DSI (RSB) will assist DL as necessary in carrying out these final steps of the resolution plan.

Robert M. Bernero

Robert M. Bernero, Director
Division of Systems Integration

Enclosure:
As stated

cc: H. Denton
E. Case
D. Crutchfield
C. Thomas
F. Miraglia
D. Brinkman
R. Birkel
T. Novak
E. Adensam
RSB S/L's

Enclosure 1

B

CATEGORY A items

SECTION 2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

The proposed T.S. requires that: "The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for four and three loop operation, respectively.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirem of Specification 6.7.1."

EVALUATION

a) Concerning the title: SAFETY LIMITS/REACTOR CORE. Clarify if the numerical values in Figure 2.1 are meant to be Safety Limits, Limiting Safety Settings or Set Points.

d) References to three loop operation should be deleted as the plant is not licensed for such operation.

"REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1."

EVALUATION

c) Where in the RCS system is the pressure limit to be observed eg Reference 10, page 15.4-20, Revision 7 first para. shows that: "To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure. What provision has been made in the specified value or related instrumentation to conservatively account for this necessary correction.

tion of the RCS pressure to its required value for steady state operation rather than within the 2735 psig limit.

Should MODE 3 also be included in the action statement for MODES 1 & 2 as generally identical concerns prevail except for the limited Applicability of Appendix G in T.S. Figs. 3.4-2.

f) Concerning MODES 3, 4 & 5.

How is the pressure limit of 2735 psig applicable to MODES 4 and 5 when reduced RCS temps. will cause consideration of constrained Pressure/Temperature limits [to Appendix G requirements] in T.S. Section 3/4.4.9.

Further, even MODE 3 has an Appendix G limits of <2500 psig at RCS temps. of <350°F; reference T.S. Figs. 3.4-2.

TABLE 2.2-1. REACTOR TRIP INSTRUMENTATION SET POINTS

These have been checked against reference 18, Westinghouse (W) RPS/ESFAS Set Point Methodology, Table 3-4 and NOTE FOR TABLE 3-4 on page 3-13, which is described as applicable to McGuire Unit 1, 50-369. At this date, the assumption has been made that this information also applies to McGuire Unit 2, Docket No. 50-370. Please docket this fact or otherwise provide the alternate information.

Item 3. Power Rate, Neutron Flux, High Positive Rate

Will a time constant of >2 seconds result in a slower response time, which is less conservative.

Item 4. Power Rate, Neutron Flux, High Negative Rate.

Will a time constant of >2 seconds result in a slower response time which is less conservative?

Reference 18 page 3-13, concerning Set Point Methodology advises that this value is not used in Safety Analyses. This appears in direct contradiction to reference 7, Section 15.2.3, page 15.2-12, revision 7, first para. The Licensee shall evaluate and propose

Item 9: Pressurizer Pressure-Low

The specified Trip Setpoint & Allowable values agree with those provided under setpoint methodology in reference 18. A disparity does exist between the related SAFETY ANALYSIS LIMITS given as used in Safety Analysis, i.e, 1845 psig in SETPOINT METHODOLOGY/Reference 18, Table 3-4, column 12 and the FSAR value for the same analysis in reference 7, Table 15.1.3-1 as 1835 psig. The Licensee shall identify the correct value. [Note also disparity with reference 7, "Analysis of Inadvertent Operation of ECCS During Power Operation", page 15.2-40, revision 43 item 7, "Reactor Trip ----- is initiated by low pressure at 1800 psia;" This is however relatively conservative with respect to the other values used above.]

The Licensee shall review and clarify.

TS Page 3/4 [illegible] MINIMUM TEMPERATURE FOR CRITICALITY

The existing minimum temperature for criticiality (in MODES 1 and 2) is given as 551°F. Please advise why this value is less than the programmed set point minimum value of 557°F in reference 20, fig. 5.3.3-1. Accident evaluations for events from zero power are predicated upon this set point of 557°, and any variation therefrom in either direction would be unacceptable. Reference our comments under Section 2.1.1.f.

T.S. Page 3/4 1-10 Concerning: CHARGING PUMPS - OPERATING AND APPLICABILITY MODES 1, 2, 3 and 4

This is directly related to the proposed changes under Item T.S. Page 3/4 1-8 of this report. Consistent with that discussion, the title should be changed to delete MODE 4, and MODE 3 conditioned to (down to 1000 psig/425°F) Item 4.1.2.4.2 under SURVEILLANCE REQUIREMENTS does not now apply since it refers to conditions ≤ 300°F which are not now covered by this section, being limited to a minimum of 1000 psig/425°F in MODE 3. The same comment applies to footnote #__ concerning one only centrifugal charging pump at ≤ 300°F.

The proposed T.S. is non-conservative with respect to the Licensing Basis. The Licensee shall evaluate and propose

T.S. Page 3/4 1-11 Concerning: BORATED WATER SOURCE - SHUTDOWN

Additionally, [by letter to reference 17] the Licensee has committed to provide and T.S. an operable level detection system with a specified "minimum level". This has not been included in the T.S. and it is proposed that it form the subject of an additional item 3.1.2.5.a.4). Surveillance requirements should be included under 4.1.2.5.a.4) in which the borated water source would be demonstrated OPERABLE by verifying minimium levels in the system.

T.S. Page 3/4 1-12 concerning: BORATED WATER SOURCES - OPERATING (in related Applicable MODES 1, 2, 3 and 4)

Additionally, [by letter to reference 17] the Licensee did commit to provide and T.S. an operable level detection system with a specified minimum level. This has not been included in the T.S. and it is proposed that it form the subject of an additional item 3.1.2.6.a.4). Additional surveillance requirements should be included under 4.1.2.6.a.4) in which the borated water source would be demonstrated OPERABLE by verifying minimum levels in the system.

Clarify whether the LCO values given are Safety Analysis Limits or Set Point Limits.

An appropriate modification may need to be made to the Boron Concentrations and volumetric requirements in the Boric Acid Storage System in MODE 3 down to 1000 psig/425°F to provide for the increased Boron Concentrations required from the Licensing Basis in this MODE discussed in this report under TS page 3/4 1-1, 2 and 2a.

T.S Page 3/4 3-13a. Proposed concerning: INSTRUMENTATION IN MODES 3, 4, 5 and 6

SER Supp 1, reference 11 page 15-2 requires a Technical Specification that "During startup and shutdown, the applicant will rely on the source range high flux alarms to alert the operator that a dilution event is occurring. This assessment is based on setting the alarm at a level of 5 times the background level. The licensee is to maintain the source range alarm setpoint at this level or lower any time the plant is in the cold shutdown Mode. The set point is to be checked and adjusted on a weekly basis if in the cold shutdown mode for an extended period."

Section 3/4.2.5 D[illegible] [illegible]AMETERS AND TABLE 3.2-1 DNB PARAMETERS

The current information does not adequately represent all those parameters necessary to ensure "acceptable" RCS operations, including DNB, under all Licensing Basis Conditions II, III and IV.

The necessary parameters are discussed and described under Section 2.1.1 Reactor Core, item f, of this report. If they are logically represented under 2.1.1. [and elsewhere], why are they also represented here?

b) Concerning Table 3.2-1. The value for Reactor Coolant System T_{avg} given as $\leq$ 593°F is not in accordance with the FSAR, reference 3, Figure 5.3.3-1 where a value of 588.1°F is given as the programmed T_{avg} for RATED THERMAL POWER Conditions. Please explain the difference and explain why setpoint and allowable values should not be provided. As a Setpoint, the proposed TS value is non-conservative with respect to the Licensing Basis.

Please explain why a related power level has not been ascribed to this temperature.

Please explain why programmed T_{avg} of 557.0°F (also reference 3, Figure 5.3.3-1 has not been given for zero power operation (Reference again our Section 2.1.1 item f).

c) Concerning Table 3.2-1 Pressurizer Pressure. Please explain the basis for the given value of $\geq$ 2230 psia when information in reference 20, Table 4.1-1 (1 of 3) shows a "System Pressure, Nominal" of 2250 psia and Section 15.1.2.2, Table 15.1.2-2 makes provision for a total of 30 psi for steady state fluctuations and measurement error. Have you quoted a Setpoint value, or an allowable value; both should be available. As a Setpoint, the proposed T.S. value is non-conservative with respect to the Licensing Basis for DNBR, and conservative for overpressure protection.

d) Why should not programmed T_{avg} be provided under T.S. Section 2.1.1

e) Why should not Pressurize Pressure be included both under T.S. Section 2.1-1 and T.S. Section 3/4.4.3 Pressurizer.

TABLE 3.3-1 REACTOR TRIP SYSTEM INSTRUMENTATION

T.S. Page 3/4 3-2.

Item 6c: Source Range, Neutron Flux

During shutdown in MODES 3, 4 and 5, with reactor trip system breakers open, Source Range, Neutron Flux, channel operability requirements specify only one channel operable, and if this same channel is being used to meet the Boron dilution alarm requirements of proposed T.S. Page 3/4 1-13 (a), then it is not in accordance with the Boron Dilution Requirements of the FSAR for which at least 2 operable channels would be required; reference 8, page Q212-24, item 212.58. The Licensee shall evaluate and propose. Currently, this appears non-conservative.

Item 6a: This Technical Specification concerning Operability of the Source Range Neutron Flux is unclear. It species operability of the Source Range Neutron Flux trip Below the P-6 (intermediate Range Neutron Flux Setpoint) during startup in MODE 2; the Licensee shall advise if this "start up" channel is required to be Operable to get Reactor trip in MODES 3, 4 and 5.

Items 1 through 5: The FSAR, Reference 5, Table 7.2.1-4 1 of 5 shows the Power-Range Neutron Flux Trip Low Setpoint and High Setpoint, and the Intermediate Range High Neutron Flux Trip, and the Source Range High Neutron Flux Trip, all being used on events being initiated from a "subcritical" condition. However, Table 3.3-1 shows that except for the Source Range Neutron Flux items 6b and 6c, all the Trips are inoperable in the subcritical MODES 3 through 5. Further, there is a note d) in the column entitled Tech. Spec(c) of Table 7.2.1-4 which states that "A technical specification is not required [for the Intermediate Range High Neutron Flux Trip and Source Range High Neutron Flux Trip] because the trip function is not assumed to function in Accident Analyses. Please note further that this position is followed through in Table 3.3-2 Items 5 and 6 in that a response time is not provided for the Intermediate and Source Range Neutron Flux trips, because it is proposed as NA (Not Applicable). Please evaluate the apparent paradox that the Source Range Trip is the only nuclear Flux trip required to be OPERABLE in the subcritical MODES 3 through 5, and yet there is no Tech Spec proposed for it. At this moment, absence of OPERABILITY requirements for the Power Range Neutron Flux Trip, Low Setpoint, in MODES 3 through 5 would appear to constitute a disparity with the Licensing Basis FSAR and in a less than conservative manner. The Licensee shall evaluate and propose, those safety-related neutron Flux trips which would be appropriate to use and available to trip the reactor for any of those events causing a return to power and under circumstance in which a safety injection initiator is not available, during MODES 3, 4 and 5; and provide the related Set Points, Allowable Values and Safety Analysis Limits. Alternately, the Licensee shall define and T.S. those conditions and parameters in accordance with 10 CFR 50.36, which would prevent any such event occurring.

Why should not this be required for MODES 3, 4 and 5 (with closed loops) to embrace the possibility of a return to nuclear power under these conditions. Further, Steam Generator Operability is also required in these Modes to remove decay heat, and Low-Low level alarms are derived from the steam generator low-low instrument channels. Reference 5, Figure 7.2.1-1. The Licensee shall evaluate and propose.

Item 17: Safety Injection Input From ESF.

The proposed T.S. proposes that Reactor Trip on ESFAS (or S.I) is not required to be OPERABLE in MODES 3 and 4. Why is reactor trip not required in these MODES when Table 3.3-3 for ESFAS Instrumentation, and more particularly Functional Unit 1, including Reactor Trip, shows operability requirements down to and including MODE 4. Further, the licensing basis provides that SI, including reactor trip, be initiated automatically and manually down to MODE 4; see Licensing Basis information in later Section 4.5, EMERGENCY CORE COOLING SYSTEMS, under GENERAL, of this review.

This proposed T.S requirement is therefore non-conservative with respect to the Licensing Basis which requires that Reactor Trip on ESFAS (or SI) be Operable in MODES 1, 2, 3 and 4. The Licensee shall evaluate and propose.

TABLE 3.3-2 REACTOR TRIP INSTRUMENTATION RESPONSE TIMES

Items 5 and 6: Intermediate Range and Source Range Neutron Flux Trips.

As indicated under item Table 3.3-1, items 1-5, these items are proposed as not being protective actions necessary for the FSAR. Analyses already requested will provide a base for determining whether those trips are necessary to protect the plant in MODES 3 through 5. If so, please provide the necessary technical specifications for these response time in conformance with 10 CFR 50.46. If these values are not provided, all related return to reactivity events shall be evaluated by the Licensee with current FSAR requirements for the Safety Analyses Limit of the power range, neutron flux, low setpoint trip which will be required to be OPERABLE.

The current proposals for these trips is non-conservative with respect to other proposals in the T.S; the Licensee shall evaluate and propose.

Item 8: Overpower ΔT.

No response time is provided by the Licensee who proposes that a T.S. on this is Not Applicable.

Please comment on the fact that this reactor trip is proposed in Reference 5 Table 7.2.1-3 (3 of 5) as applying to five (5) separate Condition II through IV licensing basis occurrences. Also that Reference 5, Page 7.2-14 Rev.42, item 1 c) specifies a maximum of 6.0 seconds (including a transport time of 2 secs) and which is confirmed by Reference 7, Table 15.1.3-1 [alongside Overpower ΔT].

The proposed T.S is non-conservative with respect to the Licensing Basis. The Licensee shall evaluate and propose.

Item 9: Pressurizer Pressure - Low

The TS specifies a Response Time of $\leq$2.0 secs. Reference 7, Table 15.1.3-1 provides a time delay of 2.0 secs for these events which conflicts with a value of 1.0 secs in Reference 5, page 7.2-14, rev. 42, item 1(e). The Licensee shall clarify.

Item 11: Pressurizer Water Level - High

No response time is provided because it it considered Not Applicable (NA).

The trip is shown as having a protective function for two Condition II occurrences in Reference 5, Table 7.2.14 (4 of 5) and a potential protective function in a Condition IV occurrence in Reference 7 page 15.4-13, item 16 c.

Additional protective functions are discussed earlier under Table 3.3-1, item 11.

Reference 5, page 7.2-14, Revision 42, Item 1 f provides a reactor trip response time at 1 sec.

Reference our earlier review under Table 2.2-1, item 18.c.(ii).

In view of the above information, the proposed T.S. is non-conservative with respect to the Licensing Basis. The Licensee shall evaluate and propose.

~~Items 8 & 11 General~~

~~Although the above two items are not apparently the primary reactor trips used as the basis for calculating protection in the Accident Analyses in reference 7, those Analyses represent a limited number of events which are proposed as "expected" to bound all possible events at the plant in terms of severity. There is no guarantee that the large number of other possible events will never use these two protection items to primary advantage.~~

Item 17, [Reactor Trip on] Safety Injection Input from ESF

This description is a misnomer and should be replaced by the description proposed under Table 2.21, Item 17 of this document.

The proposed T.S. states that the response time requirement is NA (Not Applicable). This is incorrect as a separate Reactor Trip is an essential part of all ESFAs functions during which safety injection is initiated. The required information is in fact supplied in T.S. Page 3/4 3-30 Table 3.3-5, under the already revised headings proposed above, reference items 1i, 2b, 3b, 4b.

This table, under response time, should replace the description as recommended above and alongside each, reference the entry in T.S. Table 3.3-5.

The response given in the Technical Specifications (except for Manual actuation of SI) are quoted as $\leq$ 2 secs. No docketed information is available on what values were used in accident analysis, and particularly for MSLB, SBLOCA and LOCA events. The licensee should provide this information and confirm its conservatism against the T.S. value, eg. reference 5, Table 7.2.1-4 (5 of 5) and related note e. on page entitled "Notes for Table 7.2.1-4" confirms that Pressurized Low Pressure - Low Level is the first out trip of Safety Injection for the event of "Accidental Depressurization of the Main Steam System." The licensee shall explain this terminology - whether we have Reactor Trip on Pressurizer Pressure - Low which is available at the maximum power output at which this particular event is evaluated, or Pressurizer Pressure - Low (Safety Injection) and provide the associated response time to validate proposed T.S.

Item 1a): Manual Initiation

This should read as: Manual Safety Injection Actuation. [There is not a separate Manual Actuation for each of the functional units listed.]

Item 1e:

The proposed T.S. for SI on Steam Line Pressure - Low is qualified in MODE 3 by a 3## which is identified on T.S. Page 3/4 3-23 as a situation in which the function may be blocked below P-12 (Low-Low T_{avg} Interlock) setpoint.

Reference 5, Table 7.3.1-3 (1 of 2) and (2 of 2) item P-1, shows the appropriate interlock for this purpose is P-11. Item P-12 of the same Table makes no provision for this proposed T.S. position.

However, reference 5 figure (6 of 16) does not use the same manual block (at P-11) for Pressurizer Pressure - Low (SI) as for Steam Line Pressure - Low (SI) (and implementation of Negative Steam Line Pressure Rate) on reference 5, Figure (7 of 16). The Licensee is required to confirm that no parameter other than the value of Pressurizer Pressure (at P-11) is used to condition the manual blocks relating to the steam line; if other parameters are used, the Licensee shall evaluate and propose. The Licensee shall also advise of other parameters which may be used to condition the manual block of Pressurizer Pressure - Low (SI).

If the Table 7.3.1-3 (1 of 2) and (2 of 2) is correct, then condition MODE 3## should be changed to condition MODE 3# which becomes the correct description.

Item 3.b3): Containment Phase B Isolation on Containment Pressure - High High

Operability of this isolation is not provided in MODE 4. The Licensee should advise why this is not necessary for safety when the previous item No.1.e.

showed reference in the Licensing Basis of protection against Steam Line Break inside containment and Large Break LOCA in this mode. It should be noted that T.S. Item 3.4.6.1 requires containment integrity in MODES 1 through 4.

Further Operability of Auto-Actuation Logic is required through MODE 4 [Containment Pressure-High only effects Containment Isolation A and not Containment Isolation B which is necessary to establish Containment Integrity].

The proposed T.S. is non-conservative. The Licensee shall evaluate and propose.

Item 4d: Negative Steam Line Pressure Rate - High

Operability requirements are given as MODE 3 and 4. MODE 3 should be conditioned as MODE 3# indicating it is only available below P-11 Interlock. The Licensee shall evaluate and propose.

Item 7.e: Start Turbine-Driven Pumps (by SI)

This functional unit proposes that the Turbine Driven AFW pumps are started by the SI signal. This conflicts with reference 5, Fig. 7.2.1-1 (15 of 16) I&C system Logic Diagram where the initiation of the turbine driven pumps on SI is not shown. Also, in a like manner, with related section 7.4.1.1.2.1. and reference 22, section 10.4.7.2.2.6. Also see reference 14 Section II.E.1.2 page 22-41. It is now noted that the recent T.S. has been corrected to show that the Turbine Driven AFW pump does not start on Safety Injection.] The Licensee shall clarify.

Item 7.g: Trip of Main Feedwater Pumps (MFWP) - Starts Motor Driven Pumps

The T.S. proposed only 1 channel per pump to trip. [This is different to the FSAR, reference 22, page 10.4-14, rev. 7, item 30 which specifies that loss of <u>all</u> main feedwater pumps is required. The licensee should evaluate and propose.

Applicable modes: The current T.S. proposes Modes 1 and 2#. Condition 2# is an invalid MODE since # identifies the P-11 interlock which can be manually effected only at approx. 1900 psig and which can only occur in MODE 3, i.e., the condition should be 3#. The licensee should explain and propose.

Please advise why this limitation at MODE 2 [or 3]# is proposed and how it may relate to plant operating procedures in MODES 3 and 4 and whether this Block is in conformance with regulatory requirements.

TABLE 3.3-4: ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) INSTRUMENTATION TRIP SET POINTS

Item 4d. Negative Steam Line Pressure Rate - High [For isolation of the MSIVs below P-11 Block]

The trip set point is currently specified at -100 psi/sec. Westinghouse Set Point Methodology for Unit 1, reference 18, shows this value to be "-110 psi"; an additional descriptor is also necessary reading: "with a time constant of 50 secs". The current "Allowable Value" in the T.S. is -120 psi/sec, the same reference 18 Table 3-4 shows this value to be -100 psi; this should again have the additional descriptor reading: "with a time constant of 50 secs".

To discuss negative values and related conservatisms, it is clear to delete the - in -100 as the description reads : "Negative Steam Line Pressure Rate - High so that T.S. values should read as <u>100 psi</u> and <u>110 psi</u>. This is also internally consistent with the descriptor in Table 2.2-1, Item 4, namely: Power Range, Neutron Flux High Negative Rate, 5% of R.T.P with a time constant of 2 seconds.

Item 6a & b. Containment Pressure Control System

The licensee should provide the basis for these Set Points and Allowable Values.

Items 7c(1) and (2): Concerning start of Motor Driven and Turbine Driven Pumps

This technical specification provides that the motor-driven AFW Pumps start on low-low in one SG whereas the turbine driven pumps require low-low in two SGs. This appears to be in conflict with the accident evaluation in the Licensing Basis FSAR as elaborated below. [This however is not conflict with the Instrumentation & Control Logic of the FSAR.]

Item 8: Automatic Switchover to Recirculation

The Licensee shall provide the basis for the set point values of the RWST levels specified. What are the allowable values for [drift and] total channel errors and the related Safety Analysis Limit.

Item: General

The Licensing Basis FSAR, reference 7, Section 15.2.9 under LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES describes a set of Reactor Protection System and Engineered Safeguards Features Actuation Responses for the Plant, to ensure its safety. Why is this particular set of ESFA's Functional Units and related Instrumentation Set Points not provided in this item under Table 3.3-4?

Absence of this information makes the proposed T.S. non-conservative. The Licensee shall evaluate and propose.

Item 9: Loss of Power

Confirm the bases for the set points and allowable values specified.

Item 10b: ESFAS Interlock T_{avg}-P_{12}.

The basis for this interlock on T.S. Page B 3/4 3-2 states that:

"On decreasing reactor coolant loop temperature, P-12 automatically removes the arming signal from the steam dump system." This is not substantively consistent with Reference 5, Figure 7.2.1-1 which shows that it is the arming signal for the condenser dump valves and atmospheric dump valves which is removed and then with the exception of 3 cooldown dump valves (to the condenser). The steam generator Power Operated [atmospheric] Relief Valves (SG PORVs), are not affected. Please correct the Basis.

Item 11 Proposed:

There is a need to add a new Functional Unit not addressed in the current T.S., but which is a part of ESFAS. This is:

"Close Feedwater Isolation Valves & Close Feedwater Main & Bypass Modulating Valves." (See Reference 5, Figure 7.2.1-1 (13 of 16) Revision 34.)

This Functional Unit is initiated by:

a. Reactor Trip P-4, & Low T_{avg}.

b. Reactor Trip P-4, & Steam Generator Level - High High P-14.

c. Steam Generator Level - High High P-14 (see 5 above).

d. Safety Injection (see 1 above)."

Trip Set Points would be in accordance with the related values in earlier Items 10 and 5 of this section.

TABLE 3.3-5 ENGINEERED SAFETY FEATURES RESPONSE TIMES

Item 2a: Initiation of Safety Injection by: Containment Pressure-High.

A value of $\leq$ 27 secs (without offsite power) is given.

Reference 5, page 7.3-8 shows that initiation time of ESFAS from this source is a maximum of 1 sec.

No events in Reference 7, Section 15, have been directly analyzed using this sensor as the prime initiator above the P-11 interlock although it is relied upon for diverse protection. However, it is the only automatic initiation of Safety Injection protection below [P-11]. Other events dependent upon a SI generating signal, particularly circumstances descibed under items 3a and 4a below, shows safety analyses limits of $\leq$ 12 secs. (with offsite power) and $\leq$ 22 secs (without off site power).

At this time, the proposed T.S. value is less conservative than others used in Safety Analysis. The licensee shall evaluate this difference and propose accordingly.

Item 2b: Initiation of "Reactor Trip (From SI)" by Containment Pressure-High

The descriptor (From SI), should be deleted as it is incorrect.

The response time is give is $\leq$ 2 secs and this different from the FSAR, Reference 5, page 7.3-8 which gives a maximum time of 1 sec.

This value is less conservative than the FSAR and the licensee shall evaluate and propose accordingly.

Item 2d: Containment Isolation - Phase A, from Containment Pressure-High

The proposed T.S. values are 18(3) (with offsite power) and 28(4) without offsite power.

Reference 5, page 7.3-8 shows that initiation of ESFAS from this source is 1 sec.

Table 3.6-2 shows Maximum Isolation Times of up to 15 secs for Reactor Coolant Pressure Boundary Isolation valves. A minimum total time to containment and isolation [for the RCPB] of 16 secs seems feasible, plus 10 secs giving 26 secs total without offsite power.

The proposed T.S. values should be checked against those used as Safety Analysis limits for related Conditions II, III, and IV occurrences using SI. Values used by licensee shall be provided, compared with Item 2d. and any differences evaluated.

Item 2e: Containment Purge and Exhaust Isolation, from Containment Pressure-High

This is given as N.A. This is not so; response times have be used to minimize offsite consequences of any Condition occurring whilst containment purge & exhaust is being used. This proposed T.S. is less conservative than the licensing basis. The licensee shall evaluate & propose.

Item 2f: Initiation of Auxiliary Feedwater from Containment Pressure-High.

The licensee proposes N.A. but earlier review shows AFW initiation on Containment Pressure-High and especially in MODES 3 and 4.

This is less conservative than the licensing basis; the licensee shall evaluate and propose.

Item 3(a): "Safety Injection (ECCS)" on Pressurizer Pressure-Low [SI]

Values of $\leq 27^{(1)}/12^{(3)}$ secs are proposed.

Reference 5, page 7.3-8, shows a maximum initiating time of ESFAS 1.0 secs for this signal.

The value of 12 secs (with offsite power) is consistent with safety analysis limits given for the MSLB in reference 7, page 15.4-10, Section 7 where "In 12 seconds, the valves are assumed to be in their final position and pumps are assumed to be at full speed." For the other case with Loss of Offsite Power (LOOP) "an additional 10 secs. delay is assumed to start the diesels and to load the necessary equipment onto them." Further, this particular analysis appears to initiate the event on Pressure Pressure-Low (SI).

The proposed value of $\leq$ 12 secs appears within the licensing basis of 12 secs.

The proposed value of 27 secs (with LOOP) is however larger than the value of 22 seconds from the reference described above (i.e., 12 secs + 10 secs delay for start of diesel). This value of 27 secs therefore appears less conservative than the FSAR, reference 7, page 15.4-10, and the licensee shall evaluate and propose.

Item 3b: "Reactor Trip (from SI)" on Pressurizer Pressure-Low [SI]

The descriptor (from SI) is incorrect and should be deleted.

A value of $\leq$ 2 secs is proposed. The FSAR in Reference 5, page 7.3-8 quotes a value of $\leq$ 1 secs.

The proposed T.S. value appears less conservative than the Safety Analysis Limit and the licensee should evaluate and propose.

Item 3d: "Containment Isolation - Phase A" from Pressurizer Pressure-Low (SI)

The proposed T.S. is $\leq 18^{(3)}/28^{(4)}$ secs.

Reference our comments and requirements under 2.d. above.

Item 3e: "Containment Purge & Exhaust Isolation" From Pressurizer Pressure-Low (SI)

The proposed T.S. is NA.

Reference our comments and requirements under 2.e. above.

Item 3f: "Auxiliary Feedwater" Initiation by Pressurizer Pressure-Low (SI)

The licensee proposes NA (not applicable).

Safety injection logic closes the main feedwater isolation valves for every event in which SI is initiated (reference earlier sections of this review Table 3.3-4, proposed item c). Therefore, every such event initiated by a SI initiator must be analyzed with a restoration of AFW and a related response time.

It is outside the licensing basis, not to a propose a value for this response time. This T.S. value is therefore non-conservative; the licensee shall evaluate and propose.

Item 4e: "Containment Purge and Exhaust Isolation" on Steam Line Pressure-Low

The proposed T.S. is NA.

Reference our comments and requirements under item 2d. above.

Item 4h: Steam Line Isolation on Steam Line Pressure-Low.

The proposed TS value is ≤ 9 secs.

Reference 5, page 7.3-8 states that the maximum allowable times for generating steam break protection are (1) from steam line pressure rate, 2 secs, and (2) from steam line pressure-low, 2 secs. Further, Reference 7, page 15.4-6 states that the fast acting steam line stop valves are "designed so close in 5 secs...". A minimum closure of 7 secs seems likely.

For actual safety analysis limits, Reference 7, Table 15.4-1 (1 of 4) and 15.4-1 (2 of 4) both show a difference of seven (7) secs between arriving at the "Low Steam Line Pressure Setpoint" and "All main Steamline Isolation Valves Closed." [In the case of Feedwater System Pipe Rupture]

The proposed TS value of ≤ 9 secs is therefore greater than the Safety Analysis Limit.

The proposed TS must therefore be considered less conservative for this event. The licensee shall evaluate and propose.

Item 5a: "Containment Spray" - Initiated on Containment Pressure-High-High

Licensee shall provide the Safety Analysis Limit and compare with the proposed value of ≤ 45 secs. Evaluate and propose as necessary.

Item 6b: "Feedwater Isolation" Initiated by Steam Generator Water Level-High High

The proposed T.S. is ≤ 13 secs.

Reference 7, Table 15.1.3-1 shows that "High Steam Generator level trip of the feedwater pumps and closure of feedwater system valves, and turbine trip" is based on an ESFAS time delay of 2.0 seconds.

Table 3.6.2 of the T.S. provides isolation times of ≤ 5 secs for main feedwater containment isolation and ≤ 10 secs for main feedwater to Auxiliary Feedwater Isolation.

A total time to isolation of MFW of ≤ 13 secs seems appropriate to available equipment.

... of the current safety analysis depending on this response time is that for the Excessive Cooldown occurrence under Reference 7, page 15.2-28, and for this, no value is quoted for isolation of main feedwater which is the initiator of the event. However, Figure 15.2.10-2 shows that with initiation of the event caused by one faulty control valve, it takes 32 secs to reach the SG-High-High Level with a mass increase of 35% of initial, and thereafter does not increase further. This implies zero closure time. Since it is expected to take another 13 secs to actually isolate, we could assume an additional mass increase of another 13% to give a total of approx. 1.48 the initial value.

The above additional Main Feedwater level can affect the consequences of the event at power, if there has been a trip, with a potential for power restoration and/or overfill of the S-G to cause water ingress into the main steam lines. Additionally, it can have consequences of potentially larger importance for the event occurring from zero subcritical power.

Reference also our concerns under item Table 3.3-4, item 11b and 11a above.

The licensee shall evaluate the related concerns, including the extended MFW valve isolation times, to determine their safety significance, and propose as required. Until that time, it must be concluded that since a zero (0) value has been used in the current analysis, that the licensee has a potentially non-conservative situation with respect to Regulatory Requirements of Reactivity Control and Regulatory Concerns for Flooding of the Main Steam Lines.

Item 12: "Automatic Switchover to Recirculation" on Low RWST Level

Response time proposed as $\leq$ 60 secs

The licensee shall provide the bases for this value and evaluate against this $\leq$ 60 secs, and propose as necessary.

T.S. Page 3/4 4-2: RCS HOT STANDBY

The Action Statement allowing 72 hours with only one RCS loop operable is non-conservative with respect to the current Safety Analysis Limits.

T.S. Page 3/4 4-3. REACTOR COOLANT SYSTEM - HOT SHUTDOWN.

APPLICABILITY: MODE 4. [Less than 425 psig/350°F]"

The licensee shall evaluate as outlined earlier under Item, General, for RCS loops operability requirements and make proposals relative to the status of many elements of the protection and operations system to ensure that RCS safety is maintained for related Condition II, III and IV occurrences. At this time, with the proposed TS in which limited boration is used and Reactor Trip System Safety Related Instrumentation and Safety Injection Instrumentation are all but eliminated, the safety status of the facility is outside the Licensing Basis of the FSAR in a non-conservative manner.

Each of the OPERABLE loops, whether RCS or RHR, are to be energized from separate power divisions to protect against single failure of a bus or distribution system. When the RCS systems are used, the related Auxiliary Feedwater systems are also required to be operable.

The additional requirement proposed, for two RCS loops to be operable whenever RHR loop/s are in operation, is based upon reference 8, page Q 212-55 and 56, to provide for the failure of a single motorized valve in the RHR/RCS suction line in both MODEs 4 and 5 and possible non-availability of offsite power sources. The FSAR provides, that on failure of the valve:

> "Approximately 3 hours are available to the operator to establish an alternate means of core cooling. This is the time it would take to heat the available RCS volume from 350°F to the saturation temperature for 400 psi (445°F), assuming the maximum 24 hours decay heat load.
>
> To restore core cooling, the operator only has to return to heat removal via the steam generators. The operator can employ either steam dump to the main condenser or to the atmosphere, with makeup to the steam generators from the auxiliary feedwater system. The time required to establish the alternate means of heat removal is only the few minutes necessary to open the steam dump valves and to start up the auxiliary feedwater system."

The APPLICABILITY MODE 4, is necessarily qualified by [less than 425 psig/350°F] by the LOCA analyses already referenced above under our review Section 3/4 4.1 Subsection G.2.6.3 "Concerning Large Break Loss of Coolant Accident." See reference 8, page Q 212-47.d where it is described that

> "After several hours into the cooldown procedure (a minimum time is approximately four hours) when the RCS pressure and temperature have decreased to 400 psig and 350°F."

And arising from a later revision 25, the FSAR advises on page Q 212-61b revision 29 concerning ECCS calculations in a later submittal under Revision 28 that

> "The response provided in Revision 28 addressed the subject of operator actions and ECCS availability. Consistent with the information provided in Revision 28, a postulated LOCA in the RHR mode at 425 psig RCS pressure has been assessed."

Surveillance requirement 4.4.1.3.2 should verify S.G. water level at the Safety Analysis Limit for the Licensing Basis, which is the no-load programmed level, not the current proposed TS value which is the S.G. Low-Low Level [Reactor Trip] and AFW actuation. This proposed TS is non-conservative with respect to the current Safety Analysis Limits and the licensee shall evaluate and propose.

Surveillance requirement 4.4.1.3.3 verifying one loop in operation every 12 hours, is unsupportable as all protective trips on low flow in the RCP loops in this condition have been removed. If low flow channel trips on the RCP loops are not required to be operable why should the related Alarm be operable. A low flow alarm for the RHR has been provided by the FSAR under reference 8, page Q 212-56, item:

"Case 1: The Reactor Coolant System is closed and pressurized.

The operator would be alerted to the loss of RHR flow by the RHR low flow alarm. (This alarm has been incorporated into the McGuire design)."

Since currently, these two types of alarms are the only means of alerting the operator to a Loss of Flow condition in the loop, which is beyond the Safety Analysis Limits, then the alarms on both the RCS and Loop Flows should be Safety Related and included within the T.S.; and without further analysis at this time, two loops should be placed in operation. A proposal is made by the NRC for low flow alarms in each of the separated cooling systems, under Proposed T.S. Page 3/4 4-6a of this review. Regular surveillance should be proposed to ensure they remain operable as appropriate, over a specified surveillance period.

The Surveillance requirement, every 12 hours is intended to ensure not only that the system is operating, but that it is operating at process conditions which can be evaluated to show that the equipment is capable of performing its design basis Safety Function. The current surveillance requirements for this item, i.e., for the RCS and RHR systems in Hot Shutdown in T.S. Item 4.4.1.3.3, are absent this information; it is therefore non-conservative and the licensee shall evaluate and propose.

Item 4.4.1.4.4 (Proposed). It is proposed that an additional item be inserted which reads: "The related auxiliary Feedwater System shall be determined OPERABLE as per the requirements of T.S. 3.7.1.2 [and 3.7.1.2.a as applicable]." Current proposed T.S.s on T.S. page 3/4 7-4 are non-conservative in this matter by not providing any operability requirements for AFW in this MODE. The licensee shall evaluate and propose.

An additional item is also required in which Atmospheric Dump Valves operability is established. The current T.S. are non-conservative in this matter; they make no provision for operability of this item (see later proposed T.S. page 3/4 7-8a). [General comment: Operability of each of S.G. water level, AFW and ATMOSPHERIC DUMP VALVES in this MODE is probably better defined under each of these items in their particular sections of the T.S. See later sections of this review as identified above.]

<u>T.S. Page 3/4 4-5: COLD SHUTDOWN [MODE 5] WITH LOOPS FILLED</u>.

Use of secondary side water level of at least two steam generators is discussed in reference 14 for circumstances in which the RHR is isolated from the RCS and its final acceptability for licensing purposes is still not resolved. This, in addition to its temperature limitation means that it cannot be proposed as an alternate means of removing decay heat during Cold Shutdown. The proposed T.S. is therefore not in accordance with current Safety Analysis Limits, and also non-conservative.

As discussed in the previous item T.S. Page 3/4 4-3, what is required by the current Licensing Basis in Mode 5, is to have available two OPERABLE RCS loops [including AFW, SG and SG/PORVs] to meet the circumstances of failure closed of the RHR isolation valve and in which case the RCS returns to MODE 4 with its particular MODE 4 requirements as discussed earlier. The absence of this as an LCO requirement in the proposed T.S. makes it non-conservative with respect to the Licensing Basis. The Licensee shall evaluate and propose.

Footnote*: This item proposes that an only available operational RHR pump may be de-energized for up to 1 hr. This event has not been evaluated, is not within the Licensing Basis, and is non-conservative. The licensee should define the circumstances, analyze and evaluate and propose.

The proposed surveillance requirement/4.4.1.4.1.2 provides that "At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours. The items of significance here are Operable Safety Related Flow Alarms with a surveillance frequency ensuring high probability of alarm in the event of an RHR flow failure, and a related concern for overpressure protection and recovery. The licensee shall evaluate and propose.

The surveillance requirement, every 12 hours, is intended to ensure not only that the system is operating, but that it is operating at process conditions which can be evaluated to show that the equipment is capable of performing its Licensing Basis Safety Function. The current requirements for this information for the RHR systems in T.S. 4.4.1.4.1.2 are absent; it is therefore non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

T.S. SECTION 3/4.4.2 SAFETY VALVES

SHUTDOWN (MODES 4 and 5)

The T.S. requires that:

"3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%."

The Surveillance Requirements should contain the minimum discharge capacity required of this valve as defined in the Licensing Basis. They should also ensure the maintenance of satisfactory environmental conditions consistent with reliable valve operability. The licensee shall evaluate and propose.

T.S. Section 3/4.4.3 PRESSURIZER

T.S. Page 3/4 4-9

The APPLICABILITY MODES are proposed as 1, 2 and 3.

Item: Pressurizer Level:

The response of all the analyses of Condition II, III and IV events in references 7 and 8 depend upon an initial level of water in the Pressurizer which is programmed as a varying value dependent upon the Nuclear Power Level. Additionally, the response of all Condition I events which determine the most conservative set of parameters from which to start Condition II, III and IV events, are also so dependent upon this same programmed pressurizer level.

Since therefore this pressurizer level is used in establishing an acceptable outcome of these analyses in terms of the issuance of the operating license, they also represent limiting conditions of operation as defined in 10 CFR 30.46. On this basis therefore, the licensee should provide details of the programmed pressurizer level set points with allowable values consistent with the related channel errors and Safety Analysis Limits used in the FSAR, Section 15 in reference 7. The licensee shall evaluate and propose.

APPLICABILITY MODES: Pressurizer level should be proposed for MODES 1, 2, 3, and 4 (with steam bubble). Down to MODE 4 is provided to cover LOCA and MSLB events considered in reference 8. Also, the plant can then be placed on Automatic Level Control. Appropriate ACTION and SURVEILLANCE procedures should be proposed. Licensee shall evaluate and propose.

Item: Pressurizer Pressure

The responses of all the analyses of Condition II, III and IV events in references 7 and 8 depend upon an initial value of pressure in the pressurizer (and which is not programmed at a varying value in MODES 1 and 2). Additionally, the responses of all Condition I events which determine the most conservative set of parameters from which to start Condition II, III and IV events, are also so dependent upon this same pressurize pressure.

Since therefore this value of pressurizer pressure is used in establishing an acceptable outcome of these analyses in terms of the issuance of the operating license, they also represent limiting conditions of operation as defined in 10 CFR 30.46. On this basis, therefore, for each of MODES 1 through 5, the licensee should provide details of the pressurizer pressure Set points with allowable values consistent with the related channel errors and Safety Analysis Limits used in the Licensing Basis in the FSAR in Section 15 in reference 7, and reference 8. The licensee shall evaluate and propose.

Appropriate ACTION and SURVEILLANCE procedures should be proposed. The licensee shall evaluate and propose.

a) S.G. Levels

A number of the Accident Analyses in reference 7 depend upon an initial level of water in the Steam Generator. A specific example is the Main Feedwater Line Rupture Event of Section 15.4.2.2.2 in which AFW auto-start signal on SG low-low level occurs 20 secs are main feedline rupture occurs; reference related Table 15.4-1, Page 1 of 4].

Since this, and other events, depend upon a "programmed" water level in the steam generators for an acceptable outcome in terms of the issuance of the operating license, these water levels also represent limiting conditions of operation in respect of 10 CFR 50.46. Please provide details of such SG levels including related Safety Analysis Limits, and respond to the proposition that such values should be included as Set Point values and Allowable values in the proposed T.S. as Limiting Conditions of Operation for the facility with appropriate Action Statements. The proposed T.S. is nonconservative by their absence.

b) Steam Generator Pressures

Since Steam Generator Pressures and related Saturation Temperatures under normal steady state operation can be a significant determinant of system responses for Condition II through IV occurrences analyzed in the Licensing Basis including Section 15 of reference 7, and reference 8, please provide the values used as Safety Analysis Limits in related analyses and again respond to the proposition that such values should be included as Set Point and Allowable values as Limiting Conditions of Operation for the facility with appropriate Action Statements. The proposed T.S. is nonconservative with respect to the Licensing Basis, by their absence.

c) Please respond to the proposition that this section should also adequately identify the maximum allowable Steam Generator Pressure under Transient and Accident conditions with appropriate Action Statements. Maximum SG pressure is one of the Acceptance Criteria for safety. The current very limited basis for Steam Generator Pressure integrity is completely inadequate. Please clarify apparent discrepancy between reference 4, Table 5.5.2-1 in which the steam side design pressure for the Steam Generator is given as 1285 psig and the value quoted in the T.S. Basis Page B 3/4 7-1 at 1185 psig.

The proposed T.S. is nonconservative with respect to the Licensing Basis, by this absence.

d) APPLICABILITY MODES 1, 2, 3, and 4:

The current applicability requirements relate to Structural Integrity considerations.

On inclusion of Steam Generator Level and Pressure as determinants of Operability, the licensee should evaluate and propose APPLICABILITY MODES consistent with RCS/SG loop requirements discussed in this review under separate sections and particularly under Reactor Coolant System and Residual Heat Removal sections in MODES 1 through 5. This will embrace operability requirements from MODES 1, 2, 3 and 4 through 5. The proposed T.S. is nonconservative with respect to the Licensing Basis, by the absence of this information. The licensee shall evaluate and propose.

T.S. SECTION 3/4.5 EMERGENCY CORE COOLING SYSTEMS

The operability requirements from the McGuire Units 1 & 2 Licensing Basis FSAR are markedly different from those of the W Standard Technical Specifications which have been adopted by the Licensee in his proposed T.S.

The Licensing Basis FSAR requirements are summarized under "General."

General

FSAR Reference 8, page Q 212-47, Revision 25, item 212-75, describes the following Operator Instructions and Operator Actions During Shutdown.

"The sequences of events associated with shutdown will be described. The procedures associated with startup will be the same except they will be in reverse order. The startup procedures are not presented here to avoid unnecessary duplication.

I Operator Instructions During Shutdown

A) At 1900 psig, the operator is instructed to manually block the automatic safety injection signal. This action disarms the SI signals from the pressurizer pressure transmitters and from the steamline pressure transmitters. The SI signal on containment high pressure signal continues to be armed and will actuate safety injection if the setpoint is exceeded. Manual safety injection actuation is also available. Also, at 1900 psig, the operator is instructed to close and gag UHI discharge valves. The UHI hydraulic pump and the gag motors for the UHI isolation valves are de-energized and tagged.

B) At 1000 psig, the operator closes the cold leg accumulator isolation valves. He then racks out, locks and tags the breakers for these valves. He also opens locks and tags the breakers for all safety injection pumps and all but one charging pump. At this time, one charging pump and two residual heat removal (RHR) pumps would be available for either automatic or manual SI actuation.

C) At less than 400 psig and 350°F, the operator aligns the Residual Heat Removal System. The valves in the line from the RWST are closed.

II Operator Actions During Shutdown

A) Between 1900 psig and 1000 psig, the ECCS can either be actuated automatically by the high containment pressure signal or manually by the operator.

B) Between 1000 psig and 400 psig, a portion of the ECCS can be actuated automatically (containment high pressure signal) or manually by the operator. The equipment that can be energized are two RHR pumps and one charging pump. The operator would have to reinstitute power at the motor control centers or switchgear to the remaining safety injection pumps, charging pump, and the accumulator isolation valves.

C) Below 400 psig, the system is in the RHR cooling mode. The RHR system would have to be realigned as per plant startup procedure. The operator would place all safeguards systems' valves in the required positions for plant operation and place the safety injection, centrifugal charging, and residual heat removal pumps along with SI accumulator in ready and then manually actuate SI."

In response to additional questions, the following information was provided under FSAR reference 8, page Q 212-61, revision 28, item 212.90(6.3); page Q 212-61a, revision 28, pages Q 212-61b, revision 29 and Q 212-61c, revision 29

"In spite of the low probability of occurrence and the fact that certain failure modes for pipe rupture do not exist during cooldown at an RCS pressure of 1000 psig, the following items have been incorporated into the station operating procedures:

1. At 100[0] psig, the operator will maintain pressure and proceeed to cool down the RCS to 425°F.

2. At 1000 psig and 425°F, the operator will close and lock out the accumulator isolation valves.

The above plant operating procedures will ensure that the accumulator isolation valves will not be locked out prior to about 2-1/2 hours after reactor shutdown for a cooldown rate of 50°F/hr.

A conservative analysis has defined that the peak clad temperature resulting from a large break LOCA would be significantly less than the 2200°F Acceptance Criteria limit using the ECCS equipment available 2-1/2 hours after reactor shutdown.

The following assumptions were used in the analysis:

1. The RCS fluid is isothermal at a temperatu[r]e [o]f 425°F and a pressure of 1000 psig.

2. The core and metal sensible heat above 425°F has been removed.

3. The hot spot occurs at the core midplane.

4. The peak fuel heat generation during full power operation of 12.88 kW/ft (102% of 12.63 kW/ft) will be used to calculate adiabatic heatup.

5. At 2-1/2 hours decay heat in conformance with Appendix K of 10 CFR 50, the peak heat generation rate is 0.179 kW/ft.

6. Two low head safety injection pumps and one high head charging pump are available from either manual Safety Injection actuation or automatic actuation by the containment Hi-1 signal.

7. No liquid water is present in the reactor vessel at the end of blowdown.

8. A large cold leg break is considered.

For a postulated LOCA at the cooldown condition of 1000 psig, previous calculations show that the clad does not heat up above its initial temperature during blowdown. Proceeding from the end of blowdown and assuming adiabatic heatup of the fuel and clad at the hot spot, an increase of 446°F was calculated during the lower plenum refill transient of 89 seconds. During reflood, the core and downcomer water levels rise together until steam generation in the core becomes sufficient to inhibit the reflooding rate. At that time, heat transfer from the clad at the hot spot to the steam boiloff and entrained water will commence. This heat removal process will continue as the water level in the core rises while the downcomer is being filled with safety injection water. The reflood transient was evaluated by considering two bounding cases:

1. Downcomer and core levels rise at the same rate. No cooling due to steam boiloff is considered at the hot spot. Quenching of the hot spot occurs when the core water level reaches the core midplane.

2. Core reflooding is delayed until the SI pumps have completely filled the downcomer. No cooling due to steam boiloff is considered at the hot spot until the downcomer is filled. The full downcomer situation may then be compared with the results of the ECCS analysis in the SAR to obtain a bounding clad temperature rise thereafter.

For Case 1 described above, the water level reached the core midplane 43.2 seconds after bottom of core recovery. The temperature rise during reflood at the hot spot from adiabatic heatup is 216°F, which results in a peak clad temperature of approximately 1086°F.

For Case 2, the delay due to downcomer filling is 54.4 sec. The corresponding temperature rise at the hot spot form adiabatic heatup is 272°F, which gives a hot spot clad temperature of 1143°F.

The clad temperatures at the time when the downcomer has filled for the DECLG, C_D = 0.6 submitted to satisfy 10 CFR 50.46 requirements are 1620°F and 1774°F at the 6.0 and 9.0 foot elevations, respectively.

Core flooding in the shutdown case under consideration will be more rapid from this point on due to less steam generation at the lower core power level in effect; decay heat input at any given elevation is less in the shutdown case. The combination of more rapid reflooding and lower power in the fuel insures that the clad temperature rise during reflood will be less for the shutdown case than for the design basis case.

Repeating the above calculation assuming the loss of a low head safety injection pump yields clad temperature of 1653°F and 1760°F for Cases 1 and 2, respectively. These results provide additional assurance that the peak clad temperature will not exceed 2200°F because, as stated above, in the shutdown case more rapid reflooding and lower power in the fuel insures that the clad temperature rise during reflood will be less than for the design basis case.

Based upon the analysis as presented above, it can be concluded that in the unlikely event of a LOCA at shutdown conditions, the peak clad temperature will be less limiting than that of the design base calculation.

The response provided in Revision 28 [above] addressed the subject of operator actions and ECCS availability. Consistent with the information provided in Revision 28, a postulated LOCA in the RHR mode at 425 psig RCS pressure has been assessed. The initial conditions would be reached four hours after reactor shutdown. The integrity of the core after a postulated LOCA is assured if the top of the core remains covered by the resultant two-phase mixture. A conservative indication of time available for operator action is obtained by calculating the time required for the top of the core to just uncover. A calculation has been performed to confirm that margin for operator action does exist to prevent core uncovery. This conclusion persists even under an assumption of ten minute delay for operator reaction time.

Assumptions:

(a) The system pressure essentially reaches equilibrium with containment by the time the volume of water above the bottom of the hot legs is removed.

(b) Upper plenum fluid volume between the top of the core and bottom of hot legs is the only upper plenum fluid considered.

(c) Volume between the core barrel and baffle is conservatively neglected.

(d) 120% of the ANS decay heat curve for four hours after shutdown is utilized.

Using the void fractions developed from the Yeh correlations and utilizing a hydrostatic pressure balance, the height of the steam-water mixture in the upper plenum was generated. Incorporating the plant geometry, the total liquid mass in the downcomer, core, and upper plenum was calculated, i.e., a mass-initial condition. Again by hydrostatic pressure balance, the height of liquid in the downcomer when the top of the core is just about to uncover was calculated. This information along with core volume is used to develop a mass-final condition. That is, the mass is liquid contained just before the core is uncovered. Utilizing the boil-off rate for the four hour time after shutdown, the time needed to evaporate a mass of mass-initial minus mass-final is calculated. This time was compared to the ten minute assumption for operator reaction time.

Utilizing the preceding approach, the time calculated to just initiate an uncovery of the core is 13 minutes. The conclusion is that even for the conservative method outlined above, there exists adequate margin to retain a safe core condition even in relation to a ten minute operator-response-time assumption."

These operator requirements are verified, in general, by reference 12, SER Supplement 2 page 6.6-6.8 under "Emergency Core Cooling System - Performance Evaluation," and pages 7-1 and 7-2 under "Upper Head Injection Isolation Valves."

Additionally, the status of the ECCS systems from entry into the RHR MODE through cooldown, i.e., from 425 psig/350°F through MODE 5 is clarified by the following extract from reference 11, Suppl. SER No 1, pages 5-1 and 5-2 which confirms continuance of the alignment at the end of MODE 3 425 psig/350°F through both MODES 4 and 5.

T.S. SECTION 3/4 5.1 ACCUMULATORS/COLD LEG INJECTION

Item: APPLICABILITY MODE

The Applicability Mode, given as MODES 1, 2 and 3* where 3* is 1000 psig, should be amended to include 425°F; as 1000 psig/425°F. Reference the basis in the previous section entitled "General."

Since the proposed T.S. does not contain this temperature constraint, it is non-conservative. A pressure of 1000 psig on the current Appendix G curve, and T.S. temperature constraints, would permit an RCS temp of 557°F. The only available analysis in the Licensing Basis, see earlier under "General," shows that cooling down to [1000 psig]/425°F is necessary to reduce the thermal burden on the ECCS so that the reduced ECCS capability can mitigate the consequences of a LOCA to 10 CFR 50.46 requirements; reference 8, pages Q 212-61, revision 28 and Q 212-61a, revision 28. The current T.S. is therefore non-conservative in this matter, and the licensee must evaluate and propose. Note; the "Footnote* Pressurizer Pressure above 1000 psig" also needs amendment.

Item: 3.5.1.1.d.

Nitrogen cover pressure is quoted at between 400 and 454 psig. The Licensing Basis FSAR, reference 4, page 1 of 5 revision 39 in Table 6.3.2-1 specifies a normal operating pressure of 427 psig. Making an allowance for channel error and drift should not this value be a higher set point of approx. 450 psig. The specified set point values proposed in the T.S. of 400 to 454 psig can therefore give actual values which are lower than in the Licensing Basis FSAR and be non-conservative. The Licensee shall evaluate and propose.

Item 3.5.1.1.f Proposed

The NRC proposes that an additional item limiting the range of actual water temperature in the accumulator between 60-150°F in accordance with Licensing Basis FSAR reference 29, Table 6.3.2-1 is necessary to confirm Safety Analysis Limits for this accumulator. Its absence from the proposed T.S. renders it potentially non-conservative. Further Item 4.5.1.1.1.a. concerning verification parameters should include Temperature of Accumulator Water. The licensee shall evaluate and propose.

ACTION Items a and b require HOT SHUTDOWN generally, except for closed isolation valves. This may be too conservative - the licensee should review specific cases identified under 3.5.1.1.a-f and decide whether HOT SHUTDOWN is necessary instead of to 1000 psig/425°F. Further, is there any conservative direction of the error which may minimize his need to suspend operations at power, or allow him to operate at reduced levels. This licensee proposal may be unecessarily conservative. The licensee may evaluate and propose.

Item 4.5.1.1.c requires that "once per 31 days when the RCS pressure is above 2000 psig, it is verified that power to the isolation valve on the Cold Leg Injection Accumulator is disconnected. What is the safety basis for this action, and where is it discussed in the Licensing Basis FSAR.

Item 4.5.1.1.1.d.1 requires that

"At least once per 18 months verify that each accumulator isolation valve opens automatically under each of the following conditions:

1) When an actual or a simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint,"

We are not aware that this actually occurs; the licensee shall review and advise of the related details within the FSAR on other licensing basis records. This action is not described in FSAR reference 7, under Table 7.3.1-3 (1 of 2) and (2 of 2) revision 35, "Interlocks for ESFAS," nor in the related Logic Diagrams.

The LCOs of the Licensing Basis FSAR require that this Cold Leg Injection Accumulator be made operable whenever plant conditions exceed 1000 psig/425°F which is at a lower pressure than the current P-11 set point of 1955 psig; reference earlier T/S Section 3/4.5 under "General." This P-11 logic which would propose that this isolation valve is to be closed at RCS pressures between 1955 to 1000 psig is therefore non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

The licensee shall verify that the set points for the relief valve on the Accumulators are included in the Inservice Testing Program at the facility.

Item 3.5.1.2.d: Proposed.

It is proposed that an additional item limiting the range of actual water temperatures in the accumulator to between 70 and 100°F in accordance with reference 29, Page (1 of 5), revision 39, in Table 6.3.2.1 is necessary to confirm the Safety Analysis Limits for the UHI Accumulator. It is also proposed that it be added as an additional surveillance element to item 4.5.1.2.a. Its absence from the proposed T.S. renders it potentially non-conservative with pespect to the Licensing Basis. The licensee shall evaluate and propose.

Action Items a & b require HOT STANDBY, generally, except for closed isolation valves, followed by HOT SHUTDOWN. This may be too conservative - the licensee should review specifically each of the Operability items b, c and proposed d, and decide whether HOT STANDBY leading ultimately to HOT SHUTDOWN is necessary. Further, he should assess if either boundary value, upper or lower, can be conservative, and by how much, and evaluate whether he should take an ACTION STATEMENT under "conservative" conditions. The licensee may evaluate and propose.

The licensee shall verify that the relief valve set point on the Accumulator is included in the In Service Testing Program at the facility.

T.S. Section 3/4.5.1.b (Proposed)

An additional T.S. item is proposed that provides specifically for the fact that "UPPER HEAD INJECTION SYSTEM ISOLATION VALVES" at APPLICABLE CONDITIONS of MODE 3 (< 1900 psig and > 425°F), MODE 4 and MODE 5, would have a "LIMITING CONDITION OF OPERATION" providing that "Each upper head injection system isolation valve" is closed and gagged. The UHI hydraulic pump and the gag motors for the UHI isolation values are de-energized and tagged. Appropriate Action Statements and Surveillance Procedures would be provided. This in accordance with the LCOs of the Licensing Basis FSAR as described in earlier items T.S. 3/4.5, "GENERAL" and T.S. 3/4.5.1 of this review.

Absence of this specific provision makes the current T.S. non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

Item 4.5.2.h.: concerning flow balance tests in the ECCS system. The licensee shall provide the bases for the flow distributions specified and further advise how they might meet minimum flow conditions to intact loops dating Accident Occurrences.

T.S. Section 3/4.5.3 ECCS Subsystem - Tavg ≤ 350°F

This T.S. does not disallow the additional CCP and 2 Safety Injection Pumps (SIPs) from 350°F down to 300°. This again is non-conservative with respect to the LCDs of the Licensing Basis FSAR which allows only one (1) CCP, and the remainder i.e., one (1) CCP and any other reciprocating charging pump and 2 SIPs are to be electrically isolated against inadvertent operation. This proposed T.S. is again non-conservative in respect of overpressure protection when compared with the current Licensing Basis. The licensee shall evaluate and propose.

The proposed T.S. allows one (1) CCP and one (1) SIP whenever the RCS temp is less than 300°F. The LCO of the Licensing Basis FSAR allows only one (1) CCP because of OVEPRESSURE PROTECTION; reference earlier information under earlier T.S. Section 3/4.5. Item: "General". The proposed T.S. is therefore non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

T/S Section 3/4.5.4 BORON INJECTION SYSTEM/BORON INJECTION TANK.

Item: APPLICABILTY MODES 1, 2, and 3 with the current proposed T.S. should be changed to include MODE 4 in accordance with the Licensing Basis FSAR which evaluates MSLB and LOCA events down to and including this MODE. Adoption of the Licensing Basis FSAR mode of boration control may eliminate this need. With proposed T.S., however, the absence of the BIT tank in Mode 4 must be considered non-conservative. The licensee should evaluate and propose.

The licensee shall clearly indicate, that this item is not applicable to Unit 2 by reason of a recent SER from NRC.

T.S. Page 3/4 7-4: AUXILIARY FEEDWATER SYSTEMS

Item 3.7.1.2.b. The licensee has deleted OPERABILITY requirements for the Steam-Turbine driven auxiliary feedwater pump at steam pressures of less than 900 psig. This is not in accord with current Accident Analyses and no justification has been provided. Reference 15, Recommendation GL-3, requires the Steam-Turbine AFW pump in the event of complete loss of AC power for a period of 2 hrs and beyond. This will require operability down to the lowest pressures for which the Turbine is provided as described in reference 22, Table 10.4.7-6 where the range of operating pressures provided for is from 110 psig to 1205 psig. This will also provide for operabilty down to and including MODES 4 (and availability from MODE 5) to cover licensing requirements discussed elsewhere under Table 3.3-3, ESFAS INSTRUMENTATION, Items 7a through f.

We note two principal features relating to the service conditions of the Turbine Driven Feedwater Pumps:

a. They are supplied with steam from two steam generators from main steam lines after the flow restriction orifices at outlets from the Steam Generators.

b. They would normally be expected to perform early in the transient and continue to function to design flow requirements throughout the Occurrence.

... explain how the proposed TS ensures that the Turbine Driven pump maintains its flow performance required by Accident Analysis when steam line pressures could drop substantially below the Steam Generator Pressures due to presence of the SG flow restrictions and until main steam isolation valves are isolated on steam line pressure of less than 565 psig (< provides for channel drift and errors).

The licensee shall evaluate the above comments and propose technical specifications which will ensure operability of the Turbine-Driven AFW Pump over the range of conditions expected from Design Basis Accident Analysis, and other less bounding events, down to and including MODE 4 as discussed in the Licensing Basis.

In his evaluation, the licensee should advise if Item 1e of Table 3.3-5 ESFAS INSTRUMENTATION, Steam Line-Pressure Low is derived from steam line sensors and after the SG orifices, or if it is taken from pressure sensors on the Steam Generator. The licensee should then advise what has been used in assessing Steam Generator Pressure Response and Turbine Driven AFW pump response in the

T.S. Page 3/4 7-8: MAIN STEAM ISOLATION VALVES

Item 3.7.1.4. The proposed T.S. provides that: "each main steam line isolation valve (MSLIV) shall be OPERABLE with APPLICABILITY MODES 1, 2, and 3.

The requirements within the Licensing Basis for Main Steam Line Isolation are discussed in this review under Table 3.3-4, Item 4. The Licesing Basis does require operability in MODE 4, in addition to MODES 1, 2, and 3 already provided.

We also note that the Main Steam Isolation Valves are Containment Isolation Valves as defined by 10 CFR 50 App. A Criterion 57 - "Closed System Isolation" and the Licensing Basis FSAR under reference 4 Table 6.2.4-1 (sheet 7 of 11) Revision 4 and that Primary Containment Integrity is required in MODES 1, 2, 3, and 4 according to proposed T.S. Section 3/4.6.1, T.S., Page 3/4 6-1.

The proposed T.S. is non-conservative with respect to the Licensing Basis; the Licensee shall evaluate and propose.

T.S. Page 3/4 7-8a Proposed: STEAM GENERATOR POWER OPERATED RELIEF VALVES (SG PORVs)

The proposed T.S. does not include these valves which are required to enable the plant to be cooled down under natural circulation conditions [under Loss of Offiste Power]. The Licensing Basis requirement for this is described in SER Supp No. 4 reference 14 page 5-7.

... required for natural circulation has not been established in the Licensing Basis. Reference 15, page 15.2-28, revision 15, under section 15.2.9.2 discusses natural circulation as verified by Table 15.2.9-1 which is at a maximum of 4%. This review, under earlier Table 2.2-1 Item 18b, shows how the existing Control Logic can place this plant into a natural circulation Occurrence, without reactor trip at a nominal power level of 10% Rated, and the review under Table 3.3-1 under Item: Concerning Prescribed Values for % Rated Thermal Power DURING START UP (MODE 1) AND POWER OPERATION (MODE 2) shows how the resulting residual nuclear power levels could actually be the order of 20%. Therefore, in addition to the evaluation required of the Licensee to meet those circumstances as described therein, he shall consider the consequences of the very limited SG PORVs capacity currently available to meet this situation. The Licensing Basis FSAR, reference 9, page 10.1-2, revision 8, para 3 shows a capacity of only 10% [without single failure]. This means that in addition to the potential inability of the RCS to provide the requisite cooling capacity under natural circulation for a nominal 10%, and potential 20%, power level, the SG PORV capacity is insufficient in the event of a single failure (of 4 available) for nominal conditions, and severely under capacity for a possible 20% power level. At this time, until further evaluation has been completed, the Licensee should ensure, within the T.S., a potential atmospheric relieving capacity of 20%, allowing for a single failure. This should include all his SG PORVs, plus elements of the additionally available 45% (of full load main steam flow to atmosphere) described under reference 22, page 10.1-2, revision 8, para 3, if they can be available under Loss of Offsite Power. An appropriate Action Statement should be provided. If the additional atmospheric relief is not available on LOOP, the Licensee must further evaluate and propose necessary corrective actions.

The current omission of SG PORVs from the T.S. is non-conservative with respect to the Licensing Basis. The current omission of relieving capacity additional to the SG PORVs is contrary to Regulatory Requirements which have been excluded from the Licensing Basis. The Licensee shall evaluate and propose.

T.S. Section 3/4.7.3: COMPONENT COOLING WATER SYSTEM

The proposed T.S. requires that:

3.7.3 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The SER for the plant under reference 10, summarizes the following Licensing Basis for the Component Cooling System:

The component cooling system provides cooling water to selected nuclear auxiliary components during normal plant operation and cooling water to safety-related systems during postulated accidents.

The component cooling system is designed to: (1) remove residual and sensible heat from the reactor coolant system via the residual heat removal system during shutdown; (2) cool the letdown flow to the chemical and volume control system during power operation; (3) cool the spent fuel pool water; and (4) provide cooling to dissipate waste heat from various primary station components during normal operation and postulated accident conditions. Active system components necessary for safe plant shutdown are designed to include at least 100 percent redundancy. The component cooling water for each unit includes two component cooling heat exchangers, four component cooling pumps and a split-volume component cooling surge tank. Two pumps and one heat exchanger per unit provide the necessary cooling water for normal operation, cooldown, refueling, and postulated accidents. The remaining pumps and heat exchangers serve as standby. An assured supply of makeup is provided from the nuclear service water system to each redundant loop.

The component cooling water system is designed to seismic Category I requirements, except for certain branches to non-essential equipment. The component cooling water pumps are powered by redundant emergency buses. The portion of the component cooling water system serving the residual heat removal system meets the single failure criterion for active components.

Based on our review, we conclude that the component cooling system design is in conformance with the requirements of General Design Criterion 44 of Appendix A to 10 CFR Part 50 regarding the capability of the system to transfer heat from systems and components important to safety to an ultimate heat sink and provisions of suitable redundancy for safe cool-down. We further conclude that the system design meets the requirements of General Design Criteria 45 and 46 of Appendix A to 10 CFR Part 50 regarding system design that allows performance of periodic inspections and testing. We conclude that the component cooling water system is acceptable.

Detailed reference to Operability and Operating requirements in the Licensing Basis in MODES 5 and 6 can be found in reference 22, page 92-17 and Component Cooling System.

The proposed T.S. completely ignores, without any evaluation, the Licensing Basis requirement for this system in MODES 5 & 6. The current T.S. are non-conservative with respect to the Licensing Basis. The Licensee shall evaluate and propose.

This T.S. is a prime example of a Standard Technical Specification which completely ignores the Licensing Basis for all Nuclear Power Plants. This reflects a very serious Safety Issue for all standard T.S. and which cannot await an extended "Generic" Resolution.

T.S. Section 3/4.7.4 NUCLEAR SERVICE WATER SYSTEM

APPLICABILITY MODES proposed are 1, 2, 3, 4. These should be extended to MODES 5 and 6.

Within the Licensing Basis FSAR, reference 6, [vol 8] page 9.2-5, "The Nuclear Service Waste System (NSWS) is designed to meet single failure criteria with two redundant channels [per unit] to serve components essential for safe station shutdown." The equipment requiring NSWS also includes all RPS and ESFS systems, many of which are necessary in MODES 5 and 6 to the above redundancy and single failure criteria.

Examples include: MODE 5 is required to service AFW alternate cooling requirements in event of a fail-closed RHR/RCS isolation valve in the RHR line, and in MODES 5 and 6 it is needed to service necessary redundant RHR Trains. Reference our related evaluations in this review concerning RHR operability requirements in MODES 5 and 6.

The proposed T.S. is nonconservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

T.S. Section 3/4.9 REFUELING OPERATIONS

T.S. Item 3/4 9.1 BORON CONCENTRATION

Additional LCOs are necessary to meet the requirements of reference 8, page 15.2 - 14, revision 10 concerning Accident Evaluation for Section 15.2.4, Uncontrolled Boron Dilution. The boron dilution analyses of this reference 7, provides that, during refueling:

a. "A minimum water volume in the Reactor Coolant System is considered. This corresponds to the volume necessary to fill the reactor vessel above the nozzles to ensure mixing via the residual heat removal loop."

b. Neutron sources are installed in the core and the source range detectors outside the reactor vessel are active and provide an audible count rate.

c. A high flow alarm at the discharge of the CVCS (from flow element INVFE 5630) is active providing an alarm to the operator when the flow rate from the charging pumps exceeds 175 gpm.

d. The charging pumps are inoperative.

Additionally, an appropriate condition which must be attached to a) above is that any such minimum volume should be such that the level of water in or above the loop provide acceptable flow, including NPSH conditions, at inlet to the RHR pumps.

These conditions are appropriate LCO's to 10 CFR 50.36; their current absence from the T.S. for this MODE is a non-conservative situation in respect of the Licensing Basis, and the Licensee shall evaluate and propose.

The current SER, Supplement No. 1, reference 11, 15-1, provides that:

> "During refueling the applicant has committed to isolate all sources of unborated water connected to the primary system refueling/canal/spent fuel.

We do note that Surveillance Requirement T.S. 4.9.1.3 does provide for verifying that valve No. INV-250 is closed, under administrative control in support of this. However we do note that according to reference 7, page 15.2-15, item Q 212-58, this valve INV-250 is to be locked closed during refueling. The current position could be non-conservative if the valve is not specifically locked under the proposed administrative control. Also notice, that reference 7, page 15.2 - 14, revision 10 states that:

> "The other two paths are through 2 inch lines, one of which leads to the volume control tank with the other bypassing this tank. These lines contain flow control valves INV171A and INV175A respectively."

Why are T.S.s not applied to the closure of these valves also. The proposed T.S. may be nonconservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

We also note an apparent non-conservative discrepancy between the basis for the specified reactivity condition of "a k_{eff} of 0.95 or less" without any specification of the position of movable control assemblies. We also note the need to add, according to reference 7, page 15.2-14, revision 10, that the boron concentration is to give a shutdown margin of at least 5 per cent delta k <u>with all the rod cluster control assemblies out</u>. The additional requirement underlined should be a part of the LCO for this T.S. item. Without this provision in the proposed T.S, it could be interpreted as non-conservative in respect of the Safety Analysis Limits for the plant. The licensee shall evaluate and propose.

<u>T.S. Item 3/4 9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION; HIGH WATER LEVEL</u>

The ACTION STATEMENT provides that with no RHR loop operable, the containment should be closed within 4 hours. Information in reference 8, page Q 212-56 under Case 2 shows that if RHR is absent [by isolation of the RCS/RHR inlet valve] that:

> "Approximately 2.5 hours are available to the operator to establish an alternate means of core cooling. This is the time it would take to heat 300,000 gallons of water in the refueling canal from 140°F to 212°F, assuming the maximum 24 hours decay heat load."

The current value of 4 hours appears less conservative than this calculated value of 2½ hours within the FSAR. The licensee shall evaluate and propose.

Review of available responses to the consequences of a fail closed RCR/RHR isolation valve, include many procedures using the containment sump. To allow for this single failure contingency, the licensee should therefore ensure that the containment sump will be operable during this mode, and with an appropriate surveillance procedure. There should also be provision for available fire pumps and necessary hoses to be assuredly available to enable use of the alternate procedures which have been described in reference 8, pages Q 212-56 and 57, revision 25. The current T.S. must be considered non-conservative. The licensee shall evaluate and propose

T/S Page 3/4 9-11 REFUELING OPERATIONS LOW WATER LEVEL

Additionally, the above information defines an LCO of a minimum volume of water for the related event in which the RCS is drained to just below the level flange. A further requirement (LCO) is that any such minimum volume should be such that the level of water in or above the loop provides acceptable flow, including NPSH conditions, over the range of temperatures expected at inlet to the RHR pumps. Absent those required conditions from the Limiting Conditions of Operation makes them non-conservative in respect of the Licensing Basis. The licensee shall evaluate and propose.

Footnote *: provides that,

> "*Prior to initial criticality the RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs."

This is an invalid request as all CORE ALTERATIONS are only permissible under TS 3/4 9.9 HIGH WATER LEVEL - REACTOR VESSEL. This is a non-conservative T.S proposal. The Licensee shall propose and evaluate.

The current ACTION STATEMENT calls for containment closure in 4 hours [i.e. 240 mins]. Earlier conservative calculations for this MODE show that loss of all RHR in this MODE can cause boiling in 5 minutes and core uncovery in 100 mins. Given the circumstances, containment enclosure should be effected immediately, commencing RHR low flow alarms. The licensee shall evaluate, and propose. The current T.S. appears nonconservative with respect to the Licensing Basis.

Addenda

T.S. SECTION 3/4.5 EMERGENCY CORE COOLING SYSTEMS

T.S. SECTION 3/4.4.4.1 RCS LOOPS AND COOLANT CIRCULATION/HOT SHUTDOWN MODE 4

More recent information, and a detailed check on certain elements of the proposed T.S. relevant to the above section, and the Licensing Basis FSAR, and particularly reference 5, Section 7.4.1.6 Emergency Core Cooling Systems and Section 7.4.1.5 Residual Heat Removal System, does not appear to provide acceptable surety that:

a) The Reactor Coolant Pressure Boundary (RCPB) valves on the RHR/RCS suction line are confirmed closed in MODES 1, 2, & 3.

b) That the RCPB valves in the RHR/RCS suction line are individually identified as opened in the RHR MODE.

Item 13:

Refer
item

Item 13: Concerning Steam Generator Level-Low, Low

Reference 18, page 3-13 Note 12 describes the Safety Analysis Limit for this item as the value in Table 2.2-1 of the W STS plus 10%. For conservatism, should the Safety Analysis Limit be the W STS value less 10%; is this necessarily conservative for all Licensing Basis occurrences.

Item 18b: Low Power Reactor Trips Block, P-7

Until the required re-evaluation is completed, the proposed T.S. must be considered non-conservative in respect to Regulatory Requirements. Additionally it can be interpreted as a Generic Issue.

b) The current description of this Functional Unit is incorrect. It is not "Lower Power Reactor Trips Block P-7." It is: "High Power Reactor Trips Block," by absence of Permissive P-7 and occurs when:

1) P-10 is less than the Trip Set Point and
2) P-13 is less then the Trip Set Point

c) This TS provides that when power level is less than Permissive P7 (with P10 (Nuclear) or P13 (Turbine) powers of less than 10%), reactor trip on Pressurizer Pressure-Low and Pressurizer Water Level-High are both blocked.

c(i) Concerning Block of Pressurizer Pressure Low - Reactor Trip:

The FSAR in reference 5, item 7.2.1.1.2.C.1 states that this trip is not required at low power levels.

Accidental Depressurization of the main steam system is from zero load. It is unclear from reference 5 Table 7.2.1-4 (5 of 5) if for this event, reactor trip on Pressurizer Low Pressure is expected to occur before Safety Injection (when it would not be available at zero power) or whether it is expected to occur from the pressurizer pressure low - (Safety Injection) signal if it initiates S.I., or from S.I. initiated by other initiators. The Licensee shall clarify, and hence its validity with respect to the absence of the signal caused by P7.

SECTION 3.4.1 REACTIVITY CONTROL SYSTEMS

Section 3/4.1.1 BORATION CONTROL /APPLICABLE MODES 1, 2*, 3 and 4.

T.S. Pages 3/4 1-1, 2, 2a: Reference 16; page Q 212-47e states "Operating instructions require that boron concentration be increased to at least the cold shutdown boron concentration before cooldown is initiated. This requirement insures a minimum of 1% delta k/k shutdown margin at an RCS temperature of 200°F." This is used as a means of protecting against NON-LOCA Accidents during startup and shutdown.

Since this proposal to increase boron concentration is a limiting condition for operation required for safe operation of the facility from and including MODE 3 down to and including MODE 5, please advise why this does not appear in the Technical Specifications in accordance with 10 CFR 50.36(c)(2).

SECTION 2.1 SAFETY LIMITS

enclosure 2
Category B items

2.1.1 REACTOR CORE

The proposed T.S. requires that: "The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for four and three loop operation, respectively.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded th. appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1."

EVALUATION

b) Concerning Figs 2.1-1 What is the licensing basis for this type of representation, i.e., RCS T_{avg} (°F) vs Fraction of Rated Thermal Power, and the values in this figure. Reference 7, Figure 15.1.1-1, revision 7 is the existing licensing basis; it provides different ordinates, T_{avg} vs ΔT and includes descriptions of related ac eptance criteria and limits which should also include boiling in the hot legs; it also provides direct links to the plant protection systems based on 2 out of 4 ΔT loop (individual) compared with ΔT loop set point (individual), in the reactor protection system. Any such representation should also provide the basis for the SET-POINT methodology for each unit including values of all the parameters necessary to calculate OVERTEMPERATURE ΔT and OVERPOWER ΔT SET POINTS of related Table 2.2-1, REACTOR TRIP SYSTEM INSTRUMENT TRIP SET POINTS; this will ensure a complete set of Licensing Basis data against which the proposed plant settings can be verified and amended as appropriate.

c) Representations of overpower protection (including reporting requirements) by neutron flux monitors on the Figure 2.1-1 are inappropriate. Neutron flux limits and related action statements are addressed under T.S. Section 3.4, [Nuclear] Power Distribution Limits.

The FSAR does describe a constrained set of thermal hydraulic parameters for the Reactor Coolant System under steady state normal operating conditions upon which "plant safety" under Condition II, III and IV Occurrences is established. These are generally described in reference 7, under Section 15.1.2, Table 15.1.2-2, and the programmed T_{avg} provided under reference 3, Figure 5.3.3-1; pressurizer pressure is provided under Table 5.1-1. (Related pressurizer level and steam generator levels will be discussed under T.S. Sections 3/4.4.3 and 3/4.4.5) Should not these values be included in the Technical Specifications (in appropriate set point methodology) to meet the requirements of 10 CFR 50.36.

For the thermal-hydraulic parameters represented in Section 2, the steady state set points would be represented by a single line showing programmed Tavg against programmed ΔT for the given pressurizer pressure with provision for a band of values to "allowable values". Appropriate action statements would be formulated providing a limited period of operation outside the range. Any changes proposed to such conditions need T.S. amendments as they are part of the Licensing Basis.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

EVALUATION

a) Is there not a need to forewarn the operator that as for 2.1.1, for normal steady state operation, the RCS pressurizer pressure shall not exceed the values defined in Section 3/4.2.5 and 3/4.4.3. Safety evaluations for all occurrences are predicated on those values and are invalidated if they are not sustained. If restoration cannot be achieved, there is a change from the existing Licensing Basis and an appropriate request for a T.S. change

b) Please clarify that the value of 2735 psig is an actual Safety Limit, being 110% of the Design Pressure of 2485 psig (reference 3, Table 5.2.2-2 and how is such a value determined by the operator when no set point, allowable values and channel errors are provided for or defined.

TABLE 2.2-1. REACTOR TRIP INSTRUMENTATION SET POINTS

These have been checked against reference 18, Westinghouse (W) RPS/ESFAS Set Point Methodology, Table 3-4 and NOTE FOR TABLE 3-4 on page 3-13, which is described as applicable to McGuire Unit 1, 50-369. At this date, the assumption has been made that this information also applies to McGuire Unit 2, Docket No. 50-370. Please docket this fact or otherwise provide the alternate information.

The writer finds the general approach to representing Trip Setpoints as ≥ or ≤ a certain value is less than satisfactory; it is open-ended allowing overly conservative setpoints with unnecessary reactor trips. It appears that the Set-Point methodology may already have provided for expected errors in setting SETPOINTS so that this open-ended uncertainty is eliminated to a satisfactory "manageable" quantity. The Licensee should clarify.

Item 14: When two or more RCP circuit breakers open, above Permissive 7 (10% power), Reactor Trip deriving from undervoltage of the Reactor Coolant Pumps is also initiated, reference 7 Section 15.2.5.1 and reference 5, figure 7.2.1-1

ciii) Concerning Block of Pressurizer Water Level-High Trip

This pressurizer water level-high trip is a principal element of the Overpressure Protection System for W PWRs as fully discussed in Topical Report to reference 27.

Amongst Licensing Basis events, this trip is used as primary or back up on Uncontrolled Rod Cluster Control Assembly at Power. Uncontrolled withdrawal from a subcritical condition (at below P10) is protected primarily by other trips.

Among Licensing Basis events this trip is also used on Loss of External electric load and/or Turbine Trip. Most severe design basis consequences are from full power. Such an event at less than the 10% Set Point [P-10 & P13] is within the normal control range of the reactor (without steam dump) with the expectancy of no values exceeding normal control band [and thereby not approaching T.S. limits]

The blockage of these trips is consistent with the Design Basis Events and expected behavior of the Control System. However this does not address the fact 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. It appears singularly inappropriate to remove these protection devices which could play a primary or backup role in such circumstances. For example, reference 5, page 72-27 item 7.2.2.3.4, "Pressurizer Water Level," describes the role of the Pressure Water Level trip in preventing liquid Coolant discharge through the safety valves during a failure of the Pressurizer Water Level (PWL) controller at full power. Failure of PWL controller could fill the pressurizer within ½ hour or longer, but T.S. Table 4.3-1 shows a channel check on only a shift basis. Further, a single channel failure to low could cause overfill of the pressurizer (through the level control system) and with subsequent permissable failure of a second channel could remove the alarm expected from 2 out of 3 so that no alert is given the operator which would be contrary to the requirement of the FSAR.

There is no discussion on the importance of its use at low powers although the general System Description provided under Section 7.2.1.1 and its protective actions is no less appropriate at 0-10% power, as it is at higher power levels.

It is proposed, reference 5 page 7.2-8 that Pressurizer Water Level-High Trip below P-7 is automatically blocked to permit start up. Whereas this is understandable in MODES 6, 5 and part of 4, it is not a valid proposition once a bubble is formed in the pressurizer in MODE 4 and the Pressurizer Level Control can be placed in AUTO. Considering the attention required of all other manual actions during MODES 4 through 2, it is not appropriate to remove the automatic protection of the RCS boundary. Further, in MODES 4 and 3 it could be one of the only effective trips available because of the potential non-viability of Pressurizer Pressure High and non-applicability of existing Pressurizer Pressure-Low.

The Licenee should evaluate the impact on safety by blocking the Pressure Water Level-High trip below P-7, including all the concerns discussed above. This item can be interpreted as a generic issue. This could be considered non-conservative in respect to Regulatory Requirements because of the absence of automatic protection in accordance with 10 CFR 50, GDC 20 "Protection System Functions," both for reactivity control systems, and overpressure protection systems.

c(iii) The absence of permissive P-7 [on P-10 and P-13] introduces new events to evaluate for safety. This requires related Safety Analyses Limits and the Licensee shall advise what these are for each of P-10 and P-13 and how these are combined for P-7.

Item: Loss of "POWER"

Their is a need to prescribe the conditions under which a reactor would trip directly from a "Loss of Power" condition other than those deriving from other Functional Units. This is a substantial omission from the Technical Specifications.

SECTION 3.4.1 REACTIVITY CONTROL SYSTEMS

Section 3/4.1.1 BORATION CONTROL /APPLICABLE MODES 1, 2*, 3 and 4.

Reference 11, page 15-2, first para. precludes any boron dilution after a reactor scram until the neutron flux level is below the level of the source range high flux level alarm. This is effectively an LCO that is not included in the proposed T.S.

The proposed T.S is non-conservative with respect to the Licensing Bases.

The Licensee shall evaluate our concerns under this Section 3/4.1.1 and propose.

T.S. Page 3/4 1-9 concerning: CHARGING PUMP-SHUTDOWN

Consistent with the work of the previous TS Section 3/4 1-7 of this report, this title should be changed to: CHARGING PUMP - "Standbye (at 1000 psig/425°F) through to MODE 5. Additionally, under subsection 3.1.2.3 modify to only <u>one centrifugal</u> charging pump shall be OPERABLE. APPLICABILITY is changed from MODES 5 and 6 to MODE 3 (at < 1000 psig/425°F), 4 and 5. MODE 6 is deleted.

T.S. Page 3/4 1-11 Concerning: BORATED WATER SOURCE - SHUTDOWN

Further, an additional surveillance should verify the availability of Level Detection (2 indicators/tank) and related high, low and low-low level alarms.

T.S. Page 3/4 1-12 concerning: BORATED WATER SOURCES - OPERATING (in related Applicable MODES 1, 2, 3 and 4)

Additional surveillance requirements should be included under 4.1.2.6.a.4) in which the borated water source would be demonstrated OPERABLE by verifying minimum levels in the system.

Further, an additional surveillance should verify the availability of Level Detection (2 indicators/tank) and related high, low and low-low level alarms.

T.S. Page 3/4 1-21 Concerning: CONTROL ROD INSERTION LIMITS

b) Overpower (ΔT) and overtemperature (ΔT) protection systems incorporate automatic limits (Rod stops) on control rod insertion to maintain Safety Analysis Limits on "Power Distribution" in the Reactor Core during power runback. Please advise why there are no surveillance limits and requirements for these Rod stops in your Technical Specifications to meet the requirements of 10 CFR 50.36. Without these, the proposed T.S. must be considered non-conservative.

Section 3/4.2 POWER DISTRIBUTION LIMITS

Section 3/4.2.5 DNB PARAMETERS AND TABLE 3.2-1 DNB PARAMETERS

*) As discussed in Section 2.1.1, Subsection f, additional parameters necessary to the validity of Accident Analyses in Section 15 include Pressurizer Level (See our review under Section 3.4.4.3, T.S. Page 3/4 4-5) and Steam Generator Levels under Section 3/4.4.5 T.S. Page 3/4 4-11).

TABLE 3.3-1 REACTOR TRIP SYSTEM INSTRUMENTATION

T.S. Page 3/4 3-2.

See our comments on Table 2.2-1, Item 17 on a proposed revised description for this term to "Reactor Trip from ESFAS.

> Item 17: The existing descriptor "Safety Injection Input from ESF" should be replaced by "Reactor Trip from ESFAS."

The Licensee shall evaluate the safety consequences of the fact that in the event of a Main Stream Line Break below the P-11 interlock, Reactor Trip will not be initiated by the Negative Steam Line Pressure Rate - High signal. If the break is outside containment is there is no other parameter remaining which will cause the reactor trip; if the break is inside containment will Containment Pressure-High initiate reactor trip within an acceptable time. What are the consequences of a small to intermediate size break inside containment where, such Containment Pressure - High may not occur. We appreciate that Source Range and Intermediate Range Nuclear Flux trips could trip the reactor under these circumstances, on any return to power, but their current proposed status as not being necessary for protection because they are not required in the Safety Analyses would leave only the Power Range Low Setpoint Trip, and related resulting power levels of 35% as a Safety Analysis Limit would be unacceptable without a substantive analysis of the event. Please comment in terms of Reactor Trip System Instrumentation Requirements to meet these circumstances. The proposed T.S is non-conservative in respect of Regulatory Requirements in meeting these circumstances; the Licensee shall evaluate and propose.

In actual fact, the operability positions defined in Table 3.3-1 reflect an interface between MODE 1 and MODE 2 determined by Permissive P-7 at a nominal 10% Rated Power Level. Further, in this review, under Section entitled TABLE 2.2-1, REACTOR TRIP SYSTEM INSTRUMENTATION SET POINTS, item 18 c(iii) we have identified the need for Safety Analyses Limits for P-10, P-13 and in combination for P-7, so that the outer Limits of Power level of this safety control logic can be identified for safety evaluation purposes. For example, the Safety Analyses Limit used in the FSAR for the Power Range, Neutron Flux - Low Set Point is + 10% on the Set Point of 25% to give 35% as the conservative outer limit. If this same (total channel error) margin was applicable to both the P-10 and P-13 channels to give a P-7 Safety Analysis Limit of 10% + 10%, i.e, 20% RATED THERMAL POWER, then the importance to related safety-related issues is substantively increased.*

TABLE 3.3-2 REACTOR TRIP INSTRUMENTATION RESPONSE TIMES

Item 21, Proposed (Reactor Coolant Pump Breaker Position Trip)

As discussed earlier, under table 2.21, Item 14, this trip is provided as an adjunct to Undervoltage - Reactor Coolant Pump Trip. The Licensee shall evaluate and propose.

TABLE 3.3-3 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) INSTRUMENTATION

Item 3c: Purge and Exhaust Isolation

An additional Item: 3c.4 Containment Radioactivity, is proposed to effect Purge and Exhaust Isolation as this is part of ESFAS Logic in reference 8, figure 7.2.1-1 (8 of 16), revision 34. The Licensing Basis for this requirement lies inside the analysis of consequences deriving from accidental events whilst the Purge and Exhaust Isolation Valves are open. [Refce CSB]

The proposed T.S. is non-conservative with respect to the Licensing Basis; the Licensee shall evaluate and propose.

Item 7; Auxiliary Feedwater (AFW):

General: Operability Requirements:

Requirements for ESFAS operability in AFW are generally limited to MODES 1, 2 and 3. However, provision is made in the FSAR for operation in MODE 4, and to be available in MODE 5.

For MODE 5, Reference 8 page Q 212-56 rev. 25 where RCS cooling is required to be available in the event of failure of one of the isolation valves in the line leading from the RCS hot leg to the suction of the RHR, causing flow blockage. Available Operability during MODE 5 is necessitated to facilitate conversion to effectively MODE 4 operation, as described in reference 8, page Q 212-56, rev. 25, since "only a few minutes" is proposed as necessary "to open the steam dumps and to start up the auxiliary feedwater system." It is proposed by NRC, that such a rapid startup of the AFW system can only be achieved by having available the Automatic Actuation Logic and Actuation Relays, and all related ESF equipment so that the automatic logic can be initiated manually. The licensee shall evaluate and propose. The proposed T.S. items 7a through 7g are generally non-conservative with respect to the Licensing Basis in this matter. The Licensee shall evaluate and propose on each of these items including consideration of our related reviews.

consequences of appropriate condition II, III and IV occurrences including Steam Line and Feedwater Line Breaks, which are analyzed assuming automatic initiation. Reference also proposed T.S. pages 3/4 4-3 for requirements for operable RCS systems in MODE 4. The proposed T.S. items 7a through 7g are generally non-conservative with respect to the Licensing Basis in this matter. The Licensee shall evaluate and propose on each of these items, including consideration of our related review.

Item 7.a: AFW/manual initiation

Item b: AFW/Auto Actuation Logic and Actuation Relays:

Operability is currently not required in MODES 4 and 5. Operability should be provided for both modes to meet the licensing requirements, i.e., manual initiation of Automatic Actuation Logic and Actuation Relays: reference General above.

Item 7.c.1: Start Motor Driven Pumps:

Should be operable in both MODES 4 and 5 and especially to counter non-availability of Turbine Driven Pumps early into MODE 4 during the cooldown.

Item 7.c.2): Start Turbine Driven Pumps:

Should be operable in 4. Although not capable of operating at lower temperatures of MODE 4, and MODE 5, it should nevertheless be available for use to counter consequences described in "General" above, including a station blackout.

Item 7.d): Auxiliary Feedwater Suction Pressure Low:

This proposed T.S description of a functional unit is invalid. The Functional Unit to be provided is:

d) Automatic Re-alignment of Suction Supply [This is the functional unit], on

Low Auxiliary Feedwater Suction Pressure [This is the parameter causing the change]

Operability requirements should identify how many AFW pumps are required to be "tripped" deficient in suction, to effect re-alignment.

The licensee should identify those instrument/control channels, and particular engineering alignments, which result in a re-alignment of redundant AFW supplies to the only safety-related supply available, from the Nuclear Service Water Pond, and define related operability and surveillance requirements. The mixed nonsafety and safety-related supplies on the McGuire units make it necessary to separately define and T.S. those safety-related elements, under 10 CFR 50.46: see reference 14, page 10-2.

Applicable Modes in the current T.S. is limited to 1, 2 and 3. The licensee shall evaluate why this should not be extended to MODES 4 and 5 to meet the FSAR requirements described in "General" above.

Item 7.e: Start Motor-Driven Pumps (by Safety Injection)

Applicable Modes have not been identified. NRC proposes MODES 1, 2, 3 and 4 and 5 to meet the requirements of Item 7: General, discussed earlier.

Item 7.f: Station Blackout - Start Motor Driven and Turbine Driven Pumps:

Provision for operability is only in applicable MODES 1, 2 and 3. Consistent with previous considerations, operability should be required in MODE 4, with provision for immediate operability from MODE 5.

Item 8: Automatic Switchover to Recirculation on RWST Level:

This is limited in Applicability to MODES 1, 2, 3 by the proposed T.S.

Since a LOCA in MODE 4 is part of the Licensing Basis, see later Section 3/4.5 ECCS under GENERAL, the licensee should evaluate the reasons for, and the consequences of, not proposing this OPERABLE IN MODE 4, and not being available in MODE 5, to counter the consequences of potential LOCAs and loss of RHR cooling in these MODES. The proposed T.S. is non-conservative with respect to the Licensing Basis; the Licensee shall evaluate and propose.

Item 11 proposed:

There is a need to add a new Functional Unit not addressed in the current T.S., but which is a part of ESFAS.

This is:

"Close All Feedwater Isolation Valves" and "Close the Feedwater Main and Bypass Modulating Valves"

See reference 5, Figure 7.2.1-1 (13 of 16) revision 34 for the related unique control logic.

This Function is initiated by:

11a. Reactor Trip P-4, and Low Tavg.
11b. Reactor Trip P-4, and Steam Generator Level - High High P-14.
11c. Steam Generator Level - High High P-14 (see 5 above)
11d. Safety Injection (See 5 above).

Operability for 11a would be in accordance with 10c (above) and later evaluation under Table 3.3-4 Item 11a (Proposed). Operability for 11b would be in accordance with the evaluations in 10c and d above.

Operability for 11c and 11d would be by reference to items 5, 5abc.

TABLE 3.3-3: TABLE NOTATION

The uncertainty of the notation under ## is discussed in Item 1e earlier. Please amend as required in accordance with the related resolution.

TABLE 3.3-4: ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) INSTRUMENTATION TRIP SET POINTS

Item 3c.4 (Proposed):

Reference 5, Figure 7.2.1-1 (8 of 16) revision 34 shows that "Containment Radioactivity" initiates containment ventilation (Purge and Exhaust) isolation. Please explain why it is not included as, e.g., a proposed Item 4). The proposed T.S. is non-conservative with respect to the Licensing Basis. The Licensee shall evaluate and propose.

TABLE 3.3-5 ENGINEERED SAFETY FEATURES RESPONSE TIMES

Item 6a: Turbine Trip on Steam Generator Water Level-High High

The proposed T.S. is NA, i.e., not applicable.

Reference the licensee to our comments under Table 3.3-2, Item 16 where it is shown that it is used within the Licensing Basis.

The proposed position is non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose in accordance with our review under Table 3.3-2, Item 16.

Section 3/4.4 REACTOR COOLANT SYSTEM

Section 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

Item: GENERAL

G.1 INTRODUCTION

Concerning RCS Operability requirements, in MODE 3-5:

We refer to our earlier discussions & licensee requirements - and especially under Section 3/4.1.1, T.S. Page 3/4 1-1, 2 & 2a on Boration Control, T.S., Page 3/4 1-20 & 1-21 concerning SHUTDOWN AND CONTROL ROD INSERTION LIMITS and TABLE 3.3-1 REACTOR TRIP SYSTEM INSTRUMENTATION - generally, including more particularly items 2-21 (selected) and items 12, 14, 15 and 21.

Under our item T.S. TABLE 3.3-1, items 2, 5 & 6 et al, the licensee has been required to "Provide an anlaysis and evaluation of the consequences of Applicable Condition II, III and IV Occurrences, in MODES 3 through 5, for an appropriate set of Technical Specification requirements to ensure Conformance to Acceptable Regulatory Criteria, and from this establish an appropriate range of Reactor Trip System Instrumentation to Safety Related Requirements. This evaluation shall be undertaken in conjunction with our concerns for current technical specifications under section 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION of this review.

As part of this review, and as a safety justification for our concerns, we require inclusion of the following Occurrences and Considerations in the program, and as early determinants of our proposals in respect of RCS Loop Operability requirements in MODES 3, 4 and 5 (with loops filled).

G.2 DISCUSSION

Item: CONSIDERATION

A number of factors determine our concern:

G.2.1 The increased boron concentration discussed under Section 3/4.1.1 of this review.

G.2.1.1 Increases shut down margin at temperatures above 200°F, and thereby reduces the severity of any occurrences giving a return to power, but only after reactor trip. Further the T.S. proposed by the licensee does not include the increased boron concentration and RCS Operability requirements are judged against those circumstances.

G.2.1.2 Because increased shutdown margins are available, in MODES 3, 4 and 5, the licensee may now increase the level of withdrawal of all movable control assemblies and still remain within the unchanged T.S. condition of the allowable reactivity condition, keff of $\leq$ 0.99. Consequently, it does not benefit those Occurrences initiated by fast positive reactivity excursions in which maximum power levels ultimately reached are substantively determined by given Response Times to Trip. Further, events giving a return to power after reactor trip do not have improved initial protection; the reactor must still be tripped prior to effecting the increased shut down margin, and the elimination of virtually all "Safety Related" levels of neutron flux trip protection in TABLE 3.3-1 removes all current confidence in "available" Reactor Trips on Neutron Power; the only Safety Related Neutron Flux Trip from zero power subcritical conditions is the Power Range Neutron Flux Low Set Point and the proposed T.S. removes this from operability in MODES 3, 4 and 5. Further it has a Safety Analysis Limit of 35% power (25% Set Point) and together with related high peaking flux factors under these conditions is sufficient to require all 4 RCPs running to ensure R.C.S. Safety in at least MODE 3.

G.2.1.3 The increased boron concentrations give less negative and more positive moderate coefficients which changes the complexion and nature of expected responses from "Licensing Bases Events." Under these circumstances, it may not be possible to validly deduce the resulting responses and consequences without related analyses.

G.2.1.4 At this time we see no protection against positive temperature coefficients in MODE 3 [4, 5 & 6]. Proposed T.S. page 3/4 1-4 concerning MODERATOR TEMPERATURE COEFFICIENT requires only that:

> "the moderate temperature coefficient (MTC) shall be: 3.1.1.3.b. Less negative than - 4.1 delta k/k °F for all the rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition." The T.S. proposes that this is "Applicable to MODES 1, 2 and 3" only. The licensee should also clarify this T.S. requirement which is apparently in error and applicable to MODES 1 & 2 only because of the "RATED THERMAL POWER Condition."

G.2.2 Removal of operability requirements for all safety related reactor trips (except SI) in Modes 3, 4 and 5, has placed the reactor in nonconformance with the requirements of 10 CFR Appendix A GDC 20, "Protection System Functions" and GDC 22, "Protection System Independence For All Occurrences Not Inititating Safety Injection."

Further, only a limited number of automatic trips (6) are blocked by existing plant permissive. P-7, 2 are blocked by P-8. This leaves an additional 9 from which automatic protection can potentially be provided and which have been removed by unique action of the T.S. without any Safety Evaluation.

The proposed T.S. are nonconservative with respect to Regulatory Requirements. They are also nonconservative in respect to the Licensing Basis. The Licensee shall evaluate and propose.

G.2.3 In MODE 3, down to P-11, for events initiating Safety Injection, the engineering within the existing Licensing Basis, might allow 10 CFR 50 Appendix A GDC 20 and 22 to be satisfied in respect to reactor trip and diversity. However, the proposed T.S. does not propose operability of Reactor Trip from SI in this mode and offers no Safety Evaluation for the proposed change. Reference our review under Table 3.3-1, Item 17.

The proposed T.S. is not in conformance with the Licensing Basis, and is nonconservative. The licensee shall evaluate and propose.

G.2.4 In MODE 3, from P-11, to MODE 5, for events initiating SI, the plant is engineered and can be operated so that only one automatic trip of the reactor may be available; that from containment pressure-high.

On the above bases, plant engineering and operations would not be in conformity with regulatory requirements. The Licensee shall evaluate and propose.

It may be possible for the plant to be operated in a manner to conform by not manually blocking the Main Steam Line Pressure-Low Trip [at P-11] but constraining this blockage to a point at which SG pressure during cooldown is within an acceptable error band of the related Set Point Value. Under these circumstances, two (2) diverse automatic protections on reactor trip may be available.

In addition the proposed T.S.s do not require operability of the Reactor Trip/ESF channel in this phase of operations below MODE 3 [at P-11], to MODE 4 even though this is engineered into the Facility. No Safety Evaluation of this omission is provided. The FSAR assumes Safety Injection Protection in MODES 3 and 4. The proposed T.S. is not in accord with the Licensing Basis and is nonconservative. The Licensee shall evaluate and propose.

G.2.5 Diversity of Safety Injection to the maximum extent for related Accident Circumstances can only be retained within existing plant engineering by requiring that manual block of the Steam Line Pressure-Low be delayed until SG pressures are within an appropriate error band of the Steam Line Pressure-Low Set Point. This could be down to a temperature of approximately 485-490°F in the RCS which would be in MODE 3 before 1000 psig/425°F. (485-490°F is the saturation temperature equivalent to 565 psig + 30 psig [channel error] i.e., approximately 595 psig in the SG.

The licensee shall evaluate and propose.

G.2.6 <u>EVENTS OF CONCERN (A LIMITED SELECTION)</u>

G.2.6.1 OCCURRENCES WITH RAPID REACTIVITY INCREASE

Concerning "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from Sub-Critical Condition."

Current Docketed Analysis in reference 7, section 15.2.1, page 15.2-2 is based on four operating loops. This event is possible down to and including Mode 5. Current FSAR analysis trips the reactor on Power Range, Neutron Flux-Low Set

Point (25%) at a Safety Analysis Limit of 35% (reference page 15.2-3, item 3). The principal determinant of ultimate power level is Doppler coefficient; contribution of moderator reactivity coefficient is negligible (reference page 15.2-3, items 1 & 2). The event is initiated from hot zero power (reference 7, page 15.2-4 item 3). 4 RCS pumps are operating.

Given the circumstances of the proposed T.S., any T.S. allowing OPERABILITY of less than 4 RCS Loop in MODE 3 would be in nonconformance with the current FSAR in a nonconservative manner, and the licensee would be required to evaluate and propose.

Furthermore; increased boron concentrations would not change this requirement.

Additional events of a similar nature, with a rapid increase in reactivity include:

a) Uncontrolled Boron Dilution (reference 7, pages 15.2-13)

b) Startup of an Inactive Reactor Coolant Loop (reference 7, page 15.2-19, revision 7)

c) Excessive Heat Removal Due to Feedwater System Malfunction (reference 7, page 15.2-30, revision 7) concerning initiation with the reactor at zero power). Until the licensee clarifies availability of MFW during MODES 3 through 5, this must be considered a potential occurrence.

d) Single rod cluster control assembly withdrawal (reference 7, Page 15.3-9, revision 7). Although the Licensing Basis is at 100% power, the circumstances from zero power should be reviewed.

e) Rupture of a Control Rod Drive Mechanism Housing, at Zero Power (reference 7, Page 15.4-30; revision 42).

f) Major Rupture of a Main Steam Line (see below).

G.2.6.2 STEAM LINE BREAKS: OCCURRENCES

Concerning "Major Rupture of a Main Steamline"

This event is discussed in Accident Analyses in Reference 7, section 15.4.2 and Reference 8 item 212.75 page Q 212-47c & e, item 25. Reference 8 proposes that the resulting impact on shutdown margins from this event during MODES 3, 4 and 5 are improved over that of the design basis (of zero power, just critical, Tavg = 557°) as:

> "Operating Instructions require that the boron concentration be increased to at least the cold shutdown boron concentration before cooldown is initiated. This requirement insures a minimum of 1% $\Delta k/k$ shutdown margin at a Reactor Coolant System temperature of 200°F. This condition assures that the minimum shutdown margin experienced during the streamline rupture from zero power shown in the safety analysis is less than the case where safety injection

actuation is manually blocked on low steamline pressure and low pressurizer pressure."

This position gives no measure of the resulting shutdown margins and/or power level and, the consequences of a stuck rod, with only 2 RC loops operating instead of four. It is conceivable that two loop operation may be less conservative than either 4 RCPs continuing to operate or 4 RCPs tripped on Safety Injection, due to an increased cooldown in the core due to circulation (compared to the tripped case) but a much decreased core flow rate to handle the event. The potential short term consequences of bulk voiding and loss of circulation in the non-operable loops cannot be ignored.

If during cooldown, an MSLB cools the RCS down to 212°F e.g., the residual shutdown will be at 1% delta k/k whereas the proposed T.S. margin at Zero Power according to T.S. Page 3/4 1-1 was 1.6 delta k/k. Please clarify, and at what condition during cooldown the 1.6% delta k/k is reached.

Given the circumstances that the "Operating Instructions" described above are not a part of the proposed T.S., any T.S. allowing operability of less than 4 RCS Loops in MODE 3 would be in non-conformance with the current Licensing Basis Safety Analysis in the FSAR in a non-conservative manner, and the licensee would be required to evaluate and propose.

For this licensing basis event, from Zero Power, Reactor Trip does not occur on Power Flux Trip, but on Pressurizer Pressure-Low (SI) (above P-11) [reference our required confirmation of this in an earlier item] so the Power Flux Trip is not required to be Operable.

At less than P-11, these circumstances are changed for the MSLB, and Reactor Trip does not occur until Containment-Hi is achieved, for a break inside containment.

For a break outside containment, however, high negative steam rate isolates main steam isolation valves only, but their is no Safety Injection, no Reactor Trip (on SI), and under the exisiting proposed T.S. no safety related Reactor Trip System Instrumentation of any nature to Trip the Reactor and Insert the movable control rods to benefit from potentially increased available shutdown margin. In addition to all this, the licensee proposes that MSIV closure times under these conditions in Not Applicable.

Given the circumstances of the proposed T.S., and T.S. allowing OPERABILITY of less than 4 RCS Loop in MODE 3 under these circumstances would be in nonconformance with the current Licensing Basis FSAR in a nonconservative manner, and the licensee would be required to evaluate and propose.

Additional events which exhibit a rapid cooldown and derressurization of the RCS; are:

a) Accidental Depressurization of the main steam system at no load, (reference 7, page 15.2-35, revision 36).

b) Minor Secondary System Pipe Breaks [at no load]; reference 7, page 15.3-4, revision 27).

G.2.6.3 LOSS OF PRIMARY COOLANT: OCCURRENCES

Concerning: "Small Break LOCA"

This is discussed in reference 7, section 15.3.1 for a SBLOCA from rated power, and reference 8, item 212.75 page Q 212-47b for a SBLOCA between RCS conditions of 1900 psig and 1000 psig/425°F in Hot Standby, and Q 212-64, item 3 together with SER Supp. No.2, reference 12, page 6-8 for the remaining situations. See also in general, reference 12 pages 6-6 to 6-8 in respect of ECCS System Performance Evaluation from Hot Standbye to and including RHR.

The FSAR analysis for SBLOCA in reference 7, Section 15.3.1 states that:

> "During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following trip: therefore upward flow through the core is maintained."

Topical Report, WCAP 8356 (reference 19) is the basis (reference 8, page Q 212-47b last paragraph) for the SBLOCA calculations to the same reference 8. These were undertaken with all pumps initially running followed by either a) all pumps tripped or b) continuing to run. The general conclusion from this report, reference 27, page 4-31, is that:

> "Due to the action of the running (non-tripped) pumps, less negative core flow occurs from the flow reversal compared to the case [ ] where pumps are immediaely tripped." and "The net result of these effects is a smaller peak clad temperature for the pumps running case compared to the pumps tripped case. Hence, for ECCS analysis for W 4 loop plants the reactor coolant pumps are assumed to be tripped at the initialization of a postulated LOCA and a locked rotor pump resistance is used for reflood."

At this time therefore, the NRC must conclude that RCS pump operation and coast down is important to reducing the loss of core level subsequent to the event; also in maintaining unseparated two phase flow conditions and in ensuing rapid Boron (mixing and) Injection to the core. Rapid boron injection would not be an important issue if boron concentrations are already at cold shut down values, but minimizing loss of core level is important.

Until further evaluations are made, we must conclude that the current Safety Analysis Limits of the SBLOCA event is 4 RCS pumps OPERABLE in MODE 3 down to 425 psig/350°F. The current proposed T.S. are therefore non-conservative and the licensee must evaluate and propose.

Given the circumstances of the proposed T.S., operability of less than 4 RCS Loops in MODE 3 would be in non-conformance with the Current Safety Analyses Limits in a non-conservative manner and the licensee is required to evaluate and propose.

Additional events of a similar nature to the SBLOCA events include:

a) Accidental Depressurization of the Reactor Coolant System (reference 7, page 15.2-33, revision 7).

b) Steam Generator Tube Rupture (reference, page 15.4 - 13a, revision 38).

c) Rupture of a Control Rod Drive Mechanism Housing at Zero Power (reference 7, page 15.4.6, revision 42).

Both events, a) and b), are analyzed in the Licensing Bases at Full Power, and use Pressurizer Pressure-Low as a first reactor trip. At zero power, with current proposed T.S. this reactor trip is proposed as Not Operable.

For event c), from Zero Power, Power Range Neutron Flux, High Set Point Trips the Reactor; Pressurizer Pressure-Low (SI) initiates Safety Injection; reference 7, page 15.4-29, revision 43, paras. 1 and 5. Whereas both these protections are proposed by the T.S. in MODE 2, they are not proposed for MODE 3 which differs from the circumstances of MODE 2 by only a marginal reduction in RCS Temperature.

The FSAR, reference 7, Table 15.4.6-1, revision 42, shows this occurrence as being the only event at Zero Power, analyzed to a smaller N° of RCPs than 4; it has been analyzed for 2 only. This is an accident with substantial but "acceptable to Condition IV occurrences" consequences in terms of fuel cladding damage and RCS overpressurization, but it required at least two RCPs to achieve that (in the Licensing Basis). Even the two RCPs required in this event are not proposed as being required for MODE 3.

The proposed circumstances in MODE 3 are clearly non-conservative with respect to the Licensing Bases. The licensee shall evaluate and propose.

Concerning the Large Break "Loss of Coolant Accident."

This is discussed in Accident Analyses in Reference 7, section 15.4.1 for a LOCA from rated power; in Reference 8, item 212.75 page Q 212.47, for a LOCA between RCS conditions of 1900 psig and 1000 psig/425°F in Hot Standbye; in item 212.90(6.3), page 212-61, for a LOCA at and less than 1000 psig/425° in Hot Standbye, and on page Q 212-61b, item 29 for a LOCA in the RHR Mode at 425 psig/350°F.

As for the Small Break LOCA, these analyses are presumably based on 4 RCS loop operation, with in general, loss of power to RCS Pumps on Safety Injection.

The large break LOCA analyses used the Topical Report WCAP-8479, reference 7, page 15.4-1. At this time, we expect no difference in the importance of RCPs to that discussed under the paragraph commencing "Concerning Small Break LOCA" which used the W Topical Report WCAP 8356 (reference 19) and which applied to both Large and Small Break LOCAs.

Given the circumstances of the proposed T.S., any T.S. allowing OPERABILITY of less than 4 RCS Loop in MODE 3 would be in nonconformance with the Licensing Basis FSAR in a nonconservative manner, and the licensee is required to evaluate and propose.

G.2.6.4 OCCURRENCES CAUSING AN INITIAL INCREASE OF RCS TEMPERATURE

Those events causing increases in RCS temperature are of concern because of the potential influence of the positive moderator temperature coefficient resulting from the increased boron concentration. These could be:

a) Main Rupture of a Main Feed Line (Reference 7, page 15.4-10, revision 30), although this is normally evaluated at Rated power with no provision for evalua' on as zero power.

b) Start up of an Inactive Reactor Coolant Loop

c) Loss of Offsite Power (reference 7, page 15.2-19, revision 7)

d) Partial Loss of Forced Reactor Coolant Flow (Reference 7, page 15.2-16, revision 7)

e) Complete Loss of Forced Reactor Coolant Flow (Reference 7, page 15.3-7, revision 7)

Except for item b; all these events are licensing bases events from Rated power, and not zero power, so that their importance would normally be minimal except for the positive Moderator Temperature Coefficient and the complete lack of Safety Related Reactor Trip protection proposed with the Reactor Trip System Instrumentation T.S.

At this time we see no protection against positive temperature coefficients in MODE 3 [4, 5 & 6].

Given the circumstances of the proposed T.S., Operability of less than 4 RCS Loops in MODE 3 would be in non-conformance with the current Safety Analyses Limits in a non-conservative manner and the licensee is required to evaluate and propose.

G.3 CONCLUSIONS

Occurrence II, III and IV Events in MODES 3, 4 and 5, can result in returns to power with high peaking coefficients requiring effective reactivity control and/or reactor core flow for RCS protection, including DNBR, at the very substantially reduced pressure levels in the loop [2250 psig to 425 psig and less]. Concomitant decreases in RCS temperatures are beneficial, but the importance of RCS pressure may be dominant. Acceptable RCS protection therefore requires RCS flows which are substantial, and/or effective reactivity control including combined action to limit potential reactivity excursions.

At ~~this~~ time, with the proposed T.S., 4 RCS loops (with increased Reactor Trip Protection) would be required at entry into and during MODE 3 to meet the requirements of just the Licensing Basis Events From Zero Power. In MODE 4, operation of 4 RCS Loops, whilst on RHR, may be undesirable because of the substantial additional burden on the RHR system; so, nonoperability of all RCPs must be compensated by other controllable factors such as inserting all movable control assemblies and removing power from the Reactor Trip System Breakers, closure of Main Feedwater [Containment] Isolation valves to both Main and Auxiliary Feedwater Systems, Closure of Main Steam Isolation Valves, and Boration Control measures additional to those included in the proposed T.S. An additional available alternate action is to use, within MODE 4, a minimum set of RCS pumps (and loops) as established by Safety Analysis, to cool the plant down to effectively zero pressure (gauge) in the Steam Generators [or less if the condenser was still available] before transferring the heat sink to the RHR system. This would ensure control of Steam Line Break, and LOCA events, small and large, down to RCS conditions where RCS flows are not necessary.

The current T.S. are nonconservative in respect to the Licensing Basis in respect to these concerns. The Licensee shall evaluate and propose.

T.S. Page 3/4 4-2: RCS HOT STANDBY

*This Footnote proposes that; in HOT STANDBY (MODE 3):

"*All reactor coolant pumps may be de-energized for up to 1 hour 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 saturation temperature."

This is a natural circulation condition; the only Licensing Basis calculation for this is the Natural Circulation calculations of reference 7, page 15.2-27, "Loss of Offsite Power to Station Auxiliaries"; but at MODE 2 Zero Power conditions with related programmed process conditions of Zero Load Pressure and Temperature in the loops. No basis is provided for ensuring that natural circulation will be safe over the range of conditions now expected in this MODE 3. Earlier considerations show that more comprehensive protections against the possibility of Condition II, III and IV occurrences must involve, in addition to isolation of all boron dilution sources, securing Reactor Trip System Breakers in the Open Position, closure of MFW isolation valves, isolation of MSIVs, and possibly an optimum boron concentration. At present, the only Licensing Basis for controlling this particular situation is the Emergency Operating Guidelines.

Given the circumstances of the proposed T.S., the proposal to de-energize 4 RCPs for up to one hour is outside the Safety Analysis Limits of the FSAR and is non-conservative with respect to that.

The licensee shall provide the reason for this requirement including the expected condition of the Facility, and then analyze, evaluate and propose.

Earlier concerns under General 2.6.1 addressed the need to evaluate the consequences of the Start Up of an Inactive Reactor Coolant Loop in this MODE. No apparent T.S. provision has been provided in the proposed T.S. The licensee shall evaluate and propose.

Action item b. states:

"b. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective ACTION to return the required reactor coolant loop to operation."

This instruction is invalid. The only Licensing Basis action available is the Emergency Operating Guidelines for the Natural Circulation. This proposal is non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

T.S. Section 3/4 4.2 SAFETY VALVES

OPERATING

The proposed T.S. requires all [3] pressurizer Code Safety Valves to be Operable in Applicable Modes 1, 2 and 3.

The Surveillance Requirements should contain the minimum discharge capacity required of this valve as defined in the Licensing Basis. They should also ensure the maintenance of satisfactory environmental conditions consistent with reliable valve operability. The licensee shall evaluate and propose.

T.S. Section 3/4.5.2.A Proposed

A proposed new Section which would be titled: ECCS Subsystem - Applicability between 1000 psig/425°F and 425 psig/350°F.

This would provide for: One ECCS subsystem comprising the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path.

Also, one ECCS subsystem comprising the following shall also be OPERABLE

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path.

All breakers for all safety injection pumps and all but the one operable centrifugal charging pump are opened, locked and tagged (reference earlier information).

As explained in the previous section, limited operation of the higher pressure pumps between 1000 psig/425°F and 425 psig/350°F apparently provides Low Temperature Overpressure Protection (LTOP). The proposed T.S. requires all CI and SI pumps to be available during these conditons and is therefore non-conservative with respect to the Licensing Basis and particularly in respect of Overpressure Protection. The licensee shall evaluate and propose, and in so doing provide the analyses and evaluation which required constrained operability of the higher pressure pumps in this operating phase, in his Licensing Basis FSAR.

The proposed T.S. under this Section requires a minimum of one only ECCS subsystem comprising

a. One Operable Centrifugal Charging Pump (CCP)

b. One Operable RHR Heat Exchanger

c. One Operable RHR Pump

d. An Operable Flow Path

There are no Safety Analyses or Evaluations of one only ECCS subsystem allowing for a single active failure in one only train. This proposition is therefore non-conservative with respect to the Licensing Basis FSAR. The Licensee shall evaluate and propose.

The ECCS of the Licensing Basis FSAR require the same operability of ECCS equipment as is required for TS 3/4 5.2A Proposed. So that in addition to:

- One ECCS subsystem comprising the following shall be OPERABLE:

  a. One OPERABLE centrifugal charging pump,

  b. One OPERABLE RHR heat exchanger,

  c. One OPERABLE RHR pump, and

  d. An OPERABLE flow path

which is the same as for the proposed T.S., it is also required that:

One ECCS subsystem comprising the following shall also be OPERABLE:

  b. One OPERABLE RHR heat exchanger,

  c. One OPERABLE RHR pump, and

  d. An OPERABLE flow path.

T.S. Section 3/4.7 PLANT SYSTEMS

T.S. Page 3/4 7-4: AUXILIARY FEEDWATER SYSTEMS

Item: APPLICABILITY MODES 1, 2 and 3 in the proposed T.S. should be expanded to MODES 4 and 5 in accordance with our review under Table 3.3-3 ESFAS INSTRUMENTATION, Items 7 a, b, c, d, e, and f. The conclusions from that review are: The proposed T.S. items are generally non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

T.S. Section 3/4.7.5 STANDBY NUCLEAR SERVICE WATER POND (SNSWP)

APPLICABLE MODES: The system is required in all MODES 1, 2, 3, 4, 5, & 6 to handle heat rejection requirements as the ultimate heat sink. The licensees proposal to limit this to MODES 1, 2, 3 and 4, is nonconservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

Reference 6, page 9.2-13, revision 39, states that "In the event of solid layer of ice" forms on the SNSWP, the operating train [of the Nuclear Service Water [NSW] system] is manually aligned to the SNSWP. The Licensee shall provide the Safety Related reason for this action and advise if this operator action conflicts with the Response Times proposed under Table 3.3-5. Given a Safety Related reason, surveillance requirements ensuring this action should be included under either T.S. Section 3/4.7.5 NSWS or this particular T.S. Section 3/4.7.5 STANDBY NSWP. Absent this surveillance requirement on a Safety Related Issue, the proposed T.S. would be non-conservative. The Licensee shall evaluate and propose.

T/S Page 3/4 9-11 REFUELING OPERATIONS LOW WATER LEVEL

It is proposed that an additional item be added to the current statement of APPLICABILITY to the effect that: This MODE shall not to 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. The licensee shall evaluate and propose.

The existing LCO specifies that:

> "3.9.8.2 Two independent residual heat remval (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*"

Additionally, the current FSAR requires that each of the RHR trains be provided with power from two (2) redundant electrical buses so that each pump receives power from a different source; reference 20, page 5.5-24, revision 9. Without this requirement, the T.S is less conservative than the FSAR and the licensee shall evaluate and propose.

Additionally, the current FSAR, reference 8, page Q212-57, revision 25, describes that in the event of loss of flow caused by closure of the RHR/RCS isolation valve, [and also by cessation of flow in the system]

> "The operator would be alerted to the loss of RHR flow by the RHR low flow alarm.
>
> Assuming worst case conditions (maximum 24 hours decay heat,--and the RCS drained to just below the vessel flange) and making conservative assumptions about the amount of water available to heat up and boil off, if the operator took no action, boiling would begin in about five minutes, the water level in the vessel would be down to the level of fuel in about 100 minutes."

In the event only 1 RHR loop is required to be in operation, the LCO should therefore require 2 operable safety related RHR low flow alarms on each single operating system so that the operator can respond within 10 minutes to commence operation of the redundant system. Is this time frame excessive since boiling will have commenced. It is necessary to maintain two operating RHR systems so that boiling will not occur with a single failure. The licensee shall evaluate and propose.