NRC PROJECT MANAGER BRIEFING

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OF

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

PROJECT I

OF

CONTRACT NO.: NRC-03-85-051

28 JANUARY, 1986

ACCIDENTS FROM CHAPTER 15 OF MCGUIRE FSAR

EQUIPMENT	1.1	1.2	1.3	1.4	1.5	2.2	2.3	2.6	2.7	2.8	3.1	3.2	3.3	3.4	4.1	4.2	4.3	4.4	4.6	4.8	5.1	6.1	6.2	6.4
Feedwater System	I	I	I	I	I	I	I	I	I	I	I	I	I	I		I							I	
Feedwater System Component, Control or Trip	I	I	I	Q	Q			I	I	I	I	I	I	I		I		I			I	I		
Steam Dump and Steam Dump Control System		N	N	N	N	N	N	N	N	N	N	N	N	N		N		N			N	N	N	N
Turbine Stop Valves and Control System			N			N	N														I			
Steam Generator Power Operated Relief Valves				N				N	N							N	N							
Turbine Trip from Reactor Trip																Q							Q	
RCCA Rod Withdrawal Blocks (All)															N	N								
Reactor Water Makeup-Paths, Alarms and Trips																			N					
Reactor Vessel		I	I	I	I	I	I	I	I	I	I	I	I	I		I		I			I	I		
Pressurizer Spray		N	N	N	N	N	N	N	N	N	N	N	N	N		N		N			N	N		
Reactor Trips High Pressurizer Level/Low Pressure/Low Flow		N	N	N	N	N	N	N	N	N	N	N	N	N				N			N	N	N	N
Chemical and Volume Control Specific Components (VCT)																					I			
Power Range Nuclear Flux Trips, Rate and High and Low Power Level															N					Q				
Intermediate Range Flux Trips															N									
Source Range Flux Trips															Q									
Two Reactor Coolant Pumps For 20D Withdrawal															Q									
Reactor Trips with Relation to P-7																Q								
Rod Deviation Alarm																	Q							
Three Loop Operation and P-8 Setpoint																		Q						
Trips from High-1 and High-2 Containment Pressure Signals (McGuire)					Q																			
High Containment Pressure Trips in Mode 4					Q																		Q	
Rod Insertion Limit Level Alarm																				Q				
Auxiliary Building Filters																				Q				
Containment Sump																								Q
Steam Generator Safety Valves (and MSIV)																							Q	

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (1) Feedwater System.

Findings:

Accident Analysis: Sixteen of the thirty-nine accidents analyzed referred to the main feedwater system as operable equipment.

Technical Specifications: The Westinghouse STS does not include operability requirements for the main feedwater system.

Statement of Problem: The main feedwater system is considered operating in most accident analyses but is not required by the STS.

Discussion: The feedwater system is considered a normally operating system required to achieve the basic purpose of overall plant operation. In most cases, the equipment was listed in accident analyses as operating in the initial plant condition or as equipment whose failure or misoperation contributed to initiating the accident. After initiation of accidents, no credit was taken for feedwater system operation. Normal, emergency systems and functions take over to mitigate or terminate the accident.

Recommended Resolution: No action recommended.

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Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (2 [a]) Feedwater System-Redundant Isolation of the Feedwater Lines.

Findings:

> Accident Analysis: Two of the twenty-nine accident analyses referred to the redundant isolation of the feedwater lines. The statement used in the analyses was:
>
> "In addition to normal control action which will close the main feedwater valves following reactor trips, a safety injection signal would trip the main feedwater pumps and will generate a feedwater isolation signal which will rapidly close all main feedwater control valves, isolation valves, and pump discharge valves."
>
> Technical Specifications: The Westinghouse STS could not be used to verify these actions.

Statement of Problem. This trip is one of those specifically listed in the program but it is not a trip in the Westinghouse STS Rev. 4 or 5.

Discussion: Functions such as feedwater isolation after reactor trip and the signal that closes the feedwater pump discharge valves could not be verified in the STS. These actions were; however, verified separately in system sections of the FSAR suppplemented by figure 7.2.1-1, 13 of 16.

Recommended Resolution: No action or changes recommended.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (2 [b]) Feedwater System Components and Trips Including Control System

Findings:

> Accident Analysis: Sixteen of the twenty-nine accident analyses referred to the feedwater system and twenty-three accidents referred to a component of the feedwater system or the control system for feedwater.
>
> Technical Specifications: The Westinghouse Standard Technical Specification (STS) states a limiting condition for operation (LCO) for the auxiliary feedwater system (3.7.1.2) and refers to many trips for Engineered Safety Features Activation System (ESFAS) which came from auxiliary feedwater instrumentation or actuate auxiliary feedwater trips, isolations, or pump starts.
>
> LCOs for the normal feedwater system or components do not exist in the STS.

Statement of Problem: Operability requirements do not exist in the Westinghouse STS to support the main feedwater system.

Discussion: The feedwater system is considered a normally operating system required to achieve the basic purpose of overall plant operation. In most cases, the equipment was listed in accident analyses as operating in the initial plant condition or as equipment whose failure or misoperation contributed to initiating the accident. After initiation of accidents, no credit was taken for feedwater system operation. Normal, emergency systems and functions take over to mitigate or terminate the accident.

Recommended Resolution: No action recommended.

Note: Feedwater isolation is an issue covered elsewhere in this report - item 2(a).

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (3) Steam Dump (Turbine Bypass) and Steam Dump Control System.

Findings:

Accident Analysis: Nineteen of the twenty-nine accident analyses had reference to the steam dump or steam dump control system.

Technical Specifications: The Westinghouse STS has no requirement for operability of the steam dump or its control system.

Statement of Problem: The steam dump is generally referenced in FSAR accident equipment lists but has no operability requirement in the STS.

Discussion: This system is heavily involved in reactor power operations and is a factor in plant operations in modes 1, 2 and 3 because it is in use for basic plant power control, cooldown or temperature control.

The reason it was referenced in so many of the accident analyses is because of its inclusion in the LOFTRAN analysis. The LOFTRAN analysis, however, does not take credit for the operation of non-safety systems to mitigate the accident. A typical assumption would be that the controllers remain at the pre-accident element levels. Reference WCAP-7907.

Recommended Resolution: No action is recommended.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (4) Turbine Stop Valves and Turbine Control.

Findings:

Accident Analysis: The main turbine stop valves and turbine control system are referred to in four of the twenty-nine accident analyses.

Technical Specifications: Although several interactions with turbine trip are referenced in the instrumentation portion of the Westinghouse STS, there are no operability requirements specified for the turbine stop valves or turbine control system.

Statement of Problem: No STS operability requirements exist for the main turbine stop valves and turbine control system.

Discussion: In the accident analyses, non-safety control systems are not credited with mitigating the accident. Such control systems are modeled, but any assumptions concerning control system action are selected to be conservative.

Recommended Resolution: No change is recommended to the Westinghouse STS.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered. (5) Steam Generator Power Operated Relief Valves (PORV).

Findings:

Accident Analysis: The steam generator PORVs are referred to in five of the twenty-nine accident analyses.

Technical Specifications: The steam generator PORVs do not have operability requirements in the Westinghouse STS.

Statement of Problem: The steam generator PORVs are listed in FSAR accident equipment lists but do not have operability requirements in the Westinghouse STS.

Discussion: Use of this equipment is referenced for plant cooldown and operation during transients to avoid lifting steam generator safety valves (use words from item 7 NRC).

Recommended Resolution: To be determined.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (6) Turbine Trip from Reactor Trip.

Findings:

Accident Analysis: Three of the twenty-nine accident analyses referred to trip of the main turbine from reactor trip.

Technical Specifications: This trip could not be confirmed in the Westinghouse STS.

Statement of Problem: Turbine trips from reactor trip is referenced in the FSAR accident analyses but cannot be confirmed directly in the Westinghouse STS.

Discussion: Indirectly, this trip can be confirmed bcecause it is one of the functions of interlock P-4, which is specified for modes 1, 2 and 3 in Table 3.3-3. The function of turbine trip from reactor trip can therefore be confirmed from page B 3/4 3-3, section 3/4 3.1 and 3/4 3.2 of the STS.

Recommended Resolution: No action recommended.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (7) RCCA Rod Withdrawal Blocks.

Findings:

> Accident Analysis: The RCCA rod withdrawal blocks are referenced in two accident analyses (15.4.1 and 14.4.2).
>
> Technical Specifications: The Westinghouse STS does not include operability requirements for the Rod Control System, which includes this rod block.

Statement of Problem: RCCA rod withdrawal blocks are referenced in rod withdrawal accidents but have no technical specification requirements for operability.

Discussion: Although the trip operability cannot be confirmed in the Westinghouse STS, the rod withdrawal stops are not considered used or in operation for purposes of analyzing the accidents. In fact, a rod withdrawal rate greater than the worth of two rod banks is considered until the reactor protection system terminates the accident.

It should be noted that if these rod blocks function as intended, then the accident is prevented or minimized.

Recommended Resolution: No action recommended.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (8) Reactor Water Makeup - Paths, Alarms and Trips

Findings:

> Accident Analysis: In accident analysis 15.4.6, the equipment list includes the high flow alarm at the discharge of the CVCS being active, lights in the control room panels being operable, and signals for automatic flow path linings.
>
> Technical Specifications: The Westinghouse STS does not include these alarms, lights, or automatic valve operations on the water supply systems.

Statement of Problem: The Westinghouse STS does not state operability requirements for the indicator lights, alarm at the discharge of the CVCS system, or the automatic valve linings in this system.

Discussion: The accident considered was CVCS malfunction that causes dilution of the boron concentration in the reactor coolant. The equipment and components listed are those whose failure affects the accident. There is an implied operability requirement for the equipment but nothing specific in the STS. Therefore, surveillance could be an issue.

Recommended Resolution: To be determined.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (9) Reactor Vessel.

Findings:

Accident Analysis: The reactor vessel (and sometimes the Reactor Vessel with core) was stated in seventeen of the twenty-nine accident analyses, primarily because it was stated as a considered component in the LOFTRAN computer code which was used for these accidents.

Technical Specifications: No operability requirements are included in the Westinghouse STS for the reactor vessel or reactor vessel with core.

Statement of Problem: Many accident equipment lists included reactor vessel or reactor vessel with core, but the STS has no operability requirements.

Discussion: In each case noted during the review, the designator used was an I. No specific functions of the reactor vessel or core were used in the LOFTRAN code which would relate to an operational limitation.

Recommended Resolution: No action recommended.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (10) Pressurizer Spray.

Findings:

> Accident Analysis: The pressurizer spray is considered in seventeen of the twenty-nine accidents analyzed.
>
> Technical Specifications: The Westinghouse STS specifies an LCO for the pressurizer including two banks of heaters (3.4.3) but not for the pressurizer spray.

Statement of Problem: The FSAR accident analysis equipment list includes the pressurizer spray, but the Westinghouse STS has no operability requirements for this equipment.

Discussion: It is to be noted that the effect of the pressurizer spray to reduce a pressure transient is not credited in the accident analyses.

Although the LOFTRAN and MARVEL computer codes include the effects of heaters, spray, and relief and safety valves with their appropriate control systems, safety analysis calculations are conservatively performed assuming no pressure control if such control would improve the results (WCAP-7907).

Recommended Resolution: No addition to the Standard Technical Specification is recommended.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered:

(11) Reactor Trips:
Low Pressurizer Pressure
High Pressurizer Level
Low Coolant Flow including underfrequency and undervoltage trip.

Findings:

Accident Analysis: Eighteen of the twenty-nine accident analyses credited the subject reactor trips when operational modes 2 are stated for the plant condition.

Technical Specifications: The Westinghouse STS lists the applicable modes for these trips as mode 1 only.

Statement of Problem: The Westinghouse STS does not require reactor trips for low pressurizer pressure, high pressurizer level, and low coolant flow including undervoltage and underfrequency trips in mode 2.

Discussion: More information required.

Recommended Resolution: To be determined.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Generic Technical Specifications

Equipment Considered: (12) Chemical and Volume Control Components

Findings:

Accident Analysis: The Volume Control Tank (VCT) is listed as operable equipment in accident analysis 15.5.1 - Inadvertant operation of ECCS during power operation.

Technical Specifications: No Westinghouse STS operational requirements exist for the VCT.

Statement of Problem: The VCT is listed as accident related equipment but has no operability requirements.

Discussion: This accident equipment list included the VCT because it is a source of water for the accident. Operability is implied because this equipment is an integral part of the reactor water makeup system.

Recommended Resolution: No change to the Westinghouse STS is recommended.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (13) Power Range Nuclear Flux Trips, high rate and low and high flux level.

Findings:

> Accident Analysis: The equipment list for the analysis of accident 15.4.1, uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition, includes this instrumentation. It is also listed as a question for higher numbered modes (3 and above) in accident 15.4.8.
>
> Technical Specifications: Westinghouse STS requires power range neutron flux instrumentation in Modes 1 and 2.

Statement of Problem: Power range neutron flux instrumentation is listed for accident analysis 15.4.1 but the equipment is not required by the Westinghouse STS in some modes.

Discussion: The FSAR accident analysis credits this instrumentation with Tripping the uncontrolled RCCA bank withdrawal from a subcritical condition (Table 7.2.1.4). There are specific evolutions (cold rod testing) where the reactor trip breakers are allowed to be closed in the higher numbered modes.

Recommended Resolution: Conduct evaluation necessary to add shutdown requirements to Table 3.3-1 for power range nuclear instruments to be operable in modes 3*, 4*, and 5*, where the asterisk qualifies the requirement "with reactor trip breakers closed."

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (14) Intermediate Range Nuclear Flux Instrumentation in Modes 3, 4 and 5.

Findings:

Accident Analysis: The equipment list for the analysis of accident 15.4.1, uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition includes this instrumentation.

Technical Specifications: Westinghouse STS requires intermediate range neutron flux instrumentation in Modes 1 and 2.

Statement of Problem: Power range neutron flux instrumentation is listed for accident analysis but the equipment is not required by the STS.

Discussion: The FSAR accident analysis credits this instrumentation with Tripping the uncontrolled RCCA bank withdrawal from a subcritical condition (Table 7.2.1.4). There are specific evolutions (cold rod testing) where the reactor trip breakers are allowed to be closed in the higher numbered modes.

Recommended Resolution: Conduct evaluation necessary to add shutdown requirements to Table 3.3-1 for power range nuclear instruments to be operable in modes 3*, 4*, and 5*, where the asterisk qualifies the requirement "with reactor trip breakers closed."

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (15) Source Range Neutron Flux Instrumentation.

Findings:

> Accident Analysis: The equipment list for the analysis of accident 15.4.1, uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition includes this instrumentation.
>
> Technical Specifications: Westinghouse STS requires source range neutron flux instrumentation in Modes 1 and 2.

Statement of Problem: Source range neutron flux instrumentation is listed for accident analysis but the equipment is not required by the STS.

Discussion: The FSAR accident analysis credits this instrumentation with Tripping the uncontrolled RCCA bank withdrawal from a subcritical condition (Table 7.2.1.4). There are specific evolutions (cold rod testing) where the reactor trip breakers are allowed to be closed in the higher numbered modes.

Recommended Resolution: Conduct evaluation necessary to add shutdown requirements to Table 3.3-1 for power range nuclear instruments to be operable in modes 3*, 4*, and 5*, where the asterisk qualifies the requirement "with reactor trip breakers closed."

Note: This source range instrumentation operability requirement was included in rev. 4 of the Westinghouse STS.

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Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (16) Two reactor coolant pumps (for rod withdrawal).

Findings:

Accident Analysis: Accident analysis 15.4.1 - Uncontrolled RCCA bank withdrawal from subcritical or low power startup condition - listed two reactor coolant pumps as operable equipment.

Technical Specifications: The Westinghouse STS states reactor coolant loop operability as follows:

Mode_ : and 2 -	all loops operating
Mode 3 -	at least 2 loops operable with 1 in operation
Mode 4 -	two reactor coolant and/or RHR lo ps operable with 1 in operation
Mode 5 -	at least 1 RHR loop operable and in operation with another RHR loop operable or 2 steam generators filled on secondary side.

Statement of Problem: For rod withdrawal situations in modes 3, 4, and 5, the Westinghouse STS does not require minimum reactor coolant flow of 2 loops for DNB conservatism assumed in FSAR accident analysis. There are specific evolutions (cold rod testing) where the reactor trip breakers are allowed to be closed in the higher numbered modes.

Discussion: The analysis for the accident requires two reactor coolant pumps to be in operation to be conservative with respect to DNB.

Recommended Resolution: Add a requirement to the Westinghouse STS to run at least two reactor coolant pumps during RCCA withdrawal. This may be placed in special test exceptions section or as a note in modes 3, 4, and 5.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (17) Reactor Trip on High Pressurizer Level (2 of 3) Above P-7.

Findings:

> Accident Analysis: This trip was referenced in accident 15.4.2 - Uncontrolled RCCA withdrawal at power - with a limitation that it applied with reactor power above the P-7 setpoint.
>
> Technical Specifications: This trip is specified in STS rev. 5 Table 3.3-1 as functional unit 11 but no reference is made to any limitation of the P-7 permissive. It is shown as applicable in only mode 1.

Statement of Problem: The power limitation of the high pressurizer level reactor trip due to permissive P-7 is not shown in Table 3.3-1.

Discussion: None.

Recommended Resolution: It is recommended that a notation be added that this trip is applicable above the P-7 (words from results of discussion) setpoint as follows:

Note: This same note applies to six trips:

- Low Pressurizer Pressure
- High Pressurizer Level
- Low Flow 1 Loop
- Underfrequency on Reactor Coolant Pump Bus
- Undervoltage on Reactor Coolant Pump Bus
- Turbine Trip

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Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (18) Rod Deviation Alarm.

Findings:

> Accident Analysis: The rod deviation alarm was referenced in accident analysis 15.4.3.
>
> Technical Specifications: This equipment is not referenced in the Westinghouse STS.

Statement of Problem: Equipment listed in FSAR accident analysis is not referenced in the Westinghouse STS.

Discussion: This alarm is generated by the McGuire unit computer and provides a visual printout and an auditable alarm whenever an individual rod position signal deviates from the other rods in the bank by a preset limit (McGuire FSAR Vol. 7 Section 7.7.1.3.4).

For purposes of accident analysis, this alarm was not actually considered or credited as the rod movement was allowed to progress to reactor trip or limit without operator action to stop or reverse rod motion. Therefore, an STS LCO is not warranted for this alarm.

Recommended Resolution: No action recommended.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (19) Three-Loop Operation with Relation to the P-8 Setpoint Reset for Three-Loop Operation.

Findings:

Accident Analysis: The review of the analysis of accident 15.4.4 - Startup of an inactive reactor coolant pump at an incorrect temperature resulted in questions as to the tech spec operability requirements to support the stated trip.

Technical Specifications: This trip can be confirmed in the Westinghouse STS Table 3.3-1 item 19.C where the words are stated: "Power Range Neutron Flux." Also, item 12.a - Single loop reactor coolant low flow is required operable in mode 1.

Statement of Problem: The interaction of the interlocks/permissives with regard to loss-of-flow in one loop requires an extensive knowledge of the details of the permissives and trips. Although the interactions can be worked through in the existing STS (and no change is recommeded to the STS), this is the sort of problem that should be assessed in a major revision to the STS.

Discussion: None

Recommended Resolution: No action recommended.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (20) Trips from High-1 and High-2 Containment Pressure Signals. (McGuire Problem)

Findings:

Accident Analysis: A question was generated during the review of accident 15.1.5 - Steam System Piping failure - because the safety injection activation was stated to be generated from two of four high containment pressure signals and the feedwater isolation was stated to be generated by two of three high-high containment pressure signals.

Technical Specifications: The Westinghouse STS Rev. 5 has both these trips but they are reversed in the McGuire accident analysis writeup.

Statement of Problem: High containment pressure trip operability requirements are reversed in the McGuire T.S.

Discussion: None.

Recommended Resolution: Correct the McGuire TS to correspond with the Westinghouse STS.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (21) Containment Pressure Trips Applicablity in Mode 4.

Findings:

> Accident Analysis: Accident 15.1.5 - Steam System Piping Failure, and 15.6.2 - Steam Generator Tube Failure, Equipment Lists include high containment pressure trips for safety injection.
>
> Technical Specifications: The Westinghouse STS requires operability of containment pressure trips in Modes 1, 2, and 3.

Statement of Problem: There is no operability requirement in mode 4 for high containment pressure trip.

Discussion:

Recommended Resolution: Contingent on additional information.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (22) Rod Insertion Limit Level Alarm.

Findings:

Accident Analysis: This alarm was questioned in review of accident analysis 15.4.8 " Spectrum of Rod Cluster Control Assembly Ejection Accidents."

Technical Specifications: This alarm is not specified in the Westinghouse STS.

Statement of Problem: This alarm was listed in FSAR equipment list but has no operability requirements in the Westinghouse STS.

Discussion: This rod insertion limit level alarm was not actually considered or credited as the rod movement was allowed to the point of reactor trip or limit without operator action to stop or reverse rod motion. Therefore, no STS - LCO is warranted.

Recommended Resolution: No action recommended.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (23) Auxiliary Building Filters.

Findings:

Accident Analysis: The auxiliary building filters were referenced in accident analysis 15.4.8. Credit for these filters was, however, not credited in the accident analysis.

Technical Specifications: This filter train is not referenced in revision 5 of the STS.

Statement of Problem: Filters were included in FSAR accident equipment list but not required by STS.

Discussion: Filters were not credited as functional in the analysis.

Recommended Resolution: No action recommended.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (24) Containment Sump.

Findings:

Accident Analysis: The containment sump was listed as operable equipment in the analysis of accident 15.6.4 - Loss of Coolant Accidents.

Technical Specifications: The Westinghouse STS has no operability requirement for the containment sump.

Statement of Problem: Containment sump was listed as operable equipment for FSAR accident analysis but was not required operable by STS.

Discussion: The containment sump was stated as a source of water for the RHR pump during the later phases of the LOCA. There is an implied requirement for operability of the containment sump. To add an operability requirement for this equipment would unnecessarily expand the Westinghouse STS.

Recommended Resolution: No action recommended.

Discussion of Items from Comparison of McGuire FSAR Accident Equipment Lists with Standard Technical Specifications

Equipment Considered: (25) Steam Generator Safety Valves and Main Steam Isolation Valves (MSIV) Operability (in mode 4).

Findings:

> Accident Analysis: Accident analysis 15.6.2 - Steam Generator Tube Failure - equipment list included the Steam Generator Safety Valves. The MSIVs were added because of the potential need to maintain the steam generator isolated in Mode 4.
>
> Technical Specifications: The Westinghouse STS states operability requirements for the Steam Generator Safety Valves and MSIVs for modes 1, 2 and 3.

Statement of Problem: Operability of MSIVs are required in modes 1, 2, and 3 but not in Mode 4 where events such as steam generator tube leakage could require their operation.

Discussion: The steam generator tube rupture in Mode 4 could require use of the Steam Generator Safety Valves and MSIVs. The interaction of MSIVs with the containment isolation system is not clear in mode 4, where containment integrity is applicable but the MSIV operability is not.

Recommended Resolution: To be determined.

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June 10, 1986

[redacted] Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: B.J. Youngblood, Director
PWR Project Directorate #4

Subject: McGuire Nuclear Station
Docket Nos. 50-369 and 50-370
NRC DPO Concerns on McGuire Technical Specifications

Dear Mr. Denton:

Mr. T.M. Novak's (NRC/ONRR) July 9, 1985 letter to Mr. H.B. Tucker (DPC) indicated that a review of the McGuire Unit 1 and 2 Technical Specifications was being conducted in response to concerns raised by a member of the NRC staff in a differing professional opinion (DPO) resulting from a review of the proof and review copy of the McGuire Unit 1/2 combined Technical Specifications which existed in mid-January 1983. Duke Power Company's comments were requested on certain plant-specific concerns contained in the DPO (other concerns contained in the DPO were either being considered by the NRC for generic resolution, had been closed by NRC internal review, or were still under review).

Attached is Duke Power Company's response to these concerns. This response is limited to the specified plant-specific concerns and does not address any generic aspects of these specified concerns. Note that the response has potential plant-specific impacts on the station's Technical Specifications (e.g. question nos. 6a, 7d (and 7i, 7k), and 7n) and FSAR (e.g. questions 4a&b, and 4c). Duke will pursue appropriate plant-specific Technical Specification and FSAR revisions following NRC concurrence with the positions contained herein. The Westinghouse Standard Technical Specification issues identified in this response should be resolved on a generic basis (note that Westinghouse review/input was utilized in the development of this response). Note also that generic Technical Specification improvement efforts currently underway by industry (e.g. AIF, WOG, B&WOG) and NRC (TSIP) may impact the DPO's concerns and the resolutions proposed by this response.

As indicated above, the NRC is requested to approve this response prior to Duke proceeding with the appropriate Technical Specification change submittals and inclusion of the informaiton in a future FSAR update. Should there

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be any questions regarding this matter or if additional information is required, please advise.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

PBN/jgm

Attachment

xc: Dr. J. Nelson Grace, Regional Administrator
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Duke Power Company

McGuire Nuclear Station

Response to NRC DPO Concerns

(Question 1)

TABLE 2.2-1

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.

Response: The data contained in Reference 18 has been confirmed to be valid for both McGuire Unit 1 and Unit 2. The instrumentation hardware (racks, transmitters) are the same for both Units 1 and 2. While the Steam Generators are different (D-2 for Unit 1 and D-3 for Unit 2), there are no differences in the Safety Analysis values. Therefore it can be concluded that the Setpoint Study performed for Unit 1 is applicable, in it's entirety, to Unit 2. The safety analysis performed is valid for both units and use the same equipment/instrumentation resulting in uncertainty values being valid for both units.

(Question 1a)

TABLE 2.2-1, Item 3

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

Response: The dynamic response of the High Positive Rate trip function is similar to the rate/lag function associated with the ΔT trips. The responses of the various dynamic functions are demonstrated in Appendix A of WCAP-8745 (Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions). As may be seen in the above mentioned figures, an increased time constant results in faster response and thus results in a shorter time from initiation of transient to reactor trip. Therefore, the >2 seconds Tech Spec requirement for the time constant is conservative.

(Question 1b)

TABLE 2.2-1, Item 4

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.

Response: The dynamic response of the High Negative Rate trip function is similar to the rate/lag function associated with the ΔT trips. The responses of the various dynamic functions are demonstrated in Appendix A of WCAP-8745 (Design Bases for the Thermal Overpower ΔT and Thermal Over temperature ΔT Trip Functions). As may be seen in the above mentioned figures, an increased time constant results in faster response and thus results in a shorter time from initiation of transient to reactor trip. Therefore, the >2 seconds Tech Spec requirement for the time constant is conservative.

The Revision 7 FSAR analysis referred to in this inquiry was performed prior to the NRC review and approval of WCAP 10297-P-A (Dropped Rod Methodology For Negative Flux Rate Plants). The methodology used prior to WCAP-10297-P-A did not involve an actual determination of the negative flux rate setpoint and/or determination of the maximum dropped rod(s) worths which might not result in a reactor trip. The statement in the FSAR (RCCA group results in reactivity insertion of ~-1200 pcm which results in a reactor trip within ~ 2.5 seconds) was meant only to offer support for the DNB analysis performed at lower rod worths but did not actually demonstrate the adequacy of the negative flux rate setpoint.

Upon determination of possible nonconservatisms in the analytical methodology, Westinghouse developed the dropped rod methodology outlined in WCAP-10297-P-A. The revised methodology links the assumptions regarding the negative flux rate setpoint, rod worths and locations, control system behavior, and other factors which influence plant behavior following a dropped rod(s) event. The setpoint thus becomes an integral part of the safety analysis and the table in reference 18 is revised to show a safety analysis limit of 6.9% RTP. The adjustments made to account for various uncertainties results in an STS Trip Setpoint of 5.0% RTP and an STS Allowable Value of 5.5% RTP. Details regarding the revised methodology and basis for the setpoint may be found in WCAP-10297-P-A.

(Question 1c)

TABLE 2.2-1, Item 9

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 15.2.3-1 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.

Response: The analysis of the inadvertent operation of ECCS during power operation had assumed a low pressure setpoint of 1800 psia while other analyses assumed a setpoint of 1835 psig. The reference 18 value for the safety analysis limit was in error but was conservative and since margin exists between implemented and required setpoints, the conservatism did not impact the trip setpoint and allowable values.

The transient analyses have been reanalyzed as a result of the transition to optimized fuel assembly design. The revised analyses assumed a safety analysis limit of 1850 psia (1835 psig) for all transients.

(Question 1d)

TABLE 2.2-1, Item 13

Reference 18, page 3-13, Note 12 describes the Safety Analysis Limit for this item as a 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?

Response: The analysis in effect at the time this question was posed is no longer applicable. At present the bounding analysis for the steam generator lo-lo level is the feedbreak analysis. This analysis is done assuming the system starts at full power. In this analysis the safety analysis limit is 23% of narrow range span. As is indicated in the technical specifications this corresponds to a nominal trip setpoint of 40% narrow range span at 100% RATED THERMAL POWER.

(Question 1e)

TABLE 2.2-1, Item 18b

Accidental Depressurization of the main steam system is from zero load. It is unclear from reference 5 Table 7.2.1-4, (page 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 SI, or from SI initiated by other initiators. The Licensee shall clarify, and hence its validity with respect to the absence of the signal caused by P-7.

Response: Protection against accidental depressurization of the main steam system is provided by the overpower reactor trips (neutron flux and ΔT) and by the reactor trip which results from the receipt of the safety injection (SI) signal. The safety injection signal is actuated by low steamline pressure, low pressurizer pressure, or high containment pressure. The analysis performed results in SI initiation on low pressurizer pressure and reactor trip will either occur concurrently due to the trip on SI actuation or will occur prior to SI on the overpower trips. The main steam depressurization analyzed in the FSAR is initiated from hot shutdown conditions at time zero (i.e. reactor tripped) since this represents the most conservative initial condition. Thus no explicit assumption is made regarding the cause of reactor trip for the FSAR analysis. As noted in the FSAR and above, should the reactor be just critical or operating at power a reactor trip would occur on the overpower trips or from an SI actuation. In either case, no credit is taken for the reactor trip on pressurizer pressure when reactor power is below the P-7 interlock.

(Question 2)

T.S. Page 3/4 1-6

The existing minimum temperature for criticality (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°F, and any variation therefrom in either direction would be unacceptable.

Response: FSAR Figure 5.3.3-1 gives the normal relationship between reactor coolant system temperature and power. The hot zero power temperature employed at McGuire and used in the safety analysis is 557°F. The minimum temperature for criticality is determined such that the moderator temperature coefficient is within its analyzed temperature range, the trip instrumentation is within its operating range, the pressurizer is capable of being in an operable status with a steam bubble, and the reactor vessel is above its minimum RT_{NDT} temperature. The minimum temperature for criticality limit in the McGuire Technical Specifications is 551°F.

The difference between the HZP temperature and minimum temperature for criticality limit is required in order to allow for measurement of the moderator temperature coefficient. Since the moderator coefficient is confirmed to be within safety analysis assumptions at conditions of approximately 551°F - 557°F, the only input parameter to the safety analysis of concern is the initial temperature. The change in initial conditions from 557°F to 551°F for transients occurring at HZP would have a negligible impact on results and would be a less representative input since the majority of time spent at HZP conditions includes temperatures of ~557°F. As noted, the accidents analyzed at hot zero power (HZP) assume an RCS temperature of 557 °F. The FSAR notes that use of a higher initial system temperature yields a large fuel-water heat transfer coefficient, larger specific heats, and a less negative (smaller absolute magnitude) Doppler feedback effect for fast reactivity addition transients like the RCCA Bank Withdrawal from Subcritical and HZP Rod Ejection events. The reduced feedback results in a higher neutron flux peak. For a Steamline Break event, starting from a higher initial RCS temperature results in a greater increase in coolant density from the cooldown. More reactivity is added due to the positive moderator density coefficient and a higher return to power results when compared with the case of a lower initial RCS temperature. Based on these considerations, a higher initial RCS temperature is conservative for the analysis of events from power. The statement that any variation in HZP temperature is unacceptable is also not consistent with the general conservative philosophy used to evaluate nuclear plant safety since only limited analyses are performed to demonstrate adequate safeguards for a range of plant conditions.

(Question 3)

TABLE 3.3-1, Item 6c

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 Q 212-24, Item 212.58. The Licensee shall evaluate and propose. Currently, this appears non-conservative.

Response: A review of FSAR Section 15.4.6 (Boron Dilution Accident) does not indicate the number of Source Range Channels required operable; however, These channels are mentioned for Refueling (MODE 6) and start up (MODE 2) Dilution Accidents. For these cases, two channels are required per Tech. Specs. Additionally, MODES 3,4, and 5 are not addressed by this FSAR Section. Boron Dilution analyses during MODES 3,4, and 5 are not part of the McGuire plant licensing basis. As such, any channel operability requirements would not be based on the FSAR analysis.

Generic Letter 85-05 dated January 31, 1985 informed licensees of the Staff position resulting from the evaluation of Generic Issue 22 "Inadvertent Boron Dilution Events". The Staff concluded that the consequences of such events are not severe enough to jeopardize the health and safety of the public. Furthermore, while NRC stated that it would "not require operating plant backfits for boron dilution events at this time, the staff would regard an unmitigated boron dilution event as a serious breakdown in the licensee's ability to control its plant, and strongly urges each licensee to assure itself that adequate protection against boron dilution events exists in its plants". McGuire personnel believe that adequate protection against boron dilution events exists and that no changes to technical specifications are warranted in this instance.

(Quest on 4a and 4b)

TABLE 3.3-2, Items 9 & 10

The T.S. specifies a response time of ≤ 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 .0 secs in Reference 5, page 7.2-14, rev. 42, item 1(e). The Licensee shall clarify.

Response: The Technical Specification limit of ≤ 2.0 seconds for the time delay of pressurizer pressure trip functions (low and high) is based upon the FSAR Chapter 15 transient analysis which assumed a delay of 2.0 seconds. The values for trip response times in chapter 7 are "typical maximum allowable time delays" and are not necessarily the same as the McGuire specific assumptions. For the sake of clarity, the values provided in chapter 7 will be revised to agree with Chapter 15 and Technical Specifications in a future FSAR update.

(Question 4c)

TABLE 3.3-2, Item 17

The proposed T.S. states that the response time requirement is NA (Not Applicable). This is incorrect since a separate Reactor Trip is an essential pa of all ESFAS functions during which safety injection is initiated. The r red 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 ≤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, e.g. reference 5, Table 7.2.1-4 (5 of 5) and related Note e on the page entitled "Notes for Table 7.2.1-4" confirms that Pressurizer 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. values.

Response: The NA enter for the required response time of reactor trip upon SI actuation is consistent with the Bases which states that trip functions not utiized in the FSAR transient analyses will have the requirement indicate not applicable in Table 3.3-2 (Reactor Trip System Instrumentation Response Times). However, as stated in Table

3.3-5 (Engineered Safety Features Response Times). The terminology in Note e, Table 7.2.1-4, should be Pressurizer Pressure-Low (Safety Injection). This wording will be corrected in a future update of the FSAR.

(Question 5a)

TABLE 3.3-3, Item 7g

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.

Response: The defeat of auxiliary feedwater pump auto-start is accomplished by depressing a switch that is interlocked with the P-11 permissive. Thus the auto-start can only be defeated below a pressurizer pressure of 1955 psig. However, the same defeat will prevent auto-start on low-low steam generator level (Table 3.3-3, Item 7c(1)). Since this auto-start capability is required in MODES 1, 2, and 3, the defeat switch is not used in these modes. Therefore the entry for APPLICABLE MODES on Item 7g is not important as it is controlled by the more limiting Item 7c(1).

The statement that P-11 can only occur in MODE 3 is not accurate. MODE 2 is defined as operation with T-avg. $\geq$350°F, K_{eff} $\geq$0.99, and power $\leq$5% RTP. Therefore, subcritical operation with T-avg. $\geq$350°F is in Mode 2 if K_{eff} is not less than 0.99. Critical operation is restricted to T-avg. $\geq$551°F, but even then the pressure-temperature operating limits permit pressures below 1955 psig. As a practical matter, pressure is maintained in the normal operating range (~2235 psig) during MODE 2. The 2# referred to in the question is retained to require that MODE 2 operation above P-11 is with the Item 7g auto-start enabled.

(Question 5b)

TABLE 3.3-3, Item 8

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

Response: This specification is consistent with other standard technical specificaions which require operator action to mitigate the consequences of a LOCA in these modes.

(Question 6a)

TABLE 3.3-4, Item 4d

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 re. g: "with a time constant of 50 secs". The current "Allowable Value" in th 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 i. clear to delete the - in -100 as the description reads: "Negative Steam Line Pressure Rate - High so that T.S. values should read as 100 psi and 110 psi. 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 RTP with a time constant of 2 seconds.

Response: Since no safety analysis limit exists for the negative steam line pressure rate setpoint (i.e., it is not assumed in transient analyses), the Setpoint Methodology (Reference 18) listed the T.S. values. The T.S. limits were revised at a later date and thus a discrepancy between the Reference 18 and T.S. values exists.

In order to correct a typographical error and adequately define the setpoint, a T.S. revision will be pursued in the following form:

	Trip Setpoint	Allowable Value
4d. Negative Steam Line Pressure Rate-High	≤100 psi with a rate/lag function time constant ≥50 seconds	≤120 psi with a rate/lag function time constant ≥50 seconds

(Question 6b)

TABLE 3.3-4, Items 7c(1) and (2)

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.]

- Reference 7 Related Section 15.4.2.2.2 concerning Main Feed Line Rupture (MFLR) under the Title of Major Assumption 10.

"The auxiliary feedwater system is actuated by the low-low Steam Generator Water Level Signal. The auxiliary feedwater system is assumed to supply a total of 450 gpm to three intact steam generators.

* Reference 5, Section 10.4.7.2.2 states that "Travel stops are set on the steam generator flow control valves such that the turbine driven pump can supply 450 gpm to three intact steam generators while feeding one faulted generator and both motor driven pumps together can supply 450 GPM to three intact steam generators while feeding one faulted generator. The Throttle positions allow all three pumps to supply a total flow of 1400 gpm to 4 intact steam generators".

* Reference 7 Related Section 15.4.2.2.2, page 15.4-13a (revision 38), states: "The single active failure assumed in the analysis is the turbine driven auxiliary feedwater pump. The motor driven pump that is headered to the steam generator with the ruptured main feedline supplies 110 gpm to the intact steam generator. The motor driven pump that is headered to two intact steam generators supplies 170 gpm to each. This yields a total flow of 450 to the intact steam generators one minute after reactor trip. At 30 minutes following the rupture, the operator is assumed to isolate the auxiliary feedline to the ruptured steam generator which results in an increase in injected flow of 80 gpm".

The sequence of events in the accident evaluation in Reference 7, Table 15.4-1 shows that after the accident is initiated at a programmed value of SG level, the low-low SG level in the ruptured SG is reached at 20 secs. later, and auxiliary feedwater [at 450 gpm] is delivered to the intact steam generators in 61 sec.

It appears, based on the above information, that on SG low-low in the ruptured SG, both the motor driven and the turbine driven pumps are initiated (with the single failure being in the turbine driven pumps). This is not in accord with the T.S. If it is assumed that low-low level in the other SGs is also reached at the same time by bubble collapse, please justify. We note that the Reactor & Turbine Control System is designed so that under normal operation, collapse of SG level on Turbine Trip will not cause a reactor trip; also at this time, main steam from intact SGs is being lost to the faulted SG so that whereas inventory is lost, a full collapse need not occur.

The proposed T.S.s Item 7c(1) and 7c(2) appear to be non-conservative in respect of accident analysis used in the Licensing Bases. The licensee shall clarify, evaluate and propose.

Response: It appears that the question is "Since one motor-driven pump supplies 110 gpm to an intact generator and the other otor driven pump supplies 170gpm to intact generators, where does the remaining 170 gpm (450 - 110 - 170), supplied to the intact generators, come from if not from the turbine-driven pump?". The new FSAR Chapter 15 analyses for optimized fuel make clear that the "two motor-driven pumps together deliver 450 gpm to the three intact steam generators allowing for spillage out of the break (Sect 15.2.8.2, page 15.2.8, 1984 Update). To clarify exactly the analysis assumption - One moter driven auxiliary feedw... ...mp supplies 110 gpm to an intact steam generator (the remainder spills out the break in the faulted loop) and the other motor driven pump supplies 170 gpm to each of the other two intact steam generators, this totals to 450 gpm.

If the failure of a motor driven pump is assumed, the turbine driven pump alone would supply at least 450 gpm to the intact loops. The turbine driven pump is actuated on low-low level in at least two steam generators. It is assumed that low-low level is reached in the other (non-faulted) steam generators as a result of the bubble collapse following turbine trip when the low-low level reactor trip is actuated from the faulted loop. This occurs because for this accident condition (i.e. not normal operation) the mass inventory in the intact steam generators is reduced significantly prior to reactor trip on low-low level in the faulted loop. The shrinkage caused by bubble collapse from this reduced mass condition would cause low-low level to be reached in the other steam generators.

(Question 6c)

TABLE 3.3-4, Item 9

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

Response: The bases for the setpoints and allowable values specified are to ensure Auxiliary Feedwater capability upon loss of power while minimizing the possible initiation of the sequence with the voltage greater than the limits of associated motors.

(Question 7a)

TABLE 3.3-5, Item 2a

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 described under Items 3a and 4a below, shows safety analyses limits of $\leq$ 12 secs (with offiste power) and $\leq$ 22 secs (without offsite 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.

Response: The entry for Table 3.3-5, Item 3a is identical to Item 2a for the loss of off site power case, i.e., each is 27 seconds. As explained in the Notes for Table 3.3-5, the difference between Item 4a and Item 2a and 3a is that 4a does not include a delay for the RHR pumps to attain their discharge pressure. This is appropriate since Item 4a deals with steam line break protection, as opposed to LOCA protection. The RHR pumps, although started for a steam line break, are not expected to deliver flow because of the higher RCS pressure. Therefore, the additional 5 second delay for these pumps to attain their discharge pressure is not relevant to ESF response time for this actuating signal.

(Question 7b)

TABLE 3.3-5, Item 2b

The descriptor (From SI), should be deleted as it is incorrect. The response time given is ≤ 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.

Response: The descriptor "(From SI)" is correct in that the allowable delay for a reactor trip due to the SI actuation signal is 2 seconds. This value is independent of the setpoint and associated delay of the initiator of SI. The reference 5, page 7.3-8 maximum time of 1.0 second is the limit on the delay associated with SI actuation upon exceeding the high containment pressure setpoint.

No credit is taken for reactor trip signal resulting from safety injection signal in any LOCA analysis. In the McGuire Unit 1 initial core large break LOCA analysis no credit is taken for reactor trip (rod insertion) at all. In the McGuire Unit 1 initial core small break LOCA a low pressurizer signal causes the reactor to trip. No credit for the control rods is taken until they are fully inserted.

(Question 7c)

TABLE 3.3-5, Item 2d

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.

Response: Following a design basis large LOCA, the isolation valve closure time depends upon the time when fuel failure occurs and fission products are released to the containment environment. The only isolation valves explicitly considered in the radiological consequences analysis of a LOCA are those in the containment purge and pressure relief lines which connect containment to the environment. For isolation valves in lines filled with process fluid a relatively long time is needed for the associated piping system to drain of fluid and expose the valve seat to the containment gases or for activity to migrate, due to the concentration gradient, through the process fluid and out the isolation valve. Hence, as long as isolation valve closure times for process lines are short (less than 1 min. per ANS 56.2) they need not be modeled in the dose calculations.

(Question 7d)

<u>TABLE 3.3-5, Item 2e</u>

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

Response: Section 15.B.2 of the McGuire FSAR considers the case of a LOCA concurrent with lower containment pressure relief. The results of the additional offsite dose due to this accident are presented in table 15.0.11-1. One of the parameters used to evaluate this case is the isolation time for the Containment Air Release and Addition (VQ) System valves which are used in venting lower containment. Table 15.B.2-1 indicates the isolation time for these valves is 4 seconds. Section 9.5.12.3 indicates that these valves auto close on a containment isolation, and that they have a 3 second closure time.

A technical specification revision to show a response time of $\leq$ 4 seconds for this item will be pursued. This would be consistent with the allowable 1 second for generating an ESF response as indicated on page 7.3-8 of the McGuire FSAR and the 3 second valve closing time as indicated above.

(Question 7e)

TABLE 3.3-5, Item 2f

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.

Response: No credit is taken for AFW flow being initiated from a Containment Pressure - High signal in analyses.

(Question 7f)

TABLE 3.3-5, Item 3a

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 from 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.

Response: This question is related to the question on Item 2a. For a steam line break the RHR pumps are not expected to deliver inventory and the additional 5 second delay for them to attain their discharge pressure is not included in the safety analysis.

(Question 7g)

TABLE 3.3-5, Item 3b

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.

Response: The descriptor "(from SI)" is correct in that the allowable delay for a reactor trip due to the SI actuation signal is 2.0 seconds. This value is independent of the setpoint and associated delay of the initiator of SI. The Reference 5, page 7.3-8, maximum time of 1.0 second is the limit on the delay associated with SI actuation upon exceeding the Pressurizer Pressure - Low setpoint.

The chapter 15 safety analyses do not take credit for a reactor trip from an SI signal initiated by low-low pressurizer. (Ref. Question 7b Response).

(Question 7h)

TABLE 3.3-5, Item 3d

The proposed T.S. is $\leq 18^{(3)}/28^{(4)}$ secs. Reference our comments and requirements under Item 2d above.

Response: Reference our response under item 2d above.

(Question 7i)

TABLE 3.3-5, Item 3e

The proposed T.S. is NA. Reference our comments and requirements under 2e. above.

Response: Reference our response under Item 2e above.

(Question 7j)

TABLE 3.3-5, Item 3f

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

Response: The only non-LOCA transient which assumes ESF actuation on Pressurizer Pressure Low-Low is the Main Steamline Depressurization event (Inadvertent Opening of a Steam Generator Safety, Relief, or Dump Valve). For this event it is conservatively assumed that auxiliary feedwater is actuated at the maximum flow rate at the initiation of the event to accentuate the cooldown. Any delay in auxiliary feedwater actuation would be beneficial and therefore a response time requirement is not applicable or appropriate.

(Question 7k)

TABLE 3.3-5 Item 4e

The proposed T.S. is NA. Reference our comments and requirements under Item 2d above.

Response: Reference our response under Item 2e above.

(Question 7l)

TABLE 3.3-5, Item 4h

The proposed T.S. value is $\leq$ 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 Steam Isolation Valves Closed." [In the case of Feedwater System Pipe Rupture].

The proposed T.S. value of $\leq$ 9 secs is therefore greater than the Safety Analysis Limit.

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

Response: Item 4h in Technical Specification Table 3.3-5 has been changed to a limit of $\leq$ 7 seconds (Ref. Amendment nos. 29 (Unit 1) and 10 (Unit 2)).

(Question 7m)

TABLE 3.3-5, Item 5a

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

Response: The response time for containment spray following a high containment pressure signal is specified at 45 seconds in the McGuire Technical Specifications. This value is consistent with the FSAR containment analysis actuation assumption as shown in FSAR Table 6.2.1-13c. Event times from the McGuire limiting case break mass/energy release analysis are reported in Table 6.2.1-29; the time of spray a. uation has no effect on the mass/energy releases calculated.

(Question 7n)

TABLE 3.3-5, Item 6b

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.

However 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 approximately 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 SG to cause water ingress into the main steam lines. Additionally, it can have consequences of potentially larger importance for the event occurring from subcritical zero power.

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

The licensee shall evaluate the related concerns, including the e ided MFW valve isolation times, to determine thei. safety significance, and ,ropose as required. Until that time, it must be concluded that since a zero (0) value has been used in the current analysis, 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.

Response: Excessive Feedwater Flow at Full Power is analyzed in Section 15.1.2 of the McGuire FSAR. Table 15.1.2-1, page 1 of 2, 1984 Update, gives the sequence of events for this analysis. The High-High SG Level setpoint is reached at 27 seconds with feedwater isolation occurring 9 seconds later. This 9 second value agrees with the values used for feedwater isolation on Safety Injection.

To be consistent with the current safety analysis the Technical Specifications value for item 6b of Table 3.3-5 should be ≤ 9 seconds. Another alternative is to reanalyze the Excessive Feedwater Flow event with the longer delay time. Duke will pursue a Technical Specification revision or reanalysis.

(Question 7o)

TABLE 3.3-5, Item 12

Response time proposed as ≤ 60 secs.

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

Response: The automatic switchover to recirculation is initiated when the level setpoint in the RWST is reached. The setpoint determination includes allowances for switchover delay ≥ 60 seconds and plant procedures test to ensure switchover delay ≤ 60 seconds per Table 3.3-5, Item 12.

General Response to Questions 8a-8e:

These questions in general deal with the conservatism of the FSAR Chapter 15 safety analyses for events initiated from MODES 3-5. Specifically the question of the number of RCS loops in operation, for heat removal or other purposes, appears many times. Since the McGuire Technical Specifications and Westinghouse Standard Technical Specifications are identifical for MODES 3-5 for T.S. 3.4.1, Reactor Coolant Loops and Coolant Circulation, any questions regarding these matters should be resolved on a generic basis and are not specific to McGuire. Therefore, the responses to each question will deal only with items which are specific to McGuire.

(Question 8a)

SECTION 3/4.4.1, 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

Setpoint (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, page 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).

Response: No McGuire specific concerns are raised in this question. Refer to the general response to Questions 8a-8e.

(Question 8b)

<u>SECTION 3/4.4.1, G.2.6.2 STEAM LINE BREAKS</u>

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 212-47d & 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 (hot 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, a MSLB cools the RCS down to 212°F e.g., the residual shutdown margin will be 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 there is no Safety Injection, no Reactor Trip (on SI), and under the existing 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 is Not Applicable.

Given the circumstances of the proposed T.S., the 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 depressurization 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).

Response: Changes in the Technical Specifications and plant procedures have occured since the DPO questions were posed (boration to cold shutdown prior to starting cooldown is no longer required). The required shutdown margin for RCS temperatures above 200°F is 1.3% Δk/k. The shutdown margin requirement for temperatures equal to or less than 200°F is 1.0% Δk/k. Variations in initial conditions for the steamline break transient were analyzed in WCAP-9226 and support the conservative assumptions in the FSAR analysis.

Closure times for the Main Steam Isolation Valves (MSIVs) are implied in the Technical Specifications. In Table 3.3-5, Items 4h, 5c, and 8, response times are given for the Steam Line Isolation function. This time includes the MSIV closure time. Other concerns raised in this question are generic. Refer to the general response to Questions 8a-8e.

(Question 8c)

SECTION 3/4.4.1, G.2.6.3 LOSS OF PRIMARY COOLANT

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 Q212-64, 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 Standby 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 immediately 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 analyses for W 4 loop plants the reactor coolant pumps are assumed to be tripped at the initiation 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 coastdown is important in 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 shutdown 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 nonconservative 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, the Power Range Neutron Flux, High Setpoint 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 No 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 nonconservative 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 Standby; in Item 212.90 (6.3), pa. 212-61, for a LOCA at and less than 1000 psig/425° in Hot Standby, 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 paragrapn 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 fewer than 4 RCS loops in MODE 3 would be in nonconformance with the Licensing Bases FSAR in a nonconservative manner, and the licensee is required to evaluate and propose.

Response: No McGuire specific concerns are raised in this question. Refer to the general response to Questions 8a-8e.

(Question 8d)

SECTION 3/4.4.1, G.2.6.4 OCCURRENCES CAUSING AN INITIAL INCREASE IN 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 evaluation at zero power.

b) Startup 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 nonconformance with the current Safety Analyses Limits in a nonconservative manner. The licensee is required to evaluate and propose.

Response: No McGuire specific concerns are raised in this question. Refer to the general response to Questions 8a-8e.

(Question 8e)

T.S. 3.4.1 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 RCPs (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 steamline break, and LOCA events, small and large, down to 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.

Response: No McGuire specific concerns are raised in this question. Refer to the general response to Questions 8a-8e.

(Question 9)

T.S. Page 3/4 4-2

Earlier concerns under General 2.6.1 addressed the need to evaluate the consequences of the startup 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 b. states:

> "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 natural circulation. This proposal is nonconservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

Response: The actions included in ACTION b. are 1) suspend deboration operations and 2) immediately initiate action to restore forced circulation. The actions are obviously valid responses to the condition. There is no Emergency Operating Procedure at McGuire for natural circulation. There is Abnormal Procedure AP/1&2/A/5500/09, Plant Operations During Natural Circulation, which addresses the initiation, verification, and maintenance of natural circulation. This procedure would be implemented under this condition.

(Question 10)

T.S. Page 3/4 4-3

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 T.S. 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 nonconservative 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 the 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 sturation 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 necessry 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 212-47d 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 SG water level at the Safety Analysis Limit for the Licensing Basis, which is the no-load programmed level, not the current proposed T.S. valve which is the S.G. Low-Low Level [Reactor Trip] and AFW actuation. This proposed T.S. is nonconservative 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, 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 that 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 nonconservative 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 nonconservative 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 nonconservative 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 SG 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].

Response: Several separate questions are raised here. The McGuire specific ones are answered as follows:

1) Each RHR train is powered from a separate 4160V bus in the Essential Auxiliary Power System. Each reactor coolant pump is powered from a separate 6900V bus in the Normal Auxiliary Power System.

2) It should be noted that the requirement of maintaining a specific level in the steam generator to verify operability was imposed by the NRC and has no firm basis within Westinghouse. However, for an RCS loop to be operable, sufficient inventory is required in the secondary side for heat removal. In MODE 4 this can be assured by keeping the tube bundle covered. A reasonable way of ensuring this is to require that the secondary side level indicates within the narrow range span. Accounting for errors, an indicated level at the low-low level setpoint assures that the level is at least at the bottom of the narrow range span.

The safety analysis limit for reactor trip on lo-lo SG level is a function with a value of 0% at no-load conditions. Adding allowances for reference leg heatup and instrument error gives the value of 12% used as the T.S. trip setpoint. The T.S. value is therefore conservative with respect to the safety analysis limit.

3) The low flow alarms on the RHR loops are to alert the operator to insufficient flow under RHR conditions. They have no relation to the low flow reactor trip which inserts the control rods to control reactivity during low flow conditions at power. Boron is employed for reactivity control in the shutdown modes while rod insertion is impossible (if the rods are already inserted) or unnecessary (because of the boration).

The current surveillance 4.4.1.3.3 requires verifying one RCS or RHR loop in operation at least every 12 hours. The concern raised apparently centers around the assertion that core cooling could be lost without the knowledge of the operator since no protective functions or alarms are required to be operable by the technical specifications. However, it is expected that there would be multiple indications of any problems that could cause a loss of coolant loop. Although the appropriate alarms are not required by the technical specifications to be operable, there is no reason to believe that all relevant alarms and other indicators would be inoperative during this mode.

The other issues raised in this question are not specific to McGuire. Refer to the general response to Questions 8a-8e.

(Question 11a)

T.S. SECTION 3/4.5

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.

Response: This "question" is merely a statement of operator action to align RHR. It remains true and requires no response.

(Question 11b)

T.S. 3.5

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.

Response: This "question" is merely a statement of operator action to align the ECCS for use from a shutdown condition. It remains true and requires no response.

(Question 11c)

T.S. 3.5

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 hydrostastic 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.

Response: This "question" is largely a quotation from the FSAR. The last two paragraphs, while not from the FSAR, are simply statements introducing a quotation from the SER. Therefore, this requires no response.

(Question 12a)

T.S. 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 setpoint of approximately 450 psig? The specified setpoint 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.

Response: The bases for the T.S. 3.5.1 limit of Cold Leg Accumulator cover pressure of between 400-454 psig is the assumed value in the LOCA analysis (FSAR Chapter 15). Allowance for channel error and drift are accounted for in the determination of the T.S. requirements. The numbers in Table 6.3.2-1 are nominal and minimum values as required by T.S. 3.5.1 and are in agreement with the T.S. 3.5.1 limits. Recent Technical Specification changes (Ref. unit 1/2 License Amendments 57/38) associated with the removal/isolation of the UHI System involve revising the Cold Leg Accumulator cover pressure to between 585 and 639 psig.

(Question 12b)

T.S. 4.5.1.1.1.d.1

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.

Response: The Cold Leg Accumulators Relief Valves (NI-52, 63, 74, and 86) are not required to perform a safety function either to shutdown the reactor or to mitigate the consequences of an accident. The inservice testing program requirement to test all class 1, 2, & 3 valves was changed to valves which are required for safe shutdown of the reactor or mitigating the consequences of an accident.

Consequently these relief valves are not included in the McGuire Nuclear Station pump and valve inservice testing program required by 10 CFR 50.55a(g). These valves (and setpoints) are tested following maintenance only.

(Question 13)

T.S. 3.5.1.2.d

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 T.S. 4.5.1.2.a. Its absence from the proposed T.S. renders it potentially non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

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

Response: FSAR Table 6.3.2.1 provides the expected operating temperature range for the UHI accumulator water and not Safety Analysis limits as stated above. The Safety Analysis value related to UHI water temperature is assumed to be the upper bound value of 100°F.

The Upper Head Injection Accumulator Relief Valve (NI-279) is not required to perform a safety function either to shutdown the reactor or to mitigate the consequences of an accident. The Inservice Testing Program requirement to test all class 1, 2, & 3 valves was changed to valves which are required for safe shutdown of the reactor or mitigating the consequences of an accident. Consequently this relief valve is not included in the McGuire Nuclear Station pump and valve inservice testing program required by 10CFR 50.55a(g). This valve (and setpoint) is tested following maintenance only.

(Question 14)

T.S. 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 during accident occurrences.

Response: The bases for the limits as specified in T.S. 4.5.2.h are the assumed ECCS flows used in the LOCA analysis. ECCS flow injected to the broken cold leg is assumed to spill in LOCA analyses, so limits are placed on the branch line totals to ensure that adequate flow reaches the intact loops.

(Question 15)

T.S. SECTION 3/4.5.3

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 LCOs 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 overpressure 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.

Response: This question appears to be related to the discussion of FSAR Section 5.2.2, "Overpressurization Protection". Although it is stated in two places that Technical Specification 3.5.3.a violates the FSAR Licensing Basis, Section 5.2.2 contains no discussion of ECCS pump operability between 300°F and 350°F. It is further stated, in the discussion of Section 5.2.2., that the McGuire Technical Specification 3.5.3.a. differs markedly from the Westinghouse Standard Technical Specification 3.5.3.a. Comparing the two we find no differences in the number or type of ECCS pumps required to be operable or inoperable. The McGuire lower limit is 300°F compared with Standard lower limit of 275°F. We therefore conclude that the McGuire Specification does not differ from the Standard one in a non-conservative manner.

(Question 16)

T.S. 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 hours 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 operability down to and including MODE 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 according to design flow requirements throughout the occurrence.

The licensee should explain how the proposed T.S. ensures that the turbine driven pump maintains its flow performance required by accident analyses 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 analyses, 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 Condition III and especially Condition IV occurrences of the Licensing Basis, and if the existing accident analyses remain valid.

Response: The footnote deleting operability requirements for the Steam Turbine-Driven Auxiliary Feedwater Pump (TDAFP) at steam pressures <900 psig was added in an attempt to correct a conflict between the LCO with its applicability of Modes 1, 2, and 3 and Surveillance Requirement 4.7.1.2.a.2 which defines operability of the TDAFP as developing a discharge pressure of $\geq$ 1210 psig at a flow of $\geq$900 gpm when the secondary steam supply pressure is >900 psig (to delevlop a discharge pressure of 1210 psig the TDAFP requires steam at $\geq$ 900 psig, but supply steam pressure can be <900 psig during startups/shutdowns). The Technical Specification's bases for operability of the Auxiliary Feedwater System is to ensure that the Reactor Coolant System can be cooled down to <350°F from normal operating conditions in the Event of a total loss of offsite power, with the TDAFP capable of delivering a total feedwater flow of 900 GPM at a pressure of 1210 psig to the entrance of the Steam Generators to meet this function. Under normal operating conditions source steam at >900 psig is Available and the TDAFP is capable of performing this function. However, as indicated in Question 16 and Items 1 and 2 below, the TDAFP is also required with steam pressures <900 psig.

1. During a condition IV feedline break all steam generators will depressurize prior to closure of the Main Steamline Isolation Valves (MSIV's). The low steamline pressure set point for closing the MSIV's is about 585 psig. However, errors due to seismic and environmental conditions as well as instrumentation inaccuracies may result in a steam generator pressure as low as 285 psig prior to MSIV closure. Therefore the turbine driven Auxiliary Feedwater pumps must be capable of delivering the minimum required flow for feedline break with a steam generator motive supply pressure as low as 285 psig.

2. The ability to commence a plant cooldown must be maintained following transient and accident conditions. Following design basis faulted conditions with specific single failure assumptions, it may be necessary to commence a plant cooldown with only a turbine driven Auxiliary Feedwater System pump available. Consequently the turbine driven pump must be capable of delivering the minimum required flow for cooldown with a steam generator motive supply pressure as low as 100 psia corresponding to a primary side hot leg temperature of 350°F during a natural circulation cooldown, which is maximum operating temperature for Residual Heat Removal System Operation.

Therefore, The Tech. Spec's Surveillance requirements/Bases do not adequately define the operability requirements for the TDAFP and consequently the Technical Specification does not ensure operability of the TDAFP over the range of conditions expected from Design Basis Accident Analysis and other less bounding events. All other circumstances (or accident conditions) besides the limiting condition of loss of Offsite Power during full power operation pose less severe demands on the TDAFP. For the Main Steamline Break, the intact Steam Generator is fully capable of supplying the steam requirements of the pump turbine. With source steam < 900 psig the TDAFP is capable of providing feed flow but at a discharge pressure below 1210 psig. Since the McGuire Technical Specification is essentially indentical to the Westinghouse Standard Technical Specification (with the exception of the "correcting" footnote), this discrepancy between the LCO and the Surveillance Requirements/Bases should be resolved on a generic basis and is not specific to McGuire.

With regard to providing operability down to and including Mode 4 (and availability from Mode 5), the bases of the auxiliary Feedwater System Technical Specification is that its operability (including the capacity of the TDAFP) ensures that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System Temperature to <350°F (i.e. Mode 4) when the RHR System may be placed into operation. Therefore the bases does not require System Operability in Modes 4 or 5. Since the McGuire and Westinghouse standard technical specifications bases are essentially identical, any desired changes to this bases should be pursued on a generic basis.

Item 1e of T.S. Table 3.3-3 "Steam Line Pressure-Low" is derived from steam line sensors downstream of the steam generator flow restriction orifices.. The steam flow restrictors do not cause a significant pressure drop except during a double ended steam line break. The blowdown phase of this accident lasts only a few seconds. The accurate pressure sensing in the steam lines (i.e. generation of a "Steam Line Pressure-Low" signal) takes less than 2 seconds and steam line isolation less than 7 seconds. (The main steam line break accident is descussed in Chapters 6 and 15 of the FSAR).

(Question 17)

T.S. SECTION 3/4.7.5

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.

Response: This action has been deleted. See Section 9.2.2, Nuclear Service Water System and Ultimate Heat Sink, 1984 Update.

(Question 18)

T.S. 3/4.9.1

The current SER, Supplement No.1, reference 11, page 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.1NV-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 1NV-250 is to be locked closed during refueling. The current position could be nonconservative 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 1NV-171A and 1NV-175A 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.

Response: Valve 1NV-250 is specifically required to be locked closed under the Administrative Controls (i.e. Station Procedures). This Valve is upstream of valves 1NV-171A and 1NV-175A and isolates the flow path.

(Question 19)

T.S. SECTION 3/4.9.8

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 in the FSAR. The licensee shall evaluate and propose.

Review of available responses to the consequences of a fail closed RCS/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.

Response: The McGuire Technical Specification 3.9.8 is the same as the Westinghouse Standard Technical Specification (STS) 3.9.8. Since there is nothing unique about McGuire's 3411 MWt power level, its decay heat characteristics, or its 23 feet level requirement, this question should be addressed on a generic basis.

(Question 20)

T.S. SECTION 4.9.8.2

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.

Response: See the response to the previous item since McGuire is also in accordance with Westinghouse Standard Technical Specification on this item.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 15 1989

MEMORANDUM FOR: James H. Sniezek, Deputy Director
Office of Nuclear Reactor Regulation

FROM: Robert B. A. Licciardo, Reactor Engineer (Nuclear)
Plant Systems Branch
Division of Engineering and Systems Technology
Formerly: Reactor Systems Branch
Division of Systems Integration

566 RA

SUBJECT: CLOSURE OF OUTSTANDING TECHNICAL SPECIFICATION CONCERNS DERIVING FROM R. LICCIARDO'S DPO REVIEW OF THE MCGUIRE TECHNICAL SPECIFICATIONS

During a meeting with the NRR/Plant Systems Branch in 1988 to discuss Branch and individual concerns, the writer informed you of his concerns for the delayed closure of Technical Specification deficiencies arising from his Differing Professional Opinion (DPO) Review of the McGuire Technical Specifications. As a consequence, you directed the writer to forward to the Chief, Technical Specifications Branch (TSB), for implementation, those items of that review which had already been confirmed by NRR for incorporation into the W Standard Technical Specifications (W STS). Enclosure 1 is a copy of the memorandum that forwards these items to the TSB for that purpose, to the subject: Incorporation Of Items Into W STS Deriving From NRC Confirmation Of Generic And Multiplant Actions From R. Licciardo's DPO Review Of The McGuire Technical Specifications.

Of the original 380 concerns (also contained in Enclosure 1), 220 items were originally selected by the Division of Systems Integration/Reactor Systems Branch (DSI/RSB) for review by the Division of Licensing (DL) and in the enclosed document 90 are finally categorized as Generic, and an additional 12 as Multi Plant Action (a sub-category of Generic). Additionally, 45 Plant Specific items remain for final review, but these are not separately identified in this document at this time. Implementation of your direction will close out the Generic and Multiplant Action Items.

As a result of the NRC open door policy meeting the writer had in 1985 with Dr. Nunzio J. Palladino, then Chairman of the NRC, Dr. Palladino wrote a letter to the U.S. Congress dated May 17, 1985 (Enclosure 2) in which he promised an accelerated closure of all outstanding items of the McGuire DPO review before the end of 1985 if the writer's concerns were confirmed. Because of this commitment and the generic lack of licensing basis safety in Westinghouse (W) facilities (including McGuire Units 1 and 2) deriving from the current open status of these items, I now ask that priority be given to their final closure including completion of the remaining 45 Plant Specific items for each of the McGuire units for which responses have been available from the licensee since June 10, 1986 (Enclosure 3). Further, because of the

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writer's manifest detailed experience and maturity of judgement that resulted in the initiation and final substantive confirmation of the issues of this DPO, he offers his services to facilitate the early closure of these items with appropriate management consent.

R. Licciardo

Robert B. A. Licciardo,
B. Mech. E; B. Comm.

Professional Nuclear Engineer:
No. NU 001056 (California)

Professional Mechanical Engineer:
No. M015380 (California)

Enclosures:
1. Memo from Robert B. A. Licciardo to Edward Butcher, dated March 15, 1989.
2. Letter from Dr. Nunzio J. Palladino (Chairman, NRC) to Hon. E. J. Markey (U.S. House of Representatives), dated May 17, 1985.
3. Letter from H. B. Tucker (DPCo) to H. R. Denton, dated June 10, 1986.

cc w/enclosures:
Chairman Zech
Commissioner Roberts
Commissioner Carr
Commissioner Rogers
Commissioner Curtiss
SECY
OPE
OGA
CA
V. Stello
T. E. Murley
S. A. Varga
G. C. Lainas
J. W. Craig
R. Licciardo
D. Hood

writer's manifest detailed experience and maturity of judgement that resulted in the initiation and final substantive confirmation of the issues of this DPO, he offers his services to facilitate the early closure of these items with appropriate management consent.

Original signed by

Robert B. A. Licciardo,
B. Mech. E; B. Comm.

Professional Nuclear Engineer:
No. NU 001056 (California)

Professional Mechanical Engineer:
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Enclosure 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 15 1989

MEMORANDUM FOR: Edward Butcher, Chief
Technical Specifications Branch
Division of Operational Events Assessment

FROM: Robert B. A. Licciardo, Reactor Engineer (Nuclear)
Plant Systems Branch
Division of Engineering and Systems Technology

SUBJECT: INCORPORATION OF ITEMS INTO W STS DERIVING FROM NRC CONFIRMATION OF GENERIC AND MULTI-PLANT ACTION CONCERNS FROM R. LICCIARDO'S DPO REVIEW OF THE MCGUIRE TECHNICAL SPECIFICATIONS

During a meeting with James H. Sniezek, the Deputy Director for NRR asked the writer to forward to your branch for implementation, those items of the writers DPO review of the McGuire Technical Specifications already confirmed for incorporation into the Westinghouse Standard Technical Specifications (W STS). The subject information is contained within the attachment entitled "Identification of Generic Items Confirmed for Westighouse Standard Technical Specifications." Of the original 380 concerns, 220 items were selected by Division of Systems Integration/Reactor Systems Branch (DSI/RSB) for review by Division of Licensing (DL) and in the attached document 90 are finally categorized as Generic, and 12 as Multiplant Action. Residual plant specific items numbering 45 are not identified in this document at this time.

The writer's review of the McGuire Technical Specifications (see Reference 1 in the attached List of References), reported 380 items of concern. Reactor Systems Branch (RSB) subsequently selected 220 of these for verification and these were sent to the Division of Licensing (DL) for that purpose (Ref. 2). DL responded to this request by memo dated May 28, 1985 (Ref. 3).

In its review, (Ref. 3) DL categorized three groups of conclusions: "Generic" (G), "Plant Specific" (PS), and "Closed" (C). The Generic items were referred to DSI/RSB for consideration for incorporation into the next periodic update of the W STS in accordance with the provisions of NRR Office Letter No. 38. The PS Items were to be forwarded to the licensee (Ref. 5), and upon their response (Ref. 6), DL was to work with appropriate branches to achieve their resolution.

The generic items ultimately arising out of the review are identified in the attachment. The original generic conclusions of DL are marked as G. Subsequent review by the writer and B. Sheron, Chief, DSI/RSB, of the original dispositions by DL, resulted in a transfer of a number of the items from the C and PS categories to the Generic category, including a new Multiplant

Action (MPA) sub-category. These are identified as G (RSB) and MPA (RSB) respectively. A number of additional Generic items arise from the joint response to the PS concerns by the licensee and Westinghouse under Reference 6, and these are marked as G (W).

R. Licciardo

Robert B. A. Licciardo, Reactor Engineer (Nuclear)
Plant Systems Branch
Division of Engineering and Systems Technology

Attachments:
As stated

cc: T. Murley
J. Sniezek

List of References

1) Memorandum for Brian W. Sheron, Chief, Reactor Systems Branch, Division of Systems Integration, from Robert B. A. Licciardo, Nuclear Engineer, Subject: "Review of McGuire Technical Specifications," dated June 11, 1984.

2) Memorandum for Darrell G. Eisenhut, from Robert M. Bernero, "Concerns on McGuire Technical Specifications," dated August 30, 1984.

3) Memo for Robert M. Bernero, Director, Division of Systems Integration from Hugh L. Thompson, Director, Division of Licensing, Subject: Disposition of Concerns Raised by R. Licciardo in His DPO on the McGuire Technical Specification, dated May 28, 1985.

4) Letter from Nunzio J. Palladino, Chairman, USNRC, to the Honorable Edward J. Markey, Chairman Subcommittee on Energy and Commerce, U.S. House of Representatives, dated May 17, 1985.

5) Letter to H. B. Tucker (Duke Power Company) from Thomas M. Novak (DL), Subject: "Request for Comments on McGuire Technical Specification Concerns Resulting From Differing Professional Opinion," dated July 9, 1985.

6) Letter to H. R. Denton (NRC) from H. B. Tucker (DPCo) on Subject: "NRC DPO Concerns on McGuire Technical Specification," dated June 10, 1986.

IDENTIFICATION OF GENERIC ITEMS CONFIRMED

FOR WESTINGHOUSE STANDARD TECHNICAL SPECIFICATIONS

BASED ON THE DETAILED REVIEW OF

THE "PROOF & REVIEW" COPY

OF

MCGUIRE UNITS 1 & 2: PROPOSED TECHNICAL SPECIFICATIONS

PREPARED BY

Robert B. A. Licciardo

Reactor Engineer (Nuclear)

United States Nuclear

Regulatory Commission

Date: March 9, 1989

The generic items are identified by marginal marking in this attachment which is a copy of the original DPO "Review of McGuire Technical Specifications" of June 11, 1984. The original generic conclusions of the Division of Licensing (DL) are marked as G. Subsequent review by the writer and B. Sheron, Chief, Division of Systems Integration/Reactor Systems Branch (DSI/RSB), of the original dispositions by DL, resulted in a transfer of a number of these items (from the closed (C) and Plant Specific (PS) categories) to the Generic category, including a new Multiplant Action (MPA) sub-category. These are identified as G (RSB) and MPA (RSB) respectively. A number of additional Generic items arise from the joint response to the PS concerns by the licensee and Westinghouse under Reference 6, and these are marked as G (W).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 1 1 1984

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

FROM: Robert B. A. Licciardo
Nuclear Engineer
Reactor Systems Branch
Division of Systems Integration

SUBJECT: REVIEW OF MCGUIRE TECHNICAL SPECIFICATIONS

REFERENCE: a) Memo from Harold R. Denton, Director
Office of Nuclear Reactor Regulation
for Darrell G. Eisenhut, Director
Division of Licensing and
Roger J. Mattson, Director
Division of Systems Integration
on the Subject: DIFFERING PROFESSIONAL OPINION OF MR. LICCIARDO REGARDING MCGUIRE TECHNICAL SPECIFICATION and dated: March 21, 1984

b) Memo from Brian W. Sheron, Chief, RSB, DSI to Robert Licciardo RSB, DSI dated April 11, 1984 on the Subject: MCGUIRE TECHNICAL SPECIFICATIONS ASSIGNMENT

I reference your memo to reference b) requesting review of the McGuire Technical Specifications to an acceptable format, in response to the requirement of reference a) for a coordinated review of the concerns arising from the writer's earlier DPO.

Please find attached copy of a document entitled "McGuire Units 1 & 2: Proposed Technical Specifications; Review of Proof and Review Copy," which is in response to your request.

The review is composed of two sections. The first section is entitled "Pre Review Information" which details the Basis, Purpose and Resources, Schedule, Evaluation Method, Regulatory Requirements and Licensing Consequences of the Review. The second section contains the Detailed Review.

Since the staff required this detailed review to be conducted without any formal, or substantive informal discussion, both within and without RSB, I presume that it is to be used as a basis for the coordination stated in Harold R. Denton's letter to reference a), namely that "The Division of Systems Integration, in coordination with DL, shall have people that are knowledgeable about the technical subjects raised by Mr. Licciardo, the standard technical specifications, and the McGuire technical specifications review the broad technical subjects and subgroups raised in the DPO." The

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SURNAME ▸							
DATE ▸							

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writer considers that such a coordinated review including constructive critique is an essential consequence of any such document. The writer also believes that such construction must be developed on the basis of responsible written and signed comment within the Regulatory Framework. The writer would be pleased to participate in this coordination as required.

The writer is aware that RSB staff has received copies of the writer's initial proposed memo to T. M. Novak from R. W. Houston on the subject of: "STAFF REVIEW OF PROOF AND REVIEW COPY OF PROPOSED TECHNICAL SPECIFICATIONS FOR MCGUIRE UNITS 1 & 2" dated 06/15/83, and through this action is pleased to have made an early contribution to recent reviews of Technical Specifications for Operating License Applications.

Further, the writer has been informed that the above referenced memo (of 06/15/83) was also provided to Westinghouse (W) and notes two subsequent developments of significance:

1) In response to a question from M. Wigdor concerning "Vogtle," on "Cold Overpressure Mitigation", W has now recently submitted a Topical report entitled "Cold Overpressure Mitigating Systems," dated February 1984, for review by NRC.

2) W has recently reviewed its position on Reactor Coolant System (RCS) Operability requirements in MODE 3 and from this has determined the need for additional operable RCS pumps over those required in the W STS for the case of "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition."

Both of the above items 1) and 2) were the subject of specific concern in the referenced memo proposed by the writer, and it is encouraging to note the early response by W to those safety issues.

R. Licciardo

R. B. A. Licciardo

Attachment: As stated

cc: H.R. Denton
R. Mattson
R.W. Houston w/attachment
N. Lauben w/attachment

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MCGUIRE UNITS 1 & 2: PROPOSED TECHNICAL SPECIFICATIONS

REVIEW OF "PROOF & REVIEW COPY"

Prepared By

ROBERT B. A. LICIARDO

Nuclear Engineer

RSB/DSI/RSB

Date: June 12, .9[illegible]

~~8406210191~~ XA 141 pp

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APPENDIX A: Technical Specifications - Selected Relevant Regulations

INTRODUCTION

By letter to reference 1), the licensee proposed Technical Specifications for McGuire Unit 2 which were to be an integral part of the Operating License.

The Licensee also proposed that these same Technical Specifications include detailed references to Unit 1 in a manner which did not impede its effective use for Unit 2 but which would enable its use for Unit 1 at a later date. The Licensee considered an ultimate position in which both McGuire Units 1 and 2, would use the same Technical Specifications, with marginal adaptations. The application of these Technical Specifications to Unit 1 was achieved by application for a proposed, and issuance of a subsequent, licensing amendment at a later date.

The Proof and Review copy which has been reviewed by the writer comprises a Westinghouse Standard Technical Specification, Revision 4, which had been marked up by the Licensee as a proposal for Units 2 (and 1). This mark up was further reviewed by SSPB for conformance to the Westinghouse Standard Technical Specifications, and, by mutual agreement between the Licensee, NRR/DL and SSPB, subsequent changes had been made. This subsequent document presented to RSB for review, contained no record of, or, safety evaluation reports on, these changes which had been made including any relationship to the then existing McGuire Unit 1 Technical Specification and the Final Safety Analysis Reports, or the Safety Evaluation Reports, for McGuire Units 1 & 2.

The writer has conducted the RSB portion of the review by a more detailed examination of those sections and related systems which are its primary responsibility as defined by the Standard Review Plan. These sections have been reviewed against the information in the Final Safety Analysis Report, the related Safety Evaluation Reports and additional information, as contained in references 1 through 29.

The items reviewed are listed in Table 1 and the pages affected are listed in Table 2.

PRE REVIEW INFORMATION

Basis of Review

The starting basis for this review was the proposed memo to T. M. Novak from R. W. Houston dated 6/15/83, on the subject of: "Draft Review Of Proof and Review Copy Of Proposed Technical Specifications For McGuire Units 1 & 2."

The Proof and Review Copy of the Proposed Technical Specifications For McGuire Units 1 and 2 from which the material for review by RSB was extracted, was attached to a memo from C. O. Thomas (SSPB) to Brian W. Sheron (RSB) on the subject of "Proof and Review of McGuire - Units 1 and 2, Technical Specifications" and dated January 14, 1983.

Purpose of Review and Resources

The purpose of this review has been to enable a document which could be used to serve the purpose of the request by Harold R. Denton in Reference a) namely:

> "The Divison of Systems Integration, in coordination with DL, shall have people that are knowledgeable about the technical subjects raised by Mr. Licciardo, the standard technical specifications, and the McGuire technical specifications review the broad technical subjects and subgroups raised in the DPO."

For this purpose, RSB asked the writer to identify the specific disparities of his concern, and his basis for them. Commencement of the task, as described under the section on "Schedule and Resources," disclosed more items of concern. To facilitate the preparation of a set of information within a time frame consistent with the proposed purpose and schedule, the writer was asked by RSB to complete his task with minimal interchange both within and without RSB. This document presents the best evaluations by the writer under these conditions and must be considered as a starting basis for the follow-on coordinated review required from reference a).

The writer wishes to acknowledge that during this review he has received the benefit of active discussions with ICSB personnel, namely T. G. Dunning, Section Leader, and F. Burrows, Reactor Engineer (Instr), on clarifying significant aspects of Plant Instrumentation Logic. The responsibility for interpretation and conclusions in this document remains the writer's.

Schedule

The starting basis for this review was the writer's proposed memo to T. M. Novak from R. W. Houston on the subject of Staff Review of Proof and Review Copy of the Proposed Technical Specifications for McGuire Units 1 & 2.

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By memo to reference a) dated March 21, 1984, Harold R. Denton required that: "The Division of Systems Integration, in coordination with DL, shall have people that are knowledgeable about the technical subjects raised by Mr. Licciardo, the standard technical specifications, and the McGuire technical specifications review the broad technical subjects and subgroups raised in the DPO. As soon as the review approach is selected, you are to provide me with a brief plan that describes how you plan to conduct the review, who is involved and your schedule for concluding the review. You should plan to document your review not later than July 1, 1984 or provide a status report with a schedule by May 15, 1984."

Commencing week ending March 31, 1984 the writer was asked by B. W. Sheron, Branch Chief, to develop a series of questions in accordance with his later memo of April 11, 1984 for completion by April 27, 1984.

On commencing this task, an audit was taken on other issues within the T.S. which had not received detailed attention because of relative priorities and the probabilities that because of the relatively simple nature of the related operations, that the T.S. would be complete and accurate. This audit revealed that such was not the case and that relatively complex safety issues resided in many locations of lesser perceived importance including footnotes, and descriptions in the Basis, attached to the T.S. These concerns have required a near item by item check to ensure a maximum of surety. The schedule has been extended on that basis but the need for closure has left a certain minimal area of unconfirmed concern.

However, the above approach should now convince the licensee of his primary responsibility to ensure the accuracy and completeness of the Technical Specifications including a final detailed check and evaluation of not only the items that are covered above, but residuals in the area of unconfirmed concern for RSB.

Evaluation Method

The evaluation has focused on the requirements of the process systems to meet Condition 1 Occurrences under normal operation in MODES 1 through 6. It has also focused on the capability of these same systems, and their protection systems [both Reactor Trip and Engineered Safeguards Features] to be available and to perform in accordance with acceptable calculated consequences of Condition II, III and IV Occurrences, and other (Licensing Basis) events, as identified and evaluated in the Licensing Basis for MODES 1 through 6.

The term "evaluate," used throughout this review as e.g., in the phrase "The licensee shall evaluate and propose" is to be interpreted as synonymous with the term "Safety Evaluation" as used in 10 CFR and includes the requirement to submit such an evaluation in response to related circumstances.

The term "propose" is also synonymous with the term "propose" as used in 10 CFR 50.34(b)(6)(vi) "Proposed Technical Specifications prepared in accordance with the requirements of §50.36" and 10 CFR §50.59 "Changes, tests and experiments" in respect of "proposed change, test or experiment."

Regulatory Requirements

To facilitate ready reference, a set of "Selected Relevant Regulations" is provided in Appendix A, of which the following is a brief summary:

10 CFR 50.36 "Technical Specifications." This defines the principal Requirements which will be included in the Technical Specifications. These include:

10 CFR 50.36(c)(1)	"Safety limits, limiting safety system settings and limiting control settings."
10 CFR 50.36(c)(2)	"Limiting conditions for operation"
10 CFR 50.36(c)(3)	"Surveillance requirements"
10 CFR 50.36(c)(4)	"Design Features"
10 CFR 50.36(c)(5)	"Adminstrative controls"

10 CFR 50.11 "Exceptions and Exemptions from Licensing Requirements"

10 CFR 50.12 "Specific Exemptions"

These two Regulations define the basis for granting exemptions from the requirements of 10 CFR.

10 CFR 50.34 "Contents of Applications: Technical Information"

This provides the regulatory basis for

a) Necessary descriptions of the facility and the need for related Safety Evaluations for both the PSAR and the FSAR.

b) Within the PSAR, an identification and justification for the selection of those variables, conditions, or other items which are determined as the result of preliminary safety analysis and evaluation to be probable subjects of technical specifications for the facility, with special attention given to those items which may significantly influence the final design. Reference 10 CFR 50.34,(a)(5).

c) Within the FSAR, proposed technical specifications prepared in accordance with the requirements of §50.36. Reference 10 CFR 50.34(b)(6)(vi)

10 CFR 50.57 "Issuance of Operating License"

The particular relevant subsections are:

10 CFR 50.57(a)(1) - This ensures that the facility has been substantially constructed, in conformity with the construction permit and the application as <u>amended</u>.

10 CFR 50.57(a)(2) - which requires that "The facility will operate in conformity with the application as <u>amended</u>,..."

10 CFR 50.57 (b) - "Each operating license will include appropriate provisions with respect to any uncompleted items of construction and such limitations or conditions as are required to assure that operation during the period of the completion of such items will not endanger public health and safety."

10 CFR 50.59 "Changes, Tests and Experiments"

Sections of particular relevance are:

10 CFR 50.59(a)(1) - This permits changes from the FSAR providing they

- involve no change in the Technical Specification
- do not involve an unreviewed safety question.

10 CFR 50.59(a)(2) - Defines an unreviewed safety question.

10 CFR 50.59(b) - Requires the licensee to keep a record of all changes made from the original FSAR and the related Safety Evaluation, whether involving an unreviewed safety question or not.

10 CFR 50.59(c) - provides that for these changes, tests and experiments involving an unreviewed safety question, the licensee shall submit an application for amendment of his license pursuant to 10 CFR 50.90.

10 CFR 50.90 "Application for amendment of license or construction permit"

This provides that: "Whenever a holder of a license or construction permit desires to amend the license or permit, application for an amendment shall be filed with the Commission, <u>fully describing the changes desired, and following as far as applicable the form prescribed for original applications</u>."

10 CFR 50.100 "Revocation, suspension, modification of licenses and construction permits for cause."

<u>Licensing Consequences of Review</u>

The consequences of the review in terms of the types of problems encountered in meeting regulatory requirements may be categorized as follows:

1) Descriptions which are incomplete, ambiguous and errored, varying from relatively minor matters to matters of substantial importance to safety.

   Except for relatively minor matters, this category has been considered non conservative since they provide no sound basis for ensuring that the detailed requirements of the Licensing Basis are specified for the operating facility.

2) Plant Engineering providing for unlimited operability of Process and Protection Elements. Safety Evaluations have been submitted and accepted creating an element of the Licensing Basis [within the boundaries of unlimited operability].

The Technical Specifications are not in accordance with the Licensing Basis by removing Operability Requirements without submitting necessary evaluations and proposals for evaluation by the NRC.

For this situation, the general situation is that "The Licensee shall evaluate and propose."

Examples include deletion of Operability Requirements for RHR, Component Cooling, RCS Loops, Elements of Reactor Trip System Instrumentation, and Engineered Safety Features Actuation System Instrumentation.

3) a) Plant Engineering with Operability Status limited by Plant Control or Protection Logic to certain MODES (and phases) of operation. Safety Evaluations for the limited Operability Status have been submitted and accepted as an element of the Licensing Basis.

The Technical Specifications are not in accordance with the Licensing Basis Plant Protection Logic on which the safety was assessed e.g., Reactor Trip on ESFAS initiation in MODES 3 and 4 is not provided for in the Technical Specifications.

The Licensee shall evaluate and propose.

3) b) Plant Engineering with Operability Status limited by Plant Control Logic and related Safety Evaluations submitted. Review of submittals for Amendment may include an interfacing branch. SER issued contrary to Regulations pertaining to that Branch. Examples include proposed deletion of auto initiation of MD-AFW pumps below P-11 by manual block, and deletion of Pressurizer Water Level - High trip.

The proposed Technical Specification is in accordance with the Licensing Basis, but not in full accordance with Regulatory Requirement. The licensee [should or] shall evaluate and propose.

This circumstance also introduces mixed and deficient protection rationale for a large number of occurrences requiring protection under Regulatory Requirements.

4) Plant Engineering with Operability limited by Plant Control Logic. However, no Safety Evaluation has been submitted for the limited Operability circumstances, which introduces unreviewed safety questions in the form of unforeseen and non-analyzed events. Examples include the absence of any "Low Flow" Reactor Trips below the P-7 permissive, and absence of many other Reactor Trips.

The plant is inside the Licensing Basis Engineering which however has not been adequately evaluated. This is a situation in which Regulatory Requirements have not been met within the ensuing Licensing Basis since an adequate clarification of and evaluation of the circumstances has not been undertaken.

The licensee shall evaluate and propose.

5) The Safety Analysis Limits (in the form of response times) provided in the FSAR for ESFAs are in general less conservative than used in the evaluations of the Licensing Basis.

The Licensee shall evaluate and propose.

6) The response time provided may closely conform or agree to the Licensing Basis value, but the Licensing Basis value is contrary to Regulatory Requirements e.g., the Licensing Basis uses response times for AFW from non-safety related sources; whereas safety grade sources have a significantly greater response time. This delay may also impact response times for other ESFAs equipment.

The plant is inside the Licensing Basis Engineering which however has not been evaluated to Regulatory Requirements.

The Licensee shall evaluate and propose.

7) a) Proposed Technical Specifications for major plant protection activities which do not [appear to] conform with the principal procedures described in the Licensing Basis. So that whilst the proposed Technical Specifications are not in accordance and also non-conservative, with respect to the Licensing Basis, they are also contrary to Regulatory Requirements.

This applies particularly to Boration Control in MODES 1, 2, 3 and 4 and Emergency Core Cooling Systems in MODES 3, 4, and 5. No evaluation and proposals are submitted.

The Licensee shall evaluate and propose.

7) b) Also, as a result of 7)a), we have discussed possible modifications to these proposed Technical Specifications, which may make them acceptable providing appropriate protections are added and suitable evaluations proposed.

Examples include the virtual absence of any necessary protection (including constraints) to ensure RCS safety to Regulatory Requirements under Condition II, III and IV occurrences in MODES 3, 4 and 5 due in part to the Boration Control disparity mentioned in 7 a) above.

8) The absence of necessary correlations between surveillance requirements for equipment performance and that performance necessary to achieve the required Plant Protection under Condition II, III and IV Occurrences.

An example includes Aux FW distribution to remaining intact Steam Generators in a Main Feed Line Rupture Event in which two Steam Generators providing steam to the Turbine Driven AFW Pump are ultimately faulted.

The licensee shall evaluate and propose.

9) It is a fact that engineering and construction of a nuclear facility must be checked on an element by element basis to ensure that the enormity of all the interfaces meet as required to enable final assembly and startup. Similarly, with Technical Specifications, unless they are likewise checked on an element by element basis, there will be no guarantee that the plant will have the level of safety proposed in the Licensing Basis Documents.

The Licensee has primary responsibility for this element by element check and our review together with responses from the requested evaluations and proposals will reflect the consequences of the exercise of that responsibility.

Invitation For Comment

The writer would welcome written and signed comments within the Regulatory Framework, on this Review.

References

a) Memo from Harold R. Denton, Director
Office of Nuclear Reactor Regulation
for Darrell G. Eisenhut, Director
Division of Licensing and
Roger J. Mattson, Director
Division of Systems Integration
on the Subject: DIFFERING PROFESSIONAL
OPINION OF MR. LICCIARDO REGARDING MCGUIRE
TECHNICAL SPECIFICATION and dated: March 21, 1984

b) Memo from Brian W. Sheron, Chief, RSB, DSI to
Robert Licciardo RSB, DSI dated April 11, 1984
on the Subject: MCGUIRE TECHNICAL SPECIFICATIONS
ASSIGNMENT

MCGUIRE UNITS 1 & 2: PROPOSED TECHNICAL SPECIFICATIONS

DETAILED REVIEW OF "PROOF & REVIEW" COPY

PREPARED BY
Robert B. A. Licciardo
Nuclear Engineer
RSB/DSI/RSRS

Date: June 12, 1984

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 requirements 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.

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 acceptance 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.

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

e) Concerning description under Section 2.1.1 above. We propose this description should clarify that the "combinations" presented are those allowed under "Anticipated Operational Occurrences" and not steady state conditions.

f) 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.

SUMMARY

The current method of representing Reactor Core Safety Limits is not clearly in accord with the Licensing Basis. Therefore it must be considered non-conservative and the Licensee shall evaluate and propose.

"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

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 would be necessary.

b) As for Section 2.1.1 above, is it not appropriate to clarify that the RCS Coolant System pressure shall not exceed [2735] psig under any Anticipated Operational Occurrence or Design Basis Accident.

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.

d) 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.

e) Concerning Action Statement: MODES 1 & 2. This should consider restoration 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.

SUMMARY

The current representation of Safety Limits for RCS pressure in this Section 2.1.2 is non-conservertive with respect to the Licensing Basis. The Licensee shall evaluate and propose.

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 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 5: TS incomplete; should read as: Intermediate Range, [High] neutron flux.

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.

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

The following items should be added, because they initiate Reactor Trip directly and independently of the SI signal.

17a) Pressurizer - Low Pressure (Safety injection)

The additional qualifier (SI) is generally used to distinguish this from item 5, Reactor Trip on Pressurizer Pressure-Low

17b) Containment Pressure-High

17c) Low Steam Line Pressure (subject to P-11 block)

17a) Manual Safety Injection

Item 12: Low Reactor Coolant Flow

a. Concerning Reactor Trip on "Low-Reactor Coolant Flow in One Loop."

Reference 7, Section 15.2.5.1 states that "Above approximately 50% power, Permissive P8 allows low fl[illegible] any one loop to actuate a reactor trip."

Please explain why there is no anticipatory signal for this circumstance ie under frequency, undervoltage, loss of RCP breaker. Such anticipatory signals are provided below P-8 when safety consequences are more conservative for this facility. (See later 12b.) Is this adequate conformance to diversify requirements of Criterion 22 - Protection system independence.

b. Concerning Reactor Trip on "Low Reactor Coolant Flow "In Two Loops Below P-8.

The plant is not licensed for operation with only 3 loops operating in MODES 1 and 2 below P-8. Please explain why you therefore propose a trip based on Loss of Flow in 2 loops instead of only one, at these conditions and which is not in conformance with GDC 20, "Protection System Functions." Information is provided under reference 7, Section 15.3.4.1 to show that Acceptance Criteria would not be exceeded but as indicated above it is outside the current licensing basis and should therefore be excluded.

This licensee should evaluate our concerns in items 12a and 12b above in conjunction with those of item 18.b.a of this same review of Table 2.2-1, and propose. This can be interpreted as a generic issue.

Item 13: Concerning Steam Generator Level-Low, Low

Reference 16, 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 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 note 4. It is proposed that a notation to this effect should appear under this item.

Item 21 (Proposed): [Reactor Trip on] Reactor Coolant Pump Breaker Position

Proposed: In accordance with the Licensing Basis FSAR, indicating that opening of two or more circuit breakers actuates the corresponding undervoltage trip relay above Permissive 7 (10% power); reference 7, section 15.2.5.1.

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

a) This T.S. provides that when power level is less then Permissive P7 (with P10 (Nuclear) or P13 (turbine) powers of less than 10%) the undervoltage (and RCP breaker position), under frequency and low flow reactor trips are blocked and will allow the reactor to remain untripped, and therefore at 10% power, on loss of offsite power.

   The FSAR in reference 5, item 7.2.2.1.2d which describes this permissive provides no safety evaluation of the consequences. Accident Analysis in Reference 7, section 15.2.9 for "Loss of Offsite Power to the Station Auxiliaries" is based on protection provided by these trips which are now blocked, and no evaluation is provided to show an acceptable RCS response under these particular circumstance. The existing FSAR, reference 7, Section 15.2.9.2 and related Table 15.2.9-1 shows acceptable natural circulation, but at a maximum power level of only 5%.

Accident Analysis in Reference 7, Section 15.3.4 "Complete Loss of Forced Reactor Coolant Flow" also deperds on this protection, and no evaluation is provided to show an acceptable response by the RCS system from the P-7 power levels. This also applies to Section 15.4.4, "Single Reactor Coolant Pump Locked Rotor."

There are additional events potentially arising from this item which have not been analyzed. These include a circumstance in which a normal turbine load rejection from just below the P-8 power level could result in a sequence in which power to RCPs are lost after both Nuclear and Turbine Power signals are reduced below 10% (P-7) so that reactor trip on this loss of power event could not occur, but with residual core heat fluxes at substantially greater than 10% in the early phase of the event followed by a 10% steady power level [Note also, that below P-7, a number of other reactor trips are also blocked including Pressurizer Water Level-High, Pressurizer Pressure-Low and Pressurizer Pressure-High]

The situation is one in which Condition II, III and IV occurrences are not protected in accordance with GDC 20, Protection System Functions: "The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences." It also introduces an additional occurrence, i.e., a failure to automatically trip the reactor, on top of the initial occurrence, and which in itself, and in combination with the initiating occurrence has not been evaluated.

It has not been Regulatory Practice to allow a Condition II occurrence to be followed by a Condition III or IV occurrence in the course of protective actions.

The licensee should evaluate the restoration of reactor trip on "low flow" trips down to and including MODE 2 (MODES 3-5 are discussed later) to be in conformance with G.D.C. 20 "Protection System Functions," and propose. As part of this evaluation, the Licensee should verify performance under these T.S. conditions and review for, and evaluate, Licensing Basis Occurrences affected by this T.S. requirement to show that all Regulatory Acceptance Criteria for Abnormal Operating Occurrences and Postulated Accidents are currently satisfied, making appropriate allowances for any manual Operator Action required. These events should include Loss of Off-Site Power to the Station Auxiliaries, Complete Loss of Forced Reactor Coolant Flow and Single Reactor Coolant Pump locked Rotor. [It should be noted that other reactor trips such as Pressurizer Water Level-High and Pressurizer Pressure - Low are also blocked under these conditions. Steam Generator Water Level-Low Low remains available together with Auto-initiation of AFW pumps. Steam Generator High High Turbine Trip is available, but does not trip the Reactor at these low power conditions (below P-8).]

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: G

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.

The pressurizer pressure low - reactor trips are used as both primary and back up in a number of Condition II Condition III and Condition IV occurrences, all involving breaks in the primary and secondary systems, reference 7, table 7.2.1-4 (3 of 5). Although safety injection is subsequently employed in almost all these situations, earlier reactor trip on pressurizer pressure low - is depended upon instead of the later reactor trip on pressurizer pressure low - (Safety Injection). The worst situation for most of these accidents is that of maximum power level reference 7, Table 15.1.2-2. No evaluations are provided for zero power level.

It is possible for these breaks in the primary and secondary systems to occur at less than 10% power level down to and including the startup condition (with 4 RCS loops running) ie MODES 1 & 2. (Such breaks in MODES 3-5 are discussed later). With the proposed TS, reactor trips for these breaks would be <u>delayed</u> to be initiated later by the ESFAS (SI) related signals. The licensee should provide a safety evalution of these circumstances and which is not based upon arguments relating to probability of the events. The evaluation should provide for the event to occur immediately subsequent to any normal operating transient providing the most conservative set of conditions prior to the event such as a complete load rejection using steam dumps from the P-8 level.

Until there has been a re-evaluation of these circumstances, the proposed T.S. must be considered non-conservative in respect to Regulatory Requirements. Additionally it can be interpreted as a Generic Issue.

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.

G

cii) 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 7.2-6 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. G

Item 18(f). Proposed new item: High Power Reactor Trip on Turbine Trip; Block by absence of P-8.

The Anticipatory Reactor Trip on Turbine Trip required by TMI Action Plan II.K.3.12, is bypassed below P-8. The SER is provided in reference 15, Item II.K.3.12, and reference 21 for McGuire Unit 1. We have issued no related final SER for McGuire 2 at this time. Note the related Basis will need to be amended.

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.

Item: General - This is a need to identify potential blockage of each of these Reactor Trip Functions by Plant Logic and any related manual action, e.g., $\leq$ P-7, $\leq$ P-11 with manual blockage etc. This enables improved perception of real levels of engineered protection than is currently available. Table 3.3-1 contains only approximate information concerning plant situations at which protection levels are changed. It also contains NON-OPERABILITY MODES which are not pre-determined by Plant Logic.

SECTION 3.4.1 REACTIVITY CONTROL SYSTEMS

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

G. (RSB)

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).

T.S. Page 3/4 1-1 and 2 specifying a shutdown margin of 1.6% delta K/K over MODES 1 through 4 should be modified to exclude MODES 3 and 4, and SHUTDOWN MARGIN T_{AVG} should be changed from >200°F to ≥557°.

A new T.S. Page 3/4 1-2(a) should be added for BORATION CONTROL SYSTEMS in MODES 3 through 5, from T_{avg} < 557°F through 140°F, providing that the boron concentration in the RCS shall be increased to a value which will give a shutdown margin of 1% delta K/K at 200°F.

Safety Signficance: These actions are necessary to bring the safety status of the plant into conformance with the Licensing Basis. Without this, the plant is in a less than conservative MODE which has not been evaluated. Further, it appears that OPERABILITY REQUIREMENTS of Table 3.3-1, REACTOR TRIP SYSTEM INSTRUMENTATION and TABLE 3.3-3 ESFAS INSTRUMENTATION may be conditioned on these higher Boron Concentrations so that ommission of Additional Boron Concentration in accordance with Reference 16, page Q-212-47e makes for an inconsistent and nonconservative level of protection for all NON-LOCA events for $T_{avg} \leq 557°F$.

The proposed T.S. might be acceptable if all events were analyzed in MODES 3 through 5 and the OPERABILITY REQUIREMENTS OF TABLES 3.3-1 and 3.3-3 reviewed.

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.

TS Page 3/4 1-6. 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.

An example of a safety impact is for the Design Basis Main Steam Line Break Event which is initiated from zero power in MODE 2 from a Set Point Tmin of 557°F. Any "increase" in this value (at given shutdown margin) would lead to conditions less conservative than the design basis.

To be within the Licensing Basis, this TS Section 3.1.1.4 should therefore provide that the Temperature for criticality [at zero power] shall be a set point value of 557°F with appropriate surveillance requirements. The Applicability is for MODES 1 and 2.

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

Section 3/4.1.2 BORATION SYSTEMS

T.S. Page 3/4 1-7: Concerning "BORATION SYSTEM, FLOW PATH - SHUTDOWN. APPLICABLE MODES 5 and 6:

The current T.S. requires an (unidentified) charging pump to supply Boron to the RCS. Current Licensing constraints on ECCS operation discussed under Section 3/4.5 Emergency core cooling systems" require that only one centrifugal charging pump is permitted to be in operation from a condition of 1000 psig/425°F in MODE 3 down to RHR operation commencing with MODE 4. In MODE 4, a similar and parallel requirement for overpressure protection in the RHR mode with water solid operation extends this requirement through MODE 4 to MODE 5; reference 11, page 5-1 where it is described that under RHR operation, the "only remaining centrifugal charging pump could cause an overpressure transient as a result of inadvertent start" but that "The Licensee has shown that [in this case] the 10 CFR 50 Appendix G Limit is not reached.

Charging pump requirements in MODE 6 are defined by reference 10, Section 15.2.4.2, item 3 under "Dilution During Refueling" in which a precondition for the "uncontrolled Boron Dilution Event" is that "the charging pumps are inoperative."

These circumstance permit only one charging pump, which must be a centrifugal pump only, in operation from "standby (at 1000 psig/425°F) through to MODE 5"; therefore the term SHUTDOWN in the title and the APPLICABLE MODES 5 and 6 should be replaced by these conditions. Also, the description of the charging pump should be expanded by the term "centrifugal" together with the proviso that "this centrifugal charging pump also be the same and only pump allowed for ECCS and other operations under these circumstances."

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

T.S. Page 3/4 1-8. Concerning: "FLOW PATHS - OPERATING" in APPLICABLE MODES 1 2, 3 and 4.

The Licensing Basis ECCS requirements discussed under Section 3/4.5 EMERGENCY CORE COOLING SYSTEMS of this report do not constrain charging pump operation above 1000 psig/425°F. Therefore the existing provisions on this T.S. page for charging pumps remain valid with the exception that APPLICABLE MODE 4 should be deleted and MODE 3 must be conditioned as MODE 3 (Down to 1000 psig/425°F). Further the title should be changed to incorporate these constraints.

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

The ACTION statement should be revised to be consistent with the Boration Requirements adopted out of item "Section 3/4.1.1" of this report.

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 one centrifugal 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.

Surveillance Requirements under subsection 4.1.2.3.2 must reflect the requirements of later SECTION 3/4.5 ECCS of this report in which "All centrifugal, [and reciprocating] charging pumps excluding the required OPERABLE pump shall be demonstrated inoperable by" additional features to those already described in this subsection, namely, "by verifying that the motor circuit breakers are secured in the open position by being opened, locked and tagged; the alternate of isolation from the Reactor Coolant System by at least two isolation valves with breakers for the valve operators being open, locked and tagged has not been provided. (reference 12, page 6-6 concerning racking and locking out of pumps; also reference 11, pages Q212-47 and 47a)

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

G (RSB)

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 $\leq$ 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 $\leq$ 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

This title (and related Applicability MODES 5 and 6) should be changed to BORATED WATER SOURCE - MODE 3 (1000 psig/425°F) THROUGH TO MODE 5, to be compatible with the changed title to TS pages. 3/4 1-7 and 3/4 1-9 discussed earlier since this page refers to borated water sources for situations there described.

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.

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

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

An appropriate modification may need to be made to the Boron Concentrations and volumetric requirements in the Boric Acid Storage System in these MODES 3 (1000 psig/425°) through 5 to provide for the increased Boron Concentrations required from the Licensing Basis in these MODES discussed in this report under T.S. page 3/4 1-1, 2 and 2a.

Why is the refueling water storage in MODE 5 proposed as only 26,000 gallons when reference 8, page Q212-57, revision 25, under Case-3 provides that in MODE 5, in the event of loss of cooling by a fail closed RHR/RCS isolation valve the charging pump could provide feed and bleed cooling through the PORVs for up to 5 hours from the RWST and subsequently the RHR pump and heat exchanger would re-circulate and cool from the containment sump. Would not this require an unchanged requirement from MODES 1 through 4 of at least 372,100 gallons.

The proposed T.S is non-conservative in respect to the Licensing Basis. The Licensee shall evaluate, including all our concerns above under T.S. Page 3/4 1-11, and propose.

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

This title, and related applicability modes, should be changed to: BORATED WATER SOURCES - MODES 1, 2, and 3 (Down to 1000 psig/425°F) to be compatible with the changed title to T.S. Pages 3/4 1-8 and 3/4 1-10 discussed earlier, since this page refers to borated water sources for the situations there described.

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.

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

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.

The absence of required LCOs makes the proposed T.S. less conservative than the Licensing Basis. The Licensee shall evaluate, including our concerns under TS Pages 3/4 1-12, and propose.

T.S Page 3/4 1-13a. 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."

This SER requirement has not been provided in the Technical Specifications. Please discuss provision under a proposed new item under Section 3/4.1 REACTIVITY CONTROL SYSTEMS, entitled "INSTRUMENTATION" in which these requirements would be proposed for Applicable MODES 3, 4, 5 and 6.

A similar provision is provided under Refueling, TS page 3/4 9-2 INSTRUMENTATION and is applicable only to MODE 6. Since it is a part of "Reactivity Control Systems" and applicable over additional MODES, it should be provided in this context also as discussed above.

The proposed T.S. is less conservative than the Licensing Basis. The Licensee shall evaluate and propose.

T.S. Page 3/4 1-20 Concerning: SHUTDOWN ROD INSERTION LIMITS

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

a) Specifications for limiting conditions of operation on the positions of these movable control assemblies apply only to MODES 1 & 2. There is no Technical specification on positions in MODES 3-5 although T.S. Page 3/4 1-18 concerning "Position Indication system - shutdown" requires operability of a Rod Position indication system in MODES 3 through 5 when the reactor trip system breakers are in the closed position.

It is proposed that in general, Technical specifications are required by 10 CFR 30:46 to be placed on the limits of movable control assemblies in these modes to limit the consequences of Condition II, III and IV events which may occur, unless analyses and evaluations show that these are unnecessary.

An example of the need is reflected in the memo to reference 26 in which rod positions for Boron Dilution events are specified from Refueling through to Hot standby as All Rods Out (Mode 6, Refueling) and, All Rods In with Most Reactive Rod Stuck Out, for Hot Standby through Cold shutdown. Further, applicants may opt to assume a more limiting initial control rod position - which would however need to be justified.

The Boron Dilution event for McGuire has "apparently been" made acceptable by procedures requiring the RCS to be filled with Borated (approx 2000 ppm) water from the refueling water storage tank prior to "Start Up"; reference 7, page 15.2-15, revision 10. Reference earlier discussion on TS. Pages 3/4 1-1, 2 and 2 a. This is an LCO and should appear in the proposed T.S.

With the existing T.S. without the required increase in Boron concentration, there is no guarantee that a return to power during dilution will not infringe current RCS Safety Criteria. Under those circumstances a T.S. on the Position at shutdown of Control Rods is required unless an acceptable safety evaluation is submitted to show the contrary.

In general, also, the same concern applies to any other Condition II, III and IV occurrence which can lead to a return to power in these Modes. Until these circumstances can be shown to result in acceptable consequences without a T.S. on the position of these movable rods, then 10 CFR 30:46 would require such a Technical specification. In this evaluation, cognizance also needs to be given to the reduced operability requirements for all Reactor Trip Instrumentation and Engineered Safety Features Actuation Instrumentation in these MODES (3 through 5). This is particularly significant with the proposed T.S. on Boration Control where resulting shutdown margins are substantially less than these provided by the current Licensing Basis.

The Licensee shall provide analyses and related safety evaluations to justify his current absence of Technical Specifications in respect of SHUTDOWN and CONTROL ROD positions during MODES 3 through 5. Without this, the proposed T.S are non-conservative with respect to the Licensing Basis.

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 3/4.2.1 THROUGH 3/4.2.4 POWER DISTRIBUTION LIMITS

RSB has not reviewed these sections on the understanding that they are the primary responsibility of Core Performance Branch.

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

The current information does not adequately represent all those perameters 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?

Evaluation

a) DNB presents only one Acceptance Criteria for acceptable operation of the RCS: There are others including Fuel element clad failure and Appendix K requirements depending upon the occurrence being considered. Additionally there are RCS overpressure, steam generator overpressure and Hot Leg Boiling Criteria.

As indicated in our comment in Section 2.1.1, item f, initial conditions which cover a larger N° of variables than those presented in Table 3.2.1, in combination, determine RCS safety in the necessarily broadest sense.

It is suggested that this section be deleted, and the relevant information be supplied under T.S. Sections 2.1.1 where it belongs and where it has been discussed.

G b) Concerning Table 3.2-1. The value for Reactor Coolant System T_{avg} given as ≤ 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 B

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

G 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).

G c) Concerning Table 3.2-1 Pressurizer Pressure. Please explain the basis for the given value of ≥ 2220 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 G CRSB

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

f) 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 4-9) and Steam Generator Levels under Section 3/4.4.5 T.S. Page 3/4 4-11). G CRSB,

CONCLUSION

The parameters proposed by the T.S. as "DNBR PARAMETER" under TABLE 3.2-1 are an incomplete set and inadequately defined in terms of Set Points, Allowable Values and Safety Analysis limits. All this necessary information is available from the existing Licensing Basis and their incomplete and inadequate representation creates a non-conservative situation with respect to the Licensing Basis. The Licensee shall evaluate and propose. This is only partly a generic problem arising from an inadequate representation in the W STS.

TABLE 3.3-1 REACTOR TRIP SYSTEM INSTRUMENTATION

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

Item 6c: Source Range, Neutron Flux

Does this channel provide an alarm only function, or an alarm plus trip function.

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 dilut'... alarm requirements of proposed T.S. Page 3/4 1-13 (a), then it is not in acc...ance 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 . 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.

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Please evaluate the conformance with 10 CFR 50 App. A, GDC 20 and 22 of using the Source Range Neutron Flux as a non-diverse reactor trip under circumstances in (MODES 3 through 5) in which there is no Technical Specification on movable control assemblies, and which instrumentation consists of only two channels. Also for circumstances in which all normally available other backup trip functions such as pressurizer pressure - high and low, and water level high and "low reactor coolant flow", are not specified to be OPERABLE in Table 3.3-1. The Licensee shall propose on the basis of this evaluation.

Items 7 & 8 Overtemperature ΔT and Overpower ΔT.

The current T.S. provides for operability of these trips in in MODES 1 & 2, and not 3.

Occurrences using these reactor trips include events which can be initated from subcritical Zero Power in MODE 2 (Reference 5, Table 7.2.1-4 and Reference 7, Table 15.1.2-2). With the proposed T.S. in which no difference in Reactivity Condition k_{eff} and Shut Down margin is required between MODES 2 & 3, how can the Licensee justify removal of these trips on entry into MODE 3 in which the only difference in RCS conditions is a marginal reduction in temperature, from the Programmed No Load T_{avg}.

Item 11: Pressurizer Water Level - High

Operability considerations from MODE 2 down to and including water solid conditions in the RHR MODE are discussed under Section 2.1.1 18 c(ii.) with a proposal that exclusion of this trip for all these MODES is non-conservative in respect to 10 CFR 50, GDC 20 "Protection System Functions" both for reactivity control systems and overpressure protection systems.

The necessity for this trip is increased when reviewed against the totality of the proposed exclusions for Reactor Trip System Instrumentation discussed in the following section under items 2-21 (selected).

Items 2-21 (selected):

Items 2, 5 and 6:	Power Range, Intermediate Range and Source Range Neutron Flux Trips
Item 9:	Pressurizer Pressure - Low
Item 10:	Pressurizer Pressure - High
Item 11:	Pressurizer Water Level - High
Item 12:	Low Reactor Coolant Flow
Item 14:	Undervoltage Reactor Coolant Pumps
Item 15:	Underfrequency Reactor Coolant Pumps
Item 21:	(Proposed) Reactor Coolant Pump Breaker Position Trip.

At this time, in MODE 3, 4, and 5, the proposed Technical Specifications for the plant do not provide any neutron flux trip for Accident Analysis requirements, although the FSAR would require the Power-Range Neutron Flux Trip, Low Setpoint; no insertion limits on movable control assemblies, Reactor Coolant Pump (RCP) operability requirements permitting less than four (4) RCPs in operation, a Boron Concentration Control which provides less shutdown margin capability than the FSAR requirements, no trip of RCPS on Loss of Flow or Undervoltage or Underfrequency or Opening of RCP breakers, and in addition it is proposed that no trip be provided for Pressurizer Pressure-High, Pressurizer Pressure - Low, and Pressurizer Water Level - High. And for these circumstances we have no well defined evaluation as to why these reduced protections adequately protect the plant against any of the appropriate Condition II, III and IV occurrences in these MODES except a Large and Small Break LOCA, and Steam Line Break.

We realize the interdependence of many of these factors in setting a minimum acceptable level of Reactor Trip Protection and that relatively simple solutions are possible, but at this time we do not have available an acceptable analysis and evaluation justifying the proposed T.S. position.

The Licensee shall provide an analysis and evaluation of the circumstances under applicable Conditions 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 he will establish an appropriate range of Reactor Trip System Instrument on to Safety Related Requirements. The evaluation shall be undertaken in c nction with our concerns for current Technical Specifications under Se ion 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION of this report.

Items: 12 Low Reactor Coolant Flow Trip

14 Undervoltage - Reactor Coolant Pumps

15 Underfrequency - Reactor Coolant Pumps

21 (Proposed) Reactor Coolant Pump Breaker Position Trip

All these Reactor Trip Functions concern potential for a loss of Reactor Coolant Flow. The proposed T.S. deletes all operability requirements in MODES 3 through 6. [It also deletes in MODE 2, but this has been discussed earlier under TABLE 2.2-1 items 18.b.a and 12a and 12b]. We have discussed our related concerns and requirements for analyses and evaluations in MODES 3, 4 and 5 under Items 2-21 (selected) above.

A loss of Coolant Flow in the RCS places the plant in an Emergency Operating Mode. Please advise therefore why such an event should not automatically trip the Reactor in MODES 3 through 5 with the Boron Concentrations being considered for the proposed Technical Specifications. Why should we not use the reactor trip as a device to ensure complete shutdown of all movable control rods during any time that a minimum set of RCPs in accordance with operability requirements of the T.S., are not available since RCPs may be required for accident mitigation in MODES 3 through 5 as appropriate. The Licensee shall evaluate and propose.

Item 13: Steam Generator Water Level - Low Low:

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. G (RSB)

Item 17: Safety Injection Input From ESF.

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

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. G (RSB)

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.

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. G (RSB)

Item: Concerning Proscribed Values For % RATED THERMAL POWER DURING STARTUP (MODE 2) AND POWER OPERATION (MODE 1)

We note that operability requirements for Reactor Trip System Operation when expressed in terms of MODES 1 and 2 are inaccurate and do not represent the actual situation at the plant. T.S. Page 1-9, Table 1.2 defines Power Operation (MODE 1) at > 5% Rated Thermal Power and Startup (MODE 2) at ≤ 5% Rated Thermal Power.

G 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.

The discrepancy identified is non-conservative and important on at least 2 counts:

1) A non-conservative discrepancy between the fundamental maximum T.S. Limit of 5% power level in MODE 2 as given on T.S Page 1-9, Table 1-2 and the nominal value of 10% with a real Safety analysis Limit of 10% plus a Total Channel Error as yet unspecified.

2) The elimination of most reactor trip Functions (and many ESFAS Functions) at this non-conservative power level without a separate comprehensive Safety Evaluation with respect to Regulatory Requirements and the existing Licensing Basis.

   The Licensee shall evaluate, including our concerns enpressed above, and propose.

TABLE 3.3-2 REACTOR TRIP INSTRUMENTATION RESPONSE TIMES

Item 1: Manual Reactor Trip

At this time, the licensee proposes that the Response Time (RT) for manual reactor trip is not required by safety analysis. Furthermore he proposes that in MODES 3 through 5, the only remaining operable trips are those using Source range neutron Flux and they also are not required by Safety Analyses.

Under TABLE 3.3-1, items 2-21 (selected) we have already required the licensee to re-evaluate his position in respect of what neutron Flux trips he intends to propose, together with their related Tech specs to place the reactor in a safe condition in respect to Condition II, III and IV Occurrences in MODES 3 through 5. Until this evaluation and proposal are accepted, the Licensee shall have a Safety Related Manual Trip System to assist in meeting minimum Regulatory Requirements in 10 CFR 50, APP. A. III. Protection and Reactivity Control Systems, and the Licensee shall evaluate and propose as a priority issue. At this time, the proposed T.S is non-conservative in respect to Regulatory Requirements for 10 CFR 50, App. A. III.

G
(RSB)

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 30.46. 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.

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(RSB)

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 d) 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.

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Item 9: Pressurizer Pressure - Low

Item 10: Pressurizer Pressure - High

The TS specifies a Response Time of ≤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.

G
(RSB)

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 16, Turbine Trip

A response time for Reactor Trip on Turbine Trip is not provided in the Technical Specifications. Reference 7, Table 15.1.3-1 advises that the response time for such a trip is 1.0 sec. but that it is not applicable to the analysis used.

Reference 7, Section 15.2.10.3, concerning Excessive Heat Removal Due To Feedwater System Malfunctions. Under the title of "Results" on page 15.2-30, the second paragraph describes how for this particular event at full power "A turbine trip and reactor trip are actuated when the steam generator level reaches the high-high level set point."

Also, for the Occurrence of "Inadvertent Operation of the ECCS During Power Operation under reference 7, Section 15.2.14.3, page 15.2-40, revision 43, under Conclusions states that: "If the reactor does not trip immediately, the low pressure reactor trip is actuated. This trips the turbine and prevents excess cooldown thereby expediting recovery from the incident.

Under these circumstances therefore, Reactor Trip on Turbine Trip is necessary to automatically terminate the event. The Licensee should review the response time used in the above calculation and provide an evaluation of its decision is respect of placing it in the T.S. under the requirements of 10CFR50.36

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.

G (RSB) 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. G (RSB)

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. values. G (RSB)

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.

Item 1: Safety Injection, Reactor Trip, Feedwater Isolation, Component Cooling Water, Start Diesel Generators, and Nuclear Service Water.

This description of Item 1 lists the various functions initiated by given signals (which are generally those initiating SI).

However, Reference 5, Figure 7.2.1-1 (8 of 16) revision 34 and Figure 7.2.1-1 (13 of 16) revision 34, shows that the term "Feedwater Isolation" used in this Item 1 is actually comprised of four (4) separate Logic Functions, namely "Turbine Trip", "Trip of Feedwater Pumps", "Close All Feedwater Isolation Valves" and "Close the Feedwater Main and Bypass Modulating Valves.

The term Feedwater Isolation is therefore an inaccurate term to use. It should be removed from this descriptor and replaced by the four separate functions, as each of them can be initiated separately and or together dependent upon the initiating Logic.

Further we also note that this functional unit is also that initiated by Steam Generator Water Level High-High (P14) reference 5, figure 7.2.1-1 (13 of 16) revision 34. and figure 7 of 16; revision 41.

Further, the function to be initiated by Steam Generator Water Level - High High is function 5 of the same Table which is again incompletely described and should be changed (see item 5 later) to clearly identify these same 4 elements. Under these circumstances, the current description for Item 1 should delete the term "Feedwater Isolation" and Item 5 (see later) should be expanded to include an additional Functional Unit identified as Safety Injection.

Additionally, the Function "Annulus Ventilation" needs to be added to the descriptor (reference 5, figure 7.2.1-1 (8 of 16) revision 34).

Also, the function unit description "Nuclear Service Water" should include [isolation and startup] of Nuclear Service Water.

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 1c: Containment Pressure - High/Applicable MODES 1, 2, 3.

The Current T.S. does not provide for initiation of SI on Containment Pressure - High, in MODE 4.

This is contrary to reference 8, pages Q212-47e, item 24, Q212-61b item 29, Q 212-61d, item 212.91 (15.4) wherein small and large breaks in the Steam Line and Reactor Coolant System are discussed down to and including MODE 4. Discussing NON-LOCA Accidents (in MODES 3, 4) below the P-11 (1900 psig) block of SI on Pressurizer Pressure - Low (SI) and Steam Line Pressure - Low, provision is made that if a MSLB occurs inside containment [so that MSIV Isolation on

Negative Steam Line Pressure Rate - High does not contain the event for the faulted SG] then Safety injection will be activated by Containment Pressure-High.

Note: Automatic logic for realignment to SI is already provided in the T.S. in MODES 3 and 4. This MODE 4 Operability requirement for Containment Pressure-High would also facilitate re-alignment of equipment from RHR to ECCS alignment in the event of a large break LOCA under these circumstances as described in reference 8, page Q212-47a, item II.C.

The Licensee shall evaluate why his proposed T.S. is an acceptable change from the existing Licensing Basis, or include the operability requirement in his T.S. The proposed T.S. position is non-conservative.

Item 1d: Pressurizer Pressure-Low

This is the same title as used for Reactor Trip on Pressurizer Pressure-Low. This particular/ESFAS actuation is set at a lower pressure and should be described as: Pressurizer Pressure-Low [Safety Injection].

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.

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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 2c: Containment Pressure-High-High.

Operability is not required in MODE 4. This should be required to be consistent with the evaluation under Item 3.b.3. below.

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

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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 I 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.

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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 5, 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 4, Steam Line Isolation

4b: Automatic Actuation Logic and Actuation Relays

The proposed T.S. does not require Operability of Steam Line Isolation Auto Actuation Logic in MODE 4. However, this will be required if the Operability requirements of Steam Line Isolation on Negative Steam Line Pressure Rate - High, already specified in item 4d for MODE 4, are to be met. The proposed T.S. is non-conservative with respect to the Licensing Basis; the Licensee shall evaluate and propose.

Item 4a: Manual Initiation [of steam line isolation]

1) System
2) Individual

Operability requirements for manual initiation of Steam Line Isolation are not required by the current T.S. in MODE 4. This however will be necessary to allow the operator to manually isolate small breaks which do not activate the Negative Steam Line Pressure Rate - High signal or the Containment Pressure-High High signal.

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

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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 5: Turbine Trip and Feedwater Isolation

Reference earlier Item 1 in whic.. this title for Item 5 should be more accurately described as "Turbine Trip, Trip of Feedwater pumps, Close Feedwater Isolation Valves, Close F dwater Main and Bypass Modulating Valves. The Licensee shall clarify, evaluate and propose. Lack of accuracy can be non-conservative with respect to the Licensing Basis.

Item 5a: Automatic Actuation Logic and Actuation Relay [to effect Turbine Trip, Feedwater Pump Trip, Closure of Feedwater Isolation Valves and Closure of Feedwater Modulating Valves]/APPLICABLE MODES 1 & 2

The Applicable Modes of this Auto Actuation Logic need to be extended down to MODES 3 and 4 to be available to respond to the Safety Injection signals which are expected from the Licensing Basis (reference later Section 3/4.5, Emergency Core Cooling Systems, under GENERAL). The proposed T.S. is non-conservative with respect to the current Licensing Basis and the Licensee shall evaluate and propose.

Item 5b: Steam Generator Water Level - High High [to effect Turbine Trip, Feedwater Pump Trip, Closure of Feedwater Isolation Valves and Closure of Feedwater Modulating Valves]/APPLICABLE MODES 1 & 2.

The Licensee should evaluate the need to extend the operability requirements of this functional unit from current MODES 1 and 2 down to and including MODE 4. The determining factor may be the availablity of Main Feedwater Pumps during these MODES. Plant Operating Procedures which permit Main Feedwater Pumps to be available can cause An Excessive Heat Removal Due To Feedwater System Malfunction and/or Steam Generator overfill unless Safety Related isolation at the Main Feedwater [containment] isolation valves is incorporated into the T.S.

The Logic of reference 5, figure 7.2.1-1, (13 of 16), revision 34, involving signal inputs: Steam Generator Hi-Hi P-14, Safety Injection, Reactor Trip P4, and Low T_{avg} would need to be carefully reviewed, especially since there is currently little or no Sat ty Related Reactor Trip Protection in MODES 3 through 4 so that reactor trip P4 may not be available in conjunction with Low T_{avg} (during cooldown) to effect Feedwater Isolation, and Closure of Modulating Valves, as an inbuilt protection against such circumstances.

The proposed T.S. does represent a non-conservative position in respect to the Licensing Basis, as there is no prerequisite that Main Feedwater is isolated at the Containment Isolation Valves as an LCO, during MODES 3 and 4. The Licensee shall evaluate and propose.

Item 5c (Proposed): Safety Injection [to effect Turbine Trip, Feedwater Pump Trip, Closure of Feedwater Isolation Valves and Closure of Feedwater Modulating Valves]/Applicable MODES PROPOSED AS 1, 2, 3 and 4.

This trip is relocated from Functional Unit 1 to Functional Unit 5 in accordance with our earlier reviews under Item 1C and Item 5.

OPERABILITY is required in all Modes 1, 2, 3, 4, because SI protection has been found necessary within the Licensing Basis. The protection was already intended in the proposed T.S. this action represents a more accurate description of the Functional Unit and an improved placement in the T.S. The Licensee shall evaluate and propose.

Item 7; Auxiliary Feedwater (AFW):

<u>General</u>: 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.

Operability in MODE 4 is required by the FSAR to generally counter the 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 30.46: 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.

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[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 7.4.1.1.1.1. 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.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.

G [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 all 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.

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 9: Loss of Power: Emergency Bus Undervoltage - Grid Degrade Voltage:

Item 9: 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 ESFAS Functional Units and related Response Times not provided under Table 3.3-3.

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

What does this functional unit do. Please explain, and how many busses to be tripped for the action to be defined. If it is meant to initiate AFW: what pumps etc., and if so operability requirements should be extended to MODE 5. Lack of any clarity makes this proposed T.S. non-conservative. The Licensee shall clarify, evaluate and propose.

Item 10.a)a.: Pressurizer Pressure P-11:

Applicable MODES are 1, 2, 3.

Explain the consequences of this non-operability in MODE 4 on availability of dependent protective actions, e.g., main steam line isolation, which is considered under Item 4.b above. If main steam isolation is negated, it should be restored to conform to Regulatory Protection Requirement. The Licensee shall evaluate and propose.

Concerning P-11 Interlock and AFW Pumps.

The basis provided on proposed T.S. Page B 3/4 3-2 states that:

"P-11 (i.e., on system pressure increasing to P-11 valve) ---- Defeats the manual block of the motor driven AFW pumps on trip of the main feedwater pumps and Low-Low Steam Generator level."

The following information provides the current Licensing Basis on the particular proposed interlock P-11 in respect of AFW Pumps:

The Table 3.3-3, Item 7.c.1, in reference 5, for start of motor driven AFW pumps, does not provide for the above condition.

The P-11 interlock and its provision for automatic defeat [above P-11 setpoint] do not appear in reference 5, Table 7.3.1-3, Rev-35, 'Interlocks for ESAS and Figure 7.2.1-1 (15 of 16), revision 34, I&C Logic Diagram.

Reference 5, Section 7.4.1.1.6 describes this action under "Bypasses and Interlocks" and that wherever it is present, an alarm exists in the Control Room. This allows the operator to stop AFW pumps during shutdowns.

Supplement No. 5, reference 15, page 22-22 evaluates the use of the P-11 interlock as described in the above Basis and concludes that the situation is acceptable. However, the basis for the SER Supp 5 conclusion was that a possible steam line rupture or feedwater line break were not likely to occur in the proposed MODES when the P-11 is in effect. This is a mistake, all the earlier work of this review has disclosed that the premise of these events being not likely to occur has been rejected for these MODES 3 to 5, and detailed attention has been given to their possible occurrence together with the possibility of Auto Initiation and the consequences of automatic protective action. Where the P-11 lockout has been present on other protective actions, the consequences have been fully evaluated. There has never been a related evaluation on the absence of auto-initiation of motor-driven AFWS as now proposed.

If the Licensee wishes to pursue this he should evaluate all the events considered in the FSAR below the P-11 setpoint with manual initiation of MD AFW and making due allowance for all the relative reduced and changed protections available and the time frames which must allow for all other actions, e.g., isolation of a ruptured SG is expected to take 30 mins, see reference 7, section 15.4.2.2.2 page 15 4-13a, Revision 38. Further, the detailed review of this T.S. has been based on this availability.

We note that in his submittals concerning this matter, dated March 9, 1981 concerning TMI items, the Licensee states that "the turbine driven auxiliary feedwater pumps do not have a bypass feature." Yet we also note on his T.S. page 3/4 7-4 that the Turbine Driven pump is not required to be operable when steam generator pressures are less than 900 psig; this would require only approx. 20 mins. into standby cooldown to achieve. The result is that there would be absolutely no automatic supply of feedwater for <u>any</u> event beyond approx. 20. min into cooldown.

At this time, the current Accident Analyses in the Licensing Basis FSAR support the necessity for not using the current bypass for the Motor-Driven Pumps.

The Licensee shall advise what safety-related reasons require that he must bypass automatic startup of the motor-driven auxiliary feedwater pumps on top of both main feed pumps, and on SG Low Low-Level in the final stages of plant shutdown. Also, what prevents him from installing automatic restoration on receipt of the related protection signal.

Item 10.b; Interlock; Low-Low Tavg P-12:

Applicable MODES are 1, 2, 3.

Reference Item Table 3.3-4, Item 10b, of this document.

Since Interlock P-12 effectively provides and limits steam dump capability, including accidental blowdown, by constraining it to 3 cool down dumps to the condenser; why remove this interlock in MODE 4 and MODE 5 and remove its potential availability for related Licensing Basis requirements. The proposed T.S. is non-conservative with respect to the Licensing Basis; the Licensee shall evaluate and propose.

Item 10.c; Interlock; Reactor Trip P-4:

The eight separate functions affected by this interlock are described in reference 5, Table 7.3.1-3 (1 of 2). Please evaluate how the absence of this will affect the various functions to be performed and how they will impact the FSAR requirements for plant protection in MODES 4 and 5. This should be for both the "Reactor tripped" and "Reactor not tripped" conditions considering that the reactor can be in both situations during these Modes. Licensees evaluation to items 5a, b and c above should be also considered in this evaluation.

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

Item 10.d); Interlock; Steam Generator Level-High High, P-14:

Operability is not required by the T.S. in MODES 4 and 5. The need for this interlock in these Modes will be established by the Licensee in his response to items 5a, b and c above. The licensee shall provide his evaluation and propose. Until Safety Related Isolation of Main Feedwater

Containment Isolation Valves is included in the T.S., this proposed T.S. must be considered non-conservative with respect to Regulatory Requirements.

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.

G.

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.

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

General: These have been checked against the information in reference 18, table 3-4 and related NOTES FOR TABLE 3-4 on page 3-13 and which is described as being applicable to McGuire Unit 1, 50-369. At this time, the assumption is made that this information also applies to McGuire Unit 2, Docket No. 50-370. The licensee will docket this fact or otherwise docket the alternate information.

Item No. 1:

The description for this Functional Unit should be clarified and modified in accordance with our remarks under TABLE 3.3-3; Item 1.

Item No. ~~1d~~ 1a:

The description for this Functional Unit should more accurately read as "Manual Safety Injection Actuation." See reference 5, Figure 7.2.1-1 (8 of 16), Revision 34.

Item 1c:

Modify the description in accordance with our earlier comment under Table 3.3-3 1d to: Pressurizer Pressure - Low (Safety Injection)

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.

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.

Please discuss the logic of the values in reference 18. A Trip Set Point of a negative rate of 110 psi with an allowable value of 100 psi (both with a time constant of 50 psi) would provide that an earlier isolation of the MSIVs is less conservative, and this is not so for the MSLB event. The expectations are that negative rate for the allowable value would be higher than for the Set Point. Please clarify.

Further, the same reference 18 Table 3-4, column 12, states under notation (5) that this value is not used in the safety analyses. Since this ESFAS signal provides Main Steam Valve Isolation on Main Steam Line Break below the P-11 block point (instead of by Steam Line Pressure - Low) please describe how the plant is otherwise protected through the proposed T.S. Otherwise, please provide analyses which show that the plant is protected by this proposed setting under proposed T.S. requirements. This item is related to our other concerns on Technical Specifications on Boration Control under earlier Section 3/4.1.1 Boration Control. The proposition that this value is not used in Safety Aanlysis is non-conservative. The Licensee shall evaluate and propose.

Item 5: The description of this Functional Unit should be revised and clarified to our recommendations under Table 3.3-3, Item 5.

Item 5c: Proposed new item as "Safety Injection"

This should be included in accordance with the evaluation under Table 3.3-3, Item 5c)

Item 6a & b. Containment Pressure Control System

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

Item 7(c): Steam Generator Water Level - Low-Low

The licensee should respond to our concern under Table 2.2-1, item 13.

Item 7(d): Auxiliary Feedwater Suction Pressure Low

The description should be revised as proposed under our earlier Table 3.3-3 item 7d. Provide the basis for the values given.

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 7c:

- Reference (7) related Section 15.4.2.2.2 concerning Main Feed Line Rupture (MFLR) under the title of Major Assumption 10.

  "The auxiliary feedwater system is actuated by the low-low Steam Generator Water Level Signal. The auxiliary feedwater system is assumed to supply a total of 450 gpm to three intact steam generators.

- Reference 5, Section 10.4.7.2.2 states that "Travel stops are set on the steam generator flow control valves such that the turbine driven pump can supply 450 gpm to three intact steam generators while feeding one faulted generator and both motor driven pumps together can supply 450 gpm to three intact steam generators while feeding one faulted generator. The throttle positions allow all three pumps to supply a total flow of 1400 gpm to 4 intact steam generators."

- Reference 7 related Section 15.4.2.2.2, page 15.4-13a (Revision 38), states: "The single active failure assumed in the analysis is the turbine driven auxiliary feedwater pump. The motor driven pump that is headered to the steam generator with the ruptured main feedline supplies 110 gpm to the intact steam generator. The motor driven pump that is headered to two intact steam generators supplies 170 gpm to each. This yields a total flow of 450 gpm to the intact steam generators one minute after reactor trip. At 30 minutes following the rupture, the operator is assumed to isolate the auxiliary feedline to the ruptured steam generator which results in an increase in injected flow of 80 gpm."

The sequence of events in the accident evaluation in Reference (7), Table 15.4-1 shows that after the accident is initiated at a programmed value of SG level, the low-low SG level in the ruptured SG is reached 20 secs. later, and auxiliary feedwater [at 450 gpm] is delivered to the intact steam generators in 61 sec.

It appears, based on the above information, that on SG low-low in the ruptured SG, both the motor driven and the turbine driven pumps are initiated (with the single failure being in the turbine driven pumps). This is not in accord with the T.S. If it is assumed that low-low level in the other SGs is also reached at the same time by bubble collapse, please justify. We note that the Reactor & Turbine Control System is designed so that under normal operation, collapse of SG level on Turbine Trip will not cause a reactor trip; also at this time, main steam from intact SGs is being lost to the faulted SG so that whereas inventory is lost, a full collapse need not occur.

The proposed T.S.s 7cD and 7.c(2) appear to be non-conservative in respect of Accident Analysis used in the Licensing Bases. The licensee shall clarify, evaluate and propose; this should be in conjunction with our other concerns on this event noted later in Sections of this review.

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 9: Loss of Power

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

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 10a: ESFAS Interlock Pressurizer Pressure, P-11.

Actuation of this interlock substantively reduces ECCS protection against Conditions II, III, and IV Accidental Occurrences.

The FSAR has analyzed the consequences of this reduced level of protection for a limited number of these occurrences and this has been based on a system pressure of 1900 psig; Reference 8, page Q212-47, item 212-75 1A. Why then is a trip set point of $\leq$1955 psig used. This set point value should be below 1900 psig with appropriate allowances for drift and channel errors to the limiting value used in the Safety Analysis of 1900 psig. The current specification is non-conservative with respect to the Licensing Basis FSAR & therefore not in accordance with 10 CFR 50.36. The licensee shall provide a safety evaluation for the difference, for approval, or restore the set point to be a valid T.S. value.

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

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.

A set point of 553-551°F is provided. Provide the basis for this which should be consistent with our query under earlier Section 3/4.1.1. Boration Control concerning T.S. page 3/4 1-6, "Minimum Temperature For Criticality."

Item 10e. (Proposed).

To complete the list of ESFAS interlocks, it is necessary to add an item identified as 10e. Low T_{avg}.

The safety reasons for this are described under the later Item 11.b (Proposed) of this section.

Item 10c: Interlock, Reactor Trip, P-4.

This currently reads as: "Reactor, Trip, P-4, with NA (Not Applicable) trip setpoint & Allowable values." However, should not this item read as:

10c. P-4-with Trip Setpoint and Allowable values defined as in Reactor Trip to Table 2.2-1, with the exception of: "Power Range, Neutron Flux, High Negative Rate."

The basis for this is provided in Reference 5, Figure 7.2.1-1 (2 of 16), Revision 42. The licensee should explain why Reactor Trip Signals initiating P-4 include all items in Table 2.2-1 with the exception of "Power Range, Neutron Flux, High Negative Rate." The licensee shall evaluate and propose

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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 5 above). "

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

Reference Item 11b above, involving Reactor Trip P-4 & Steam Generator High High Level P-14.

The NRC has observed potential situations of concern involving this interlock.

NRC Safety Concern A: A review of the logic of this interlock, Reference 7, Figure 7.2.1-1, (13 of 16), Revision 42 shows that if a SG-Hi Hi occurs, Turbine Trip, Trip of MFW Pumps, closure of MFW isolation and control valves occur, but the reactor is not tripped if the Nuclear Power Level is below P-8 (48% Power Level), Reference 7, Figure 7.2.1-1, Revision 42, (18 of 18). This would then cause another occurrence which would be effectively a loss of main feedwater to the reactor at a nominal power level of 48%.

NRC Safety Concern B: The existing FSAR, Reference 7, Section 15.2.10.1, Revision 15, shows that a feedwater malfunction at full power is not terminated by a neutron Flux Power trip, but by a SG-Hi Hi (i.e., P-14) signal initiating Turbine Trip, Trip of MFW Pumps, Closure of MFW Isolation and MFW modulating valves. Turbine Trip will trip the reactor (if initial power level is above P-8). However, if the feedwater malfunction is initiated at zero power FSAR, Reference 7, Section 15.2.10.2, "Results," first paragraph, the consequences are a rapid increase in nuclear power which will cause a reactor trip from the neutron flux low power, 25%, setpoint, and 35% (Limiting Safety Value in Analysis) and hence generate a P-4 signal, but will not correct the initiating cause of the faulted main feedwater control system until SG-Hi Hi level is subsequently initiated and effects closure of MFW isolation valves. Whereas the FSAR evaluates the first event of this sequence by reference to the event of "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From A Subcritical Condition," the FSAR provides no evaluation of the subsequent event including the DNBRs resulting from any restoration of reactivity before SG-Hi Hi ultimately effectively closes MFW isolation valves. This latter event from zero power can also occur at any intermediate power level, with and without automatic rod control, and there is currently no analysis which evaluates the worst case.

NRC Safety Concern C: The licensee has provided no information on "Safety Analysis Limits" that would be applicable to Permissive P-8 in evaluating the above events. If the allowance is ultimately of the same order as for the Power Range, Neutron Flux - High and Low Set Point Trips, i.e., approx. +10 percentage point, then Safety Concerns A and B could be occurring at up to 58% power level.

In respect of NRC Safety Concerns A, B, and C above, we consider the proposed T.S. in respect of the related permissives and interlocks to be non-conservative with respect to Regulatory Requirements. The licensee should review the safety consequences of each of these potential NRC concerns and respond with a safety evaluation with proposed changes to the T.S. as appropriate. This could be considered a Generic Issue.

General: In view of the consequences of the bypass of reactor trip on turbine trip below P-8 for the events protected by trip of turbine on

Steam Generator Hi Hi., the licensee should review the analyses for all other Condition II through IV occurrences to determine whether the conclusions deriving from the existing evaluations need to be altered. This could be considered a Generic Issue.

<u>Reference Item 11(a) above, involving Reactor Trip P-4 and Low T_{avg}.</u>

Reactor Trip P-4 together with Low-T_{avg} causes closure of the MFW isolation valves and MFW Modulating (Control valves) thereby isolating the reactor from any faulted [on non faulted] feedwater system.

The safety significance of the parameter, Low T_{avg}, as expressed in the FSAR derives (a) from its inclusion in the ESFAS under Reference 5, Figure 7.2.1-1, (13 of 16), Revision 34 and (b) a description in Reference 5, Section 7.7.1.7 under the title Steam Generator Water Level Control, in the following terms:

> "Continued delivery of feedwater to the steam generators is required as a sink for the heat stored and generated in the reactor following a reactor trip and a turbine trip. An override signal closes the feedwater-valves when the reactor coolant is below a given temperature, and the reactor has tripped. Manual override of the feedwater control system is available at all times."

This P-4/Low T_{avg} combination does perform a safety function in preventing excessive cooldown after the reactor is tripped, but has never been incorporated, or discussed in the Section 15 FSAR analyses (Reference 7) for this purpose.

Within the FSAR under Reference 7, Section 15.2.10.1 "Excessive HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS" state that:

> "An accidental full opening of one feedwater control valve with the reactor at zero power and the above mentioned assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in Subsection 15.2.1, Uncontrolled Control RCCA Bank Withdrawal from a Subcritical Condition, and therefore, the results of the analyses are not presented. It should be noted that if the incident occurs with the unit just critical at no load, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 25 percent."

> "For all excessive feedwater cases continuous addition of cold feedwater is prevented by closure of all feedwater control valves, a trip of the feedwater pumps, and closure of the feedwater pump discharge valves on steam generator high-level."

event from zero and higher power levels (already discussed under ...lier Item 11b) is initially protected by the high neutron fluxtrip; ...ever whilst this provides immediate protection, the main feedwater is ... isolated and continue to cooldown the reactor with continued reactivity addition. The licensee must confirm that acceptance criteria for the reactor system are not exceeded if further protection must wait for Steam

Generator Hi Hi Level to trip the MFW pumps, and together with existing Reactor Trip to provide Main Feedwater Isolation. Or, is it necessary to depend on an earlier "Isolation of Main Feedwater" from the combination of the existing reactor trip P-4 signal already provided and a related Low T_{avg}.

Inclusion of the P-4 and Low T_{avg} interlock into the T.S. would provide more reliability in protection for this event in conformance with the diversity criteria of 10 CFR 50 Appendix A, GDC Criterion 22 in support GDC 20. Without this, there is no diversity for protection from this continuing event. The proposed T.S. should require T_{avg} Low to be incorporated into the T.S. to meet the above Regulatory Criteria. The licensee shall evaluate and propose.

The licensee shall evaluate this issue with our concerns expressed under Table 3.3-4, Item 11 proposed, Reference Item 11(b) above, NRC Safety Concerns B and C to which this is directly related.

The presence of Low T_{avg}, without T.S. considerations of Set Point, Maximum Errors, Channel Reliability, Applicability MODES and Action Statements raises concerns about the consequences of a single failure. For example, a failure low, remaining undetected, could combine with a Reactor Trip from full power to close Main Feedwater [containment] Isolation valves and Main Feedwater Modulating valves and cause a more severe transient than would otherwise be necessary. The Licensee should evaluate the consequences of single failure on appropriate Conditions II, III, and IV Occurences, and propose as necessary.

Item: Reference 7, Section 15.2.14, page 15.2-38, Revision 43, which is the Accident Analysis for "Inadvertent Operation of ECCS During Power Operation," states that:

Spurious ECCS operation at power could be caused by operator error or a false electrical actuating signal. Spurious actuation may be assumed to be caused by any of the following:

1. High Containment pressure

2. Low pressurizer pressure

3. High steam line differential pressure

4. High steam line flow with either low average coolant temperature or low steam line pressure.

Please explain the signals 3 and 4 since they do not appear in the TABLE 3.3-4 just reviewed, nor do they seem to appear in the Logic Diagrams of the Licensing Basis in the FSAR to reference 5. The Licensee shall evaluate and propose.

Item": Reference 5, Figure 7.2.1-1 (2 of 16) Reactor Trip Signals

The reference to Safety Injection Signal (Sheet 8) is inaccurate. This signal is from the ESFAS and not directly from the SI signal.

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 2c: "Feedwater Isolation" from Containment Pressure-High

The response time is given as $\leq$ 9 secs.

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

Table 3.6.2 of the T.S. provides isolation times of $\leq$ 5 secs for main feedwater containment isolation and $\leq$ 10 secs for main feedwater to Auxiliary Feedwater Isolation. A total time to isolation of MFW, from Containment Pressure-High, of $\leq$ 11 secs seems appropriate to available equipment.

There would then be a conflict between the response time of $\leq$ 9 secs in the proposed T.S. and the potential value of up to 11 sec from other licensing basis information.

No event in Reference 7, Section 15.1 through 4. uses this particular isolation in time Analyses. However, this is a important factor for containment integrity during a Main Steam Line Break in containment. The value used as the Safety Analysis Limit shall be provided by the licensee, compared with proposed T.S. Item 2c and any differences evaluated, and T.S. proposed as appropriate.

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 2g: Initiation of Nuclear Service Water (NSW) from Containment Pressure-High

This response time is given as $\leq 65^{(3)}/76^{(4)}$ secs.

The superscript 3 does not seem appropriate; whilst the related Notation on T.S. Page 3/4 3-33 refers to absence of diesel delay (i.e., no loss of offsite power), it describes start up of ECCS equipment but does not include the requirement for "Isolation and Startup of Nuclear Service Water Pumps as described in Functional Unit 1 of Tables 3.3-3 and 3.3-4. The same comment applies t superscript 4 which applies to the circumstances without offsite power. The licensee should propose an accurate description of these circumstances; the current description does not meet the intent.

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

No other information is available on Safety Analysis Limits because, contrary to Regulatory Requirements, this value has not been used in the Safety Analysis of the FSAR in respect of AFW supplies. In other sections of this review, the licensee has been asked to re-evaluate Safety Analyses to recognize this fact. Parallel with this, the licensee shall identify the Actual Safety Analysis Limit to be used for this response, compare with the proposed T.S., and repropose as appropriate. Any Occurrences required to utilize Nuclear Service Water must be considered non-conservative with respect to these values currently presented in the FSAR to Reference 7, Section 15.

Item 2h: Initiation of Component Cooling Water from Containment Pressure-High

This response time is given as $65^{(3)(3)}/76^{(4)(2)}$ secs.

The description of superscript 2 under Table Notation on T.S. Page 3/4 3-33 is incomplete. The licensee shall propose an accurate description of these circumstances including its dependence on Nuclear Service Water; the licensee should confirm that this cooling water supply information is for this safety related service.

Reference 5, page 73-8 shows the initiation of ESFAS from this source is 1 sec.

No other information is available on Safety Analysis Limits used in the FSAR. The licensee shall provide this information for related Conditions II, III, and IV Occurrences for both on-site and offsite power. This information shall be evaluated and the licensee shall propose. At this time, considering the non-conservative circumstance with NSW AFW supply, it must be presumed that any Occurrence required to utilize the Nuclear Service Water must be considered non-conservative with respect to the values currently presented in the FSAR, Reference 7, Section 15.

Item 2i: "Start Diesel Generators" from Containment Pressure-High

A response time of $\leq$ 11 secs is given.

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

No evaluation in Reference 7, uses this sensor as the prime initiator above the P-11 Interlock, although it is relied upon for protection above, and directly for protection below [P-11]. Other events dependent upon a SI generating signal particularly, items 3a & 4a below, show safety analysis limits of $\leq$ 10 secs for this value.

In respect of current safety analyses limits, therefore, it appears that the proposed value is less conservative than the Safety Analysis Limits. The licensee shall evaluate and propose.

We note that Reference 5, page 8.3-6, describes testing of diesels on 11 second starts and if initiating times of 1 and 2 seconds were allowed for, this would mean actual times of 12 and 13 secs from the initiating signal. The licensee shall clarify, evaluate and propose.

Item 3: Pressurizer Pressure-Low

This title should be modified to read as Pressurizer Pressure-Low (Safety Injection) as Pressurizer Pressure-Low Is a Reactor Trip only.

The initiation time of all ESFAS Functions from this sensor is $\leq$ 1 sec (Reference 5, page 7.3-8). This is also the same initiation time for Containment Pressure-High. Since both or either of these initiators can be available in Occurrences involving SI, and initiation times are the same, our comments and conclusions under earlier Item 2 can be directly referenced for items under Item 3 in cases where the proposed response time is the same for a given ESFAS function.

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 3c: "Feedwater Isolation" From Pressurizer Pressure-Low (SI)

The proposed T.S. is $\leq$ 9 secs.

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

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 3g: "Nuclear Service Water System" Initiation from Pressurizer Pressure-Low SI

The T.S. value is given as $76^{(1)}/65^{(3)}$ secs.

Our comments on $65^{(3)}$ are as for our earlier 2g.

With respect to superscript$^{(1)}$ on 76; why is this different to Containment Pressure High which is $76^{(3)}$ when the concomitant SI signal generates the same equipment requirements. Superscript $^{(1)}$ now provides for SI and RHR pumps whereas $^{(3)}$ did not. Also, superscript $^{(1)}$, if it is to be used should include Isolation and Start of Nuclear Service Water System (NSW).

Reference our comments and requirements under earlier 2g.

Item 3: General

The licensee is to evaluate each of his superscripts $^{(1)}$, $^{(2)}$, $^{(3)}$ and $(_4)$ and ensure that they are complete, accurate and consistent with all the related ESFAS initiating signals and functions.

This position appears inaccurate & confusing to the extent that it must be considered non-conservative.

Item 3h: Initiation of Component Cooling Water from Pressurizer Pressure-Low (SI)

The proposed T.S. is $\leq 76^{(1)}/65^{(2)(3)}$ secs.

See our comments and requirements under 2h. and 3. General above.

Item 3i: Start Diesel Generators from Pressurizer Pressure-Low (SI)

The T.S. value is $\leq$ 11 secs.

See our comments under 2i. above which are substantively applicable to this item. Therefore, the proposed item is less conservative than the safety analysis limits; the licensee shall evaluate and propose.

Item 4: Steam Line Pressure-Low

The initiation time for all ESFAS functions for this sensor is given as $\geq$ 2.0 sec in Reference 5, page 7.3-8. This compares with only 1 sec for Item 2, Containment Pressure-High and Item 3, Pressurizer Pressure-Low (SI). Since again, all these 3 initiators can be available in occurrences involving SI, our comments and conclusions under 2 and 3 can be referenced with the condition that actual response times under item 4 could be 1 sec longer. We note however, that functional response times specified under 4 remain the same (in general) as under Items 3 and 2 and do not apparently provide for this differential of 1 sec. The licensee shall evaluate and propose.

Item 4a: "Safety Injection (ECCS)" Initiation on Steam Line Pressure-Low

These values of $\leq 12^{(3)}/22^{(4)}$ agree with the Safety Analysis Limits of the Licensing Basis FSAR.

Item 4b: "Reactor Trip (From SI)" from Steam Line Pressure-Low.

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

This value of $\leq$ 2 secs agrees with Reference 5, page 7.3-8.

Item 4c: "Feedwater Isolation" from Steam Line Pressure-Low

The proposed T.S. is $\leq$ 9 secs.

Reference our comment and requirements under 2c. above modified by the fact that there appears to be a larger conflict between the response time of $\leq$ 9 secs and the potential value of up to 11 + 1 = 12 seconds from Licensing Basis Information.

Item 4d: "Containment Isolation - Phase A" on Steam Line Pressure-Low

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

Reference our comments and requirements under 2d. above, modified in that proposed T.S. times appear feasible with the additional delay of 1 sec.

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 4f: "Auxiliary Feedwater Pumps" initiated by Steam Line Pressure-Low

The proposed T.S. is NA.

Reference our comments and requirements under 3f. above.

Item 4g: "Nuclear Service Water" initiated on Steam Line Pressure-Low

The proposed T.S. is $\leq 65^{(3)}/76^{(4)}$ secs.

Reference our comments, requirements, and remarks under 2g., 3g., and 3 General above.

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

The proposed TS value is $\leq$ 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 $\leq$ 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 4i: "Component Cooling Water" Initiation by Steam Line Pressure-Low

Proposed T.S. value is $65^{(2)(3)}/76^{(2)(4)}$.

Reference our earlier comments and requirements under 2h and 3h. above.

Item 4j: "Start Diesel Generators" by Steam Line Pressure-Low.

Proposed T.S. value is $\leq$ 11 secs.

Reference our comments and requirements under 2i above.

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

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

Item 5b: Containment Isolation - Phase B on Containment Pressure-High-High

This is proposed as Not-Applicable. The licensee should propose why this is so when it appears that TS Table 3.6-2 Containment Isolation valves, Maximum Isolation Time (secs), applies only to closure from receipt of signal, and may not include the ESFAS Response Time. Reference especially T.S. page 3/4 6-30 where main steam line isolation is specified at 5 secs compared with the same value quoted on Reference 7, page 15.4-6 which states that these fast acting steam line valves are designed to close in 5 secs and Safety Analysis Limits have been shown as 7 secs under Item 4h. above.

What is needed to supplement the information in T.S. Table 3.6-2 is the ESFAS response time as defined in Reference 5, page 7.3-7, Revision 36, and which values are quoted at 1.0 sec for initiation from containment pressure (related page 7.3-7), and also as 1 sec for closing main steam line stop valves on Containment Pressure-High [High]. It appears this item should read as:

5b. ESFAS Input to Containment Isolation - Phase B 1 sec

The licensee shall clarify, identify the related Safety Analysis Limits, and evaluate as appropriate. Until then, the proposed T.S. must be considered non-conservative with respect to the Licensing Basis.

Item 5c: Steam Line Isolation on Containment Pressure High-High

The proposed T.S. value is $\leq$ 9 secs.

Reference 5, page 3.7-8 shows containment pressure initiating ESFAS signals with a $\leq$ 1 response time. Item 4h. above shows fast acting stop valves closing in 5 secs. giving a total time of $\leq$ 6 secs.

Since MSIV actuation under Containment-Hi Hi can be caused by MSLB which provides for a maximum of 7 secs above, the proposed value of 9 secs appears less conservative.

A comparison also with values used in assessing environmental releases from containment should also be made.

The licensee shall identify the Safety Analysis Limits used for this Steam Line Isolation, including the MSLB in containment, evaluate against the proposed T.S. value and propose as appropriate. Until such time, the current value appears non-conservative.

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.

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

The proposed T.S. is $\leq$ 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 $\leq$ 5 secs for main feedwater containment isolation and $\leq$ 10 secs for main feedwater to Auxiliary Feedwater Isolation.

A total time to isolation of MFW of $\leq$ 13 secs seems appropriate to available equipment.

However 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 7a: "Motor-Driven Auxiliary Feedwater Pumps" initiated by SG Level-Low Low

Item 7b: "Turbine-Driven Auxiliary Feedwater Pumps" initiated by SG Level-Low Low

Proposed T.S. response times are given as $\leq$ 60 secs.

The FSAR Safety Analysis Limit is 61 secs; Reference 7, Table 15.4-1 (1 of 4) and 15.4-2 (2 of 4) where the difference between SG Low-Low and auxiliary feedwater delivered to steam generators is 61 secs. The current proposed T.S. value is therefore conservative with respect to the current safety analysis limit.

However, the current safety analysis limit of 61 secs currently used appears to be a mistake and not in accordance with Regulatory requirements.

The only safety related water source available for Auxiliary Feedwater, is the Nuclear Service Water System.

Reference 22, page 10.4-14a, states that "All three AFS pumps are normally supplied from a common leader which can be aligned to the upper surge tank, the auxiliary condensate storage tank, or the condenser hotwell. Each of these sources are provided with motor operated valves with control room operation. The assured AFS pump suction is from the Nuclear Service Water System. The A motor drive is aligned to the A NSWS header and the B motor driven pump is aligned to the B NSWS header. The turbine driven pump is aligned to both channels. Each source is provided with diesel aligned motor operated valves which open automatically on how suction pressure" [with a proposed T.S. response time of 13 secs].

Earlier information under this T.S. Table 3.3-5 shows that the response time for Nuclear Service Water Supply is 65 secs, assuming offsite power available and 76 secs assuming loss of offsite power whereas the Safety Analysis Limit used in the FSAR is only 61 secs. On this basis, all Conditions II, III, and IV occurrences involving AFW supply would need to be re-evaluated to establish acceptability.

The NRC does notice from Reference 5, Table 8.1.2.1 entitled "Maximum Loads to be supplied from one of the Redundant Essential Auxiliary Power Systems" that the related loading sequences for pumping equipment, alone, might enable an earlier response time then given in Table 3.3-5, e.g., Nuclear Service Water Pumps can be available 35 secs and AFW, 40 secs, after Blackout or LOCA signal [further, the Table notation of Table 3.3-5 is inadequate to clarify the position].

The licensee shall clarify the available response time for AFW supply from the Safety Related Nuclear Service Water system, and include the consequences of additional delays due to inadequate suction pressure under

Item 11, below. If this is confirmed at from 65 to 70 secs, or any longer time than used as the existing Safety Analysis Limit in the FSAR, then ~~acceptable re-~~evaluation of all Conditions II, III, and IV occurrences involving AFW supply, are required by 10 CFR 50.36.

Our current evaluation is that the response times in the proposed T.S. are non-conservative in respect of Regulatory requirements.

Item 8: "Steam Line Isolation" on Negative Steam Line Pressure Rate-High

Proposed T.S. value is $\leq$ 9 sec.

Reference 5, page 7.3-8 states that the maximum allowable time for generating the ESFAS MSIV isolation signal from a Steam Line Pressure Rate circumstance is 2 secs, the same as for item 4h. above.

Our comments and requirements therefore are the same as under item 4h.

We appreciate that this signal is generated at below P-11, but with the existing proposed Boration Control T.S. we must continue to evaluate this value as non-conservative.

The proposed T.S. value is greater than the Safety Analysis Limit of seven (7) secs and must be considered less conservative for this event. The licensee must evaluate this difference and propose.

Item 11: "Automatic Re-alignment of AFW Supply on Low Suction Line Pressure" [The existing description should be changed to more accurately state this action]

Proposed T.S. value is 13 secs.

Note our comments under 7a. and 7b. above. Although this response time may be in accordance with current plant engineering, it is not in accordance with the existing Safety Analysis Limit for Auxiliary Feedwater Supply which, on current information, has pre supposed no such transfer time. If a tank has been lost because of seismic action, we cannot assume a residual 15 secs supply at this time.

At this time, until the evaluation of 7a. and 7b. above is completed, we must evaluate this delay as non-conservative with respect to currently used Safety Analysis Limits which in themselves are non-conservative with respect to Regulatory requirements.

The licensee will evaluate and propose.

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.

Item 13: Station Blackout

Item 13: General

The Licensing Basis FSAR, reference 6, page 9.2-10 describes how station blackout causes startup of all Emergency diesel generators and alignment of [NSWS and CCW]. Why is this not included under this item 13 "Station Blackout."

The Licensing Basis FSAR, reference 7, Section 15.2.9 under LOSS OF OFF-SITE POWER TO THE STATION AUXILIARIES describes a set of Protection Actions for the plant, all which have related response times. Why is this information not provided under this heading?

The absence of most of the information on Functional Units and Related Response times required to protect the facility on Station Blackout conditions makes the proposed T.S. non-conservative with respect to the Licensing Basis. The Licensee shall evaluate and propose.

Item 13a: "Start Motor-Driven AFW Pumps" on Station Blackout

Item 13b: "Start Turbine-Driven AFW Pumps" on Station Blackout

Proposed T.S. response times are $\leq$ 60 secs.

Reference our comment under 7a. and 7b. above.

These values are non-conservative with respect to Regulatory requirements and the licensee shall evaluate and propose.

Item 14: "Start Motor-Driven Auxiliary Feedwater Pumps" on Trip of Main Feedwater Pumps

Proposed T.S. value is < 60 secs.

Reference our comments under 7a. and 7b. above together with the necessity for licensee action.

At this time, these values are non-conservative with respect to regulatory requirements, and the licensee shall evaluate and propose.

Item 15: Loss of Power: "4 Kv Emergency Bus Undervoltage-Grid Degraded Voltage."

Proposed T.S. response time of $\leq$ 11 secs.

Reference our comments under T.S. Table 3.3-3 Item 9 and Table 3.3-4 Item 9 and provide appropriate clarification.

No evaluation is possible at this time.

Item 15: Loss of Power

Item 15: General

Our review comments under item 13 "Station Blackout" are fully applicable to this item with the related conclusion that:

The absence of most of the information on Functional Units and related Response Times required to Protect the Facility on Loss of Power makes the proposed T.S. non-conservative with respect to the Licensing Basis. The Licensee shall evaluate and propose.

Item [Foot] Note: Response time for Motor-Driven Auxiliary Feedwater Pump Starts on All SI signals.

This is proposed as $\leq$ 60 secs.

Reference our earlier comments for its inclusion in Items 2f., 3f., and 4f. above together with the necessary Licensee Actions.

Reference our earlier comments under 7a. and 7b. above together with the necessity for licensee action.

At this time, these values are non-conservative with respect to Regulatory requirements and the licensee must evaluate and propose.

Item: Table 3.3-5, TABLE NOTATION on T.S. Page 3/4 3-33

These notations 1, 2, 3, and 4 must be expanded to include Component Cooling Water System Isolation and Pumps, Nuclear Service Water System (NSWS) Isolation & Pumps, and AFW re-alignment to NSWS and alternate sources as necessary. This will also enable verifiable consistency with the Notations used in the table.

See our comment under items 2g., 2h., 3g., 3h., 4g., and 4i. above.

Notation 2 of this Table states that:

(2) Valves 1KC305B and 1KC315B for Unit 1 and Valves 2KC305B and 2KC315B for Unit 2 are exceptions to the response times listed in the table. The following response times in seconds are the required values for these valves for the initiating signal and function indicated:

2.b	$< 30^{(3)}/40^{(4)}$
3.b	$\leq 30^{(3)}$
4.b	$\leq 30^{(3)}/40^{(4)}$

Since the functions 2b, 3b and 4b are all Reactor Trip functions, please explain.

Since these descriptors are apparently incorrect, provide the correct descriptors.

Since supercripts (3) and (4) used above make no mention of Component Cooling Water, [from which the valves derive] what do they mean?

What is meant by the Statement that the valves specified are exceptions to the response times listed in the Table. How do they affect the response times - do they increase, or decrease them, or have no effect. If they increase response time, by how much and what is the effect on the Actual overall response time, and has this been incorporated into the Safety Analysis of the Licensing Basis.

The Licensee shall clarify, evaluate and propose. Lack of accurate information on response times must be considered as non-conservative.

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).

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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 ≤ 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

G 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

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: G

> "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.

G 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

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

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 psi/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

G.2.6 EVENTS OF CONCERN (A LIMITED SELECTION)

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.

G. (W) (RSB) 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:

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

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

G. (W) (RSB) 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.

G (W) (RSB) 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.

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

G (W) RSB) 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 212-47d & 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." G (W) (RSB

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. G (W)(RS

. 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. G (W)(RS

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. G (W)(RS)

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. G (W)(RS.

Additional events which exhibit a rapid cooldown and depressurization 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.

G. (W) (RSB) 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.

G (W) (RSB) 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: G (W) (RA)

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). G (W) (RSB)

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

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. G. (W) (RSB)

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.

G (W) (RSB)

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:

G (W) (RSB)

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 evaluation as zero power.

G. (W) (RSB)

b) Start up of an Inactive Reactor Coolant Loop

G. (W)(RSB)

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

G (W)(RSB)

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

G (W)(RSB)

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 <u>CONCLUSIONS</u>

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.

G (w)(RSI

T.S. SECTION 3/4.4.1: RCS LOOPS AND COOLANT CIRCULATION

START UP (MODE 2) AND POWER OPERATION (MODE 1).

The LCO requires all [4] reactor coolant loops to be in operation in MODES 1 & 2.

The ACTION Statement requires that in the event of loss of 1 [of 4] RCS Loop in MODES 1 & 2, the licensee is required to be in at least HOT STANDBY within 1 hr.

The current Safety Analysis Limits in the FSAR, reference 7, page 15.2-16, revision 7, requires an immediate trip of the reactor to RTI & ESFAS response times in the event of loss of 1 RCS pump. Also, placement of the RCS in hot Standby with less than one loop operable [without other compensating conditions] would be non-conservative in respect of the existing FSAR.

The Action Statement is non-conservative with respect to the current licensing basis and the licensee shall evaluate and propose.

T.S. surveillance requires verification of Reactor Coolant Loop (RCL) circulation once every 12 hours. This is unacceptable considering the Safety Analysis limits required above for loss at one pump. In the event of failure of the Low Reactor Coolant Flow Reactor Trip; the operator should respond immediately to the related Alarm to trip the reactor, if it remains. Reference to earlier work of this review will show that there is no alternate, or diverse, sensor for low flow in one Reactor Coolant Loop. Further the FSAR analysis does not provide an evaluation of the consequences of a 10 min delay by the operator on hearing the Alarm - if it has remained operable from available [3 channel] LOGIC. Additionally, the FSAR proposes no alternate trips for the reactor, with related evaluation, such as over temperature leading to Pressurizer Level-High and Pressurizer Pressure-High. The Action Statement would place the plant outside the current licensing basis for normal operation and is non-conservative with respect to that. The licensee shall evaluate and propose.

Further it can be proposed, for this event analyzed in ref. 7, page 15.2-16, revision 7, that Criterion 22, Protection System Independence has not been met:

"Criterion 22- Protection system independence. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function."

The Facility is non-conservative with respect to this Regulation, the licensee shall evaluate and propose. This is a generic issue.

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 Functions. The proposed T.S. requirements are absent in this information; it is therefore non-conservative and the licensee shall evaluate and propose.

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

The current T.S. requires only 2 RCS loops to be in operation in this MODE 3. The basis for this requirement on TS Page B 3/4 4-1 says only: "In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however single failure considerations require that at least two loops be OPERABLE." This basis is unacceptable since the facility is required, within this condition of normal operation, and its existing licensing basis, to also be able to withstand related valid Condition II, III and IV occurrences; and earlier work has shown the Safety Analysis Limits for the plant currently requiring at least 4 RCS pumps for this MODE.

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

At this time, any No. of loops less than 4 in MODE 3 is non-conservative with respect to the existing FSAR and the plant should be transferred to operation in MODE 4 under these circumstances, with approved maximum normal cooldown rates.

It is recognized there are many protective actions which may provide more flexibility in this MODE within NRC/RCS Safety Criteria but they are not included within the current T.S. proposed by the licensee; further that final choice of such actions may be determined by "additional" protective procedures already in place at the plant, but not included in the T.S. where they are required by 10 CFR 50-36. Also, the particular combinations of protections which could be proposed may depend on providing the facility with maximum flexibility in other operations in this MODE 3 consistent with meeting Regulatory Safety requirement. See our earlier review under General.

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

It further follows, that the proposed surveillance requirement T. S. item 4.4.1.2.3 that at least one reactor coolant loop shall be verified in operation and circulating reactor coolant at least once 12 hours is also invalid and should be changed.

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

Surveillance requirements for the S.G. call for a level of 12% at least once per 12 hours. This is not in accordance with the Licensing Basis; this level is the S.G. Low - Low Trip Set Point. All conditions II, III and IV occurrences require in general, for this S.G. level to be at the programmed Set Point for the Zero Power Condition with automatic actuation; we have no evaluation at alternate conditions. Therefore this exlisting proposal is outside the current Licensing Basis and non-conservative. Reference our earlier comments under Item 2.1.1, Item f. The licensee shall evaluate and propose.

*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. Page 3/4 4-3. REACTOR COOLANT SYSTEM - HOT SHUTDOWN.

The proposed T.S. should be supplemented by the conditions contained within the brackets [ ]:

"3.4.1.3 At least two of the reactor coolant and/or residual heat removal (RHR) loops listed below shall be OPERABLE [and energized from separate power divisions] and at least one of the above reactor coolant and/or RHR loops shall be in operation:** [Additionally two RCS loops must always be OPERABLE whenever RHR loops are in operation]

a. Reactor Coolant Loop A and its associated steam generator [including related auxiliary feedwater pumps] and reactor coolant pump,*

b. Reactor Coolant Loop B and its associated steam generator [including related auxiliary feedwater pumps] and reactor coolant pump,*

c. Reactor Coolant Loop C and its associated steam generator, [including relating auxiliary feedwater pumps] and reactor coolant pump,*

d. Reactor Coolant Loop D and its associated steam generator, [including related auxiliary feedwater pumps] and reactor coolant pump,*

e. RHR Loop A,*** and

f. RHR Loop B.***

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

MPA (RSB)

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. MPA (RSB)

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 MPA (RSB)

"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."

The additional Action statement that:

b. "With no reactor coolant or RHR 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 coolant loop to operation."

and the additional notation that

"***All reactor coolant pumps and RHR 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."

are unsupportable by present analyses in the FSAR. These proposed T.S.s are the same as for MODE 3 and our relevant comments and requirements under T.S. Page 3/4 4-2: RCS HOT STANDBY should be applied to MODE 4. Emergency Operating Guidelines Apply. This proposed T.S. is non-conservative with respect to the Licensing Basis. The licensee shall provide the reason for the requirement including the expected condition of the facility, and then analyze evaluate and propose.

MPA (RSB) 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)."

MPA (RSB) 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.

MPA (RSB) 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. MPA (RSB

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.] MPA (RSB)

The FSAR addresses the consequence of a failure, closed, of the isolation valve in the RCS/RHR line; it addresses the analysis from 350°F in the RHR MODE when a bubble is present in the pressurizer. This will also be valid down to the RCS temperature at which the bubble will be established, i.e., below 300°F according to reference 19, page 52-21a, revision 33, first para. If the licensee does operate the plant so that the system is water solid between 200°F and 300°F in MODE 4, a loss of cooling could result in a potential overpressurization of the system and the reviewer is not aware of any evaluation of the adequacy of the existing Low Temperature Overpressure Protection System to accommodate that event. The licensee shall evaluate and propose.

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

The current proposed T.S. provides:

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 12%.

The current FSAR requires two (2) OPERABLE RHR trains on two (2) redundant electrical buses so that each pump receives power from a different source, reference 20, Pages 5.5-24. In the event of Loss of Offsite Power, the pumps are automatically transferred to a separate emergency diesel power supply. Therefore; the current licensing basis is that 2 residual heat removal loops shall be operable. The above provision for either an RHR loop or two steam generators is therefore not in accordance with the Licensing Basis. The proposed T.S. in this respect is also non-conservative as it would necessarily require S.G. temperatures greater than 212°F (Atmos Press in SGs) which would place it outside the Cold Shutdown MODE into the Hot Shutdown MODE - which is outside the required Functional MODE.

The T.S. requirement for one RHR loop in operation and one to be available OPERABLE is currently not supportable by analysis evaluating the situation in which all RHR cooling is lost in a water solid condition; reference our immediately preceeding item T.S Page 3/4 4-3. In this case, if one only RHR loop is operating, loss of that single loop cause overheating in a water solidstate with potential overpressurization. Does the alarm of loss of RHR Flow which is required, and an operator response time of 10 mins, provide sufficient time to commence operations of the second RHR loop to the extent necessary to mitigate the consequences of any potential overpressure event in an acceptable manner. The licensee shall evaluate and propose.

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. G

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. G

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. G

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. G

T.S. Page 3/4 4-6. REACTOR COOLANT SYSTEM - COLD SHUTDOWN, LOOPS ARE NOT FILLED

Item 3.4.1.4.2 requires that:

"3.4.1.4.2 Two residual heat removal (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 (2) redundant electrical buses so that each pump receives power from a different source; reference 20, pages 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 Q 212-57, revision 25, describes that in the event of loss of flow caused by isolation 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 conditons (maximum 24 hours decay heat, air in the steam generator tubes, 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, and the pressure would increase to 550 psi in about 40 minutes (the pressure rise could be limited to about 550 psi by opening the pressurizer power operated relief valves)."

In the event only 1 RHR loop is required to be in operation,the LCO should therefore require 2 operable Safety Related RHR flow alarms on each single operating RHR system so that the operator can respond within 10 mins to commence operation of the redundant system. However, this time frame is excessive since boiling will have commenced. It is necessary to maintain two operating RHR systems so that boiling may be eliminated on single failure. The licensee shall evaluate and propose.

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 Reactor Vessel flanges and which minimum volume shall be included in the T.S. as an LCO with appropriate surveillance and Action Statements. A further T.S. requirement is that any such min volume should be such that the level of water in or above the RCS loops be such as to provide 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 to the Licensing Basis. The licensee shall evaluate and propose.

Concerning Action item b., this provides that

b. With no RHR 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 RHR loop to operation.

Further: In the event that RHR cooling cannot be restored in "sufficient" time, the FSAR states that, in the event of loss of flow caused by the single RCS/RHR motorized valve:

> "To restore core cooling, the operator would first attempt to fill and pressurize the reactor coolant system with the centrifugal charging pumps. If the system can be pressurized to the range of 400-500 psi, the

operator could return the plant to heat removal via the steam generators. To do this the operator would have to jog the reactor coolant pumps to sweep the trapped air from the steam generators. He would also have to open the steam dump valves (to atmosphere or the main condenser) and start up the auxiliary feedwater system."

In this MODE therefore, it is necessary to ensure that 2 RCS loops with operable SG, AFW supply and SG/PORVs are operable from separate buses, to be available, in the event of the single failure discussed. This would also support the general concern in the event of noncapability of restoring failed RHR systems to Operability within an acceptable time frame, including the possibility of core uncovery in 100 mins. [The licensee shall also reference any Emergency Operating Guidelines in this respect]. Without provision for RCS Loop Operability required by the Licensing Basis FSAR, the current T.S. LCOs must be considered non-conservative with respect to the Licensing Basis, and the licensee shall evaluate and propose.

Item 4.4.1.4.2, A surveillance requirement, specifies:

At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

A time delay of 12 hours is excessive to verify a loop in operation, and this has been considered earlier in this section. Further 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, including instrumentation and control, which can be evaluated to show that the equipment is capable of performing its design basis Safety Function. The current requirements for this T.S. Item are absent in this information; it is therefore non-conservative and the licensee shall evaluate and propose.

Footnote*: Provides that,

"*The RHR pump 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 departure from the Licensing Basis of two available RHRs with effective cooling at all times it outside the FSAR Licensing Basis in a non-conservative manner. Further this is also supported by the earlier information of this section that boiling would commence in 5 minutes with core uncovery in 100 minutes. The provision is outside the Licensing Basis in a non-conservative manner and the licensee shall evaluate and propose.

T/S Page 3/4 4-6(a) Proposed.

A new subsection should be added entitled "REACTOR COOLANT SYSTEM, HOT SHUTDOWN TO REFUELING, APPLICABLE MODES 4, 5, & 6 which requires a LIMITING CONDITION OF OPERATION that two RHR Flow Alarms to Safety Related requirements shall be operable on each RHR loop when only one RHR loop is in operation under the provisions of the Technical Specifications. Appropriate Action Statements and surveillance requirements shall be applied.

The safety basis for this was established in the FSAR, as indicated in earlier sections, and the need for safety related redundancy arises to ensure RCS integrity to Safety Related Criteria as discussed above. The current T.S. is non-conservative with respect to the Licensing Basis.

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%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling MODE."

Reference our review comments and requirements under T.S. 3/4.4.2 SAFETY VALVES, OPERATING which are also applicable to this section. The current T.S. must be considered nonconservative with respect to the Licensing Basis. The Licensee shall evaluate and propose.

The Action statement is based (reference T.S. page B 3/4.4-2) on the premise that INOPERABILITY of the Safety Valve in Modes 4 and 5 needs to be offset by operability of pressure relief valves in the RHR systems. This is not the safety basis for Action. The safety basis is, that the Reactor Coolant Pressure Boundary has been effectively rendered inoperable requiring the operator to proceed to a cold shutdown condition with the zero pressure (gauge) in both RCS and SG systems, and related reactivity control actions to ensure that no return to nuclear power is possible. This needs to be done in a manner consistent with the nature of inoperability of the Safety Valve. The current T.S. is nonconservative with respect to the Licensing Basis; the licensee shall evaluate and propose.

Further, McGuire Units 1 and 2 do not use RHR overpressure protection of the RCS as the plant utilizes two available PORVs on the pressurizer, reset to 400 psig (reference review under T.S. Page 3/4 4-36) in the primary coolant system. In this respect, the proposed action statement is non-conservative and contrary to the Licensing Basis. The licensee shall evaluate and propose.

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. G RSR

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 Action Statement requires that

"ACTION:

With one pressurizer Code Safety Valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours."

Failure of the Pressurizer Code Safety Valve, in general, would infringe the integrity of the Reactor Coolant Pressure Boundary and the RCS should be brought to the cold shutdown condition, as rapidly as possible, with zero (gauge) pressure in both the RCS and SG, in a manner consistent with the nature of the inoperability, and potential for all positive reactivity levels eliminated.

The worst situation would be that of an "Accidental Depressurization of the Reactor "Coolant System" analyzed for the most severe conditions including maximum core power, reference 7, page 15.2-33 revision 7. This type of event would require Emergency Procedures to define the ACTION STATEMENT.

Could other types of failure allow other types of response which could be outside the Emergency Operating Procedures. The Licensee has not identified others and analyzed and evaluated the related safety to Regulatory Requirements as a basis for his proposed action.

The T.S. Bases on page B 3/4.4-2 does not exhibit an acceptable understanding of the importance of, and potential severity of, the event including failure types and appropriate Regulatory requirements including procedures.

The existing ACTION statement is inadequate within the Licensing Basis, and therefore unacceptable. The only existing Licensing Basis must be within the analyses reported in reference 7, page 15.2-33, revision 7, and the proposed Action Statement does not recognize these circumstances. The existing Action Statement is therefore nonconservative with respect to the Licensing Basis; the licensee shall evaluate and propose.

LCO and surveillance procedures must also address position indication and/or discharge flow measurement procedures, including pressurizer relief tank condition and other measures to ascertain the operability of the valve [this is necessary to satisfy 10 CFR 50 Appendix A, Criterion 20, 32 and 33]. The writer reviewed, in 1983, information pertaining to the GPU/B&W lawsuit review, and his recollection is that the TMI-2 operators "initially thought that the safety valves had developed a leak in the PORVs because the valves had lifted on a recent event." There must be a measure of acceptable leak tightness from measurable parameters "in operation" to ascertain the status of the valve so that acceptable measures can be taken.

The safety basis for the concern rests not only in the previous position addressed above, but also, that in the event of failure of control grade "pressure control devices" these valves will be challenged on the following occurrences within the Licensing Basis.

- Startup of the Inactive Coolant Loop; reference 7 Figure 15.2.6-1, revision 4
- Loss of Load Accident; reference 7, Figure 15.2.7-5, revision 38
- Loss of Normal Feedwater; reference 7, page 15.2-26, revision 7, para. 3
- Main Feedwater Line Break Accident, reference 7, Figure 15.4.2.7, revision 38
- One Locked Rotor Event; reference 7, Figure 15.4.4-1, revision 32

Safety Valve Operation could also occur on other overpressurization events if same of the early reactor trips fail to operate as expected.

In this matter, the T.S. is nonconservative with respect to Regulatory Requirements. The Licensee shall evaluate and propose. This could be a generic issue.

Surveillance Requirements should reference the documents containing the record of the Inservice Testing of the valves for inspection on a regular basis of 12 hours so that changing operating staff are kept aware of a potentially changing status on a singularly critical item.

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.

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(RSI

G (RSB)

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 MODE 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.

G (RSB)

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

G (RSB)

T.S. SECTION 3/4.4.4 RELIEF VALVES (POWER OPERATED)

The current T.S. provides that the plant may continue in operation if either one of the combination of Block Valve and PORV is INOPERABLE. This is a contravention of the regulations which provides under 10 CFR 50.2(v) that:

> (v)"Reactor coolant pressure boundary" means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves which are:
>
> (1) Part of the reactor coolant system, or
>
> (2) Connected to the reactor coolant system, up to and including any and all of the following:
>
> (i) The outermost containment isolation valve in system piping which penetrates primary reactor containment.
>
> (ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment.
>
> (iii) The reactor coolant system safety and relief valves.

Since a single failure of either the Block valve, or the PORV, will reduce the level of protection of the Reactor Coolant Pressure Boundary (RCPB) from two

(2) valves to one (1) only valve, the Regulatory Requirements are not met and the plant must proceed to a cold shutdown condition witn no potential for positive reactivity changes, within appropriate time frames.

The current T.S. is nonconservative in respect to Regulatory Requirements. The licensee shall evaluate and propose.

T.S. Section 3/4 4.5 STEAM GENERATORS

T.S. Page 3/4 4-11

a) S.G. Levels

G (RSB)

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 30.46. 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

G (RSB)

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:

G
RSB

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.

<u>T.S. Page 3/4 4-36 (REACTOR COOLANT SYSTEM) OVERPRESSURE PROTECTION SYSTEMS</u>

The current LCOs require that either of the following be Operable;

"(a) 2 PORVs with a lift setting of less than or equal to 400 psig, <u>or</u>

(b) The Reactor Coolant system (RCS) depressurized with an RCS vent of greater than, or equal to 4.5 square inches.

The Applicability is MODE 4 when the temperature of any RCS cold leg is less than or equal to <u>300°F</u>, MODE 5 and MODE 6 with the reactor vessel head on."

This section should also include the often used restraint that:

*A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 300°F unless: (1) the pressurizer water volume is less than 1600 cubic feet, or (2) the secondary water temperature of each steam generator is less than 50F° above each of the Reactor Coolant System cold leg temperatures.

It is necessary, to expand the LCOs to all those which should be incorporated into the operability requirements for the pressurizer and steam generator discussed earlier under T.S. Section 3/4.4.3 Pressurizer and T.S. Section 3/4.4.5 Steam Generators. This additional information defines necessary safety limits for the Licensing Basis event; as in reference 28, which is an early Topical Report submitted by <u>W</u> for approval. The proposed T.S. is nonconservative in the absence of this information. The licensee shall evaluate and propose.

Concerning the alternate provision that the RCS be depressurized with an RCS vent of greater than or equal to 4.5 square inches:

> We find that this should be confined only to MODE 5, COLD SHUTDOWN, LOOPS ARE NOT FILLED, and REFUELING OPERATIONS; MODE 6 HIGH WATER LEVEL and MODE 6 LOW WATER LEVEL. There are no safety analyses to support this type of operation in remaining MODES 4 and 5. The proposed TS, without this clarification, is nonconservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

We find no safety evaluation in the Licensing Basis for the alternate use of an RCS vent of greater than or equal to 4.5 square inches in the proposed T.S. 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 temperature of 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 wil 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 LCOA 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:

"5.2.2 Overpressure Protection

In the Safety Evaluation Report we indicated a concern about the possibility of reactor vessel damage as a result of overpressurization when the reactor coolant system is water-solid during startup and shutdown. We have reviewed the applicant's system for overpressure protection when the reactor coolant system is water-solid. It consists of two separate trains each containing a power-operated relief valve set to open when the system pressure reaches 400 pounds per square inch gauge should an overpressure event occur. Each train contains an annunciator which sounds to alert the operator when plant conditions require enabling of the water-solid overpressure protection system; enabling is performed manually, by turning key-lock switch. The system is automatically disabled when plant conditions no longer require it; an annuciator sounds to indicate the system is no longer needed so that the operator may turn the key-lock to disable the system until needed. In addition, each train contains an annuciator which sounds when the power-operated relief valve is open, indicating an overpressure transient is in process.

Each power-operated relief valve is supplied with nitrogen from the cold leg accumulators. No operator action is required in the event of a transient. The operator isolates the upper head injection system, the cold leg accumulators, the safety injection pumps and one centrifugal charging pump before the reactor coolant system is cooled to 300 degrees Fahrenheit; only the remaining centrifugal charging pump could cause an overpressure transient as a result of inadvertent start with concomitant mass addition. The only other overpressure event would result from an inadvertent main coolant pump start with the coolant in the secondary side of the steam generator hotter than that in the reactor coolant system. The applicant has shown that in neither case was 10 CFR Part 50, Appendix G limit reached. For the latter case (that for main coolant pump inadvertent start), the applicant assumed that the temperature of the fluid in the steam generator would exceed that in the reactor coolant system by no greater than 50 degrees Fahrenheit.

The staff requires that the technical specifications require that the reactor coolant system may not be cooled to temperatures lower than 300 degrees Fahrenheit without the overpressure protection system enabled, and unless both power-operated relief valve trains are operable, in order to assure suitable overpressure protection for the reactor coolant system when water-solid. In addition, the technical specifications will state that the temperature of the fliuid in the secondary side of the steam generator will not exceed the temperature of the fluid in the reactor coolant system by greater than 50 degrees Fahrenheit when the reactor coolant system fluid temperature is less than 300 degrees Fahrenheit since the applicant's calculations did not assume differences greater than 50 degrees Fahrenheit.

The applicant provided data to show that the power-operated relief valve opens within the time specified in the analyses.

The system meets the single failure criteria as only one of the two trains is required for overpressure mitigation. Means are provided to test and calibrate the system. It has been designed in accordance with the Institute of Electrical and Electronics Engineers Standard 279-1971, "Criteria for Protection Systems."

This system meets the staff requirements for an overpressure protection system with the reactor coolant system water-solid and is acceptable. We consider this matter resolved."

The required status of the ECCS systems required by the existing Licensing Basis FSAR are briefly summarized:

Above 1900 psig (in MODES 1, 2, and 3): All ECCS systems are OPERABLE. Between 1900 psig and 1000 psig/425°F; upper head injection isolation valves area closed and gagged, de-energized and tagged. Between 1000 psig/425° F and 425 psig/350° F (in MODE 3): Upper head injection isolation valves remain closed and gagged and de-energized; cold leg accumulator isolation valves are closed and breakers racked out, 1 centrifugal and 1 reciprocating charging pump and 2 safety injection pumps are isolated, and rendered inoperable by opening and locking the related circuit breakers. Below 425 psig/350° (in MODES 4 and 5) status of all ECCS systems remain unchanged, i.e., same (as for the preceding phase of MODE 3) with the exception that remaining equipment is re-aligned for RHR operation with the capability of re-alignment to ECCS. [UHI, Cold Leg Accumulators, 1 cent. CP & 1 Recip. CP, and 2 SI pumps are effectively electrically isolated.] RHR PORVs are rendered operable during water solid operation, below 300°F.

These requirements are substantially different from those of the W STS which the licensee has adopted for his facility contrary to his Licensing Basis as disclosed in the FSAR and SER to the above references.

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

G (RSB)

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.

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

An additional T.S. Section is proposed that provides specifically for the fact that "COLD LEG INJECTION ACCUMULATOR ISOLATION VALVES" at "APPLICABLE CONDITIONS" of MODE 3 (< 1000 psig/425°F), MODE 4 and MODE 5 would have a "LIMITING CONDITION OF OPERATION" providing that "Each Cold Leg Injection Accumulator Isolation Valve is closed with circuit breakers opened, locked and tagged." Appropriate Action Statements and Surveillance Procedures would be provided. This is in accord with the LCOs of the Licensing Basis FSAR as described under 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 proposed T.S. non-conservative. The licensee shall evaluate and propose.

T.S. Page 3/4 5-3. UPPER HEAD INJECTION

Item: APPLICABILITY MODE.

The Applicability Mode given as MODES 1, 2, and 3* where * signifies Pressurizer Pressure above 1900 psig, should be amended to include >425°F; as 1900 psig/>425°F.

The FSAR does not include the temperature constraint explicitly at 1900 psig, though it is implicit in that the next lower boundary for change is 1000 psig/425°F [Reference earlier Item: T.S. 3/4.5 under GENERAL]. Absent this condition, the related proposed T.S. is non-conservative. Appendix G curves (T.S. Page 3/4 4-32) would allow RCS temperatures down to <300°F, and one of the reasons for isolating UHI below 1900 psig, includes overpressure concerns at the reducing levels of temperature down to 425°F, reference 12, page 7-1. From his detailed analysis, the licensee should evaluate and propose a lower limit to this temperature condition of >425°F.

Item 3.5.1.2.c Nitrogen cover pressure is specified as between 1206 and 1264 psig. The Licensing Basis FSAR, reference 29, page (1 of 5), revision 39 in Table 6.3.2-1 specifies a normal operating pressure of 1220-1280 psig with a minimum of 1220 psig. Making an allowance for channel error and drift, should not T.S. setpoints be higher [at say 1240-1300 psig]. The specified minimum set point values in the proposed T.S. of 1206 would therefore require lower pressure in the RCS before actuation and is therefore non-conservative. The licensee shall evaluate and propose.

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 respect 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, a... 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)

G (RSB)

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 ($\leq$ 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.

<u>T.S. Section 3/4.5.2 ECC SUBSYSTEMS -Tavg $\geq$ 350°F</u>

The title should be amended to read as:

ECCS SUBSYSTEMS - PRESSURIZER PRESSURE $\geq$ 1000 psig/RCS Tavg$\geq$425°F

The Operability requirements of 2 full trains of ECCS equipment remains unchanged.

Absence of the pressure/temperature condition in the proposed T.S. is not in accordance with Safety Analysis Limits. Its absence permits high pressure pump operation at lower pressures and temperatures with potential infringement of related safety criteria. Related safety criteria have not been well defined, or docketed, but are apparently considerations of Low Temperature Overpressure Protection of the RCS under these and related Accident circumstances including inadvertent operation of ECCS pumps. This diversion from the Safety Analysis

Limits of the Licensing Basis FSAR must therefore be considered non-conservative and the licenseee 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 3/4.5.2.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 Overprrssure 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.

T.S. Section 3/4.5.3 ECCS Subsystem - Tavg $\leq$ 350°F

This title should be amended to read ECCS Subsystems - 425 psig/350°F to COLD SHUTDOWN

The current T.S. provides no pressure condition on the temperature of 350°F, and Appendix G Limit curves of proposed T.S. Page 3/4 4-32 would permit "maximum

RCS pressures" of 2485 psig under these circumstances. Also the proposed T.S. alignment eliminates safety injection and charging pump capacity. There is no available evaluation of the capability of the reduced ECCS system to satisfactorily mitigate the consequences of a Small Break or Large Break LOCA from 2485 psig/350°F as is provided for the values of 425 psig/350°F within the Licensing Basis as described earlier under T.S. 3/4.5, Item: GENERAL. Our evaluation is that the absence of this pressure condition is non-conservative, and especially with respect to the Safety Analysis Limits of the Licensing Basis. The Licensee shall evaluate and propose.

The proposed limit at COLD SHUTDOWN MODE 5 is conditioned by the fact that Refueling is a condition of a vented vessel with Reactor Vessel Bolts untensioned, and non-ECCS alignments are proposed to deal with related events. Reference 8 pages Q212-56 revision 25 under the Titles of Case 1 and Case 2 and page Q 212-57, revision 25, under the Title of Case 3. Overpressure Protection also, which is a principal determinant of alignment, also ceases with untensioning the Reactor Vessel bolts for refueling.

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.

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 LCOs 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.

The LCOs 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.

Additionally, that all breakers for all safety injection pumps and all but the one operable centrifugal charging pump are opened, locked and tagged. (reference earlier information) The proposed T.S. is therefore less conservative than the Licensing Basis FSAR by being deficient in ECCS total pumping capacity, and excessive in available high pressure pumping capacity so infringing LTOP. The licensee shall evaluate and propose.

Additionally the Licensing Basis requires that ech of these subsystems be independent and receive power from two (2) redundant Emergency Buses and Power Sources. The absence of any such provision in the proposed T.S. makes it 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.

G 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.

Item: The ACTION Statement should be clarified to include [ ] that in the event of inoperablity of the BIT tank, the RCS be borated to [a boron concentration which will give] a SHUTDOWN margin of 1% delta k/k at 200°F.

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

Comment: Since BIT concentrations of only 2000 ppm, only are now required, and only 900 gallons are involved compared with 372,100 gallons in the R.W.S.T, is not the proposed ACTION statement to ultimately place the plant in HOT SHUTDOWN overly conservative; if minimum volumetric requirements are necessary, can additional provision be made in the RWST. The licensee may evaluate and propose.

T.S. Section 3/4.5.5 REFUELING WATER STORAGE TANK

Item: APPLICABLITY MODES 1, 2, 3, 4.

The current MODES 1, 2, 3 and 4 which includes an LCO for 372,100 gallons must be extended to MODE 5 and MODE 6 (limited) to meet the FSAR requirements in reference 8, pages Q 212-57 and 58, revision 25, item: Case 3: [when] The RCS is depressurized and vented with the air in the steam generator tubes, with the reactor vessel head on, and tensioned - and later with open relief paths between the head and the reactor vessel cavity and refueling canal. The single failure of an RHR/RCS Isolation valve is resolved by the expected Operability of the RWST providing 5 hours of injection flow. The recovery description also means that the RWST must be available in MODE 6 until the vessel head is removed and the refueling canal is filled to its specified level. It must also be available at termination of core alterations - in Mode 6, when drainage of the refueling canal commences until the Reactor Vessel Head is tensioned, when the RCS then moves into MODE 5. The proposed T.S. is non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

Action Statement: The proposed ACTION should be modified [ ] as follows:

With the RWST Inoperable, restore the tank to OPERABLE status within 1 hour, or be in at least HOT STANDBY [and borated to a boron concentration which will give a shut down margin of 1% delta k/k at 200°F and a minimum of 2000 ppm] within [the next] 6 hours and in COLD SHUTDOWN within the following 30 hours.

The Licensing Basis FSAR requires Safety Injection of 2000 ppm Boron to mitigate the nuclear power consequences of any accidents which may initiate during this period; if the RWST is not available, then Boron Concentration in the RCS should be increased to the level required to mitigate any potential return of nuclear power. The proposed T.S. appears nonconservative.

The licensee shall evaluate and propose and in so doing he should evaluate each of the Operability requirements separately to determine if COLD SHUTDOWN is required for each INOPERABILITY REQUIREMENT, or whether alternate mitigating Actions are possible.

T.S. Section 3/4.7 PLANT SYSTEMS

T.S. Page 3/4 7-1: SAFETY VALVES

The proposed T.S. requires that:

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specfied in Table 3.7-3.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With four reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With three reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves associated with an operating loop inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN with the following 30 hours.

Our concerns in this section are parallel to those in our review under T.S. Section 3/4.4.2 SAFETY VALVES.

Failure of Steam Generator Code Safety Valves infringe basic safety criteria for Reactor Protection through its impact on SG/RCS system response under Condition II, III, and IV occurrences. It also affects the integrity of the Primary Containment Boundary.

We do not find an adequate consideration of the alternate type of Safety Valve Failure that can occur, and their related significance, upon the action statements proposed.

How sure is the Licensee that inadequacy to meet the very limited single operability requirement of the T.S. does not represent an intermittent problem leading to early opening of valves, failure to close, or failure to open under the severe conditions of Transient and Accident Events.

We find the proposed T.S. inadequate in its representation of operability, or lack there of, for these Safety Valves. Consequently, without a requirement that they all be operable in MODES 1, 2, 3, and 4, with a further requirement to be in cold shutdown in the event of failure, there of, we must consider the proposed T.S. non-conservative. The Licensee shall evaluate and propose.

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.

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 availabiilty 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.

The licensee should 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

Condition III and especially Condition IV Occurrences of the Licensing Basis, and if the existing Accident Analyses remain valid.

Item 4.7.1.2: SURVEILLANCE REQUIREMENTS

The Technical Specifications, page T.S. 3/4 7-4 requires each motor driven (MD) AFW pump to supply 450 pgm at greater than or equal to 1210 psig. This is at entrance to the Steam Generators according to the T.S. Basis on T.S. page B 3/4 7-2.

However, we note that the FSAR Accident Evaluation; reference 7, section 15.4.2.2.2, and the description of the AFW system in reference 5, refer to a total supply of 450 gpm from MDAFW pumps to three intact steam generators.

Further, this is parallel with a description in the Accident Analysis on page 15.4 - 13 a (Revision 38) in which the MDAFW pump headered to two intact steam generators supplies 170 gpm each whilst the one headered to the faulted Steam Generator suppies 110 gpm to the intact steam generator.

The SER supplement, reference 14, page 10-2 requires that the licensee confirm the capability of each of the Motor Driven and Turbine Driven AFW Pump systems to meet the flow distribution requirements of that particular Safety Evaluation Report, with a faulted steam generator associated with the ruptured main feedline and a second steam generator (SG) faulted with a failed open code Safety Valve or SG PORV, and both these SGs supply the Turbine Driven AFW pump. The Licensee committed to establish and verify by test, the valve throttle positions necessary to achieve this, during the initial startup test programs.

In addition, under SER supplement, reference 15, page 22-15, under the title of <u>Recommendation GS-6</u> the licensee agreed to propose Technical Specifications to assure that prior to plant startup following an extended shudown, a flow test would be performed to verify the normal flowpath from the primary AFW system to the steam generator. The flow test should be conducted with AFW system valves in their normal alignment.

At this time, we do not see a proposed T.S. which ensures that the required subdivision of flow between 3 intact and 1 faulted steam generator, and 2 Intact and 2 "Faulted" Steam Generators associated with the Turbine-Driven AFW Pump, required by the Licensing Basis is achieved, and we do not see any test period recommended such as following an extended cold shutdown to ensure that the required flow division is maintained in an acceptable manner. At this time we must conclude that the current T.S. is nonconservative in respect to the Licensing Basis. The licensee shall evaluate and propose.

<u>T.S. Page 3/4 7-5c Proposed: CONDENSATE STORAGE TANK SYSTEMS</u>

It is proposed that a new item be added to the Technical Specifications to the above title and to include an LCO providing "The Condensate Storage Tank System (CTS) comprising available usable storage from the upper surge tank, auxiliary feedwater condensate storage tank and condenser hot well shall be operable with a contained water volume of at least 175,000 gallons of water.

APPLICABILITY MODES proposed are 1, 2 and 3, with lesser volumes required in MODES 4 and 5.

ACTION STATEMENT should include a provision that, with the condensate storage tank inoperable, within 4 hours either

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

Demonstrate the OPERABILITY of the Nuclear Service Water System and Standby Nuclear Source Water Pond (alternate water source) as a backup supply, and align to the auxiliary feedwater pumps, and restore the condensate storage tank to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS should include

a. The condensate storage tank system shall be demonstrated OPERABLE at least once per 12 hours by appropriate measures when the tank is the supply source for the auxili[illegible] feedwater pumps.

b. The Nuclear Service Water System and Standby Nuclear Source Water Pond shall be demonstrated OPERABLE at least once per 12 hours by appropriate measures.

Additionally, an evaluation of and provision will need to be made concerning potential loss of AFW supplies during loss of suction and change-over to alternate AFW sources.

The safety basis for these requirements are

a. Our earlier review under TS. Table 3.3-5 Items 7a and 7b show that whereas all safety evaluations involving AFW supply have assumed a Safety Analysis Limit of 61 sec. response time, this is only available from nonsafety related water sources. Further, that the safety related supply from the Nuclear Service Water Pond may take an extra 15 secs which is substantially non-conservative in respect of the related safety analysis.

Therefore, at this time, until the licensee has evaluated our concerns and made acceptable proposals, the NRC will require technical specifications on this non safety-related water storage of the above nature. The proposed T.S. are nonconservative with respect to Regulatory Requirements. The licensee shall evaluate and propose.

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

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.

G 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.

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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.

The minimum number of valves 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
G 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. G

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:

9.2.4 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.

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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

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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 3/4.7.5 STANDBY NUCLEAR SERVICE WATER POND (SNSWP)

Item 3.7.5.b, an LCO, should be amended to read that the nuclear service water pond shall be operable with

> "an average water temperature of not less than 70°F or greater than 94°F ....in the intake structure"

The Licensing Basis FSAR, reference 6, page 9.2 - 12(a), revision 39, item 39, provides for an allowable maximum of 94° which meets both maximum allowable temperatures for all Safety Related Components including NPSH requirements (reference 6, page 9.2-13, last para).

An average water temperature of 70°F has been selected by RSB as a potential design basis for Condition II, III and IV occurrences. The licensee has provided little information on the range of AFW temperatures used in his analyses and the related sensitivity of results to AFW temperature variations. In the Major Rupture of A Main Feedline, reference 7, page 15.4 - 13, it is stated that a "relatively cold (120°F) AFW temperature was used (after purging the feedwater lines)." "Excessive Heat Removal" analyses in reference 7, page 15.2 - 29, uses a "conservatively low feedwater temperature of 70°F."

We note that reference 6, page 9.2-13, revision 39, item 8 discusses ice formation on the surface of the pond which would imply near freezing temperatures for water supply. At this time, we have no record of any Safety Analysis being undertaken at such low inlet temperatures and on this basis we must consider any such low value as non-conservative.

The licensee will advise the range of AFW temperatures used in Condition II, III and IV events, their sensitivity to AFW temperature values, and from this his bases for setting any alternate values proposed to the water temperatures in the standby nuclear service water pond. The proposed TS maximum value of 78°F is conservative with respective to certain Accident Analyses; the lack of a minimum temperature of 70°F including possible near-freezing temperatures must be considered as nonconservative in respect of certain events. The Licensee shall evaluate and propose.

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. G

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. 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. G

In the Licensing Basis FSAR reference 8, page Q 212-24, item 212.57, it is required that the reactor makeup water pumps shall be removed from the loads supplied by the emergency power supplies. This is to prevent inadvertent boron dilution during certain Occurrences in which electrical loads are disconnected from, and returned to, the Emergency Buses. Provision should be made so that at the end of refueling, before start-up, a surveillance procedure will confirm that this Licensing Basis FSAR requirement continues to be met. Absence of confirmation of this LCO is a non-conservative condition; 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 LCO provides that:

3.9 8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

The Licensing Basis, reference 20, Page 5.5-23, under Refueling, and page 5.5-24 under 5.5.7.3.1, System Availability and Reliability, last paragraph, shows the licensing of the RHR system is never based on only one RHR system being operable. Two are always to be available. This proposal is therefore outside the LCO for the FSAR in a non-conservative manner. The Licensee shall evaluate and propose

In his Basis, on T.S. Page 3/4 9-2, last para., the licensee has proposed that:

> "With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core."

In the FSAR, reference 8, page Q 212-56 under Case 2, it has been estimated that on loss of all RHR Cooling due to a fail closed RHR/RCS isolation valve, it will take 2½ hours for the available water inventory to boil. In that case, a number of alternates are proposed to resolve the situation and almost invariably, electric power is required, and in most cases the RHR equipment is used. If the basis for the licensee's request here is to enable him to operate with only one available electrical bus, it is unacceptable, as the loss of one operable RHR on loss of the only available electrical bus, with containment isolation required in 2½ hours, has not been evaluated. At this time we have no acceptable safety basis for allowing the proposed deviation from the Limiting Conditions of Operation of the Licensing Basis FSAR which is that 2 RHR loops from separate emergency buses be operable. The proposal is therefore non-conservative and the licensee must evaluate and propose.

Furthermore, the licensee must provide that the level of water in or above the loops be such as to provide acceptable flow, including NPSH conditions, at inlet to the RHR pumps. Absent those required conditions from the Limiting Conditions of Operation could make them non-conservative. The licensee shall evaluate and propose.

G (W)

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.

The current surveillance requirement:

> 4.9.8.1 "At least one RHR loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours."

is deficient in that the thermal performance of any one RHR system to Licensing Basis safety requirements is not being verified. The T.S. is therefore non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

Footnote *: The licensee also proposes that,

> "The [only operable] 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."

The licensee shall provide the basis for this proposal including safety evaluation, any related compensating actions, and a related proposal. [It should be noticed that such an action could increase pool temperature by 35° and in so doing decrease the available response to handle a loss of cooling capacity from 2½ hours down to 1½ hours, and for a considerable period of time thereafter whilst temperatures are again being reduced to the required value of 140°F.] This proposed T.S. is outside the Licensing Basis in a nonconservative manner. 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.

G (W)

T/S Page 3/4 9-12 REFUELING OPERATIONS

The subtitle should read as 3/4.9.9 HIGH WATER LEVEL

Clarify by addition of the term HIGH

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

APPLICABILITY: MODE 6 when the water level above the top of the reactor vessel flange is less than 23 feet.

GENERAL REVIEW: Whereas the existing FSAR under reference 20, page 5.1-7 discusses Refueling, it does not provide for a sustained period of normal operations under these Low Water Level conditions. The FSAR provides that:

> "Refueling
>
> Before removing the reactor vessel head for refueling, the system temperature has been reduced to 140°F or less and hydrogen and fission product levels have been reduced. The Reactor Coolant System is then drained until the water level is below the reactor vessel flange. The vessel head is then raised as the refueling canal is flooded. Upon completion of refueling, the system is refilled for startup."

Furthermore, we find that the FSAR analyses of the single failure of the RHR/RCS isolation valve is not predicated upon operations at "Low Water Level" so that no specific analyses and/or protective actions have not been developed for these circumstances. However analyses have been undertaken for the water inventories and temperatures in the RCS system that might apply under those conditions. Presumably therefore, the "OPERATING MODE - LOW LEVEL" is a long term changing condition following Cold Shutdown, with loops drained and bolts tensioned changing to bolts untensioned and removal of the head, as concomitant flooding of the reactor vessel cavity continues. At this time therefore, we cannot presume that the consequences of the case of single failure of the RHR/RCS isolation valve used as Case 3 in FSAR reference 8, page Q21-57, does not also apply under this MODE. We will use these consequences to evaluate.

Further, since this is effectively a long term changing condition, in the FSAR, it is not acceptable to allow some of the provisions requested such as one hour for the performance of CORE ALTERATIONS--which by T.S 3/4 9.9 are only permissible under that specification with at least 23 feet of water over the reactor vessel flange.

G

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.

G

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.

G

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.

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.

G (RSB)

Item 4.9.8.2, a surveillance requirement, specifies:

"At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours."

A time delay of 12 hours is excessive to verify a loop in operation, and this has been considered earlier in this section.

Further, 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, including instrumentation and control, which can be evaluated to show that the equipment is capable of performing its Licensing Basis safety function. The current requirements for this item are absent most of this information; it is therefore non-conservative and the licensee shall evaluate and propose.

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.

G (W)

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

G (RSB)

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.

c) That in RHR MODE 4, the RHR system must be capable of automatic re-alignment to the ECCS mode with residual ECCS equipment, in the event of a SI signal, including automatic closure of the RCPB Isolation valves on the RHR/RCS Suction Line in accordance with 10 CFR 50 App A Criterion 55(4) and subsequent automatic opening of valves to the RWST in accordance with 10 CFR 50 App A, Criterion 20 [with appropriate provision for RHR pump protection].

The current position in respect of c above appears to be absent those requirements and therefore non-conservative. The Licensee shall evaluate and propose.

The T.S. should provide the LCOs and surveillance in the overpressurization protection system of the RHR system as described in Licensing Basis FSAR, reference 3, page 5-5-24.

Proposed T/S Page 3/4 5-6, item 4.5.2.d, 1) b) appears incorrect: it provides that, in establishing ECCS operability:

d. At least once per 18 months by:

1) Verifying automatic isolation and interlock action of the RHR System from the Reactor Coolant System by ensuring that:

a) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened, and

b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to 560 psig the interlocks will cause the valves to automatically close.

Item b) above is incorrect in that it should ensure that with a simulated or actual Reactor Coolant System pressure signal greater than 475 psig, the interlocks will cause the valves to automatically close, reference 4, section 5.5.7.3.3 and reference 5, section 7.4.1.5.4. and is therefore non-conservative

A The proposed T.S. closes the valves when they are in fact required to be open and is therefore non-conservative. Further, the lower pressure of
m / 475 psig required to close is more conservative than a valve of 560 unless there are Set Point and Channel considerations - The pressure is less conservative than the Licensing Basis FSAR value.

interlocks will cause the valves to automatically close, reference 4, section 5.5.7.3.3 and reference 5, section 7.4.1.5.4.

The proposed T.S. closes the valves when they are in fact required to be open and is therefore non-conservative. Further, the lower pressure of 475 psig required to close is more conservative than a valve of 560 unless there are Set Point and Channel considerations - The pressure is less conservative than the Licensing Basis FSAR value.

LIST OF REFERENCES

1. Letter from H. B. Tucker (D.P.Co) to H. R. Denton (NRC) dated September 27, 1982 to the subject of "McGuire Nuclear Station."

2. Memo from C. O. Thomas (SSPB) to Brian W. Sheron (RSB) on the subject of "Proof and Review of McGuire - Units 1 and 2, Technical Specifications." Dated January 14, 1983.

3. U.S. Nuclear Regulatory Commission, Final Safety Analysis Report, Volume 4, Duke Power Company, McGuire Nuclear Station, Units 1 and 2.

4. U.S. Nuclear Regulatory Commission, Final Safety Analysis Report, Volume 5, Duke Power Company, McGuire Nuclear Station, Units 1 and 2, Rev. 45.

5. U.S. Nuclear Regulatory Commission, Final Safety Analysis Report, Volume 7, Duke Power Company, McGuire Nuclear Station, Units 1 and 2, Rev. 45.

6. U.S. Nuclear Regulatory Commission, Final Safety Analysis Report, Volume 8, Duke Power Company, McGuire Nuclear Station, Units 1 and 2, Rev. 45.

7. U.S. Nuclear Regulatory Commission, Final Safety Analysis Report, Volume 10, Duke Power Company, McGuire Nuclear Station, Units 1 and 2, Rev. 45.

8. U.S. Nuclear Regulatory Commission, Final Safety Analysis Report, Volume 11, Duke Power Company, McGuire Nuclear Station, Units 1 and 2, Rev. 45.

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11. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Safety Evaluation Report, McGuire Nuclear Station Units 1 and 2, Duke Power Company," NUREG-0422, Supp. 1, on Docket Nos. 50-369 and 50-370, May 1978.

12. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Safety Evaluation Report, McGuire Nuclear Station, Units 1 and 2, Duke Power Company," NUREG-0422, Supp. No. 2, on Docket Nos. 50-369 and 50-370, March 1979.

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15. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Safety Evaluation Report, McGuire Nuclear Station Units 1 and 2, Duke Power Company," NUREG-0422, Supp. No. 5, on Docket Nos. 50-369 and 50-370, April 1981.

16. Memo from R. W. Houston to T. M. Novak on the subject of "Staff Review and Input to SER Supplement No. 6 for McGuire Nuclear Station Units 1 and 2". Dated February 08, 1983.

17. Letter from H. B. Tucker (D.P.Co) to H. R. Denton (NRC) on the subject of McGuire Nuclear Station, Units 1 and 2, filing amendment No. 71 to its Application for License for the McGuire Nuclear Station and Submitting Revision 45 to the Final Safety Analysis Report. Dated February 16, 1983.

18. Letter from W. O. Parker (D.P.Co) to H. R. Denton (NRC), dated Oct. 8, 1981 on the subject of McGuire Nuclear Station, Unit 1 and submitting copies of Report identified as "Westinghouse Reactor Protection System/Engineered Safety Features Actuation System Setpoint Methodology, Duke Power Company, McGuire Unit 1," by C. R. Tuley et al. and dated April 1981, published by Westinghouse Electric, Nuclear Energy Systems, PROPRIETARY.

19. Westinghouse Electric Corporation, PWR Systems Division "Westinghouse Emergency Core Cooling System - Plant sensitivity studies, WCAP-8356. August 1,1974.

20. U.S. Nuclear Regulatory Commission, Final Safety Analysis Report, Volume 4, Duke Power Company, McGuire Nuclear Station, Units 1 and 2, Rev. 45.

21. Letter from T. M. Novak (NRC) to H. B. Tucker (D.P.Co), dated May 17, 1983 on the subject of OL Condition 2.C.(11)g, Anticipatory Reactor Trip (II.K.3.10) (McGuire Nuclear Station, Unit 1).

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26. Memo for RSB, CPB, ICSB Members from Brian W. Sheron (RSB), Carl H. Berlinger (CPB), Faust Ross (ICSB) dated April 12, 1983 on the Subject of Inadvertent Boron Dilution Events.

27. Westinghouse Electric Corporation, Nuclear Energy Systems Topical Report, Overpressure Protection for Westinghouse Pressurized Water Reactors, WCAP-7769, Rev. 1, June 1972.

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29. U.S. Nuclear Regulatory Commission, Final Safety Analysis Report, Volume 6, Duke Power Company, McGuire Nuclear Station, Units 1 and 2, Rev. 45.

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mined that there are no unresolved safety issues relating to the additional activities that may be authorized pursuant to this paragraph that would constitute good cause for withholding authorization.

(4) Any activities undertaken pursuant to an authorization granted under this paragraph shall be entirely at the risk of the applicant and, except as to matters determined under paragraphs (e)(2) and (e)(3)(ii), the grant of the authorization shall have no bearing on the issuance of a construction permit with respect to the requirements of the Act, and rules, regulations, or orders promulgated pursuant thereto.

(Secs. 101, 185, 68 Stat. 936, 956, as amended (42 U.S.C. 2131, 2235; sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1242, Pub. L. 94-79, 89 Stat. 413 (42 U.S.C. 5841); sec. 161 as amended, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201))

[21 FR 355, Jan. 19, 1956, as amended at 25 FR 8712, Sept. 9, 1960; 33 FR 2381, Jan. 31, 1968; 35 FR 11460, July 7, 1970; 37 FR 5748, Mar. 21, 1972; 39 FR 14508, Apr. 24, 1974; 39 FR 26279, July 18, 1974; 39 FR 33202, Sept. 16, 1974; 42 FR 22887, May 5, 1977; 43 FR 6924, Feb. 17, 1978]

§ 50.11 Exceptions and exemptions from licensing requirements.

Nothing in this part shall be deemed to require a license for:

(a) The manufacture, production, or acquisition by the Department of Defense of any utilization facility authorized pursuant to section 91 of the Act, or the use of such facility by the Department of Defense or by a person under contract with and for the account of the Department of Defense:

(b) Except to the extent that Administration facilities of the types subject to licensing pursuant to section 202 of the Energy Reorganization Act of 1974 [1] are involved:

(1)(i) The processing, fabrication or refining of special nuclear material or the separation of special nuclear material, or the separation of special nuclear material from other substances by a prime contractor of the Department under a prime contract for:

(A) The performance of work for the Department at a United States government-owned or controlled site:

(B) Research in, or development, manufacture, storage, testing or transportation of, atomic weapons or components thereof; or

(C) The use or operation of a production or utilization facility in a United States owned vehicle or vessel; or

(ii) By a prime contractor or subcontractor of the Commission or the Department under a prime contract or subcontract when the Commission determines that the exemption of the prime contractor or subcontractor is authorized by law; and that, under the terms of the contract or subcontract, there is adequate assurance that the work thereunder can be accomplished without undue risk to the public health and safety;

(2)(i) The construction or operation of a production or utilization facility for the Department at a United States government-owned or controlled site, including the transportation of the production or utilization facility to or from such site and the performance of contract services during temporary interruptions of such transportation; or the construction or operation of a production or utilization facility for the Department in the performance of research in, or development, manufacture, storage, testing, or transportation of, atomic weapons or components thereof; or the use or operation of a production or utilization facility for the Department in a United States government-owned vehicle or vessel: *Provided*, That such activities are conducted by a prime contractor of the

[1] The Department facilities identified in section 202 are:

(1) Demonstration Liquid Metal Fast Breeder reactors when operated as part of the power generation facilities of an electric utility system, or when operated in any other manner for the purpose of demonstrating the suitability for commercial application of such a reactor.

(2) Other demonstration nuclear reactors, except those in existence on January 19, 1975, when operated as part of the power generation facilities of an electric utility system, or when operated in any other manner for the purpose of demonstrating the suitability for commercial application of such a reactor.

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Department under a prime contract with the Department.

(ii) The contruction or operation of a production or utilization facility by a prime contractor or subcontractor of the Commission or the Department under his prime contract or subcontract when the Commission determines that the exemption of the prime contractor or subcontrator is authorized by law; and that, under the terms of the contract or subcontract, there is adequate assurance that the work thereunder can be accomplished without undue risk to the public health and safety.

(c) The transportation or possession of any production or utilization facility by a common or contract carrier or warehousemen in the regular course of carriage for another or storage incident thereto.

[40 FR 8788, Mar. 3, 1975]

§ 50.12 Specific exemptions.

(a) The Commission may, upon application by any interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest.

(b) Any person may request an exemption permitting the conduct of activities prior to the issuance of a construction permit prohibited by § 50.10. The Commission may grant such an exemption upon considering and balancing the following factors:

(1) Whether conduct of the proposed activities will give rise to a significant adverse impact on the environment and the nature and extent of such impact, if any;

(2) Whether redress of any adverse environment impact from conduct of the proposed activities can reasonably be effected should such redress be necessary;

(3) Whether conduct of the proposed activities would foreclose subsequent adoption of alternatives; and

(4) The effect of delay in conducting such activities on the public interest, including the power needs to be used by the proposed facility, the availability of alternative sources, if any, to meet those needs on a timely basis and delay costs to the applicant and to consumers.

Issuance of such an exemption shall not be deemed to constitute a commitment to issue a construction permit. During the period of any exemption granted pursuant to this paragraph (b), any activities conducted shall be carried out in such a manner as will minimize or reduce their environmental impact.

[37 FR 5748, Mar. 21, 1972, as amended at 39 FR 26279, July 18, 1974; 40 FR 8789, Mar. 3, 1975]

§ 50.13 Attacks and destructive acts by enemies of the United States; and defense activities.

An applicant for a license to construct and operate a production or utilization facility, or for an amendment to such license, is not required to provide for design features or other measures for the specific purpose of protection against the effects of (a) attacks and destructive acts, including sabotage, directed against the facility by an enemy of the United States, whether a foreign government or other person, or (b) use or deployment of weapons incident to U.S. defense activities.

[32 FR 13445, Sept. 26, 1967]

Classification and Description of Licenses

§ 50.20 Two classes of licenses.

Licenses will be issued to named persons applying to the Commission therefor, and will be either class 104 or class 103.

§ 50.21 Class 104 licenses; for medical therapy and research and development facilities.

A class 104 license will be issued, to an applicant who qualifies, for any one or more of the following: to transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, or use.

(a) A utilization facility for use in medical therapy; or

(b)(1) A production or utilization facility the construction or operation of

(4) The information described in paragraphs (a)(1) and (2) of this section shall be submitted as a separate document prior to any other part of the license application as provided in paragraph (b) and in accordance with § 2.101 of this chapter.

(b) Except as provided in paragraph (d), any person who applies for a class 103 construction permit for a nuclear power reactor on or after July 28, 1975 shall submit the document titled "Information Requested by the Attorney General for Antitrust Review" at least nine (9) months but not more than thirty-six months prior to the date of submittal of any part of the application for a class 103 construction permit.

(c) [Reserved]

(d) Any person who applies for a class 103 construction permit for a nuclear power reactor pursuant to the provisions of § 2.101(a-1) and Subpart F of Part 2 of this chapter shall submit the document title "Information Requested by the Attorney General for Antitrust Review" at least nine (9) months but not more than thirty-six months prior to the filing of part two or part three of the application, whichever part is filed first, as specified in § 2.101(a-1) of this chapter.

(e) Any person who applies for a class 103 construction permit for a uranium enrichment or fuel reprocessing plant shall submit such information as may be requested by the Attorney General for antitrust review, as a separate document as soon as possible and in accordance with § 2.101 of this chapter.

(Sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1242, Pub. L. 94-79, 89 Stat. 413 (42 U.S.C. 5841))

[39 FR 34395, Sept. 25, 1974, as amended at 42 FR 22887, May 5, 1977; 42 FR 25721, May 19, 1977; 43 FR 49775, Oct. 25, 1978; 44 FR 60716, Oct. 22, 1979]

§ 50.34 Contents of applications; technical information.

(a) *Preliminary safety analysis report.* Each application for a construction permit shall include a preliminary safety analysis report. The minimum information[1] to be included shall consist of the following:

(1) A description and safety assessment of the site on which the facility is to be located, with appropriate attention to features affecting facility design. Special attention should be directed to the site evaluation factors identified in Part 100 of this chapter. Such assessment shall contain an analysis and evaluation of the major structures, systems and components of the facility which bear significantly on the acceptability of the site under the site evaluation factors identified in Part 100 of this chapter, assuming that the facility will be operated at the ultimate power level which is contemplated by the applicant. With respect to operation at the projected initial power level, the applicant is required to submit information prescribed in paragraphs (a)(2) through (8) of this section, as well as the information required by this paragraph, in support of the application for a construction permit.

(2) A summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

(3) The preliminary design of the facility including:

(i) The principal design criteria for the facility.[2] Appendix A, General Design Criteria for Nuclear Power Plants, establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units;

[1] The applicant may provide information required by this paragraph in the form of a discussion, with specific references, of similarities to and differences from, facilities of similar design for which applications have previously been filed with the Commission.

[2] General design criteria for chemical processing facilities are being developed.

(ii) The design bases and the relation of the design bases to the principal design criteria.

(iii) Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.

(4) A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of (i) the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and (ii) the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of § 50.46 of this part for facilities for which construction permits may be issued after December 28, 1974.

(5) An identification and justification for the selection of those variables, conditions, or other items which are determined as the result of preliminary safety analysis and evaluation to be probable subjects of technical specifications for the facility, with special attention given to those items which may significantly influence the final design: *Provided, however*, That this requirement is not applicable to an application for a construction permit filed prior to January 16, 1969.

(6) A preliminary plan for the applicant's organization, training of personnel, and conduct of operations.

(7) A description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the requirements for quality assurance programs for nuclear power plants and fuel reprocessing plants. The description of the quality assurance program for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of Appendix B will be satisfied.

(8) An identification of those structures, systems, or components of the facility, if any, which require research and development to confirm the adequacy of their design; and identification and description of the research and development program which will be conducted to resolve any safety questions associated with such structures, systems or components; and a schedule of the research and development program showing that such safety questions will be resolved at or before the latest date stated in the application for completion of construction of the facility.

(9) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.

(10) A discussion of the applicant's preliminary plans for coping with emergencies. Appendix E sets forth items which shall be included in these plans.

(11) On or after February 5, 1979, applicants who apply for construction permits for nuclear powerplants to be built on multiunit sites shall identify potential hazards to the structures, systems and components important to safety of operating nuclear facilities from construction activities. A discussion shall also be included of any managerial and administrative controls that will be used during construction to assure the safety of the operating unit.

(b) *Final safety analysis report.* Each application for a license to operate a facility shall include a final safety analysis report. The final safety analysis report shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and shall include the following:

(1) All current information, such as the results of environmental and meteorological monitoring programs, which has been developed since issu-

ance of the construction permit, relating to site evaluation factors identified in Part 100 of this chapter.

(2) A description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

(i) For nuclear reactors, such items as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent.

(ii) For facilities other than nuclear reactors, such items as the chemical, physical, metallurgical, or nuclear process to be performed, instrumentation and control systems, ventilation and filter systems, electrical systems, auxiliary and emergency systems, and radioactive waste handling systems shall be discussed insofar as they are pertinent.

(3) The kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in Part 20 of this chapter.

(4) A final analysis and evaluation of the design and performance of structures, systems, and components with the objective stated in paragraph (a)(4) of this section and taking into account any pertinent information developed since the submittal of the preliminary safety analysis report. Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of § 50.46 for facilities for which a license to operate may be issued after December 28, 1974.

(5) A description and evaluation of the results of the applicant's programs, including research and development, if any, to demonstrate that any safety questions identified at the construction permit stage have been resolved.

(6) The following information concerning facility operation:

(i) The applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements.

(ii) Managerial and administrative controls to be used to assure safe operation. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the requirements for such controls for nuclear power plants and fuel reprocessing plants. The information on the controls to be used for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of Appendix B will be satisfied.

(iii) Plans for preoperational testing and initial operations.

(iv) Plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components.

(v) Plans for coping with emergencies, which shall include the items specified in Appendix E.

(vi) Proposed technical specifications prepared in accordance with the requirements of § 50.36.

(vii) On or after February 5, 1979, applicants who apply for operating licenses for nuclear powerplants to be operated on multiunit sites shall include an evaluation of the potential hazards to the structures, systems, and components important to safety of operating units resulting from construction activities, as well as a description of the managerial and administrative controls to be used to provide assurance that the limiting conditions for operation are not exceeded as a result of construction activities at the multiunit sites.

(7) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.

(8) A description and plans for implementation of an operator requalifi-

connected to the containment atmosphere. (II.E.4.1)

(vii) Provide a description of the management plan for design and construction activities, to include: (A) The organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort. (II.J.3.1)

(g) *Conformance with the Standard Review Plan (SRP).* (1)(i) Applications for light water cooled nuclear power plant operating licenses docketed after May 17, 1982 shall include an evaluation of the facility against the Standard Review Plan (SRP) in effect on May 17, 1982 or the SRP revision in effect six months prior to the docket date of the application, whichever is later.

(ii) Applications for light water cooled nuclear power plant construction permits, manufacturing licenses, and preliminary or final design approvals for standard plants docketed after May 17, 1982 shall include an evaluation of the facility against the SRP in effect on May 17, 1982 or the SRP revision in effect six months prior to the docket date of the application, whichever is later.

(2) The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for a facility and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where such a difference exists, the evaluation shall discuss how the alternative proposed provides an acceptable method of complying with those rules or regulations of Commission, or portions thereof, that underlie the corresponding SRP acceptance criteria.

(3) The SRP was issued to establish criteria that the NRC staff intends to use in evaluating whether an applicant/licensee meets the Commission's regulations. The SRP is not a substitute for the regulations, and compliance is not a requirement. Applicants shall identify differences from the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the Commission's regulations.

(Secs. 161b, 161i, Pub. L. 83-703, 68 Stat. 948, secs. 201, 204(b)(1), Pub. L. 93-438, 88 Stat. 1242, 1243, 1245 (42 U.S.C. 2201, 5841, 5844); sec. 7, Pub. L. 93-377, 88 Stat. 475; sec. 161i, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201))

[33 FR 18612, Dec. 17, 1968, as amended at 34 FR 6037, Apr. 3, 1969; 34 FR 6770, Apr. 23, 1969; 35 FR 10499, June 27, 1970; 35 FR 19567, Dec. 24, 1970; 36 FR 3256, Feb. 20, 1971; 36 FR 4861, Mar. 13, 1971; 36 FR 18201, Sept. 11, 1971]

EDITORIAL NOTE: For additional FEDERAL REGISTER citations affecting § 50.34 see the List of CFR sections Affected in the Finding Aids section of this volume.

§ 50.34a Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors.

(a) An application for a permit to construct a nuclear power reactor shall include a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences. In the case of an application filed on or after January 2, 1971, the application shall also identify the design objectives, and the means to be employed, for keeping levels of radioactive material in effluents to unrestricted areas as low as is reasonably achievable. The term "as low as is reasonably achievable" as used in this part means as low as is reasonably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeco-

(b) A construction permit will constitute an authorization to the applicant to proceed with construction but will not constitute Commission approval of the safety of any design feature or specification unless the applicant specifically requests such approval and such approval is incorporated in the permit. The applicant, at his option, may request such approvals in the construction permit or, from time to time, by amendment of his construction permit. The Commission may, in its discretion, incorporate in any construction permit provisions requiring the applicant to furnish periodic reports of the progress and results of research and development programs designed to resolve safety questions.

(c) Any construction permit will be subject to the limitation that a license authorizing operation of the facility will not be issued by the Commission until (1) the applicant has submitted to the Commission, by amendment to the application, the complete final safety analysis report, portions of which may be submitted and evaluated from time to time, and (2) the Commission has found that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the facility in accordance with the requirements of the license and the regulations in this chapter.

(Sec. 185, 68 Stat. 955; 42 U.S.C. 2235)

[27 FR 12915, Dec. 29, 1962, as amended at 31 FR 12780, Sept. 30, 1966; 35 FR 5318, Mar. 31, 1970; 35 FR 6644, Apr. 25, 1970; 35 FR 11461, July 7, 1970]

§ 50.36 Technical specifications.

(a) Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section. A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications.

(b) Each license authorizing operation of a production or utilization facility of a type described in § 50.21 or § 50.22 will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to § 50.34. The Commission may include such additional technical specifications as the Commission finds appropriate.

(c) Technical specifications will include items in the following categories:

(1) *Safety limits, limiting safety system settings, and limiting control settings.* (i)(A) Safety limits for nuclear reactors are limits upon important process variables which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor shall be shut down. The licensee shall notify the Commission, review the matter and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude reoccurrence. Operation shall not be resumed until authorized by the Commission.

(B) Safety limits for fuel reprocessing plants are those bounds within which the process variables must be maintained for adequate control of the operation and which must not be exceeded in order to protect the integrity of the physical system which is designed to guard against the uncontrolled release of radioactivity. If any safety limit for a fuel reprocessing plant is exceeded, corrective action shall be taken as stated in the technical specification or the affected part of the process, or the entire process if required, shall be shut down, unless such action would further reduce the margin of safety. The licensee shall notify the Commission, review the matter and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude reoccurrence. If a portion of the process or the entire process has been shut down, operation shall not be resumed until authorized by the Commission.

(ii)(A) Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor. He shall notify the Commission, review the matter and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude reoccurrence.

(B) Limiting control settings for fuel reprocessing plants are settings for automatic alarm or protective devices related to those variables having significant safety functions. Where a limiting control setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that protective action, either automatic or manual, will correct the abnormal situation before a safety limit is exceeded. If, during operation, the automatic alarm or protective devices do not function as required, the licensee shall take appropriate action to maintain the variables within the limiting control-setting values and to repair promptly the automatic devices or to shut down the affected part of the process and, if required, to shut down the entire process for repair of automatic devices. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude reoccurrence.

(2) *Limiting conditions for operation.* Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met. When a limiting condition for operation of any process step in the system of a fuel reprocessing plant is not met, the licensee shall shut down that part of the operation or follow any remedial action permitted by the technical specification until the condition can be met. In the case of either a nuclear reactor or a fuel reprocessing plant, the licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude reoccurrence.

(3) *Surveillance requirements.* Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within the safety limits, and that the limiting conditions of operation will be met.

(4) *Design features.* Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of this section.

(5) *Administrative controls.* Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

(d)(1) This section shall not be deemed to modify the technical specifications included in any license issued prior to January 16, 1969. A license in which technical specifications have not been designated shall be deemed to include the entire safety analysis report as technical specifications.

(2) An applicant for a license authorizing operation of a production or utilization facility to whom a construction permit has been issued prior to January 16, 1969, may submit technical specifications in accordance with this section, or in accordance with the requirements of this part in effect prior to January 16, 1969.

(3) At the initiative of the Commission or the licensee, any license may be amended to include technical specifications of the scope and content which would be required if a new license were being issued.

§ 50.38 Ineligibility of certain applicants.

Any person who is a citizen, national, or agent of a foreign country, or any corporation, or other entity which the Commission knows or has reason to believe is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government, shall be ineligible to apply for and obtain a license.

(Sec. 161, as amended, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1243 (42 U.S.C. 5841))

[21 FR 355, Jan. 16, 1956, as amended at 43 FR 6924, Feb. 17, 1978]

§ 50.39 Public inspection of applications.

Applications and documents submitted to the Commission in connection with applications may be made available for public inspection in accordance with the provisions of the regulations contained in Part 2 of this chapter.

STANDARDS FOR LICENSES AND CONSTRUCTION PERMITS

§ 50.40 Common standards.

In determining that a license will be issued to an applicant, the Commission will be guided by the following considerations:

(a) The processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals, in regard to any of the foregoing collectively provide reasonable assurance that the applicant will comply with the regulations in this chapter, including the regulations in Part 20, and that the health and safety of the public will not be endangered.

(b) The applicant is technically and financially qualified to engage in the proposed activities in accordance with the regulations in this chapter. However, no consideration of financial qualifications is necessary for an electric utility applicant for a license for a production or utilization facility of the type described in § 50.21(b) or § 50.22.

(c) The issuance of a license to the applicant will not, in the opinion of the Commission, be inimical to the common defense and security or to the health and safety of the public.

(d) Any applicable requirements of Part 51 have been satisfied.

[21 FR 355, Jan. 19, 1956, as amended at 36 FR 12731, July 7, 1971; 39 FR 26279, July 18, 1974; 47 FR 13754, Mar. 31, 1982]

§ 50.41 Additional standards for class 104 licenses.

In determining that a class 104 license will be issued to an applicant, the Commission will, in addition to applying the standards set forth in § 50.40 be guided by the following considerations:

(a) The Commission will permit the widest amount of effective medical therapy possible with the amount of special nuclear material available for such purposes.

(b) The Commission will permit the conduct of widespread and diverse research and development.

(c) An application for a class 104 operating license as to which a person who intervened or sought by timely written notice to the Commission to intervene in the construction permit proceeding for the facility to obtain a determination of antitrust considerations or to advance a jurisdictional basis for such determination has requested an antitrust review under section 105 of the Act within 25 days after the date of publication in the FEDERAL REGISTER of notice of filing of the application for an operating license or December 19, 1970, whichever is later, is also subject to the provisions of § 50.42(b).

(42 U.S.C. 2132-2135, 2239)

[21 FR 355, Jan. 19, 1956, as amended at 35 FR 19660, Dec. 29, 1970]

§ 50.42 Additional standards for class 103 licenses.

In determining whether a class 103 license will be issued to an applicant, the Commission will, in addition to applying the standards set forth in § 50.40, be guided by the following considerations:

(a) The proposed activities will serve a useful purpose proportionate to the quantities of special nuclear material or source material to be utilized.

(b) Due account will be taken of the advice provided by the Attorney General, pursuant to subsection 105c of the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. If a purge system is used as part of the repressurization system, the purge system shall be designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. The containment shall not be repressurized beyond 50 percent of the containment design pressure.

(g) For facilities with respect to which the notice of hearing on the application for a construction permit was published on or before December 22, 1968, if the combined radiation dose at the low population zone outer boundary from purging (and repressurization if a repressurization system is provided) and the postulated LOCA calculated in accordance with § 100.11(a)(2) of this chapter is less than 25 rem to the whole body and less than 300 rem to the thyroid, only a purging system is necessary, provided that the purging system and any filtration system associated with it are designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. Otherwise, the facility shall be provided with another type of combustible gas control system (a repressurization system is acceptable) designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. If a purge system is used as part of the repressurization system, it shall be designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. The containment shall not be repressurized beyond 50 percent of the containment design pressure.

(h) As used in this section: (1) Degradation, but not total failure, of emergency core cooling functioning means that the performance of the emergency core cooling system is postulated, for purposes of design of the combustible gas control system, not to meet the acceptance criteria in § 50.46 and that there could be localized clad melting and metal-water reaction to the extent postulated in paragraph (d) of this section. The degree of performance degradation is not postulated to be sufficient to cause core meltdown.

(2) A combustible gas control system is a system that operates after a LOCA to maintain the concentrations of combustible gases within the containment, such as hydrogen, below flammability limits. Combustible gas control systems are of two types: (i) Systems that allow controlled release from containment, through filters if necessary, such as purging systems and repressurization systems, and (ii) systems that do not result in a significant release from containment such as recombiners.

(3) A purging system is a system for the controlled release of the containment atmosphere to the environment through filters if needed.

(4) A repressurization system is a system used to dilute the concentration of combustible gas within containment by adding inert gas or air to the containment. Dilution of the combustible gas results in a delay in time until a flammable concentration is reached and permits fission product decay. Operation is limited to a containment repressurization to 50 percent of the containment design pressure. A purging system is normally part of the repressurization system.

(Sec. 161, as amended, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1242, Pub. L. 94-79, 89 Stat. 413 (42 U.S.C. 5841))

[43 FR 50163, Oct. 27, 1978, as amended at 46 FR 58486, Dec. 2, 1981]

§ 50.45 Standards for construction permits.

An applicant for a license or an amendment of a license who proposes to construct or alter a production or utilization facility will be initially granted a construction permit, if the application is in conformity with and acceptable under the criteria of §§ 50.31 through 50.38 and the standards of §§ 50.40 through 50.43.

§ 50.46 Acceptance criteria for emergency core cooling systems for light water nuclear power reactors.

(a)(1) Except as provided in paragraph (a)(2) and (3) of this section, each boiling and pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy cladding shall be provided with an emergency core cooling system (ECCS) which shall be designed such that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance shall be calculated in accordance with an acceptable evaluation model, and shall be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated loss-of-coolant accidents is covered. Appendix K, ECCS Evaluation Models, sets forth certain required and acceptable features of evaluation models. Conformance with the criteria set forth in paragraph (b) of this section with ECCS cooling performance calculated in accordance with an acceptable evaluation model, may require that restrictions be imposed on reactor operation.

(2) With respect to reactors for which operating licenses have previously been issued and for which operating licenses may issue on or before December 28, 1974:

(i) The time within which actions required or permitted under this paragraph (a)(2) must occur shall begin to run on February 4, 1974.

(ii) Within six months following the date specified in paragraph (a)(2)(i) of this section an evaluation in accordance with paragraph (a)(1) of this section shall have been submitted to the Director of Regulation of the Atomic Energy Commission. The evaluation shall have been accompanied by such proposed changes in technical specifications or license amendments as may be necessary to bring reactor operation in conformity with paragraph (a)(1) of this section.

(iii) Any licensee may have requested an extension of the six-month period referred to in paragraph (a)(2)(ii) of this section for good cause. Any such request shall have been submitted not less than 45 days prior to expiration of the six-month period, and shall have been accompanied by affidavits showing precisely why the evaluation is not complete and the minimum time believed necessary to complete it. The Director of Regulation of the Atomic Energy Commission shall have caused notice of such a request to be published promptly in the FEDERAL REGISTER; such notice shall have provided for the submission of comments by interested persons within a time period established by the Director of Regulation. If, upon reviewing the foregoing submissions, the Director of Regulation concluded that good cause had been shown for an extension, he may have extended the six-month period for the shortest additional time which in his judgment will be necessary to enable the licensee to furnish the submissions required by paragraph (a)(2)(ii) of this section. Requests for extensions of the six-month period submitted under this subparagraph will have been ruled upon by the Director of Regulation prior to expiration of that period.

(iv) Upon submission of the evaluation required by paragraph (a)(2)(ii) of this section (or under paragraph (a)(2)(iii), if the six-month period is extended) the facility shall continue or commence operation only within the limits of both the proposed technical specifications or license amendments submitted in accordance with this paragraph (a)(2) and all technical specifications or license conditions previously imposed by the Atomic Energy Commission, including the requirements of the Interim Policy Statement (June 29, 1971, 36 FR 12248) as amended December 18, 1971, 36 FR 24082).

(v) Further restrictions on reactor operation will be imposed if it is found that the evaluations submitted under paragraphs (a)(2) (ii) and (iii) of this section are not consistent with paragraph (a)(1) of this section and as a result such restrictions are required to protect the public health and safety.

(vi) Exemptions from the operating requirements of paragraph (a)(2)(iv) of this section may be granted for good cause. Requests for such exemption shall be submitted not less than 45 days prior to the date upon which the plant would otherwise be required to operate in accordance with the procedures of said paragraph (a)(2)(iv) of this section. Any such request shall be filed with the Secretary of the Commission, who shall cause notice of its receipt to be published promptly in the FEDERAL REGISTER, such notice shall provide for the submission of comments by interested persons within 14 days following FEDERAL REGISTER publication. The Director of Nuclear Reactor Regulation shall submit his views as to any requested exemption within five days following expiration of the comment period.

(vii) Any request for an exemption submitted under paragraph (a)(2)(vi) of this section must show, with appropriate affidavits and technical submissions, that it would be in the public interest to allow the licensee a specified additional period of time within which to alter the operation of the facility in the manner required by paragraph (a)(2)(iv) of this section. The request shall also include a discussion of the alternatives available for establishing compliance with the rule.

(3) Construction permits may have been issued after December 28, 1973 but before December 28, 1974 subject to any applicable conditions or restrictions imposed pursuant to other regulations in this chapter and the Interim Acceptance Criteria for Emergency Core Cooling Systems published on June 29, 1971 (36 FR 12248) as amended (December 18, 1971, 36 FR 24082): *Provided, however*, that no operating license shall be issued for facilities constructed in accordance with construction permits issued pursuant to this paragraph, unless the Commission determines, among other things that the proposed facility meets the requirements of paragraph (a)(1) of this section.

(b)(1) *Peak cladding temperature*. The calculated maximum fuel element cladding temperature shall not exceed 2200° F.

(2) *Maximum cladding oxidation*. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

(3) *Maximum hydrogen generation*. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) *Coolable geometry*. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) *Long-term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

(c) As used in this section: (1) Loss-of-coolant accidents (LOCA's) are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.

(2) An evaluation model is the calculational framework for evaluating the

behavior of the reactor system during a postulated loss-of-coolant accident (LOCA). It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

(d) The requirements of this section are in addition to any other requirements applicable to ECCS set forth in this part. The criteria set forth in paragraph (b), with cooling performance calculated in accordance with an acceptable evaluation model, are in implementation of the general requirements with respect to ECCS cooling performance design set forth in this part, including in particular Criterion 35 of Appendix A.

[39 FR 1002, Jan. 4, 1974, as amended at 39 FR 27121, July 25, 1974; 40 FR 8789, Mar. 3, 1975]

§ 50.47 Emergency plans.

(a)(1) Except as provided in paragraph (d) of this section, no operating license for a nuclear power reactor will be issued unless a finding is made by NRC that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

(2) The NRC will base its finding on a review of the Federal Emergency Management Agency (FEMA) findings and determinations as to whether State and local emergency plans are adequate and whether there is reasonable assurance that they can be implemented, and on the NRC assessment as to whether the applicant's onsite emergency plans are adequate and whether there is reasonable assurance that they can be implemented. A FEMA finding will primarily be based on a review of the plans. Any other information already available to FEMA may be considered in assessing whether there is reasonable assurance that the plans can be implemented. In any NRC licensing proceeding, a FEMA finding will constitute a rebuttable presumption on questions of adequacy and implementation capability. Emergency preparedness exercises (required by paragraph (b)(14) of this section and Appendix E, Section F of this part) are part of the operational inspection process and are not required for any initial licensing decision.

(b) The onsite and, except as provided in paragraph (d) of this section, offsite emergency response plans for nuclear power reactors must meet the following standards:[1]

(1) Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.

(2) On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available and the interfaces among various onsite response activities and offsite support and response activities are specified.

(3) Arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and local staff at the licensee's near-site Emergency Operations Facility have been made, and other organizations capable of augmenting the planned response have been identified.

(4) A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance

[1] These standards are addressed by specific criteria in NUREG-0654; FEMA-REP-1 entitled "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in support of Nuclear Power Plants—for Interim Use and Comment", January 1980.

Footnotes to § 50.55a:

[1] [Reserved]

[2] Components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary defined in § 50.2(v) need not meet these requirements, provided:

(a) In the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only, or

(b) The component is or can be isolated from the reactor coolant system by two valves (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and, assuming the other valve is open, its closure time must be such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

[3] Copies may be obtained from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. Copies are available for inspection at the Commission's Public Document Room, 1717 H St. NW., Washington, D.C.

[4] USAS and ASME Code addenda issued prior to the Winter 1977 Addenda are considered to be "in effect" or "effective" 6 months after their date of issuance and after they are incorporated by reference in paragraph (b) of this section. Addenda to the ASME Code issued after the Summer 1977 Addenda are considered to be "in effect" or "effective" after the date of publication of the addenda and after they are incorporated by reference in paragraph (b) of this section.

[5] For ASME Code Editions and Addenda issued prior to the Winter 1977 Addenda, the Code Edition and Addenda applicable to the component is governed by the order or contract date for the component, not the contract date for the nuclear energy system. For the Winter 1977 addenda and subsequent editions and addenda the method for determining the applicable Code editions and addenda is contained in Paragraph NCA 1140 of Section III of the ASME Code.

[6] ASME Code cases which have been determined suitable for use by the Commission staff are listed in NRC Regulatory Guide 1.84, "Code Case Acceptability—ASME Section III Design and Fabrication" and NRC Regulatory Guide 1.85, "Code Case Acceptability—ASME Section III Materials." The use of other Code cases may be authorized by the Commission upon request pursuant to § 50.55a(a)(2)(ii).

[7] For purposes of this regulation, the proposed IEEE 279 became "in effect" on August 30, 1968, and the revised issue IEEE 279-1971 became "in effect" on June 3, 1971. Copies may be obtained from the Institute of Electrical and Electronics Engineers, United Engineering Center, 345 East 47th Street, New York, NY 10017. A copy is available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

[8] Where an application for a construction permit is submitted in four parts pursuant to the provisions of § 2.101(a-1) and Subpart F of Part 2 of this chapter, "the formal docket date of the application for a construction permit" for purposes of this section shall be the date of docketing of the information required by § 2.101(a-1) (2) or (3), whichever is later.

§ 50.56 Conversion of construction permit to license; or amendment of license.

Upon completion of the construction or alteration of a facility, in compliance with the terms and conditions of the construction permit and subject to any necessary testing of the facility for health or safety purposes, the Commission will, in the absence of good cause shown to the contrary issue a license of the class for which the construction permit was issued or an appropriate amendment of the license, as the case may be.

(Sec. 185, 68 Stat. 955; 42 U.S.C. 2235)

[21 FR 355, Jan. 19, 1956, as amended at 35 FR 11461, July 17, 1970]

§ 50.57 Issuance of operating license. [1]

(a) Pursuant to § 50.56, an operating license may be issued by the Commission, up to the full term authorized by § 50.51, upon finding that:

(1) Construction of the facility has been substantially completed, in conformity with the construction permit and the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

[1] The Commission may issue a provisional operating license pursuant to the regulations in this part in effect on March 30, 1970, for any facility for which a notice of hearing on an application for a provisional operating license or a notice of proposed issuance of a provisional operating license has been published on or before that date.

(2) The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

(3) There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations in this chapter; and

(4) The applicant is technically and financially qualified to engage in the activities authorized by the operating license in accordance with the regulations in this chapter. However, no finding of financial qualifications is necessary for an electric utility applicant for an operating license for a production or utilization facility of the type described in § 50.21(b) or § 50.22.

(5) The applicable provisions of Part 140 of this chapter have been satisfied; and

(6) The issuance of the license will not be inimical to the common defense and security or to the health and safety of the public.

(b) Each operating license will include appropriate provisions with respect to any uncompleted items of construction and such limitations or conditions as are required to assure that operation during the period of the completion of such items will not endanger public health and safety.

(c) An applicant may, in a case where a hearing is held in connection with a pending proceeding under this section make a motion in writing, pursuant to this paragraph (c), for an operating license authorizing low-power testing (operation at not more than 1 percent of full power for the purpose of testing the facility), and further operations short of full power operation. Action on such a motion by the presiding officer shall be taken with due regard to the rights of the parties to the proceedings, including the right of any party to be heard to the extent that his contentions are relevant to the activity to be authorized. Prior to taking any action on such a motion which any party opposes, the presiding officer shall make findings on the matters specified in paragraph (a) of this section as to which there is a controversy, in the form of an initial decision with respect to the contested activity sought to be authorized. The Director of Nuclear Reactor Regulation will make findings on all other matters specified in paragraph (a) of this section. If no party opposes the motion, the presiding officer will issue an order pursuant to § 2.730(e) of this chapter, authorizing the Director of Nuclear Reactor Regulation to make appropriate findings on the matters specified in paragraph (a) of this section and to issue a license for the requested operation.

[35 FR 5318, Mar. 31, 1970, as amended at 35 FR 6644, Apr. 25, 1970; 37 FR 11873, June 15, 1972; 37 FR 15142, July 28, 1972; 40 FR 8790, Mar. 3, 1975; 47 FR 13755, Mar. 31, 1982]

§ 50.58 Hearings and report of the Advisory Committee on Reactor Safeguards.

(a) Each application for a construction permit or an operating license for a facility which is of a type described in § 50.21(b) or § 50.22, or for a testing facility, shall be referred to the Advisory Committee on Reactor Safeguards for a review and report. An application for an amendment to such a construction permit or operating license may be referred to the Advisory Committee on Reactor Safeguards for review and report. Any report shall be made part of the record of the application and available to the public, except to the extent that security classification prevents disclosure.

(b) The Commission will hold a hearing after at least 30 days notice and publication once in the FEDERAL REGISTER on each application for a construction permit for a production or utilization facility which is of a type described in § 50.21(b) or § 50.22 or which is a testing facility. When a construction permit has been issued for such a facility following the holding of a public hearing and an application is made for an operating license or for an amendment to a construction permit or operating license, the Commission may hold a hearing after at least 30 days notice and publication once in the FEDERAL REGISTER or, in the absence of a request therefor by any person whose interest may be affected, may issue an operating license or an amendment to a construction permit or operating license without a hearing, upon 30 days notice and publication once in the FEDERAL REGISTER of its intent to do so. If the Commission finds that no significant hazards consideration is presented by an application for an amendment to a construction permit or operating license, it may dispense with such notice and publication and may issue the amendment.

[27 FR 12186, Dec. 8, 1962, as amended at 33 FR 8570, June 12, 1968; 35 FR 11461, July 17, 1970; 39 FR 10555, Mar. 21, 1974]

§ 50.59 Changes, tests and experiments.

(a)(1) The holder of a license authorizing operation of a production or utilization facility may (i) make changes in the facility as described in the safety analysis report, (ii) make changes in the procedures as described in the safety analysis report, and (iii) conduct tests or experiments not described in the safety analysis report, without prior Commission approval, unless the proposed change, test or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question.

(2) A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

(b) The licensee shall maintain records of changes in the facility and of changes in procedures made pursuant to this section, to the extent that such changes constitute changes in the facility as described in the safety analysis report or constitute changes in procedures as described in the safety analysis report. The licensee shall also maintain records of tests and experiments carried out pursuant to paragraph (a) of this section. These records shall include a written safety evaluation which provides the bases for the determination that the change, test or experiment does not involve an unreviewed safety question. The licensee shall furnish to the appropriate NRC Regional Office shown in Appendix D of Part 20 of this chapter with a copy to the Director of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, annually or at such shorter intervals as may be specified in the license, a report containing a brief description of such changes, tests, and experiments, including a summary of the safety evaluation of each. Any report submitted by a licensee pursuant to this paragraph will be made a part of the public record of the licensing proceeding. In addition to a signed original, 39 copies of each report of changes in a facility of the type described in § 50.21(b) or § 50.22 or a testing facility, and 12 copies of each report of changes in any other facility, shall be filed. The records of changes in the facility shall be maintained until the date of termination of the license, and records of changes in procedures and records of tests and experiments shall be maintained for a period of five years.

(c) The holder of a license authorizing operation of a production or utilization facility who desires (1) a change in technical specifications or (2) to make a change in the facility or the procedures described in the safety analysis report or to conduct tests or experiments not described in the safety analysis report, which involve an unreviewed safety question or a change in technical specifications, shall submit an application for amendment of his license pursuant to § 50.90.

[39 FR 10555, Mar. 21, 1974, as amended at 41 FR 16446, Apr. 19, 1976; 41 FR 18302, May 3, 1976; 42 FR 20139, Apr. 18, 1977]

INSPECTIONS, RECORDS, REPORTS, NOTIFICATIONS

§ 50.70 Inspections.

(a) Each licensee and each holder of a construction permit shall permit in-with the regulations in this chapter and will not be inimical to the common defense and security or to the health and safety of the public.

(b) If the application demonstrates that the dismantling of the facility and disposal of the component parts will be performed in accordance with the regulations in this chapter and will not be inimical to the common defense and security or to the health and safety of the public, and after notice to interested persons, the Commission may issue an order authorizing such dismantling and disposal, and providing for the termination of the license upon completion of such procedures in accordance with any conditions specified in the order.

[26 FR 9546, Oct. 10, 1961, as amended at 32 FR 3090, Feb. 21, 1967]

AMENDMENT OF LICENSE OR CONSTRUCTION PERMIT AT REQUEST OF HOLDER

§ 50.90 Application for amendment of license or construction permit.

Whenever a holder of a license or construction permit desires to amend the license or permit, application for an amendment shall be filed with the Commission, fully describing the changes desired, and following as far as applicable the form prescribed for original applications.

§ 50.91 Issuance of amendment.

In determining whether an amendment to a license or construction permit will be issued to the applicant the Commission will be guided by the considerations which govern the issuance of initial licenses or construction permits to the extent applicable and appropriate. If the application involves the material alteration of a licensed facility, a construction permit will be issued prior to the issuance of the amendment to the license. If the amendment involves a significant hazards consideration, the Commission will give notice of its proposed action pursuant to § 2.105 of this chapter before acting thereon. The notice will be issued as soon as practicable after the application has been docketed.

[39 FR 13258, Apr. 12, 1974]

REVOCATION, SUSPENSION, MODIFICATION, AMENDMENT OF LICENSES AND CONSTRUCTION PERMITS, EMERGENCY OPERATIONS BY THE COMMISSION

§ 50.100 Revocation, suspension, modification of licenses and construction permits for cause.

A license or construction permit may be revoked, suspended, or modified, in whole or in part, for any material false statement in the application for license or in the supplemental or other statement of fact required of the applicant; or because of conditions revealed by the application for license or statement of fact or any report, record, inspection, or other means, which would warrant the Commission to refuse to grant a license on an original application (other than those relating to §§ 50.51, 50.42(a), and 50.43(b) of this part); or for failure to construct or operate a facility in accordance with the terms of the construction permit or license, provided that failure to make timely completion of the proposed construction or alteration of a facility under a construction permit shall be governed by the provisions of § 50.55(b); or for violation of, or failure to observe, any of the terms and provisions of the act, regulations, license, permit, or order of the Commission.

§ 50.101 Retaking possession of special nuclear material.

Upon revocation of a license, the Commission may immediately cause the retaking of possession of all special nuclear material held by the licensee.

[21 FR 355, Jan. 19, 1956, as amended at 40 FR 8790, Mar. 3, 1975]

§ 50.102 Commission order for operation after revocation.

Whenever the Commission finds that the public convenience and necessity, or the Department finds that the production program of the Department requires continued operation of a production or utilization facility, the license for which has been revoked, the Commission may, after consultation with the appropriate federal or state regulatory agency having jurisdiction, order that possession be taken of such facility and that it be operated for a period of time as, in the judgment of the Commission, the public convenience and necessity or the production program of the Department may require, or until a license for operation of the facility shall become effective. Just compensation shall be paid for the use of the facility.

[40 FR 8790, Mar. 3, 1975]

§ 50.103 Suspension and operation in war or national emergency.

(a) Whenever Congress declares that a state of war or national emergency exists, the Commission, if it finds it necessary to the common defense and security, may,

(1) Suspend any license it has issued.

(2) Cause the recapture of special nuclear material.

(3) Order the operation of any licensed facility.

(4) Order entry into any plant or facility in order to recapture special nuclear material or to operate the facility.

(b) Just compensation shall be paid for any damages caused by recapture of special nuclear material or by operation of any facility, pursuant to this section.

(Sec. 108, 68 Stat. 939, as amended; 42 U.S.C. 2138)

[21 FR 355, Jan. 19, 1956, as amended at 35 FR 11416, July 17, 1970; 40 FR 8790, Mar. 3, 1975]

BACKFITTING

§ 50.109 Backfitting.

(a) The Commission may, in accordance with the procedures specified in this chapter, require the backfitting of a facility if it finds that such action will provide substantial, additional protection which is required for the public health and safety or the common defense and security. As used in this section, "backfitting" of a production or utilization facility means the addition, elimination or modification of structures, systems or components of the facility after the construction permit has been issued.

(b) Nothing in this section shall be deemed to relieve a holder of a construction permit or a license from compliance with the rules, regulations, or orders of the Commission.

(c) The Commission may at any time require a holder of a construction permit or a license to submit such information concerning the addition or proposed addition, the elimination or proposed elimination, or the modification or proposed modification of structures, systems or components of a facility as it deems appropriate.

[35 FR 5318, Mar. 31, 1970]

ENFORCEMENT

§ 50.110 Violations.

An injunction or other court order may be obtained prohibiting any violation of any provision of the Atomic Energy Act of 1954, as amended, or Title II of the Energy Reorganization Act of 1974, or any regulation or order issued thereunder. A court order may be obtained for the payment of a civil penalty imposed pursuant to section 234 of the Act for violation of section 53, 57, 62, 63, 81, 82, 101, 103, 104, 107, or 109 of the Act, or section 206 of the Energy Reorganization Act of 1974, or any rule, regulation, or order issued thereunder, or any term, condition, or limitation of any license issued thereunder, or for any violation for which a license may be revoked under section 186 of the Act. Any person who willfully violates any provision of the Act or any regulation or order issued thereunder may be guilty of a crime and, upon conviction, may be punished by fine or imprisonment or both, as provided by law.

[40 FR 8790, Mar. 3, 1975, as amended at 42 FR 25721, May 19, 1977]

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Enclosure 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CHAIRMAN

May 17, 1985

The Honorable Edward J. Markey, Chairman
Subcommittee on Energy Conservation and Power
Committee on Energy and Commerce
United States House of Representatives
Washington, DC 20515

PUBLIC DOCUMENT ROOM
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TIME REQUESTED

Dear Mr. Chairman:

Recently, Mr. Licciardo, an NRC staff member, met with me under NRC's Open Door Policy regarding the Commission's letter to you dated December 20, 1984 on the subject of erroneous McGuire Technical Specifications. He felt that the December 20, 1984 letter mischaracterized his involvement in the review of the McGuire Technical Specifications and that his actions were inaccurately cited as the main cause for delay in resolving his differing professional opinion (DPO) on these same specifications. This letter is intended to correct any mischaracterizations or misrepresentations regarding Mr. Licciardo in our December 20 letter.

Our December 20 letter should not have inferred that Mr. Licciardo introduced unnecessary delays nor that the detailed attention provided during the staff's review resulted in unwarranted or avoidable delays. The problem is complex and, as such, is not subject to singling out one cause of delay. Due to the sheer magnitude of his concerns, over 300 in all, it took a significant amount of time for Mr. Licciardo to provide the required bases for each item. Likewise, a significant and lengthy staff effort was necessary to evaluate each item.

Based on my conversation with Mr. Licciardo and his subsequent discussions with my personal staff, I believe the pace of the staff's review is acceptable to Mr. Licciardo. The staff found in February 1984 that none of the McGuire concerns presented an imminent public health or safety problem. Given this finding and the increased attention afforded by the staff to this matter, I believe that the McGuire Technical Specification evaluation is proceeding at a satisfactory pace.

Mr. Licciardo also indicated that the December 20, 1984 letter to you mischaracterized the present state of the McGuire Technical Specifications. However, I have not been able to confirm Mr. Licciardo's claim. As I noted above, the staff made an initial finding that there was no imminent safety

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problem with the Technical Specifications. The 380 items identified by Mr. Licciardo were evaluated by a team of reactor systems technical managers. That team concluded that 160 of the items did not warrant further attention either because:

(1) Mr. Licciardo's assessment of the issue was incorrect, or

(2) the management team (all of whom were experienced reactor systems reviewers) could not understand Mr. Licciardo's description of the issue.

The management team concluded that the remaining 220 did warrant additional NRC evaluation. The present schedule calls for completion of the staff evaluation and categorization of those 220 items by late spring of this year. Upon completion of this categorization a letter will be forwarded to the licensee requesting his response to plant specific issues within three months. The remaining issues of the 220 items which are generic in nature will be handled as part of our generic issues program with a target date for final resolution by the end of this year. This letter and all subsequent letters, will be a matter for the public record, and, as such, will be docketed. If any information becomes available which causes us to reconsider the staff's initial finding, the schedule will be accelerated.

I appreciate Mr. Licciardo's sincerity and conscientiousness in bringing his concerns to my attention. I trust that this letter will further clear the air on his involvement in the schedule of resolving the concerns arising from his Differing Professional Opinion.

Sincerely,

Nunzio J. Palladino

Nunzio J. Palladino

cc: Rep. Carlos Moorhead

Enclosure 3

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RECIP NAME RECIPIENT AFFILIATION
DENTON, H R Office of Nuclear Reactor Regulation, Director (post 851125
YOUNGBLOOD, B J PWR Project Directorate 4

SUBJECT: Response to 850709 ltr re plant-specific concerns from review of mid-Jan 1983 Tech Specs. Tech Spec & FSAR revs will be pursued upon NRC concurrence w/positions.

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HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

June 10, 1986

[redacted] Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: B.J. Youngblood, Director
PWR Project Directorate #4

Subject: McGuire Nuclear Station
Docket Nos. 50-369 and 50-370
NRC DPO Concerns on McGuire Technical Specifications

Dear Mr. Denton:

Mr. T.M. Novak's (NRC/ONRR) July 9, 1985 letter to Mr. H.B. Tucker (DPC) indicated that a review of the McGuire Unit 1 and 2 Technical Specifications was being conducted in response to concerns raised by a member of the NRC staff in a differing professional opinion (DPO) resulting from a review of the proof and review copy of the McGuire Unit 1/2 combined Technical Specifications which existed in mid-January 1983. Duke Power Company's comments were requested on certain plant-specific concerns contained in the DPO (other concerns contained in the DPO were either being considered by the NRC for generic resolution, had been closed by NRC internal review, or were still under review).

Attached is Duke Power Company's response to these concerns. This response is limited to the specified plant-specific concerns and does not address any generic aspects of these specified concerns. Note that the response has potential plant-specific impacts on the station's Technical Specifications (e.g. question nos. 6a, 7d (and 7i, 7k), and 7n) and FSAR (e.g. questions 4a&b, and 4c). Duke will pursue appropriate plant-specific Technical Specification and FSAR revisions following NRC concurrence with the positions contained herein. The Westinghouse Standard Technical Specification issues identified in this response should be resolved on a generic basis (note that Westinghouse review/input was utilized in the development of this response). Note also that generic Technical Specification improvement efforts currently underway by industry (e.g. AIF, WOG, B&WOG) and NRC (TSIP) may impact the DPO's concerns and the resolutions proposed by this response.

As indicated above, the NRC is requested to approve this response prior to Duke proceeding with the appropriate Technical Specification change submittals and inclusion of the informaiton in a future FSAR update. Should there

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be any questions regarding this matter or if additional information is required, please advise.

Very truly yours,

Hal B. Tucker/pa

Hal B. Tucker

PBN/jgm

Attachment

xc: Dr. J. Nelson Grace, Regional Administrator
U.S. Nuclear Regulatory Commission - Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

Mr. Darl Hood
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
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Mr. W.T. Orders
Senior Resident Inspector
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Westinghouse Electric Corp.
MNC West Tower
P.O. Box 355
Pittsburgh, PA 15230

Duke Power Company

McGuire Nuclear Station

Response to NRC DPO Concerns

(Question 1)

TABLE 2.2-1

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.

Response: The data contained in Reference 18 has been confirmed to be valid for both McGuire Unit 1 and Unit 2. The instrumentation hardware (racks, transmitters) are the same for both Units 1 and 2. While the Steam Generators are different (D-2 for Unit 1 and D-3 for Unit 2), there are no differences in the Safety Analysis values. Therefore it can be concluded that the Setpoint Study performed for Unit 1 is applicable, in it's entirety, to Unit 2. The safety analysis performed is valid for both units and use the same equipment/instrumentation resulting in uncertainty values being valid for both units.

(Question 1a)

TABLE 2.2-1, Item 3

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

Response: The dynamic response of the High Positive Rate trip function is similar to the rate/lag function associated with the ΔT trips. The responses of the various dynamic functions are demonstrated in Appendix A of WCAP-8745 (Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions). As m y be seen in the above mentioned figures, an increased time constant results in faster response and thus results in a shorter time from initiation of transient to reactor trip. Therefore, the >2 seconds Tech Spec requirement for the time constant is conservative.

(Question 1b)

TABLE 2.2-1, Item 4

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.

Response: The dynamic response of the High Negative Rate trip function is similar to the rate/lag function associated with the ΔT trips. The responses of the various dynamic functions are demonstrated in Appendix A of WCAP-8745 (Design Bases for the Thermal Overpower ΔT and Thermal Over temperature ΔT Trip Functions). As may be seen in the above mentioned figures, an increased time constant results in faster response and thus results in a shorter time from initiation of transient to reactor trip. Therefore, the >2 seconds Tech Spec requirement for the time constant is conservative.

The Revision 7 FSAR analysis referred to in this inquiry was performed prior to the NRC review and approval of WCAP 10297-P-A (Dropped Rod Methodology For Negative Flux Rate Plants). The methodology used prior to WCAP-10297-P-A did not involve an actual determination of the negative flux rate setpoint and/or determination of the maximum dropped rod(s) worths which might not result in a reactor trip. The statement in the FSAR (RCCA group results in reactivity insertion of ~-1200 pcm which results in a reactor trip within ~ 2.5 seconds) was meant only to offer support for the DNB analysis performed at lower rod worths but did not actually demonstrate the adequacy of the negative flux rate setpoint.

Upon determination of possible nonconservatisms in the analytical methodology, Westinghouse developed the dropped rod methodology outlined in WCAP-10297-P-A. The revised methodology links the assumptions regarding the negative flux rate setpoint, rod worths and locations, control system behavior, and other factors which influence plant behavior following a dropped rod(s) event. The setpoint thus becomes an integral part of the safety analysis and the table in reference 18 is revised to show a safety analysis limit of 6.9% RTP. The adjustments made to account for various uncertainties results in an STS Trip Setpoint of 5.0% RTP and an STS Allowable Value of 5.5% RTP. Details regarding the revised methodology and basis for the setpoint may be found in WCAP-10297-P-A.

(Question 1c)

TABLE 2.2-1, Item 9

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 15.2.3-1 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.

Response: The analysis of the inadvertent operation of ECCS during power operation had assumed a low pressure setpoint of 1800 psia while other analyses assumed a setpoint of 1835 psig. The reference 18 value for the safety analysis limit was in error but was conservative and since margin exists between implemented and required setpoints, the conservatism did not impact the trip setpoint and allowable values.

The transient analyses have been reanalyzed as a result of the transition to optimized fuel assembly design. The revised analyses assumed a safety analysis limit of 1850 psia (1835 psig) for all transients.

(Question 1d)

TABLE 2.2-1, Item 13

Reference 18, page 3-13, Note 12 describes the Safety Analysis Limit for this item as a 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?

Response: The analysis in effect at the time this question was posed is no longer applicable. At present the bounding analysis for the steam generator lo-lo level is the feedbreak analysis. This analysis is done assuming the system starts at full power. In this analysis the safety analysis limit is 23% of narrow range span. As is indicated in the technical specifications this corresponds to a nominal trip setpoint of 40% narrow range span at 100% RATED THERMAL POWER.

(Question 1e)

TABLE 2.2-1, Item 18b

Accidental Depressurization of the main steam system is from zero load. It is unclear from reference 5 Table 7.2.1-4, (page 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 SI, or from SI initiated by other initiators. The Licensee shall clarify, and hence its validity with respect to the absence of the signal caused by P-7.

Response: Protection against accidental depressurization of the main steam system is provided by the overpower reactor trips (neutron flux and ΔT) and by the reactor trip which results from the receipt of the safety injection (SI) signal. The safety injection signal is actuated by low steamline pressure, low pressurizer pressure, or high containment pressure. The analysis performed results in SI initiation on low pressurizer pressure and reactor trip will either occur concurrently due to the trip on SI actuation or will occur prior to SI on the overpower trips. The main steam depressurization analyzed in the FSAR is initiated from hot shutdown conditions at time zero (i.e. reactor tripped) since this represents the most conservative initial condition. Thus no explicit assumption is made regarding the cause of reactor trip for the FSAR analysis. As noted in the FSAR and above, should the reactor be just critical or operating at power a reactor trip would occur on the overpower trips or from an SI actuation. In either case, no credit is taken for the reactor trip on pressurizer pressure when reactor power is below the P-7 interlock.

(Question 2)

T.S. Page 3/4 1-6

The existing minimum temperature for criticality (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°F, and any variation therefrom in either direction would be unacceptable.

Response: FSAR Figure 5.3.3-1 gives the normal relationship between reactor coolant system temperature and power. The hot zero power temperature employed at McGuire and used in the safety analysis is 557°F. The minimum temperature for criticality is determined such that the moderator temperature coefficient is within its analyzed temperature range, the trip instrumentation is within its operating range, the pressurizer is capable of being in an operable status with a steam bubble, and the reactor vessel is above its minimum RT_{NDT} temperature. The minimum temperature for criticality limit in the McGuire Technical Specifications is 551°F.

The difference between the HZP temperature and minimum temperature for criticality limit is required in order to allow for measurement of the moderator temperature coefficient. Since the moderator coefficient is confirmed to be within safety analysis assumptions at conditions of approximately 551°F - 557°F, the only input parameter to the safety analysis of concern is the initial temperature. The change in initial conditions from 557°F to 551°F for transients occurring at HZP would have a negligible impact on results and would be a less representative input since the majority of time spent at HZP conditions includes temperatures of ~557°F. As noted, the accidents analyzed at hot zero power (HZP) assume an RCS temperature of 557 °F. The FSAR notes that use of a higher initial system temperature yields a large fuel-water heat transfer coefficient, larger specific heats, and a less negative (smaller absolute magnitude) Doppler feedback effect for fast reactivity addition transients like the RCCA Bank Withdrawal from Subcritical and HZP Rod Ejection events. The reduced feedback results in a higher neutron flux peak. For a Steamline Break event, starting from a higher initial RCS temperature results in a greater increase in coolant density from the cooldown. More reactivity is added due to the positive moderator density coefficient and a higher return to power results when compared with the case of a lower initial RCS temperature. Based on these considerations, a higher initial RCS temperature is conservative for the analysis of events from power. The statement that any variation in HZP temperature is unacceptable is also not consistent with the general conservative philosophy used to evaluate nuclear plant safety since only limited analyses are performed to demonstrate adequate safeguards for a range of plant conditions.

(Question 3)

<u>TABLE 3.3-1, Item 6c</u>

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 Q 212-24, Item 212.58. The Licensee shall evaluate and propose. Currently, this appears non-conservative.

Response: A review of FSAR Section 15.4.6 (Boron Dilution Accident) does not indicate the number of Source Range Channels required operable; however, These channels are mentioned for Refueling (MODE 6) and start up (MODE 2) Dilution Accidents. For these cases, two channels <u>are</u> required per Tech. Specs. Additionally, MODES 3,4, and 5 are not addressed by this FSAR Section. Boron Dilution analyses during MODES 3,4, and 5 are not part of the McGuire plant licensing basis. As such, any channel operability requirements would not be based on the FSAR analysis.

Generic Letter 85-05 dated January 31, 1985 informed licensees of the Staff position resulting from the evaluation of Generic Issue 22 "Inadvertent Boron Dilution Events". The Staff concluded that the consequences of such events are not severe enough to jeopardize the health and safety of the public. Furthermore, while NRC stated that it would "not require operating plant backfits for boron dilution events at this time, the staff would reqard an unmitigated boron dilution event as a serious breakdown in the licensee's ability to control its plant, and strongly urges each licensee to assure itself that adequate protection against boron dilution events exists in its plants". McGuire personnel believe that adequate protection against boron dilution events exists and that no changes to technical specifications are warranted in this instance.

(Question 4a and 4b)

TABLE 3.3-2, Items 9 & 10

The T.S. 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.

Response: The Technical Specification limit of $\leq$ 2.0 seconds for the time delay of pressurizer pressure trip functions (low and high) is based upon the FSAR Chapter 15 transient analysis which assumed a delay of 2.0 seconds. The values for trip response times in chapter 7 are "typical maximum allowable time delays" and are not necessarily the same as the McGuire specific assumptions. For the sake of clarity, the values provided in chapter 7 will be revised to agree with Chapter 15 and Technical Specifications in a future FSAR update.

(Question 4c)

TABLE 3.3-2, Item 17

The proposed T.S. states that the response time requirement is NA (Not Applicable). This is incorrect since 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, e.g. reference 5, Table 7.2.1-4 (5 of 5) and related Note e on the page entitled "Notes for Table 7.2.1-4" confirms that Pressurizer 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. values.

Response: The NA enter for the required response time of reactor trip upon SI actuation is consistent with the Bases which states that trip functions not utiized in the FSAR transient analyses will have the requirement indicate not applicable in Table 3.3-2 (Reactor Trip System Instrumentation Response Times). However, as stated in Table

3.3-5 (Engineered Safety Features Response Times). The terminology in Note e, Table 7.2.1-4, should be Pressurizer Pressure-Low (Safety Injection). This wording will be corrected in a future update of the FSAR.

(Question 5a)

TABLE 3.3-3, Item 7g

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.

Response: The defeat of auxiliary feedwater pump auto-start is accomplished by depressing a switch that is interlocked with the P-11 permissive. Thus the auto-start can only be defeated below a pressurizer pressure of 1955 psig. However, the same defeat will prevent auto-start on low-low steam generator level (Table 3.3-3, Item 7c(1)). Since this auto-start capability is required in MODES 1, 2, and 3, the defeat switch is not used in these modes. Therefore the entry for APPLICABLE MODES on Item 7g is not important as it is controlled by the more limiting Item 7c(1).

The statement that P-11 can only occur in MODE 3 is not accurate. MODE 2 is defined as operation with T-avg. $\geq$350°F, K_{eff} $\geq$0.99, and power $\leq$5% RTP. Therefore, subcritical operation with T-avg. $\geq$350°F is in Mode 2 if K_{eff} is not less than 0.99. Critical operation is restricted to T-avg. $\geq$551°F, but even then the pressure-temperature operating limits permit pressures below 1955 psig. As a practical matter, pressure is maintained in the normal operating range (~2235 psig) during MODE 2. The 2# referred to in the question is retained to require that MODE 2 operation above P-11 is with the Item 7g auto-start enabled.

(Question 5b)

TABLE 3.3-3, Item 8

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

Response: This specification is consistent with other standard technical specificaions which require operator action to mitigate the consequences of a LOCA in these modes.

(Question 6a)

TABLE 3.3-4, Item 4d

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 100 psi and 110 psi. 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 RTP with a time constant of 2 seconds.

Response: Since no safety analysis limit exists for the negative steam line pressure rate setpoint (i.e., it is not assumed in transient analyses), the Setpoint Methodology (Reference 18) listed the T.S. values. The T.S. limits were revised at a later date and thus a discrepancy between the Reference 18 and T.S. values exists.

In order to correct a typographical error and adequately define the setpoint, a T.S. revision will be pursued in the following form:

	Trip Setpoint	Allowable Value
4d. Negative Steam Line Pressure Rate-High	$\leq$100 psi with a rate/lag function time constant $\geq$50 seconds	$\leq$120 psi with a rate/lag function time constant $\geq$50 seconds

(Question 6b)

TABLE 3.3-4, Items 7c(1) and (2)

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.]

- Reference 7 Related Section 15.4.2.2.2 concerning Main Feed Line Rupture (MFLR) under the Title of Major Assumption 10.

"The auxiliary feedwater system is actuated by the low-low Steam Generator Water Level Signal. The auxiliary feedwater system is assumed to supply a total of 450 gpm to three intact steam generators.

- Reference 5, Section 10.4.7.2.2 states that "Travel stops are set on the steam generator flow control valves such that the turbine driven pump can supply 450 gpm to three intact steam generators while feeding one faulted generator and both motor driven pumps together can supply 450 GPM to three intact steam generators while feeding one fault d generator. The Throttle positions allow all three pumps to supply a t al flow of 1400 gpm to 4 intact steam generators".

- Reference 7 Related Section 15.4.2.2.2, page 15.4-13a (revision 38), states: "The single active failure assumed in the analysis is the turbine driven auxiliary feedwater pump. The motor driven pump that is headered to the steam generator with the ruptured main feedline supplies 110 gpm to the intact steam generator. The motor driven pump that is headered to two intact steam generators supplies 170 gpm to each. This yields a total flow of 450 to the intact steam generators one minute after reactor trip. At 30 minutes following the rupture, the operator is assumed to isolate the auxiliary feedline to the ruptured steam generator which results in an increase in injected flow of 80 gpm".

The sequence of events in the accident evaluation in Reference 7, Table 15.4-1 shows that after the accident is initiated at a programmed value of SG level, the low-low SG level in the ruptured SG is reached at 20 secs. later, and auxiliary feedwater [at 450 gpm] is delivered to the intact steam generators in 61 sec.

It appears, based on the above information, that on SG low-low in the ruptured SG, both the motor driven and the turbine driven pumps are initiated (with the single failure being in the turbine driven pumps). This is not in accord with the T.S. If it is assumed that low-low level in the other SGs is also reached at the same time by bubble collapse, please justify. We note that the Reactor & Turbine Control System is designed so that under normal operation, collapse of SG level on Turbine Trip will not cause a reactor trip; also at this time, main steam from intact SGs is being lost to the faulted SG so that whereas inventory is lost, a full collapse need not occur.

The proposed T.S.s Item 7c(1) and 7c(2) ppear to be non-conservative in respect of accident analysis used in the Licensing Bases. The licensee shall clarify, evaluate and propose.

Response: It appears that the question is "Since one motor-driven pump supplies 110 gpm to an intact generator and the other otor driven pump supplies 170gpm to intact generators, where does the remaining 170 gpm (450 - 110 - 170), supplied to the intact generators, come from if not from the turbine-driven pump?". The new FSAR Chapter 15 analyses for optimized fuel make clear that the "two motor-driven pumps together deliver 450 gpm to the three intact steam generators allowing for spillage out of the break (Section 15.2.8.2, page 15.2.8, 1984 Update). To clarify exa ly the analysis assumption - One moter driven auxiliary feedwater pump supplies 110 gpm to an intact steam generator (the remainder spills out the break in the faulted loop) and the other motor driven pump supplies 170 gpm to each of the other two intact steam generators, this totals to 450 gpm.

If the failure of a motor driven pump is assumed, the turbine driven pump alone would supply at least 450 gpm to the intact loops. The turbine driven pump is actuated on low-low level in at least two steam generators. It is assumed that low-low level is reached in the other (non-faulted) steam generators as a result of the bubble collapse following turbine trip when the low-low level reactor trip is actuated from the faulted loop. This occurs because for this accident condition (i.e. not normal operation) the mass inventory in the intact steam generators is reduced significantly prior to reactor trip on low-low level in the faulted loop. The shrinkage caused by bubble collapse from this reduced mass condition would cause low-low level to be reached in the other steam generators.

(Question 6c)

TABLE 3.3-4, Item 9

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

Response: The bases for the setpoints and allowable values specified are to ensure Auxiliary Feedwater capability upon loss of power while minimizing the possible initiation of the sequence with the voltage greater than the limits of associated motors.

(Question 7a)

TABLE 3.3-5, Item 2a

A value of ≤ 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 described under Items 3a and 4a below, shows safety analyses limits of ≤ 12 secs (with offiste power) and ≤ 22 secs (without offsite 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.

Response: The entry for Table 3.3-5, Item 3a is identical to Item 2a for the loss of off site power case, i.e., each is 27 seconds. As explained in the Notes for Table 3.3-5, the difference between Item 4a and Items 2a and 3a is that 4a does not include a delay for the RHR pumps to attain their discharge pressure. This is appropriate since Item 4a deals with steam line break protection, as opposed to LOCA protection. The RHR pumps, although started for a steam line break, are not expected to deliver flow because of the higher RCS pressure. Therefore, the additional 5 second delay for these pumps to attain their discharge pressure is not relevant to ESF response time for this actuating signal.

(Question 7b)

TABLE 3.3-5, Item 2b

The descriptor (From SI), should be deleted as it is incorrect. The response time given is ≤ 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.

Response: The descriptor "(From SI)" is correct in that the allowable delay for a reactor trip due to the SI actuation signal is 2 seconds. This value is independent of the setpoint and associated delay of the initiator of SI. The reference 5, page 7.3-8 maximum time of 1.0 second is the limit on the delay associated with SI actuation upon exceeding the high containment pressure setpoint.

No credit is taken for reactor trip signal resulting from safety injection signal in any LOCA analysis. In the McGuire Unit 1 initial core large break LOCA analysis no credit is taken for reactor trip (rod insertion) at all. In the McGuire Unit 1 initial core small break LOCA a low pressurizer signal causes the reactor to trip. No credit for the control rods is taken until they are fully inserted.

(Question 7c)

TABLE 3.3-5, Item 2d

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.

Response: Following a design basis large LOCA, the isolation valve closure time depends upon the time when fuel failure occurs and fission products are released to the containment environment. The only isolation valves explicitly considered in the radiological consequences analysis of a LOCA are those in the containment purge and pressure relief lines which connect containment to the environment. For isolation valves in lines filled with process fluid a relatively long time is needed for the associated piping system to drain of fluid and expose the valve seat to the containment gases or for activity to migrate, due to the concentration gradient, through the process fluid and out the isolation valve. Hence, as long as isolation valve closure times for process lines are short (less than 1 min. per ANS 56.2) they need not be modeled in the dose calculations.

(Question 7d)

<u>TABLE 3.3-5, Item 2e</u>

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

Response: Section 15.B.2 of the McGuire FSAR considers the case of a LOCA concurrent with lower containment pressure relief. The results of the additional offsite dose due to this accident are presented in table 15.0.11-1. One of the parameters used to evaluate this case is the isolation time for the Containment Air Release and Addition (VQ) System valves which are used in venting lower containment. Table 15.B.2-1 indicates the isolation time for these valves is 4 seconds. Section 9.5.12.3 indicates that these valves auto close on a containment isolation, and that they have a 3 second closure time.

A technical specification revision to show a response time of $\leq$ 4 seconds for this item will be pursued. This would be consistent with the allowable 1 second for generating an ESF response as indicated on page 7.3-8 of the McGuire FSAR and the 3 second valve closing time as indicated above.

(Question 7e)

TABLE 3.3-5, Item 2f

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.

Response: No credit is taken for AFW flow being initiated from a Containment Pressure - High signal in analyses.

(Question 7f)

TABLE 3.3-5, Item 3a

Values of $\leq 27^{(1)}/12^{(2)}$ secs are proposed. Reference 5, page 7.3-8, shows a maximum initiating time of ESFAS 1.0 secs from 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.

Response: This question is related to the question on Item 2a. For a steam line break the RHR pumps are not expected to deliver inventory and the additional 5 second delay for them to attain their discharge pressure is not included in the safety analysis.

(Question 7g)

TABLE 3.3-5, Item 3b

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.

Response: The descriptor "(from SI)" is correct in that the allowable delay for a reactor trip due to the SI actuation signal is 2.0 seconds. This value is independent of the setpoint and associated delay of the initiator of SI. The Reference 5, page 7.3-8, maximum time of 1.0 second is the limit on the delay associated with SI actuation upon exceeding the Pressurizer Pressure - Low setpoint.

The chapter 15 safety analyses do not take credit for a reactor trip from an SI signal initiated by low-low pressurizer. (Ref. Question 7b Response).

(Question 7h)

TABLE 3.3-5, Item 3d

The proposed T.S. is $\leq 18^{(3)}/28^{(4)}$ secs. Reference our comments and requirements under Item 2d above.

Response: Reference our response under item 2d above.

(Question 7i)

TABLE 3.3-5, Item 3e

The proposed T.S. is NA. Reference our comments and requirements under 2e. above.

Response: Reference our response under Item 2e above.

(Question 7j)

TABLE 3.3-5, Item 3f

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

Response: The only non-LOCA transient which assumes ESF actuation on Pressurizer Pressure Low-Low is the Main Steamline Depressurization event (Inadvertent Opening of a Steam Generator Safety, Relief, or Dump Valve). For this event it is conservatively assumed that auxiliary feedwater is actuated at the maximum flow rate at the initiation of the event to accentuate the cooldown. Any delay in auxiliary feedwater actuation would be beneficial and therefore a response time requirement is not applicable or appropriate.

(Question 7k)

TABLE 3.3-5 Item 4e

The proposed T.S. is NA. Reference our comments and requirements under Item 2d above.

Response: Reference our response under Item 2e above.

(Question 7l)

TABLE 3.3-5, Item 4h

The proposed T.S. 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 Steam Isolation Valves Closed." [In the case of Feedwater System Pipe Rupture].

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

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

Response: Item 4h in Technical Specification Table 3.3-5 has been changed to a limit of ≤ 7 seconds (Ref. Amendment nos. 29 (Unit 1) and 10 (Unit 2)).

(Question 7m)

TABLE 3.3-5, Item 5a

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

Response: The response time for containment spray following a high containment pressure signal is specified at 45 seconds in the McGuire Technical Specifications. This value is consistent with the FSAR containment analysis actuation assumption as shown in FSAR Table 6.2.1-13c. Event times from the McGuire limiting case break mass/energy release analysis are reported in Table 6.2.1-29; the time of spray actuation has no effect on the mass/energy releases calculated.

(Question 7n)

TABLE 3.3-5, Item 6b

The proposed T.S. is $\leq$ 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 $\leq$ 5 secs for Main Feedwater Containment Isolation and $\leq$ 10 secs for Main Feedwater to Auxiliary Feedwater Isolation.

A total time to isolation of MFW of $\leq$ 13 secs seems appropriate to available equipment.

However 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 approximately 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 SG to cause water ingress into the main steam lines. Additionally, it can have consequences of potentially larger importance for the event occurring from subcritical zero power.

Reference also our concerns under item Table 3.3-4, Items 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, 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.

Response: Excessive Feedwater Flow at Full Power is analyzed in Section 15.1.2 of the McGuire FSAR. Table 15.1.2-1, page 1 of 2, 1984 Update, gives the sequence of events for this analysis. The High-High SG Level setpoint is reached at 27 seconds with feedwater isolation occurring 9 seconds later. This 9 second value agrees with the values used for feedwater isolation on Safety Injection.

To be consistent with the current safety analysis the Technical Specifications value for item 6b of Table 3.3-5 should be ≤ 9 seconds. Another alternative is to reanalyze the Excessive Feedwater Flow event with the longer delay time. Duke will pursue a Technical Specification revision or reanalysis.

(Question 7c)

TABLE 3.3-5, Item 12

Response time proposed as ≤ 60 secs.

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

Response: The automatic switchover to recirculation is initiated when the level setpoint in the RWST is reached. The setpoint determination includes allowances for switchover delay ≥ 60 seconds and plant procedures test to ensure switchover delay ≤ 60 seconds per Table 3.3-5, Item 12.

General Response to Questions 8a-8e:

These questions in general deal with the conservatism of the FSAR Chapter 15 safety analyses for events initiated from MODES 3-5. Specifically the question of the number of RCS loops in operation, for heat removal or other purposes, appears many times. Since the McGuire Technical Specifications and Westinghouse Standard Technical Specifications are identifical for MODES 3-5 for T.S. 3.4.1, Reactor Coolant Loops and Coolant Circulation, any questions regarding these matters should be resolved on a generic basis and are not specific to McGuire. Therefore, the responses to each question will deal only with items which are specific to McGuire.

(Question 8a)

SECTION 3/4.4.1, 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

Setpoint (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, page 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).

Response: No McGuire specific concerns are raised in this question. Refer to the general response to Questions 8a-8e.

(Question 8b)

<u>SECTION 3/4.4.1, G.2.6.2 STEAM LINE BREAKS</u>

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 212-47d & 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 (hot 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, a MSLB cools the RCS down to 212°F e.g., the residual shutdown margin will be 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 there is no Safety Injection, no Reactor Trip (on SI), and under the existing 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 is Not Applicable.

Given the circumstances of the proposed T.S., the 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 depressurization of the RCS; are:

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

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

Response: Changes in the Technical Specifications and plant procedures have occured since the DPO questions were posed (boration to cold shutdown prior to starting cooldown is no longer required). The required shutdown margin for RCS temperatures above 200°F is 1.3% Δk/k. The shutdown margin requirement for temperatures equal to or less than 200°F is 1.0% Δk/k. Variations in initial conditions for the steamline break transient were analyzed in WCAP-9226 and support the conservative assumptions in the FSAR analysis.

Closure times for the Main Steam Isolation Valves (MSIVs) are implied in the Technical Specifications. In Table 3.3-5, Items 4h, 5c, and 8, response times are given for the Steam Line Isolation function. This time includes the MSIV closure time. Other concerns raised in this question are generic. Refer to the general response to Questions 8a-8e.

(Question 8c)

SECTION 3/4.4.1, G.2.6.3 LOSS OF PRIMARY COOLANT

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 Q212-64, 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 Standby 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 immediately 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 analyses for W 4 loop plants the reactor coolant pumps are assumed to be tripped at the initiation 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 coastdown is important in 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 shutdown 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 nonconservative and the licensee must evaluate and propose.

Given the circumstances of the proposed T.S., operability of 1 [illegible] RCS loops in MODE 3 would be in non-conformance with the curren[illegible] [illegible]es Limits in a non-conservative manner and the licensee is requi[illegible] to eva[illegible]ate 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, the Power Range Neutron Flux, High Setpoint 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 No 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 nonconservative 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 Standby; in Item 212.90 (6.3), page 212-61, for a LOCA at and less than 1000 psig/425° in Hot Standby, 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 fewer than 4 RCS loops in MODE 3 would be in nonconformance with the Licensing Bases FSAR in a nonconservative manner, and the licensee is required to evaluate and propose.

Response: No McGuire specific concerns are raised in this question. Refer to the general response to Questions 8a-8e.

(Question 8d)

<u>SECTION 3/4.4.1, G.2.6.4 OCCURRENCES CAUSING AN INITIAL INCREASE IN RCS TEMPERATURE</u>

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 evaluation at zero power.

b) Startup 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 nonconformance with the current Safety Analyses Limits in a nonconservative manner. The licensee is required to evaluate and propose.

Response: No McGuire specific concerns are raised in this question. Refer to the general response to Questions 8a-8e.

(Question 8e)

T.S. 3.4.1 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 RCPs (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 steamline break, and LOCA events, small and large, down to 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.

Response: No McGuire specific concerns are raised in this question. Refer to the general response to Questions 8a-8e.

(Question 9)

T.S. Page 3/4 4-2

Earlier concerns under General 2.6.1 addressed the need to evaluate the consequences of the startup 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 b. states:

> "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 natural circulation. This proposal is nonconservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

Response: The actions included in ACTION b. are 1) suspend deboration operations and 2) immediately initiate action to restore forced circulation. The actions are obviously valid responses to the condition. There is no Emergency Operating Procedure at McGuire for natural circulation. There is Abnormal Procedure AP/1&2/A/5500/09, Plant Operations During Natural Circulation, which addresses the initiation, verification, and maintenance of natural circulation. This procedure would be implemented under this condition.

(Question 10)

T.S. Page 3/4 4-3

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 T.S. 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 nonconservative 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 the 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 sturation 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 necessry 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 212-47d 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 veri SG water level at the Safety Analysis Limit for the Licensing Basis, which is the no-load programmed level, not the current proposed T.S. valve which is the S.G. Low-Low Level [Reactor Trip] and AFW actuation. This proposed T.S. is nonconservative 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, 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 that 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 nonconservative 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 nonconservative 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 nonconservative 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 SG 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].

Response: Several separate questions are raised here. The McGuire specific ones are answered as follows:

1) Each RHR train is powered from a separate 4160V bus in the Essential Auxiliary Power System. Each reactor coolant pump is powered from a separate 6900V bus in the Normal Auxiliary Power System.

2) It should be noted that the requirement of maintaining a specific level in the steam generator to verify operability was imposed by the NRC and has no firm basis within Westinghouse. However, for an RCS loop to be operable, sufficient inventory is required in the secondary side for heat removal. In MODE 4 this can be assured by keeping the tube bundle covered. A reasonable way of ensuring this is to require that the secondary side level indicates within the narrow range span. Accounting for errors, an indicated level at the low-low level setpoint assures that the level is at least at the bottom of the narrow range span.

The safety analysis limit for reactor trip on lo-lo SG level is a function with a value of 0% at no-load conditions. Adding allowance for reference leg heatup and instrument error gives the value of 12% used as the T.S. trip setpoint. The T.S. value is therefore conservative with respect to the safety analysis limit.

3) The low flow alarms on the RHR loops are to alert the operator to insufficient flow under RHR conditions. They have no relation to the low flow reactor trip which inserts the control rods to control reactivity during low flow conditions at power. Boron is employed for reactivity control in the shutdown modes while rod insertion is impossible (if the rods are already inserted) or unnecessary (because of the boration).

The current surveillance 4.4.1.3.3 requires verifying one RCS or RHR loop in operation at least every 12 hours. The concern raised apparently centers around the assertion that core cooling could be lost without the knowledge of the operator since no protective functions or alarms are required to be operable by the technical specifications. However, it is expected that there would be multiple indications of any problems that could cause a loss of coolant loop. Although the appropriate alarms are not required by the technical specifications to be operable, there is no reason to believe that all relevant alarms and other indicators would be inoperative during this mode.

The other issues raised in this question are not specific to McGuire. Refer to the general response to Questions 8a-8e.

(Question 11a)

T.S. SECTION 3/4.5

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.

Response: This "question" is merely a statement of operator action to align RHR. It remains true and requires no response.

(Question 11b)

T.S. 3.5

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.

Response: This "question" is merely a statement of operator action to align the ECCS for use from a shutdown condition. It remains true and requires no response.

(Question 11c)

T.S. 3.5

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 hydrostastic 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 min" e 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.

Response: This "question" is largely a quotation from the FSAR. The last two paragraphs, while not from the FSAR, are simply statements introducing a quotation from the SER. Therefore, this requires no response.

(Question 12a)

T.S. 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 setpoint of approximately 450 psig? The specified setpoint 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.

Response: The bases for the T.S. 3.5.1 limit of Cold Leg Accumulator cover pressure of between 400-454 psig is the assumed value in the LOCA analysis (FSAR Chapter 15). Allowance for channel error and drift are accounted for in the determination of the T.S. requirements. The numbers in Table 6.3.2-1 are nominal and minimum values as required by T.S. 3.5.1 and are in agreement with the T.S. 3.5.1 limits. Recent Technical Specification changes (Ref. unit 1/2 License Amendments 57/38) associated with the removal/isolation of the UHI System involve revising the Cold Leg Accumulator cover pressure to between 585 and 639 psig.

(Question 12b)

T.S. 4.5.1.1.1.d.1

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.

Response: The Cold Leg Accumulators Relief Valves (NI-52, 63, 74, and 86) are not required to perform a safety function either to shutdown the reactor or to mitigate the consequences of an accident. The inservice testing program requirement to test all class 1, 2, & 3 valves was changed to valves which are required for safe shutdown of the reactor or mitigating the consequences of an accident.

Consequently these relief valves are not included in the McGuire Nuclear Station pump and valve inservice testing program required by 10 CFR 50.55a(g). These valves (and setpoints) are tested following maintenance only.

(Question 13)

T.S. 3.5.1.2.d

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 T.S. 4.5.1.2.a. Its absence from the proposed T.S. renders it potentially non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

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

Response: FSAR Table 6.3.2.1 provides the expected operating temperature range for the UHI accumulator water and not Safety Analysis limits as stated above. The Safety Analysis value related to UHI water temperature is assumed to be the upper bound value of 100°F.

The Upper Head Injection Accumulator Relief Valve (NI-279) is not required to perform a safety function either to shutdown the reactor or to mitigate the consequences of an accident. The Inservice Testing Program requirement to test all class 1, 2, & 3 valves was changed to valves which are required for safe shutdown of the reactor or mitigating the consequences of an accident. Consequently this relief valve is not included in the McGuire Nuclear Station pump and valve inservice testing program required by 10CFR 50.55a(g). This valve (and setpoint) is tested following maintenance only.

(Question 14)

T.S. 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 during accident occurrences.

Response: The bases for the limits as specified in T.S. 4.5.2.h are the assumed ECCS flows used in the LOCA analysis. ECCS flow injected to the broken cold leg is assumed to spill in LOCA analyses, so limits are placed on the branch line totals to ensure that adequate flow reaches the intact loops.

(Question 15)

T.S. SECTION 3/4.5.3

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 LCOs 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 overpressure 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.

Response: This question appears to be related to the discussion of FSAR Section 5.2.2, "Overpressurization Protection". Although it is stated in two places that Technical Specification 3.5.3.a violates the FSAR Licensing Basis, Section 5.2.2 contains no discussion of ECCS pump operability between 300°F and 350°F. It is further stated, in the discussion of Section 5.2.2., that the McGuire Technical Specification 3.5.3.a. differs markedly from the Westinghouse Standard Technical Specification 3.5.3.a. Comparing the two we find no differences in the number or type of ECCS pumps required to be operable or inoperable. The McGuire lower limit is 300°F compared with Standard lower limit of 275°F. We therefore conclude that the McGuire Specification does not differ from the Standard one in a non-conservative manner.

(Question 16)

T.S. 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 hours 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 operability down to and including MODE 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 according to design flow requirements throughout the occurrence.

The licensee should explain how the proposed T.S. ensures that the turbine driven pump maintains its flow performance required by accident analyses 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 Condition III and especially Condition IV occurrences of the Licensing Basis, and if the existing accident analyses remain valid.

Response: The footnote deleting operability requirements for the Steam Turbine-Driven Auxiliary Feedwater Pump (TDAFP) at steam pressures <900 psig was added in an attempt to correct a conflict between the LCO with its applicability of Modes 1, 2, and 3 and Surveillance Requirement 4.7.1.2.a.2 which defines operability of the TDAFP as developing a discharge pressure of $\geq$ 1210 psig at a flow of $\geq$900 gpm when the secondary steam supply pressure is >900 psig (to delevlop a discharge pressure of 1210 psig the TDAFP requires steam at $\geq$ 900 psig, but supply steam pressure can be <900 psig during startups/shutdowns). The Technical Specification's bases for operability of the Auxiliary Feedwater System is to ensure that the Reactor Coolant System can be cooled down to <350°F from normal operating conditions in the Event of a total loss of offsite power, with the TDAFP capable of delivering a total feedwater flow of 900 GPM at a pressure of 1210 psig to the entrance of the Steam Generators to meet this function. Under normal operating conditions source steam at >900 psig is Available and the TDAFP is capable of performing this function. However, as indicated in Question 16 and Items 1 and 2 below, the TDAFP is also required with steam pressures <900 psig.

1. During a condition IV feedline break all steam generators will depressurize prior to closure of the Main Steamline Isolation Valves (MSIV's). The low steamline pressure set point for closing the MSIV's is about 585 psig. However, errors due to seismic and environmental conditions as well as instrumentation inaccuracies may result in a steam generator pressure as low as 285 psig prior to MSIV closure. Therefore the turbine driven Auxiliary Feedwater pumps must be capable of delivering the minimum required flow for feedline break with a steam generator motive supply pressure as low as 285 psig.

2. The ability to commence a plant cooldown must be maintained following transient and accident conditions. Following design basis faulted conditions with specific single failure assumptions, it may be necessary to commence a plant cooldown with only a turbine driven Auxiliary Feedwater System pump available. Consequently the turbine driven pump must be capable of delivering the minimum required flow for cooldown with a steam generator motive supply pressure as low as 100 psia corresponding to a primary side hot leg temperature of 350°F during a natural circulation cooldown, which is maximum operating temperature for Residual Heat Removal System Operati n.

Therefore, The Tech. Spec's Surveillance requirements/Bases do not adequately define the operability requirements for the TDAFP and consequently the Technical Specification does not ensure operability of the TDAFP over the range of conditions expected from Design Basis Accident Analysis and other less bounding events. All other circumstances (or accident conditions) besides the limiting condition of loss of Offsite Power during full power operation pose less severe demands on the TDAFP. For the Main Steamline Break, the intact Steam Generator is fully capable of supplying the steam requirements of the pump turbine. With source steam < 900 psig the TDAFP is capable of providing feed flow but at a discharge pressure below 1210 psig. Since the McGuire Technical Specification is essentially indentical to the Westinghouse Standard Technical Specification (with the exception of the "correcting" footnote), this discrepancy between the LCO and the Surveillance Requirements/Bases should be resolved on a generic basis and is not specific to McGuire.

With regard to providing operability down to and including Mode 4 (and availability fro~ Mode 5), the bases of the auxiliary Feedwater System Technical Specification is that its operability (including the capacity of the TDAFP) ensures that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System Temperature to <350°F (i.e. Mode 4) when the RHR System may be placed into operation. Therefore the bases does not require System Operability in Modes 4 or 5. Since the McGuire and Westinghouse standard technical specifications bases are essentially identical, any desired changes to this bases should be pursued on a generic basis.

Item 1e of T.S. Table 3.3-3 "Steam Line Pressure-Low" is derived from steam line sensors downstream of the steam generator flow restriction orifices.. The steam flow restrictors do not cause a significant pressure drop except during a double ended steam line break. The blowdown phase of this accident lasts only a few seconds. The accurate pressure sensing in the steam lines (i.e. generation of a "Steam Line Pressure-Low" signal) takes less than 2 seconds and steam line isolation less than 7 seconds. (The main steam line break accident is descussed in Chapters 6 and 15 of the FSAR).

(Question 17)

T.S. SECTION 3/4.7.5

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.

Response: This action has been deleted. See Section 9.2.2, Nuclear Service Water System and Ultimate Heat Sink, 1984 Update.

(Question 18)

T.S. 3/4.9.1

The current SER, Supplement No.1, reference 11, page 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.1NV-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 1NV-250 is to be locked closed during refueling. The current position could be nonconservative 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 1NV-171A and 1NV-175A 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.

Response: Valve 1NV-250 is specifically required to be locked closed unuer the Administrative Controls (i.e. Station Procedures). This Valve is upstream of valves 1NV-171A and 1NV-175A and isolates the flow path.

(Question 19)

T.S. SECTION 3/4.9.8

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 in the FSAR. The licensee shall evaluate and propose.

Review of available responses to the consequences of a fail closed RCS/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 an` propose.

Response: The McGuire Technical Specification 3.9.8 is the same as the Westinghouse Standard Technical Specification (STS) 3.9.8. Since there is nothing unique about McGuire's 3411 MWt power level, its decay heat characteristics, or its 23 feet level requirement, this question should be addressed on a generic basis.

(Question 20)

T.S. SECTION 4.9.8.2

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.

Response: See the response to the previous item since McGuire is also in accordance with Westinghouse Standard Technical Specification on this item.