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September 20, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20005

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

Dear Mr. Denton:

Per discussions with Mr. Calvin Moon and numerous NRC reviewers, listed below and attached are those pages which Commonwealth Edison understands have been accepted by the principal reviewers as a result of our meeting in Bethesda, Maryland Sept. 18,1984 thru Sept. 20,1984.

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Also included with this letter are the following pages which are being formally submitted for your review. These page changes were discussed with the individual reviewers during the meetings and commonwealth Edison feels that mutual concurrence was achieved:

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1/1

The following open items remain, as of 9/20/84, as part of the continued Commonwealth Edison/NRC Tech Spec review. The item number identified below corresponds to the Attachment number of the handwritten list provided to Calvin Moon by Commonwealth Edison.

Attachment #	Tech Spec	Subject	Status
1	4.7.7.d.3	-1/4" pressure	Conference call scheduled for 9/24/84
2	4.3.4.2	Turbine Overspeed	NRC awaiting formal letter from CECo(9/24/84)
8	Table 4.11-2	Gaseous Waste Analyses	Conference call scheduled for 9/24/84
17	4.7.1.2.3 4.7.5.3 4.7.10.1.3 4.8.1.1.2	Diesel Oil Sampling	NRC reviewing
25	3.7.6 3.7.7 3.9.12	Ventilation	NRC/CECo telcon set for 9/24/84
27		RSB Questions	NRC to determine impact on Tech Specs
29	3.7.5	Ultimate Heat Sink	a.) NRC/CECo Mtg. set for 9/24/84 b.) NRC to determine how contingency plan affects Tech Specs
30	Misc.	Omissions, typo's from Final Draft	NRC to provide pages
31	Table 3.4-1	RHR Suction Valves	CECo reviewing NRC changes
34	3/4.3-2	Undervoltage Surveillance Int.	NRC reviewing

Commonwealth is proceeding with the Tech Spec Certification process based on the August 28 Final Draft, the 71 pages of corrections, and the attached marked up page changes.

Commonwealth requests an expeditious NRC closure of the remaining open items to support our Tech Spec Certification.

Very truly yours,

Dale E. Shelton for

T. R. Tramm
Nuclear Licensing Administrator

cc: Byron Resident Inspector
Senior Tech Spec Coor.
Calvin Moon

LIMITING SAFETY SYSTEM SETTINGS

FINAL DRAFT

9/19/84

~~AUG 2 8 1984~~

BASES

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

P-6 On increasing power, P-6 allows the manual block of the Source Range Reactor trip (i.e., prevents premature block of Source Range trip), provides ~~a~~ an automatic backup block for Source Range Neutron Flux doubling, and the manual block that de-energizes the high voltage to the source range detectors. On decreasing power, Source Range Level trips and Neutron Flux Doubling circuits are automatically reactivated and high voltage restored.

P-7 On increasing power, P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, more than one reactor coolant pump breaker open, reactor coolant pump bus undervoltage and underfrequency, Turbine trip, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. On decreasing power, the P-8 automatically blocks the single loop low flow trip.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range Reactor trip and the Low Setpoint Power Range Reactor trip; and automatically blocks the Source Range Reactor trip and de-energizes the Source Range high voltage power. On decreasing power, the Intermediate Range Reactor trip and the Low Setpoint Power Range Reactor trip are automatically reactivated. Provides input to P-7.

provides an automatic backup function to

and Source Range high voltage to the detectors is restored if power decreases below the P-6 setpoint

P-13 Provides input to P-7.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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
7. Automatic Opening (Continued)								
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	~~N.A.~~ S	R	~~N.A.~~ M	N.A.	~~N~~ N.A.	N.A.	N.A.	1, 2, 3, 4
Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements							
8. Loss of Power								
a. ESF Bus Undervoltage	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Grid Degraded Voltage	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Feature Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
c. Low-Low T_{avg}, P-12	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Generator Water Level, P-14 (High-High)	S	R	M	N.A.	M(1)	M(1)	Q	1, 2, 3

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

OK pw 9/18/84 MTG. [illegible]

FINAL DRAFT AUG [illegible] 9/19/84

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REACTOR COOLANT SYSTEM

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HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 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:*

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 only one reactor coolant loop in operation, restore at least two loops to operation within 72 hours or open reactor trip breakers within 1 hour.

c. 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 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.

** See Special Test Exception 3.10.4

Discussed with Cal Moon

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3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water level of between ~~34%~~ 31 and ~~66%~~ 63, ×

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

d. A nitrogen cover-pressure of between 617 and 662 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one 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 within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

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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 70 gallons by verifying the boron concentration of the accumulator solution,

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that power to the isolation valve operator is disconnected from the circuit by ~~removing the control fuses~~.

verifying the associated MCC compartment is out of service.

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE at least once per 18 months by the performance of a CHANNEL CALIBRATION.

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3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen monitors shall be OPERABLE.*

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

b. With both hydrogen monitors inoperable, restore at least one monitor to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK and a check that the monitor is in standby mode at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and at least once per 92 days ~~on a STAGGERED TEST BASIS~~ by performing a CHANNEL CALIBRATION using ~~sample gas containing:~~

~~a. Nominal one volume percent hydrogen, balance nitrogen, and~~

~~b. Nominal four volume percent hydrogen, balance nitrogen.~~

*The monitors must be in standby mode to meet the requirement in NUREG-0737, Item II.F.1.6.

five gas samples which shall cover the range from zero volume percent hydrogen (100% N_2) to greater than 20 volume percent hydrogen, balance nitrogen.

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PLANT SYSTEMS

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3/4.7.12 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.12 The temperature limit of each area shown in Table 3.7-6 shall not be exceeded ~~be maintained within the limits indicated in Table 3.7-6.~~ for more than 8 hours, or by more than 30°F.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 for more than 8 hours, prepare and submit to the Commission within 30 days, pursuant to Specification ~~6.7.2,~~ 6.9.2, a Special Report that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 by more than 30°F, prepare and submit a Special Report as required by ACTION a. above, and within 4 hours either return the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.12 The temperature in each of the areas shown in Table 3.7-6 shall be determined to be within its limit at least once per 12 hours.

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TABLE 3.7-6

AREA TEMPERATURE MONITORING

AREA	TEMP. °F
1. Misc. Electric Equipment and Battery Rooms	108
2. ESF Switchgear Rms	108
3. Division 12 Cable Spreading Rm	108
4. Upper and Lower Cable Spreading Rms	90
5. Diesel-Generator Rms	132
6. Diesel Oil Storage Rooms	132
7. Aux. Building Vent Exhaust Filter Cubicle	122
8. Centrifugal Charging Pump Room	122
9. Containment Spray Pump Rooms	130
10. RHR Pump Rooms	130
11. Safety Injection Pump Room	130

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TABLE 3.8-2

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION DEVICES

VALVE NUMBER	FUNCTION
0OG059	Unit 1 Suct Isol Vlv H_2 Recomb
0OG060	Unit 1 Discharge Isol Vlv H_2 Recombiner
0OG061	Unit Discharge Xtie for H_2 Recombiner
0OG062	Unit Xtie on Discharge of H_2 Recombiner
0OG063	Unit Suction Xtie for H_2 Recombiner
0OG064	Unit Suction Xtie for H_2 Recombiners
0OG065	OB H_2 Analyzer Inlet Isol Vlv
0OG066	OB H_2 Recomb Disch Isol Vlv
1OG057A	OA H_2 Recomb Disch. Isol. Valve
1OG079	H_2 Recomb Disch. Cnmt. Isol. Valve
1OG080	H_2 Recomb Suct. Cnmt. Isol. Valve
1OG081	H_2 Recomb Suction Cnmt. Isol. Valve
1OG082	OA H_2 Recomb Disch Cnmt Isol Vlv
1OG083	OA H_2 Recomb Disch Cnmt Isol Vlv
1OG084	OA H_2 Recomb Cnmt Outlet Isol Vlv
1OG085	H_2 Recomb Cnmt Outlet Isol Vlv
1AF006A	1A AF Pp SX Suct Isol Vlv
1AF006B	1B AF Pp SX Suct Dwst Isol Vlv
1AF013A	AF Mtr Drv Pmp Disch Hdr Dwst Isol Vlv
1AF013B	AF Mtr Drv Pmp Dsch Hdr Dwst Isol Vlv
1AF013C	AF Mtr Drv Pp Disch Hdr Dwst Isol Vlv
1AF013D	AF Mtr Drv Pp Disch Hdr Dwst Isol Vlv
1AF013E	AF Dsl Drv Pm Dsch Hdr Dwst Isol Vlv
1AF013F	AF Dsl Drv Pp Dsch Hdr Dwst Isol Vlv
1AF013G	AF Dsl Drv Pp Dsch Hdr Dwst Isol Vlv
1AF013H	AF Dsl Drv Pp Dsch Hdr Dwst Isol Vlv
1AF017A	1A AF Pp SX Suct Upst Isol Vlv
1AF017B	1B AF Pp SX Suct Upst Isol Vlv
1CC685	RCP Thermal Barrier Outlet Hdr Cnmt Isol Vlv
1CC9412A	CC to RH HX 1A Isol Vlv
1CC9412B	CC to RH HX 1B Isol Vlv
1CC9413A	RCP CC Supply Dwst CNMT Isol
1CC9413B	RCPs CC Supply Upst CNMT Isol
1CC9414	CC Water from RCPs Isol. Valve
1CC9415	Unit 1 Serv. Loop Isol Vlv
1CC9416	CC Wtr from RCPS Isol. Valve
1CC9438	CC Wtr from RC Pumps Thermal Bar Isol. Valve
1CC9473A	Disch Hdr X-tie Isol Vlv
1CC9473B	Disch Hdr X-tie Isol Vlv.

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION DEVICES

VALVE NUMBER	FUNCTION
1CS001A	1A CS Pp Suct from RWST
1CS001B	1B CS Pp Suction from RWST
1CS007A	CS Pp 1A Disch Line Dwst Isol Vlv
1CS007B	CS Pp 1B Disch Line Downstream Isol Vlv
1CS009A	1A Pump Suction from 1A Recirc Sump
1CS009B	1B CS Cont Recirc Sump B Suct Isol Vlv to CS
1CS019A	CS Eductor 1A Suction Conn Isol Vlv
1CS019B	CS Eductor 1B Suction Conn Isol Vlv
1CV112B	MOV VCT Outlet Upstm Isol VCT Vlv
1CV112C	MOV VCT Outlet ~~Upstm~~ Dwnstm Isol VCT Vlv
1CV112D	MOV RWST to Chg Pp Suct Hdr
1CV112E	MOV RWST to Chg Pp Suct Hdr
1CV8100	MOV RCP Seal Leakoff Hdr Isol
1CV8104	MOV Emerg Boration Vlv
1CV8105	MOV Chrg Pps Disch Hdr Isol Vlv
1CV8106	MOV Chrg Pps Disch Hdr Isol Vlv
1CV8109	MOV PD Chrg. Pp Miniflow Recirc. Vlv
1CV8110	MOV A & B Chg. pp Recirc Downstream Isol
1CV8111	MOV A & B Chg Pp Recirc Upstream Isol
1CV8112	RC Pump Seal Water Return Isol. Valve
1CV8355A	MOV RCP 1A Seal Inj Inlet to containment Isol
1CV8355B	MOV RCP 1B Seal Inj Inlet Isol
1CV8355C	MOV RCP 1C Seal Inj Isol
1CV8355D	MOV RCP 1D Seal Inj Isol
1CV8804A	MOV RHR Sys X-Tie Vlv to Chrgng Pump Suction Hdr A.B.
1RC8001A	RC Loop 1A Hot Leg Stop Valve
1RC8001B	RC Loop 1B Hot Leg Stop Valve
1RC8001C	RC Loop 1C Hot Leg Stop Valve
1RC8001D	RC Loop 1D Hot Leg Stop Valve
1RC8002A	RC Loop 1A Cold Leg Stop Valve
1RC8002B	RC Loop 1B Cold Leg Stop Valve
1RC8002C	RC Loop 1C Cold Leg Stop Valve
1RC8002D	RC Loop 1D Cold Leg Stop Valve
1RC8003A	RC Loop 1A Bypass Leg Stop Valve
1RC8003B	RC Loop 1B Bypass Leg Stop Valve
1RC8003C	RC Loop 1C Bypass Leg Stop Valve
1RC8003D	RC Loop 1D Bypass Leg Stop Valve
1RH610	RH PP 1RH01PB Recirc, Line Isol.
1RH611	RH PP 1RH01PB Recirc, Line Isol.
1RH8701A	RC Loop 1A to RHR Pump Isol. Valve

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION DEVICES

VALVE NUMBER	FUNCTION
1RH8702A	RC Loop 1C to RHR Pump Isol. Valve
1RH8701B	RC Loop 1A to RHR Pump Isol. Valve
1RH8702B	RC Loop 1C to RHR Pump Isol. Valve
1RH8716A	RH HX 1RH02AA Dwnstrm Isol Vlv
1RH8~~0~~16B x 7	RH HX 1RH02AB Dwnstrm Isol Valve
1RY8000A	Prz. Relief Isol. Valve 1A
1RY8000B	Prz. Rélief Isol. Valve 1B
1SI8801A	SI Charging Pump Disch Isol Vlv
1SI8801B	SI Charging Pump Disch Isol Vlv
1SI8802A	SI PP 1A Disch Line Dwst Cont Isol Vlv
1SI8802B	SI PP 1B Disch Line Dwst Isol Vlv
1SI8804B	SI Pump 1B Suct X-tie from RHR HX
1SI8806	SI Pumps Upstream Suction Isol
1SI8807A	SI to Chg PP Suction Crosstie Isol Vlv
1SI8807B	SI to Chg PP Suction Crosstie Isol Vlv
1SI8808A	Accum. 1A Disch. Isol. Valve
1SI8808B	Accum. 1B Disch. Isol. Valve
1SI8808C	Accum. 1C Disch. Isol. Valve
1SI8808D	Accum. 1D Disch. Isol. Valve
1SI8809A	SI RX HX 1A Dsch Line Dwst Isol Vlv
1SI8809B	SI RX HX 1B Dsch Line Dwst Isol Vlv
1SI8811A	SI Cnmt Sump A Outlet Isol Vlv
1SI8811B	SI Cnmt Sump B Outlet Isol Vlv
1SI8812A	SI Rwst to RH Pp 1A Outlet Isol Vlv
1SI8812B	SI Rwst to RH Pp 1B Outlet Isol Vlv
1SI8813	SI Pumps 1A-1B Recirc Line Dwst Isol
1SI8814	SI Pump 1A Recirc Line Isol Vlv
1SI8835	SI Pumps X-tie Disch Isol Vlv
1SI8840	SI RHR HX Disch Line Upstrm Cont Pen Isl Vlv
1SI8821A	SI PP 1A Disch Line X-tie Isol Vlv
1SI8821B	SI Pump 1B Disch Line X-tie Isol Vlv
1SI8920	SI Pump 1B Recirc Line Isol Vlv
1SI8923A	SI PP 1A Suction Isol Vlv
1SI8923B	SI Pump 1B Suct Isol Valve
1SI8924	SI Pump 1A Suction X-tie Dwnstrm Isol Vlv
1SX016B	RCFC B&D Sx Supply MOV
1SX016A	RCFC A&C SX Supply MOV
1SX027A	RCFC A&C Return
1SX027B	RCFC B&D SX Return MOV

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION DEVICES

VALVE NUMBER	FUNCTION
0SX007	CC HX Outlet Vlv
0SX063A	SX to Cont Rm Refrig Cdsr 0A
0SX063B	SX to Cont Rm Refrig Cdsr 0B
0SX146	CC Hx "0" return Vlv to Unit 1 MDCT
0SX147	CC Hx "0" return Vlv to Unit 2 MDCT
0SX157A	SX M/U Pp 0A Supply Fill to MDCT
0SX157B	SX M/U Pp 0B Supply to MDCT 0B MOV
0SX158A	SX M/U Pp 0A Supply Fill to MDCT MOV
0SX158B	SX M/U Pp 0B Supply to MDCT 0B MOV
0SX162A	MDCT 0A Bypass to basin MOV
0SX162B	MDCT 0B Bypass to basin MOV
0SX162C	MDCT 0A Bypass to basin MOV
0SX162D	MDCT 0B Bypass to basin MOV
0SX163A	MDCT 0A Riser Isol Vlv MOV
0SX163B	MDCT 0A Riser Isol Vlv MOV
0SX163C	MDCT 0A Riser Isol Vlv MOV
0SX163D	MDCT 0A Riser Isol Vlv MOV
0SX163E	MDCT 0B Riser Isol Vlv MOV
0SX163F	MDCT 0B Riser Isol Vlv MOV
0SX163G	MDCT 0B Riser Isol Vlv MOV
0SX163H	MDCT 0B Riser Isol Vlv MOV
1SX001A	1A SX Pp Suct Vlv MOV
1SX001B	1B SX Pp Suct Vlv MOV
1SX004	U-1 SX Supply to U-1 CCW HX MOV
1SX005	1B SX Pp Supply to 0 CCW HX MOV
1SX007	CC HX Outlet Vlv
1SX010	U-1 Trn A return Vlv AB
1SX011	Trn A Trn B Unit 1 return X-tie Vlv AB
1SX033	1A SX Pp Disch X-tie MOV
1SX034	1B SX Pp Disch X-tie MOV
1SX136	Unit 1 Trn B return Vlv AB
1SX150A	Sx strn drn to waste treatment bldg MOV
1SX150B	Sx strn drn to TR bldg MOV
1W0006A	CHILLED Wtr Coils 1A & 1C Supply Isol Vlv
1W0006B	CHILLED Wtr Coils 1B & 1D Supply Isol Vlv
1W0020A	CHILLED Wtr Coils 1A & 1C Return Isol Vlv
1W0020B	CHILLED Wtr Coils 1B & 1D Return Isol Vlv
1W0056A	CHILLED Water Cnmt Isol Valve
1W0056B	CHILLED Water Cnmt Isol Valve

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Special Test Exceptions

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3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of startup and PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

Replace with Calloway

Section 3/4.10.4

9/19/84

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of the following requirements may be suspended:

a. Specification 3.4.1.1 - During the performance of startup and PHYSICS TESTS in MODE 1 or 2 provided:

1) The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

2) The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

b. Specification 3.4.1.2 - During the performance of hot rod drop time measurements in MODE 3 provided at least three reactor coolant loops as listed in Specification 3.4.1.2 are OPERABLE.

APPLICABILITY: During operation below the P-7 Interlock Setpoint or performance of hot rod drop time measurements.

ACTION:

a. With the THERMAL POWER greater than the P-7 Interlock Setpoint during the performance of startup and PHYSICS TESTS, immediately open the Reactor trip breakers.

b. With less than the above required reactor coolant loops OPERABLE during performance of hot rod drop time measurements, immediately place two reactor coolant loops in operation.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

4.10.4.3 At least the above required reactor coolant loops shall be determined OPERABLE within 4 hours prior to initiation of the hot rod drop time measurements and at least once per 4 hours during the hot rod drop time measurements by verifying correct breaker alignments and indicated power availability.

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5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of [redacted] grams uranium. The initial core loading shall have a maximum enrichment of ~~3.10~~ weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of ~~3.50~~ 4.0 weight percent U-235.

less than 3.20

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. All control rods shall be hafnium, clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,257 cubic feet at a nominal T_{avg} of 588.4°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.~~1~~ The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 3.31% $\Delta k/k$ for uncertainties as described in Section 9.1 of the FSAR; and

b. A nominal 14 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.~~2~~ 1 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 423 feet 2 inches.

CAPACITY

* 5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than ~~1050~~ 1060 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

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UNIT STAFF (Continued)

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., licensed Senior Operators, licensed Operators, health physics personnel, equipment operators, and key maintenance personnel.

The amount of overtime worked by Unit staff members performing safety-related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).

6.2.3 ~~INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)~~ ONSITE NUCLEAR SAFETY GROUP (ONSG)

FUNCTION

6.2.3.1 The ~~ISEG~~ ONSG shall function to examine plant operating characteristics, NRC issuances, industry advisories, REPORTABLE EVENTS and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety. The ~~ISEG~~ ONSG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Manager of Nuclear Safety, and the Superintendent, Byron Station.

COMPOSITION

6.2.3.2 The ~~ISEG~~ ONSG shall be composed of at least four, dedicated, full-time engineers located on site.

RESPONSIBILITIES

6.2.3.3 The ~~ISEG~~ ONSG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ~~ISEG~~ ONSG shall be prepared, maintained, and forwarded ~~each calendar month~~ to the Manager of Nuclear Safety, and the Superintendent, Byron Station.

6.2.4 SHIFT TECHNICAL ADVISOR

The Station Control Room Engineer (SCRE) may serve as the Shift Technical Advisor (STA) during abnormal operating or accident conditions. During these conditions the SCRE or other on duty STA shall provide technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the unit.

*Not responsible for sign-off function.

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6.5 REVIEW INVESTIGATION AND AUDIT (Continued)

OFFSITE

6.5.1 The Supervisor Manager of Nuclear Safety of the Offsite Review and Investigative Function shall be appointed by the ~~Executive Vice President~~ responsible for nuclear activities. The audit function shall be the responsibility of the Manager of Quality Assurance and shall be independent of operations.

a. Offsite Review and Investigative Function

The Supervisor of the Offsite Review and Investigative Function shall: (1) provide directions for the review and investigative function and appoint a senior participant to provide appropriate direction, (2) select each participant for this function, (3) select a complement of more than one participant who collectively possess background and qualifications in the subject matter under review to provide comprehensive interdisciplinary review coverage under this function, (4) independently review and approve the findings and recommendations developed by personnel performing the review and investigative function, (5) approve and report in a timely manner all findings of non-compliance with NRC requirements to the Station Superintendent, Division Vice President and General Manager - Nuclear Stations, Manager of Quality Assurance, and the Vice President - Nuclear Operations. During periods when the Supervisor of Offsite Review and Investigative Function is unavailable, he shall designate this responsibility to an established alternate, who satisfies the formal training and experience for the Supervisor of the Offsite Review and Investigate ive Function. The responsibilities of the personnel performing this function are stated below. The Offsite Review and Investigative Function shall review:

1) The safety evaluations for: (1) changes to procedures, equipment, or systems as described in the safety analysis report, and (2) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question. Proposed changes to the Quality Assurance Program description shall be reviewed and approved by the Manager of Quality Assurance;

2) Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

3) Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

4) Proposed changes in Technical Specifications or this Operating License;

OFFSITE (Continued)

5) Noncompliance with Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures, or instructions having nuclear safety significance;

6) Significant operating abnormalities or deviation from normal and expected performance of plant equipment that affect nuclear safety as referred to it by the Onsite Review and Investigative Function;

7) All REPORTABLE EVENTS;

8) All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components;

9) Review and report findings and recommendations regarding all changes to the Generating Stations Emergency Plan prior to implementation of such change; and

10) Review and report findings and recommendations regarding all items referred by the Technical Staff Supervisor, Station Superintendent, Division Vice President and General Manager - Nuclear Stations, and Manager of Quality Assurance.

b. Audit Function

The audit function shall be the responsibility of the Manager of Quality Assurance independent of the Production Department. Such responsibility is delegated to the Director of Quality Assurance ~~for~~ Operating and the ~~Staff Assistant to the manager of~~ General Supervisor Quality Assurance ~~for~~ Maintenance ~~quality assurance activities~~.

Either shall approve the audit agenda and checklists, the findings and the report of each audit. Audits shall be performed in accordance with the Company Quality Assurance Program and Procedures. Audits shall be performed to assure that safety-related functions are covered within the period designated below:

1) The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

2) The adherence to procedure, training, and qualification of the station staff at least once per 12 months;

3) The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or methods of operation that affect nuclear safety at least once per 6 months;

4) The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

OFFSITE (Continued)

5) The Facility Emergency Plan and implementing procedures at least once per 12 months;

6) The Facility Security Plan and implementing procedures at least once per 12 months;

7) Onsite and offsite reviews;

8) The Facility Fire Protection programmatic controls including the implementing procedures at least once per 24 months by qualified QA personnel;

9) The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

10) The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months;

11) The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months;

12) The PROCESS CONTROL PROGRAM and implementing procedures for solification of radioactive wastes at least once per 24 months; and

13) The performance of activities required by the Company Quality Assurance Program for effluent and environmental monitoring at least once per 12 months.

Report all findings of noncompliance with NRC requirements and recommendations and results of each audit to the Station Superintendent, ~~Director~~ Manager of Nuclear Safety, the Division Vice President and General Manager - Nuclear Stations, Manager of Quality Assurance, the Vice Chairman, and the Vice President - Nuclear Operations.

c. Authority

The Manager of Quality Assurance and the Manager of ~~reports to the Vice Chairman and the Supervisor of the Offsite Review and Investigative Function reports to the Director~~, Nuclear Safety report to the Chairman and President. Either the Manager of Quality Assurance or the ~~Supervisor of the Offsite Review and Investigation Function~~ Manager of Nuclear Safety has the authority to order unit shutdown or request any other action which he deems necessary to avoid unsafe plant conditions.

OFFSITE (Continued)

d. Records

1) Reviews, audits, and recommendations shall be documented and distributed as covered in Specification 6.5.1a. and 6.5.1b.; and

2) Copies of documentation, reports, and correspondence shall be kept on file at the station.

e. Procedures

Written administrative procedures shall be prepared and maintained for the offsite reviews and investigative functions described in Specification 6.5.1a. and for the audit functions described in Specification 6.5.1b. Those procedures shall cover the following:

1) Content and method of submission of presentations to the Supervisor of the Offsite Review and Investigative Function,

2) Use of committees and consultants,

3) Review and approval,

4) Detailed listing of items to be reviewed,

5) Method of: (1) appointing personnel, (2) performing reviews, investigations, (3) reporting findings and recommendations of reviews and investigations, (4) approving reports, and (5) distributing reports, and

6) Determining satisfactory completion of action required based on approved findings and recommendations reported by personnel performing the review and investigative function.

f. Personnel

1) The persons, including consultants, performing the review and investigative function, in addition to the Supervisor of the Offsite Review and Investigative Function shall have expertise in one or more of the following disciplines as appropriate for the subject or subjects being reviewed and investigated:

a) nuclear power plant technology,
b) reactor operations,
c) utility operations,
d) power plant design,
e) reactor engineering,
f) radiological safety,
g) reactor safety analysis,

OFFSITE (Continued)

h) Instrumentation and Control

Engineering graduate or equivalent with at least 5 years of experience in instrumentation and control design and/or operation.

i) Metallurgy

Engineering graduate or equivalent with at least 5 years of experience in the metallurgical field.

3) The Supervisor of the Offsite Review and Investigative Function shall have experience and training which satisfy ANSI N18.1-1971 requirements for plant managers.

ONSITE

6.5.2 The Onsite Review and Investigative Function shall be supervised by the Station Superintendent.

a. Onsite Review and Investigative Function

The Station Superintendent shall: (1) provide directions for the Review and Investigative Function and appoint the Technical Staff Supervisor, or other comparably qualified individual as the senior participant to provide appropriate directions; (2) approve participants for this function; (3) assure that at least two participants who collectively possess background and qualifications in the subject matter under review are selected to provide comprehensive interdisciplinary review coverage under this function; (4) independently review and approve the findings and recommendations developed by personnel performing the Review and Investigative Function; (5) report all findings of noncompliance with NRC requirements, and provide recommendations to the Division Vice President and General Manager - Nuclear Stations and the Supervisor of the Offsite Review and Investigative Function; and (6) submit to the Offsite Review and Investigative Function for concurrence in a timely manner, those items described in Specification 6.5.1a which have been approved by the Onsite Review and Investigative Function.

b. Responsibility

The responsibilities of the personnel performing this function are:

1) Review of: (1) procedures required by Specification 6.8.1 and changes thereto, (2) all programs required by Specification 6.8.4 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the ~~Plant~~ Station Superintendent to affect nuclear safety;

2) Review of all proposed tests and experiments that affect nuclear safety as defined in 10CFR50.59;

ONSITE (Continued)

3) Review of all proposed changes to the Technical Specifications;

4) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety as defined in 10CFR50.59;

5) Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Division Vice President and General Manager - Nuclear Stations and to the Supervisor of the Offsite ~~Nuclear~~ Review and Investigative Function;

6) Review of all REPORTABLE EVENTS;

7) Performance of special reviews and investigations and reports thereon as requested by the Supervisor of the Offsite Review and Investigative Function;

8) Review of the Station Security Plan and implementing procedures and submittal of recommended changes to the Division Vice President and General Manager - Nuclear Stations;

9) Review of the Emergency Plan and station implementing procedures and shall submit recommended changes to the Division Vice President and General Manager - Nuclear Stations;

10) Review of Unit operations to detect potential hazards to nuclear safety;

11) Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Division Vice President and General Manager - Nuclear Stations and the Supervisor of the Offsite ~~Nuclear~~ Review and Investigative Function; and

12) Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL, and the Radwaste Treatment Systems.

c. Authority

The Technical Staff Supervisor is responsible to the Station Superintendent and shall make recommendations in a timely manner in all areas of review, investigation, and quality control phases of plant maintenance, operation, and administrative procedures relating to facility operations and shall have the authority to request the action necessary to ensure compliance with rules, regulations, and procedures when in his opinion such action is necessary. The Station Superintendent shall follow such recommendations or select a course

HIGH RADIATION AREA (Continued)

source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Rad/Chem Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area; or

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the Radiation Work Permit.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

(more substantial than a rope)

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS
Indicated Reactor Coolant System T_{avg}	≤ ~~592°F (592.5)~~ 591.2
Indicated Pressurizer Pressure	≥ ~~2205 psig* (2220 psig)~~ 2219 psig*

Changes Discussed with Fred Anderson

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*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.

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	2	1	2	3, 4, 5	5
7. Overtemperature ΔT	4	2	3	1, 2	6#
8. Overpower ΔT	4	2	3	1, 2	6#
9. Pressurizer Pressure-Low (Above P-7)	4	2	3	1	6#

ok @ 9/15/84 mtg.

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~~AUG 1984~~ Sep 19 1984

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TABLE 3.3-1 (Continued)

ok per 9/11/84 mts

TABLE NOTATIONS

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

#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.

** The Boron Dilution Protection System may be blocked when rods are being withdrawn.

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.

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint; and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

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

a

ACTION 5 - 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 within the next hour open the reactor trip breakers, suspend all operations involving positive reactivity changes, and verify valves 1CV111B, 1CV8428, 1CV-8439, 1CV-8441 and 1C-8435 are closed and secured in position. *

See Insert "AA"

b.

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.

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Insert "AA" TO PAGE 3/4 3-6

D Two less than the Minimum Channels OPERABLE requirement, verify the Reactor trip breakers are open, suspend all operations involving positive reactivity changes, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2 as applicable within 1 hour, and verify valves 1CV-111B, 1CV-8428, 1CV-8439, 1CV-8441 and 1CV-8435 are closed and secured in position within 4 hours.

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Byron 84-1106

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 Pump	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 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	19*
d. Undervoltage - RCP Bus-Start Motor-Driven Pump and Diesel-Driven Pump	4-1/bus	2	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	2	2	1, 2, ~~3~~, 4	~~18~~ 25*

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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 Feed-water Pump Suction Pressure-Low (Transfer to Essential Service Water)	2	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	14
b. RWST Level - Low-Low Coincident With	4	2	3	1, 2, 3, 4	16
Safety Injection	See Item 1. above for Safety Injection initiating functions and requirements.				
8. Loss of Power					
a. ESF Bus Undervoltage	2/Bus	2/Bus	~~1~~ 2/Bus	1, 2, 3, 4	~~19*~~ 25*
b. Grid Degraded Voltage	2/Bus	2/Bus	~~1~~ 2/Bus	1, 2, 3, 4	~~19*~~ 25*

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NEW

Action 25

With the number of OPERABLE channels one less than the Minimum Numbers 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 inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (SE)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generators, Containment Cooling Fans, Control Room Isolation, Phase "A" Isolation, Turbine Trip, Auxiliary Feedwater, Containment Vent Isolation and Essential Service Water)					
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-1	5.7 ~~6.0~~	0.71	1.5	≤ 3.4 ~~2.7~~ psig	≤ 5.8 psig
d. Pressurizer Pressure-Low (Above P-11)	16.1	14.41	1.5	≥ 1829 psig	≥ 1823 psig
e. Steam Line Pressure-Low (Above P-11)	21.2	14.81	1.5	≥ 640 psig*	≥ 617 psig*
2. Containment Spray					
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-3	8.0	0.71	1.5	≤ 20.0 psig	≤ 21.0 psig

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REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least ~~three~~ two of the reactor coolant loops listed below shall be OPERABLE and at least two of these reactor coolant loops shall be in operation* when the Reactor Trip System breakers are closed and at least one reactor coolant loop shall be in operation when the Reactor Trip System breakers are open:*

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 only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position~~, restore at least two loops to operation within 72 hours or~~ open the reactor trip breakers within 1 hour.

c. 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 loops 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 two~~ The required 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.

** See Special Test Exception 3.10.4

Changes discussed with J. WERMIEL 9/19/84

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REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Particulate ~~and Gaseous~~ Radioactivity Monitoring System,

b. The Containment Floor Drain and Reactor Cavity Flow Monitoring System, and

c. The ~~containment air pressure instrumentation and reactor containment fan cooler outlets and inlets Dewcell and dry bulb temperature instrumentation~~ Containment Gaseous Radioactivity Monitoring System.

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

ACTION:

a. With ~~only two~~ a. or c. of the above required Leakage Detection Systems ~~OPERABLE~~ inoperable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed for gaseous and particulate radioacitivity at least once per 24 hours when the required Gaseous or Particulate Radioactivity Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Insert "B"

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION, and DIGITAL CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment Floor Drain and Reactor Cavity Flow Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months, and

c. Verify the oil separator portion of the containment floor drain collection sump has been filled to the level of the overflow to the containment floor drain unidentified leakage collection weir box once per 18 months, following refueling, and prior to initial startup.

~~d. Containment air pressure and reactor containment fan cooler outlet and inlet temperatures-performance of CHANNEL CALIBRATION at least once per 18 months.~~

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b. With b. of the above required Leakage Detection Systems inoperable be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a. and c. of the above required Leakage Detection Systems inoperable:

1) Restore either a. or c. of the above required Leakage Detection Systems to OPERABLE status within 72 hours and

2) Obtain and analyze a grab sample of the containment atmosphere for gaseous and particulate radioactivity at least once per 24 hours and

3) Perform a Reactor Coolant System water inventory balance at least once per 8 hours

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -0.1 and ~~±0.5~~ +1.0 psig.

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

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

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Containment Systems

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CONTAINMENT PURGE VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valves shall be OPERABLE and:

a. Each 48-inch containment shutdown purge supply and exhaust isolation valve shall be closed and power removed, and

b. The 8-inch containment purge supply and exhaust isolation valve(s) may be open, ~~for up to 1000 hours during a calendar year~~ provided no more than ~~one~~ two lines ~~is~~ are open at one time, for purging and/or venting as required for safety related purposes such as:

1)
See Insert "A"
2)

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

ACTION:

a. With a 48-inch containment purge supply and/or exhaust isolation valve open and/or powered, close and remove power to isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the 8-inch containment purge supply and/or exhaust isolation valve(s) open for reasons other than those stated in Specification 3.6.1.7b ~~more than 1000 hours during a calendar year~~, close the open 8-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.7.3 and/or 4.6.1.7.4, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

INSERT "A" TO PAGE 3/4 6-11 9/19/84

1) Maintaining containment pressure within the limits of Specification 3.6.1.4

2) Reducing the containment atmosphere airborne radioactive or gaseous material to an acceptable level for personnel safety.

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SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 48-inch containment purge supply and exhaust isolation valve(s) shall be verified closed and power removed at least once per 31 days.

4.6.1.7.2 ~~The cumulative time that all 8-inch containment purge supply and/or exhaust isolation valves have been open during a calendar year shall be determined at least once per 7 days.~~

Documentation shall be reviewed every 18 months to confirm the purging and venting was performed in accordance with Specification 3.6.1.7

4.6.1.7.3 At least once per 6 months on a STAGGERED TEST BASIS, the inboard and outboard valves with resilient material seals in each closed 48-inch containment purge supply and exhaust penetration shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.05 L_a when pressurized to at least P_a, 43.6 psig.

4.6.1.7.4 At least once per 3 months, each 8-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.01 L_a when pressurized to at least P_a, 43.6 psig.

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3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The personnel hatch should have a minimum of one door closed at any one time and the equipment hatch shall be in place and held by a minimum of four bolts, or the equipment hatch removed pursuant to Surveillance Requirement 4.9.4.2.

b. A minimum of one door in the personnel emergency exit hatch is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1) Closed by an isolation valve, blind flange, or manual valve, or

2) Capable of being closed by an OPERABLE automatic containment purge isolation valve.

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

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge isolation valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment purge isolation valves per the applicable portions of Specification 4.6.3.2.

4.9.4.2 See Insert

INSERT

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4.9.4.2 Verify that the Fuel Handling Building Exhaust Filter Plenums maintain the Fuel Building at a negative pressure of greater than or equal to 1/4 inch water gauge relative to the outside atmosphere with the equipment hatch removed,

a. Prior to CORE ALTERATIONS or movement of irradiated fuel and

b. At least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel

Verification of negative pressure will be performed with systems in the normal REFUELING MODE.

SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that the exhaust filter plenum satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1.0%, when using the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the flow rate is 21,000 cfm ± 10%;

2) Verifying, within 31 days afer removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, and by showing a methyl iodine penetration of less than 4.3%;

3) Verifying a flow rate of 21,000 cfm ± 10% through the exhaust filter plenum during operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 4.3%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the exhaust filter plenum at a flow rate of 21,000 cfm ± 10%.

2) Verifying that on a Safety Injection test signal and a High Radiation test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks; and

3) Verifying that the exhaust filter plenum maintains the fuel building at a negative pressure of greater than or equal to 1/4 inch Water Gauge relative to the outside atmosphere during operations involving movement of fuel within the storage pool, or crane operation with loads over the storage pool.

Changes discussed with W. Jensen and F. Anderson prior to mtg. 9/19/84

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect flow degradation which could lead to operation outside the acceptable limit.

add back in

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2 hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such ACTION does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a design DNBR throughout each analyzed transient. The calculated values of the DNB-related parameters will be an average of the indicated values for the OPERABLE channels.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

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 the applicable design bases DNBR 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 6 hours.

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~~. See Insert "C"

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.

The restrictions on starting a reactor coolant pump with one or more RCS cold legs less than or equal to 350°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.

The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the stop valves provides a reassurance of the adequacy of the boron concentration in the isolated loop.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops.

Insert "C" to pg B 3/4 4-1

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even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

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~~AUG 28 1984~~

Discussed with Col Moon

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3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met. A contained borated water level between ~~34%~~ 31 and ~~66%~~ 63 ensures a volume of greater than or equal to 6995 gallons but less than or equal to 7217 gallons.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

The requirement to verify accumulator isolation valves shut with power removed from the valve operator when the pressurizer is solid ensures the accumulators will not inject water and cause a pressure transient when the Reactor Coolant System is on solid plant pressure control. ×

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

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3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 0.1 psig, and (2) the containment peak pressure does not exceed the design pressure of 50 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a cold leg double-ended break event is 43.6 psig. The limit of ~~0.3~~ 1.0 psig for initial positive containment pressure will limit the total pressure to ~~43.9~~ 44.3 psig, which is ~~less than design pressure and is consistent with the safety analyses.~~ higher than the FSAR Chapter 15 accident analysis expected peak pressure but it is considerably less than the design pressure of 50 psig

3/4.9 REFUELING OPERATIONS 3/19/84 FINAL DRAFT

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The limitation on Keff of no greater than 0.95 is sufficient to prevent reactor criticality during refueling operations and includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portions of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

The Byron Station is designed such that the containment opens into the fuel building through the personnel hatch or equipment hatch. In the event of a fuel drop accident in the containment, any gaseous radioactivity escaping from the containment building will be filtered through the Fuel Handling Building Exhaust Ventilation System.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.