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

Attachment 1

DEC 0 7 1983

MEMORANDUM FOR: G. Norman Lauben, Section Leader
Section A
Reactor Systems Branch, DSI

FROM: R. B. A. Licciardo, Nuclear Engineer
Section A
Reactor Systems Branch, DSI

SUBJECT: DIFFERING PROFESSIONAL OPINION RELATED TO TECHNICAL SPECIFICATION FOR MCGUIRE UNIT 2 (AND PROPOSED FOR MCGUIRE UNIT 1)

This memorandum constitutes formal submission of a Differing Professional Opinion (DPO) in accordance with NRC Manual Chapter NRC-4125 and NRC Appendix 4125.

This DPO relates to the operation of McGuire Unit 2 (also proposed for McGuire Unit 1) safety systems necessary to provide assurance of public health and safety.

Disparities existing between current Technical Specifications relating to these systems and the safety analyses of record within the existing licensing basis, suggest that the existing regulatory requirements identified in 10 CFR Parts 50.36, 50.46 and 50, Appendix A could be compromised. This compromise could manifest itself in increased risk to public health and safety beyond that intended in the existing licensing basis. As an example, the mitigating effects of the Emergency Core Cooling System (ECCS) could be compromised.

In accordance with NRC Appendix 4125 G.2.a, I request that this DPO be presented to an impartial peer review group for review, evaluation and comment.

R. Licciardo

R. B. A. Licciardo, Nuclear Engineer
Section A
Reactor Systems Branch, DSI
U.S. Nuclear Regulatory Commission

Attachment 2
[illegible]

06/15/85 (For [illegible])

Memorandum To: Thomas M. Novak, Assistant Director for Licensing

From: R. Wayne Houston, Assistant Director for Reactor Safety
Division of Systems Integration [illegible]

Subject: STAFF REVIEW OF PROOF AND REVIEW COPY OF PROPOSED TECHNICAL SPECIFICATIONS FOR MCGUIRE UNITS 1 & 2

Reference: 1) Letter from H. B. Tucker (D.P.Co) to H. R. Denton (NRC) dated September 27, 1982 to the Subject of "McGuire Nuclear Station"

2) Memo from C. O. Thomas (SSPB) to Brian W. Sheron (RSB) dated January 14, 1983 on the Subject of "Proof and Review of McGuire - Units 1 and 2 Technical Specifications.

Plant Name:	McGuire Nuclear Station, Units 1 and 2
Docket Number:	Unit 1, 50-369; Unit 2, 50-370
Licensing Stage:	Unit 1-OR; Unit 2-OR
License No.:	Unit 1, NPF-9; Unit 2, NPF-17
Responsible Branch:	Licensing Branch #4
Project Manager:	Ralph A. Birkel
Review Branch:	Reactor Systems Branch
Review Status:	Complete with ~~need for confirmatory information~~ open items.

The Reactor Systems Branch has completed its review of the Subject document.

This review is in response to the request of Reference 2.

Our review has been based on the information in the Final Safety Analysis Reports for the McGuire Nuclear Station, Units 1 and 2, up to and including revision 45, and the Safety Evaluation Report for McGuire Nuclear Station, Units 1 and 2, NUREG 0422 and related Supplements 1 through 6.

Summarily, the staff finds a number of substantive differences between the Proposed Technical Specifications supplied do this Division Under Reference 2 and the Docketed Information, which could ultimately have a significant impact on Public Health and Safety. We have proposed Technical Specifications to correct this situation.

Wayne Houston,
Assistant Director for
Reactor Safety
Division of Systems Integration

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Wayne Houston,
Assistant Director for
Reactor Safety
Division of Systems Integration

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Safety Evaluation Report
McGuire Nuclear Station: Units 1 and 2
Proposed Technical Specifications

INTRODUCTION

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

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

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

The RSB staff has conducted its portion of the review by a more detailed examination of those sections and related systems which are its primary responsibility as defined by the Standard Review Plan. These sections has been reviewed against the information in the Final Safety Analysis Report in references 4 through 8, and the Safety Evaluation Reports to references 10 through 15, together with the additional information in references 16 through 20.

The sections reviewed are listed in table 1. Detailed changes are principally identified in the Proof and Review Copy of the Technical Specifications. This SER summarizes the related significant evaluations.

A ~~number of the proposed changes which could~~ld be readily i~~ntroduced inside the structure of the existing Standard Technical Specifications have already been incorporated into the Technical Specifications~~ for ~~McGuire Unit 2 issued with License No. NPF-17 on March 1~~983. ~~A number of other proposed changes were not incorporated at the same time, since they involved changes to the Standard Technical Specifications necessitating substantive~~ ~~NRC review~~, a~~nd because the Safety Analysis Reports stated that some of the more~~ significant ~~changes would~~ nevertheless ~~be a part of Plant Operating Procedures and therefore of plant operations. It was evaluated that relative short term operation of the facility under these different administrative but identical engineering circumstances would~~ not ~~have significantly impact public health and safety~~ whilst ~~the principal longer term regulatory~~ ~~issues of incorporating such procedures into the~~ ~~[illegible] Technical Specifications received the required consideration.~~

Evaluation

Section 2.1 SAFETY LIMITS

T.S. Pages 2-1, 2: It is necessary to emphasize that the "acceptable (region of) operation" in Fig. 2.1-1 does not represent an allowable region of Steady State Operation for the reactor core; that actual steady state operation is confined to a very narrow band of operation on the Figure so that the Safety Limits shown are not exceeded under Anticipated Operational Occurrences.

Table 2.2-1 REACTOR TRIP SYSTEM INSTRUMENTATION SET POINTS

All related values have been checked. There is no documentary basis for confirming the S-G Water Level -Low-Low trip. The Reactor Coolant Pump Breaker position Trip must be added; [redacted].
Reference to the bypass of the Anticipatory Turbine trip at P-8 is necessary. Interlocks, P-12 and P-13 should be described.

Section 3/4.1.1 BORATION CONTROL

T.S. Pages 3/4 1-1, 2, 2a: Reference 16; page Q 212-47e states that boron concentration be increased to at least the cold shutdown boron concentration before cooldown is initiated, as a means of protecting against NON-LOCA Accidents during start up and shut down. The proposed changes incorporate this requirement.

T.S. Page 3/4 1-6: Changes to the "Minimum Temperature For Criticality" recognize that the prescribed value is not only a minimum, but also a maximum. The worst case MSLB event is at End of Cycle at no load; and the analysis of record is presumably at this value. Any increase would be less conservative with respect to the current Docket.

The minimum temp. of 551° F must be reconciled with the value of 557° F for programmed no-load Tavg given in reference 20 Fig. 5.3.3-1.

Section 3/4.1.2 BORATION SYSTEMS

T.S. Pages 3/4 1-7, 8, refce. Flow Path~~s~~- Shutdown and Operating: Changes reflect the current boration requirements prior to cooldown from no load programmed Tavg.

T.S. Pages 3/4 1-9, 10, refce. Charging Pumps: The changes reflect the current restrictions on charging pump operation arising from prescribed ECCS operation in Reference 12, page 6-6 and Reference 11, pages Q 212-47 and 47a.

T.S. Pages 3/4 1-11, 12, refce. Borated Water Source - Shutdown and Operating: Level detection systems need T.S. at minimum levels (in reference 17).

T.S. Page 3/4 1-13a), refce. Proposed Subsection on INSTRUMENTATION: This meets the specific requirement (to guard against the Boron Dilution Event) described in neférence 11), page 15-2.

T.S. Page 3/4 1-20a), refce. Proposed Section on SHUTDOWN ROD INSERTION LIMITS IN MODES 3, 4 and 5: Technical Specification is necessary to minimize the level of detail in the T-S necessary to protect against LOCA and Non LOCA events from HOT STANDBY (MODE 3) down to COLD SHUTDOWN (MODE 5).

T.S. Page 3/4 1-21, refce. CONTROL ROD INSERTION LIMITS: Overpower and over temperature ΔT protection systems incorporate rod insertion limit stops. These limit stops should be checked. Control Rod Insertion Limits should be specified for MODES 3, 4, and 5 to minimize the level of detail in the T.S. necessary to protect against LOCA and Non LOCA events from HOT STANDBYE (MODES 3) to COLD SHUTDOWN (MODE 5).

<u>Section 3/4.2 POWER DISTRIBUTION LIMITS</u>

RSB has not reviewed this section on the understanding that it is the primary responsibility of Core Performance Branch.

<u>Section 3/4.2.5 DNB (AND RCS PRESSURE) PARAMETERS</u>

T.S. Pages 3/4 2-15, 16: The additional information is necessary for DNB purposes, and also provides necessary conditions for protection against RCS overpressurization.

Values in Table 3.2-1 should be given as indicated (Setpoint) values. Table 3.2-1 shows marked differences from Reference 20, figure 5.3.3-1. These differences must be resolved.

TABLE 3.3-2 REACTOR TRIP INSTRUMENTATION RESPONSE TIMES

Item 7: Overpower ΔT response time is necessary. Reference 10, Table 15.1.3-1 gives a value of 6 secs; extensive reference is made to its potential protective function.

Item 11: Pressurizer Water Level - High is necessary. Reference 10, page 15.4-13, item 16 specifically mentions its potential protective function. Reference also previous evaluation of Table 3.3-1, item 11.

Although the above two items are not apparently the ultimate basis for protection in the Accident Analysis in reference 10, those Analyses represent a limited number of events which are proposed as "expected " to bound all possible events at the plant in terms of severity. There is no guarantee that the large number of other possible events will never use these two protection items to primary advantage; for example without the trip, failure of the control grade Pressurizer Level Control System could result in overfill of the pressurizer and water ingress into the RCS relief system at power.

Item 16, Turbine Trip: A Response Time of 1.0 secs is docketed in Reference 7, Table 15.1.3-1, as used in Accident Analyses.

Item 17, Safety Injection Input from ESF: This should be provided unless it is covered in overall response times in Table 3.3-5.

Item 21, Reactor Coolant Pump Breaker Position Trip: Response time should be provided.

TABLE 3.3-3 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

Item 1c: Containment Pressure High in Mode 4 is required. See reference 8, page Q 212-47e, item 24. [redacted]

Item 1e: Steam Line Pressure-Low. Correction to 3## has been made. The ## referred to a MSLB protection system which was not applicable to McGuire Units 1 and 2.

Item 3.b3, Phase B Isolation on Containment A Hi-Hi should be provided in Mode 4 to protect against LOCA and MSLB events. ~~Note: T.S. item 34.6.1 requires~~ [redacted]t integrity in Modes 1 through 4.

Item 4, Steam Line Isolation: Automatic initiation is required in Mode 4 to support the Negative Steam Line Pressure Rate - High signal from a MSLB type event. Manual Initiation is also necessary for a small break event. ([redacted].

Item 4d: Identify that this item is not available above the P-11 interlock.

Item 7d: This item in reality is a re-alignment of the suction valves on the auxiliary feedwater suction lines to alternate water sources.

Item 7g: Need to identify that channels are required to trip on all main feedwater Pumps as well as the Trip Signal. [redacted] TBD, [redacted] TBD

T.S. 3/4 3-23, Item #: Provide complete details of the P-11 function; the information is in the Basis, which however is not legally a part of the Technical Specifications.

TABLE 3.3-4 ESF ACTUATION SYSTEM INSTRUMENTATION SET POINTS

There are two sets of values docketed: Reference 7, Table 15.1.3.1 and reference 18).

There is uncertainty in the use of reference 18; the report references Unit 1 only, and has erred in including a Main Steam Line Isolation Logic which does not apply to either of Units 1 and 2. For values used in Accident Analysis, differences of concern included for: Pressurizer Pressure - low Safety Injection; 1685 psig (reference 18) to 1835 psig (reference 11); Pressurizer Pressure - Low Reactor Trip, 1845/psig rafce (18) to 1835 psig (reference 11).

There is no basis for checking other FSAR values against reference 18, as they have not been provided.

Reference 18), table 3-4 also provides set points and allowable values (i.e., with drift allowance) for Tech. Spec. use. These agree except for the Negative Steamline Pressure Rate - High; the amended values in the T.S. for this parameter are from reference 18.

Item 7(c): Steam generator water level - Low Low, Need to confirm correspondence of the Trip Set Point with calculation values used in Reference 7.

Item 10a: Engineered Safety Features Actuation Interlocks, Pressurizer Pressure, P-11.

The Trip Set Point of 1955 psig is unacceptable, and should be reduced to 1900 psig less the allowable maximum error. Safety Injection is required down to 1900 psig according to the Docketed Analysis, in reference 8, page Q212-47, item 212.75 1A. This should be verified before exceeding (TBD) power.

TABLE 3.3-5 ENGINEERED SAFETY FEATURES RESPONSE TIMES

In general, Safety Injection on ECCS using onsite power is less conservative by 2 secs i.e., 27 secs instead of 25 secs in the Docket (reference 7). Also "Diesel Generator Start" times are less conservative by 1 sec, i.e., 11 secs instead of 10 secs. Of principal concern is the time for start of Auxiliary Feedwater Pumps; these are generally given as 60 secs on the assumption of immediate availability of water supply, from non-seismic non-safety classified condensate storage tanks. The assured water supply is from the Seismic Category I Nuclear Service Water System and the standby Nuclear Source Water Pond which are permitted up to 76 secs from onsite power sources i.e., 16 secs less conservative. To immediately offset this non-conservatism, we have required in a new TS 3/4 7-5(c), a minimum of non-seismic, non-safety classified Auxiliary Feedwater Condensate Storage, of 175,000 gallons until such time as the appropriate Accident Analysis are repeated with the less conservative value.

The Licensee is required to incorporate the corrected value for Engineered Safety Feature Response times in his Safety Analyses required to be submitted for Cycle 2 re-load. We evaluate that until that time, the probability of an event requiring the Nuclear Service water pond is such as to permit continued plant operation. We also evaluate the significance of the less conservative diesel generator and safety injection response times as again providing an acceptable level of safety until the commencement of Cycle 2 operation. pages T.S ~~T.S. [illegible]~~ -32. The significant information on this page requires substantive clarification.

Section 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

T.S. Pages 3/4 4-1, 2, 3, 4: The following additional transients and accidents are also determinants of RCS Loop OPERABILITY in MODES 3, 4 and 5 (with loops filled):

Concerning "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Sub-Critical Condition." Current Docketed Analysis in reference 10, section 15.2.1, page 15.2-2 is based on four operating loops. This event is possible down to and including Mode 5. At present (without further analysis, we propose that) with Boration to the Cold Shut Down Boron concentration value prior to commencement of cooldown from 557° F, we recommend OPERABILITY and operation of 4 RCS loops through Mode 3. Since the potential for return to criticality will increase as the available shutdown margin is reduced at lower temperatures, special additional considerations may be necessary to prevent this event in Modes 4 and 5.

Concerning "Loss Of Offsite Power to Station Auxiliaries." Accident Analyses in reference 10, table 15.2.9-1 provides resulting natural circulation flows based on four (4) loop operation. Until further justification and evaluation is provided, four operating RCS loops will be required to mitigate this possible event during a cooldown through MODE 3.

Concerning "Accidental Depressurization of the Main Steam System" discussed in Accident Analyses in reference 10, section 15.2.13.1. This event is covered by consideration of the major rupture of a main steamline.

Concerning "Start Up of the Inactive Reactor Coolant Loop"; discussed in reference 7, section 15.2.6, page 15.2-19. Under operating conditions, substantial short term increases in reactivity can occur. The c[redacted]nces of [redacted] including particularly Steam Generator conditions, un[redacted]can [redacted] prior [redacted] Ejection [redacted]

Concerning Small Break LOCA discussed in reference 7, section 15.3.1. and reference 8, item 212.75 page Q 212-47, and item 212.90(6.3) page Q[redacted] These analyses are[redacted]ased on 4 RCS loop operation, with loss of power to RCS pumps on safety injection. RCS pump operation and coast down is important to reducing the loss of core level subsequent to the event by maintaining unseparated two phase flow conditions and in ensuing rapid Boron (mixing and) Injection to the core. Rapid boron injection is not an important issue with boron concentrations already at cold shut down values, but minimizing loss of core level is important. Until further evaluations are made, we must propose OPERABILITY of all 4 RCS loops as a necessary requirement throughout MODE 3 down to 425 psig/300° F.

Concerning the "Loss of Coolant Accident" discussed in Accident Analyses in Reference 7, section 15.4.1, and in Reference 8, item 212.75 page Q 212.47, item 212.90(6.3) page 212-61b. As for the Small Break LOCA, these analyses a[redacted]ased on 4 RCS loop operation, with in general, loss of power to RCS Pumps on Injection. Plant sensitivity studies in reference 19 examined the effects of reactor coolant pump operation on a LOCA by examining the consequences of two cases:

a) where pumps continued to run after the event, and

b) where pumps were tripped on the event.

Due to the action of the running (non-tripped) pumps, less negative flow occurs in the core after the flow reversal compared to the case when the pumps are immediately tripped. The net result is a smaller peak clad temperature for the pumps running case compared to the pumps tripped case. Hence, for ECCS Analyses for Westinghouse four loop plants, the RCPs are assumed to be tripped at the initialization of a postulated LOCA and a locked pump resistance is used for reflood. We conclude, that the analyses of record for MODE 3 discussed in the preceeding reference 8 is presumably based on four RCS pumps operating followed by loss of Reactor Coolant Pumps on initiation of the event. Under these circumstances, the current T.S. position to require only two loop operation in MODES is less conservative, and even more so, when it is considered that the break could occur in the non-circulatoring loop. It is proposed that 4 RCS loops be OPERABLE for MODE [redacted] ided.

Concerning "Major Rupture Of A Main Steamline" discussed in Accident Analyses in Reference 7, section 15.4.2 and Reference 8 item 212.75 page Q 212-47d. Reference 8) proposes that the resulting impact on shutdown margins from this event during MODES 3, 4 and 5 are improved over that of the design basis (of zero power, just critical, Tavg = 557°) as operating Instructions <u>require</u> that boron concentration be increased to at least the cold shutdown boron concentration before cooldown is initiated. This position gives no measure of the resulting shut down margins and/or power level and, the consequences of a stuck rod, with only 2 RC loops operating instead of four. It is conceivable that two loop operation may be less conservative than either 4 RC Pumps continuing to operate or 4 RCPS tripped Safety Injection, due to an increased cooldown in the core due to circulation (compared to the tripped case) but a much decreased core flow rate to handle the event. The potential short term consequences of bulk voiding and loss of circulation in the non-operable loops cannot be ignored. It is recommended that four RCS loops be OPERABLE and operating in MODE 3, to mitigate the consequences of this event [redacted] and justification is provided.

In summary, with the available knowledge, and particularly the evaluations of record, we conclude it is necessary to have four RCS loops operable down to 1000 psig/425° F, and because of the lack of sufficient data to otherwise confirm satisfactory margins, that it also necessary to have 4 RCS loops OPERABLE down to 425 psig/350° F. Additional Technical Specification consideration is necessary to the consequences of start up of an Inactive Reactor Coolant Loop under these circumstances.

T.S. Pages 3/4 4-3, 4, refce. Hot Shutdown: Each of the OPERABLE loops, whether RCS or RHR, are to be energized from separate power divisions to protect against single failure of a bus or distribution system. When the RCS systems are used, the related Auxiliary Feedwater systems are also required to be operable. [redacted] e.

T.S. Page 3/4 4-5, refce. Cold Shutdown - Loops Filled: The [redacted]h. [redacted]ERABILITY [redacted]ements for [redacted]s. [redacted] it [redacted] Also, where is the justification for the proposition that two only Steam Generators can remove the decay heat under "natural circulation" at RCS loop temperatures and "voidage" conditions, which are acceptable."

T.S. Page 3/4 4-6, refce. Cold Shutdown - Loops Not Filled: Add provision for ensuring loop water level is adequate for satisfactory inlet conditions to the RHR pumps.

T.S. Page 3/4 4-6, (proposed). Instrumentation: Reference 8, page Q212-56 ~~can remove the decay heat under "natural circulation" at RCS loop temperatures, and "voidage" conditions which are acceptable.~~ requires an audible low flow alarm in the RHR system to protect against the possible failure of one of the two isolation valves in the single RHR suction line from the R.C.S.

Section 3/4.4.3 PRESSURIZER

T.S. Page 3/4 4-9: System transient responses are dependent upon the pressurizer water levels at event initiation, and are based upon the values programmed by the Pressurizer Level Control System and used in the Accident Analyses of Reference (7).

Section 3/4.4.4 REACTOR COOLANT SYSTEM RELIEF VALVES

T.S. Page 3/4 4-10: 10 CFR 50 Appendix A requires two operable valves for the primary coolant system pressure boundary. The proposal to allow continued operation with such valves inoperable, contravenes the Regulations.

T.S. Page 3/4 4-36: Add paragraph on limiting RCP operation as this is an essential element of overpressure protection.

Section 3/4.5 EMERGENCY CORE COOLING SYSTEMS

T.S. Pages 3/4 5-1, 2, 2a, 2b, 3, 4, 4a, 4b, 8, 9, 10: Proposed Technical Specifications are at marked variance from the conditions described in the FSAR and the Safety Evaluation Reports.

[redacted] Technical Specifications provide for [redacted] ILITY [redacted] ly) [redacted] ed [redacted] $_{avg}$ ≥ 557°F [redacted] $_{avg}$ [redacted]

Conditions [redacted]d for the ECCS Systems fo[redacted] described in Reference 12, sections 6.3.4, page 6.6 to 6-8 and section 7.3.5 pages 7-1 and 7-2; [redacted] at [redacted]-47 [redacted]) a[redacted]-68. [redacted] as $\geq$ 1900 psig, < 1900 psig to 1000 psig/425°, and < 1000 psig/425° F to 425 psig/350° F; in MODE 4, one set of conditions is used. A pressurizer pressure of 1900 psig defines the pressure condition during cooldown at which the Pressurizer Pressure - Low Safety Injection signal, and the Low Pressure in the Main Steam Line Safety injection and Main Steam Isolation Valve signals can be manually blocked, and which manual blocking also causes the high negative pressure rate signal in the Main Steam Line (causing Main Steam Line Isolation), to be activated. At RCS pressures less than 1900 psig various combinations of ECCS equipment are effectively isolated from operation and the balance rendered OPERABLE to ensure sufficient protection against LOCA, Small Break LOCA, and small and large Main Steam Line Break events in conjunction with reduced Reactor Protection and Engineered Safeguards Features signal initiation capability; additionally, overpressure protection for the Primary Coolant Loop below 1000 psig/425° in MODE 3, and below 425 psig/350° F in MODES 4 and 5 (with loops filled), are parallel primary determinants.

The required status of the ECCS systems are briefly summarized:

Above 1900 psig (in MODES 1, 2 and 3) : All ECCS systems are OPERABLE. Between 1900 psig and 1000 psig/425° F; upper head injection isolation valves are closed and gagged, de-energized and tagged. Between 1000 psig/425° F and 425 psig/350° F (in MODE 3): Upper head injection isolation valves remain closed and gagged and de-energized; cold leg accumulator isolation valves are closed and breakers racked out, 1 centrifugal and 1 reciprocating charging pump and 2 safety injection pumps are isolated, and rendered inoperable by opening and locking the related circuit breakers. Below 425 psig/350° (In MODE 4) status of all ECCS systems remain unchanged, i.e., same (as for MODE 3, phase 3) with the exception [redacted] re-alignement to ECCS. [redacted]

1 [redacted]].

At this time, ~~we~~ RSB cannot accept the Proposed by the Licensee Technical Specification and which is based on the Standard Technical Specifications, in this matter. We will require changes to the McGuire Units 1 and 2 Technical Specifications unless the Licensee can explain the basis for these differences. Verbal discussion with the Licensee has not resulted in an acceptable response.

Relevant process conditions for the Cold Leg Injection and Upper Head Injection Accumulators are described in Reference 7, section 6, Page 6.3.2-1, and Reference 8, item 212.78, page Q 212-49b and item 212.97(15.4) page Q 212-71. The proposed values in the Technical Specifications on TS Pages 3/4 5-1 and 3/4 5-3 have been changed to conform with docket information to the above references. An inconsistency remains, in that UHI actuation pressure used in the FSAR analysis is 1300 psia (Reference 8, page Q 212-49b) and the proposed values in Reference 7 are less at 1220-1280 psig. Surveillance is also required on UHI water temperature.

Section 3/4.5.4 BORON INJECTION SYSTEM

T.S. Page 3/4 5-11: Applicability modes have be changed to reflect changes in TS section 3/4.1.1 BORATION CONTROL.

Section 3/4.5.5 REFUELING WATER STORAGE TANK

T.S. Page 3/4 5-12: Changes to Unit 1 reflect the reduced quantity of boron immediately available in the BIT tank in its reduction from 20,000 ppm to 2000 ppm. Changes for Unit 2 reflect complete dependence for Boron on the Refueling Water Storage Tank.

Section 3/4.7 PLANT SYSTEMS: AUXILIARY FEEDWATER SYSTEM

T.S. Page 3/4 7-4: The staff has deleted the provision for NON-OPERABILITY of the Steam-Turbine driven auxiliary feedwater pump at steam pressures of less than 900 psig. This is not in accord with current Accident Analyses and no justification has been provided.

T.S. Page 3/4 7-5(a)(Proposed): These additional OPERABILITY requirements are necessitated by the option to use reactor coolant loops for cooling in MODES 4 and 5.

T.S. Page 3/4 7-5(c)(Proposed): The basis for this additional OPERABILITY requirement concerning auxiliary feedwater condensate storage tank, is described earlier in this SER under the heading of TABLE 3.3-5.

T.S. 3/4 7-8 MAIN STEAM LINE ISOLATION VALVES: OPERABILITY is required in MODE 4 to limit the effects of a Main Steam Line Break as per the T.S. TABLE 3.3-3.

T.S. Page 3/4 7-8(a)(Proposed): ATMOSPHERIC DUMP VALVES; Surveillance for OPERABILITY should be provided, and especially for manual capability, since reference 14, section 5.3.3 specifies their use in natural circulation cooldown to MODE 5.

T.S. Page 3/4 7-10 COMPONENT COOLING WATER SYSTEM: Are two loops not required to be OPERABLE for MODE 5 (with RCS loops Filled).

Section 3/4.7.4 NUCLEAR SERVICE WATER SYSTEM

T.S. Page 3/4 7-11: Are 2 OPERABLE systems not required in MODE 5 (with RCS Loops Filled).

Section 3/4.7.5 STANDBYE NUCLEAR SERVICE WATER POND

T.S. Page 3/4 7-12: Reference 6, page 92-12(a) documents an acceptable maximum temp of 94° F. Accident Analysis, documents an auxiliary feedwater intake temperature of 70° F for evaluation (Provision can be made to reduce the minimum value of 70° F by an allowance for heat up in pumping, and in any heat exchangers, prior to entry into the Steam Generators. The Pond is still required in MODE 5 with the RC Loops Filled.

Section 3/4.8 ELECTRICAL POWER SYSTEM (TBD)

T.S. Page 3/4 8-1: Provision for only one emergency bus in MODE 5 (loops filled) is doubtful (TBD).

Section 3/4.9.1 BORON CONCENTRATION

T.S. Page 3/4 9-1: The changes are required by reference 11, page 15-2.

Section 3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION [In MODE 6 with water level less than 23 feet above ~~I~~ d.R.V. FLANGE]

T.S. Page 3/4 9-11, refce. Low Water Level: We cannot have 2 "independent" residual heat removal loops if we have only one emergency bus provided for under TS item 3.8.3.2 (page 3/4 8-17). Further, ~~my~~ RSB understanding is that a minimum level requirement of less than 23 feet is not permitted [TBD]. Further, are RHR pumps capable of being operated at these proposed low water level conditions; refer to our provision under T.S. Page 3/4 4-6.

CONCLUSION

Our review shows no close correlation between the docketed information for the Facilities and the proposed Technical Specifications, in a number of important areas. This position may derive from a desire by the licensee to conform to the Standard Technical Specifications which in themselves do not conform to the requirements of the Docket. The items of concern are primarily with the requirements for "Boration Control" and the "Emergency Core Cooling Systems" (ECCS); and from this there are significant repercussions on the Reactor Trip and ESF Actuation System, instrumentation requirements.

Concerning Boration Control: [redacted] the mai[redacted] ement [redacted] part [redacted] increased boron concent[redacted] ameliorate [redacted]ions [redacted]

Concerning the ECCS: [redacted] one centrifugal and one reciprocating charging pump, as well as two safety injection pumps, be rendered inoperable by electrical isolation. (Standard Technical Specifications do not require this action.) This electrical isolation of selected ECCS equipment together with restricted capability to initiate safety injection below 1900 psig, and 350°F (425 psig), has required more attention to the temperatures and pressures at which the electrical isolation are initiated; this has included both small break and large break LOCA analyses with parallel attention to the capability of a timely operator response to prevent uncovery of the core. Further, in addition, special [emergency] procedures are to be provided by the Licensee to enable small break accidents in an RHR loop to be adequately handled. The Plant [normal] Operating Procedure which defines ECCS alignments is not addressed in the current Standard Technical Specifications upon which the current McGuire Technical Specification has been developed but are required to be so included for their validation. Because of this, there are apparently substantive differences between the Safety Analyses of Record and the current McGuire Technical Specifications. We have proposed changes which would correct this deficiency.

Additionally, the standard Technical Specifications do not address the following significant issues:

a) Thermal Hydraulic parameters which are necessary inputs for Safety Analysis and which are partly ignored in the Technical Specifications. These are principally the programmed values for Operating Pressures, Temperatures, and Pressurize Water Levels, for the Reactor Coolant System. Although these values are controlled by Control Grade Systems, the resulting programmed settings ~~(value)~~ are safety related ~~[illegible]~~ and therefore require Technical Specification.

b) There is no Technical Specification on Control Rod Position in any Modes other than 1 or 2. There is Technical Specification on required Shut Down Margins - but this is a potential value - not the actual value; so that shutdown and/or control rods can be withdrawn, and provided actual reactivity is ≤ 0.99 and the desired shut down margin can actually be achieved by tripping the reactor, then the reactor is in conformance [with the Technical Specifications]. As a result of these circumstances, a number of substantial modifications are required to the Reactor Trip and ESF Actiuation System, Instrumentation requirements to ensure adequate Reactor Trip and ESF Actuation protection against a number of LOCA and NON-LOCA events, including reactivity excursions, during operation in Modes 3 through 5.

We also find, that during normal cooldown, the limiting Conditions of Operation for the "Reactor Coolant Loops and Coolant Circulation" for the McGuire T.S. (and the standard T.S.) are different to what is justified by the Docketed information, and in addition we find more detailed attention necessary to the Electrical Divisions being used, to guard against single failure considerations.

We do find a number of disparities between T.S. values and the Docket for some parameters, some omissions in the T.S., and a lack of justification for some of the T.S. values used.

It is likely that these proposed Technical Specifications could be reduced by detailed attention to Technical Specifications on Rod Control Insertion limits during Modes 3 through 5 as well as a review of the resulting reactivity excursions with RCS Borated to the actual concentration required to achieve cold shutdown - prior to cooldown in MODE 3. The information for this review is not generally currently available.

Summarily we find a number of substantive differences between the proposed Technical Specifications and the Docketed information which could ultimately have a significant impact on Public Health & Safety and we have proposed Technical Specifications to correct this situation.

LIST OF REFERENCES

1) Letter from H. B. Tucker (D.P.Co) to H. R. Denton (NRC) dated September 27, 1982 to the Subject of "McGuire Nuclear Station".

2) Memo from C. O. Thomas (SSPB) to Brian W. Sheron (RSB) on the Subject of "Proof and Review of McGuire - Units 1 and 2, Technical Specifications."

3) U.S. Nuclear Regulatory Commission, Final Safety Analysis Report, Volume 4, Duke Power Company, McGuire Nuclear Station, Units 1 and 2.

4) U.S. Nuclear Regulatory Commission, Final Safety Analysis Report, Volumes 5 and 6, Duke Power Company, McGuire Nuclear Station, Units 1 and 2, Rev. 45.

5) U.S. Nuclear Regulatory Commission, Final Safety Analysis Report, Volume 7, Duke Power Company, McGuire Nuclear Station, Units 1 and 2, Rev. 45.

6) U.S. Nuclear Regulatory Commission, Final Safety Analysis Report, Volume 8, Duke Power Company, McGuire Nuclear Station, Units 1 and 2, Rev. 45.

7) U.S. Nuclear Regulatory Commission, Final Safety Analysis Report, Volume 10, Duke Power Company, McGuire Nuclear Station, Units 1 and 2, Rev. 45.

8) U.S. Nuclear Regulatory Commission, Final Safety Analysis Report, Volume 11, Duke Power Company, McGuire Nuclear Station, Units 1 and 2, Rev. 45.

9) Deleted

10) U.S. Nuclear Regulatory Commission; Office of Nuclear Reactor Regulation; "Safety Evaluation Report; McGuire Nuclear Station Units 1 and 2, Duke Power Company," NUREG 0422, on Docket Nos. 50-369 and 50-370, March 1, 1978.

11) U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Safety Evaluation Report, McGuire Nuclear Station Units 1 and 2, Duke Power Company," NUREG 0422, Supp. 1, on Docket Nos. 50-369 and 50-370, May 1978.

12) U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Safety Evaluation Report, McGuire Nuclear Station, Units 1 and 2, Duke Power Company," NUREG 0422 Supp. No. 2, on Docket Nos. 50-369 and 50-370, March 1979.

13) U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Safety Evaluation Report, McGuire Nuclear Station, Units 1 and 2, Duke Power Company," NUREG 0422, Supp. No. 3, on Docket Nos. 50-369 amd 50-370, May, 1980.

14) U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Safety Evaluation Report, McGuire Nuclear Station, Units 1 and 2, Duke Power Company," NUREG 0422, Supp. No. 4, on Docket Nos. 50-369 and 50-370, January 1981.

15) U.S. Regulatory Commission, Office of Nuclear Reactor Regulation "Safety Evaluation Report, McGuire Nuclear Station Units 1 and 2, Duke Power Company," NUREG 0422, Supp. No. 5, on Docket Nos. 50-369 and 50-370, April 1981.

16) Memo from R. W. Houston to T. M. Novak on the subject of "Staff Review and Input to SER Supplement No. 6 for McGuire Nuclear Station Units 1 and 2."

17) Letter from H. B. Tucker (D.P.Co) to H. R. Denton (NRC) on the subject of McGuire Nuclear Station, Units 1 and 2, filing amendment No. 71 to its Application for License for the McGuire Nuclear Station and submitting Revision 45 to the Final Safety Analysis Report. Dated February 16, 1983.

18) Letter from W. O. Parker (D.P.Co) to H.R. Denton (NRC), dated Oct. 8, 1981 on the subject of McGuire Nuclear Station, Unit 1 and submitting copies of Report identified as "Westinghouse Reactor Protection System/Engineered Safety Features Actuation System Setpoint Methodology, Duke Power Company, McGuire Unit 1", by C. R. Tuley et al. and dated April, 1981, published by Westinghouse Electric, Nuclear Energy Systems, PROPRIETARY.

19) Westinghouse Electric Corporation, PWR Systems Division "Westinghouse Emergency Core Cooling System - Plant sensitivity studies, WCAP - 8356.

20) U.S. Nuclear Regulatory Commission, Final Safety Analysis Report, Volume 4, Duke Power Company, McGuire Nuclear Station, Units 1 and 2, Rev. 45.

TABLE 1

SECTIONS REVIEWED BY REACTOR SYSTEMS BRANCH

SECTION	PAGE

SECTION		PAGE

SECTION PAGE

TABLE 2

TECHNICAL SPECIFICATION PAGES AFFECTED

The following pages of the Technical Specifications are affected by this review:

T.S. Pages	2-1, 2
TABLE 2.2-1, T.S. Pages	2-5 2-6 2-7
T.S. Pages	3/4 1-1 3/4 1-2 3/4 1-2a proposed 3/4 1-6 3/4 1-7 3/4 1-8 3/4 1-9 3/4 1-10 3/4 1-11 3/4 1-12 3/4 1-13 3/4 1-13a) 3/4 1-20a) 3/4 1-21
T.S. Pages	3/4 2-15 16

TABLE 3.3-1, T.S. Pages	3/4	3-2 3-3 3-4 3-5 3-6
TABLE 3.3-2, T.S. Pages	3/4	3-9 3-10
TABLE 3.3-3, T.S. Pages	3/4	3-16 3-18 3-19 3-21 3-22 3-23
TABLE 3.3-4, T.S. Pages	3/4	3-25 3-26 3-27 3-28 3-29
TABLE 3.3-5, T.S. Pages	3/4	3-30 3-31 3-32 3-33
T.S. Pages	3/4	4-1 4-2 4-3 4-4 4-5 4-6 4-6(a) proposed 4-9 4-10 4-36

T.S. Pages	3/4 5-1, 5-2 5-2a) proposed 5-2b) proposed 5-3 5-4 5-4a) proposed 5-4b) proposed 5-8, 5-9, 5-10 5-11 5-12
T.S. Pages	3/4 7-4 7-5(a) proposed 7-5(c) proposed 7-8 7-8(a) proposed 7-10 7-11 7-12
T.S. Page	3/4 8-1
T.S. Page	3/4 9-1
T.S. Page	3/4 9-11

Attachment 3

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS UNDER ANTICIPATED OPERATIONAL OCCURRENCES

REACTOR CORE

under Anticipated Operational Occurrences

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for four ~~and three~~ loop operation, ~~respectively~~. TBL

APPLICABILITY: MODES 1 ~~and~~ 2 and 3

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

under Anticipated Operational Occurrences

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

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

ACTION:

MODES 1 ~~and~~ 2 and 3

appropriate T.S. limits for NORMAL OPERATION

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

MODES ~~3~~, 4 and 5

appropriate T.S. limits for NORMAL OPERATION

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

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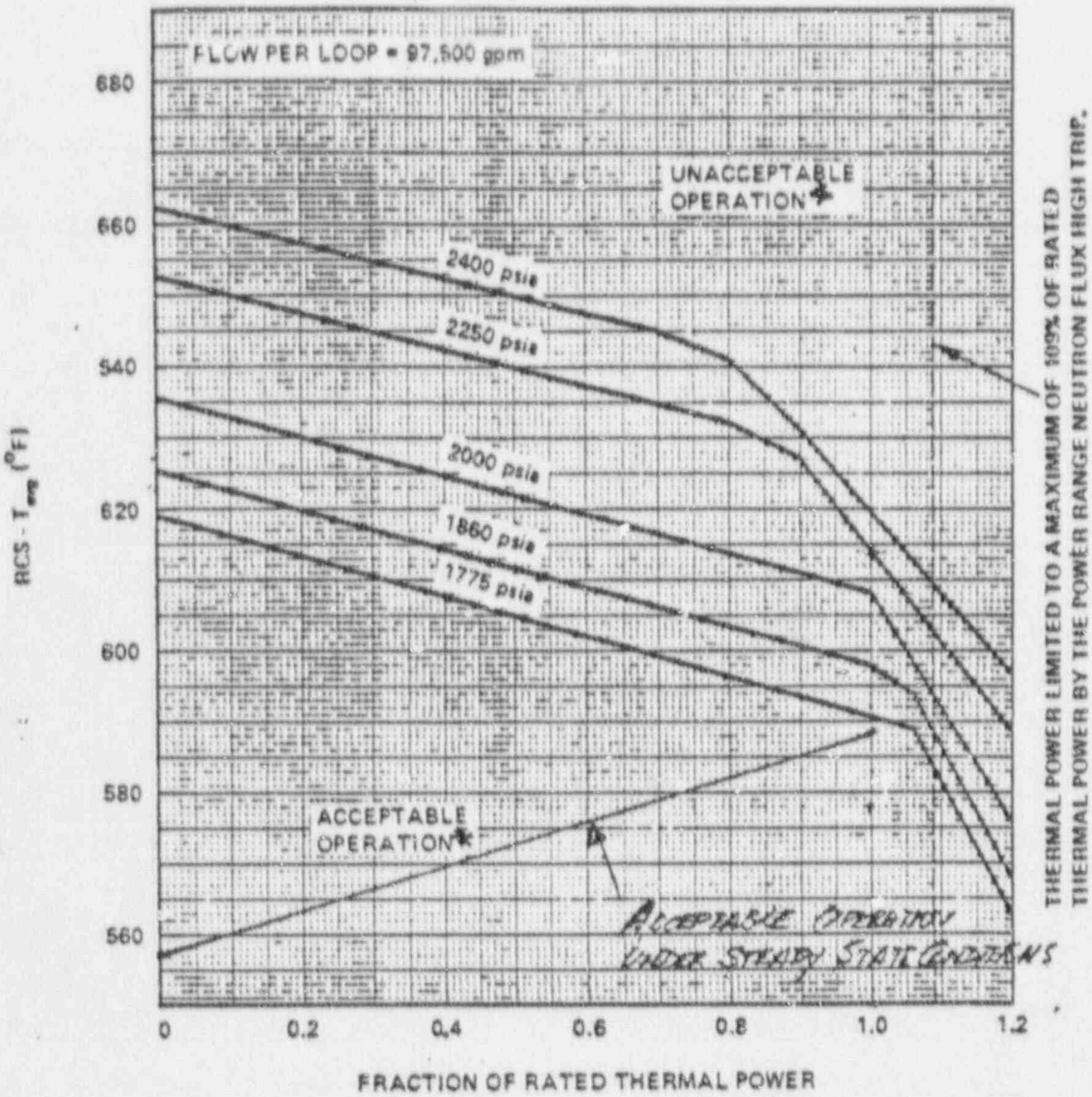

FIGURE 2.1-1
REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

* Under Anticipated Operational / Occurrences G

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, high Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	$\geq$ 1945 psig	$\geq$ 1935 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Low Reactor Coolant Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% of design flow per loop*

*Design flow is 97,500 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	$\geq$ 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to $\geq$ 54.9% of span at 100% of RATED THERMAL POWER.	$\geq$ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing to 53.9% of span at 100% of RATED THERMAL POWER. TBD
14. Undervoltage-Reactor Coolant Pumps	$\geq$ 5082 volts-each bus	$\geq$ 5016 volts-each bus
15. Underfrequency-Reactor Coolant Pumps	$\geq$ 56.4 Hz - each bus	$\geq$ 55.9 Hz - each bus
16. Turbine Trip		
a. Low Trip System Pressure	$\geq$ 45 psig	$\geq$ 42 psig
b. Turbine Stop Valve Closure	$\geq$ 1% open	$\geq$ 1% open
17. Safety Injection Input from ESF	N.A.	N.A.

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
18. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
b. Low Power Reactor Trips Block, P-7		
1) P-10 Input	10% of RATED THERMAL POWER	$\geq$ 9%, $\leq$ 11% of RATED THERMAL POWER
2) P-13 Input	$\leq$ 10% RTP Turbine Impulse Pressure Equivalent	$\leq$ 11% RTP Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	$\leq$ 48% of RATED THERMAL POWER	$\leq$ 49% of RATED THERMAL POWER
d. Low Setpoint Power Range Neutron Flux, P-10	10% of RATED THERMAL POWER	$\geq$ 9%, $\leq$ 11% of RATED THERMAL POWER
e. Turbine Impulse Chamber Pressure, P-13	$\leq$ 10% RTP Turbine Impulse Pressure Equivalent	$\leq$ 11% RTP Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.
21 Reactor Coolant Pump Breaker Position Trip	TBD	TBD

(Refer SAR 15.2 5.1)

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3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} ~~[illegible]~~ Programmed No Load Tavg [557°F]

Tavg 553

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% delta k/k for four loop operation.

APPLICABILITY: ~~MODES 1, 2*, 3, and 4.~~ **

ACTION:

With the SHUTDOWN MARGIN less than 1.6% delta k/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% delta k/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e., below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

Reword unnecessary

*See Special Test Exception 3.10.1.

** Before Any Cooldown from ~~[illegible]~~ Programmed No Load Avge Temp [557.0°F)

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SURVEILLANCE REQUIREMENTS (Continued)

Rework — as necessary

e. When in MODE 3** ~~or 4~~, at least once per 24 hours by consideration of the following factors:

1) Reactor coolant system boron concentration,

2) Control rod position,

3) Reactor coolant system average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN: TAVG < PROGRAMMED NO LOAD TAVG [ ]

and ≥ 200°F

LIMITING CONDITION FOR OPERATION

The boron concentration in the RCS shall be increased to a value which will give a SHUTDOWN MARGIN of ~~[illegible]~~ 1% delta k/k at 200 °F

APPLICABILITY MODES 3^{**}, 4, ~~[illegible]~~ and 5.

ACTION

(TBD)

SURVEILLANCE REQUIREMENTS

(TBD)

** During Cooldown at ~~[illegible]~~ Tav. < Programmed No Load Average Temperature.

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MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) at zero Thermal Power shall be ~~greater than or~~ equal to 551°F, and shall not exceed this value in MODE 3

APPLICABILITY: MODES 1 and 2#*, 3.

TBD — should then be 557°F

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 551°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be ~~greater than or~~ equal to 551°F: — should then be 557°F (TBD)

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 561°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

#With K_{eff} greater than or equal to 1.0.
*See Special Test Exception 3.10.3.

PL

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3/4.1.2 BORATION SYSTEMS

FLOW PATH - ~~SHUTDOWN~~ STANDBY** AND REFUELING

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from a boric acid tank via a boric acid transfer pump and a centrifugal charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.5a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a centrifugal charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b. is OPERABLE.

APPLICABILITY: MODES 3**, 4, 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path is greater than or equal to 65°F when a flow path from the boric acid tanks is used, and

b. At least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

** At less than 1000 psig/425 °F

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - ~~OPERATING~~ POWER OPERATION, START UP, STANDBY, COOLDOWN TO 1000 psig/425°F

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LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from a boric acid tank via a boric acid transfer pump and a charging pump to the Reactor Coolant System, and

b. Two flow paths from the refueling water storage tank via charging pumps to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3, ~~and 4#~~ **

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is greater than or equal to 65°F when it is a required water source;

b. At least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal; and

d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the Reactor Coolant System.

~~#Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F.~~

** Down to 1000 psig/425°F

CHARGING PUMP - ~~SHUTDOWN~~ STANDBY/REFUELING

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LIMITING CONDITION FOR OPERATION

3.1.2.3 One centrifugal charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES ~~5~~ 3, 4 and 6.

ACTION:

With no charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, on recirculation flow, a differential pressure across the pump of greater than or equal to 2380 psid is developed when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All centrifugal charging pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable at least once per 31 days, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position or by verifying the discharge of each charging pump has been isolated from the Reactor Coolant System by at least two isolation valves with ~~power removed~~ from the valve operators. — being open, locked & tagged

by being opened, locked and tagged

breakers

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CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

centrifugal

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3,xx and 4#

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE by verifying, on recirculation flow, a differential pressure across each pump of greater than or equal to 2380 psid is developed when tested pursuant to Specification 4.0.5.

4.1.2.4.2 All centrifugal charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 31 days whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F by verifying that the motor circuit breakers are secured in the open position or by verifying the discharge of each charging pump has been isolated from the Reactor Coolant System by at least two isolation valves with power removed from the valve operators.

#A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F.

what about Reciprocating Pump

xx If ≥ 1000 psig, 425°F

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BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System and at least one associated heat tracing system with:

1) A minimum contained borated water volume of 5100 gallons,

2) Between 7000 and 7700 ppm of boron, and

3) A minimum solution temperature of 65°F.

4) Level Detection system at a minimum level.

b. The refueling water storage tank with:

1) A minimum contained borated water volume of 26,000 gallons,

2) A minimum boron concentration of 2000 ppm, and

3) A minimum solution temperature of 70°F.

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.

4) Verify minimum level indication

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 70°F.

c) Level Detection (2 Indicators/Tank) and related high, low and low-low level alarms

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BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A Boric Acid Storage System and at least one associated heat tracing system with:

1) A minimum contained borated water volume of 19,500 gallons,

2) Between 7000 and 7700 ppm of boron, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1) A contained borated water volume of at least 372,100 gallons,

2) Between 2000 and 2100 ppm of boron,

3) A minimum solution temperature of 70°F, and

4) A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the Boric Acid Storage System inoperable and being used as one of the above required borated water sources, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water volume of the water source, and

3) Verifying the Boric Acid Storage System solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 70°F or greater than 100°F.

3/4.1 REACTIVITY CONTROL SYSTEMS

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~~REFUELING OPERATIONS~~

3/4.1.4 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE and operating with Alarm Setpoints at 0.5 decade above steady-state count rate, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE ~~6~~, 3, 4, 5, 6

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.4 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

(MODES. 3 — 5)

A statement is required here on any necessary positions as there is a prospect for simplifying the T.S. in so far as ~~necessary details~~ ~~for~~ Accident Protection is concerned

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[TBD] There should be a check on the Insertion Limit Stops.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figures 3.1-1 and 3.1-2.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the above figures, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

Insertion Limit Stops should be checked

*See Special Test Exceptions 3.10.2 and 3.10.3.
#With K_{eff} greater than or equal to 1.0.

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3/4.2.5 DNB PARAMETERS AND RCS PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}, and

b. Pressurizer Pressure.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

Check

TABLE 3.2-1

DNB PARAMETERS (& RCS Pressure)

PARAMETER	LIMITS Four Loops In Operation	Three Loops In Operation	Rated Thermal Power level
Reactor Coolant System T_{avg}	≤ 593°F 588.1	(**)	100%
Pressurizer Pressure	~~≥ 2230 psia*~~ 2235 psig ± TBD	(**)	100%
Reactor Coolant System Tavg	TBD 557.0 = 551°F ± TBD	(**)	0
Pressurizer Pressure	= 2235 psig ± TBD	(**)	0%

Presumably these are Control Room Instrument Values

: they must be the nominal setpoint values for Programmed Tavg at NO LOAD and RATED POWER levels and the nominal system pressure of 2235 psig, which have been the starting basis for all ACCIDENT EVALUATIONS

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

**These values left blank pending NRC approval of three loop operation.

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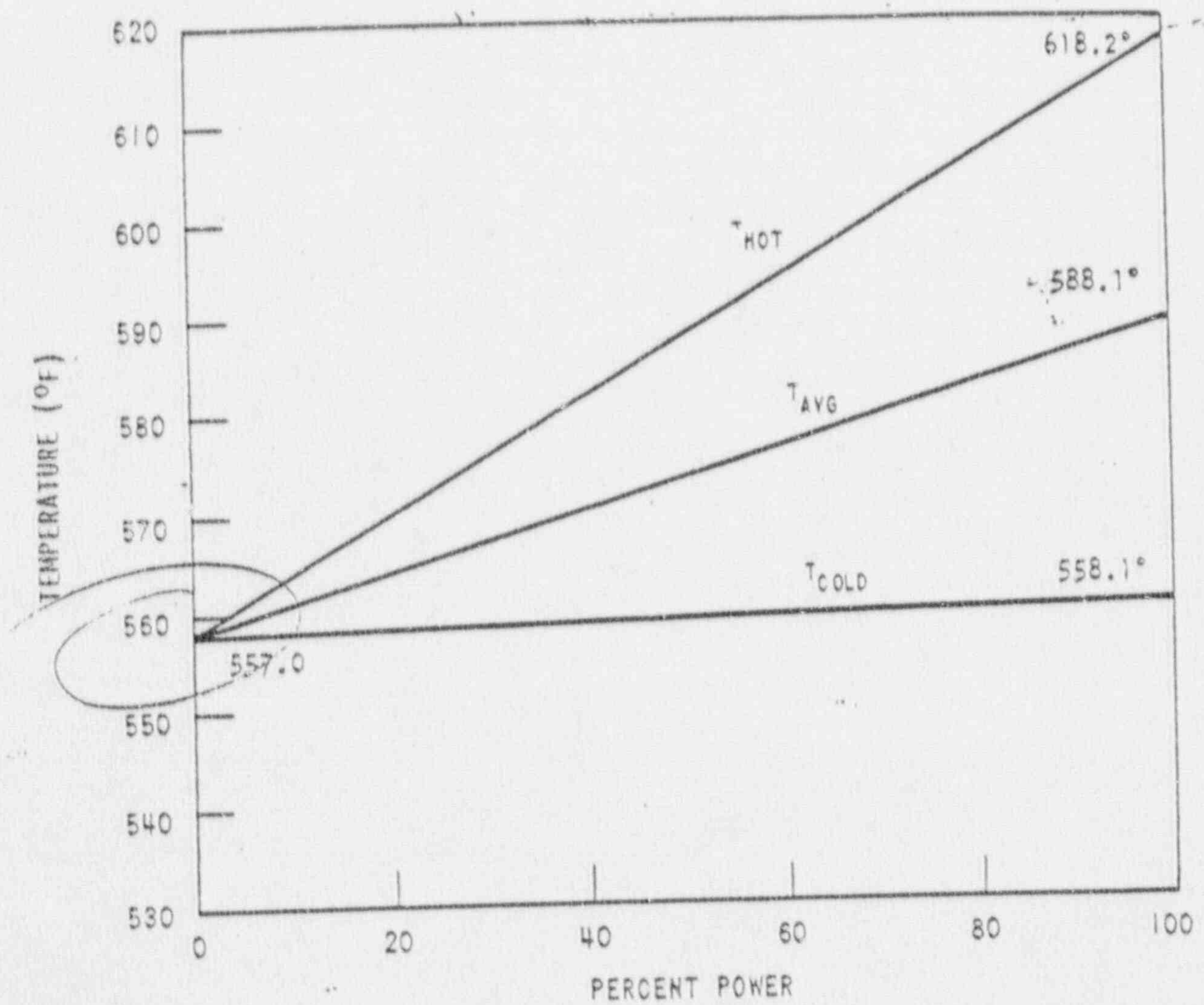
There is need to clarify apparent ambiguities on the 0% RTP level.

The T.S says approx 551°F, Fig 5.3.3-1 of FSAR says 557.0°F

6234-70



RELATIONSHIP BETWEEN REACTOR COOLANT SYSTEM TEMPERATURE AND POWER

McGUIRE NUCLEAR STATION

Figure 5.3.3-1

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux - High Setpoint	4	2	3	1, 2	2#
Low Setpoint	4	2	3	1###, 2, 3@	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2, 3	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2 ~~3~~	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2, 3,	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3*, 4*, 5*	10
c. Shutdown	2	0	1	3, 4, and 5	5
7. Overtemperature ΔT					
Four Loop Operation	4	2	3	1, 2	6#
Three Loop Operation	(**)	(**)	(**)	(**)	(**)

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TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Overpower ΔT					
Four Loop Operation	4	2	3	1, 2	6#
Three Loop Operation	(**)	(**)	(**)	(**)	(**)
9. Pressurizer Pressure-Low	4	2	3	1, 2, 3**	6#
10. Pressurizer Pressure--High	4	2	3	1, 2, 3.	6#
11. Pressurizer Water Level--High	3	2	2	1, 2, 3, 4, 5	#
12. Low Reactor Coolant Flow					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1,	7#
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1,	7#
13. Steam Generator Water Level--Low-Low	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2, 3, 4, 5	7#

** Prior to commencement of cooldown from 557°F

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TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
14. Undervoltage-Reactor Coolant Pumps (above P-7)	4-1/bus	2	3	1	6#
15. Underfrequency-Reactor Coolant Pumps (above P-7)	4-1/bus	2	3	1	6#
16. Turbine Trip					
a. Low Fluid Oil Pressure	3	2	2	1	7#
b. Turbine Stop Valve Closure	4	4	1	1	11#
17. Safety Injection Input from ESF	2	1	2	1, 2, 3, 4.	9
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	8
or					
P-13 Input	2	1	2	1	8
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Low Setpoint Power Range Neutron Flux, P-10	4	2	3	1, 2	8
e. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
19. Reactor Trip Breakers	2	1	2	1, 2	9
	2	1	2	3*, 4*, 5*	10
20. Automatic Trip and Interlock Logic	2	1	2	1, 2	9
	2	1	2	3*, 4*, 5*	10
21 Reactor Coolant Pump Breaker Position Trip /				1, 2, 3, [4, 5] — TBD	

P Should provide for trip in the Reactor circuit that no RCP is operating in 2, 3, 4 & 5 during cooldown. In addition to protection against failure of one pump only at MODES 1, 2 & 3.

NOTE: NO RCPS IN MODE 3 IS NATURAL CIRCULATION WHICH IS AN EMERGENCY CONDITION

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TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the Reactor Trip System breakers in the closed position, the Control Rod Drive System capable of rod withdrawal.

what about SD Rods

**Values left blank pending NRC approval of three loop operation.

#The provisions of Specification 3.0.4 are not applicable.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	≤ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second* #C
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	≤ 4.0 seconds*, (6 secs in [illegible])
8. Overpower ΔT	~~N.A.~~ Provide [TBD. ≤ 6 secs]
9. Pressurizer Pressure--Low	≤ 2.0 seconds
10. Pressurizer Pressure--High	≤ 2.0 seconds
11. Pressurizer Water Level--High	~~N.A.~~ Provide. [mentioned in SAR 15-A-13 (18)]

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

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TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Low Reactor Coolant Flow	
a. Single Loop (Above P-8)	$\leq$ 1.0 second
b. Two Loops (Above P-7 and below P-8)	$\leq$ 1.0 second
13. Steam Generator Water Level--Low-Low	$\leq$ 2.0 seconds
14. Undervoltage-Reactor Coolant Pumps	< 1.5 seconds
15. Underfrequency-Reactor Coolant Pumps	< 0.6 second
16. Turbine Trip	
a. Low Fluid Oil Pressure	~~N.A.~~ 1.0 second
b. Turbine Stop Valve Closure	~~N.A.~~ 1.0 second
17. Safety Injection Input from ESF	~~N.A.~~ Necessary [Refer Table 3.3-1 Item 1)
18. Reactor Trip System Interlocks	N.A.
19. Reactor Trip Breakers	N.A.
20. Automatic Trip and Interlock Logic	N.A.
21. Reactor Coolant Pump Breaker Position Trip	Necessary

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TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection, Reactor Trip, Feedwater Isolation,*** Component Cooling Water, Start Diesel Generators, and Nuclear Service Water					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High	3	2	2	1, 2, 3, 4	15*
d. Pressurizer Pressure - Low Safety Inj.	4	2	3	1, 2, 3#	19*
e. Steam Line Pressure-Low					
Four Loops Operating	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3# No	15*
Three Loops Operating	(**)	(**)	(**)	(**)	(**)

Ref Q212-47c of FSAR, Vol III

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (continued)					
b. Phase "B" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure--High-High	4	2	3	1, 2, 3 4.7B	16
c. Purge and Exhaust Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	17
3) Safety Injection	See item 1. above for all Safety Injection initiating functions and requirements				

TBD x) Radioact In Containment

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FSAR Vol II, pg. Q212-47e, item 24.

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation					
a. Manual Initiation					
1) System	2	1	2	1, 2, 3 4*	22
2) Individual	1/steam line	1/steam line	1/operating steam line	1, 2, 3 4*	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3 4*	21
c. Containment Pressure--High-High	4	2	3	1, 2, 3	16
d. Negative Steam Line Pressure Rate - High					
Four Loops Operating	3/steam line	2/steam line in any steam line	2/steam line	▲ 3, 4	15*
Three Loops Operating	(**)	(**)	(**)	(**)	(**)
e. Steam Line Pressure - Low					
Four Loops Operating	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	15*
Three Loops Operating	(**)	(**)	(**)	(**)	(**)

Ref. 15.2-3.5 4(a)

▲ This Is Not Operable Above P-11 [Interlock] · [clock]

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Auxiliary Feedwater					
a. Manual Initiation	2	1	2	1, 2, 3	22
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Stm. Gen. Water Level-Low-Low					
1) Start Motor-Driven Pumps	4/stm. gen.	2/stm. gen. in any operating stm gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	19*
2) Start Turbine-Driven Pump	4/stm. gen.	2/stm. gen. in any 2 operating stm. gen.	3/stm. gen in each operating stm. gen	1, 2, 3	19*
d. Auxiliary Feedwater Suction Pressure - Low	2/pump	2/pump	2/pump	1, 2, 3	24
e. Safety Injection Start Motor-Driven Pumps and Turbine-Driven Pump	See Item 1. above for all Safety Injection initiating functions and requirements				

Realignment of Suction Valves on condensate storage supply

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Auxiliary Feedwater (continued)					
f. Station Blackout Start Motor-Driven Pumps and Turbine-Driven Pump	6-3/Bus	2/Bus Either Bus	2/Bus	1, 2, 3	19*
g. Trip of Main Feedwater Pumps Start Motor-Driven Pumps	2/pump	1/pump	1/pump	1, 2#	14
8. Automatic Switchover to Recirculation					
RWST Level	3	2	2	1, 2, 3	16
9. Loss of Power					
4 kV Emergency Bus Undervoltage-Grid Degraded Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	19*
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	20
c. Reactor Trip, P-4	2	2	2	1, 2, 3	22
d. Steam Generator Level, P-14	3/stm gen.	2/stm gen. in any operating stm gen.	2/stm gen. in each operating stm gen.	1, 2, 3	15*

What Trip

FSAR 10.4-14 says all pumps

All Pumps

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TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be manually blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

~~##Trip function may be blocked in this MODE below the P-12 (Low-Low T_{avg} Interlock) Setpoint.~~ By RSB

~~###The channel(s) associated with the protective functions derived from the out-of-service reactor coolant loop shall be placed in the tripped mode.~~ By LQB

*The provisions of Specification 3.0.4 are not applicable.

**These values left blank pending NRC approval of three loop operation.

xxx Close all ~~feedwater~~ control, and discharge valves; trip main ~~feedwater~~ pumps; close containment isolation valves

ACTION STATEMENTS

ACTION 14 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 16 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

xxxx Trip function automatically blocked above P-11 and may be blocked (manually) below P-11 when [illegible]

Steam Line Pressure-[illegible] is not manually blocked

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Safety Injection, Reactor Trip, Feedwater Isolation,* Component Cooling Water, Start Diesel Generators, and Nuclear Service Water.		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High	≤ 1.1 psig	≤ 1.2 psig
d. Pressurizer Pressure--Low Safety Injection	≥ 1845 psig	≥ 1835 psig
e. Steam Line Pressure - Low	≥ 585 psig	≥ 565 psig
2. Containment Spray		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	≤ 2.9 psig	≤ 3.0 psig

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	
b. Phase "B" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure--High-High	$\leq$ 2.9 psig	$\leq$ 3.0 psig
c. Purge and Exhaust Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	

TBD Radioactivity in Containment

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	$\leq$ 2.9 psig	$\leq$ 3.0 psig
d. Negative Steam Line Pressure Rate - High	~~-100~~ psi/sec -110	~~-120~~ psi/sec -100
e. Steam Line Pressure - Low	$\geq$ 585 psig	$\geq$ 565 psig
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water level--High-High (P-14)	< 82% of narrow range instrument span each steam generator	< 83% of narrow range instrument span each steam generator
6. Containment Pressure Control System		
a. Start Permissive	$\leq$ 0.25 psid	$\leq$ 0.25 psid
b. Termination	$\leq$ 0.25 psid	$\leq$ 0.25 psid

TBD. [illegible] ≤ means this context. More or less negative

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CHECK AGAIN

SD are suffice not to start according to SAP TBD

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Auxiliary Feedwater		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level--Low-Low		
1) Start Motor-Driven Pumps	> 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to > 54.9% of span at 100% of RATED THERMAL POWER.	> 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to > 53.9% of span at 100% of RATED THERMAL POWER.
2) Start Turbine-Driven Pumps	> 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to > 54.9% of span at 100% of RATED THERMAL POWER.	> 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to > 53.9% of span at 100% of RATED THERMAL POWER.
d. Auxiliary Feedwater Suction Pressure - Low	≥ 2 psig	≥ 1 psig
e. Safety Injection - Start Motor-Driven Pumps and Turbine-Driven Pump	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	
f. Station Blackout - Start Motor-Driven Pumps and Turbine-Driven Pump	3464 ± 173 volts with a 8.5 ± 0.5 second time delay	≥ 3200 volts
g) Trip of Main Feedwater Pumps - Start Motor-Driven Pumps	N.A.	N.A.

Re-aligns Water Resource
Does not re-start Pumps
Describe Re-alignment

TBD
Explain ... SG ...

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Automatic Switchover to Recirculation		
RWST Level	≥ 90 inches	≥ 80 inches
9. Loss of Power		
4 kV Emergency Bus Undervoltage-Grid Degraded Voltage	3464 ± 173 volts with a 8.5 ± 0.5 second time delay	≥ 3200 volts
10. Engineered Safety Features Actuation System Interlocks	1870 psig ~~1900 psig~~	1880 psig ~~1890 psig~~
a. Pressurizer Pressure, P-11	≤ 1955 psig	≤ 1965 psig
b. T_{avg}, P-12	≥ 553°F	≥ 551°F
c. Reactor Trip, P-4	N.A.	N.A.
d. Steam Generator Level, P-14	See Item 5. above for all Trip Setpoints and Allowable Values.	

Ref
Q&A
FSAR 11
Pg 212-47c/e

TBD - Is this authorized
Thus ?le
aut. SI on
SLP-low

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1900 1+

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TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Containment Isolation	
Phase "A" Isolation	N.A.
Phase "B" Isolation	N.A.
Purge and Exhaust Isolation	N.A.
d. Steam Line Isolation	N.A.
e. Feedwater Isolation	N.A.
f. Auxiliary Feedwater	N.A.
g. Nuclear Service Water	N.A.
h. Component Cooling Water	N.A.
i. Reactor Trip (from SI)	N.A.
j. Start Diesel Generators	N.A.
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq$ ~~21~~ 25 (1) Ref FSAR Pg 15 3-3
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 9
d. Containment Isolation-Phase "A" (2)	$\leq 18^{(3)}/28^{(4)}$
e. Containment Purge and Exhaust Isolation	N.A.
f. Auxiliary Feedwater	N.A.
g. Nuclear Service Water	$\leq 65^{(3)}/76^{(4)}$
h. Component Cooling Water	$\leq 65^{(3)(2)}/76^{(4)(2)}$
i. Start Diesel Generators	$\leq$ ~~12~~ 10

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low Low	
a. Safety Injection (ECCS)	≤ ~~27~~ 25(1)/12(3) Ref SAR Pag 15.3.3
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation (from SI)	≤ 9 TBD
d. Containment Isolation-Phase "A"(2) (from SI)	≤ 18(3)/28(4)
e. Containment Purge and Exhaust Isolation (from SI)	N.A.
f. Auxiliary Feedwater (from SI)	N.A.
g. Nuclear Service Water System (from SI)	≤ 76(1)/65(3)
h. Component Cooling Water (from SI)	≤ 76(1)/65(2)(3)
i. Start Diesel Generators (from SI)	≤ ~~21~~ 10 Ref SAR 15.4-10 2nd Para consistent with
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 12(3)/22(4)
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation (from SI)	≤ 9 TBD
d. Containment Isolation-Phase "A"(2) (from SI)	≤ 18(3)/28(4) TBD
e. Containment Purge and Exhaust Isolation (from SI)	N.A.
f. Auxiliary Feedwater Pumps (from SI)	N.A.
g. Nuclear Service Water (from SI)	≤ 65(3)/76(4)
h. Steam Line Isolation	≤ 9
i. Component Cooling Water (from SI)	≤ 65(2)(3)/76(2)(4)
j. Start Diesel Generators (from SI)	≤ ~~12~~ 10 Ref SAR 15.4-10 2nd Para
5. Containment Pressure--High-High	
a. Containment Spray	≤ 45
b. Containment Isolation-Phase "B"	N.A.
c. Steam Line Isolation	≤ 9
6. Steam Generator Water Level--High-High	
a. Turbine Trip	N.A.
b. Feedwater Isolation	≤ 13

TBD - what does (from SI) mean. From SI signal [illegible]

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
7. Steam Generator Water Level - Low-Low	
a. Motor-driven Auxiliary Feedwater Pumps	≤ 60 /(5)
b. Turbine-driven Auxiliary Feedwater Pumps	≤ 60 (5)
8. Negative Steam Line Pressure Rate - High	
Steam Line Isolation	≤ 9
9. Start Permissive	
Containment Pressure Control System	N.A.
10. Termination	
Containment Pressure Control System	N.A.
11. Auxiliary Feedwater Suction Pressure - Low	
Auxiliary Feedwater Pumps (Suction Supply Automatic Realignment)	≤ 13
12. RWST Level	
Automatic Switchover to Recirculation	≤ 60
13. Station Blackout	
a. Start Motor-Driven Auxiliary Feedwater Pumps	≤ 60 /(5)
b. Start Turbine-Driven Auxiliary Feedwater Pump	≤ 60 (5)
14. Trip of Main Feedwater Pumps	
Start Motor-Driven Auxiliary Feedwater Pumps	≤ 60/(5)
15. Loss of Power	
4 kV Emergency Bus Undervoltage-Grid Degraded Voltage	≤ 11
Note: Response time for Motor-Driven Auxiliary Feedwater Pumps on All SI Signal Starts	≤ 60 /(5)

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TABLE 3.3-5 (Continued)

TABLE NOTATION

TBD

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish Safety Injection path and attainment of discharge pressure for centrifugal charging pumps, Safety Injection and RHR pumps.
LPSI

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

2.b	$\leq 30^{(3)}/40^{(4)}$	what are these valves, what response time do they impact TBD
3.b	$\leq 30^{(3)}$	
4.b	$\leq 30^{(3)}/40^{(4)}$	

(3) Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish Safety Injection path and attainment of discharge pressure for centrifugal charging pumps and safety injection pumps (TBD) Ref SIR 15.4-7 table E) Min Cap
and LPSI [RHR] pumps

(4) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish Safety Injection path and attainment of discharge pressure for centrifugal charging pumps and safety injection pumps (TBD)
and LPSI (RHR) pumps

(5) must show total time allowing for necessary use of STANDBY NUCLEAR SERVICE WATER POND

This significant information requires substantive clarification before it can be accepted

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3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENT

4.4.1.1 The above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exception 3.10.4.

~~** At No Load Programmed Tavg~~

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HOT STANDBY **

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least ~~two~~ four of the reactor coolant loops listed below shall be OPERABLE and ~~at least one of the above reactor coolant loops~~ shall be in operation:* four loops

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump, and

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours. ?

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

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 12% of narrow range instrument at least once per 12 hours.

No Basis for this value

~~4.4.1.2.3 At least one reactor coolant loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.~~

*All reactor coolant pumps may be de-energized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

** Down to 425 psig / 350°F

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REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the reactor coolant and/or residual heat removal (RHR) loops listed below shall be OPERABLE and at least one of the above reactor coolant and/or RHR loops shall be in operation:** —

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

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

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,* (including related auxiliary feedwater pump)

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

e. RHR Loop A, and

f. RHR Loop B.

Additionally: one RCS loop must always be OPERABLE whenever RHR loops are inoperable

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

ACTION:

a. With less than the above required reactor coolant and/or RHR loops OPERABLE, immediately initiate corrective ACTION to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

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

STET

*A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 300°F unless: (1) the pressurizer water volume is less than 1600 cubic feet, or (2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

Since it is an overpress protn reqmt

**All reactor coolant pumps and RHR pumps may be de-energized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 12% at least once per 12 hours. TBD BATH

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.4.4 The related auxiliary feedwater subsystem shall be determined OPERABLE as per the requirements of T.S 3.7.1.2 and [3.7.1.2-a as applicable]

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COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE#, or
One Reactor Coolant loop and its ~~associated~~ steam generator
b. The secondary side water level of at least two steam generators ~~energize~~ shall be greater than 12%. from a separate emergency ~~power~~ division

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator level, immediately initiate corrective ACTION to return the inoperable RHR loop to OPERABLE status or restore the required steam generator level as soon as possible.

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

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 300°F unless: (1) the pressurizer water level is less than 92% (1600 cubic feet), or (2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

* The RHR pump may be de-energized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) loops shall be OPERABLE# and at least one RHR loop shall be in operation.*

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required RHR loops OPERABLE, immediately initiate corrective ACTION to return the required RHR loops to OPERABLE status as soon as possible.

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

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

The level of water in the loops shall ensure adequate NPSH at, and satisfactory flow conditions at inlet to, the RHR Pumps and this shall be determined once every 8 hrs and

One RHR loop may be inoperable for up to _ hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

* The RHR pump may be de-energized for up to hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

[after every change of level.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN — REFUELING

~~3~~ — INSTRUMENTATION — RHR LOW FLOW ALARM

LIMITING CONDITION FOR OPERATION

As a minimum, two RHR flow meters shall be operable and operating with alarm set points, each with continuous visual and audible indication in the control room ~~and must be [illegible] in the~~

Proposed page 3/4 4-6(a)

Proposed page 3/4 4-6(a)

TBD

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REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

at the programmed values of ---% of Rated core power and ---% of Rated thermal power

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with ~~a~~ water levels of ~~less than or equal to 92% (1600 cubic feet)~~, and at least two groups of pressurizer heaters each having a capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c) Insert values required

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water level shall be determined to be within its at its programmed values ~~limit~~ at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually transferring power from the normal to the emergency power supply and energizing the heaters.

[illegible]
with one block valve; also with no PORVs operable

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3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

consistent with 10 CFR 50 App. A

ACTION:

a. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s) or close the PORV and remove power from its associated solenoid valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the valve through one complete cycle of full travel.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION a. in Specification 3.4.4.

4.4.4.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by:

a. Manually transferring motive and control power from the normal to the emergency power supply, and

b. Operating the valves through a complete cycle of full travel.

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LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with a lift setting of less than or equal to 400 psig, or

b. The Reactor Coolant System (RCS) depressurized in Modes 5 & 6 with an RCS vent of greater than or equal to 4.5 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 300°F, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 4.5 square inch vent(s) within the next 8 hours.

b. With both PORVs inoperable, depressurize and vent the RCS through at least a 4.5 square inch vent(s) within 8 hours.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient, and any corrective ACTION necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

Same comments 3.4.1.3 for overpressure protection

*A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 300°F unless: (1) the pressurizer water volume is less than 1600 cubic feet, or (2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION : Tavg ≥ 425°F and Pressurizer Pressure ≥ 1000 psig

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 8251 and 8496 gallons, TBD

c. A boron concentration of between 1900 and 2100 ppm,

d. A nitrogen cover-pressure of between ~~400~~ 400+ and 454+ psig, and TBD

e. A water level and pressure channel OPERABLE.

f. A water temperature within the range of 60 - 150°F

Table 6.3.2-1 Min of 400 psig + Allowable error

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN at less than 1000 psig and 425°F within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 1 hour and in HOT SHUTDOWN at less than 1000 psig and 425°F within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume, temperature, and nitrogen cover-pressure in the tanks, and TBD

2) Verifying that each cold leg injection accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig. and Tavg ≥ 425°F

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution;

c. At least once per 31 days when the RCS pressure is above ~~2000~~ 1000 psig by verifying that power to the isolation valve operator is disconnected by ~~removal of the breaker from the circuit,~~ and opening, locking out and tagging the related circuit breaker

removed

d. At least once per 18 months by verifying that each accumulator isolation valve opens automatically under each of the following conditions:

This is an incorrect description

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

2) Upon receipt of a Safety Injection test signal.

TBD

4.5.1.1.2 Each cold leg injection accumulator water level and pressure and temperature channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

Need a regular test on the Relief Valve Set Point at 700 psig

EMERGENCY CORE COOLING SYSTEMS

3/4.5.1

COLD LEG INJECTION ACCUMULATOR ISOLATION VALVES

≤ Pressure Pressure 1000 psig and Tavg ≤ 425°F

LIMITING CONDITION OF OPERATION

3.5.1.1 Each cold leg injection accumulator isolation valve is closed with the circuit breakers opened, locked and tagged.

APPLICABILITY: MODES 3**, 4, 5***

ACTION a) With one or more isolation valve inoperable within 1 (TBD) hour, either restore to OPERABLE status or be in at least COLD SHUTDOWN within the following 30 (TBD) hrs

SURVEILLANCE REQUIREMENTS

Each cold leg injection accumulator isolation valve shall be demonstrated OPERABLE:

a) At least once per 12 hrs by verifying that each cold leg injection accumulator isolation valve is closed

** Limited as above *** With heads bolted and

3/4 5-2 b)

$T_{avg} < 200\ ^{\circ}F$

EMERGENCY CORE COOLING SYSTEMS

UPPER HEAD INJECTION ; MODE 3, Pressurizer Pressure ≥ 1900 psig Tavg > 425°F

LIMITING CONDITION FOR OPERATION

3.5.1.2 Each Upper Head Injection Accumulator System shall be OPERABLE with:

FSAR Value needs revised

a. The isolation valves open.

b. The water-filled Upper Head Injection accumulator containing a minimum of 1850 cubic feet of borated water having a concentration of between 1900 and 2100 ppm of boron, ~~and~~ at a temperature of 70 - 100°F TBD

FSAR Table 6.3.2-1

c. The nitrogen bearing accumulator pressurized to between ~~1966~~ 1220 + Error Allow. and 1264 psig. 1280 + Error Allowance

Ref 3.5.1.2.1 on page

APPLICABILITY: MODES 1, 2 and 3 (with Pressurizer Pressure ≥ 1900 psig and Tavg ≥ 425°F)

ACTION:

a. With the Upper Head Injection Accumulator System inoperable, except as a result of a closed isolation valve(s), restore the Upper Head Injection Accumulator System to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the Upper Head Injection Accumulator System inoperable due to the isolation valve(s) being closed, either immediately open the isolation valve(s) or be in HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.2 Each Upper Head Injection Accumulator System shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume, temperature TBD and nitrogen pressure in the accumulators, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer Pressure above 1900 psig.

The basis for the UHI pressure setting will have to be explained against the SER [illegible]

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the solution in the water-filled accumulator;

c. At least once per 18 months by:

1) Verifying that each accumulator isolation valve closes automatically when the water level in the accumulator is 76.6 ± 0.5 inches for atmospheric pressure (72.5 ± 0.5 inches for blowdown pressure) above the bottom inside edge of the water-filled accumulator, and

2) Verifying that the total dissolved nitrogen and air in the water-filled accumulatory is less than 80 scf per 1800 cubic feet of water (equivalent to 5×10^{-5} pounds nitrogen per pounds water).

d. At least once per 5 years by replacing the membrane installed between the water-filled and nitrogen bearing accumulators and verifying that the removed membrane burst at a differential pressure of 40 ± 10 psi.

e) Need regular test on relief valve set point of 1620 psig (on UHI Accumulator

EMERGENCY CORE COOLING SYSTEMS

UPPER HEAD INJECTION SYSTEM ISOLATION VALVES [illegible] Pressurizer Pressure ≤1900 psig [illegible] ≤425 F

LIMITING CONDITION FOR OPERATION

3.1.5.12.1 All Upper Head Injection System Isolation Valves shall be OPERABLE by being closed and gagged

APPLICABILITY MODES 3**, 4, 5***

ACTION a. With one or more Isolation Valves inoperable, within 1 hour, either restore to OPERABLE status otherwise be in at least COLD SHUTDOWN within the following 30 hrs

SURVEILLANCE REQUIREMENTS

Each Upper Head Injection System

Isolation Valve shall be demonstrated

OPERABLE:

a) At least once per 12 hours by

verifying that each accumulator

isolation valve is closed

** Pressurizer Pressure less than 1900 psig

***

NOT CONSISTENT WITH SER NUREG-0422 SUPP No2. [check Supp] - #0460

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3/4.5.2 ECCS SUBSYSTEMS - ~~$T_{avg} \geq 350°F$~~ Pressurizer Pressure [$\geq$ 1000 psig and $T_{avg} \geq 425°F$]

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Safety Injection pump,

c. One OPERABLE RHR heat exchanger,

d. One OPERABLE RHR pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and ~~3*~~ (with Pressurizer Pressure [$\geq$ 1000 psig and] [$T_{avg} \geq 425°F$])

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

~~* Subject to Pressurizer Pressure being greater than ...~~

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SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
NI162A	Cold Leg Recirc.	Open*
NI121A	Hot Leg Recirc.	Closed
NI152B	Hot Leg Recirc.	Closed
NI183B	Hot Leg Recirc.	Closed
NI173A	RHR Pump Discharge	Open*
NI178B	RHR Pump Discharge	Open*
NI100B	SI Pump RWST Suction	Open
FW27A	RHR/RWST Suction	Open*
NI147A	SI Pump Mini flow	Open

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2) Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1) Verifying automatic isolation and interlock action of the RHR System from the Reactor Coolant System by ensuring that:

a) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened, and

b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to 560 psig the interlocks will cause the valves to automatically close.

*Valves may be realigned to place RHR System in service and for testing pursuant to Specification 4.4.6.2.2.

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2) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

1) Verifying that each automatic valve in [illegible] flow path actuates to its correct position on Safety In[illegible] actuation and automatic switchover to Containmen[illegible] irculation test signals, and

2) Verifying that each of the followin[illegible] start automatically upon receipt of a Safety Injection [illegible] test signal:

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1) Centrifugal charging pump $\geq$ 2380 psid,

2) Safety Injection pump $\geq$ 1430 psid, and

3) RHR pump $\geq$ 160 psid.

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

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2) At least once per 18 months.

Boron Injection Throttle Valves	Safety Injection Throttle Valves
Valve Number	Valve Number
NI-480	NI-488
NI-481	NI-489
NI-482	NI-490
NI-483	NI-491

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For centrifugal charging pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 345 gpm, and — measured TBD

b) The total pump flow rate is less than or equal to 550 gpm. TBD

2) For Safety Injection pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 462 gpm, and

b) The total pump flow rate is less than or equal to 660 gpm.

3) For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3575 gpm.

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3/4.5.3 ECCS SUBSYSTEMS - ~~Tavg < 350°F~~ Between Pressurizer Pressure < 1000 psig and Tavg < 425° and Pressurizer Pressure ≥ 425 psig and Tavg ≥ 350°

LIMITING CONDITION FOR OPERATION

3.5.3 ~~As a minimum, one ECCS subsystem comprised of~~ the ECCS subsystem comprising the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank (RWST) upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE ~~4~~ 3 between Pressurizer Pressure < 1000 psig and Tavg < 425° and Pressurizer Pressure ≥ 425 psig and Tavg ≥ 350

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the RHR heat exchanger or RHR pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

~~#A maximum of 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.~~

An additional ECCS subsystem and comprising the following, shall be OPERABLE:

a) One OPERABLE RHR Heat Exchanger

b) One OPERABLE RHR pump, and

d) an OPERABLE flowpath capable of taking suction from the RWST upon being manually re-align and transferring suction to the

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SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

4.5.3.2 All charging pumps and Safety Injection pumps, except the above required OPERABLE pumps, 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 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.~~

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EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 ECCS SUBSYSTEMS - $T_{avg} \leq 350°F$ and Pressurizer Pressure < 425 psig

LIMITING CONDITION FOR OPERATION

3.5.3 ~~As a minimum, one ECCS subsystem comprised of~~ One ECCS subsystem comprising the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

Why manual realignment TBD

APPLICABILITY: MODE 4. and 5

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the RHR heat exchanger or RHR pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

~~#A maximum of 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.~~

An additional ECCS subsystem comprising the following shall be OPERABLE:
b) One OPERABLE RHR Heat Exchanger
c) One OPERABLE RHR pump, and
d) an OPERABLE flowpath capable of taking suction from the RWST upon being manually realigned and transferring suction to the [illegible]

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SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

4.5.3.2 All charging pumps and Safety Injection pumps, except the above required OPERABLE pumps, 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 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.~~

opened, locked and Tagged

circuit breakers opened, locked and Tagged,

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EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 BORON INJECTION SYSTEM

BORON INJECTION TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The boron injection tank shall be OPERABLE with:

a. A minimum contained borated water volume of 900 gallons, and

b. Between 2000 and 4000 ppm of boron.

APPLICABILITY: MODES 1, 2 and 3, 4 (Unit 1 only)

ACTION:

Boron concentration which will give a

With the boron injection tank inoperable, restore the tank to OPERABLE status within 1 hour or be in HOT STANDBY and borated to a SHUTDOWN MARGIN ~~equivalent~~ of ~~to~~ 1% delta k/k at 200°F within the next 6 hours; restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The boron injection tank shall be demonstrated OPERABLE by:

a. Verifying the contained borated water volume at least once per 7 days, and

b. Verifying the boron concentration of the water in the tank at least once per 7 days.

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3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained borated water volume of at least 372,100 gallons,

b. A boron concentration of between 2000 and 2100 ppm of boron,

c. A minimum solution temperature of 70°F, and

d. A maximum solution temperature of 100°F.

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

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 70°F or greater than 100°F.

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3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained borated water volume of at least 372,100 gallons,

b. A boron concentration of between 2000 and 2100 ppm of boron,

c. A minimum solution temperature of 70°F, and

d. A maximum solution temperature of 100°F.

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

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours ~~and in COLD SHUTDOWN within the following 30 hours.~~ - boron concentration which will give a SHUTDOWN MARGIN and borated to a ~~equivalent~~ of ~~≥~~ 1% delta K/K at 200°F.

- restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 70°F or greater than 100°F.

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PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

MODES 1, 2 & 3.

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY witin 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective ACTION to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1210 psig at a flow of greater than or equal to 450 gpm;

2) Verifying that the steam turbine-driven pump develops a discharge pressure of greater than or equal to 1210 psig at a flow of greater than or equal to 900 gpm when the secondary steam supply pressure is greater than 900 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

~~*not applicable with steam pressure less than 900 psig.~~

check why not [illegible]

3) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position;

4) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER; and

5) Verifying that the isolation valves in the auxiliary feedwater suction line from the upper surge tanks are open with power to the valve operators removed.

b. At least once per 18 months during shutdown by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an Auxiliary Feedwater Actuation test signal,

2) Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal, and

3) Verifying that the valve in the suction line of each auxiliary feedwater pump from the Nuclear Service Water System automatically actuates to its full open position within less than or equal to 13 seconds on a Low Suction Pressure test signal.

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PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

MODES. 4 & 5.

LIMITING CONDITION FOR OPERATION. IF REQUIRED FOR REACTOR COOLANT OPERABILITY FOR TS. 3.4.1.3 AND/OR TS 3.4.14.1

3.7.1.2a) At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

b. ~~One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.~~

APPLICABILITY: MODES ~~1, 2, and 3~~ 4 and 5* (with reactor coolant loops filled)

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. } MODIFY TO COMPATIBILITY.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY witin 6 hours and in HOT SHUTDOWN within the following 6 hours. } MODIFY TO COMPATABILITY

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective ACTION to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible. } [illegible]

C with TS 3.4.1. and TS 3.4.14.1

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1210 psig at a flow of greater than or equal to 450 gpm;

2) Verifying that the steam turbine-driven pump develops a discharge pressure of greater than or equal to 1210 psig at a flow of greater than or equal to 900 gpm when the secondary steam supply pressure is greater than 900 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

~~*Not applicable with steam pressure less than 900 psig.~~

check whether it will require

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SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position;

4) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER; and

5) Verifying that the isolation valves in the auxiliary feedwater suction line from the upper surge tanks are open with power to the valve operators removed.

b. At least once per 18 months during shutdown by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an Auxiliary Feedwater Actuation test signal,

2) Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal, and

3) Verifying that the valve in the suction line of each auxiliary feedwater pump from the Nuclear Service Water System automatically actuates to its full open position within less than or equal to 13 seconds on a Low Suction Pressure test signal.

AUXILIARY FEEDWATER

PLANT SYSTEMS

CONDENSATE STORAGE TANK SYSTEM

comprising [demineralized water storage] Upper Surge Tank, Auxiliary Feedwater Condensate Storage Tank, and Condenser Hot Well

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tanks (CSTS) shall be OPERABLE with a contained water volume of at least [75,000] gallons of water.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the (alternate water source) Nuclear Service Water System and Standby Nuclear Service Water Pond as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storagr tank system shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The ~~(alternate water source)~~ Nuclear Service Water System and Standby Nuclear Service Water Pond shall be demonstrated OPERABLE at least once per 12 hours by (method dependent upon alternate source) whenever the ~~(alternate water source)~~ ~~is~~ these systems are the supply source for the auxiliary feedwater pumps.

PLANT SYSTEMS

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MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.4 Each main steam line isolation valve (MSLIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, ~~and~~ 3 and 4

ACTION:

MODE 1 - With one MSLIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5% of RATED THERMAL POWER within 2 hours.

MODES 2 and 3 - With one MSLIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided:

a. The isolation valve is maintained closed, and

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. and in COLD SHUTDOWN in

Only one Safety Iso Valve in line

SURVEILLANCE REQUIREMENTS

4.7.1.4 Each MSLIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5.

(ADD)

RL

PLANT SYSTEMS

ATMOSPHERIC DUMP VALVES

Ref NUREG -0423 Supp. No 2

Section 5.3.3 specifies their use during natural circulation cooldown even though they are not safety grade they should be Tech Spec'd with provision for a special action statement in the event they are not OPERABLE

Otherwise - all related T.S's (in these T.S's) which use the RCS/SG loop as an alternate Cooldown Technique are not admissable, including natural circulation.

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3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3, ~~and~~ 4 and 5* (with RCS loops filled)

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At least once per 18 months during shutdown, by verifying that:

   1) Each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection test signal, and

   2) Each component cooling water pump starts automatically on a Safety Injection and Station Blackout test signal.

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3/4.7.4 NUCLEAR SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent nuclear service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4 and 5 (with RCS loops filled)

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.4 At least two nuclear service water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At least once per 18 months during shutdown, by verifying that:

   1) Each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection test signal, and

   2) Each nuclear service water pump starts automatically on a Safety Injection and Station Blackout test signal.

BC locally

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3/4.7.5 STANDBY NUCLEAR SERVICE WATER POND

LIMITING CONDITION FOR OPERATION

3.7.5 The standby nuclear service water pond shall be OPERABLE with:

a. A minimum water level at or above elevation 739.5 feet Mean Sea Level, USGS datum, and

b. An average water temperature of not less than greater than 70°F or ~~equal to~~ 78°F at elevation 700 feet in the intake structure. $\boxed{\le 94°F}$ Refer FSAR Pg 9.2-12(

APPLICABILITY: MODES 1, 2, 3, and 4 and 5* (with RCS loops filled)

ACTION: (Units 1 and 2)

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. this is NOT Approp'l

NO

SURVEILLANCE REQUIRMENTS

4.7.5 The standby nuclear service water pond shall be determined OPERABLE:

a. At least once per 24 hours by verifying the water level to be within its limit,

b. At least once per 24 hours during the months of July, August and September by verifying the water temperature to be within its limit, and

c. At least once per 12 months by visually inspecting the dam and verifying no abnormal degradation, erosion, or excessive seepage.

Bootstrap Education !!!

Q7

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3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite essential auxiliary power system, and

b. Two separate and independent diesel generators, each with:

1) A separate day tank containing a minimum volume of 120 gallons of fuel,

2) A separate fuel storage system containing a minimum volume of 28,000 gallons of fuel,

3) A separate fuel transfer pump,

4) Lubricating oil storage contains a minimum total volume of (   ) gallons of lubricating oil, and

5) Capability to transfer lubricating oil from storage to the diesel generator unit.

APPLICABILITY: MODES 1, 2, 3, ~~and~~ 4, and 5*(with Reactor Coolant Loop Filled)

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specifications 4.8.1.1.1a. and 4.8.1.1.2a.4) within 1 hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specifications 4.8.1.1.1a. and 4.8.1.1.2a.4) within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one diesel generator inoperable in addition to ACTION a. or b. above, verify that:

1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

All potential sources of unborated water supply shall be isolated from the Reactor Coolant System, Refueling Canal and Spent Fuel Pool

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3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2000 ppm.

APPLICABILITY: MODE 6*, with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

Insert an additional valve to ensure isolation

4.9.1.3 NV-250 shall be verified closed under administrative control at least once per 72 hours.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.9

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LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 6 when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective ACTION to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective ACTION to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

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

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

January 27, 1984

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

FROM: Lake H. Barrett, Deputy Program Director
TMI Program Office

SUBJECT: STATUS REPORT ON DPO OF ROBERT LICCIARDO ON SAFETY SYSTEMS

On December 29, 1983, I was designated the responsible manager for the DPO raised by Mr. Licciardo concerning safety systems and related technical specifications for the McGuire reactor. Since then I have met several times with Mr. Licciardo to assist him in preparing a more explanatory statement of his concerns which are multiple and complex. I believed this was the necessary first step to identify how to precede with the resolution. As a result of these meetings Mr. Licciardo has signed a January 26, 1984, memorandum which describes his concerns and breaks them down into groups and subgroups.

The initial scheduled date for resolution of the DPO was February 1, 1984. Because of the time needed to obtain a better definition of the DPO, additional time will be required for DPO resolution. Therefore, I am requesting an extension until February 29, 1984.

Lake

Lake H. Barrett
Deputy Program Director
TMI Program Office

cc:
R. Licciardo
N. Lauben
B. Sheron
R. Mattson
E. Eisenhut
R. Houston
R. Majors
R. Licciardo DPO File

JAN 27 1984

50-369

MEMORANDUM FOR: Lake H. Barrett, Deputy Program Director
TMI Program Office, TMI Site

FROM: Robert B. A. Licciardo, Reactor Systems Branch, DSI/NRR

SUBJECT: DPO REGARDING MCGUIRE TECHNICAL SPECIFICATIONS

Reference:

1. Memo for G. N. Lauben (RSB) from R.B.A. Licciardo (RSB), dated Dec. 07, 1983, Subject: Differing Professional Opinion Related to Technical Specifications for McGuire Unit 2 (And Proposal for McGuire Unit 1).

2. Memo for L. H. Barrett (TMI Program) from R. B. A. Licciardo (RSB), dated January 26, 1984, Subject: Differing Professional Opinion - McGuire Technical Specifications.

The following material is an elaboration of the principal elements of the Differing Professional Opinion submitted formally by the writer to reference 1.

Background
The DPO developed out of an action arising from a memo by C. O. Thomas (SSPD/DL) to Brian W. Sheron (RSB/DSI) et.al dated January 14, 1983, on the subject: Proof and Review of McGuire - Units 1 and 2 Technical Specification. This memo requested reviews of the relevant sections of the Proof and Review copy of the technical specification proposed for McGuire Units 1 and 2, reference Appendix B.

Arising from the above requirement, the writer ultimately prepared a final draft memorandum, for concurrence, to T. M. Novak (AD/L) from R. W. Houston (AD/RS) dated 06/15/1983 on the subject: Staff Review of Proof and Review Copy of Proposed Technical Specification for McGuire Units 1 and 2. This memo was attached to a final draft copy, for concurrence, of a document entitled, "Safety Evaluation Report, McGuire Nuclear Station Units 1 and 2, Proposed Technical Specifications", dated 06/15/1983; this document includes a mark-up of the Proof & Review copy of the McGuire Technical Specifications. Proposed concurrence chain for these copies was R. Licciardo, N. Lauben, J. E. Rosenthal, B. Sheron, W. Houston; one copy each of this material was provided to N. Lauben (current section leader) and J. E. Rosenthal (former section leader) on

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June 24, 1983. The proposed SER summarized the results of the review on an item-by-item basis for comparability with the Safety Analyses Report and the Amendments thereto, in accordance with 10 CFR 50.36 for McGuire Unit 1 and Unit 2, and 10 CFR 50.59(c) for McGuire Unit 1. Copies of these documents have been provided under reference 2..

The proposed SER dated 06/15/1983 and referenced above contains the detailed substance from which the DPO is drawn.

The Introduction to the SER describes the basis for the information within the Proof and Review Document supplied for review. This Document contained no record of, or safety evaluation report on, those changes which were made to the Westinghouse Standard Technical Specifications including any relationship to the existing McGuire Unit 1 Technical Specification, and the Final Safety Analyses Reports, and Amendments thereto, or the Safety Evaluation Reports for, McGuire Units 1 and 2. The proof and review requested therefore required a detailed check of every relevant item for evaluation purposes. The Introduction also summarized a proposed RSB staff position as the bases for the plant being allowed to proceed to power:

> "A number of the proposed changes which could be readily introduced inside the structure of the existing Standard Technical Specifications have already been incorporated into the Technical Specifications for McGuire Unit 2 issued with License No. NPF-17 on March 1983. A number of other proposed changes were not incorporated at the same time, since they involved changes to the Standard Technical Specifications necessitating substantive NRC review, and because the Safety Analysis Reports stated that the more significant changes would nevertheless be a part of Plant Operating Procedures and therefore of plant operations. It was evaluated that relative short term operation of the facility under these different administrative but identical engineering circumstances would not significantly impact public health and safety while the principal longer term regulatory issues of incorporating such procedures into the Standard Technical Specifications received the required consideration."

The Conclusion to the SER briefly summarized the writers findings from the review as:

> "Our review shows no close correlation between the docketed information for the Facilities and the proposed Technical Specification, in a number of important areas. This position may derive from a desire by the licensee to conform to the Standard Technical Specifications which in themselves do not conform to the requirements of the Docket. The items of concern are primarily with the requirements for "Boration Control" and the "Emergency Core Cooling Systems (ECCS); and from this there are significant repercussions on the Reactor Trip and ESF Actuation System, instrumentation requirements."

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DIFFERING PROFESSIONAL OPINION RELATED TO TECHNICAL SPECIFICATION FOR MCGUIRE UNIT 2 (AND PROPOSED FOR MCGUIRE UNIT 1)-AN ELABORATION

The NRC Manual Chapter NRC-4125, Differing Professional Opinions, recommends that the statement [of DPO] include:

(1) A summary of the originators perception of the prevailing staff review, existing management decision or stated position, or the proposed or established agency practice.
No written response of a non-personal nature has been received by the writer from management of his verbal and written submittals in respect of his Proposed Safety Evaluation Report for McGuire Units 1 and 2 [PMSER]. The only observation of an implied formal response has been manifest in the fact that the plant was allowed to start up on during March, 1983 without reference to the writer and that he was assigned to another task (under verbal protest) and the related Planned Accomplishment (PA) number was no longer available to conclude the review. Subsequent to a discussion with his Branch Chief, about mid-year 1983, of his concern for a continuation of this situation, the writer's then former Section Leader verbally advised that he had given verbal approval of the Proposed Technical Specifications for RSB/DSI, which permitted start up of Unit 2 during March 1983.

The staff review and evaluation (by others) on the adequacy of the Technical Specifications (T.S.) for McGuire Unit 2 at the time of issuance of its low power [5%] license on March 5, 1983 is in NUREG-0422, March 1, 1978, Safety Evaluation Report by the Office of Nuclear Reactor Regulation. The staff review and evaluation (by others) on the adequacy of the same Technical Specifications for McGuire Unit 1 is within the SER associated with the issuance of Amendment No. 19 to the license for McGuire Unit 1 on March 29, 1983.

(2) A description of the originators opinions and how they differ from any items discussed in 1 above.

The writers opinions are detailed on a line-by-line basis in the Proposed McGuire Safety Evaluation Report (PMSER) including the proposed mark-up of the Proof and Review Copy; copies of these have been attached to reference 2. These are summarized in the related Conclusions in pages 17-20, leading to the statement within the his formal DPO that:

> "Disparities existing between current Technical Specifications relating to these systems and the safety analyses of record within the existing licensing basis, suggest that the existing regulatory requirements identified in 10 CFR Parts 50.36, 50.46 and 50, Appendix A could be compromised."

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Additionally, the writer proposes that the "safety review" of the RSB section of the "proof and review" copy permitting startup of the plant by others was inadequate and not properly evaluated and documented as required by 10 CFR.

A categorization of principal concerns including disparities, within the current technical specifications for McGuire Units 1 and 2 has already been provided under reference 2, a copy of which is Appended to this memo.

(3) A statement of the originators assessment of the resulting consequences if the differing professional opinion is not adopted by the agency.

In his formal DPO, the writer has submitted that:

"Disparities existing between current Technical Specifications relating to these systems and the safety analyses of record within the existing licensing basis, ------ ----------- could manifest itself in increased risk to public health and safety beyond that intended in the existing licensing basis. As an example, the mitigating effects of the Emergency Core Cooling System (ECCS) could be compromised."

The particular circumstances for the ECCS are summarized in reference 2, and in the PMSER on pages 18 and 19 and detailed on pages 13-15.

In his PMSER, on the issue of public health and safety the writer concludes that

"We find a number of substantive differences between the proposed technical specifications and the docketed information which could ultimately have a significant impact on public health and safety and we have proposed technical specifications to correct this situation."

(4) The status of related efforts with which the originator is familiar and their potential contribution toward resolution of the originators Differing Professional Opinion.

During the first 6 weeks of this review, (January-March 1983) the writer was in active discussion with his then Section Leader who also sought substantive information from other persons and branches, in attempting to resolve the disparities which had been observed. The writer is also aware, that since the submittal of his written version of his DPO on November 07, 1983, his current Section Leader has been actively involved in discussions with the writer in resolution of these issues. Further, by memo to Robert Licciardo from Brian W. Sheron dated December 9, 1983 on the subject: McGuire Technical Specification Issue, B. Sheron records that [in mid November] he did propose to evaluate the issue and determine

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JAN 27 1984

if any actions should or could be taken to address [these] concerns and that I should defer formal issuance of my DPO until that approach had been tried. Under the totality of the circumstance preceeding the formal submitted of December 7, 1983, this approach was not acceptable to the writer. Since January 24. 1984, the writer has been active with L. Barrett (TMI-site) in furthering resolution of this DPO.

Original signed by:

R. B. A. Licciardo, Nuclear Engineer
Section A
Reactor Systems Branch, DSI

Enclosure: As stated

cc: w/o encl.

H. Denton	R. Mattson
R. Houston	D. Eisenhut
B. Sheron	B. Paul Cotter, ASLAB
N. Lauben	A. Rosenthal, ASLAB
T. Novak, DL	R. Birkel, DL
E. Adensam	Licciardo DPO File
N. Brinkman	F. Miraglia
C. Thomas	J. Rosenthal

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UNITED STATES
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WASHINGTON, D. C. 20555

Appendix A

JAN 2 6 1984

MEMORANDUM FOR: Lake H. Barrett, Deputy Program Director
TMI Program Office, NRR

FROM: Robert B. A. Licciardo, Reactor System Branch
DSI, NRR

SUBJECT: DIFFERING PROFESSIONAL OPINION - MCGUIRE TECHNICAL SPECIFICATIONS

On December 7, 1983, I submitted my DPO (Attachment 1) concerning disparities between the McGuire technical specifications and the FSAR safety analyses. Since that time, I have met several times with you to discuss my DPO and am documenting the following further description and elaboration of my DPO in accordance with the guidance of paragraph C.2 of Manual Chapter 4125, Differing Professional Opinions.

The DPO contains multiple complex issues of various types and subgroups. The first type of issues are technical based on some McGuire FSAR safety analyses differing in various respects from the McGuire proof and review technical specifications such that parts of the technical specifications are non-conservative or contradictory. These issues, which can be divided into four subgroups are typified as follows:

1) Boron limits

The FSAR analyses states that the reactor coolant system is borated to cold shutdown concentrations prior to cooling below 557°F whereas the technical specifications requires only a boron concentration necessary to provide a minimum normal shutdown margin of 1.6% delta k/k; i.e., a boron concentration that is lower than cold shutdown. This lower boron concentration may not be adequate to assure fuel protection under non LOCA events; e.g. main steam line break. I propose that the FSAR higher boron limits be used in the technical specifications, or that analyses be performed to assure that adequate fuel protection will be maintained under accident conditions with the lower boron concentration requirements in the technical specifications.

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2) ECCS Pump Operability Requirements

The FSAR analyses (and staff SER) establishes the ECCS pump operability requirements after careful consideration of sufficient capacity for decay heat removal and boration while assuring adequate overpressure protection when the RCS is cooled down. The McGuire technical specifications do not fully reflect these ECCS operability considerations because they require HPSI and charging pump operability contrary to the FSAR analyses which state that these pumps are non-operable because of overpressure considerations. This contradiction may lead to operator confusion and/or improper plant procedures.

3) Reactor Trip Instrumentation and ESF Actuation Response Times and ESF Actuation Set Points

The FSAR analyses assume certain response times and set points for various reactor trip and ESF actuation instrumentation. The McGuire technical specifications specify various response times and set points that are sometimes different from the FSAR analyses which could result in a reduced level of protection for the reactor. I propose that the FSAR response times and set points be used in the technical specifications or that analyses be performed to assure that adequate reactor protection is provided by the technical specifications.

4) ESF Actuation Instrumentation

The FSAR analyses assume that certain ESF actuation instrumentation; e.g., High Containment Pressure and Main Steam Line Isolation in Mode 4, is operable. The McGuire technical specifications do not require these instruments operable in the modes addressed in the FSAR. I propose that the mode addressed in the FSAR be included in the technical specifications or analyses performed to assure that they are not necessary for safety.

The second type of concern is more judgemental in nature in that I submit that 10 CFR 50.36, Technical Specifications, requires that the McGuire technical specifications contain more safety restrictions; e.g. LCOs, than is presently incorporated in the McGuire or Westinghouse Standard Technical Specifications. I submit that a thorough review of the McGuire FSAR "analyses of record" would establish more restrictions; e.g. LCOs, and that those restrictions should be in the McGuire technical specifications or that analyses should be performed (specifically for McGuire or generic enveloping analyses) to provide the legal/technical basis that the present technical specifications are adequate and appropriately implement 10 CFR 50.36, 50.46, and the GDC (Appendix A). Examples of FSAR limitations that should be so addressed are as follows:

1) Control Rod Insertion and Reactor Trip System Operability Limits

FSAR analyses assume certain control rod positions and reactor protection system availability when in modes 3 through 6. The McGuire technical specifications do not impose any limitations on control rod position during these modes. Therefore, the positions of the McGuire control rods could be different from those used in the FSAR analyses and could result in less conservative reactor protection for non LOCA events. I propose that the McGuire technical specifications include either limitations on control rod positions or a revision and re-validation of the availability of the reactor protection system, during modes 3 though 6.

2) RCS Loop Operability Limits

The FSAR analyses requires that an RCS loop be available when the plant is in mode 4 to assure decay heat removal during a single failure event; i.e. an RCS/decay heat removal system isolation valve. The McGuire technical specifications do not require an RCS loop to be operable in this mode (4). I propose to determine the need for RCS loop(s) operability by reviewing and/or performing analyses of accidents during cooldown to establish a more reliable basis than is currently available in the FSAR for the current LCOs in the technical specifications.

3) Thermal-Hydraulic Limits

The FSAR specifies certain thermal hydraulic parameters; e.g., RCS pressure, temperature and pressurizer water level, as initial conditions for various accident analyses. The McGuire technical specifications do not adequately specify these conditions. There is a need to clarify and verify the present specifications which could allow reactor conditions that could be less conservative then the design bases. I propose that Table 3.2-1 and Section 2 need to be revised to more accurately reflect the FSAR programmed operating conditions and eliminate ambiguities.

The third type of concern involves internal staff practices for reviewing and issuing the technical specifications when licensing a reactor. Based on my McGuire experience, I submit that the "safety review" of the RSB section of the "proof and review" technical specifications, which permitted start up of the plant by others, was inadequate and not properly justified and documented as required by 10 CFR. My review shows that a thorough review of the McGuire FSAR "analyses of record" indicates significant inconsistencies with the McGuire technical specifications (and its parent Westinghouse Standard Technical Specifications). I propose that responsible technical branches work more closely with the SSPB/DL group during the entire licensing review period, and that the staff adopt improved internal administrative procedures to document reviews that justify the adequacy of the final issued technical specifications. I suggest that the staff internally use a 10 CFR 50.59 methodology for its technical specification reviews to confirm that the technical specifications maintain the reactor within the FSAR safety analysis envelope and clearly articulate and justify the rationale for any less restrictive criteria.

I believe my above description describes how my December 7, 1983 DPO differs from the existing staff positions concerning the RSB Section of McGuire proof and review technical specifications. Supporting documents are attached as follows:

Attachment 2: My draft SER for the McGuire Technical Specifications (dated June 15, 1983)

Attachment 3: My proposed McGuire Technical Specifications (dated June 15, 1983)

R. Licciardo

Robert B. A. Licciardo
Reactor System Branch
DSI, NRR

Attachments:
As Stated

cc wo/attachments:
H. Denton
N. Lauben
T. Novak
E. Adensam
R. Mattson
D. Eisenhut
B. Cotter
A. Rosenthal
R. Birkel
F. Miraglia
C. Thomas

cc w/attachments:
R. Licciardo DPO File
B. Sheron
R. Houston
D. Brinkman

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

Appendix B

JAN 1 = 1983

Docket Nos. 50-369/370

MEMORANDUM FOR: Those on Attached List

FROM: Cecil O. Thomas, Chief
Standardization & Special Projects Branch
Division of Licensing

SUBJECT: PROOF AND REVIEW OF MCGUIRE - UNITS 1 AND 2 TECHNICAL SPECIFICATIONS

Attached please find the Appendix A Technical Specifications for McGuire-Units 1 and 2 for proof and review. It should be noted that Unit 1 was licensed in January 1981. These Technical Specifications will combine the requirements on Units 1 and 2 as well as upgrade the Unit 1 requirements to current standards. You should consider this aspect during your review. We request that you review those sections of the attached Technical Specifications which pertain to your particular area of responsibility and that the results of this review, identifying those sections of the Technical Specifications reviewed, be forwarded to the Standardization & Special Projects Branch by February 9, 1983. This abbreviated schedule is due to an expected licensing date of March 1, 1983.

If you have no comments or suggestions, i.e., if you are in agreement with the Technical Specifications content in your area of review, it is requested that a response to that effect identifying those sections of the Technical Specifications reviewed be provided by the above specified date.

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization & Special
Projects Branch
Division of Licensing

Attachment:
McGuire - Units 1 and 2
Technical Specifications

cc: w/o attachment
F. Miraglia
D. Eisenhut

Contact: F. Anderson, x27803

Dupe PDR
~~8301200161~~

ATTACHED LIST

Thomas A. Ippolito, Chief
Operating Reactors Assessment Branch

George W. Knighton, Chief
Licensing Branch No. 3

B. D. Liaw, Chief
Materials Engineering Branch

Victor Benaroya, Chief
Chemical Engineering Branch

Vincent S. Noonan, Chief
Equipment Qualifications Branch

Walter P. Haass, Deputy Branch Chief
Quality Assurance Branch

Robert J. Bosnak, Chief
Mechanical Engineering Branch

Franz P. Schauer, Chief
Structural Engineering Branch

Robert E. Jackson, Chief
Geosciences Branch

George Lear, Chief
Hydrologic & Geotechnical Engineering Branch

William H. Regan, Chief
Siting Analysis Branch

Brian W. Sheron, Chief
Reactor Systems Branch

Faust Rosa, Chief
Instrumentation & Control Systems Branch

Walter R. Butler, Chief
Containment Systems Branch

M. I. Srinivasan, Chief
Power Systems Branch

Carl Berlinger, Chief
Core Performance Branch

Olan D. Parr, Chief
Auxiliary Systems Branch

Lewis G. Hulman, Chief
Accident Evaluation Branch

William P. Gammill, Chief
Meteorology and Effluent Treatment
Systems Branch

Frank Congel, Chief
Radiological Assessment Branch

Voss A. Moore, Jr., Chief
Human Factors Engineering Branch

Harold R. Booher, Chief
Licensee Qualifications Branch

Dennis L. Ziemann, Chief
Procedures and Test Review Branch

Regional Administrator
Region II

FEB 22 1984

MEMORANDUM FOR: Lake Barrett, Deputy Program Director, TMI Program Office

FROM: Brian W. Sheron, Chief, Reactor Systems Branch, DSI

SUBJECT: DPO OF ROBERT LICCIARDO ON MCGUIRE TECHNICAL SPECIFICATIONS

The purpose of this memorandum is to advise you of the results of our recent efforts to assist in the resolution of Bob Licciardo's DPO on the McGuire Tech. Specs. Over the past two weeks, Norm Lauben and I met with him on four separate occasions, each lasting approximately 2-3 hours, to discuss and try to resolve as many as possible of each of the items identified in his proposed SER on the McGuire Tech. Specs. During these meetings, we addressed about 25% of the items. Most of our time was spent trying to understand the basis for his proposed revisions or additions to the Tech. Specs.

As a result of the four meetings, I have concluded that a fundamental problem he has with the technical specifications is that there is no basis, or traceability provided with the Tech. Specs. so the reviewer can understand where a particular number came from. Lacking this basis information, he went to the docket material, such as the FSAR, and staff SERs. What he found were discrepancies between that information and the Tech. Specs. His approach was to propose modifications to the Tech. Specs. that he felt would make them consistent with the docket material ("the analyses of record"). This approach was consistent with the Standardization and Special Projects Branch's specific request to provide comments in the form of mark-ups.

Based on our experience in these meetings, I have proposed to Bob that he take his proposed SER, with his marked-up Tech. Specs., and transform it into a set of specific questions. For example, if two numbers are inconsistent, he should prepare a question, asking why the numbers were inconsistent. I asked that he clearly document where he found his numbers (i.e., FSAR Amendment No., FSAR chapter, page no.).

We estimated it would take him about three to four weeks to prepare such a question set. Both Norm Lauben (his section leader) and I would review the questions, resolve those that could be resolved within RSB, and, once we were satisfied, we would forward the remainder to the Standardization and Special Projects Branch in DL.

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~~PDR~~

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I met with Cecil Thomas, Chief of the Standardization and Special Projects Branch, and explained to him my proposed plan. He agreed that his branch would review the questions submitted to SSPB and they would try to resolve as many as they could. Following SSPB's review, those items that still remained and appeared to be discrepancies would be brought to the attention of the licensee (Duke Power). Cecil thought that the licensee would voluntarily want to correct any inconsistencies brought to their attention. For items of more substance, such as a proposed additional Tech. Spec., we would follow the guidance of Mr. Denton's October 25, 1983, letter regarding plant-specific backfits. Any items deemed generic during the course of the review would be handled according to the provisions of NRR Office Letter No. 38.

My understanding is that by acknowledging that his DPO merits attention and that a resolution plan has been identified, this would be an acceptable basis for him to consider his DPO resolved.

Original signed by:
Brian W. Sheron

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

cc: H. Denton
R. Mattson
R. W. Houston
N. Lauben
C. Thomas
R. Birkel

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50-370

February 27, 1984

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

FROM: Lake H. Barrett, Deputy Program Director
TMI Program Office

SUBJECT: RECOMMENDED RESOLUTION OF R. LICCIARDO DPO

As requested by your memo of December 29, 1983, I have conducted an independent assessment of the December 7, 1983 R. Licciardo DPO (Enclosure 1) concerning disparities between the McGuire reactor safety system technical specifications and the safety analyses of record in the licensing documents. Mr. Licciardo provided further description and elaboration on his DPO in memoranda to me dated January 26 and 27, 1984 (Enclosures 2 and 3). At my request, during the month of February, Reactor Systems Branch management spent many hours reviewing with Mr. Licciardo the technical substance and bases of his specific concerns. DSI supplied information regarding this DPO is attached as Enclosures 4 through 9.

I have evaluated the documents and met with the various parties and have concluded that the issue raised in Mr. Licciardo's DPO warrants further staff attention. I recommend the following actions:

1) In accordance with NRC Manual Appendix 4125, Section G.1.a, adopt the views of Mr. Licciardo's December 7, 1983, DPO. This DPO addresses apparent disparities between the McGuire reactor safety system technical specifications and the safety analyses of record within the licensing documents.

2) Develop and implement a plan for timely identification and resolution of the McGuire disparities.

3) Perform a review of staff procedures and practices used for the review of technical specifications when issuing operating reactor licenses. It is my understanding the DL presently has such an effort underway.

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It is difficult to assess the safety significance of this disparity issue before a more complete technical review of the McGuire disparities is completed. Based on my discussions with Mr. Licciardo and other staff members I consider this issue important deserving staff attention. As Mr. Licciardo states in his DPO the disparities "suggest" that regulations "could be compromised" and that compromises "could manifest" in increased risk. My limited review of Mr. Licciardo's elaboration of the disparities in his

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January 26, 1983 memorandum (Enclosure 2) indicates that the disparities in the McGuire technical specifications would not reduce overall safety margins to a point resulting in an unacceptable public risk and, therefore at this time, do not require extraordinary regulatory actions. Once the staff has completed its safety review of the disparities appropriate regulatory action can then be determined.

Mr. Licciardo's December 7, 1983 DPO requested a impartial peer group review in accordance to Manual Chapter 4125. I have since discussed my above proposed resolution with Mr. Licciardo and he has agreed to waive the peer group review provided the above resolution is adopted.

ORIGINAL SIGNED BY:

Lake H. Barrett
Deputy Program Director
TMI Program Office

Enclosures:
1. Memo, December 7, 1983, R. B. A. Licciardo to G. Norman Lauben 8312220004
2. Memo, January 26, 1984, Robert B. A. Licciardo to Lake H. Barrett 8402070496
3. Memo, January 27, 1984, Robert B. A. Licciardo to Lake H. Barrett 8402070493
4. Memo, December 9, 1983, Brian W. Sheron to Robert Licciardo 8312220006
5. Memo, December 13, 1983, G. Norman Lauben to Brian W. Shero 8312230216
6. Memo, December 15, 1983, Brian W. Sheron to Harold R. Denton 8401110577
7. Memo, December 15, 1984, R. Wayne Houston to Roger J. Mattson 8312230113
8. Memo, February 1, 1984, Brian W. Sheron to Lake H. Barrett 8404110467
9. Memo, February 22, 1984, Brian W. Sheron to Lake H. Barrett 8403020416

8404110470

cc w/enclosures:
R. Licciardo
R. Mattson
D. Eisenhut
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NRC FORM 318 (10-80) NRCM 0240

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ENCLOSURE 1

DEC 0 7 1983

MEMORANDUM FOR: G. Norman Lauben, Section Leader
Section A
Reactor Systems Branch, DSI

29444

FROM: R. B. A. Licciardo, Nuclear Engineer
Section A
Reactor Systems Branch, DSI

SUBJECT: DIFFERING PROFESSIONAL OPINION RELATED TO TECHNICAL SPECIFICATION FOR MCGUIRE UNIT 2 (AND PROPOSED FOR MCGUIRE UNIT 1)

This memorandum constitutes formal submission of a Differing Professional Opinion (DPO) in accordance with NRC Manual Chapter NRC-4125 and NRC Appendix 4125.

This DPO relates to the operation of McGuire Unit 2 (also proposed for McGuire Unit 1) safety systems necessary to provide assurance of public health and safety.

Disparities existing between current Technical Specifications relating to these systems and the safety analyses of record within the existing licensing basis, suggest that the existing regulatory requirements identified in 10 CFR Parts 50.36, 50.46 and 50, Appendix A could be compromised. This compromise could manifest itself in increased risk to public health and safety beyond that intended in the existing licensing basis. As an example, the mitigating effects of the Emergency Core Cooling System (ECCS) could be compromised.

In accordance with NRC Appendix 4125 G.2.a, I request that this DPO be presented to an impartial peer review group for review, evaluation and comment.

R. Licciardo

R. B. A. Licciardo, Nuclear Engineer
Section A
Reactor Systems Branch, DSI
U.S. Nuclear Regulatory Commission

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

ENCLOSURE 2

JAN 26 1984

MEMORANDUM FOR: Lake H. Barrett, Deputy Program Director
TMI Program Office, NRR

FROM: Robert B. A. Licciardo, Reactor System Branch
DSI, NRR

SUBJECT: DIFFERING PROFESSIONAL OPINION - MCGUIRE TECHNICAL SPECIFICATIONS

On December 7, 1983, I submitted my DPO (Attachment 1) concerning disparities between the McGuire technical specifications and the FSAR safety analyses. Since that time, I have met several times with you to discuss my DPO and am documenting the following further description and elaboration of my DPO in accordance with the guidance of paragraph C.2 of Manual Chapter 4125, Differing Professional Opinions.

The DPO contains multiple complex issues of various types and subgroups. The first type of issues are technical based on some McGuire FSAR safety analyses differing in various respects from the McGuire proof and review technical specifications such that parts of the technical specifications are non-conservative or contradictory. These issues, which can be divided into four subgroups are typified as follows:

1) Boron limits

The FSAR analyses states that the reactor coolant system is borated to cold shutdown concentrations prior to cooling below 557°F whereas the technical specifications requires only a boron concentration necessary to provide a minimum normal shutdown margin of 1.6% delta k/k; i.e., a boron concentration that is lower than cold shutdown. This lower boron concentration may not be adequate to assure fuel protection under non LOCA events; e.g. main steam line break. I propose that the FSAR higher boron limits be used in the technical specifications, or that analyses be performed to assure that adequate fuel protection will be maintained under accident conditions with the lower boron concentration requirements in the technical specifications.

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2) ECCS Pump Operability Requirements

The FSAR analyses (and staff SER) establishes the ECCS pump operability requirements after careful consideration of sufficient capacity for decay heat removal and boration while assuring adequate overpressure protection when the RCS is cooled down. The McGuire technical specifications do not fully reflect these ECCS operability considerations because they require HPSI and charging pump operability contrary to the FSAR analyses which state that these pumps are non-operable because of overpressure considerations. This contradiction may lead to operator confusion and/or improper plant procedures.

3) Reactor Trip Instrumentation and ESF Actuation Response Times and ESF Actuation Set Points

The FSAR analyses assume certain response times and set points for various reactor trip and ESF actuation instrumentation. The McGuire technical specifications specify various response times and set points that are sometimes different from the FSAR analyses which could result in a reduced level of protection for the reactor. I propose that the FSAR response times and set points be used in the technical specifications or that analyses be performed to assure that adequate reactor protection is provided by the technical specifications.

4) ESF Actuation Instrumentation

The FSAR analyses assume that certain ESF actuation instrumentation; e.g., High Containment Pressure and Main Steam Line Isolation in Mode 4, is operable. The McGuire technical specifications do not require these instruments operable in the modes addressed in the FSAR. I propose that the mode addressed in the FSAR be included in the technical specifications or analyses performed to assure that they are not necessary for safety.

The second type of concern is more judgemental in nature in that I submit that 10 CFR 50.36, Technical Specifications, requires that the McGuire technical specifications contain more safety restrictions; e.g. LCOs, than is presently incorporated in the McGuire or Westinghouse Standard Technical Specifications. I submit that a thorough review of the McGuire FSAR "analyses of record" would establish more restrictions; e.g. LCOs, and that those restrictions should be in the McGuire technical specifications or that analyses should be performed (specifically for McGuire or generic enveloping analyses) to provide the legal/technical basis that the present technical specifications are adequate and appropriately implement 10 CFR 50.36, 50.46, and the GDC (Appendix A). Examples of FSAR limitations that should be so addressed are as follows:

1) Control Rod Insertion and Reactor Trip System Operability Limits

FSAR analyses assume certain control rod positions and reactor protection system availability when in modes 3 through 6. The McGuire technical specifications do not impose any limitations on control rod position during these modes. Therefore, the positions of the McGuire control rods could be different from those used in the FSAR analyses and could result in less conservative reactor protection for non LOCA events. I propose that the McGuire technical specifications include either limitations on control rod positions or a revision and re-validation of the availability of the reactor protection system, during modes 3 though 6.

2) RCS Loop Operability Limits

The FSAR analyses requires that an RCS loop be available when the plant is in mode 4 to assure decay heat removal during a single failure event; i.e. an RCS/decay heat removal system isolation valve. The McGuire technical specifications do not require an RCS loop to be operable in this mode (4). I propose to determine the need for RCS loop(s) operability by reviewing and/or performing analyses of accidents during cooldown to establish a more reliable basis than is currently available in the FSAR for the current LCOs in the technical specifications.

3) Thermal-Hydraulic Limits

The FSAR specifies certain thermal hydraulic parameters; e.g., RCS pressure, temperature and pressurizer water level, as initial conditions for various accident analyses. The McGuire technical specifications do not adequately specify these conditions. There is a need to clarify and verify the present specifications which could allow reactor conditions that could be less conservative then the design bases. I propose that Table 3.2-1 and Section 2 need to be revised to more accurately reflect the FSAR programmed operating conditions and eliminate ambiguities.

The third type of concern involves internal staff practices for reviewing and issuing the technical specifications when licensing a reactor. Based on my McGuire experience, I submit that the "safety review" of the RSB section of the "proof and review" technical specifications, which permitted start up of the plant by others, was inadequate and not properly justified and documented as required by 10 CFR. My review shows that a thorough review of the McGuire FSAR "analyses of record" indicates significant inconsistencies with the McGuire technical specifications (and its parent Westinghouse Standard Technical Specifications). I propose that responsible technical branches work more closely with the SSPB/DL group during the entire licensing review period, and that the staff adopt improved internal administrative procedures to document reviews that justify the adequacy of the final issued technical specifications. I suggest that the staff internally use a 10 CFR 50.59 methodology for its technical specification reviews to confirm that the technical specifications maintain the reactor within the FSAR safety analysis envelope and clearly articulate and justify the rationale for any less restrictive criteria.

JAN 26 1984

I believe my above description describes how my December 7, 1983 DPO differs from the existing staff positions concerning the RSB Section of McGuire proof and review technical specifications. Supporting documents are attached as follows:

Attachment 2: My draft S R for the McGuire Technical Specifications (dated June 15, 1983)

Attachment 3: My proposed McGuire Technical Specifications (dated June 15, 1983)

R. Licciardo

Robert B. A. Licciardo
Reactor System Branch
DSI, NRR

Attachments
As Stated

cc wo/attachments:
H. Denton
N. Lauben
T. Novak
E. Adensam
R. Mattson
D. Eisenhut
B. Cotter
A. Rosenthal
R. Birkel
F. Miraglia
C. Thomas

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R. Licciardo DPO File
B. Sheron
R. Houston
D. Brinkman