Attachment

COMMENTS ON RESOLUTION OF PLANT SPECIFIC DPO ISSUES CONCERNING

MC GUIRE TECHNICAL SPECIFICATIONS

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RESOLUTION OF PLANT-SPECIFIC

DPO ISSUES CONCERNING

MCGUIRE TECHNICAL SPECIFICATIONS

BY

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APRIL 1990

PLUS SEPARATE COMMENTS

BY

ROBERT B. A. LICCIARDO

JUNE 1990

PRIMARILY: COMMENTS ARE INSERTED INTO A COPY OF THE NRC'S REPORT IMMEDIATELY FOLLOWING THE RELATED RESOLUTION

SECONDARLY: WRITTEN COMMENTS ARE MADE ON EACH TABLE 1 THROUGH 5

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATION

TABLE-1	PLANT-SPECIFIC DPO ISSUES RESOLVED BY TECHNICAL SPECIFICATION AMENDMENT
TABLE-2	PLANT-SPECIFIC DPO ISSUES RESOLVED BY UPDATING FSAR & SET POINT METHODOLOGY (SPM)
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TABLE-4	DPO ISSUES CONSIDERED AS GENERIC ISSUES TO BE RESOLVED BY THE CTSE UNDER TS IMPROVEMENT PROGRAM (LICENSEE IDENTIFIED THESE ISSUES IN THEIR SUBMITTAL DATED JUNE 1986).
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TABLE-I

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

PLANT-SPECIFIC DPO ISSUES RESOLVED BY TECHNICAL SPECIFICTION AMENDMENT

QUESTION*	TS	SUBJECT	TS AMENDMENT NO. UNIT 1	TS AMENDMENT NO. UNIT 2
6a	Table 3.3-4, Item 4d	Steam Line Isolation Trip Setpoint	102	84
7d	Table 3.3-5, Item 2e	Containment Purge and Exhaust Isolation Response Time	102	84
7i	Table 3.3-5, Item 3e	" " "	102	84
7k	Table 3.3-5, Item 4e	" " "	102	84
7l	Table 3.3-5, Item 4h	Steam Line Isolation Response Time	29	10
7n	Table 3.3-5, Item 6b	Feedwater Isolation Response Time	102	84
15	TS 3/4.5.3	ECCS - Subsystems (Low Temperature Overpressure Protection	The licensee is in process to revise the TS. Incorporate Additional Requirements.	

Reference Additional Items Identified Under Table 3.

*Questions numbers are from reference 4.

TABLE-2

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

PLANT-SPE^IFIC DPO ISSUES RESOLVED BY UPDATING FSAR AND SETPOINT METHODOLOGY (SPM.

QUESTION*	TS	SUBJECT	UPDATE REFERENCE
4a/4b	Table 3.3-2, Items 9/10	Reactor Trip-Response Time	FSAR Page 7.2-15
4c	Table 3.3-2, Item 17	Reactor Trip-Response Time	Licensee response dated June 10, 1986 made a commitment to update the FSAP Table 7.2.1-4, Note e.

Reference Additional Items Identified Under Table-3

*Questions numbers are from reference 4.

TABLE-3

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

PLANT-SPECIFIC DPC ISSUES REQUIRING NO LICENSEE ACTION

QUESTION*	TS	SUBJECT	STATUS	
1	Table 2.2-1	Steam Generator-Setpoint	Complete - Staff agrees with the licensee response and that no licensee action required. Enclosure 3 provides the details of resolution.	
1a	Table 2.2-1, Item 3	Reactor Trip-Setpoint	" "	
1b	Table 2.2-1, Item 4	Reactor Trip-Setpoint	" "	Change FSAR & SPM
1c	Table 2.2-1, Item 9	Reactor Trip-Setpoint	" "	
1d	Table 2.2-1, Item 13	Reactor Trip-Setpoint	" "	change FSAR & SPM
1e	Table 2.2-1, Item 18b	Reactor Trip-Setpoint	" "	
2	TS Page 3/4.1-6, (TS 3.1.1.4)	Minimum Temperature for Criticality	" "	Unacceptable.
3	Table 3.3-1, Item 6c	Reactor Trip Instrumentation	" "	Revise T.S
5a	Table 3.3-3, Item 7g	Auxiliary Feedwater Mode Applicability	" "	Revise T.S

*Questions numbers are from reference 4.

TABLE-3 (continued)

QUESTION	TS	SUBJECT	STATUS	
6b	Table 3.3-4, Items 7c (1) and (2)	Auxiliary Feedwater-Trip Setpoints & Flow Distribution	Complete - Staff agrees with the licensee response and that no licensee action required. Enclosure 3 provides the details of resolution.	Flow Distribution should Transfer with Quest. 8 of Table 4
6c	Table 3.3-4, Item 9	Loss of Power To ESF Busses -Trip Setpoint	" "	Transfer to Generic Issues
7a	Table 3.3-5, Item 2a	Safety Injection (ECCS) - Response Time	" "	Amend TSs
7b	Table 3.3-5, Item 2b	Reactor Trip (from SI) - Response Time	" "	Amend TSs
7c	Table 3.3-5, Item 2d	Containment Isolation - Phase "A" (2) - Response Time	" "	Further review by Lic
7e	Table 3.3-5, Item 2f	Auxiliary Feedwater - Response Time	" "	Amend TS
7f	Table 3.3-5, Item 3a	Safety Injection (ECCS) - Response Time	" "	Amend TS
7g	Table 3.3-5, Item 3b	Reactor Trip-Response Time	" "	Amend TS

TABLE 3 (continued)

QUESTION	TS	SUBJECT	STATUS	
7h	Table 3.3-5, Item 3d	Containment Isolation Phase "A" – Response Time	Complete - Staff agrees with the licensee response and that no licensee action required. Enclosure 3 provides the details of resolution.	Further Review by Lee
		Phase "A" (2) - Response Time	" "	
7j	Table 3.3-5, Item 3f	Auxiliary Feedwater (5) - Response Time	" "	Amend T.S
7m	Table 3.3-5, Item 5a	Containment Spray - Response Time	" "	
7o	Table 3.3-5, Item 12	Automatic Switchover to Recirculation-Response Time	" "	Further clarification by Lee.
9	TS Page 3/4 4-2 (TS 3.4.1)	Natural Circulation Cooldown	" "	Unacceptable to Lee. Amend TS
11a	TS 3/4.5	ECCS	" "	
11b	TS 3.5	ECCS LOCA in Modes 3,4 & 5	" "	Reference calculations by [illegible]
11c	TS 3.5	ECCS LOCA in Modes 3,4 & 5	" "	

Table-3 (continued)

QUESTION	TS	SUBJECT	STATUS	
12a	Table 3.5.1.1.d	Cold Leg Injection Accumulator Nitrogen Cover Pressure	Complete - Staff agrees with the licensee response and that no licensee action required. Enclosure 3 provides the details of resolution.	Amend TS
12b	TS 4.5.1.1.1.1.d.1	Accumulator Relief Valve Setpoints Testing	" "	Amend TS/ IST program
13	TS 3.5.1.2.d	Upper Head Injection Accumulator	" "	Amend TS
14	TS 4.5.2.h	ECCS - Subsystems	" "	Add to FSAR
17	TS 3/4.7.5	Standby Nuclear Service Water Pond		Amend TS
18	TS 3/4.9.1	Boron Concentration	" "	Amend TS

TABLE-4

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

DPO ISSUES CONSIDERED AS GENERIC ISSUES TO BE RESOLVED BY THE OTSB

UNDER TS IMPROVEMENT PROGRAM

QUESTION*	TS	SUBJECT	STATUS
5b	Table 3.3-3, Item 8	Automatic Switchover to Recirculation and Loss of RHR Cooling (Modes 4 and 5)	Open
8a	TS 3/4.4.1 G.2.6.1	Rapid Reactivity Increase in Lower Modes 3, 4 & 5	"
8b	TS 3/4.4.1 G.2.6.2	Steam Line Breaks-Modes 3, 4 & 5	"
8c	TS 3/4.4.1 G.2.6.3	Loss of Primary Coolant - Modes 3, 4 & 5	"
8d	TS 3/4.4.1 G.2.6.4	Increase in RCS Temperature	"
8e	TS 3.4.1	RCS Loops - Modes 3, 4 & 5	"
10	TS Page 3/4 4-3	RCS - ~~Hot Shutdown~~ Modes 4 & 5	"
16	TS 3.7.1.2.6	Auxiliary Feedwater Operability	" Add Related TS item 4.7.1.2 of Ref 30
19	TS 3/4.9.8	Refueling Operations	"
20	TS 4.9.8.2	Refueling Operations	"

*Questions numbers are from reference 4.

TABLE 5

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

DPO ISSUES CONSIDERED AS GENERIC ISSUES TO BE RESOLVED BY THE OTSB

UNDER TS IMPROVEMENT PROGRAM

CONCERN*	TS	SUBJECT	STATUS		MODES APPLICABILITY
9A	3/4.2.5	DNB parameters	To be covered in bases	Re-Consider this STATUS	1,2,3,4 5 & 6.
10A	3/4.3.1	Source Range Neutron Flux	In proposed STS (NRC markup)		—
14A	Table 3.3.3	ESFAS instrumentation containment phase "B" isolation pressure hi-hi	In proposed STS (NRC markup)		—
15A	Table 3.3-4	ESFAS trip setpoints feedwater isolation	Under review		—
18A (Quest. 10)	3/4.4	RCS-hot shutdown	Under review		Shutdown
19A	3/4.4	Cold shutdown with loop filled	Under review		Shutdown

*Concerns and questions are from references 3 and 4 respectively.

CONCERN*	TS	SUBJECT	STATUS	MODES APPLICABILITY
29A (Quest. 16)	3/4.7	a. AFW system operability b. AFW instrumentation And related T.S items 4.7.1.2 of Rev. 30.	Covered by proposed STS	—
30A	3/4.7	MSIV's operability	Covered by proposed STS	Shutdown
31A	3/4.7	ADV's	Covered by new STS	—
32A	3/4.7.3	CCW-operability modes 5 & 6	Covered by definition of operability - no new spec. Re-Consider this STATUS	Cold Shutdown & Refueling
33A	3/4.7.4	SWS-operability modes 5 & 6	See 32A	—
35A (Quest. 19)	3/4.9.8	RHR-high water level	Under review	Refueling - High Water Level
36A (Quest. 20)	Page 3/4.9-11	Refueling operations - low water level	Under review	~~Shutdown~~ Refueling - Mid loop operations
38A	Table 2.2-1	RTS setpoints - low power reactor trip	In proposed STS (NRC markup)	—

CONCERN*	TS	SUBJECT	STATUS	MODES APPLICABILITY
3B	Table 2.2-1	a. P-7 permissive b. pressurizer water level high	In proposed STS (NRC markup)	___
10B	3/4.3	P-11 interlock	Under review	___
12B (Quest. 5b)	Table 3.3-3	ESFAS-autoswitchover on RWST level	In proposed STS (NRC markup)	___
15B (Quest. 8a, 8b, 8c, 8d, & 8e)	3.4.4.1	RCS loops	Under review	___
20B	3/4.7.5	Ultimate heat sink operability modes 5 & 6	See 32A	Cold Shutdown & Refueling
21B	Page No 3/4.9-11	Refueling operations-low water level	Under review	~~Shutdown~~ Refueling - Mid loop operations
	Page No 3.4.4-6.	Reactor Coolant System - Cold Shut Down, loops Are Not Filled	Should add to Review (Diablo Canyon Event)	Mode 5.

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

ENCLOSURE-1 PLANT-SPECIFIC DPO ISSUES RESOLVED BY TECHNICAL SPECIFICATION AMENDMENT

ENCLOSURE-2 PLANT-SPECIFIC DPO ISSUES RESOLVED BY UPDATING FSAR AND SETPOINT METHODOLOGY (SPM)

ENCLOSURE-3 PLANT-SPECIFIC DPO ISSUES REQUIRING NO LICENSEE ACTION

ENCLOSURE 1

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS
PLANT SPECIFIC DPO ISSUES RESOLVED
BY TECHNICAL SPECIFICATION AMENDMENT

Question 6a
Table 3.3-4,
Item 4d
(Reference 4)

Include response time in the definition of of the setpoint and provide appropriate descriptors for the values in the TS.

Issue

Technical Specifications Table 3.3-4 specifies the Engineered Safety Features Actuation System Instrumentation trip setpoints and allowable values for various functional units. Item 4d addresses Negative Steam Line Pressure-Rate-High for Steam Line Isolation.

TS Values' descriptors are inconsistent in their format with respect to setpoint methodology values and inclusion of a negative sign is redundant to the setpoint definition.

Resolution

The licensee changed the descriptor in the TS to make it consistent with the descriptor for the setpoint methodology values and eliminated a negative sign for better clarity.

These TS changes are administrative in nature. The staff approved these changes in TS Amendment 102 (Unit 1) and TS Amendment 84 (Unit 2) respectively.

Questions 7d, 7i and 7k, Table 3.3-5, Item 2e Table 3.3-5, Item 3e Table 3.3-5, Item 4e	Clarify the inconsitency between the TS values and FSAR values for these items.

Issue

TS Table 3.3-5, lists the engineered safety features response time. Items 2e, 3e and 4e indicate that response time is "N.A." for the Containment Purge and Exhaust Isolation Systems for Containment Pressure-High, Pressurizer Pressure-Low-Low and Steam Line Pressure-Low initiating signals.

FSAR offsite consequences accident analyses took credit for the contaimnent purge and exhaust system isolation and assumed 4 seconds as response time in the analyses. FSAR Section 9.5.12.3 indicates closure time for these valves is 3 seconds and FSAR Section 7.3.1.2.6 indicates a 1 second response time for generating an engineering safety feature actuation signal.

Resolution

The licensee proposed a TS change to make safety analysis values and TS values consistent by including 4 second response times for items 2e, 3e and 4e in TS table 3.3-5.

The staff approved these changes in the TS Amendment #102 (Unit 1) and TS Amendment #84 (Unit 2) respectively.

Question 71 Table 3.3-5, Item 4h	Clarify the inconsistency between the safety analysis value and the TS Value for steam line isolation response time.

Issue

FSAR feedwater system pipe break analysis sequence of events Table 15.2.3-1 indicates that the low steam line pressure setpoint is reached in the ruptured steam generator in 420 seconds, and that all main steam line isolation valves would close in 427 seconds. Based on this information, the response time assumed in the safety analysis for steam line isolation is 7 seconds. The TS allows steam line isolation time of 9 seconds.

Resolution

The licensee propsed a TS change to make the allowed steam line isolation response time 7 seconds which is consistent with the FSAR. This TS change was approved by the staff in the TS Amendment #29 (Unit 1) and TS Amendment #10 (Unit 2) respectively.

Question 7n
Table 3.3-5,
Item 6b

Clarify the inconsistency between the safety analysis value and the TS value for feedwater isolation response time.

Issue

Table 15.1.2-1 in the FSAR indicates that following an excessive feedwater flow event at full power, a High-High Steam Generator water level signal is generated in 27 seconds and feedwater isolation valves close in 36 seconds. Consequently, the actual feedwater isolation time is 9 seconds; however, the TS lists 13 seconds for feedwater isolation.

Resolution

The licensee proposed a TS change to make feedwater isolation response time in the TS 9 seconds, which is consistent with the FSAR. This TS change was approved by the staff in the TS Amendment 102 (Unit #1) and 84 (Unit #2) respectively.

Question 15
TS 3/4.5.3

Clarify the inconsistency between the TS and FSAR concerning the number of ECCS pumps operable when the RCS temperature is less than or equal to 300°F with respect to low temperature overpressure protection (LTOP).

Issue

TS 3.5.3 presents ECCS subsystems - Tavg $\leq$ 350°F during Mode 4 operation. The footnote states that a maximum of two ECCS pumps--one centrifugal charging pump and one safety injection--pump shall be operable whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F.

The licensee performed the low temperature overpressure protection analysis (FSAR 5.2.2.3) assuming only one pump operation when the RCS temperature is less than or equal to 300°F.

Resolution

The footnote for TS 3.5.3 calls for two pumps to be operable, however, the plant procedures permit only the centrifugal pump to be lined-up for injection to the reactor vessel. The safety injection pump will be operable and may be run in the recirculation mode; however, the safety injection pump flow path to the reactor vessel is normally blocked with closed valves not actuated on safety injection. Thus, only centrifugal charging pump could inadvertently inject during this mode which is consistent with the FSAR analysis. However, the licensee is in process to revise the footnote to make it consistent with the FSAR analysis.

During the review process, the staff found that TS 3.4.9 concerning pressure and temperature limits for heatup and cooldown curves had been revised such that the threshold for LTOPs protection shifted to 320°F from 300°F; but that the reference to this temperature threshold in the footnote to TS 3.5.3 had not been revised accordingly. This inconsistency was not identified as a DPO issue; but rather, found incidentally during the review of the above DPO issue. The staff has discussed this subject with the licensee and Darl Hood, the NRC Project Manager for McGuire. The licensee is in process of revising the TS 3.5.3 to be consistent with the TS 3.4.9.

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Question 15 . TS 3/4.5.3

Resolution:

1. The licensee action will only be in conformance with the FSAR if the TS includes the specific requirement that under these conditions the Breakers for the Safety injection pumps will be opened, locked and tagged.

2. The necessary reduction in threshold temperature was also identified inside the DPO review under TS Section3/4.5., Item "5.2.2 Overpressure protection", page 87 of the original review (Ref.30), last para. Excluded from consideration in the original 220 items, other FSAR commitments essential to the same over-pressure protection need to be addressed by this Review although they are currently excluded, by earlier selection: They are detailed in the writer's original review to Reference 30.

Action:

1 Licensee Amendment of the TS should incorporate the provisions described above.

2. Potential additional necessary TS Amendments deriving from additional essential protections outside the current review but included in the original review to reference 30 , should now be evaluated

ENCLOSURE 2

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS
PLANT-SPECIFIC DPO ISSUES RESOLVED BY UPDATING FSAR

Question 4a/4b TS Table 3.3-2, Items 9 and 10 (Reference 4)	Resolve the inconsistency between the TS response time value of $\leq$ 2.0 secs with respect to the value for pressurizer pressure (low and high) on page 7.2-14 of the FSAR.

Issue

TS Table 3.3-2, items 9 and 10 provide the maximum allowable pressurizer pressure (low and high) reactor trip response time which are greater than the nominal value given in chapter 7 of the FSAR.

Resolution

The licensee has updated page 7.2-15 in the FSAR to make reactor trip response time consistent with the TS for pressurizer pressure (low and high) trip functions.

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Question 5a. Table 3.3-3. Item 7g:

Issues:
An additional Issue was the validity of preventing automatic actuation of the motor driven auxiliary feedwater pumps below 3# because of the licensing basis need for protection under these circumstances, including Mode 4.

Resolution:

The following comments are made on Resolutions identified as 1,2,and 3 and should be incorporated into the final report:

Resolution 1: This statement is categorically incorrect. The definition of the Operational Mode inside the TS Table on TS Page 1-9 is Grossly Deficient. The definitive limits are established by Process Safety Analysis Limits used in calculating some of the most severe Licensing Basis Transients and Accidents for the facility and which occur from Zero Power in Operational Mode 2, Start Up: For Mc Guire these are represented by a Plant Setpoints for an Operating Pressure of 2235 psig, and of 557.1 deg F for the average temperature of the RCS: Any Operating Pressure less than this value in non-conservative, and the non-conservatism of a lesser value of Average RCS Temperature has already been discussed under Question 2 of this evaluation.

Action: An Applicability Mode 2# is invalid and thereby cannot be used. If the * condition were to be used, it would have read as Modes 1,2 and 3# as proposed by the writer.

Resolution 2: The licensee has agreed with the proposition that the blockage of the trip in Mode 3 below Mode 3# is not acceptable. The licensee has not responded on the need for operability in Mode 4: Application to this Mode is necessary , as the RHR system for each of the Units is subject to complete loss by a single failure thereby requiring the steam generator system as the only alternate mode of decay heat removal.

Action: The licensee should now change the TS's to revise the Applicability to at least Modes 1,2,3 and 4. Further it is unnacceptable and unsafe and invalid to delay action until the new STS development program is established at the Mc Guire Facility .

Question 6b
Table 3.3-4,
Items 7c(1) and (2)

Clarify TS items 7c(1) and 7c(2) concerning the Auxiliary Feedwater system initiation and the flow distribution following a feedwater line break.

Issue

TS Table 3.3-3 presents Engineered Safety Features Actuation System Instrumentation. Items 7c(1) and (2) discuss the auxiliary feedwater system initiation by the steam generator water level-low-low signal. Information in the table indicates that low-low level in one steam generator is necessary to start the motor driven pumps and low-low level in at least two steam generators is necessary to start the turbine driven pump. The reviewer questions whether the level in the intact steam generator will be low enough during the feedline break incident to result in a start of the turbine driven AFW pump.

Resolution

In the case of a feedwater line break, the auxiliary feedwater system is designed to deliver 450 GPM by either turbine driven pump or two motor-driven pumps to three intact steam generators while feeding one faulted generator.

In the McGuire feedwater line break analysis, it was assumed that: (1) the turbine driven pump failed as the single failure consideration; (2) One motor driven auxiliary feedwater pump supplies 110 gpm to an intact SG (the remainder spills out the break in the faulted loop); and (3) the other motor-driven pump supplies 170 gpm to each of the other two intact steam generator; thus maintaining

450 gpm as total flow to three intact system generators. These assumptions are consistent with the design of the AFW system instrumentation and TS requirements for that instrumentation.

In the case of a single failure of a motor driven pump, it is assumed that the turbine driven pump can actuate on low-low level in at least two steam generators. The licensee has calculated that during this accident condition, 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 the bubble collapse from this reduced mass condition would cause low-low level to be reached in the other steam generators.

Thus, in the case of a motor-driven pump single failure consideration, the turbine-driven pump can actuate on low-low level in two steam generators and would maintain 450 gpm flow distribution similar to the motor-driven pump to the intact SGs. Thus, with either motor-driven pump or turbine drivin pump single failure consideration, the auxiliary feedwater system can deliver the designed flow of 450 gpm.

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Question 6b. Table 3.3-4. Items 7c(1) and (2)

Resolution

Comments by the reviewer:

1. The TS Items 7c(1) and 7c(2) are correct.

2. Ref. para. 3. second sentence : The licensee has assumed , not calculated , that during this accident condition , the mass inventory in the intact steam generators is reduced significantly prior to reactor trip on low-low level . The Topical Report on this issue , reference 32. makes no such " a priori " assumption.

3. Ref. para.2. first sentence , their is no information in either Section 7 or 10 of the FSAR,which would show how the specified flow distribution is obtained . This however remains part of a complete set of other concerns arising from this review and detailed under a) Question 16. Table 4 of this Review , for evaluation as a Generic Issue,and to which should be added b) the directly related TS Item 4.7.1.2., Surveillance Requirements from the writers initial review (Refce. 30).

Question 6c
Table 3.3-4,
Item 9

Confirm the bases for the setpoints and allowable values as specified in the TS.

Issue

TS Table 3.3-4, Item 9 presents ESFAS instrumentation trip setpoint and allowable value for 4KV Emergency Bus Undervoltage-Grid Degraded Voltage (Loss of Power). Reviewer requested that bases for setpoints be confirmed.

Resolution

The NRC staff issued a generic letter, dated August 12, 1976 requesting all licensees to analyze their Class 1E electrical distribution system to determine if the operability of safety related equipment could be adversely affected by short term or long term degradation of grid system voltage. A supplemental generic letter issued June 2, 1977 provided staff positions pertaining to degraded grid voltage protection and the selection of voltage and time setpoints, and appropriate technical specifications. The licensee's responses, including setpoints, were reviewed by the staff and found acceptable as discussed on Page 8-1 of Supplement 1 to the SER.

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Question 4a//4b, TS Table 3.3-2, Items 9 and 10(Reference 4)

Resolution:

The following comments are made on the existing resolution :

1. The answer provided by the licensee is invalid. The question is, what is used in the Safety Analyses?, and this is the value required for the FSAR and the starting basis required for determining the TS.

2 Reference 8 , page 212-59, Rev.28 shows that delay time for these two initiators was increased from an old value of 1.0 secs. to 2.0 secs.; the 2 secs. is therefore the correct value, and the value of 1 secs. currently in the FSAR must thereby be corrected.

Action: The Licensee has already completed the necessary update the FSAR.

Question 4c TS Table 3.3-2, Item 17	Clarify whether the reactor is tripped due to pressurizer pressure-low signal or pressurizer pressure-low-low (ESFAS/safety injection) signal during an accidental depressurization of the main steam system; and if so, include the appropriate response time in Table 3.3-2. Also, clarify terminology used in Note e for Table 7.2.1-4 in the FSAR.

Issue

A. TS Table 3.3-2, lists the reactor trip instrumentation response times. Item 17 in the table lists the input response time as "N.A." for pressurizer pressure-low-low-(safety injection). This would appear to be incorrect if this trip function is relied upon to mitigate the transient associated with depressurization of the main steam system.

B. Note e for Table 7.2.1-4 in the FSAR makes reference to a pressurizer low pressure-low level trip. This should be pressurizer pressure-low-low (safety injection).

Resolution

A. During the transient associated with depressurization of the main steam system, the reactor will trip at 1945 psig with the pressurizer pressure-low function during the transient. The pressurizer pressure-low-low (SI) setpoint is 1845 psig. Since this trip function is not utilized to mitigate accidents other than LOCA, the TS will continue to list "N.A." in the TS Table 3.3-2. The actual response time of 2.0 seconds is listed for this ESFAS function under item 3b of TS Table 3.3.5. Therefore, the present TS is correct and remains the same.

B. The licensee will revise the FSAR Table 7.2.1-4, Note e for better terminology and clarity.

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QUESTION 4.C.Table 3.3.2 , Item 17 : REACTOR TRIP INSTRUMENTATION RESPONSE TIMES.

Question

The current para. should be replaced by :
Inaccurate description of Functional Units initiating Reactor Trip. Incomplete TS, and Inaccurate description in Reactor Trip Correlation Table of the FSAR.

Issues:

The current two paras. should be replaced by the following:

1. Functional Unit described as "Safety Injection Input from ESF" is incorrect. TS descriptors should be replaced by four functional units consistent with Table 3.3-5 ; i.e. by- Manual Safety Injection ,Containment Pressure-High, Pressurizer Pressure -Low (SI) and Steam Line Pressure _ Low.

2. Related Response Times ommitted from TS by proposing as Not Applicable (N.A).

3.Absence of docketed information for times used in related Accident Analyses, and particularly for MSLB , SBLOCA and LOCA events.

4. Clarify initiator of Reactor Trip for Accidental Depressurization of the Main Steam Line under Reactor Trip Correlation Table 7.2.1-4(5 of 5)

Resolution

Issue 1. No Response from Licensee

RBAL Position - Reference response under Issue 2 below. Reference also comments under Questions 7b and 7g

Proposed Action : TS descriptors should be replaced four functional units consistent with Table 3.3-5; i.e. by: Manual Safety Injection ,Containment Pressure-High, Pressurizer Pressure Low (SI) and Steam Line Pressure _ Low.

Issue 2. Licensee responds that trip functions not utilized in FSAR transient and accident analyses will have the requirement indicated as Not Applicable (N.A.).

RBAL Position- This position is incorrect and thereby Unacceptable. An essential regulatory requirement is diversity of Protection Systems so that all licensing basis transients and accidents will in general have at least two separate parameters initiating protective action. Also Transient & Accident (T&A) analysis will also generally be undertaken with the second out trip .or other later trip, giving the most conservative evaluation considered necessary for the expected consequences of the Occurrence . In this regard it should be noted that for the parameters in question , examples include LOCA and MSLB Breaks inside and outside containment , both small and large ; and such breaks in modes 3 and 4: For transients , the excessive cool down resulting from failure open of the main feedwater valves is an event where this is use as back up parameter . As a first out, or diverse protection , this reactor trip is especially important for events below the P-7 permissive when direct reactor trip from another parameter may not be available.

Proposed Action: The term NA alongside item 17 in this Table 3.3-2 should be replaced by the response times used in the Accident Analyses. Note the actual response times are included in Table 3.3-5 and under the more accurate descriptors required of Issue 1 above

Issue 3: The writer has discovered docketed information and which is different from that of existing TS values . Reference response to Questions 7b an 7g. The corrected values should be inserted in this Table 3.3-2, Item 17.

Issue 4: The licensee has agreed to revise the FSAR TABLE 7.2.1-4 . Note e to improve terminology to clarify the reactor trip initiator for the event as Pressurizer Pressue -Low (Safety Injection) (as distinct from Pressurizer Pressure- Low, which is has a higher setpoint for Reactor Trip)

ENCLOSURE 3

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

RESOLUTION OF PLANT-SPECIFIC DPO ISSUES REQUIRING NO LICENSEE ACTION

Question 1
Table 2.2-1
(Reference 4)

Confirm the validity of McGuire Units 1/2 steam generator instrumentation, setpoint and their applicability. McGuire Unit 1 has D-2 steam generators and McGuire Unit 2 has D-3 SG.

Issue

Steam Generators D-2 and D-3 have a minor design difference at SG bottom plate. Both SGs have identical instrumentation hardware and setpoint.

Resolution

The licensee performed a conservative safety analysis which is applicable to both units. Instrumentation setpoints values are based on this analysis. Westinghouse RPS/ESFAS setpoint methodology is applicable to both units and approved by the staff.

COMMENTS BY R. LICCIARDO ON K. DESAI RESOLUTION
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Question 1. Table 2-2-1. (Reference 4)

Question

The current Question should be replaced by the following:
Confirm the validity of using McGuire Unit 1 Set Point (SP) Methodology for checking the TS's for McGuire Unit 2

Issue:

Mc Guire 1 SP Methodology was used to check set points for Mc Guire 2 TS's. Is this valid ?.

Resolution:

Licensee advises the only significant difference in this respect is that Unit 1 has D-2 Type Steam Generators , whilst Unit 2 has D-3 Type with a related minor design difference at the SG bottom plate. Both types have identical instrumentation hardware , and thereby instrumentation errors and drifts, so that transient and accident evalution for both units using the same safety analysis limits gives the same TS setpoints. These circumstances validate the use of the same Set Point methodology for both units as has been done for this review.

Question 1a
Table 2.2-1
Item 3

Verify that a time constant of > 2 seconds results in a slower response time which is less conservative.

Issue

TS Table 2.2-1 represents reactors trip system instrumentation trip setpoints including response time. TS Table 2.2-1, Item 3 - concerns power range, neutron flux, high positive rate trip during a control rod ejection accident.

Resolution

An increased time constant results in a faster response and thus results in a shorter time from initiation of a transient to reactor trip. The analysis assumes a time constant of 2 seconds. Therefore, the time constant of > 2 seconds is conservative.

Question 1b Table 2.2-1 Item 4	(1)	Verify that a time constant of ≯ 2 seconds result in a slower response time which is less conservative.
	(2)	Resolve the inconsistency between setpoint methodology value and FSAR analysis value.

Issues

TS Table 2.2-1 Item 4 specifies power range - neutron flux, high negative rate during a control rod drop event. The reviewer questioned (1) the conservatism of the time constant used in processing the flux rate signal input to the RPS; and (2) the validity of statements in the setpoint methodology document which indicates that the negative flux rate setpoint was not used in the safety analysis for McGuire.

Resolution

(1) An increased time constant results in a faster response and thus results in a shorter time from initiation of a transient to reactor trip. Therefore, the time constant of ≯ 2 seconds is conservative.

(2) As indicated in the FSAR the negative flux rate trip setpoint was evaluated as part of the safety analysis for McGuire. The setpoint methodology document was indeed in error. The licensee has revised the setpoint methodology Table 3-4 to show a safety analysis limit of 6.9 % rated thermal power. TS trip setpoint and allowable values remain the same.

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Question 1b, Table 2.2-1, Item 4.

Resolution

Considering Resolution (2): This should be replaced by the following:

The negative flux rate trip setpoint was not evaluated as part of the safety analyses for Mc Guire as their was no approved Evaluation Methodology for the related Transient. The setpoint methodology document was indeed in error. A later NRC approved Evaluation Methodology has now been used and the licensee has revised the Setpoint Methodology Table 3-4 to show a safety analyses limit of 6.9% rated thermal power. This value permits the TS trip setpoint an allowable values to remain unchanged.

Action: Table 2 should show Updating of the FSAR to record these changes in the related safety evaluation, and it should also show an update of the Analysis Of Record (AOR) in that the Set Point (SP) Methodology has also been changed.

Question 1c
TS Table 2.2-1,
Item 9

Resolve the disparity between the setpoint methodology value and the FSAR safety analysis value.

Issue

The setpoint methodology safety analysis value for pressurizer pressure-low is 1845 psig. While the FSAR value for the same analysis is 1835 psig.

Resolution

The licensee has indentified the correct value to be 1835 psig. No change to the FSAR or TS was necessary.

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Question 1c. TS Table 2.2-1, Item 9

Resolution

The following should be added:

A change is also required in the Set Point Methodology to record the change in the safety analyses limit from 1845 to 1835 psig, and this should be noted in Table 2 as an related update.

We also note that this is also a non-conservative change from the original value and the licensee should Docket additional evidence as to when it occurred.

Question 1d
TS TAble 2.2-1,
Item 13

Verify that the FSAR safety analysis value assumed in the feedwater line break analysis is lower than the TS setpoint value.

Issue

TS Table 2.2-1, item 13 lists steam generator water level-low-low reactor trip setpoint and allowable value. The reviewer questions whether the allowance for instrument error and uncertainties was applied in a conservative manner to arrive at the safety analysis value listed in the setpoint methodology document.

Resolution

The setpoint specified in the setpoint methodology document does suggest a non-conservative application of the allowance for channel error and drift. However, this value (i.e W STS + 10%) was not used in the McGuire TS. As discussed below, the allowance for instrument error and other uncertainties has been properly applied for McGuire.

The licensee performed the limiting feedwater break analysis starting at full power and assuming a low water level trip setpoint of 23% narrow range span. The McGuire TS limit for the SG low-low water level trip setpoint, at 100% rated thermal power is 40% of narrow range span which exceeds the safety analysis value of 23% narrow range span by more than 10%.

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Question 1d, TS Table 2.2-1, Item 13

Resolution

This resolution should read as:

The setpoint specified in the setpoint methodology document was a non- conservative application of the allowance for channel error and drift

The licensee has changed the bounding analysis event for this parameter to that of the Main Feedwater Line Rupture initiating at full power and assuming a low-low water level Safety analyses Limit of 23% of narrow range span. The licensee now states that the Mc Guire TS setpoint for the SG low-low water level trip , at 100% rated thermal power, "is now 40% of narrow range span which exceeds the safety analyses limit value of 23% narrow range span by more than 10%".

Action: This change in Safety Anaysis Limit for the SG should be be reflected in a necessary amendment to the Set Point Methodology Report for Mc Guire Unite 1&2 .Ref. 18, and entered into Table 2 as an Update to the AOR and also as a change to the FSAR (from the original value of > or= 54.9%). It should also appear in Table 3 as an amendment to the TS.

Additional Information is also required:
Since Reactor Trip and Auxiliary Feedwater Initiation are initiated by the same sensors and Logic, please clarify why a Safety Analysis Limit of 35% is used for Reactor Trip on the Loss Of Normal Feedwater Event (FSAR page 15.2-13, 1985 Update) whilst 23% is used for the Main Feedwater Line Rupture Event. Also clarify why the Set Point of 40% in less than the 49% contained in the W Set Point Study to refce.33, fig .3-2.

Question 1e
Table 2.2-1,
Item 18b

Clarify whether pressurizer pressure - low signal or pressurizer pressure - low (safety injection) signal trip the reactor during an accidental depressurization of the main steam system from zero load.

Resolution

An accidental depressurization of the main steam system (inadvertent opening of a dump valve, safety valve or relief valve) is initiated from hot shutdown conditions at zero power which is the most conservative initial condition. Reactor is already tripped at the beginning of the transient (hot shutdown condition). Thus, no explicit assumption is made regarding the cause of reactor trip for the FSAR analysis. No credit is taken for the reactor trip on pressurizer pressure when reactor power is below the P-7 interlock.

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Question 1e Table 2.2-1, Item 18b

Question

The following comments are made :

The descriptor "18b" is incorrect and should be replaced by "18 c(i)(last para)"

Replace the Introductory paragraph with the following :

At less than 10% RTP, absence of the P-7 permissive prevents reactor trip on Pressurizer Pressure-Low : Clarify how the Reactor Trip is initiated when Accidental Depressurization of the Main steam line occurrs from these conditions, and does the current evaluation for this Occurrence remain valid under these circumstances.

Resolution:

The existing para. should be replaced by :
The licensee response is that the reactor is tripped either by the overpower delta T trip, which is not blocked by absence of P-7, and or by the the initiators of safety injection , namely : pressurizer pressure- low (safety injection), steam line pressure-low , and containment pressure -high. The safety evaluation which assumes the reactor is tripped at the commencement of the event , remains valid, as this is more conservative when the reactor is not already tripped at the initiation of the event; refce. FSAR Section 15.2.13.2, Revision 43.

Question 2
TS Page 3/4 1-6
(TS 3.1.1.4)

Clarify why the existing minimum temperature for criticality (Modes1/2) is 551°F which is less than the programmed setpoint minimum value of 557°F for events from zero power.

Issue

The reviewer is concerned that transients or accidents may be initiated at zero power conditions from a temperature lower than the programmed setpoint minimum value of 557°F, i.e. the allowed minimum temperature for criticality of 551°F.

Resolution

Accident evaluations for events from zero power are performed using the programmed setpoint minimum value of 557°F. The difference between the hot zero power temperature and minimum temperature for criticality limit is required in order to allow for measurement of the moderator temperature coefficient. For most plants the minimum temperature for criticality is lower than hot zero power temperature.

The change in initial condition from 557°F to 551°F for transients occuring at hot zero power would have a negligible impact on results and would be a less representative input condition since the majority of time spent at hot zero power conditions is at a temperature of about 557°F.

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Question 2, TS Page 3/4 1-6, (TS 3.1.1.4)

Question

The opening statement should be amended to read as:
Clarify why the existing minimum temperature for criticality (Modes 1/2) is 551 deg. F which is less than the programmed setpoint minimum value of 557 deg. F used for the Licensing Basis caculations of all Occurrences from Zero Power.

Issue

Should be amended to read as :
The reviewer is concerned that transients or accidents may be initiated at zero power conditions from a temperature outside the Licensing Basis calculations and so place the plant in and Unanalyzed Safety Condition .

Resolution

Should be replaced by :

Licensee advises that the difference between the hot zero power temperature and minimum temperature for criticality limit is required in order to allow for measurement of the moderator temperature coefficient .

The licensee provides a qualitative evaluation which does show that indeed the reduction of temperature is non - conservative in the evaluation of significant Occurrences from zero power . The licensee does not provide any details of existing margins to safety for these Occurrences nor calculations of the reductions in margins and their significance.

Licensee proposes that the change in initial condition from 557 deg.F to 551 deg F for transients and accidents occuring at hot zero power would have a negligible impact on results and would be a less representative input condition since the majority of time spent at hot zero power condition is at a temperature of about 557 deg. F.

Action:

The licensee should be advised that the Qualitative Evaluation provided is Unacceptable in meeting the Regulatory requirements for safety analy``s during the proposed experiments under 10CFR50.59, and the argunents based on probablility of being within that temperature range is an infringement of TS requirements under 10CFR50.36 .

Further-more, the unacceptability of this proposition is based not only on the non-conservatism of the reduced temperature, but also of

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the departure from other analytical condi ons with proposed operating pressures of less than 2235psig., and a k eff. of > 1, as currently provided in the Applicability LCO's. If the licensee wishes to pursue this matter , he must do so under the terms of an experiment under 10CFR50.59, in which the acceptability for protection against all related approporiate T&A's would need to be evaluated. Or other circumstances in which operating safety in MODE 3 has been fully evaluated including the totality of the particular circumstances being proposed, but as yet undefined.

The writer notices that this concern may not only restricted to Mc Guire TS's, but may also be applicable to all other facilities using Standard TS's, and thereby be a generic issue.

Question 3
TS Table 3.3-1,
Item 6c

Verify that during shutdown in Modes 3, 4 and 5 with reactor trip system breakers open, source range and neutron flux channel operability TS requirements specify only one channel operable while FSAR requires two channels to be operable.

Issue

Technical Specifications require 2 source range neutron flux channels be operable at all times except when in modes 3, 4 and 5 with the reactor trip breakers open. Reviewer suggested that assumptions of boron dilution analysis would require 2 operable channels at all times.

Resolution

The licensee has determined that boron dilution events during modes 1, 2 and 6 were analyzed for the McGuire units. Consequently, the McGuire safety analysis does not provide a basis for requiring two operable source range channels during modes 3, 4 and 5 of operation. The licensee has considered changing technical specification 3.3.1 to require two operable source range channels at all times during operation in mode 3, 4 and 5; but has instead choosen to follow staff guidance in Generic Letter 85-05 to take action to assure that adequate protective measures to avoid boron dilution events are in place.

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Question 3. TS Table 3.3.1. Item 5c.

Resolution

The following comments are made on the licensee's proposed resolution:

Ref. FSAR. Section 15.2.4.2. Revision 10 .opening para. states :

"To cover all phases of the unit operation. boron dilution during refueling , startup and power operation are considered in this analysis" .

So it was the intent of the FSAR that the evaluation of this particular Occurrence of Boron Dilution cover all operational Modes 1-6. and including 5- 2 in the general description of the start up mode and not to restrict it literally to "Start Up" as only partially defined on TS Page 1-9.

Ref FSAR Section 15.2.4. page 15.2-15 states that. "

"Dilution during start up: prior to start up , the RCS is filled with borated (app. 2000 ppm) water from the refuelling water storage tank. core monitoring is by external BF3 detectors. Mixing of the reactor coolant system is accomplished by operation of reactor coolant pumps. High source range flux level and all reactor trip alarms are effective".

Note that Mc Guire is committed to operate in a manner in which the RCS is to be retained with a boron concentration equal to that required in the Cold Shut Down Mode 5. at any time the plant is in Modes 5-2. to facilitate protection against return to power events : but by default. this requirement has never incorporated into the TS'S: reference 16. page Q212-47e. and reference 30. Section 3.4.1. page 10.

Ref. FSAR ref: 8. page Q 212-24 Revision 10 under 212.58. the staff's position is that:

"unless permanent plant alarm and indication and the temporary core monitoring systems are designed in conformance with criteria established for safety systems. they should not be used to perform functions that are essential to safety. Confirm that your design will comply with the staff's position".

Further note that SER reference 10. Section 15.2.1. last para. states that :

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"for the postulated boron dilution event at startup, the applicant relies on the neutron detector counting rates to alert the operator: we will require that a separate alarm be provided or that the applicant isolate all sources of unborated water during startup or shut down".

And that :

"During startup and shutdown, the applicant shall rely on the source range high flux alarm to alert the operator that a dilution event is o urring: thereafter he has approximately 26 minutes to dete..ine the event before all shutdown margin is lost. This assessment is based on setting the alarm at a level 5 times the background level. The staff requires that tha applicant modify his operating procedures so as to maintain the source range alarm setpoint at this level or lower(equal to or less than 5 times the background level) any time the plant is in the shutdown mode: the alarm setpoint is to be checked and adjusted, on a weekly basis. This will assure suitable time for operator action should a dilution event occur. This matter will be included in the technical specifications . In addition, we require that procedures be developed that preclude any boron dilution after reactor scram until the neutron flux level is below thw level of the source range high flux level alarm".

The Licensee was required to have these commitments incorporated into the TS's for the 1981 Start Up of McGuire 1, and the 1983 Start Up of Mc Guire 2, and by default the licensee has never proposed them.

Concerning GL 85-05 : It would not be a Backfit --These are 1978 through 1981 commitments never met for the commencement of operations in 1981, and discovered by the writer in 1983-4, and thereby remains an outstanding default.

Action: The proposed TS's were invalid and remain invalid until they conform to FSAR commitments by having at least two Source Range Neutron Flux channels being operable in Modes 5-3 with effective alarms whilst the reactor trip breakers are in the open position.

Question 5a Table 3.3-3 Item 7g	Clarify whether applicable modes, Modes 1 and 2 # is appropriate or it should be modes 1 and 3 # under P-11 interlock.

Issue

TS Table 3.3-3 presents Engineered Safety Features Actuation System Instrumentation. Item 7g specifies applicable modes and operability requirements for auto-start of motor driven auxiliary feedwater pumps (motor-driven pumps) on trip of all main feedwater pumps. The reviewer questioned whether this feature could be blocked during Mode 2 below the P-11 interlock because the threshhold for P-11 could not be reached while in mode 2.

The # sign states that trip function may be blocked in this mode below the P-11 (pressurizer pressure interlock setpoint) and which can occur only in mode 3, therefore, the reviewer believes that condition should be on mode # 3.

Resolution

The statement that P-11 can only occur in mode 3 is inaccurate. 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 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 switch will prevent auto-start on low-low steam generator water level (TS Table 3.3-3, Item 7c(1). Since this auto-start capability is required in Modes 1, 2 and 3, blocking is not allowed in these modes. The # is misleading and will be eliminated by the licensee during the new STS development program.

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Question 6c, Table 3.3-4, Item 9.

Issue:

The following should be added:
A leading purpose of the question was to discover whether the setpoint was conditioned by the undervoltage trip setpoint for the reactor on the reactor coolant pump busses.

Resolution

The following paras.. should be added:

The licensee response confirms that the setpoint for the Emergency Busses allows them to be Unloaded of all Non ESF loads during 100% normal operation of the plant. without the reactor being tripped by the Undervoltage Trip on the RCP Busses, and consequently that after being transferred to DG supply, all of the Non-ESF loads will not be restored ,with a potential for affecting the continuing safe normal operation of the plant- without an analysis of the related consequences. At present, this represents an unanalyzed condition for the operating reactors at Mc Guire.

Action: The writer is advised that this this is potentially a generic issue.

Question 7a and 7f
Table 3.3-5, Item 2a
Table 3.3.-5, Item 3a

Clarify the inconsistency between the TS response time values and the FSAR values used in the LOCA analyses.

Issue

TS Table 3.3-5, lists engineered safety features response time. Items 2a and 3a provide Safety Injection (ECCS) response time of 27 seconds (without offsite power) due to containment pressure - high and pressurizer pressure-low-low initiating signals during LOCA analyses, respectively. Reviewer questioned the response time between items 2a, 3a and 4a.

Resolution

No LOCAs were analyzed for initial condition below P-11 interlock. Low head safety injection pumps are required during the LOCA cases which results in a response time of 27 seconds (without offsite power) for Items 2a and 3a as shown in the table below. Item 4a represents the main steamline break where the low head safety injection pumps are not expected to deliver flow because of the high RCS pressure. Consequently, the response time is shorter as indicated in the table below.

Therefore, the additional 5 seconds delay for low head safety injection pumps to attain their discharge pressure is not included in the safety analysis for steam line break.

TS Table 3.3-5 Item	Initiating Signal	TS Response Time
2a. Safety Injection (ECCS)	Containment Pressure-High	27 seconds
3a. Safety Injection	Pressurizer Pressure-Low-Low	27/12 seconds (without/with off-site power)
4a. Safety Injection (ECCS)	Steam Line Pressure-Low	22/12 seconds

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Question 7a and 7f: Table 3.3-5. Item 2a: Table 3.3_5. Item 3a.

Resolution:

The following comments are made:

The first sentence of para.1 commencing with " No LOCA'S _ _ _ is incorrect and should be deleted.

The second para. should be corrected to read :
Therefore, the additional 5 seconds delay for low Pressure safety injection (RHR) pumps........

Comments :

1. LOCA's below P-11 Interlock were evaluated and are a part of the Licensing Bases for Mc Guire Units 1 and 2. Reference Question 8c of TABLE 4 of this review concerning my Item TS 3/4.4.1. G 2.6.3.

2. Be advised that FSAR TABLE 15.4.1-5 (1 of 2). Rev. 43 shows Pump Injection for Large Breaks of Cd--1.0 DECL occurring at 26.1 secs. Therefore values 3a and 4a should show a value of no more than 26. secs. The tables should be changed to reflect this value.

3. One of the most limiting requirements for the High Pressure charging pumps is for the MSLB: ref. FSAR. Page 15.4-8. Rev.7 so that 22 secs. becomes the limiting value.

4. Since high pressure Safety injection on a small MSLB or small LOCA can be initiated from different SI initiators deoending on the circumstances . items 2a and 3a should include the 22 sec. requirement for HP Pumps as
item 2a 26/22

Item 3a 26/22/12
with an additional notation to this effect.

5. Why does not Containment Pressure-High have a TS Response time of 12 secs. when On- Site Power in available

Action: Change the TS Tables in accordance with the above comments.

Question 7b and 7g
Table 3.3-5, Item 2b
Table 3.3-5, Item 3b

Clarify the 2.0 seconds TS response time value versus the 1.0 seconds value on FSAR Page 7.3-8 value. The descriptor (from SI) is incorrect and should be deleted.

Issue

TS Table 3.3-5, items 2b and 3b provide reactor trip (from SI signal) response time of $\leq$ 2 seconds for containment pressure-high and pressurizer pressure-low-low initiating signals respectively.

The lower value of 1.0 second on FSAR Page 7.3-8 is the limit on the delay in receipt of SI actuation upon exceeding the high containment pressure setpoint.

Resolution

The response time listed in TS Table 3.3-5 is not related to 1.0 second limit in FSAR page 7.3-8.

The FSAR value of 1.0 second is the time it takes to generate a safety injection signal. The description "(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.

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Question 7b and 7g. Table 3.3-5. Item 2b: Table 3.3-5. item 3b

Issue:

The second para. should be replaced by :
The lower value of 1.0 secs. on FSAR Page 7.3-8 is the time required to initiate the SI sequence after the appropriate variable exceeds the setpoint (reference FSAR Section 7.3.1.2.6.).

Resolution:

Comments on second para:

The licensee proposition that the reactor trip is initiated by effectively the safety injection acuation sequence after the initial delay time of 1 sec.. is categorically incorrect. And this very faulted interpretation is due to the manner in which it is described in the TS and is the reason for the writer's submission that the current descriptor Reactor Trip (from SI), must be replaced by only Reactor Trip . Reader's should reference FSAR Fig 7.2.1-1 (8 of 10) Revision 34: Reactor Trip is not part of that Safety injection sequence initiated after the delay of 1 sec: Reactor is tripped directly from Cont.Press.-High. Pressurizer Pressure -Low (SI). and Steam Line Pressure -Low . Their is much additional logic before the SI signal itself is initiated from these parameters.

This also confirms the writer's proposition under previous question 4.C above that the descriptors of Reactor Trip (from SI) are categorically incorrect for related Item 4b from the same Table 3.3-5.

Additional Comments:

1. The above comments are further confirmed by FSAR Table 15.4.1-5. 1 of 2. Revision 43 for LOCA'S from Cd =1.0 to 0.4 DECL in which Reactor Trip signals initiate within 0.6-1.3 secs. of initiation of LOCA. and safety injection signals of 1.1 to 4.1 secs. occurr no earlier than these values. and even later.

2. More recent results in FSAR Table 15.6.4-2 and 3. 1987 Update show Reactor Trip response time to SI initiators of 0.46 secs.with longer times of 2.6 to 2.9 secs for the Safety Injection Signal

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3. The above information establishes that the Response times for Reactor Trip in question, i.e. TS Table 3.3-5, items 2b and 3b (and also 4d) should be less than or equal to 0.46 secs.

4. The same information confirms the FSAR value of 1.0 secs. as the appropriate (conservative) time taken to generate the SI signal.

Action:

1. For TS Table 3.3-5, Items 2b,3b,and 4b, the current descriptor Reactor Trip (from SI), must be replaced by only "Reactor Trip".

2. For TS Table 3.3-5 , Items 2b,3b,and 4b, the current response times of 2 secs.must be replaced by >or = 0.46 secs.

Question 7c and 7h
Table 3.3-5, Item 2d
Table 3.3-5, Item 3d

Justify the TS values used for containment isolation valves closure time for LOCA analyses.

Issue

TS Table 3.3-5, Items 2d and 3d list containment isolation-phase "A" (2) response times of 18 and 28 seconds for containment pressure-high and pressurizer pressure-low-low initiating signals for LOCA analysis with and without offsite power respectively. The reviewer questioned the acceptability of the containment isolation response times.

Resolution

The only isolation valves explicitly considered in the radiological consequences analysis of a LOCA include the containment purge, exhaust and the process line isolation valves which connect containment to the environment. The containment purge and exhaust valves will close in 4 seconds. The process lines with fluids will take longer time to close in comparison to the purge valves. The process lines valves will close in about 18 seconds (with offsite power). However, ANSI N271-1976/ANS 56.2, "Containment Isolation Provisions for Fluid Systems" recommends that, in general, closure times should be as low as reasonably attainable, based on manufacturers' recommended times and valve sizes, but generally not less than 15 seconds and in any case, no more than one minute. If these guidelines are met, releases through these process line valves before closure need not be modeled in the dose calculation. Therefore, the TS containment isolation valves closure time of 18 seconds is acceptable.

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Question 7c and 7h Table 3.3-5. Item 2d: Table 3.3-5. Item 3d.

Issue

The following comments are made:

Containment A Isolation also includes Reactor Coolant Pressure Boundary (RCPB) Valves which might also be functioning as Containment Isolation Valves . RCPB valves are special in that loss of RCS Inventory during closure impacts validity of ECCS analyses. So that as short a time as possible should be the General Basis consistent with any specific analyses using particular valves which should already have been incorporated in relevant TS's. RCPB valves other than these already considered in other TS's should be identified and evaluated against this concern to necessarily minimise Inventory loss .

Action: Licensee should review RCPB valves isolated by the Safety Injection signal to ensure shortest possible closure times consistent with any specific analyses using particular valves which should already have been incorporated in relevant TS's. Such closure times should be incorporated into the TS's

Question 7e
Table 3.3-5,
Item 2f

Clarify the TS concerning auxiliary feedwater system initiation on Containment Pressure-High in Modes 3 and 4.

Issue

TS Table 3.3-5, Item 2f provides auxiliary feedwater system response time for actuation from a containment pressure-high initiating signal as "N.A."

Resolution

FSAR accidents analyses do not take any credit for actuation of the auxiliary feedwater system from a containment pressure-high signal. Consequently, N.A. has been entered for the response time in table 3.3-5. However, the TS Table 3.3-5, Note 5 clarifies that the response time for motor-driven auxiliary feedwater pumps on all safety injection signals shall be less than or equal to 60 seconds. Response time limit includes opening of valves to establish safety injection path and attainment of discharge pressure for auxiliary feedwater pumps. The AFW response time as "N.A." is acceptable.

COMMENTS BY R. LICCIARDO ON K. DESAI RESOLUTION
OF PLANT SPECIFIC MC GUIRE TS REVIEW OF 05/14/90

Question 7 e. Table 3.3-5. Item 2f:

Question

This should read as:
Clarify the TS concerning auxiliary feedwater system initiation on Containment Pressure-High , and especially in Modes 3 and 4.

Resolution:

The following comments are made on the licensee submittal:

The statement that FSAR accident analyses do not take any credit for actuation of the auxiliary feedwater system from a containment pressure- high signal is categorically incorrect:.

1. Licensee must recognize a whole series of Breaks of Different Sizes in the RCS, Main Steam and Feedwater lines inside containment and at above P-7 , and below P-7, that require AFW. Reference one case- FSAR, Page 15 .3-2, Revision 27 for SBLOCA, with a specific Analysis and specification of AFW. Reference also Mainfeedwater Line Rupture.

2. Licensee must also recognize Containment High - as a necessarily diverse protective signal, required by Regulation, to the Pressurizer Pressure -Low (SI) and Steam Line Pressure -Low signals .

3. For necessary operation in Modes 3 and 4 , reference our comments under previous Question 5a of this Review. See also Ref.8 Page Q212-47b et. al. Revision 24 between 1900 and 1000 psig, and down to Mode 4. See also Ref.30, section 3/4.4.1, Item - General.

4. The NRC cannot accept a "Notation" concerning AFW which is a part of Safety Analyses, and also critical to maintaining a timely heat sink for the Decay Heat Removal from the Core under all Transient and Accident Conditions.

Action:Table 3.3-5 Items 2f,3f, and 4f, shall include response times of equal to or < 60 secs. against the item of Auxiliary Feedwater Pumps.

Question 7j
Table 3.3-5,
Item 3f

Clarify the TS concerning auxiliary feedwater system under pressurizer-pressure-low-low initiation signal.

Issue

TS Table 3.3-5, Item 3f provides auxiliary feedwater system response time as "N.A." due to pressurizer pressure-low-low initiating signal. The reviewer questioned the "N.A." entry for this item.

Resolution

The main steamline depressurization event (inadvertent opening of a steam generator safety, relief or dump valve) assumes ESF actuation on pressurizer pressure-low-low initiating signal. 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.

COMMENTS BY R. LICCIARDO ON K. DESAI RESOLUTION
OF PLANT SPECIFIC MC GUIRE TS REVIEW OF 05/14/90

Question 7j. Table 3.3-5. Item 3f.

Generalized comment:

The descriptor "pressurizer pressure -low -low" should be replaced by "pressurizer pressure - low (SI)" where-ever it used in this particular discussion.

Resolution

The licensee's position is very deficient in substance . The full licensing basis response to his position has already been provided under Question 7e. Table 3.3-5.Item 2f

The licensee has not recognized that the starting conditions for analyzing various Occurrences are modified to ensure additional conservatisms in a manner considered to be prudent considering the importance of the potential related consequences deriving from the event being analyzed. For the case of the main steam line de-pressurization it is prudent to accentuate the cooling effect by assuming that the AFW is initiated at the commencement of the event instead of at 60 secs. even though the logic does not provide for this. as it results in a more severe event to mitigate and protect against. However , it is ultimately necessary to isolate this flow into the faulted generator to prevent overpressurization of containment . but it remains necessary to continue to provide AFW to the intact SG's on a 60 secs. time basis as is used in the analyses to ensure system reponse according to evaluations and also from which recovery procedures can be determined. The licensee should consult Reference 31. Section 2.2.3.

Action: Items 3f. shall include response times of equal to or < 60 secs. against the item Auxiliary Feedwater Pumps.

Question 7m
Table 3.3.-5.

Confirm that the TS containment spray response time and FSAR analysis value are consistent.

Resolution

TS Table 3.3-5, Item 5a lists containment spray response time of ≤ 45 seconds following a containment pressure-high-high initiating signal. TS response time of 45 seconds is consistent with the FSAR containment analysis actuation assumption as shown in FSAR Table 6.2.1-16.

Question 7o
Table 3.3-5,
Item 12

Confirm that the TS automatic switchover to recirculation response time is consistent with the FSAR assumption.

Issue

TS Table 3.3-5, Item 12 lists response time $\leq$ 60 seconds for automatic switchover to recirculation resulting from a refueling water storage tank (RWST) level initiating signal. The reviewer questioned the basis for this value.

Resolution

The containment sump valves are interlocked with the RWST isolation valves to the RHR pumps such that these isolation valves will close when the containment sump valves reach their full open position. This automatic switchover provides an uninterrupted flow of water to the RHR pumps.

The automatic switchover to recirculation is initiated when the level setpoint is reached in the RWST. The plant procedures as delineated in FSAR Table 6.3.2-3A/3B test to ensure switchover delay of 60 seconds which is consistent with the TS response time.

COMMENTS BY R. LICCIARDO ON K. DESAI RESOLUTION
OF PLANT SPECIFIC MC GUIRE TS REVIEW OF 05/14/90

Question 7c, Table 3.3-5, Item 12

Question:

This should read more accurately as:

> Confirm that the TS response time for automatic switchover of ECCS to recirculation is consistent with FSAR analyses.

Resolution

The following comments are made on the licensee's response:

Referring to Table 6.3.2-3B . Update. The switchover is initiated at the 117,094 Gal. RWST Volume (Note 9). it opens isolation valves in Containment Pump Suction lines to the RHR Pumps (NI 184B and NI185A) and automatically closes isolation valves in suction lines from the RFWT to the RHR Pumpd(ND 19A and ND 4B). The writer finds no identifiable relationship of the TS value of 60 secs. to the Activities in this table. Licensee's response is Unacceptable.

Action : The licensee shall specifically clarify the 60 sec. TS value in his FSAR.

Question 9
Page 3/4 4-2
TS 3.4.1.2

Justify TS action requirement to restart an idle loop when in Mode 3 with no reactor coolant loops in operation; or explain how natural circulation is accomplished with emergency procedures.

Issue

TS 3.4.1.2, Action C states, "with no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the RCS and immediately initiate corrective action to return the required reactor coolant loop to operation." The reviewer questions the basis for these procedural actions and prepares alternate action which is to implement an EOP for natural circulation.

Resolution

For the condition of no reactor coolant loops in operation while in mode 3, the licensee will immediately initiate corrective action to restart the reactor coolant pumps to operation per the Abnormal Procedure, AP/1 and 2/A 5500/09," Plant Operations During Natural Circulation." If restart of reactor coolant pumps is not successful, natural circulation cooling is verified and maintained per this same procedure actions and their sequence are standard in the industry and are acceptable to the staff. It is to be noted that EOPs can only be entered following a reactor trip or safety injection.

COMMENTS BY R. LICCIARDO ON K. DESAI RESOLUTION
OF PLANT SPECIFIC MC GUIRE TS REVIEW OF 05/14/90

Question 9, Pa/ 3/4 4-2, TS 3.4.1.2

Questi.

This should be reworded as follows:

Justify TS action requirements to restart an idle loop when in Mode 3 , as their is no licensing basis evaluation defining safety limits and thereby acceptable TS limits inside which this can be safely achieved.

Issue

This should be reworded as follows:

TS 3.4.1.2, Action C states, "with no reactor coolant loop in operation in Mode 3, suspend all operations involving a reduction in boron concentration of the RCS and immediately initiate corrective action to return the required reactor coolant loop to operation". The reviewer questions the basis for the procedural action of restart of the reactor coolant loop whilst in this Mode, and under the related existing TS for reactivity control, as this action has never been analyzed under these circumstances and therefore represents and Unanalyzed Safety Condition for the facility . The only licensing basis action available under the existing TS is that of natural circulation. The licensee has been asked to evaluate and propose.

Resolution

Comments by the reviewer:

In his response the licensee has not addressed the need to determine safety limits and thereby TS for restart of a reactor coolant loop in this mode, and thereby is unacceptable .

Restarting a RCP without an adequate recognition and analysis of the prevailing conditions and consequences con cause a significant increase in reactivity, Reactor power, and reactor pressure. The licensing bases for Mc Guire provided for substantially increased Boration concetrations to approx 2000 ppm in Modes 3 -5, to mitigate these potential circumstances; but the existing TS are in default in not providing for such Boration levels. Therefore the plant is exposed to potentially undesirable consequences if the action proposed is undertaken at this time. This concern had been recognized as a Generic item under Section 3/4.4.1, G2.6.1 and Listed under Table 4 . Question 6a of this Memorandum.

Action: The licensee should be required to re-evaluate for his current TS, or borate to the level required by his existing safety evaluation under Ref.16 page Q 212-47e before initiating cooldown in Mode 3.

Question 11a
TS Section 3.4.5

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

Resolution

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

LOCAs in lower modes of operation and loss of RHR cooling in lower modes will be addressed generically in Question 5b.

Question 11b
TS 3.5

When the sytem is in the RHR cooling modes, 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.

Resolution

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

LOCAs in lower modes of operation and loss of RHR cooling in lower modes will be addressed generically in Question 5b.

COMMENTS BY R. LICCIARDO ON K. DESAI RESOLUTION
OF PLANT SPECIFIC MC GUIRE TS REVIEW OF 05/14/90

Question 11b. TS Item 3.5

The reviewer's comments on question 11c also apply to this item, as follows:

Resolution

The following comments are made:

With respect to para. 1.:

The Question was a statement from the FSAR Evaluating a LOCA in Mode 4, (4 hrs. after reactor trip), and Mode 5, describing the necessary features of the evaluation and the Event and the resulting parameters necessary to protect the core against uncovery, and describing and referencing the equipment and procedures necessary to ensure acceptable protection

The licensee has a Licensing Basis requirement to protect against a LOCA in these Modes, and this was not manifest in the Mc Guire TS's or in the Response to this Question

This issue remains a Licensing Basis requirement for the Licensee, even though it is to be treated Generically.

Action : Even though this item is to be treated generically it is important to recognize that the licensee does have a legal commitment specifically deriving from his Licensing basis to provide the protections described and this should not be diluted or implicitly withdrawn as a result of Generic actions to which other licensee's may argue a Backfit situation . Consegently this should be addressed in the reponse to the licensee as an Outstanding Issue.

The second para. should be replaced by:
LOCA'S in lower Modes of operation, and loss of RHR cooling in lower Modes, will be addressed generically in Questions 5b, 8c, and 10, and to this should be added review of "TS page 3/4.4-6 , Reactor Coolant System - Cold Shut Down, Loops Are Not Filled" from the writer's original review , Reference 30, which is the Mid-Loop event of Mode 5 which unfortunately was omitted from selection during the original review of Reference 30. The writer also draws attention to the fact that again from his initial review to Reference 30, loss of RHR in the Refueling Mode is discussed under TS Item 3/4.9.8 Residual Heat Removal and Coolant Circulation: High Water Level, and TS Page 3/4 9-11 Refueling Operations Low Water Level - which is Mid-Loop Cooling in Mode 6. Because of

their interdependence and importance. all these items should now be evaluated, and necessarily together.

Question 11c
TS 3.5

The question is not clearly stated.

Resolution

This "question" is largely a quotation from the FSAR. The last two paragraphs are statement introducing a quotation from the SER. This question requires no response.

LOCAs in lower modes of operation and loss of RHR cooling in lower modes will be addressed generically in Question 5b.

COMMENTS BY R. LICCIARDO ON K. DESAI RESOLUTION
OF PLANT SPECIFIC MC GUIRE TS REVIEW OF 05/14/90

Question 11c, TS 3.5

Resolution

The following comments are made:

With respect to para. 1.:

The Question was a statement from the FSAR Evaluating a LOCA in Mode 4, (4 hrs. after reactor trip), and Mode 5, describing the necessary features of the evaluation and the Event and the resulting parameters necessary to protect the core against uncovery, and describing and referencing the equipment and procedures necessary to ensure acceptable protection

The licensee has a Licensing Basis requirement to protect against a LOCA in these Modes, and this was not manifest in the Mc Guire TS's or in the Response to this Question

This issue remains a Licensing Basis requirement for the Licensee, even though it is to be treated Generically.

Action : Even though this item is to be treatod generically it is important to recognize that the licensee does have a legal commitment specifically deriving from his Licensing basis to provide the protections described and this should not be diluted or implicitly withdrawn as a result of Generic actions to which other licensee's may argue a Backfit situation . Conseqently this should be addressed in the reponse to the licensee as an Outstanding Issue.

The second para. should be replaced by:
LOCA'S in lower Modes of operation, and loss of RHR cooling in lower Modes, will be addressed generically in Questions 5b, 8c, and 10, and to this should be added review of "TS page 3/4.4-6 , Reactor Coolant System - Cold Shut Down, Loops Are Not Filled" from the writer's original review . Reference 30, which is the Mid-Loop event of Mode 5 which unfortunately was omitted from selection during the original review of Reference 30. The writer also draws attention to the fact that again from his initial review to Reference 30, loss of RHR in the Refueling Mode is discussed under TS Item 3/4.9.8 Residual Heat Removal and Coolant Circulation: High Water Level, and TS Page 3/4 9-11 Refueling Operations Low Water Level - which is Mid-Loop Cooling in Mode 6. Because of their interdependence and importance, all these items should now be evaluated, and necessarily together.

Question 12a
TS 3.5.1.1.d

Explain why FSAR value for nitrogen cover-pressure of cold leg accumlators should not be of higher value to account for channel error and drift consideration.

Issue

FSAR safety analysis value is 400 psig for nitrogen cover-pressure of cold leg accumulators. TS setpoint value is also 400 psig. How do we account for channel error and drift consideration?

Resolution

Since the UHI system is removed, the licensee revised the value for nitrogen cover-pressure of cold leg accumlator to 585 psig in comparison to 400 psig with UHI accumlator. The alarm is set at 590 psig to account for channel error and drift consideration.

In the near future, the licensee will consider the channel error and drift values in the safety analysis when they revise the LOCA analyses to meet the SG tubes plugging requirement. The safety analysis value will be 564 psig and the TS value will remain the same, 585 psig.

COMMENTS BY R. LICCIARDO ON K. DESAI RESOLUTION
OF PLANT SPECIFIC MC GUIRE TS REVIEW OF 05/14/90

Question 12a. TS 3.5.1.1.d

Question

This should read as:
Explain why TS values for nitrogen cover-pressure of cold leg accumulators should not be of higher value than FSAR values quoted for normal operation

Issue

Should read as:
FSAR safety analysis shows values of 400-427 psig for nitrogen cover-pressure of cold leg accumulators. TS setpoint values are 400-450 psig. How do we account for measuring channel error and drift considerations . A positive correction to the process safety limits would normally have been necessary to allow for these considerations.

Resolution

The following comments should be added:

The current FSAR, Table 6.3.2-1, 12/88, page 1 of 5 shows minimum operating pressure of 585 psig: With an alarm now set at 590 psig, cumulative errors of >1% could result in non-conservative pressures of <585 psig inside the Accumulator.

Action: The licensee is required to revise his TS's now to conform to current analyses with qualified values of allowances for error and drift.

Question 12b
TS 4.5.1.1.1.d.1

Verify that the accumulators relief valves setpoints are included in the Inservice Testing program.

Resolution

The cold leg accumulators relief valves are not required to perform a safety function either to shutdown the reactor or to mitigate the consequences of an accident. Therefore, these valves are not included in the IST program. However, these valves are included in the licensee's preventive maintenance program at this time.

COMMENTS BY R. LICCIARDO ON K. DESAI RESOLUTION
OF PLANT SPECIFIC MC GUIRE TS REVIEW OF 05/14/90

Question 12b,TS 4.5.1.1.1.d.1

Resolution

The following comments are made on the licensee's proposal:

The Cold Leg Accumulators are Nuclear Class 2 Vessels normally isolated from the RCS by two check valves in series. These accummulators are part of the licensing basis Protective Equipment (ECCS) to mitigate the consequences of a LOCA . They are a safety related device.

The relief valves protect against loss of this protective capability by protecting against the rupture of the Accummulators from overpressurization . Therefore they come under the umbrella surveillance of the IST program as defined in the licensee's response, and must thereby be included therein.

Note : Loss of accummulators at any time represents a loss of licensing basis protective capability

Action: The licensee is required to restore relief valves on the cold leg accummulators to the IST program.

Question 13
TS 3.5.1.2.d

Verify the water temperature value used in the safety analysis for UHI accumulator.

Verify that the accumulator relief valve setpoint is included in the Inservice Testing Program.

Issue

(1) Should the accumulator water temperature value be in the technical specification?

(2) Should the accumulator relief valve setpoint be in the IST program.

Resolution

(1) The safety analysis value related to UHI accumulator water temperature is assumed to be the upper bound value of 100°F. Since the UHI accumulator is not heated or located inside containment, there is no real mechanism for increasing temperatures during operation. Therefore, there is no need for TS or UHI accumulator water temperature.

(2) The UHI accumulator relief valve is not required to perform a safety function either to shutdown the reactor or to mitigate the consequences of an accident. Therefore, it is not in the IST program.

McGuire Units 1/2 are ice condenser plants with Upper Head Injection system. Experience has demonstrated that the UHI system adds to the complexity of plant operation, requires additional maintenance and generally reduces plant availability. The TS Amendment 57 (Unit 1) and 38 (Unit 2) approved the removal of the UHI system for McGuire Units 1/2.

COMMENTS BY R. LICCIARDO ON K. DESAI RESOLUTION OF PLANT SPECIFIC MC GUIRE TS REVIEW OF 05/14/90

Question 13,TS 3.5.1.2.d

Resolution

The following comments are made on the licensee's proposal:

Concerning resolution (1): Their is much experience with Intersystem leakages as precursors to LOCA's or Feed Water Line Ruptures, causing elevated temperatures . This response is not acceptable.

Concerning resolution (2) first para.: Reference our comments on Question 12b. This response is not acceptable

Concerning resolution (2) second para.: Considering the unacceptability of the above proposals , the arguments of this para. are also generally unacceptable. The licensee has provided no information on the time schedule for removal of the UHI system and therefore we have no bases for evaluating the acceptablilty of his proposal not to implement the necessary TS's at this time . Under these circumstances we must find his proposal not to implement, Unacceptable.

Action:(1) The licensee is required to provide TS's for the UHI accumulator water temperature. (2) The licensee is required to restore inspection under the IST program to the UHI accumulator relief valves.

Question 14
TS 4.5.2.h

Verify the bases for the flow distributions in the ECCS system and how they meet minimum flow conditions to intact loops during accident occurrences.

Resolution

The ECCS flows assumed in the LOCA analyses are the bases for the limits as specified in TS 4.5.2.h.

Flow balance tests are performed during shutdown to account for any change in the subsystem flow characteristics to ensure adequate flow for ECCS consideration. ECCS flow injected to the broken cold leg is assumed to spill in LOCA analyses. The flow balance tests will place limits on the branch lines to ensure that total designated flow reaches the intact loops.

COMMENTS BY R. LICCIARDO ON K. DESAI RESOLUTION
OF PLANT SPECIFIC MC GUIRE TS REVIEW OF 05/14/90

Question 14. TS 4.5.2.h

Action: The licensee is required to place this information on the necessary distribution of ECCS flows inside the FSAR, as being a necessary set of Safety Analysis Limits

Question 17
TS 3/4.7.5

FSAR page 9.2-13, states that "In the event of solid layer of ice" forms on the Standby Nuclear Service Water Pond (SNSWP), the operating train is manually aligned to SNSWP. Provide safety-related reason for this action.

Resolution

McGuire Units 1/2 have two sources for ultimate heat sink, the primary source is a lake and the backup source is a pond. In the case of severe, prolonged cold weather, the operating train could be aligned manually from the control room to desolve the ice layer on the top of the pond. In ten years of operation, the licensee never experienced this kind of situation or any operating problems. Therefore, the licensee deleted this action and description from the FSAR and does not require any TS surveillance for this system.

COMMENTS BY R. LICCIARDO ON K. DESAI RESOLUTION
OF PLANT SPECIFIC MC GUIRE TS REVIEW OF 05/14/90

Question 17, TS 3/4.7.5

Question: Should be replaced by :

FSAR page 9.2-13, states that " In the event a solid layer of ice " forms on the Standby nuclear Service Water pond (SNSWP) , the operating train of the Nuclear Service Water System (NSWS) is manually aligned to SNSWP. Provide safety -related reason for this action, and thereby provide a related TS under TS Section 3/4.7.4 NSWS or TS Section 3/4.7.5 SNSWP.

Resolution: The following para.'s should be added:

The licensee's statement on removal of this item from the FSAR is incorrect : Ref. FSAR , Section 9.2.2.2, page 9.2-14, 1985 Update.

The purpose of this requirement is obviously to ensure that Regulatory Requirements for Redundant 100% Ultimate Heat Sinks are always available and that the safety function of this particular requirement is to protect against potential inoperability of the Pond resulting from Icing conditions, in accordance with 10.CFR. 50 GDC 2 - Design bases for protection against natural phenomena .

The very limited information provided by the licensee as a Bases for this change is not Acceptable. The licensee shall continue to ensure the availability and operability of the related system/s to perform this safety related function and incorporate the requirement for this action into the technical specifications under either TS Section 3/4.7.4 NSWS or TS Section 3/4.7.5 SNSWP.

Action: The licensee shall Amend the TS's to include this requirement under either TS Section 3/4.7.4 NSWS or TS Section 3/4.7.5. SNSWP. No change to the FSAR is necessary as the commitment remains in the document.

Question 18
TS 3/4.9.1

Why TS are not applied to flow control valves INV-171 A and INV-175 A?

<u>Resolution</u>

Surveillance Requirement 4.9.1.3 requires that valve #INV-250 shall be verified locked closed under administrative controls at least once per 72 hours during refueling operation. This valve is upstream of valves INV-171 A and INV-175 A and isolates the flow path to prevent the inadvertent dilution of the RCS boron concentration. Therefore, INV-171 A and INV-175 A are not part of TS.

COMMENTS BY R. LICCIARDO ON K. DESAI RESOLUTION
OF PLANT SPECIFIC MC GUIRE TS REVIEW OF 05/14/90

Question 18. TS 3/4.9.1

Resolution: The following comments are made and should be added to the resolution.

The TS Surveillance requirement 4.9.1.3. for valve No #INV 250 has not been changed to read "locked " closed. and should be: Administrative controls do not ensure the necessary protection against return to reactivity.

The flow control valves INV-171A and INV-175A should also be verified closed under TS's to provide effective isolation against single failure in the event #INV-250 fails open. The safety basis for this is the related diversity of protection to the source range neutron monitor system, in this Mode . This derives from the fact their are a number of additional protections which were provided within the FSAR and which the licensee has not included in the TS's and which have been excluded from incorporation by prior NRC review. It is thereby necessary to compensate for the loss of these additional diverse protective actions by ensuring this alternate fully safety related protective measure

Action: The licensee shall modify the language of his TS to require locking of the valve #INV 250 . and to verify closure of the valves INV-171A and INV-175A.

REFERENCES (For K. Desai Review)

1. Letter from Robert Licciardo to Brian Sheron, "Review of McGuire Technical Specifications," dated June 11, 1984.

2. Letter from Thomas Novak to H. B. Tucker, "Request for Comments on McGuire Technical Specifications Concerns Resulting from Differing Professional Opinion," dated July 9, 1985.

3. Letter from H. Thompson to R. Bernero, "Disposition of Concerns Raised by R. Licciardo in his DPO on the McGuire Technical Specifications," dated May 1985.

4. Letter from H. B. Tucker to Harold Denton, "NRC DPO Concerns on McGuire Technical Specifications," dated June 10, 1986.

5. Memorandum from Thomas Murley to Robert Licciardo, "December 7, 1983 Differing Professional Opinion," dated December 29, 1989.

6. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," dated March 1977.

7. NUREG-0964, "Technical Specifications McGuire Nuclear Station Unit Nos. 1 and 2," dated March 1983.

8. Letter from William Parker to Harold Denton, "Westinghouse Reactor Protection System/Engineered Safety Features Actuation System Setpoint Methodology, Duke Power Company, McGuire Unit 1," dated October 1981.

9. Duke Power Company, McGuire Nuclear Station Final Safety Analysis Report - Volumes 5, 6, 7, 9, 10 and 12.

10. ANS-56.2, "Containment Isolation Provisions for Fluid Systems," 1976.

11. Generic Letter 85-05, "Inadvertent Boron Dilution Events," January 85.

12. Letter from George Lear to D. C. Switzer, "Millstone Nuclear Power Station Units 1 and 2," dated June 1977.

LIST OF REFERENCES FOR COMMENTS BY R. LICCIARDO

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.

9. Deleted

10. 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, on Docket Nos. 50-369 and 50-370, March 1, 1978.

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.

13. 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. 3, on Docket Nos. 50-369 and 50-370, May 1980.

14. 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. 4, on Docket Nos. 50-369 and 50-370, January 1981.

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