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

SEP 4 1990

NOTE FOR: Ashok C. Thadani, Director
Division of Systems Technology

FROM: Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Technology

SUBJECT: FINAL ISSUANCE OF MCGUIRE DPO WITH REVISED RESPONSES

We have incorporated W. Russell's comments in the final McGuire DPO document. We revised our responses to the questions: 15, 1d, 2, 3, 5a, 6b, 6c, and 7b/7g as appropriate. Our responses are attached for your information.

Robert Jones

Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Technology

Enclosures:
As stated

cc w/enclosures:
D. Matthews
D. Hood
T. Reed
M. Caruso
K. Desai

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 then 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.

Clarification of R. Licciardo's comments dated June 19, 1990

The DPO reviewer raised the concern that the safety injection pump breakers should be opened, locked and tagged to be consistent with the FSAR LTOP analysis.

McGuire's LTOP analysis is based on one centrifugal charging pump mass flow. TS 3.5.3 defines the minimum number of ECCS pumps to be operable for temperature less than or equal to 350°F. Surveillance requirement, SR 4.5.3.2 specifies that all pumps, except the minimum required operable pumps (which means only one centrifugal charging pump for LTOP considerations) shall be demonstrated inoperable by verifying that the motor circuit breakers are secured in the open position or by verifying the discharge of each pump has been isolated from the RCS by at least two isolation valves (double isolation) with power removed from the valve operators at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F. Thus, there is an adequate protection provided for LTOP event.

However, there is an apparent inconsistency in the TS. The TS has a footnote that allows a maximum of one centrifugal charging pump and one SI pump to be operable whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F. This would invalidate the LTOP analysis. However, as noted in our response, plant procedures only permit the charging pump to be lined up for injection.

We have discussed this subject matter with the licensee. The licensee has committed to eliminate this inconsistency as part of their planned threshold temperature TS change of their LTOP.

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%.

Clarification to R. Licciardo's comments dated June 19, 1990

(1) The licensee's feedwater line rupture analysis assumes a SG water level-low low reactor trip setpoint of 23% narrow range span. The TS setpoint value is 40% of narrow range span. The staff has reviewed the derivation of the TS value and has concluded that the effect of instrument error and channel drift has been appropriately added to the value used in the safety analyses.

In addition, the reviewer raised a question related to the need to revise the setpoint methodology document to reflect these changes. The setpoint methodology document is a reference document which demonstrates how the instrumentation drift and other uncertainties are accounted for in setpoint determination. Unlike FSAR or TS changes, it does not require an amendment. The licensee maintains separate calculation files to support their setpoint calculations used in the TS.

(2) Feedwater line rupture and loss of normal feedwater events result in different containment environments. Instrumentation error is larger for the feedline rupture event due to the hostile environment which is created. Therefore, the SG water level-low low reactor trip setpoint for the main feedwater line rupture is lowered to 23 percent to account for the environmental effects.

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.

Clarification of R. Licciardo's comments dated June 19, 1990

The change in initial condition from 557°F to 551°F for transients occurring at hot zero power would have a negligible impact on results because the moderator temperature coefficient (MTC) is not significantly affected in this temperature band. In addition, the analyzed input condition represents the expected plant operating condition at hot zero power temperature of about 557°F.

Therefore, it is not necessary to analyze hot zero power transients at 551°F. This is a normal industry practice.

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.

Clarification of R. Licciardo's comments dated June 19, 1990

The staff reviewed the FSAR Section 15.4.6 concerning boron dilution events analyzes. We could not find the commitment made by the licensee that two source range neutron flux channels would be operable during the modes 3, 4 and 5.

The licensee complies with the staff position requiring adequate operating procedures and administrative controls to prevent boron dilution events. McGuire has both positive alarms and audible count rate meters to alert the operators to boron dilution events. Therefore, the plant complies with its licensing basis.

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.

Clarification of R. Licciardo's comments dated June 19, 1990

Auxiliary feedwater system (AFS) applicability in Mode 4 is a generic issue. Our response is provided as a resolution of Generic Issue No. 29A. (The new STS require operability of the one motor driven auxiliary feedwater system pump in Mode 4 whenever a steam generator is relied on for heat removal. This is a change over the current STS which do not require AFS operability in Mode 4).

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.

<u>Clarification of R. Licciardo's comments dated June 19, 1990</u>

Westinghouse has explicitly calculated the steam generator inventory for the case of a single failure of the motor driven pump. Actuation of turbine driven AFW pump on low-low steam generator has been demonstrated for this single failure case.

TS 3.7.1.2, auxiliary feedwater system, requires each motor-driven and steam turbine driven pump be demonstrated operable once per 31 days by verifying adequate pump flow. The flow distribution calculations are done by computerize analyses which utilize standard engineering techniques and conservative failure assumptions to minimize flows.

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.

Clarification of R. Licciardo's comments dated June 19, 1990

R. Licciardo raised the new issue and our response is as follows:

The undervoltage setpoints and the degraded voltage setpoints of the safety busses are provided to protect the equipment on the safety busses from degraded voltage conditions and to ensure availability of the safety busses following loss of offsite power. They are not designed to trip the reactor. The undervoltage trip on the reactor coolant pump provides an anticipatory trip of the reactor. The clearing of the loads from the safety busses upon undervoltage conditions, the startup of the diesel generators, and the subsequent sequencing of the safety loads onto the safety busses do not necessarily result in or require a reactor trip. Therefore, coordination between the setpoints for the undervoltage clearing of the safety busses and the setpoints for the reactor coolant pump undervoltage trip is not required nor desired. Nor is the undervoltage trip function of the reactor coolant pump compromised if there is a concurrent loss of voltage on the safety busses since functioning of the reactor trip system does not depend on the AC safety busses in the short term. This is because the reactor trip system is powered from the DC system and its associated inverters.

This is a normal industry practice and it is not a issue.

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.

Clarification of R. Licciardo's comments dated June 19, 1990

The FSAR Table uses the word "signal" to mean "setpoint" reached. TS Table 3.3-5 shows response times which is the time from the setpoint reached to full actuation of equipment. Thus, the values shown in the FSAR table are not directly related to TS Table 3.3-5.

Apparently, the DPO reviewer has concluded that the reactor trip times shown in the FSAR are based on reaching the SI actuation signals. However, these are actually based on reaching the reactor trip signal on pressurizer pressure-low. Thus, these have no relationship to TS Table 3.3-5. Therefore, no change to the TS is appropriate.

Please also see our response in Question 4c.

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

SEP 10 1990

Docket Nos. 50-369
and 50-370

MEMORANDUM FOR: Robert Licciardo, Planning, Program and Management Support Branch, PMAS
Office of Nuclear Reactor Regulation

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

SUBJECT: CLOSURE OF DPO ISSUES REGARDING MCGUIRE TECHNICAL SPECIFICATIONS (TACS 55435/55436/67757)

In accordance with my memorandum to you of December 29, 1989, the actions regarding your Differing Professional Opinion (DPO) on the McGuire Technical Specifications (TSs) have been completed or are proceeding to an established resolution plan. These actions have included:

(1) Issuance of amendments to the McGuire operating licenses changing TSs based on plant-specific issues. The DPO issues resolved in this manner are identified by Table 1 and are discussed by Enclosure 1.

(2) Changes to the McGuire FSAR by the licensee's annual updates. These are identified by Table 2 and are discussed by Enclosure 2. The licensee has stated that those changes to the FSAR identified in its letter of June 10, 1986, if not already made, are being addressed by the "1989 update" to be issued in September 1990.

(3) Reevaluations by SRXB and other technical branches for several plant-specific issues. Unlike item (1) above, these reevaluations determined that no actions need be taken by the licensee. During these reevaluations the NRC staff has had the benefit of comments by an NRC contractor, the licensee (who, in turn, reflected the results of comments by Westinghouse) and the results of previous staff reviews. DPO issues resolved by this further evaluation are identified by Table 3 and are discussed by Enclosure 3.

The staff has also had the benefit of your further comments of June 19, 1990, on A. Thadani's memorandum

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of May 14, 1990. Where appropriate, these further comments have been addressed in Enclosure 3 by adding clarifications to the prior resolutions of May 14, 1990 (dated April 1990).

(4) Evaluation by OTSB of DPOs based upon generic issues. Issues in this category were evaluated using criteria given in the Commission's Interim Policy Statement on Technical Specification Improvement to determine if they should be incorporated into the Standard Technical Specifications (STS) for the Westinghouse Owners Group (WOG). For this evaluation, OTSB had the benefit of extensive support by NRR technical branches and information from WOG. These issues are identified by Tables 4 and 5 and are discussed in Enclosure 4. The results generally indicate that the new STS have addressed several of the DPO issues, while other DPO concerns need not be added to the STS because they do not qualify under the Commission's TS criteria or for technical reasons.

The above actions have addressed all DPO issues not previously closed by R. Bernero's memorandum of August 30, 1984. Additionally, these previously closed issues (160 in all) were reviewed in light of events at Diablo Canyon and Vogtle. The review found that the original DPO issues regarding mid-loop operation have all been addressed through staff considerations and actions in response to the Diablo Canyon event, including resolution of Generic Issue 36A. With respect to the Vogtle event, none of the 160 issues included concerns regarding station blackout. Moreover, no reason to re-open any of the 160 issues was found during the review.

Accordingly, this completes NRR action on your DPO and the subject TACS are closed.

Original Signed By:

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosures:
As stated (Tables and Enclosures)

LA:PDII-3	PM:PDII-3	D:DST	D:DOEA	D:PDII-3	ADR2	D:DRP
RIngram	DHood	AThadani	ERossi	DMatthews	GLainas	SVarga
8/14/90	8/14/90	8/15/90	8/15/90	8/15/90	8/15/90	8/16/90

ADP	ADT	NRR:D
JPartlow	WRussell	TMurley
8/20/90	9/5/90	9/4/90

DISTRIBUTION Closure of McGuire DPO Issues Dated September 10, 1990

Docket File
NRC PDR
Local PDR
PDII-3 Reading
F. Miraglia
J. Partlow
S. Varga
G. Lainas
H. Smith
F. Gillespie
E. Rossi
J. Calvo
A. Thadani
R. Jones
M. Caruso
R. Emch
K. Desai
D. Matthews
D. Hood

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 RESOLVED REQUIRING NO LICENSEE ACTION
TABLE-4	DPO ISSUES CONSIDERED AS GENERIC ISSUES 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 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 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)	Complete
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 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

Complete

*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	—	Complete
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	—	
36A (Quest. 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

Complete

ENCLOSURES 1, 2, & 3

RESOLUTION OF PLANT-SPECIFIC

DPO ISSUES CONCERNING

MCGUIRE TECHNICAL SPECIFICATIONS

by

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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 RESOLVED 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 'n the definition of of the setpoint and rovide 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 #102 (Unit 1) and TS Amendment #84 (Unit 2) respectively.

Question 71
Table 3.3-5,
Item 4h

Clarify the inconsistency between the safety analysis value and the TS Value for steam line isolation response time.

Issue

FSAR feedwater system pipe break analysis sequence of events Table 15.2.3-1 indicates that the low steam line pressure setpoint is reached in the ruptured steam generator in 420 seconds, and that all main steam line isolation valves would close in 427 seconds. Based on th s 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 charge 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.

Clarification of R. Licciardo's comments dated June 19, 1990

The DPO reviewer raised the concern that the safety injection pump breakers should be opened, locked and tagged to be consistent with the FSAR LTOP analysis.

McGuire's LTOP analysis is based on one centrifugal charging pump mass flow. TS 3.5.3 defines the minimum number of ECCS pumps to be operable for temperature less than or equal to 350°F. Surveillance requirement, SR 4.5.3.2 specifies that all pumps, except the minimum required operable pumps (which means only one centrifugal charging pump for LTOP considerations) shall be demonstrated inoperable by verifying that the motor circuit breakers are secured in the open position or by verifying the discharge of each pump has been isolated from the RCS by at least two isolation valves (double isolation) with power removed from the valve operators at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F. Thus, there is an adequate protection provided for LTOP event.

However, there is an apparent inconsistency in the TS. The TS has a footnote that allows a maximum of one centrifugal charging pump and one SI pump to be operable whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F. This would invalidate the LTOP analysis. However, as noted in our response, plant procedures only permit the charging pump to be lined up for injection.

We have discussed this subject matter with the licensee. The licensee has committed to eliminate this inconsistency as part of their planned threshold temperature TS change of their LTOP.

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.

Clarification to R. Licciardo's comments dated June 19, 1990

TS Tables 3.3-2 and 3.3-5 list reactor trip response times and engineered safety features response times respectively. Response times are provided for accidents and transients as appropriate based on the trip function which is taken credit for in the safety analysis. However, the other trip functions are always available and have their surveillance requirements to demonstrate their operability. To eliminate duplicate surveillance testing requirements, trip functions response times are listed in either tables as appropriate.

This response is also applicable to Question 7b/7g and 7e comments.

ENCLOSURE 3

DPO CONCERNS ON MCGUIRE TECHNICAL SPECIFICATIONS
RESOLUTION OF PLANT-SPECIFIC DPO ISSUES RESOLVED 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 $\geq$ 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 in [illegible] 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.

Clarification to R. Licciardo's comments dated June 19, 1990

Set point methodology document is a reference document to demonstrate how the instrumentation drift and other uncertainties are accounted for in setpoint determination. Unlike FSAR or Technical Specifications changes, it does not require an amendment. Licensee keeps their files up to date with their revised calculations and can make changes in the setpoint methodology document without the staff's approval.

FSAR original analysis value of 1835 psig remains the same.

This response also applies to Question 1d.

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%.

Clarification to R. Licciardo's comments dated June 19, 1990

(1) The licensee's feedwater line rupture analysis assumes a SG water level-low low reactor trip setpoint of 23% narrow range span. The TS setpoint value is 40% of narrow range span. The staff has reviewed the derivation of the TS value and has concluded that the effect of instrument error and channel drift has been appropriately added to the value used in the safety analyses.

In addition, the reviewer raised a question related to the need to revise the setpoint methodology document to reflect these changes. The setpoint methodology document is a reference document which demonstrates how the instrumentation drift and other uncertainties are accounted for in setpoint determination. Unlike FSAR or TS changes, it does not require an amendment. The licensee maintains separate calculation files to support their setpoint calculations used in the TS.

(2) Feedwater line rupture and loss of normal feedwater events result in different containment environments. Instrumentation error is larger for the feedline rupture event due to the hostile environment which is created. Therefore, the SG water level-low low reactor trip setpoint for the main feedwater line rupture is lowered to 23 percent to account for the environmental effects.

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.

Clarification of -cciardo's comments dated June 19, 1990

The change in initial condition from 557°F to 551°F for transients occurring at hot zero powe vould have a negligible impact on results because the moderator tempe..ure coefficient (MTC) is not significantly affected in this temperature band. In addition, the analyzed input condition represents the expected plant operating condition at hot zero power temperature of about 557°F.

Therefore, it is not necessary to analyze hot zero power transients at 551°F. This is a normal industry practice.

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 channe' operable while FSAR requires two channels + .. 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.

Clarification of R. Licciardo's comments dated Jur 19, 1990

The staff reviewed the FSAR Section 15.4.6 concerning boron dilution events analyses. We could not find the commitment made by the licensee that two source range neutron flux channels would be operable during the modes 3, 4 and 5.

The licensee complies with the staff position requiring adequate operating procedures and administrative controls to prevent boron dilution events. McGuire has both positive alarms and audible count rate meters to alert the operators to boron dilution events. Therefore, the plant complies with its licensing basis.

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.

Clarification of R. Licciardo's comments dated June 19, 1990

Auxiliary feedwater system (AFS) applicability in Mode 4 is a generic issue. Our response is provided as a resolution of Generic Issue No. 29A. (The new STS require operability of the one motor driven auxiliary feedwater system pump in Mode 4 whenever a steam generator is relied on for heat removal. This is a change over the current STS which do not require AFS operability in Mode 4).

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.

<u>Clarification of R. Licciardo's comments dated June 19, 1990</u>

Westinghouse has explicitly calculated the steam generator inventory for the case of a single failure of the motor driven pump. Actuation of turbine driven AFW pump on low-low steam generator has been demonstrated for this single failure case.

TS 3.7.1.2, auxiliary feedwater system, requires each motor-driven and steam turbine driven pump be demonstrated operable once per 31 days by verifying adequate pump flow. The flow distribution calculations are done by computerize analyses which utilize standard engineering techniques and conservative failure assumptions to minimize flows.

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.

Clarification of R. Licciardo's comments dated June 19, 1990

R. Licciardo raised the new issue and our response is as follows:

The undervoltage setpoints and the degraded voltage setpoints of the safety busses are provided to protect the equipment on the safety busses from degraded voltage conditions and to ensure availability of the safety busses following loss of offsite power. They are not designed to trip the reactor. The undervoltage trip on the reactor coolant pump provides an anticipatory trip of the reactor. The clearing of the loads from the safety busses upon undervoltage conditions, the startup of the diesel generators, and the subsequent sequencing of the safety loads onto the safety busses do not necessarily result in or require a reactor trip. Therefore, coordination between the setpoints for the undervoltage clearing of the safety busses and the setpoints for the reactor coolant pump undervoltage trip is not required nor desired. Nor is the undervoltage trip function of the reactor coolant pump compromised if there is a concurrent loss of voltage on the safety busses since functioning of the reactor trip system does not depend on the AC safety busses in the short term. This is because the reactor trip system is powered from the DC system and its associated inverters.

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 P[illegible]-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

<u>Clarification of R. Licciardo's comments dated June 19, 1990</u>

Safety injection flow rate to the Reactor Coolant System as a function of the system pressure is used as part of the input in the LOCA analyses. The Safety Injection (SI) system was assumed to be delivering to the RCS 25 seconds after the low pressurizer pressure setpoint was reached. The Technical Specifications permit a delay time of 27 seconds; however, the two seconds difference is more than offset by the following three factors:

(1) These analyses assume that no safety injection flow reaches the reactor coolant system until the full 25 second delay has expired. Actually, there will be some safety injection flow during the startup of the various safety injection pumps.

(2) According to the Technical Specification requirements, the high head safety injection pumps are loaded onto the emergency buses within 13 seconds. These pumps would therefore be providing their full flow prior to 25 seconds.

(3) These analyses assume diesel generator startup upon the generation of a safety injection signal and take no credit for the start of the generator due to the loss of offsite power, which is assumed to occur concurrently with the initiation of the LOCA.

The licensee has revised the LOCA analyses and will update the FSAR by September 1991.

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.

Clarification of C. Licciardo's comments dated June 19, 1990

The FSAR Table uses the word "signal" to mean "setpoint" reached. TS Table 3.3-5 shows response times which is the time from the setpoint reached to full actuation of equipment. Thus, the values shown in the FSAR table are not directly related to TS Table 3.3-5.

Apparently, the DPO reviewer has concluded that the reactor trip times shown in the FSAR are based on reaching the SI actuation signals. However, these are actually based on reaching the reactor trip signal on pressurizer pressure-low. Thus, these have no relationship to TS Table 3.3-5. Therefore, no change to the TS is app opriate.

Please also see our response in Question 4c.

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, ..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.

Clarification of R. Licciardo's comments dated June 19, 1990

Please see our response in Question 4c.

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.

<u>Resolution</u>

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.

Clarification of R. Licciardo's comments dated June 19, 1990

The FSAR analysis documented in Table 6.3.2-3B is based on sequential operation of the sump valves (NI-184B and NI-185A) and the RWST valves (ND-4B and ND-19A). As stated in Note 10 of this table, the RWST valves (which close after the sump valves have fully opened) finish closing no later than 10 seconds into Step 5. Since this step is begun at 110 seconds, the sequential operation of each pair of valves therefore is assumed in the FSAR to require 120 seconds, reflecting the allowable stroke time of 60 seconds per valve. The valves actually stroke faster than this, allowing the 60 seconds Technical Specification to be satisfied.

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.

Clarification of R. Licciardo's comments dated June 19, 1990

Our response to Questions 11a, 11b, and 11c is available in Generic Issues resolution Item 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.

Clarification of R. Licciardo's comments dated June 19, 1990

The licensee has revised the LOCA analyses and will update the FSAR by September 1991. This new analysis value will provide about 3 percent of margin to account for drift and channel error which we find acceptable.

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.

Clarification of R. Licciardo's comments dated June 19, 1990

The cold leg accumulator relief valves will be added into the IST program at the upcoming next 10 year inservice inspection interval assuming ANSI/ASME-OM-10 standard is incorporated into the 10 CFR 50.

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.

Clarification of R. Licciardo's comments dated June 19, 1990

Upper Head Injection system is removed from the McGuire facility. The comments are no longer applicable.

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.

Clarification of R. Licciardo's comments dated June 19, 1990

We have deleted the last two sentences.

Intake structure for the pond is 40 feet below the top of the pond level which provides initial water source in the case of the ice layer on the top of the pond. Discharge water is recirculated on the top of the pond which could also melt the ice. Thus, the pond is available to satisfy all design basis events upon the loss of the lake source.

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 . 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 T and Thermal Overtemperature T 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.

ENCLOSURE 4

RESOLUTION OF GENERIC

DPO ISSUES CONCERNING

MCGUIRE TECHNICAL SPECIFICATIONS

Technical Specifications Branch

Division of Operational Events Assessment

AUGUST 3, 1990

AVAILABILITY OF RCPs DEPARTURE FROM NUCLEATE BOILING (DNB) TS

Concern 9A
Question 8e
3/4.2.5

Issue

The assertion involving Concern 9A consists of the following three parts:

I. The DPO asserts that the value for the reactor coolant system average temperature (Tavg) given in the TS Table 3.2-1 is not consistent with the value given for Tavg in FSAR Figure 5.3.3-1 for the rated power conditions. Furthermore, the DPO asserts that the following should be provided in the TS:

a.) The setpoint and allowable values of Tavg;
b.) The related power level ascribed to Tavg; and
c.) The reactor coolant system Tavg for the zero power condition.

II. The DPO asserts that the values for pressurizer pressure in TS Table 3.2-1 are not consistent with the information given in FSAR Table 15.1.2-2 and Table 4.1-1 of DPO reference 20. Also, the DPO asserts that the setpoint and allowable values of the pressurizer pressure should be provided in the TS.

III. The DPO asserts that the pressurizer pressure should be provided in TS 2.1-1 and 3/4.4.3.

Resolution - I.

The values of Tavg listed in Table 3.2-1 of the TS are Limiting Conditions of Operation (LCOs) and are derived from plant safety analyses. These limiting values are established in conjunction with limiting values for other principal thermal-hydraulic parameters to ensure sufficient DNB margin. These limits ensure that the DNB safety limit will not be violated in the event of a plant transient.

FSAR Figure 5.3.3-1 is a plot of the expected Tavg versus power level. The values of Tavg in the plot are not derived from the plant safety analyses. They are estimates of the actual values of Tavg that will exist when the plant is operated the way the licensee intends. All the plotted Tavg values are within the limits in TS Table 3.2-1.

a.) There is no instrumentation which monitors Tavg and generates a reactor trip signal based on the values in Table 3.2-1. Therefore, setpoints and allowable values corresponding to the limits in Table 3.2-1 do not need to be specified in the TS.

b.) and c.) The Tavg limits in TS Table 3.2-1 were derived by considering plant transients initiated from all power levels. Therefore, they are bounding values which are applicable at any power level. No related power level needs to be ascribed to the Tavg values in TS Table 3.2-1, and no separate Tavg limit needs to be specified in TS Table 3.2-1 for zero power operations.

Resolution - II.

The values of pressurizer pressure listed in Table 3.2-1 of the TS are LCOs and are derived from plant safety analyses. Pressurizer pressure is another principal thermal-hydraulic parameter in the calculation of DNB. These limits ensure that the DNB safety limit will not be violated in the event of a plant transient.

Since there are no automatic reactor trips actuated based on the values in TS 3.2-1, there is no need to specify setpoints or allowable values. The instrumentation that would initiate a reactor trip based on these parameters is addressed in TS 3.3.1.

The pressurizer pressure value in Table 4.1-1 of reference 20 of the DPO is the nominal design pressure for the reactor coolant system and reactor internals and is an expected value for plant operation. It is an estimate of the actual value of pressurizer pressure that will exist when the plant is operated the way the licensee intends. The nominal value is within the limits of TS Table 3.2-1.

FSAR Table 15.1.2-2 is part of the description of the plant safety analyses. These analyses include adjustments to account for steady state fluctuations and measurement error. The DPO suggests that the limits in TS Table 3.2-1 should equal the reference 20 nominal value minus the adjustment specified in the safety analyses. This suggestion is not correct. The limits in the TS are derived by making adjustments on safety analysis limiting values of the pressure - not nominal values.

Resolution - III.

In the new STS, pressurizer pressure is included in the curves in Section 2.1.1 (it is also included in the Section 2.1.1 curves of the current STS). Specification 3/4.4.3 specifies the operability of the pressurizer. The operability of the pressurizer is determined based on water volume and heater capacity; therefore, pressurizer pressure does not need to be included in TS 3/4.4.3.

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

Concern 16B
Questions 8a, 8b, 8c, 8d, and 8e
TS 3/4.4.1

Issue

The DPO asserts that four Reactor Coolant System (RCS) loops should be required to be OPERABLE in MODE 3 (Hot Standby) to meet the assumptions of the safety analysis for a number of accident scenarios. Each of Questions 8a - 8d below discusses this concern for a different type of accident.

Question 8a: OCCURRENCES WITH RAPID REACTIVITY INCREASE
Pertaining to "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from Sub-Critical Condition."

This Technical Specification (TS) at the time of the DPO submittal required that two RCS loops be OPERABLE and one RCS loop be in operation in MODE 3. The FSAR for McGuire (and other Westinghouse plants) assumes four Reactor Coolant Pumps (RCPs) are running for this event. The DPO asserts that any Technical Specification allowing operability of less than four RCS loops in MODE 3 would not be in conformance with the FSAR and is non-conservative.

Question 8b: STEAM LINE BREAKS: OCCURRENCES
Pertaining to "Major Rupture of a Main Steamline."

The McGuire FSAR states that the resulting impact on shutdown margin for this event during MODES 3, 4, and 5 is improved over that of the design basis for zero power, just critical and Tavg = 557°F. The DPO asserts, however, that the design basis case may not be the most limiting case. It states that it is conceivable that two loop operation may be less conservative than either four RCPs continuing to operate or four RCPs tripped on Safety Injection. The conclusion of the DPO is that any Technical Specification allowing operability of less than four RCS loops in MODE 3 would not be in conformance with the FSAR and is non-conservative.

Question 8c: LOSS OF PRIMARY COOLANT: OCCURRENCES
Pertaining to "Small Break LOCA (SBLOCA)."

The McGuire FSAR and WCAP 8356 describe the SBLOCA as a design basis event when it occurs from the Rated Power (MODE 1) and Hot Standby (MODE 3) conditions. The assertion contained in the DPO is that "until further evaluations are made, we must conclude that the current Safety Analysis Limits of the SBLOCA event is four RCS pumps OPERABLE in MODE 3 down to 425 psig/350°F" and that the operability of less than four RCS loops in MODE 3 would not be in conformance with the current safety analysis limits and is not conservative. The DPO also contains a similar assertion for the large break LOCA scenario.

Question 8d: OCCURRENCES CAUSING AN INITIAL INCREASE OF RCS TEMPERATURE

The assertion contained in the DPO states that the increase of RCS temperature events are of concern because of the potential influence of the positive moderator temperature coefficient. It discusses several events and states that all but one are licensing basis events from rated power. The conclusion of the DPO is that these events are important in MODE 3 due to the positive moderator temperature coefficient and states that operability of less than four RCS loops in MODE 3 would not be in conformance with the safety analyses limits and is not conservative.

Question 8e: AVAILABILITY OF RCPs

The DPO states that four RCS loops would be required in MODE 3 to meet the requirements of the licensing basis events from zero power. In addition it suggests that, in MODE 4, a minimum set of RCS pumps and loops be used to cool and depressurize the plant down to effectively zero pressure in the steam generators before transferring the heat sink to the RHR system. This is to ensure control of Steam Line Break and LOCA events down to RCS conditions where RCS flows are not necessary. The part of this question addressing MODE 4 is addressed in Concern 18A.

Resolution

In the new STS the LCO for RCS loops in MODE 3 states:

[Two] RCS loops shall be OPERABLE, and

a. [Two] RCS loops shall be in operation when the reactor trip breakers are closed, or

b. One RCS loop shall be in operation when the reactor trip breakers are open.

The numbers in brackets indicate that each plant must supply the number of pumps which is required to meet their safety analysis. For four loop Westinghouse plants, analysis indicates that two is the appropriate number of RCS loops.

At the time of the DPO submittal, this TS required that two RCS loops be OPERABLE and only one RCS loop be in operation in MODE 3. The FSAR for McGuire (and other Westinghouse plants) assumed from two to four RCPs operating for many of the accidents discussed above. Westinghouse acknowledged the discrepancy in a letter dated July 9, 1984, from E. P. Rahe to D. Eisenhut. At that time, Westinghouse reviewed the safety analyses for the accidents which are the most limiting at zero power for the reduced flow conditions of one RCP. These accidents are the steamline break, rod ejection, and control rod bank withdrawal from subcritical conditions. For the rod ejection and steamline break events, Westinghouse determined that the inconsistency between the safety analysis and the Technical Specification would not impact the conclusions presented in the FSAR. The analyses showed that the applicable accident criteria were met with only one RCS pump operating.

For the bank withdrawal from subcritical event, Westinghouse performed calculations which showed that the DNBR design basis may not be met when only one RCP is in operation. Consequently, the Westinghouse STS were changed to require at least two RCS loops in operation with the reactor trip breakers closed to meet the safety analysis limits for an inadvertent bank withdrawal from subcritical.

For the SBLOCA an analysis was conducted by Westinghouse assuming that all pumps were initially operating followed by either all the pumps tripping or all the pumps continuing to operate. The general conclusion was that there was a smaller peak clad temperature for the case of all the pumps operating when compared to the case of all the pumps tripped. This case forms the bounding analysis since the reactor coolant pumps are not automatically tripped during the SBLOCA and continue to operate after the SBLOCA. For ECCS analysis for Westinghouse four LOOP plants the most conservative results are obtained when the RCPs are assumed to be tripped at the initiation of a postulated LOCA. The DPO's assertion is unsubstantiated since the ECCS analysis demonstrated that acceptable fuel cladding temperatures resulted for the more conservative scenario which resulted when the RCPs are assumed to be shut down.

Therefore, for the limiting design basis events at zero power, the proposed new STS will ensure the safety analysis limits are met. The other events described in the DPO are not limiting design basis events at zero power and are thereby bounded by the limiting events.

AVAILABILITY OF REACTOR COOLANT SYSTEM LOOPS IN MODE 4
(HOT SHUTDOWN)

Concern 18A
Question 10
S Page 3/4 4-3

Issue

The DPO proposed two additional requirements for this specification. The first is that two RCS loops be Operable whenever RHR loops are in operation, in order to provide for the failure of a single motorized valve in the RHR/RCS suction line. The second is that surveillance requirements be added to require a determination of the operability of the associated Auxiliary Feedwater System and the Atmospheric Dump Valves.

Specifically, the DPO concerns the McGuire FSAR, which describes a scenario comprised of the failure of a single motorized valve in the RHR/RCS suction line concurrent with the loss of offsite power. For this scenario the DPO asserts that two RCS loops should be operable whenever a plant has RHR operating in Mode 4. Furthermore, the DPO asserts that the current specifications are not conservative because they lack operability requirements for the Auxiliary Feedwater Systems or Atmospheric Valves in Mode 4.

Resolution

The new Westinghouse STS require two loops consisting of any combination of RCS loop and RHR loops be Operable and at least one loop be in operation in Mode 4. The Bases for this LCO states, "Any one loop in operation provides enough flow capacity to remove the decay heat from the core with forced circulation. The second loop, which is required to be OPERABLE, meets single failure criteria." Therefore, in order for a licensee to take credit for each loop, there cannot be a single failure which could render both loops inoperable. The McGuire design which is typical of Westinghouse plants includes a single RHR suction line which connects the reactor coolant loop to the RHR pumps. This RHR suction line contains two motorized valves in series. The DPO asserts that a single failure concurrent with the loss of offsite power could cause one of these valves to fail close during Mode 4; thereby, eliminating the core cooling capability of the RHR system. These valves are opened and left open when core cooling via the RHR is initiated in Mode 4. Since motorized valves fail in the "as is" position, these suction line valves remain open after a single active failure concurrent with a loss of offsite power resulting in the RHR system maintaining its full functional capability in Mode 4. Therefore, to require in the TS that 2 RCS loops be operable whenever the RHR loop(s) are in operation is not necessary.

As discussed in the resolution of concern 29A, the new Westinghouse STS require the operability of one motor driven AFS pump in Mode 4 when a steam generator is relied on for heat removal. The new STS do not require operability of the Atmospheric Dump Valves (ADVs) in Mode 4. The preferred method of removing heat from the steam generators in Mode 4 is through the turbine bypass valves to the condenser. If this path becomes unavailable, the heat load is low enough in Mode 4 that SG secondary side steaming would take time to reach a high enough pressure to necessitate venting. Several options (including the opening of ADVs) would be available to the operators during that time to achieve venting or eliminate the need to vent. Ultimately, the safety valves would vent the pressure. The safety valve LCO does not require the safety valves to be operable in Mode 4; however, the TS definition of Operability and the ASME code require operability of the safety valves when the steam generator is operable.

The DPO also discussed concerns about the depth of the Surveillance Requirements (SR) and suggested that additional SRs be added on the systems in this LCO. The existing SRs are not intended to be complete tests of the system performance; they are quick, simple, frequent checks (every 12 hours) to ensure that the equipment is operating properly. The more detailed testing is done in chapter 6 of the STS and the inservice test program. Therefore, there is no need to supplement the existing SRs.

AVAILABILITY OF REACTOR COOLANT SYSTEM LOOPS IN MODE 5 (COLD SHUTDOWN)

Concern 19A
Question 8e
T[illegible].4.1

Issue

The DPO made the following assertions for the Cold Shutdown Mode of operation:

(1) If the steam generators are used for cooling the Auxiliary Feedwater System and Atmospheric Dump Valves should be required to be operable.

(2) There is no basis for allowing the operating RHR pumps to be de-energized for 1 hour.

(3) The surveillance requirements do not fully test all aspects of operability of the RHR and RCS loops.

Resolution

The new Westinghouse STS require one RCS loop or one RHR loop to be operating and either one additional RHR loop to be OPERABLE or the secondary-side water level of at least two steam generators to be [17]% or greater of the Low-Low Trip Setpoint. A note in the Limiting Condition for Operation (LCO) allows the RHR pump or the RHR loop in operation to be de-energized for up to 1 hour provided: (1) no operations are permitted that would cause reduction of the RCS boron concentration; and (2) core outlet temperature is maintained at least 10°F below saturation temperature. The Surveillance Requirements verify that at least one RHR or RCS loop is operating and that there is adequate water level in the SG.

In MODE 5 with the RCS loops filled, the objectives of this LCO are: (1) to remove decay heat generated in the fuel; and (2) to prevent stratification of the soluble boric acid. In MODE 5, an operating RHR or RCS loop accomplishes these functions. The other operable RHR loop or the two steam generators with adequate secondary side water level provide single failure protection. Under these conditions of low heat load, the heat sinks in the two steam generators provide adequate back-up cooling until a RCS or RHR loop can be put into operation. Also, under these low heat load conditions, operability of neither the Auxiliary Feedwater System nor the Atmospheric Dump Valves is necessary.

In the new STS, the note allowing the operating RHR or RCS pump to be de-energized for up to 1 hour is limited; it may only be exercised once in an 8 hour period. This time period is needed to perform surveillance testing. As explained above, compensatory measures including close monitoring of coolant temperatures are required to exercise the 1 hour allowance. The RHR or RCS loop would still be available to be restarted if coolant temperatures exceeded the surveillance limit in the note. Experience in the use of this note has shown that plants do not experience heating or boric acid stratification problems.

The surveillances in this LCO do not include testing of alarms and design basis flow rates. The purpose of these surveillances is to provide quick, simple, frequent checks (every 12 hours) to ensure that the equipment is operating properly. The more detailed testing is done in chapter 6 of the STS and the inservice testing program. Therefore, there is no need to augment the existing surveillance requirements.

STANDBY NUCLEAR SERVICE WATER POND
(ULTIMATE HEAT SINK)

Concern 20B
TS Section 3/4.7.4

Issue

The DPO asserts that the applicability section of the Standby Nuclear Service Water Pond (SNSWP) TS which includes Modes 1, 2, 3, and 4 should also include Modes 5 and 6.

Resolution

The need for operability of the Ultimate Heat Sink (UHS) in Modes 5 and 6 is addressed in the new STS through the definition of Operability. UHS is required as a support system for other systems such as RHR which are required by STS to be operable in Modes 5 and 6. In Modes 5 and 6 the heat load is low; therefore, the demands on the UHS as a support system would be well below the temperature and volume requirements of the UHS LCO.

REFUELING OPERATIONS - LOW WATER LEVEL

Concern 21B
TS Page 3/4.9-11

Issue

The DPO asserts that both RHR loops should be in operation in Mode 6 with less than 23 feet of water above the top of the reactor vessel flange. In support of this statement the DPO postulates the loss of the operable RHR loop without operator action; the DPO asserts that this scenario would result in boiling in 5 minutes and core uncovery in 100 minutes.

Resolution

The new STS require that one RHR loop be operating and the other RHR loop be operable under the low water level conditions. The new STS also require action to restore RHR cooling if it is lost. The DPO seems to express concern over a scenario where the operating RHR loop fails and the reactor coolant heats up and uncovers the core before the operators become aware of the inoperable RHR loop and take action to operate the other RHR loop. The operating RHR loop has an alarm for low RHR flow and other instruments provide multiple, diverse indications of loss of RHR cooling to the operators. In addition, several operations personnel would be present in the area of the reactor cavity. For these reasons it is highly unlikely that a loss of RHR flow would go unnoticed and uncorrected long enough to allow the core to become uncovered. The other RHR loop is required by STS to be operable. Through the definition of Operability the support systems necessary for operation of the other RHR loop must also be operable. Finally, both offsite and emergency diese generator power are required to be operable in Mode 6 by STS 3.8.1.2. Therefore, an additional STS requirement to have both RHR loops operating is not necessary.

AUXILIARY FEEDWATER SYSTEM

Concern 29A
TS Page 3/4.7-4

Issue

The DPO states that the TS should require operability of the steam driven auxiliary feedwater pump in Mode 4. The DPO also questions the derivation of the Steam Line - Pressure Low signal.

Resolution

The new STS require operability of the one motor driven Auxiliary Feedwater System (AFP) pump in Mode 4 whenever a steam generator is relied on for heat removal. Once the plant is switched to RHR cooling, operability of the Auxiliary Feedwater System (AFS) is no longer required. This is a change over the current STS which do not require AFS operability in Mode 4. The current STS assume that the plant switches from SG cooling to RHR cooling when a change from Mode 3 to Mode 4 occurs. During the review of the new STS, it was found that some plants maintain cooling via the steam generators into the upper temperature range of Mode 4. These plants maintain operability of the Auxiliary Feedwater System via administrative controls until cooling is switched to RHR.

The Steam Line Pressure Low Signal used in the main steam line break accident analysis is derived from steam line sensors downstream of the steam generator flow restriction orifices. This results in a conservative measure of steam generator pressure since the steam flow restrictors do not cause a significant pressure drop except during a doubled ended steam line break. The blowdown phase of the double ended steam line break lasts only a few seconds. The accurate pressure sensing in the steam lines (the generation of the steam line pressure low signal) requires less than 2 seconds and steam line isolation requires less than 7 seconds. Deriving this low pressure signal from sensors downstream of the steam generator flow restriction orifices is conservative.

MAIN STEAM ISOLATION VALVES

Concern 30A
TS Page 3/4.7-8

Issue

The DPO contains an assertion that there is a conflict between TS Sections 3.7.1.4, 3.6.3 and TS Table 3.3-4 dealing with the applicability modes for operability of the Main Steam Isolation Valves.

Resolution

The Main Steam Isolation Valves (MSIV) have two accident mitigation functions. First, during a steam line break the MSIVs close to prevent blowdown from more than one steam generator. This function is necessary in Modes 1, 2, and 3. In Mode 4, the lower reactor coolant temperature reduces the consequences of the steamline blowdown such that MSIV closure is not necessary. In the new STS, the LCOs which address this function in plant systems and instrumentation chapters require MSIV operability in Modes 1, 2, and 3.

The second accident mitigation function for the MSIVs is containment isolation. This function is necessary in Modes 1, 2, 3, and 4. In the new STS, the LCOs which address this function in the containment and instrumentation chapters require MSIV operability in Modes 1, 2, 3, and 4.

STEAM GENERATOR POWER OPERATED RELIEF VALVES (SGPORV)

Concern 31A
TS Page 3/4.7.8a

Issue

The DPO states that the TS should include the SGPORVs since under the loss of offsite power condition these valves are necessary for cooling down the plant by natural circulation. Furthermore, the DPO states that additional relieving capacity should be covered by TS since the reactor will operate at power levels as high as 20% during the loss of offsite power condition.

Resolution

The loss of offsite power will cause the Reactor Coolant Pumps (RCP) to trip since the only power source for these pumps is the offsite grid. At reactor power levels greater than or equal to 10% the tripping of the RCP will initiate a reactor scram. At reactor power levels less than 10% the reactor would be manually scrammed by the operator. The power level for either scram is equivalent to the initial decay heat power level after a scram. The required heat removal capacity is within the design limits of natural circulation.

The bases for the new STS state that the Atmospheric Dump Valves (ADVs) will be used to cool down the plant for accidents which are accompanied by a loss of offsite power. Therefore, the ADVs are part of the primary success path for such accidents and are required by the new STS in Modes 1, 2 and 3. PORVs are used to minimize the opening of the Main Steam Safety Valves (MSSVs); the MSSVs are part of the primary success path for events such as full power turbine trip without steam dump. Since the SGPORVs are not part of the primary success path, they do not meet the criteria for inclusion in TS pursuant to the Commission's Policy Statement. Therefore, operability of the SGPORVs is not required by the new STS.

COMPONENT COOLING WATER SYSTEM

Concern 32A
TS Section 3/4.7.3

Issue

The DPO states that the applicability of the Component Cooling Water System (CCWS) TS which includes Modes 1, 2, 3, and 4 should also include Modes 5 and 6.

Resolution

The need for operability of the Component Cooling Water System (CCWS) in Modes 5 and 6 is addressed in the new STS through the definition of Operability. CCWS is required as a support system for other systems such as RHR which are required by STS to be operable in Modes 5 and 6. Since the two trains of the CCWS are typically cross connected as in the McGuire Plant, one train of CCWS is adequate to meet the support function for both RHR trains in Modes 5 and 6. Both trains of the CCWS are not required to be operable to provide single failure protection in Modes 5 and 6 since the heat load is low, and there are other methods which can be instituted by the operators to handle the low heat load if the CCWS fails. Also, this allows licensees to perform necessary maintenance and system modifications.

SERVICE WATER SYSTEM

Concern 33A
TS Section 3/4.7.4

Issue
The DPO states that the applicability section of the Service Water System (SWS) TS which includes Modes 1, 2, 3, and 4 should also include Modes 5 and 6.

Resolution

The need for operability of the Service Water System (SWS) in Modes 5 and 6 is addressed in the new STS through the definition of Operability. SWS is required as a support system for other systems such as RHR which are required by STS to be operable in Modes 5 and 6. Since the trains of the SWS are typically cross connected as in the McGuire Plant, one train of SWS is adequate to meet the support function for both RHR trains in Modes 5 and 6. Both trains of the SWS are not required to be operable to provide single failure protection in Modes 5 and 6 since the heat load is low, and there are other methods which can be instituted by the operators to handle the low heat load if the SWS fails. Also, this allows licensees to perform necessary maintenance and system modifications.

RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION - HIGH WATER LEVEL

Concern 35A
TS 3/4.9.8

Issue

The DPO states that the action statement should require containment isolation within 2.5 hours when no RHR loops are operable. Also, the DPO states that the TS should require operability of the containment sump and alternate cooling methods in this Mode.

Resolution

In current STS, the action statement allows 4 hours to isolate containment when no RHR loops are operable. In the new STS, the instruction to isolate containment has been removed from the action statement. The new STS require the establishment of alternate cooling methods whenever the RHR is unavailable. The action statement concentrates on the most important task of supplying core cooling and leaves the provision of containment isolation to the licensee's contingency procedures.

The alternate cooling methods do not need to be required to be operable in Mode 6 with the cavity flooded. The Commission's Interim Policy Statement on Technical Specification Improvement states criteria for deciding which equipment and conditions should be included in TS. Under those criteria the primary success path system, RHR, is required by TS. The provision of alternate cooling methods referred to in the first paragraph is left as the responsibility of the licensee's contingency procedures. In Mode 6 with the cavity flooded, there is a large volume of water over the core and a low decay heat load. Under these conditions the operator has (1) alternative cooling methods which can handle the low decay heat load and (2) time to implement those alternatives.

REFUELING OPERATIONS - LOW WATER LEVEL

Concern 36A
TS Page 3/4.9-11

Issue

The DPO states that the action statement should require containment isolation immediately when no RHR loops are operable.

Resolution

In the current STS, the action statement allows 4 hours to isolate containment when no RHR loops are operable. In the new STS, the instruction to isolate containment has been removed from the action statement. As discussed in the resolution to concern 35A, the action statement concentrates on the most important task of supplying core cooling and leaves the provision of the containment isolation to the licensee's contingency procedures.

REACTOR TRIP INSTRUMENTATION SETPOINT

Concern 38A
Table 2.2.-1

The assertion in the DPO states that the Technical Specification nomenclature "Low Power Reactor Trips Block, P-7" is incorrect and should be labeled "High Power Reactor Trips Block".

Resolution

The nomenclature is an acceptable description for this function without change; however, information describing the P-7 permissive and the P-10 and P-13 trips is discussed in detail in the new STS Bases under the title "Low Power Reactor Trips Block, P-7."

REACTOR TRIP INSTRUMENTATION SETPOINTS

Concern 3B
Table 2.2-1

Issue

The assertion in the DPO is that the absence of the permissive P-7 [on P-10 and P-13] introduces new events to evaluate for safety. The DPO further asserts that the impact of blocking the Pressurizer Water Level-High trip below P-7 should be evaluated.

Resolution

The new STS include in Table 3.3.1-1 the P-7 [on P-10 and P-13] interlock.

Several reactor trips (including Pressurizer Water Level-High) are only required when operating above 10% power, the P-7 setpoint. The P-7 interlock enables and disables trips as reactor power passes through the 10% power setpoint. Below 10% power, the RCS is capable of sufficient natural circulation without any RCP running to prevent DNB.

The Pressurizer Water Level-High trip is a back-up signal for Pressurizer Pressure High trip and provides protection against passing water through the pressurizer safety valves. A reactor trip is actuated before the pressurizer is water solid. These level channels provide input to the pressurizer level control system and do not actuate the safety valves.

This trip must be operable in Mode 1 when there is a potential for overfilling the pressurizer. This trip is automatically enabled on increasing power by the P-7 interlock. On decreasing power the absence of P-7, automatically blocks this trip. Below the P-7 setpoint, transients which could raise pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) INSTRUMENTATION

Question 5B
Concern 12B
Table 3.3-3

Issue

The assertion in the DPO recommended that the staff consider the consequences of not requiring automatic switchover to recirculation on RWST level for Mode 4 in addition to Modes 1, 2, and 3.

Resolution

The new STS and the current STS require the operability of the switchover to containment sump on RWST level low for Modes 1, 2, 3, and 4.

SOURCE RANGE NEUTRON FLUX

Concern 10A*
TS Page 3/4.3 - Item 6c

Issue

The assertions the DPO are as follows:

1) Power range neutron flux trip (low and high) setpoints and intermediate range neutron flux are used for events being initiated in a "Subcritical" condition as described in FSAR (table 7.2.1-4); however, the TS does not require their operability in Modes 3, 4, and 5.

2) Furthermore, the source range trip is required to be operable in Modes 3, 4, and 5, yet there is no technical specification for it.

Resolution

1) The Power Range, Neutron Flux-High Setpoint and Low Setpoint do not have to be operable in Modes 3, 4, and 5, because the reactor is shutdown and the Nuclear Instrumentation System (NIS) power range detectors cannot detect neutron levels in the shutdown range. Other RTS functions and administrative controls provide protection against reactivity additions when in Modes 3, 4, and 5.

The Intermediate Range Neutron Flux trip does not have to be operable in Modes 3, 4, or 5 because the controls rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition. The core also has the required Shutdown Margin to mitigate the consequences of a positive reactivity addition accident and this margin is required to be monitored frequently. In Mode 6, all rods are fully inserted and the core has an increased Shutdown Margin. Also, the NIS intermediate range detectors cannot detect neutron levels in this range.

2) The new STS require the source range neutron flux trip function to be operable in Modes 2, 3, 4, and 5 with the reactor trip breakers closed and the rod control system capable of rod withdrawal. It is also required to be operable in Mode 3, 4, and 5 with trip breakers open when the only function of the source range monitor is indication.

P-11 INTERLOCK

Concern 10B
TS Page 3/4.3-2

Issue

The assertion in the DPO is that the licensee needed to evaluate the consequences of an event involving a Main Steam Line Break below the P-11 interlock reactor trip such that the trip will not be initiated by the Negative Steam Line Pressure Rate - High signal. This concern acknowledges the source range and intermediate range nuclear flux trips under these (small and intermediate size breaks) circumstances, on any return to power, as not being necessary because they are not required in the safety analysis. Their current proposed status precludes crediting their function capability and would leave only the power range low setpoint trip to trip the reactor. Furthermore, the resulting power levels of 35% as a safety analysis limit would be unacceptable without a substantial analysis of the event.

Resolution

The P-11 interlock permits a normal unit cooldown and depressurization without actuation of safety injection (SI) or main steam line isolation. With 2/3 pressurizer pressure channels less than the P-11 setpoint, the operator can manually block the Pressurizer Pressure - Low and Steam Line Pressure - Low SI signals and the Steam Line Pressure - Low Steam Line Isolation signal. When the Steam Line Pressure - Low Steam Line Isolation signal is manually blocked, the main steam isolation signal on Steam Line Pressure - Negative Rate - High is enabled. This provides protection for a steam line break by closure of the main steam isolation valves and initiation of a reactor trip.

ESFAS INSTRUMENTATION

Concern 14A
Table 3.3-3

Issue

The DPO asserts that the operability of the containment Phase B isolation on a Containment Pressure High-High signal should be required in Mode 4. The DPO also asserts that a containment Phase B isolation is necessary to establish containment integrity.

Resolution

The Containment Pressure High-High signal is initiated due to a large break LOCA or steam line break and it actuates containment spray and Phase B containment isolation. Containment Pressure High-High must be operable in Modes 1, 2 and 3 when there is sufficient energy in the primary and secondary sides to challenge the containment pressure High-High setpoint. In Mode 4, there is insufficient energy in the primary and secondary sides to challenge the Containment Pressure High-High set point. Therefore, operability of the Containment Pressure High-High signal is not necessary.

Containment Pressure High actuates SI and SI actuates containment Phase A isolation. Containment Phase A isolation isolates all lines into containment except those associated with the Engineered Safety Features. The CCW System, which is typically an Engineered Safety Features System as in the McGuire Plant, is not isolated by the Phase A isolation. Containment Phase A isolation establishes containment integrity and allows the continued use of the Reactor Coolant Pumps (RCPs) which rely on the CCW. The containment Phase B isolation is actuated by Containment Pressure High-High and isolates the CCW. The high pressure which causes the Containment Pressure High-High signal indicates accident conditions for which RCP operation is not necessary.

ESFAS INSTRUMENTATION TRIP SETPOINTS

Concern 15A
Table 3.3-4

Issue

The DPO asserts that a new Functional Unit which is part of ESFAS should be included in the TS. This new Functional Unit is "Closure of the Feedwater Isolation, Main Feedwater, and Bypass Modulation Valves."

Resolution

The new STS and the current STS include these valve closure functions under other functions in the ESFAS tables. The DPO acknowledges this fact, but asserts that the function needs to be included as a separate function in the ESFAS tables. The DPO gives no justification for including this separate function; therefore, no additional functional unit needs to be included in the STS.