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July 19, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Byron Generating Station Units 1 and 2
Technical Specifications
NRC Docket Nos. 50-454 and 50-455

Reference (a): December 16, 1983 memorandum from Cecil O. Thomas.

Dear Mr. Denton:

This is to provide additional comments and suggestions regarding the proof and review version of the Byron 1 Technical Specifications that was distributed in reference (a). NRC review of the specific changes proposed here is necessary before the Technical Specifications can be finalized.

Attachments A through E to this letter contain marked-up pages of various sections of the Technical Specifications. A summary explanation of the changes is provided for each attachment. Justifications are provided where appropriate. Some of the changes address questions raised by the NRC Staff in the review of changes previously proposed. We understand that the NRC will review each of these proposed changes and inform Commonwealth Edison of their acceptability.

Please direct any questions you may have regarding this matter to this office.

One signed original and fifteen copies of this letter and the attachments are provided for NRC review.

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator

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cc: Byron Resident Inspector

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ATTACHMENT A
(Section 3/4.3)

Circled items noted in this attachment have been previously submitted.

1) Table 3.3-2 (pg. 3/4 3-7) Reactor Trip System Instrumentation Response Times (RSB question 16).

   Item 7 Overtemperature ΔT response time should be changed from "≤ 4 seconds*" to "≤ 4 seconds*#".

   Add the note to the bottom of the page "# thermal lag and RTD bypass manifold delay time are not included."

   This change is provided for clarification of what the overtemperature ΔT response time includes.

2) Table 3.3-3 (pg. 3/4 3-14) Engineered Safety Features Actuation System Instrumentation (RSB question 9).

   To item 1.c. Containment Pressure - High - 1 add mode "4" to the applicable Modes column.

   Table 4.3-2 (pg. 3/4 3-33) Engineered Safety Features Actuation System Instrumentation .

   To item 1.c. Containment Pressure-High-1 add mode "4" to the "Modes for which surveillance is required" column.

   This change is necessary because Automatic Safety Injection will occur in mode 4 on Containment Pressure - High - 1.

3) Table 3.3-3 (pg. 3/4 3-19) Engineered Safety Features Actuation System Instrumentation (RSB question 35).

   Item 9.b. Reactor Trip, P-4 change the Total No. of channels from "2" to "4-2/Train" change the channels to Trip from "2" to "2/Train" change the minimum channels operable from "2" to "2/Train".

   This change is provided for clarification. There is a total of 4 breakers with 2 breakers per train.

4) Table 3.3-5 (pg. 3/4 3-29, 30, 31 and 32) Engineered Safety Features Response Times (RSB question 17).

   The Response Time Table has been updated and verified to be consistent with numbers used in the Safety Analysis. It is included in this package.

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*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	≤ ~~8~~ 4 seconds* #
8. Overpower ΔT	N.A.
9. Pressurizer Pressure-Low (Above P-7)	≤ 2 seconds
10. Pressurizer Pressure-High	≤ 2 seconds
11. Pressurizer Water Level-High (Above P-7)	N.A.

*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.

\# Thermal lag and RTD bypass manifold delay time are not included.

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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, Start Diesel Generators, Containment Cooling Fans, and Essential 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-1	3	2	2	1, 2, 3, 4	15*
d. Pressurizer Pressure-Low (Above P-9)	4	2	3	1, 2, 3#	19* ~~20*~~
e. Steam Line Pressure-Low (above P-11)	3/stm. gen.	2/stm. gen. any steam line	3/stm. gen.	1, 2, 3#	15*
2. Containment Spray					
a. Manual Initiation	2 pair	1 pair	2 pair	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	4	2	3	1, 2, 3	16

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TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generators, Containment Cooling Fans, and Essential Service Water)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3,
e. Steam Line Pressure-Low (Above P-11)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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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
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Reactor Trip, P-4	4- 2/Train	2/Train	2/Train	1, 2, 3	22
c. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	20
d. Steam Generator Water Level, P-14 (High-High)	4 ~~2~~/stm. gen.	2/stm. gen. in any operating stm. gen.	2/stm. gen. in each operating stm. gen.	1, 2, 3	20

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TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Containment Vent Isolation	N.A.
f. Steam Line Isolation	N.A.
f. Feedwater Isolation	N.A.
h. Auxiliary Feedwater	N.A.
i. Essential Service Water	N.A.
j Containment Cooling Fans	N.A.
k. Start Diesel Generator	N.A.
2. Containment Pressure-High-1	
a. Safety Injection (ECCS)	$\le 27^{(1)}/12^{(2)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\le 7^{(3)}$
3) Phase "A" Isolation	$\le \cancel{17^{(2)}/27^{(1)}}\ 2^{(6)}$
4) Containment Vent Isolation	$\le \cancel{25^{(1)}/10^{(2)}}\ 7$
5) Auxiliary Feedwater	≤ 60
6) Essential Service Water	$\le \cancel{32^{(2)}/47^{(1)}}\ 42^{(1)}$
7) Containment Cooling Fans	$\le \cancel{55^{(1)}/40^{(2)}}\ 40^{(1)}$
8) Start Diesel Generator	$\le \cancel{10}\ 12$
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\le 27^{(1)}/12^{(2)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\le 7^{(3)}$
3) Phase "A" Isolation	$\le \cancel{17^{(2)}/27^{(1)}}\ 2^{(6)}$
4) Containment Vent Isolation	$\le \cancel{25^{(1)}/10^{(2)}}\ 7$
5) Auxiliary Feedwater	≤ 60
6) Essential Service Water	$\le \cancel{47^{(1)}/32^{(2)}}\ 42^{(1)}$
7) Containment Cooling Fans	$\le \cancel{55^{(1)}/40^{(2)}}\ 40^{(1)}$
8) Start Diesel Generator	$\le \cancel{10}\ 12$

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 22(4)/12(5)
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 7(3)
3) Phase "A" Isolation	≤ ~~17(2)/27(1)~~ 2(6)
4) Containment Vent Isolation	≤ ~~25(1)/10(2)~~ 7
5) Auxiliary Feedwater	≤ 60
6) Essential Service Water	≤ ~~47(1)/32(2)~~ 42(1)
7) Containment Cooling Fans	≤ ~~55(1)/40(2)~~ 40(1)
8) Start Diesel Generator	≤ ~~10~~ 12
b. Steam Line Isolation	≤ ~~5~~ 7
5. Containment Pressure-High-3	
a. Containment Spray	≤ ~~57(1)/45(2)~~ 45(1)
b. Phase "B" Isolation	≤ ~~75(1)/65(2)~~ 22(1)/12(2)
6. Steam Generator Water Level-High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 7(3)
7. Steam Generator Water Level-Low-Low	
a. Motor-Driven Auxiliary Feedwater Pump	≤ 60
b. Diesel-Driven Auxiliary Feedwater Pumps	≤ 60
8. Containment Pressure-High-2	
Steam Line Isolation	≤ 7
9. RWST Level-Low-Low Coincident with Safety Injection	
Automatic Opening of Containment Sump Suction Isolation Valves	≤ ~~25(2)/265(1)~~ 100

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. Undervoltage RCP Bus	
a. Motor-Driven Auxiliary Feedwater Pump	≤ 60
b. Diesel-Driven Auxiliary Feedwater Pump	≤ ~~11~~ 60
11. Division 11 ESF Bus Undervoltage	
Motor-Driven Auxiliary Feedwater Pump	≤ 60
12. Loss of Power	
a. ESF Bus Undervoltage ~~(Electromechanical Relaying)~~	≤ ~~10~~ 1.9
b. Grid Degraded Voltage ~~(Solid State Relaying)~~	≤ ~~10~~ 310 ± 30 delay
13. Steam Line Pressure - Negative Rate-High (Below P-11)	
Steam Line Isolation	≤ 7
14. Phase "A" Isolation	
Containment Vent Isolation	≤ ~~5~~ 7
15. Auxiliary Feedwater Pump Suction Pressure-Low-Low	
Automatic Switchover to ESW	N.A.

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TABLE 3.3-5 (Continued)

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting and sequence loading delay not included. Offsite power available.

(3) ~~Air~~ Hydralic operated valves.

(4) Diesel generator starting and sequence loading delay included. ~~RHR pumps not included~~. Only centrifugal charging pumps included.

(5) Diesel generator starting and sequence loading delays not included. Offsite power available. ~~RHR pumps not included~~. Only centrifugal charging pumps included.

~~(6) Sequence delays not included.~~

(6) Does not include valve closure time.

ATTACHMENT B
(Section 3/4.4)

Circled items noted in this attachment have been previously submitted.

1) Section 3.4.1.2 and 4.4.1.2.3 Reactor Coolant System (pg. 3/4 4-2) RSB question 28).

   Change the first line of the LCO to read "At least three of the reactor coolant loops listed below shall be OPERABLE and at least two of these reactor coolant loops shall be in operation:*".

   Change Action statement "b" to "c".

   Add Action statement b as follows:
   "b. With only one reactor coolant loop in operation, restore at least two loops to operation within 72 hours or be in HOT SHUTDOWN within the next 12 hours."

   Change Surveillance Requirement 4.4.1.2.3 to read "At least two reactor coolant loops..."

   These changes are necessary because in mode 3, the reactor coolant loops provide sufficient heat removal capability for removing decay heat. Two loops must be in operation to meet the DNB design basis when considering an inadvertent control rod withdrawal event from subcritical. Single failure considerations require that three loops be operable.

2) Section 3.4.1.3 Reactor Coolant System (pg. 3/4 4-3) (RSB question 1).

   In note ** at the bottom of the page change "380°F" to 350°F".

   This change is necessary to maintain consistency with revised Tech Spec 3.4.9.3 also in this package.

3) Section 3.4.1.4.1 (pg. 3/4 4-5) (RSB question 1).

   In note ## at the bottom of the page change "380°F" to "350°F".

   This change is necessary to maintain consistency with revised Tech Spec 3.4.9.3 also in this package.

4) Section 3.4.4 Relief Valves (pg. 3/4 4-10) (RSB question 5).

   Delete the Action statements and replace with Insert "A" which reads as follows:

"<u>ACTION:</u>

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

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With both PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore each of the PORV(s) to OPERABLE status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With one or more block valve(s) inoperable, within 1 hour: 1) 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 its associated solenoid valve; and 2) apply the ACTION of b or c above, as appropriate for the isolated PORV(s).

e. The provisions of Specification 3.0.4 are not applicable."

This change is necessary so that the Pressurizer PORVs are not removed from service for an indefinite period. The revised Action statement is consistent with assumptions in the FSAR.

5) Section 3.4.9.3 Overpressure Protection Systems (pg. 3/4 4-35 and 3/4 4-37) (RSB question 1).

For Tech Spec 3.4.9.3 delete the LCO, Applicability, Action Statement and Surveillance Requirements and replace with the new attached pages 3/4 4-35 and 3/4 4-37. The revised Tech Spec will read as follows:

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. Two residual heat removal (RHR) suction relief valves each with a Setpoint of 450 psig $\pm$ 1%, or

b. Two power-operated relief valves (PORVs) with lift Setpoints that vary with RCS temperature which do not exceed the limit established in Figure 3.4-4, or

c. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2 square inches.

APPLICABILITY: MODES 4 and 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV and one RHR suction relief valve inoperable, either restore two PORVs or two RHR suction relief valves to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.

b. With both PORVs and both RHR suction relief valves inoperable, depressurize and vent the RCS through at least a 2 square inch vent within 8 hours.

c. In the event the PORVs, or the RHR suction relief valves, 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 6.7.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, or the RHR suction relief valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

"REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

a. For RHR suction relief valve 8708B:

1) By verifying at least once per 31 days that RHR RCS Suction Isolation Valve 1RH8702A is open with power to the valve operator removed, and

2) By verifying at least once per 12 hours that 1RH8702B is open.

b. For RHR suction relief valve 8708A:

1) By verifying at least once per 31 days that 1RH8701B is open with power to the valve operator removed, and

2) By verifying at least once per 12 hours that 1RH8701A is open.

c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

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*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position then verify these valves open at least once per 31 days."

This change is necessary to include the RHR suction relief valves in the Overpressure Protection Systems and to delete the requirement that the Overpressure Protection Systems be OPERABLE in mode 3 when the temperature of any RCS cold leg is less than or equal to 380°F.

The pressurizer PORVs and the cold Overpressure Protection System are not required for dealing with overpressure transients when the primary temperature is above 350°F. This is because at 350°F the Appendix G limit is more than 500 psi above the pressurizer safety valves opening setpoint of 2485 psig and this margin becomes greater with increasing primary temperatures. Hence the safety valves will prevent pressure transients from exceeding the Appendix G limits when the primary temperature is above 350°F.

6) BASES Section 3/4.4.1 Reactor Coolant Loops and Coolant Circulation (pg. B 3/4 4-1) (RSB questions 28 and 1).

Change the second paragraph of the Bases to read as follows:

"In Mode 3 two Reactor Coolant Loops provide sufficient heat removal capability for removing decay heat; however, single failure considerations require that three loops be OPERABLE."

This change is necessary to maintain consistency with revised Tech Spec 3.4.1.2.

In the last paragraph, second sentence change "380°F" to "350°F".

This change is necessary to maintain consistency with revised Tech Spec 3.4.9.2 also in this package.

7) BASES Section 3/4.4.9 Pressure Temperature Limits (pg. B 3/4 4-15) (RSB question 1).

In the second paragraph first line after, The Operability of two PORVs add ",or two RHR suction valves,".

In the same paragraph change "380°F" to "350°F".

Add the following as the last paragraph to Bases Section 3/4.4.9 "RHR RCS suction isolation valves 8701A and 8702A are interlocked with an "A" train wide range pressure transmitter and valves 8701B and 8702B are interlocked with a "B" train wide range pressure transmitter. Removing power from valves 8701B and 8702A, prevents a single failure from inadvertently isolating both RHR suction relief valves while maintaining RHR isolation capability for both RHR flow paths."

This change is necessary to update the bases consistent with the revised Tech Spec 3.4.9.3.

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REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

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

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~~ or

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 only one reactor coolant loop in operation, restore at least two loops to operation within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

c. ~~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 narrow range water level to be greater than or equal to 41% at least once per 12 hours.

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

*All Reactor Coolant pumps may be deenergized 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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HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

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,**

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

e. RHR Loop A, and/or

f. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required reactor coolant and/or RHR loop(s) 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.

*All reactor coolant pumps and RHR pumps may be deenergized 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.

**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~~ 350°F unless 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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REACTOR COOLANT SYSTEM

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

b. The secondary side narrow range water level of at least two steam generators shall be greater than 41%.

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.

*The RHR pump may be deenergized 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.

#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 ~~180°F~~ unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

350°F

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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.

ACTION:

Replace with Insert "A"

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 its control power; 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. of 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.~~

219

Insert "A"

ACTION:

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

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With both PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore each of the PORV(s) to OPERABLE status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With one or more block valve(s) inoperable, within 1 hour: 1) 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; and 2) apply the ACTION of b or c above, as appropriate for the isolated PORV(s).

e. The provisions of Specification 3.0.4 are not applicable.

DELETE AND REPLACE WITH NEW PAGE 3/4 4-35

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LIMITING CONDITION FOR OPERATION

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

& lift setpoint not to exceed the limits of

a. Two power-operated relief valves (PORVs) with ~~nominal setpoints~~ which vary with RCS temperature as shown on Figure 3.4-4, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2 square inches.

APPLICABILITY: MODE ~~4~~ 3 when the temperature of any RCS cold leg is less than or equal to 380°F, MODE 5 and MODE 6 with the reactor vessel head on.

Mode 4

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.

b. With both PORVs inoperable, depressurize and vent the RCS through at least a 2 square inch vent 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 6.7.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

1, 2, 8, 9

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. Two residual heat removal (RHR) suction relief valves each with a Setpoint of 450 psig ± 1%, or

b. Two power-operated relief valves (PORVs) with lift Setpoints that vary with RCS temperature which do not exceed the limit established in Figure 3.4-4, or

c. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2 square inches.

APPLICABILITY: MODES 4 and 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV and one RHR suction relief valve inoperable, either restore two PORVs or two RHR suction relief valves to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.

b. With both PORVs and both RHR suction relief valves inoperable, depressurize and vent the RCS through at least a 2 square inch vent within 8 hours.

c. In the event the PORVs, or the RHR suction relief valves, 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 6.7.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, or the RHR suction relief valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

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SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

9

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

a. For RHR suction relief valve 8708B:

1) By verifying at least once per 31 days that RHR RCS Suction Isolation Valve 1RH8702A is open with power to the valve operator removed, and

2) By verifying at least once per 12 hours that 1RH8702B is open.

b. For RHR suction relief valve 8708A:

1) By verifying at least once per 31 days that 1RH8701B is open with power to the valve operator removed, and

2) By verifying at least once per 12 hours that 1RH8701A is open.

c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

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*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

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BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR ~~above 1.30~~ during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within ~~1~~ hours.
6

ABOVE THE APPLICABLE SAFETY ANALYSIS DNBR

two

In MODE 3, ~~a single~~ reactor coolant loops provide~~s~~ sufficient heat removal capability for removing decay heat; however, single failure considerations require that ~~two~~ loops be OPERABLE.

three

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

350°F

The restrictions on starting a reactor coolant pump with one or more RCS cold legs less than or equal to ~~380°F~~ are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

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PRESSURE/TEMPERATURE LIMITS (Continued)

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, or two RHR suction valves, or an RCS vent opening of at least 2 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to ~~350°F~~ 350°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a ~~HPSI~~ centrifugal charging pump and its injection into a water solid RCS.

These two scenarios are analyzed to determine the resulting overshoots assuming a single PORV actuation with a stroke time of 2.0 seconds from full closed to full open. Figure 3.4-4 is based upon this analysis and represents the maximum allowable PORV variable setpoint such that, for the two overpressurization transients noted, the resulting pressure will not exceed the nominal 10 effective full power years (EFPY) Appendix G reactor vessel NDT limits.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

RHR RCS suction isolation valves 8701 A and 8702 A are interlocked with an "A" train wide range pressure transmitter and valves 8701 B and 8702 B are interlocked with a "B" train wide range pressure transmitter. Removing power from valves 8701 B and 8702 A, prevents a single failure from inadvertently isolating both RHR suction relief valves while maintaining RHR isolation capability for both RHR flow paths. 9,8

ATTACHMENT C
(Boron Dilution Protection Package)

1) Table 3.3-1 Reactor Trip System Instrumentation (pg. 3/4 3-2).

Item 6.b. delete the words "(Trip Brk Closed)" and the Applicable Modes shall be "3, 4, 5" not "3*, 4*, 5*".

Change the Action statement from 10 to 5. Delete item 6.c. in its entirety.

This change is necessary for the addition of the Boron Dilution Protection System (BDPS).

2) Table 3.3-1 Action Statements (pg. 3/4 3-6).

Delete Action 5 and insert the following:

"Action 5 - With the number of OPERABLE channels one less than the Minimum Channels Operable requirement,

a. Suspend all operations involving a reduction in boron concentration of the Reactor Coolant System <u>and</u>

b. Restore the inoperable channel to OPERABLE status within 48 hours <u>or</u> open the Reactor trip breakers, <u>and either</u>

1) Verify compliance with the SHUTDOWN margin requirements of Specification 3.1.1.1 or 3.1.1.2 as applicable within 1 hour and at least once per 4 hours thereafter, <u>or</u>

2) Close and secure in position by mechanical stops or by removal of air or electrical power, valves 1CV-8439, 1CV11B, 1CV8428, 1CV-8441 and 1CV-8435 within 4 hours and verify that they are secured in the closed position at least once per 12 hours thereafter."

This change is necessary for the addition of the BDPS.

3) Table 4.3-1 (pg. 3/4 3-9) Reactor Trip System Instrumentation Surveillance Requirements.

Change #6. to

"6.a. Source Range, Neutron Flux S R(4,5) S/U(1) N.A. N.A. 2##, 3, 4, 5
b. Source Range, Neutron Flux N.A. R(12) M(9) N.A. N.A. 3, 4, 5".

Item 6.b. was separated from 6.a. because of its reference to the BDPS which is only applicable in modes 3, 4, 5.

4) (pg. 3/4 9-1) Boron concentration.

4.9.1.3 change "1CV8430" to "1CV111B, 1CV8428".

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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					
a. High Setpoint	4	2	3	1, 2	2#
b. Low Setpoint	4	2	3	1###, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown ~~(Trip Brk Closed)~~	2	1	2	~~3*, 4*, 5*~~ 3, 4, 5	~~10~~ 5
~~c. Shutdown (Trip Brk Open)~~	~~2~~	~~0~~	~~1~~ 2	~~3, 4, 5~~	~~6~~
7. Overtemperature ΔT					
~~a. Four Loop Operation~~	4	2	3	1, 2	6#
~~b. Three Loop Operation~~	~~**~~	~~**~~	~~**~~	~~**~~	~~**~~
8. Overpower ΔT					
~~a. Four Loop Operation~~	4	2	3	1, 2	6#
~~b. Three Loop Operation~~	~~**~~	~~**~~	~~**~~	~~**~~	~~**~~

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

~~ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.~~

Insert "AA"

ACTION 6 - 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; and

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.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 10 - 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 open the Reactor trip breakers within the next hour.

ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 1 hour.

Insert "AA"

ACTION 5 With the number of OPERABLE channels one less than the Minimum Channels Operable requirement,

a. Suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and

b. Restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breaker and either:

   1) Verify compliance with the SHUTDOWN Margin requirements of Specification 3.1.1.1 or 3.1.1.2 as applicable within 1 hour and at least once per 4 hours thereafter ~~[illegible]~~ or

   2) Close and secure in position by mechanical stops or by removal of air or electrical power, valves 1CV111B, 1CV8428, 1CV-8439, 1CV-8441 and 1CV-8435 within 4 hours and verify that they are secured in the closed position at least once per 12 hours thereafter

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	M	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	M	N.A.	N.A.	1###, 2 X
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1),M	N.A.	N.A.	1###, 2 X
6. a. Source Range, Neutron Flux	S	R(4, 5, 12)	S/U(1),M(9)	N.A.	N.A.	2##, 3, 4, 5
b. Source Range, Neutron Flux	N/A	R(12)	M(9)	N/A	N/A	3, 4, 5
7. Overtemperature ΔT	S	R(13)	M	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	M	N.A.	N.A.	1, 2
9. Pressurizer Pressure-Low (above P-7)	S	R	M	N.A.	N.A.	1
10. Pressurizer Pressure-High	S	R	M	N.A.	N.A.	1, 2
11. Pressurizer Water Level-High (above P-7)	S	R	M	N.A.	N.A.	1
12. ~~Reactor Coolant Flow-Low~~	S	R	M	N.A.	N.A.	1

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3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

unisolated

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 either:

(1)

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

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

APPLICABILITY: MODE 6*.

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.

BORATION RATE

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.

1CV111B, 1CV8428, ~~1CV8430~~, 1CV8441, 1CV8435 and 1CV8439

4.9.1.3 Valves ~~[illegible], [illegible], [illegible], and [illegible]~~ shall be verified closed and secured in position by mechanical stops or by removal of air or electrical power at least once per 31 days.

*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.

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ATTACHMENT D

Circled items noted in this attachment have been previously submitted.

1) Pg. 3/4 9-10 Containment Purge Isolation System.

   4.9.9 Delete "Manual Initiation and on".

   There is no specific manual initiation for a Containment Purge Isolation. It occurs on a Manual Phase A or a Manual Phase B. This is covered in 3/4.3.2 Engineered Safety Features Actuation System Instrumentation.

3/4.9.9 CONTAINMENT PURGE ISOLATION ~~VENTILATION~~ SYSTEM

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LIMITING CONDITION FOR OPERATION

3.9.9 The Containment PURGE ISOLATION ~~Ventilation~~ System shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a. With the Containment PURGE ISOLATION ~~Ventilation~~ System inoperable, close each of the purge valves providing direct access from the containment atmosphere to the outside atmosphere.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment PURGE ISOLATION ~~Ventilation~~ System shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment purge isolation occurs on manual initiation and on a ESF test signal from each of the containment radiation monitoring instrumentation channels.

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ATTACHMENT E

Circled items noted in this attachment have been previously submitted.

1) Table 2.2-1 (pg. 2-6), Table 3.3-3 (pg. 3/4 3-17), Table 3.3-4 (pg. 3/4 3-25, 3-26, 3-27), Table 4.3-2 (pg. 3/4 3-36).

   The above referenced tables have been revised as noted on the attached sheets.

2) Table 3.3-3 (pg. 3/4 3-16), Table 3.3-4 (pg. 3/4 3-24), Table 4.3-2 (pg. 3/4 3-35).

   Item 5.c. has been added to each of these tables as shown on the attached sheets. This change is necessary because a Safety Injection signal will cause a turbine trip and feedwater isolation.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	2.5	1.77	0.6	≥90% of loop design flow*	≥89.2% of loop design flow*
13. Steam Generator Water Level Low-Low	27.1	~~27.18*~~ 18.28	1.5	≥40.8% of narrow range instrument span	39.1% ≥~~36%~~ of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	3.3	0	0	~~4920~~ 5268 ≥~~4830~~ volts - each bus	4728 ≥~~4760~~ volts - each bus
15. Underfrequency - Reactor Coolant Pumps	14.4	15.36 ~~13.2~~ ~~0~~	0	57.0 ≥~~57.5~~ Hz	~~57.6~~ 56.5 ≥~~57.6~~ Hz
16. ~~Turbine Trip~~ Emergency Trip Header					
a. ~~Low Fluid Oil~~ Pressure	N.A.	N.A.	N.A.	540 ≥~~X~~ psig	~~NA~~ 540 ≥~~X~~ psig
b. Turbine Stop Valve Closure Throttle	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	N.A.	N.A.

*Loop design flow = 94,400 ~~96,700~~ gpm

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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
6. 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	15*
2) Start Diesel-Driven Pump	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	15*
d. Undervoltage - RCP Bus-Start Motor-Driven Pump and Diesel-Driven Pump	4-1/bus	2 of 4	3	1, 2	19*
e. Safety Injection - Start Motor-Driven Pump and Diesel-Driven Pump	See Item 1. above for all Safety Injection initiating functions and requirements.				
f. Division 11 ESF Bus Undervoltage-Start Motor-Driven Pump (Start as part of DG sequencing)	2	~~1~~ 2	2	1, 2, 3	18

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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
6. Auxiliary Feedwater (Continued)					
g. Auxiliary Feedwater Pump Suction Pressure-Low (Transfer to Essential Service Water)	X2	2	2	1, 2, 3	15*
7. Automatic Opening of Containment Sump Suction Isolation Valves					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	21
b. RWST Level - Low-Low Coincident With Safety Injection	4	2	3	1, 2, 3, 4	16
	See Item 1. above for Safety Injection initiating functions and requirements.				
8. Loss of Power					
a. ESF Bus Undervoltage ~~(Electromechanical Relaying)~~	2/Bus	2/Bus	1/Bus	1, 2, 3, 4	19*
b. Grid Degraded Voltage ~~(Solid State Relaying)~~	2/Bus	2/Bus	1/Bus	1, 2, 3, 4	19*

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level-Low-Low-Start Motor-Driven Pump and Diesel-Driven Pump	27.1 ~~30.0~~	18.28 ~~27.18~~	1.5	40.8 ≥ ~~41%~~ of narrow range instruement span	39.1 ≥ ~~40%~~ of narrow range instrument span
d. Undervoltage-RCP Bus-Start Motor Driven Pump and Diesel-Driven Pump	N.A.	N.A.	N.A.	5268 ~~4920 volts~~ ~~≥ 70% RCP bus voltage (4890 volts)~~	~~4760~~ ~~≥ 69% RCP bus voltage (4674 volts)~~ 4728
e. Safety Injection-Start Motor-Driven Pump and Diesel-Driven Pump	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)					
f. Division 11ESF Bus Undervoltage-Start Motor-Driven Pump	N.A.	N.A.	N.A.	~~N.A.~~ 2870 volts	~~N.A.~~ 2730 volts
g. Auxiliary Feedwater Pump Suction Pressure-Low (Transfer to Essential Service Water)	N.A.	N.A.	N.A.	N.A.	N.A.
7. Automatic Opening of Containment Sump Suction Isolation Valves					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level-Low-Low Coincident with Safety Injection	N.A.	N.A.	N.A.	N.A.	N.A.
	See Item 1. above for Safety Injection Trip Setpoints and Allowable Values.				

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Loss of Power					
a. ESF Bus Undervoltage ~~(Electromechanical Relaying)~~	N.A.	N.A.	N.A.	28.? volts 1.8s time delay	≥ 2730 volts ~~2870 volts ± 143~~ ~~1.8 ± 0.1s time delay~~ ≤ 1.9s time delay
b. Grid Degraded Voltage ~~(Solid State Relaying)~~	N.A.	N.A.	N.A.	3804 volts ~~0s time delay~~ w/310s delay	≥ 3728 volts ~~3804 volts ± 76~~ ~~0 ± 0.0s time delay~~ w/310 ± 30s delay
9. Engineered Safety Feature Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤ ~~1950~~ 1930 psig	≤ 2050 psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
c. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	550°F	≥ 548°F and ≤ 552°F
d. Steam Generator Water Level, P-14 (High-High)	See Item 5.b. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.				

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater (Continued)								
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
f. Division 11ESF Bus Undervoltage	N.A.	R	N.A.	R	N.A.	N.A.	N.A	1, 2, 3
g. Auxiliary Feedwater Pump Suction Pressure-Low	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3
7. Automatic Opening of Containment Sump Suction Isolation Valves								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. RWST Level-Low-Low Coincident With Safety Injection	N.A.	R	N.A.	N.A.	M	N.A.	N.A.	1, 2, 3, 4
	See Item 1. above for all Safety Injection Surveillance Requirements							
8. Loss of Power								
a. ESF Bus Undervoltage ~~(Electromechanical Relays)~~	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Grid Degraded Voltage ~~(Solid State Relays)~~	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4

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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
4. Steam Line Isolation					
a. Manual Initiation 1) Individual	1/stm line	1/stm line	1/operating steam line	1, 2, 3	23 ~~24~~
2) System	2	1	2	1, 2, 3	~~23~~ 22
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Containment Pressure-High-2	3	2	2	1, 2, 3	15*
d. Steam Line Pressure-Low (above P-11)	3/stm. gen.	2/stm. gen. any steam line	3/stm. gen.	1, 2, 3#	15*
e. Steam Line Pressure - Negative Rate-High (below P-11)	3/stm. gen.	2/stm. gen. any steam line	2/stm. gen.	3##, 4	15*
5. Turbine Trip & Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2,	21
b. Steam Generator Water Level-High-High (P-14)	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2	19*

c. Safety Injection   See Item 1. above for all Safety Injection initiating function and requirements.

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-2	7.7 ~~6.0~~	0.71	1.5	8.2 ≤ ~~10.0~~ psig	9.2 ≤ ~~12.0~~ psig
d. Steam Line Pressure-Low (Above P-11)	21.2 ~~14.8~~	14.81 ~~10.71~~	1.5	≥ 640 psig	617 ≥ ~~610~~ psig*
e. Steam Line Pressure-Negative Rate-High (Below P-11)	8.0	0.5	0	≤ -100 psi/sec	≤ -111.5 psi ~~≤ -110.0 psi/sec~~**
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)	6.0 ~~4.0~~	4.28 ~~2.18~~	1.5	81.4 ≤ ~~82%~~ of narrow range instrument span	82.? ≤ ~~83%~~ of narrow range instrument span

c. Safety Injection See Item 1 above for all Safety Injection Setpoints and Allowable Values.

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure-High-2	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low (Above P-11)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure - ~~Negative Rate~~ - High (Below P-11)	S	R	M	N.A.	N.A.	N.A.	N.A.	3, 4
5. Turbine ~~Trip and Feedwater~~ Isolation								
a. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level-High-High (P-14)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
c. Safety Injection	See Item 1 above for all Safety Injection Surveillance requirements.							
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relay	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level-Low-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Undervoltage-RCP Bus	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2

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