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

December 29, 1989

MEMORANDUM FOR: Robert Licciardo, Plant Systems Branch
Division of Systems Technology
Office of Nuclear Reactor Regulation

FROM: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

SUBJECT: DECEMBER 7, 1983 DIFFERING PROFESSIONAL OPINION

In follouwp to our December 15, 1989 discussions, I asked Jim Sniezek to determine the review status of the issues you raised and to provide me the schedule for completion of the reviews. The following actions are being taken:

1.	January 1990	PD II-3 - issue Sholly notice on McGuire TS changes submitted on September 15, 1989 (Dave Matthews).
2.	February 1990	PD II-3 - issue TS amendment on McGuire TS (Dave Matthews).
3.	April 1990	SRXB - complete evaluation of licensee's response to other (non-amendment) McGuire specific concerns and document findings. If appropriate, prepare correspondence to McGuire for additional TS amendments (Bob Jones).
4.	June 1990	TSB - complete evaluation of generic TS issue, compare against Commission Policy Statement and include in revised STS, as appropriate (Rich Emch).
5.	July 1990	PD II-3 - prepare consolidated report of DPO re[illegible] and results (Dave Matthews).

The Director, Division of Systems Technology has been assigned overall coordination responsibility for the foregoing actions.

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

cc: A. Thadani
F. Gillespie
D. Matthews
R. Jones
R. Emch

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

January 26, 1990

MEMORANDUM FOR: Charles E. Rossi, Director
Division of Operational Events
Assessment, NRR

Steven A. Varga, Director
Division of Reactor Projects - I/II, NRR

FROM: Ashok C. Thadani, Director
Division of Systems Technology, NRR

SUBJECT: ASSIGNMENT AND SCHEDULES FOR RESOLUTION OF MCGUIRE DPO TECHNICAL SPECIFICATIONS (TACs 55435, 55436 AND 67757)

On December 18, 1989, DOEA, DST, and PD2-3 met to review the status and schedules for resolution of concerns expressed by R. Licciardo after his Differing Professional Opinion (DPO) on the 1984 "Proof and Review" version of the proposed McGuire Technical Specifications (TSs). Staff review of these concerns is proceeding in accordance with assignments in H. Thompson's memorandum of May 28, 1985, which identified 220 of 380 original items for action and divided these 220 into three groups: (1) generic, (2) plant specific, and (3) closed.

Generic

DOEA/OTSB reported that of the 220 items, the review of those designated as both open and generic in the May 28, 1985 memo is continuing. Additionally, items indicated to be generic by Duke's response of June 10, 1986, are included in the OTSB review. These open generic items are listed by Enclosure 1. Any of these items found by OTSB to satisfy criteria established under the TS Improvement Program for inclusion in TSs will be incorporated into draft STSs for review by NUMARC and the Owners Groups. Technical support for review of these items is being provided by SRXB and others as requested by OTSB. Completion of this effort and issuance of the new STS is presently scheduled for June 1990.

Plant Specific

Duke's reply of June 10, 1986, indicates that five of the plant-specific items have potential impact on the McGuire TSs. The PM reported that amendments to change the TSs for the five are in process with issuance expected February 10, 1990. Duke also replied that three of the plant-specific items have potential impact on the FSAR. Duke's next annual FSAR update will reflect changes for the three. These eight items are identified by Enclosure 2.

The remaining plant-specific responses by Duke are being reviewed by SRXB. Any item determined by SRXB to warrant plant-specific or generic change will be referred to the PM or OTSB for appropriate action.

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Enclosure 1

OTSB's Open Generic Items from H. Thompson's May 28, 1985 Memo

I. Category A**

	Page	Concern No.	Applicable Portion	Source*
1.	6	9	all except item d	1
2.	8, 9	10	item 1-5	1
3.	13, 14	14	item 3.b.3	1
4.	18, 19	15	item 11	1
5.	24-26	18	all	2
6.	26, 27	19	all	1
7.	42, 43	29	all	2
8.	43	30	all	1
9.	44	31	all	1
10.	45, 46	32	all	1
11.	46, 47	33	all	1
12.	49	35	all	2
13.	50, 51	36	3rd Evaluation/Disposition	2
14.	52	38	item 18b	1

II. Category B**

	Page	Concern No.	Applicable Portion	Source*
15.	4, 5	3	cii & ciii	1
16.	8, 9	10	2nd Evaluation/Disposition	1
17.	13	12	item 8	2
18.	13, 14	12	item 11	1
19.	15-18	15	G.1-G.2.5	1
20.	18-24	15	G.2.6-G.3	2
21.	28	20	1st Evaluation/Disposition	1
22.	29, 30	21	all	1

* 1 = Generic item as designated by H. Thompson's 5/28/85 memo to R. Bernero.
2 = Generic item as designated in H. Tucker's (Duke) letter of 6/10/86.

**Category A or Category B refers to an earlier (8/30/84) classification. It has no meaning here but it is needed for reference purposes because the Thompson May 28, 1985 memo references the items accordingly.

Enclosure 2

PM's Plant-Specific Items from H. Thompson's May 28, 1985 Memo*

I. Category A

	Page	Concern No.	Applicable Portion	Action	Question No.**
1.	15	15	Item 4.d	T.S. change	6a
2.	20	16	Item 2.e	T.S. change	7d
3.	21	16	Item 3.e	T.S. change	7i
4.	21	16	Item 4.e	T.S. change	7k
5.	22	22	Item 6.b	T.S. change	7n
6.	11	13	Item 9	FSAR Update	4a
7.	11	13	Item 10	FSAR Update	4b
8.	12	13	Item 17	FSAR Update	4c

II. Category B: None

* Items designated plant-specific by H. Thompson's 5/28/85 memo to R. Bernero and responded to by H. Tucker's (Duke) letter of 6/10/86.

** As designated in T. Novak's letter to H. Tucker (Duke) of 7/9/85 and responded to 6/10/86.

Enclosure 3

Closed Generic Issues

The following issues are considered closed in accordance with H. Thompson's May 28, 1985 memo. Although subsequently reclassified as generic and open by a March 15, 1989 memo from R. Licciardo, no documented basis exists for this change, and they are deemed closed based upon their 1985 disposition.

I. Category A

	Page	Concern No.	Applicable Portion
1.	4, 5	5	all
2.	6	9	Item d
3.	9	11	all
4.	9, 10	12	all
5.	10-12	13	Items 5, 6, 8, 11&17
6.	14	14	Items 4d, 7e&7g
7.	18	15	Item 10b
8.	23	17	all
9.	27	20	all
10.	27-29	21	all
11.	29, 30	22	a, b&d
12.	36, 37	24	Applicability Mode
13.	40, 41	25	TS 3/4.5.1.b (Proposed)
14.	42	28	all
15.	48, 49	34	Last Evaluation/Disposition
16.	50	36	2nd Evaluation/Disposition
17.	51	37	all
18.	53	39	all
II.	Category B		
19.	8	9	all
20.	10	12	Item 3c
21.	14	12	Table 3.3-3 Notation

SRXB reported that review of the plant-specific items will be completed by April 1990.

Closed

Several issues for which the disposition is "closed" in the H. Thompson May 28, 1985, memo were subsequently reclassified as generic and open by a March 15, 1989 memo from R. Licciardo to E. Butcher. These issues are listed in Enclosure 3. The NRC staff finds no documentation or other basis for this change. Accordingly, the staff considers these issues closed based upon their 1985 disposition and no further action is planned.

/s/

Ashok C. Thadani, Director
Division of Systems Technology
Office of Nuclear Reactor Regulation

Enclosures: As stated

DISTRIBUTION
Docket File
NRC PDR
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PDII-3 Reading

DST RDG	8E2	OGC (for info. only)	15-B-18
AThadani	8E2	E. Jordan	MNBB-3302
S. Varga	14-E-4	ACRS (10)	P-315
G. Lainas	14-H-3	[McGuire Plant File]	
D. Matthews	14-H-25		
R. Ingram	14-H-25		
D. Hood	14-H-25		
J. Calvo	11-F-23		
R. Licciardo	8-D-1		
H. Smith	12-H-5		
B. Jones	8-E-23		

*SEE PREVIOUS CONCURRENCES

OFC	:*LA:PDII-3	:*PM:PDII-3	:*NRR:OTSB	:*NRR:SRXB	:D:PDII-3	:AD:DRP	:
NAME	:RIngram	:DHood:cb	:JCalvo	:BJones	:DMatthews	:GLainas	:
DATE	:12/29/89	:12/29/89	:12/29/89	:1/2/90	:1/23/90	: 1/24/90	:

OFC	:D:DRP	: D:DST	:	:	:	:	:
NAME	:SVarga	: AThadani	:	:	:	:	:
DATE	: 1/24/90	: 1/25/90	:	:	:	:	:

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DST Document Name: Memo to C Rossi

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

May 14, 1990

MEMORANDUM FOR: Steven A. Varga, Director
Division of Reactor Projects - I/II

FROM: Ashok C. Thadani, Director
Division of Systems Technology

SUBJECT: RESOLUTION OF PLANT-SPECIFIC DPO ISSUES CONCERNING MCGUIRE TECHNICAL SPECIFICATIONS

Dr. Thomas Murley's memorandum dated December 29, 1989, identified the scope of work to resolve the differing of professional opinion (DPO) issues concerning McGuire Technical Specifications. The Reactor Systems Branch was assigned the responsibility to resolve all the plant-specific DPO issues by April 1990. The Technical Specifications Branch (OTSB) will complete the evaluation of all DPO generic issues by June 1990. PD II-3 will issue the final consolidated report by July 1990. The Director, DST, will coordinate the overall foregoing actions.

Plant-Specific DPO Issue - SRXB

The licensee provided their response to the plant-specific DPO issues in their submittal dated June 10, 1986 (Ref. 4). The licensee responded to 51 DPO issues in their submittal. Out of 51 issues, the licensee concluded that 41 issues are plant-specific and 10 issues are generic in nature.

In performing our review of the plant specific issues, we have discussed them with Robert Licciardo, NRR reviewers of various branches, (SRXB, SPLB, SICB, SELB, EMEB, and PRPB) and the licensee. For the most part, the issues involved inconsistencies between the FSAR safety analysis values, technical specifications values and the setpoint methodology report values. Resolution of these 41 issues involved disposition in one of the following categories:

(1) Plant-specific issues resolved by Technical Specification Amendment as listed in Table-1.

(2) Plant-specific issues resolved by updating the FSAR as listed in Table-2.

(3) Plant-specific issues determined not to require any action by the licensee as listed in Table-3.

The 10 generic issues identified by the licensee in their submittal will be resolved by the OTSB under the Technical Specifications Improvement Program by June 1990. These issues are listed in Table-4.

Each plant-specific issue and its resolution are discussed in detail in Enclosures 1, 2, and 3. These enclosures provide the resolution of the issues

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as listed in the Tables 1, 2, and 3 respectively. This completes our efforts on the DPO plant-specific issues.

Generic DPO Issues - OTSB

Table-5 lists all generic issues including the issues identified in Table-4. Most of the issues deal with mode applicability, either extending the mode applicability to the shutdown modes (Modes 5 and 6) or applying the LCO to other modes. A few may require changes to actions taken when LCOs are not met while others may require changes to surveillance requirements or the Bases. One issue requires a new Technical Specification.

OTSB has resolved these generic issues by either incorporating as LCO, action statements, or part of Bases Section at this time. The staff dispositions may change due to the interaction with the Owners Groups under the TS improvement program. These changes will be noted as a follow-up to the DPO resolution. OTSB will provide their evaluation report by June 1990.

A C Thadani

Ashok C. Thadani, Director
Division of Systems Technology

Enclosures:
As stated

cc: See next page

cc w/enclosures:
T. Murley
F. Miraglia
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A. Thadani
G. Lainas
B. Boger
G. Holahan
C. Rossi
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D. Matthews
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R. Licciardo
L. Marsh
D. Hood
R. Giardina
T. Collins
L. Phillips
P. VanDoorn, SRI
PDR
SRXB Members

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
TABLE-3	PLANT-SPECIFIC DPO ISSUES REQUIRING NO LICENSEE ACTION
TABLE-4	DPO ISSUES CONSIDERED AS GENERIC ISSUES TO BE RESOLVED BY THE OTSB UNDER TS IMPROVEMENT PROGRAM (LICENSEE IDENTIFIED THESE ISSUES IN THEIR SUBMITTAL DATED JUNE 1986).
TABLE-5	DPO ISSUES CONSIDERED AS GENERIC ISSUES TO BE RESOLVED BY THE OTSB UNDER TS IMPROVEMENT PROGRAM. (TABLE 5 INCLUDES ISSUES IDENTIFIED IN TABLE 4).

TABLE-1

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

PLANT-SPECIFIC DPO ISSUES RESOLVED BY TECHNICAL SPECIFICTION AMENDMENT

QUESTION*	TS	SUBJECT	TS AMENDMENT NO.	
			UNIT 1	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.	

*Questions numbers are from reference 4.

TABLE-2

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

PLANT-SPECIFIC DPO ISSUES RESOLVED BY UPDATING FSAR

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 FSAR Table 7.2.1-4, Note e.

*Questions numbers are from reference 4.

TABLE-3

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS
PLANT-SPECIFIC DPO 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	" "
1c	Table 2.2-1, Item 9	Reactor Trip-Setpoint	" "
1d	Table 2.2-1, Item 13	Reactor Trip-Setpoint	" "
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	" "
3	Table 3.3-1, Item 6c	Reactor Trip Instrumentation	" "
5a	Table 3.3-3, Item 7g	Auxiliary Feedwater Mode Applicability	" "

*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	Complete - Staff agrees with the licensee response and that no licensee action required. Enclosure 3 provides the details of resolution.
6c	Table 3.3-4, Item 9	Loss of Power-Trip Setpoint	" "
7a	Table 3.3-5, Item 2a	Safety Injection (ECCS) - Response Time	" "
7b	Table 3.3-5, Item 2b	Reactor Trip (from SI) - Response Time	" "
7c	Table 3.3-5, Item 2d	Containment Isolation - Phase "A" (2) - Response Time	" "
7e	Table 3.3-5, Item 2f	Auxiliary Feedwater - Response Time	" "
7f	Table 3.3-5, Item 3a	Safety Injection (ECCS) - Response Time	" "
7g	Table 3.3-5, Item 3b	Reactor Trip-Response Time	" "

TABLE 3 (continued)

QUESTION	TS	SUBJECT	STATUS	
7h	Table 3.3-5, Item 3d	Containment Isolation	Complete - Staff agrees with the licensee response and that no licensee action required. Enclosure 3 provides the details of resolution.	
		Phase "A" (2) - Response Time	"	"
7j	Table 3.3-5, Item 3f	Auxiliary Feedwater (5) - Response Time	"	"
7m	Table 3.3-5, Item 5a	Containment Spray - Response Time	"	"
7o	Table 3.3-5, Item 12	Automatic Switchover to Recirculation-Response Time	"	"
9	TS Page 3/4 4-2 (TS 3.4.1)	Natural Circulation Cooldown	"	"
11a	TS 3/4.5	ECCS	"	"
11b	TS 3.5	ECCS	"	"
11c	TS 3.5	ECCS	"	"

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.
12b	TS 4.5.1.1.1.1.d.1	Accumulator Relief Valve Setpoints Testing	" "
13	TS 3.5.1.2.d	Upper Head Injection Accumulator	" "
14	TS 4.5.2.h	ECCS - Subsystems	" "
17	TS 3/4.7.5	Standby Nuclear Service Water Pond	
18	TS 3/4.9.1	Boron Concentration	" "

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	"
8b	TS 3/4.4.1 G.2.6.2	Steam Line Breaks	"
8c	TS 3/4.4.1 G.2.6.3	Loss of Primary Coolant	"
8d	TS 3/4.4.1 G.2.6.4	Increase in RCS Temperature	"
8e	TS 3.4.1	RCS Loops	"
10	TS Page 3/4 4-3	RCS - Hot Shutdown	"
16	TS 3.7.1.2.6	Auxiliary Feedwater Operability	"
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	---
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	Covered by proposed STS	—
30A	3/4.7	MSIV's operabilit,	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.	Shutdown
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	—
36A (Quest. 20)	3/4.9	Refueling operations - low water level	Under review	Shutdown
38A	Table 2.2-1	RTS setpoints - low power reactor tr	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 31A	Shutdown
21B	3/4.9	Refueling operations-low water level	Under review	Shutdown

RESOLUTION OF PLANT-SPECIFIC

DPO ISSUES CONCERNING

MCGUIRE TECHNICAL SPECIFICATIONS

by

Kulin Desai

Reactor Systems Branch
Division of Systems Technology

APRIL 1990

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
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.

<u>Issue</u>

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.

<u>Resolution</u>

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 [illegible] (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.

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.

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.

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 safet 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.

Question 1a
Table 2.2-1
Item 3

Verify that a time constant of $\geq$ 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 $\geq$ 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.

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.

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%.

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.

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.

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.

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.

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.

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.

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

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.

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.

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.

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.

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 $\leq$ 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 ≤ 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.

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.

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 closeo.

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.

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.

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.

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.

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.

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.

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.

Question 18

TS 3/4.9.1

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

Resolution

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.

REFERENCES

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.

3/23/90

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

° Licensee Response (Duke Power) dated June 10, 1986

Plant-Specific Issues: 41

Generic Issues: 20

° Process Used to Review DPO Issues

- Discussion with
  - ° Robert Licciardo
  - ° NRR Reviewers - SRXB, SPLB, SICB, SELB AND EMEB
- Meeting with Licensee Feb. 26/27
  - ° Review of FSAR, Technical Specification and Setpoint Methodology Report

° Resolved Plant-Specific Issues into Four Categories

TS Amendment:	9 Issues Closed
FSAR Amendment:	6 Issues Closed
No Licensee Action Required:	23 Issues Closed, 3 Still Open

We will complete our evaluation of the plant specific actions by April 1990.

° Generic Issues

Approximately 60% of the generic issues have been dispositioned at this time. The remaining issues will be dispositioned by June 1990.

° Conclusion

Plant-Specific Issues - No safety concern

Generic Issues - Issues have merits and need to be studied

3/23/90

STATUS OF DPO ON MCGUIRE TECHNICAL SPECIFICATION

March 23, 1990

Plant-Specific TS Issues

SRXB had a working level meeting with the licensee in their office on Feb 26/27 to discuss the DPO issues. The licensee was co-operative and helpful providing adequate information. Our meeting was constructive.

The licensee responded to all 51 DPO issues in their submittal dated June 10, 1986. Out of 51 DPO issues, 41 issues are plant-specific and 10 issues are of generic in nature. To date, 38 plant specific issues have been resolved to SRXB satisfaction. More information from the licensee is needed to complete the review of the remaining 3 issues.

SRXB has divided these plant-specific issues into the four categories:

(1) Plant-specific DPO issues resolved by the TS amendment as listed in Table-1.

(2) Plant-specific DPO issues resolved by updating the FSAR as listed in Table-2.

(3) Plant-specific DPO issues do not require any action and the staff agrees with the licensee response as per Table-3.

(4) DPO issues considered as generic issues and to be resolved by the OTSB under TS improvement program as per Table-4.

Plant-specific issues			status
TS Amendment	:	9	complete
FSAR Update	:	6	complete
No Action Required (Staff agrees with the licensee response)	:	23	complete
Open issues (Awaiting more information)	:	3	open
		41	

Generic TS Issues

OTSB reviewed all generic issues and made engineering judgement to determine which issues will be addressed in upgraded TS and, which are not (See Table-5).

Approximately, 60% of the generic issues have been resolved by either determining the TS was correct, incorporating as LCO, Action statements or part of Bases Section at this time. The remaining issues will be dispositioned by June 1990. We consider this as resolution of the DPO and will inform Mr. Licciardo at that time.

It should be noted that actual implementation may change due to the negotiation with the Owners Groups. Implementation will be completed as part of the scheduled TS upgrade program.

If any changes occur as a result of the negotiations, we will inform Mr. Licciardo.

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

TABLE-1

PLANT-SPECIFIC DPO ISSUES RESOLVED BY TECHNICAL SPECIFICTION AMENDMENT

QUESTION	TS	SUBJECT	TS AMENDMENT NO. UNIT 1	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
12a	TS 3.5.1.1.d	Cold Leg Injection Accumulation Nitrogen Cover-pressure	57	38
13	TS 3.5.1.2.d	Upper Head Injection Accumulator	57	38
18	TS 3/4.9.1	Boron Concentration	105	87

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

TABLE-2

PLANT-SPECIFIC DPO ISSUES RESOLVED BY UPDATING FSAR

QUESTION	TS	SUBJECT	UPDATE REFERENCE
1c	Table 2.2-1, Item 9	Reactor Trip-Setpoint	FSAR Table 15.0.6-1
1d	Table 2.2.-1, Item 13	Reactor Trip-Setpoint	Licensee performed a new analysis and would update the FSAR
4a/4b	Table 3.3-2, Item 9/10	Reactor Trip-Response Time	Licensee response dated June 10, 1986 made a commitment to update the FSAR
4c	Table 3.3-2, Item 17	Reactor Trip-Response Time	Licensee response dated June 10, 1986 made a commitment to update the FSAR
7g	Table 3.3-5, Item 3b	Reactor Trip-Response Time	FSAR Page 7.2-15

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

TABLE-3

STATUS OF PLANT-SPECIFIC DPO ISSUES REVIEWED BY SRXB

QUESTION	TS	SUBJECT	STATUS
1	Table 2.2-1	Steam Generator-Setpoint	Complete - Staff agrees with the licensee clarification and that no change needed
1a	Table 2.2-1, Item 3	Reactor Trip-Setpoint	" "
1b	Table 2.2-1, Item 4	Reactor Trip-Setpoint	" "
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	" "
3	TAble 3.3-1, Item 6c	Reactor Trip Instrumentation	" "
5a	Table 3.3-3, Item 7g	Auxiliary Feedwater Mode Applicability	" "
6b	Table 3.3-4, Items 7c (1) and (2)	Auxiliary Feedwater-Trip Setpoints	" "
7a	Table 3.3-5, Item 2a	Safety Injection (ECCS) - Response Time	" "
7b	Table 3.3-5, Item 2b	Reactor Trip (from SI) - Response Time	" "

TABLE-3 (continued)

QUESTION	TS	SUBJECT	STATUS
7c	Table 3.3-5, Item 2d	Containment Isolation - Phase "A" (2) - Response Time	Complete - staff agrees with the licensee clarification and that no change needed
7e	Table 3.3-5, Item 2f	Auxiliary Feedwater - Response Time	" "
7f	Table 3.3-5, Item 3a	Safety Injection (ECCS) - Response Time	" "
7h	Table 3.3-5, Item 3d	Containment Isolation - Phase "A" (2) - Response Time	" "
7j	Table 3.3-5, Item 3f	Auxiliary Feedwater (5) - Response Time	" "
7m	Table 3.3-5, Item 5a	Containment Spray - Response Time	" "
7o	Table 3.3-5, Item 12	Automatic Switchover to Recirculation-Response Time	" "
9	TS Page 3/4 4-2 (TS 3.4.1)	Natural Circulation Cooldown	" "
11a	TS 3/4.5	ECCS	" "
11b	TS 3.5	ECCS	" "
11c	TS 3.5	ECCS	" "
14	TS 4.5.2.h	ECCS - Subsystems	" "
15	TS 3/4.5.3	ECCS - Subsystems	" "

TABLE-3 (continued)

QUESTION	TS	SUBJECT	STATUS
6c	Table 3.3-4, Item 9		Open-Under SRXB Review (Awaiting additional information from the licensee)
12b	TS 4.5.1.1.1.1.d.1	Accumulator Relief Valve Setpoints Testing	" "
17	TS 3/4.7.5	Nuclear Service Water System - Ultimate Heat Sink	" "

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

TABLE-4

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	LOCAs and loss of RHR in Lower Modes	Open
8a	TS 3/4.4.1 G.2.6.1	Rapid Reactivity Increase in Lower Modes	"
8b	TS 3/4.4.1 G.2.6.2	Steam Line Breaks	"
8c	TS 3/4.4.1 G.2.6.3	Loss of Primary Coolant	"
8d	TS 3/4.4.1 G.2.6.4	Increase in RCS Temperature	"
8e	TS 3.4.1		"
10	TS Page 3/4 4-3		"
16	TS 3.7.1.2.6	Auxiliary Feedwater Operability	"
19	TS 3/4.9.8	Refueling Operations	"
20	TS 4.9.8.2	Refueling Operations	"

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

ENCLOSURE

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

TABLE- 5

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

UNDER TS IMPROVEMENT PROGRAM

Concern	TS	Subject	Status
9A	3/4.2.5	DNB parameters	To be covered in bases
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
16A (Quest. 10)	3/4.4	RCS-hot shutdown	Under review
19A	3/4.4	Cold shutdown with loops filled	Under review
29A (Quest. 16)	3/4.7	a. AFW system operability b. AFW instrumentation	Covered by proposed STS
30A	3/4.7	MSIV's operability	Covered by proposed STS
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.
33A	3/4.7.4	SWS-operability modes 5 & 6	See 32A

Concern	TS	Subject	Status
35A (Quest. 19)	3/4.9.8	RHR-high water level	Under review
36A (Quest. 20)	3/4.9	Refueling operations - low water level	Under review
38A	Table 2.2-1	RTS setpoints - low power reactor trip	In proposed STS (NRC markup)
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	[illegible] 32A
21B	3/4.9	Refueling operations-low water level	Under review

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

June 7, 1990

Docket Nos. 50-369
and 50-370

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
P. O. Box 1007
Charlotte, North Carolina 28201-1007

Dear Mr. Tucker:

SUBJECT: STATUS OF PLANT-SPECIFIC ISSUES FROM DIFFERING PROFESSIONAL OPINION OF MCGUIRE TECHNICAL SPECIFICATIONS (TACS 55435/55436)

By letter of June 10, 1986, you responded to certain issues that are the subject of a differing professional opinion by a member of the NRC staff as a result of the 1983 review of the McGuire Technical Specifications (TSs). The enclosed memorandum indicates the currrent status and pending actions for resolution of these issues.

As noted in the enclosure, most of the plant-specific issues involving a change in documentation have now been resolved by TS amendment or FSAR annual update. The plant-specific effort will be concluded on the basis of your update to Note e of FSAR Table 7.2.1-4 and your proposed amendment to change TS 3/4.5.3 for consistency with the FSAR and with TS 3.4.9. The generic issues have been incorporated into our Technical Specification Improvement Program and should be completed later this month.

The NRC staff wishes to express its appreciation for the detailed effort provided by Mr. Jackie Lee and others of your company in support of this matter. We look forward to your timely submittals in order that we may bring the remaining issues to prompt resolution. Please advise me of your schedule to this end.

The reporting and/or recordkeeping requirements of this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

9007190081

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Original distribution on 6/13 incomplete

If you have questions regarding this matter, contact me at (301) 492-0905.

Sincerely,

/s/

Darl S. Hood, Project Manager
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

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

May 14, 1990

MEMORANDUM FOR: Steven A. Varga, Director
Division of Reactor Projects - I/II

FROM: Ashok C. Thadani, Director
Division of Systems Technology

SUBJECT: RESOLUTION OF PLANT-SPECIFIC DPO ISSUES CONCERNING MCGUIRE TECHNICAL SPECIFICATIONS

Dr. Thomas Murley's memorandum dated December 29, 1989, identified the scope of work to resolve the differing of professional opinion (DPO) issues concerning McGuire Technical Specifications. The Reactor Systems Branch was assigned the responsibility to resolve all the plant-specific DPO issues by April 1990. The Technical Specifications Branch (OTSB) will complete the evaluation of all DPO generic issues by June 1990. PD II-3 will issue the final consolidated report by July 1990. The Director, DST, will coordinate the overall foregoing actions.

Plant-Specific DPO Issue - SRXB

The licensee provided their response to the plant-specific DPO issues in their submittal dated June 10, 1986 (Ref. 4). The licensee responded to 51 DPO issues in their submittal. Out of 51 issues, the licensee concluded that 41 issues are plant-specific and 10 issues are generic in nature.

In performing our review of the plant specific issues, we have discussed them with Robert Licciardo, NRR reviewers of various branches, (SRXB, SPLB, SICB, SELB, EMEB, and PRPB) and the licensee. For the most part, the issues involved inconsistencies between the FSAR safety analysis values, technical specifications values and the setpoint methodology report values. Resolution of these 41 issues involved disposition in one of the following categories:

(1) Plant-specific issues resolved by Technical Specification Amendment as listed in Table-1.

(2) Plant-specific issues resolved by updating the FSAR as listed in Table-2.

(3) Plant-specific issues determined not to require any action by the licensee as listed in Table-3.

The 10 generic issues identified by the licensee in their submittal will be resolved by the OTSB under the Technical Specifications Improvement Program by June 1990. These issues are listed in Table-4.

Each plant-specific issue and its resolution are discussed in detail in Enclosures 1, 2, and 3. These enclosures provide the resolution of the issues

Contact: K. Desai, SRXB, x21058

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as listed in the Tables 1, 2, and 3 respectively. This completes our efforts on the DPO plant-specific issues.

Generic DPO Issues - OTSB

Table-5 lists all generic issues including the issues identified in Table-4. Most of the issues deal with mode applicability, either extending the mode applicability to the shutdown modes (Modes 5 and 6) or applying the LCO to other modes. A few may require changes t^ actions taken when LCOs are not met while others may require changes to .eillance requirements or the Bases. One issue requires a new Technic. Specification.

OTSB has resolved these generic issues by either incorporating as LCO, action statements, or part of Bases Section at this time. The staff dispositions may change due to the interaction with the Owners Groups under the TS improvement program. These changes will be noted as a follow-up to the DPO resolution. OTSB will provide their evaluation report by June 1°90.

A C Thadani

Ashok C. Thadani, Director
Division of Systems Technology

Enclosures:
As stated

cc: See next page

cc w/enclosures:
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F. Miraglia
W. Russell
A. Thadani
G. Lainas
B. Boger
G. Holahan
C. Rossi
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R. Licciardo
L. Marsh
D. Hood
R. Giardina
T. Collins
L. Phillips
P. VanDoorn, SRI
PDR
SRXB Members

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
TABLE-3	PLANT-SPECIFIC DPO ISSUES REQUIRING NO LICENSEE ACTION
TABLE-4	DPO ISSUES CONSIDERED AS GENERIC ISSUES TO BE RESOLVED BY THE OTSB UNDER TS IMPROVEMENT PROGRAM (LICENSEE IDENTIFIED THESE ISSUES IN THEIR SUBMITTAL DATED JUNE 1986).
TABLE-5	DPO ISSUES CONSIDERED AS GENERIC ISSUES TO BE RESOLVED BY THE OTSB UNDER TS IMPROVEMENT PROGRAM. (TABLE 5 INCLUDES ISSUES IDENTIFIED IN TABLE 4).

TABLE-1

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.	

*Questions numbers are from reference 4.

TABLE-2

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS

PLANT-SPECIFIC DPO ISSUES RESOLVED BY UPDATING FSAR

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 FSAR Table 7.2.1-4, Note e.

*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	" "
1c	Table 2.2-1, Item 9	Reactor Trip-Setpoint	" "
1d	Table 2.2-1, Item 13	Reactor Trip-Setpoint	" "
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	" "
3	Table 3.3-1, Item 6c	Reactor Trip Instrumentation	" "
5a	Table 3.3-3, Item 7g	Auxiliary Feedwater Mode Applicability	" "

*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	Complete - Staff agrees with the licensee response and that no licensee action required. Enclosure 3 provides the details of resolution.
6c	Table 3.3-4, Item 9	Loss of Power-Trip Setpoint	" "
7a	Table 3.3-5, Item 2a	Safety Injection (ECCS) - Response Time	" "
7b	Table 3.3-5, Item 2b	Reactor Trip (from SI) - Response Time	" "
7c	Table 3.3-5, Item 2d	Containment Isolation - Phase "A" (2) - Response Time	" "
7e	Table 3.3-5, Item 2f	Auxiliary Feedwater - Response Time	" "
7f	Table 3.3-5, Item 3a	Safety Injection (ECCS) - Response Time	" "
7g	Table 3.3-5, Item 3b	Reactor Trip-Response Time	" "

TABLE 3 (continued)

QUESTION	TS	SUBJECT	STATUS
7h	Table 3.3-5, Item 3d	Containment Isolation	Complete - Staff agrees with the licensee response and that no licensee action required. Enclosure 3 provides the details of resolution.
		Phase "A" (2) - Response Time	" "
7j	Table 3.3-5, Item 3f	Auxiliary Feedwater (5) - Response Time	" "
7m	Table 3.3-5, Item 5a	Containment Spray - Response Time	" "
7o	Table 3.3-5, Item 12	Automatic Switchover to Recirculation-Response Time	" "
9	TS Page 3/4 4-2 (TS 3.4.1)	Natural Circulation Cooldown	" "
11a	TS 3/4.5	ECCS	" "
11b	TS 3.5	ECCS	" "
11c	TS 3.5	ECCS	" "

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.	
12b	TS 4.5.1.1.1.1.d.1	Accumulator Relief Valve Setpoints Testing	"	"
13	TS 3.5.1.2.d	Upper Head Injection Accumulator	"	"
14	TS 4.5.2.h	ECCS - Subsystems	"	"
17	TS 3/4.7.5	Standby Nuclear Service Water Pond		
18	TS 3/4.9.1	Boron Concentration	"	"

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	"
8b	TS 3/4.4.1 G.2.6.2	Steam Line Breaks	"
8c	TS 3/4.4.1 G.2.6.3	Loss of Primary Coolant	"
8d	TS 3/4.4.1 G.2.6.4	Increase in RCS Temperature	"
8e	TS 3.4.1	RCS Loops	"
10	TS Page 3/4 4-3	RCS - Hot Shutdown	"
16	TS 3.7.1.2.6	Auxiliary Feedwater Operability	"
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	—
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	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.	Shutdown
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	—
[illegible]A ([illegible] 20)	3/4.9	Refueling operations - low water level	Under review	Shutdown
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	Shutdown
21B	3/4.9	Refueling operations-low water level	Under review	Shutdown

RESOLUTION OF PLANT-SPECIFIC

DPO ISSUES CONCERNING

MCGUIRE TECHNICAL SPECIFICATIONS

by

Kulin Desai

Reactor Sy.tems Branch
Division of Systems Technology

APRIL 1990

9007230392

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
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.

Questi-n 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.

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.

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.

<u>Issue</u>

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.

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.

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.

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.

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.

[illegible]
also found a error while [illegible]
FSAR revision
error
will correct
in 1985
update

<u>Issue</u>

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.

<u>Resolution</u>

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 <u>W</u> 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%.

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.

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.

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.

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.

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.

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 gri ystem 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.

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

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.

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.

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.

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.

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.

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.

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.

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.

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.

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.

Being added now

OM-10

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 on 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.

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.

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.

Not in procedure will be deleted in FSAR on August

Question 18 — TS 3/4.9.1

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

Resolution

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.

REFERENCES

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.