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July 16, 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

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

(b): March 26, 1984 letter from T. R. Tramm to H. R. Denton.

(c): A· ˙l 2, 1984 letter from T. R. Tramm t . R. Denton.

(d): April 9, 1984 letter from T. R. Tramm to H. R. Denton.

(e): May 2, 1984 letter from L. O. DelGeorge to H. R. Denton.

(f): June 20, 1984 letter from T. R. Tramm to H. R. Denton.

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 N 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.

A number of similar changes were submitted in references (b) through (f). We understand that the NRC will review each of these proposed changes and inform Commonwealth Edison of their acceptability.

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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

lm

cc: Byron Resident Inspector

8934N

ATTACHMENT A
(Bases Section 2.0)

1) Section 2.1.2 (pg B2-2) Reactor Coolant System Pressure

Change "3107 psig" to "3110 psig" in the last paragraph.

This section stated that the entire RCS is hydrotested at 3107 psig which is 125% of the system's design pressure. 125% of design pressure is 1.25 times 2500 psia = 3125 psia -15 psig = 3110 psig.

2) Section 2.2.1 (pg B2-6) Reactor Trip System Instrumentation Setpoints

The second to last line reads ". . . 10% of full equivalent." This should be changed to read ". . . 10% of full power equivalent."

This change is for clarity.

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REACTOR CORE (Continued)

imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistant with core Safety Limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping, valves, and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistant with the design criteria and associated Code requirements.

The entire RCS is hydrotested at ~~3107~~ 3110 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

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Overpower ΔT

The Overpower ΔT Reactor trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

In each of the pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full equivalent); and on increasing power, automatically reinstated by P-7.

POWER

ATTACHMENT B
(Section 3/4.0)

1) Section 4.0, Surveillance Requirements (pg 3/4 0-2)

Add the following as new item 4.0.6.

"4.0.6 Performing a surveillance that results in equipment being inoperable due to requirements of the surveillance (e.g. abnormal valve alignments) is not considered a violation of the Limiting Condition for Operation and does not need to be reported as a violation provided the equipment is made operable at the completion of the surveillance."

This is being added to clarify when Byron Station will be in violation of an LCO.

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APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, but

b. The combined time interval for any three consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i);

4.0.6 Performing a surveillance that results in equipment being inoperable due to requirements of the surveillance (e.g. abnormal valve alignments) is not considered a violation of the Limiting Condition for Operation and does not need to be reported as a violation provided the equipment is made operable at the completion of the surveillance.

ATTACHMENT C
(Section 3/4.1)

Circled items noted in this attachment have been previously submitted.

Section 3/4.1.1.5 (pg 3/4.1-6x) Boron Dilution

1) Section 3/4.1.2.1 Boration Systems (pg. 3/4 1-7)

   Delete "350°F" and insert "or equal to 330°F" in the note that has been previously submitted at the bottom of the page.

   The reason for this change is to permit the plant to do heatups and cooldowns without violating Tech Specs. This change will make it permissible to restore Charging and Safety Injection pumps to operability between 330°F and 350°F. This revision is possible because a review of the Byron cold-overpressure transient analysis reveals that above 330°F the Cold Overpressure Mitigation System should prevent Appendix G limits from being exceeded even for a worst case spurious SI actuation. The relieving capacity of a single pressurizer PORV at 1050 psig, the Cold Overpressure Mitigation System valve-opening pressure setpoint corresponding to 330°F, is adequate for dealing with the Charging and Safety Injection pump flows resulting from the SI actuation.

   Also, it should be noted that, with primary temperatures of 330°F or greater, the Appendix G overpressure limits are calculated to be above the pressurizer safety valve setting of 2485 psig. Hence, in addition to the PORV's, the Safety Valves should also prevent a spurious SI actuation transient at temperatures above 330°F from exceeding the limits.

2) Section 3/4.1.2.3 Charging Pump-Shutdown (pg. 3/4 1-9)

   - Insert, "whenever the temperature of one or more of the RCS cold legs is less than or equal to 330°F" before the first sentence in Section 4.1.2.3.2.

   - Delete the asterisk following the word "one" in Section 3.1.2.3. Add the asterisk behind Mode 4 under "applicability".

   - Delete "350°F" and insert "or equal to 330°F" in the note that was previously submitted at the bottom of the page.

   This is for the same reason as #2 above.

3) Section 3.1.3.3 (pg 3/4 1-18)

   In the LCO delete the words "shutdown or". This change is necessary because the digital rod position indication system does not indicate the actual position of the shutdown mode between 18 steps and 210 steps withdrawn. Therefore, it is not possible to verify that the DRPI agrees with the demand position indicator within 12 steps.

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3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 4* 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

* A maximum of one centrifugal charging pump shall be operable whenever the temperature of one or more of the RCS cold legs is less than ~~350°F~~ or equal to 330°F.

REACTIVITY CONTROL SYSTEMS

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CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

delete the asterisk

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

APPLICABILITY: MODES 4*, 5, and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, on recirculation flow, that a differential pressure across the pump of greater than or equal to 2416 psid is developed when tested pursuant to Specification 4.0.5.

4.1.2.3.2 ~~*~~ ~~All~~ a All charging pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable at least once per 31 days, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position. Whenever the temperature of one or more of the RCS cold legs is less than or equal to 330°F.

* A maximum of one centrifugal charging pump shall be operable whenever the temperature of one or more of the RCS cold legs is less than ~~350°F.~~ or equal to 330°F.

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POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 One digital rod position indicator (excluding bank demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each ~~shutdown or~~ control rod not fully inserted.

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

ACTION:

With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined OPERABLE by verifying ~~that the digital rod position indicator~~ agrees with the demand position indicator within 12 steps when exercised over the full-range of rod travel at least once per 18 months. The Digital Position Indication System does not indicate the actual position of the shutdown rods between 18 steps and 210 steps withdrawn.

*With the Reactor Trip System breakers in the closed position.

#See Special Test Exception 3.10.5.

ATTACHMENT D
(Section 3/4.3)

Circled items noted in this attachment have been previously submitted.

1) Table 3.3-2, item 7, (pg 3/4 3-7)

Item 7, Overtemperature ΔT response time should be changed from "≤ 6 seconds" to "≤ 4 seconds". The FSAR analysis assumes a total overtemperature ΔT response time of 6 seconds. This 6 seconds represents a pure 2 second delay plus a 4 second log allowance. This change should be made in order to maintain a consistency with the FSAR.

2) Table 3.3-3, item 1d, (Pg 3/3 3-14)

Note number "18" should be changed to "19". This maintains a consistency with Tech Spec Section 4.3.1.2, Table 3.3-1 item 9 action statement number on pg 3/4 3-3.

3) Table 3.3-11, (pg 3/4 3-57)/Inserts A & B/ (pg 3/4 3-58)

On Table 3.3-11, pg 3/4 3-57 change the following:

	Instrument Location	Change From	Change To
1.	Zone 76, Elev. 426	9	13
	Zone 7, Elev. 414	6	7
	Zone 24, Elev. 414	10	16
2.	Zone 69, Elev. 451	8	10
	Zone 75, Elev. 451	10	16
3.	Zone 77, Elev. 426	9	20
	Zone 78, Elev. 426	9	19
5.	Zone 13, Elev. 383	5	6

- Insert A to pg 3/4 3-57

Zone 45, Elev. 463	8	9
Zone 46, Elev. 463	8	9
Zone 47, Elev. 463	4	5
Zone 48, Elev. 463	4	5

ATTACHMENT D (CONTINUED)
(Section 3/4.3)

- Insert B to pg 3/4 3-57

Zone 49, Elev. 439	8	23
Zone 50, Elev. 439	8	23
Zone 51, Elev. 439	10	13
Zone 52, Elev. 439	10	13
Zone 53, Elev. 439	8	9
Zone 54, Elev. 439	8	9
Zone 55, Elev. 439	4	6
Zone 56, Elev. 439	4	6

- Table 3.3-11, pg 3/4 3-58

6. Zone 67, Elev. 451	3	11
9. Zone 52, RSH	6	8
10. Zone 39, Elev. 401	3	7

Also, the ** Table Notation, add "These switches are not 72D supervised." These changes are being made because new detectors are being installed in safety related areas to upgrade coverage.

4) Table 4.3-9, item 4, (pg 3/4 3-73)

Delete the words "a nominal" and replace with "hydrogen and nitrogen". Also, delete items 4a and 4b.

As part of the channel calibration for this equipment, Byron Station makes use of the two point method of calibration utilizing a minimum of two standard gas samples each varying in the total percentage of hydrogen and nitrogen. It is not necessary to calibrate this equipment using only the one and four volume percent hydrogen, balance nitrogen gas samples.

5) Table 4.3-9, item 5, (pg 3/4 3-73)

Delete the words "a nominal" and replace with "hydrogen and nitrogen". Also, delete items 5a and 5b.

As part of the channel calibration for this equipment, Byron Station makes use of the two point method of calibration utilizing a minimum of two standard gas samples each varying in the total percentage of oxygen and nitrogen. It is not necessary to calibrate this equipment using <u>only</u> the one and four volume percent oxygen, balance nitrogen gas samples.

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	≤ ~~6~~ 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.

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	15*
d. Pressurizer Pressure-Low (Above P-9)	4	2	3	1, 2, 3#	~~18*~~ 19*
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 3.3-11

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	INSTRUMENT TYPE*	TOTAL NUMBER OF INSTRUMENTS Heat	Flame	Smoke
1. Containment ***				
Zone 11 Elev 426	Suppression	1 **		
Zone 12 Elev 426	Suppression	1 **		
Zone 2 Elev 401	Detection			2
Zone 3 Elev 401	Detection			2
Zone 4 Elev 401	Detection			2
Zone 5 Elev 401	Detection			2
Zone 6 Elev 426	Detection			6
Zone 76 Elev 426	Detection			~~9~~ 13
Zone 7 Elev 414	Detection			~~6~~ 7
Zone 24 Elev 414	Detection			~~10~~ 16
2. Control Room				
Zone 68 Elev 451	Detection			3
Zone 69 Elev 451	Detection			~~8~~ 10
Zone 75 Elev 451	Detection			~~10~~ 16
3. Switchgear Rooms				
Zone 77 Elev 426	Detection			~~9~~ 20
Zone 78 Elev 426	Detection			~~9~~ 19
4. Upper Cable Spreading Room				
~~Zone 33 Elev 463~~	~~Suppression~~	Replace with A		4
~~Zone 34 Elev 463~~	~~Suppression~~			8
~~Zone 35 Elev 463~~	~~Suppression~~			8
~~Zone 36 Elev 463~~	~~Suppression~~			4
Lower Cable Spreading Room				
~~Zone 43 Elev 439~~	~~Suppression~~	Replace with B		8
~~Zone 44 Elev 439~~	~~Suppression~~			10
~~Zone 45 Elev 439~~	~~Suppression~~			8
~~Zone 46 Elev 439~~	~~Suppression~~			4
5. Remote Shutdown Panel				
Zone 13 Elev 383	Detection			~~8~~ 6

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				Smoke
Zone	41	Elev 463	Detection	4
Zone	42	Elev 463	Detection	4
Zone	43	Elev 463	Detection	8
Zone	44	Elev 463	Detection	8
Zone	45	Elev 463	Detection	~~8~~ 9
Zone	46	Elev 463	Detection	~~8~~ 9
Zone	47	Elev 463	Detection	~~4~~ 5
Zone	48	Elev 463	Detection	~~4~~ 5

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				Smoke
Zone	49	Elev 434	Detection	~~8~~ 23
Zone	50	Elev 434	Detection	~~8~~ 23
Zone	51	Elev 434	Detection	~~4~~ 13
Zone	52	Elev 434	Detection	~~4~~ 13
Zone	53	Elev 434	Detection	~~8~~ 9
Zone	54	Elev 434	Detection	~~8~~ 9
Zone	55	Elev 434	Detection	~~4~~ 6
Zone	56	Elev 434	Detection	~~4~~ 6

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	INSTRUMENT TYPE*	TOTAL NUMBER OF INSTRUMENTS Heat	Flame	Smoke
6. Station Battery Room				
Zone 67 Elev 451	Detection			~~3~~ 11
7. Diesel Generator Room				
Zone 37 Elev 401	Suppression	4		
Zone 38 Elev 401	Suppression	4		
Zone 71 Elev 401	Detection		1	
Zone 71 Elev 401	Detection		1	
8. Diesel Fuel Storage				
Zone 39 Elev 401	Suppression	1		
Zone 40 Elev 401	Suppression	~~1~~		
Zone 27 Elev 383	Suppression	3		~~3~~
Zone 28 Elev 383	Suppression	3		~~3~~
Zone 16 Elev 383	Detection			6
9. Safety Related Pumps				
Zone 41 Elev 383	Suppression	2		
Zone 42 Elev 383	Suppression	1		
Zone 16 Elev 364	Detection			2
Zone 18 Elev 364	Detection			~~1~~
Zone 19 Elev 364	Detection			~~3~~ 2
Zone 20 Elev 346	Detection			~~1~~
Zone 21 Elev 346	Detection			3
Zone 52 RSH	Suppression	~~4~~ 8		
10. Fuel Storage				
Zone 39 Elev 401	Detection			~~3~~ 7
Zone 38 Elev 426	Detection		3	

TABLE NOTATIONS

*A single detector in a zone marked "Detection" will alarm in the Main Control Room. A single detector in a zone marked "Suppression" will initiate suppression and alarm in the Main Control Room.

**These are Containment Ventilation temperature switches. Upon receipt of a Hi-Hi temperature, suppression must be manually initiated. These switches are not supervised. TLD

***The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A containment leakage rate tests.

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TABLE 4.3-9 (Continued)

TABLE NOTATIONS

* At all times.

** During WASTE GAS HOLDUP SYSTEM operation.

(1) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

   a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

   b. Circuit failure, or

   c. Instrument indicates a downscale failure, or

   d. Instrument controls not set in operate mode.

(2) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

   a. Instrument indicates measured levels above the Alarm Setpoint, or

   b. Circuit failure, or

   c. Instrument indicates a downscale failure, or

   d. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing ~~a nominal:~~ hydrogen and nitrogen.

   ~~a. One volume percent hydrogen, balance nitrogen, and~~

   ~~b. Four volume percent hydrogen, balance nitrogen.~~

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing ~~a nominal:~~ oxygen and nitrogen.

   ~~a. One volume percent oxygen, balance nitrogen, and~~

   ~~b. Four volume percent oxygen, balance nitrogen.~~

ATTACHMENT E
(Section 3/4.4)

Circled items noted in this attachment have been previously submitted.

1) Section 4.4.6.2.1(b), (pg 3/4 4-20)

   Delete the words "inventory and" from item b. At Byron Station there is no capability of monitoring the reactor cavity sump <u>inventory</u>. The method of calculating the containment floor drain and reactor cavity sump discharge is attached (See BOS 0.1-1 Revision 1, Page D-11).

2) Section 4.4.9.1.2, (Pg 3/4 4-30)

   Delete the word "and" between Figures 3.4.2...3.4.3. Also, following the 3.4.3 add "and 3.4.4". By adding reference to Figure 3.4-4, this ensures that the figure will be updated periodically to account for the irradiation of the pressure vessel.

3) Section 3/4.4.11 (pg 3/4 4-39)

   Add the attached <u>new</u> submittal Limiting Conditions for Operation for the Reactor Coolant System Vents.

   This is being added as a result of NUREG-0737.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor at least once per 12 hours;

b. Monitoring the containment floor drain and reactor cavity sump ~~inventory and~~ discharge at least once per 12 hours;

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± ~~[illegible]~~ ±20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

CONTAINMENT FLOOR DRAIN AND REACTOR CAVITY SUMP INVENTORY AND DISCHARGE

NOTE: Run time meters are located inside 0PL02J

1). Reactor Cavity Sump Pump 1RF01P

Shift	Previous Reading	Present Reading	Run time	Flow	Discharge
1	Min	Min	Min	71.1gpm	gals
2	Min	Min	Min	71.1gpm	gals
3	Min	Min	Min	71.1gpm	gals

2). Containment Floor Drain Sump Pump 1RF02PA

Shift	Previous Reading	Present Reading	Run time	Flow	Discharge
1	Min	Min	Min	142.5gpm	gals
2	Min	Min	Min	142.5gpm	gals
3	Min	Min	Min	142.5gpm	gals

3.) Containment Floor Drain Sump Pump 1RF02PB.

Shift	Previous Reading	Present Reading	Run time	Flow	Discharge
1	Min	Min	Min	98.5gpm	gals
2	Min	Min	Min	98.5gpm	gals
3	Min	Min	Min	98.5gpm	gals

Previous Reading (Min) - Present Reading (Min) = Run time (Min)

Run time (Min) X flow (gpm) = Discharge (gals)

FOR INFORMATION ONLY

APPROVED
JUL 18 1983
B.O.S.R.

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3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H, in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2, ~~and~~ 3.4-3 and 3.4-4.

REACTOR COOLANT SYSTEM

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least TWO REACTOR VESSEL HEAD vent PATHS, EACH consisting of two valves in series powered from emergency buses, shall be OPERABLE and closed.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the above REACTOR VESSEL HEAD vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With BOTH of the above REACTOR VESSEL HEAD vent paths inoperable; maintain the inoperable vent paths closed with power removed from the valve actuators of all the valves in the inoperable vent paths, and restore at least ONE of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11 Each REACTOR VESSEL HEAD vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.
2. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.
3. Verifying flow through the REACTOR VESSEL HEAD vent paths during venting OPERATIONS AT COLD SHUTDOWN or REFUELING.

ATTACHMENT F
(Section 3/4.5)

Circled items noted in this attachment have been previously submitted.

1) Section 3/4.5.3 ECCS Subsystems (pg. 3/4 5-7)

   Delete "350°F" and insert "or equal to 330°F" from the note at the bottom of the page.

   Section 4.5.3.2 (pg. 3/4 5-8)

   Delete "350°F" and insert "or equal to 330°F".

   This is consistent with the change which allows the Charging and Safety Injection pumps to be made operable between 330°F and 350°F.

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LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,*

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

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

APPLICABILITY: MODE 4.

ACTION:

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

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

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission within 90 days, pursuant to Specification 6.7.2, describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

* A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than ~~or equal to 350°F.~~ or equal to 330°F.

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SURVEILLANCE REQUIREMENTS

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

4.5.3.2 All charging pumps and Safety Injection pumps, except the above required OPERABLE pumps, shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than ~~or equal to 350°F.~~ or equal to 330°F.

---

* It may be desirable to operate the SI pumps for testing and also for filling the accumulator tanks while in Mode 4. It is permissable to operate the SI pumps if the appropriate valves are locked closed to prevent pressurization of the RCS. To run the A-Pump, valves SI 8802A and SI 8821A must be locked closed. The B-Pump may be used by locking closed valves SI 8835, SI 8802A and SI 8802B with valves SI 8821A and SI 8821B open. When the accumulators are being filled, the applicable block valves (SI 8808A, SI 8808B, SI 8808C and SI 8808D) must also be locked closed.

ATTACHMENT G
(Section 3/4.6)

1) Section 3.6.1.7.b (pg 3/4 6-11) and Action Statement b.

Delete the number "1000" and replace with "2000". Technical Specification 3.8.1.4 (pg 3/4 6-6) requires an extremely small primary containment internal pressure range. The 1000 hour limit for the isolation valve will be exceeded given normal operation of the unit.

2) Section 4.6.4.1 (pg 3/4 6-23)

Delete the words "on a Staggered Test Basis". Following the words "sample gas containing" add the words "hydrogen and nitrogen". Also, delete items 4.6.4.1a and 4.6.4.1.b.

As part of the channel calibration for this equipment, Byron Station makes use of the two point method of calibration utilizing a minimum of two standard gas samples each varying in the total percentage of hydrogen and nitrogen. It is not necessary to calibrate this equipment using <u>only</u> the one and four volume percent hydrogen, balance nitrogen gas samples.

The staggered test basis definition does not apply here.

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CONTAINMENT SYSTEMS

CONTAINMENT 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 ~~sealed closed~~ power removed, and

b. The 8-inch containment purge supply and exhaust isolation valve(s) may be open for up to ~~1000~~ 2000 hours during a calendar year. ~~provided no more than one line is open at one time.~~

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

ACTION:

a. With a 48-inch containment purge supply and/or exhaust isolation valve open or not sealed closed, close and/or seal close that valve 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.

b. With the 8-inch containment purge supply and/or exhaust isolation valve(s) open for more than ~~1000~~ 2000 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.

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CONTAINMENT SYSTEMS

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:

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.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK 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: hydrogen and nitrogen.

~~a. One volume percent hydrogen, balance nitrogen, and~~

~~b. Four volume percent hydrogen, balance nitrogen.~~

ATTACHMENT H
(Section 3/4.7)

1) Section 3/4.7.3 (page "AA" (2 of 3) after page 3/4 7-11)
Component Cooling Water System

Delete paragraph 4.7.3.1.b from the attached Surveillance Requirements of the Tech Specs.

There are no valves in the Component Cooling Water System that are automatically actuated on an SI signal.

2) Section 3/4.7.1 Fire Suppression Systems (page 3/4 7-28)

- Insert the word "Supply" in 5 places on this page between the words "Fire Suppression Water" and "System".

- Replace LCO 3.7.10.1.b (page 3/4 7-28) with the following:

"An OPERABLE flowpath capable of taking suction from the flume and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to: 1) the yard hydrant isolation valves (for hydrants near buildings containing safety related equipment), 2) the last valve ahead of each hose standpipe as required by Specification 3.7.10.5, 3) the last valve ahead of the deluge valve (on the diesel generator oil storage room foam system and manual containment charcoal filter deluge systems), or 4) flow alarm valves (on sprinkler systems) as required by Specification 3.7.10.2."

This change is made to limit the fire suppression system operability to safety related areas.

3) Section 3.7.10.2 (pg 3/4 7-31)

- Change section heading from "Foam Systems" to "Water Systems"

- LCO 3.7.10.2 (page 3/4 7-31) shall be modified to read as follows:

"The Water Systems including the foam systems in the diesel generator fuel oil storage tank room the sprinkler systems in the Auxiliary Building and the containment charcoal filter manual deluge system shall be OPERABLE."

Delete the word "foam" in the applicability statement and replace with "above". Delete the word "foam" in the Action Statement.

This change is made to include the containment manual deluge system and new sprinkler systems.

4) Renumber Surveillance Requirements 4.7.10.2 to "4.7.10.2.1" this is necessary so two new sections can be added.

ATTACHMENT H (CONTINUED)
(Section 3/4.7)

5) Delete the existing Surveillance Requirement 4.7.10.2.1.c(1) and insert the following:

   1) By performing a system functional test which includes:

      a) Verifying a "Fire Trouble" alarm is received when the isolation valve is closed, and

      b) Verifying a "Fire" alarm is received on actuation of the alarm switch.

   These changes are necessary to provide the surveillance requirements for the two systems added: the containment manual deluge and the new sprinkler system.

   This change is necessary to delete automatic actuation of the foam system per a NRC concern on spurious actuation of the foam system.

6) Add as new surveillance requirements 4.7.10.2.2 and 4.7.10.2.3 the following:

   "4.7.10.2.2 The above required sprinkler system shall be demonstrated OPERABLE;

   a) At least once per 12 months:

      1) By flowing water out the inspectors test connection and verifying a "Fire" alarm

      2) By cycling the isolation valve and verifying the "Fire Trouble" alarm

      3) By a visual inspection of the sprinkler header to verify integrity and that the head spray pattern is not obstructed.

      4) By verifying water flow through the 2" test drain at the riser.

4. .10.2.3 The above required manual deluge system shall be demonstrated OPERABLE.

a) At least once per 18 months

1) By cycling the isolation valve and verifying the "Fire Trouble" alarm

2) By cycling the deluge valve and verifying a "Fire" alarm

3) By a visual inspection of the deluge header to verify integrity and that the spray heads are not obstructed.

This change notes the distinction between the surveillances for the various water systems.

7) Section 3.7.10.3 CO_2 Systems (pg 3/4 7-32)

- Delete the words "low pressure" in Sections 3.7.10.3, 4.7.10.3.1 and 4.7.10.3.2. This change is for clarity. There is only one CO_2 system.

- To action statement a. add the words "(lower cable spreading room)" between the words "damaged;" and "for other areas".

- Section 4.7.10.3.2 b(1) and the word "both" after actuates, delete the words "manually and" after actuates and add the words "manually, and" at the end of the sentence.

- On insert A, replace 290 with 275.

This change is necessary in order to be consistent with vendor, the manual, Chemetron Operations & Service Manual.

8) Table 3.7-5 Fire Hose Stations (pg. 3/4 7-36)

Insert

"LOCATION	ELEVATION	HOSERACK	ANGLE
M-8 South Wall of battery room	451	279	0FP638
M-26 South Wall of battery room	451	280	0FP639"

after L-11 and before M-18.

These new hose reels have been added.

9) Section 4.7.11.1 Fire Rated Assemblies (pg. 3/4 7-38)

- Following the words a "visual inspection of" add "a) Performing".

- Also, renumber 4.7.11.1 a, b, and c to 4.7.11.1 (1), (2) and (3)

- Also, add as a new item "4.7.11.1.b

Performing a functional test of at least 10% of the fire dampers. If any nonconforming dampers are found, an additional 10% will be inspected. This process will continue until an acceptable sample is found."

This addition is being made due to NRC concerns.

AA (2 of 3)
follows pg 3/4 7-11

6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two safety loops shall be demonstrated OPERABLE:

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

b. ~~At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety-related equipment (including containment isolation valves) actuates to its correct position on a SI test signal.~~

4.7.3.2 At least two component cooling water pumps shall be demonstrated OPERABLE by performing the following:

a. The component cooling pumps shall be ~~demonstrated OPERABLE~~ operated each month. Performance will be acceptable if the pump starts upon actuation, operates for at least 4 hours and satisfies the cooling

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3/4.7.10 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SUPPLY SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.10.1 The Fire Suppression Water SUPPLY System shall be OPERABLE with:

a. Two fire suppression pumps, ~~each with a capacity of 2500 gpm~~, with their discharge aligned to the fire suppression header, and

b. ~~An OPERABLE flow path capable of taking suction from the flume and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each Deluge or Spray System required to be OPERABLE per Specifications 3.7.10.2 and 3.7.10.5.~~ Insert "XX"

Foam

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the Fire Suppression Water SUPPLY System otherwise inoperable establish a backup Fire Suppression Water SUPPLY System within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.10.1.1 The Fire Suppression Water SUPPLY System shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the contained water supply volume,

b. At least once per 31 days on a STAGGERED TEST BASIS by starting the electric motor-driven pump and operating it for at least 15 minutes on recirculation flow,

c. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position,

2,7

Insert "xx"

An operable flowpath capable of taking suction from the flume and transfering the water through distribution piping with Operable sectionalizing control or isolation valves to: 1) the yard hydrant isolation valves (for hydrants near buildings containing safety related equipment), 2) the last valve ahead of each hose standpipe as required by Specification 3.7.10.5, 3) the last valve ahead of the deluge valve (on the diesel generator oil storage room foam system and manual containment charcoal filter deluge systems), or 4) flow alarm valves (on sprinkler systems) as required by Specification 3.7.10.2.

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PLANT SYSTEMS

WATER ~~FOAM~~ SYSTEMS

LIMITING CONDITION FOR OPERATION

WATER SYSTEMS INCLUDING THE

3.7.10.2 The Foam Systems in the diesel generator fuel OIL storage tank ~~rooms~~ room, the sprinkler ~~shall be OPERABLE.~~ systems in the Auxiliary Building and the containment charcoal filter manual deluge system shall be operable.

APPLICABILITY: Whenever equipment protected by the ~~Foam~~ above System is required to be OPERABLE.

ACTION:

a. With one or more of the above required ~~Foam~~ Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.10.2.1 ~~Each of~~ The above required Foam ~~Systems~~ System shall be demonstrated OPERABLE:

a. At least once per 31 days, by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position,

b. At least once per 12 months, by cycling each testable valve in the flow path through at least one complete cycle of full travel,

c. At least once per 18 months:

1) By performing a system functional test which includes: ~~simulated automatic actuation of the system, and:~~

a) Verifying ~~that the automatic valves in the flow path actuate to their correct positions on a Fire Detection test signal, and~~ a "Fire Trouble" alarm is received when the isolation valve is closed, and

Verifying a "Fire" alarm is received on actuation of the alarm switch.

b) ~~Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.~~

2) By a visual inspection of the dry pipe ~~spray and sprinkler~~ deluge headers to verify their integrity, and

3) By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

Insert "A"

d. At least once per 3 years by performing an air flow test through each open deluge nozzle and verifying each nozzle is unobstructed.

Insert "A" on pg 3/4 7-31

4.7.10.2.2 The above required sprinkler system shall be demonstrated Operable;

a) At least once per 12 months;

1) By flowing water out the inspectors test connection and verifying a "Fire" alarm.

2) By cycling the isolation valve and verifying the "Fire Trouble" alarm.

3) By a visual inspection of the sprinkler header to verify integrity and that the head spray pattern is not obstructed.

4) By verifying water flow through the 2" test drain at the riser.

4.7.10.2.3 The above required manual deluge system shall be demonstrated Operable;

a) At least once per 18 months;

1) By cycling the isolation valve and verifying the "Fire Trouble" alarm

2) By cycling the deluge valve and verifying a "Fire" alarm

3) By a visual inspection of the deluge header to verify integrity and that the spray heads are not obstructed.

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PLANT SYSTEMS

CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.10.3 The following ~~Low Pressure~~ CO_2 Systems shall be OPERABLE:

a. Diesel generator rooms and day tank rooms,

b. Lower cable spreading room,

c. Auxiliary feedwater diesel room and day tank room, and

d. Diesel-driven Essential Service Water (ESW) ~~make-up~~ pumps and day tank rooms.

APPLICABILITY: Whenever equipment protected by the CO_2 systems is required to be OPERABLE:

ACTION:

a. With one or more of the above required CO_2 systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged (lower cable spreading room); for other areas, establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.10.3.1 Each of the above required ~~Low Pressure~~ CO_2 Systems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

4.7.10.3.2 Each of the above required ~~Low Pressure~~ CO_2 Systems shall be demonstrated OPERABLE:

Replace with (A)

a. At least once per 7 days by verifying the CO_2 storage tank level to be greater than 75% (7.5 tons) and pressure to be greater than 290 psig, and

b. At least once per 18 months by verifying:

1) The system, including associated ventilation system fire dampers, actuates, ~~manually and~~ automatically upon receipt of a simulated actuation signal, and manually, and

2) Both Flow from each nozzle during a "Puff Test."

Insert "A" to page 3/4 7-32

(A) At least once per 7 days by verifying the plant CO_2 storage tank level to be greater than 75% (7.5 tons), the river screen house CO_2 storage tank level to be greater than 50% (1 ton); and pressure of both to be greater than ~~290~~ 275 and less than 375 psig, and

SUBMITTED WITH THIS PKG.

BYRON - UNIT 1

TABLE 3.7-5 (Continued)

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK REEL	ANGLE VALVE
Aux. Bldg. (Continued)			
P-7 ~~P-17:~~ ~~By control panel 1AP76J~~ West wall 6.9 KV switchgear room.	455	20	0FP324
L-11: ~~By cooling coils + fan~~ In VC HVAC RM OA of LCSR C-1	455	22	0FP332
SEE INSERT "A"			
M-18: North wall ~~U-1 by door~~	444	238 *	0FP463
East Wall ~~L-7:~~ ~~Southeast corner of lower cable room~~ LCSR A-1	443	207 *	0FP327
In the ~~M-10:~~ ~~By east door to room 121~~ Southeast corner of LCSR A-1	443	208 *	0FP327
P-10: ~~By west door to room 121~~ Southwest " " " "	443	209 *	0FP325
M-13: South wall of ~~room 123~~ LCSR C-1	443	210 *	0FP326
P-15 ~~P-13:~~ West wall of ~~room 124~~ LCSR D-1	443	211 *	0FP328
S-21: By cabinet 2RY01EC (elec. pen. area)	431	229	0FP455
S-24: By U-2 cont. shield wall (elec. pen. area)	431	230	0FP456
S-12: By U-1 cont. shield wall (elec. pen. area)	431	237	0FP462
P-11: ~~By contaminated clothing room~~ outside laundry room	430	52	0FP313
Q-19: By U-2 ~~boron injection pumps~~ VCT valve aisle	430	54	0FP342
P-24: By radwaste evaporator	430	55	0FP343
V-17: By ~~U-1 boron injection pumps~~ east door to decon/change area	430	58	0FP319
V-17: By ~~U-1~~ door to decon/change area	430	61	0FP320
L-11: By ~~U-1~~ waste oil tank room	405	90	0FP315
P-18: By elevator	405	91	0FP318
P-23: By ~~U-2~~ spent resin pumps	405	92	0FP349
Q-11: By laundry tanks	405	93	0FP314
East of ~~S-21:~~ ~~By~~ U-2 ~~pipe tunnel trolley beam~~ hydrogen recombiner	405	94	0FP348
west of ~~V-21:~~ ~~By~~ U-2 hydrogen recombiner	405	95	0FP345
west of ~~V-15:~~ ~~By~~ U-1 hydrogen recombiner ~~control panel~~	405	96	0FP316
East of S-15: ~~By~~ U-1 ~~pipe tunnel trolley beam~~ hydrogen recombiner	405	97	0FP317
N-11: By the recycle holdup tanks	368	130	0FP373
M-14: By the stairs of U-1	368	131	0FP374
P-14: By panel 1PL84JB	368	132	0FP369
L-20: By ~~the~~ stairs of U-2	368	133	0FP355
P-21: By the blowdown condenser PW M/U Pumps	368	134	0FP356
L-25: By the ~~recycle evaporator monitor tanks~~	368	135	0FP361
N-25: By ~~regeneration waste~~ chemical drain tank	368	136	0FP357
S-18: By panel 1PL86J	368	138	0FP362
Q-11: By Aux. Bldg. floor drain tanks	368	139	0FP368
U-15: By ~~positive displacement charging pump~~ U-1, Spray Add Tank	368	140	0FP372

3/4 7-36

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"A"

LOCATION	ELEVATION	HOSE RACK	ANGLE
M-8 South wall of battery room	451	279	0FP638
M-26 South wall of battery room	451	280	0FP639

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3/4.7.11 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.11 All fire rated assemblies (walls, floor/ceilings, cable tray enclosures and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire-rated assembly penetrations (fire doors, fire windows, fire dampers, cable, piping and ventilation duct penetration seals) shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.11.1 At least once per 18 months the above required fire barrier penetrations and penetration sealing devices shall be verified OPERABLE by: ~~performing a~~

a) Performing a visual inspection of:

1 ~~a~~. The exposed surfaces of each fire rated assembly,

2 ~~b~~. Each fire window/fire damper and associated hardware,

3 ~~c~~. At least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected every 15 years~~.~~, and

b. Performing a functional test of at least 10% of the fire dampers. If any non-conforming dampers are found, an additional 10% will be inspected. This process will continue until an acceptable sample is found.

4.7.11.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the ~~automatic hold-open,~~ release and closing mechanism and latches at least once per 6 months, and by verifying:

X

a. The OPERABILITY of the Fire Door Supervision System for each electrically supervised fire door by performing a TRIP ACTUATING DEVICE OPERATIONAL TEST at least once per 31 days,

b. That each locked closed fire door is closed at least once per 7 days, and

c. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

ATTACHMENT I
(Section 3/4.8)

1) Section 3.8.1.1 Action Statement C.2. (pg 3/4 8-2)

Insert between the words "pump" and "and" the words "is operable". Also, delete the words "Division 21 diesel generator is operable" and replace with "2A diesel generator is capable of providing power to Bus 141." (See Below)

2) Section 3.8.1.3 (pg 3/4 8-X)

The attached section represents a new tech spec submittal for the 2A diesel generator. This LCO and Surveillance represents the Byron Station action statement and surveillance requirements for the 2A diesel generator. The above change to pg 3/4 8-2 maintains a consistency with this addition.

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2A diesel generator is capable of providing power to Bus 141.

ACTION (Continued)

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

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

2. When in MODE 1, 2, or 3, the diesel-driven auxiliary feedwater pump is operable and the ~~Division 21 diesel generator is OPERABLE~~, if the inoperable diesel generator is the emergency power supply for the motor-driven auxiliary feedwater pump.

If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Specification 4.8.1.1.2a.4) within 1 hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring manually unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

3.4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A. C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.3 The 2A Diesel Generator shall be capable of being manually started and crosstied to Bus 141.*

APPLICABILITY: Modes 1, 2, and 3

ACTION: With the 2A Diesel Generator incapable of being manually started and crosstied to Bus 141, restore the diesel generator to capable status within 7 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.4 The 2A diesel generator shall be demonstrated capable of providing power to ~~1A auxiliary feedwater pump.~~ Bus 141

a. At least once per day by:

1. Verifying the day tank level is greater than 450 gallons.

2. Verifying DC control power is available to the 2A Diesel Generator local control panel (2PL07J).

3. Verifying that at least one starting air receiver is at greater than 175 psig.

4. Verifying the Essential Service Water System is ~~properly aligned~~ available to supply cooling requirements.

b. At least once per 31 days by:

1. Verifying the diesel generator starts manually and operates with a load of greater than or equal to 5500 KW for one half hour.

c. At least once per 18 months by:

1. Verifying the diesel generator can be crosstied to Bus 141.

* The reporting requirements of 10CFR50.36(c)(2) do not apply during the first eight hours in the Action Statement

(0464M)

ATTACHMENT J
(Section Bases 3/4)

1) Section 3/4.1.2 (pg. B 3/4 1-2) Boration Systems

Change "200°F" to "350°F" in the Second paragraph.

(pg B 3/4 1-3)

Change "200°F" to "350°F" in the first paragraph.

This section specifies that at least two boron injection flow paths shall be operable in modes 1, 2, and 3. However, the bases for this section explained the requirement as being applicable above 200°F.

Change "350°F" to "330°F" in the second paragraph.

2) Section 3/4.4.1 (pg. B 3/4 4-1) Reactor Coolant Loops and Coolant Circulation

Change "1 hour" to "6 hours" in the first paragraph.

The action statement indicated that with less than the above required reactor coolant loops in operation be in at least HOT STANDBY within 6 hrs. This was not consistent with the bases provided for this Technical Specification which required placing the plant in HOT STANDBY within 1 hour with less than the required reactor coolant loops in operation.

3) Section 3/4.4.2 Safety Valves (pg B 3/4 4-2)

In the second paragraph following the words "loss-of-load assuming no reactor trip" delete the words "until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor Trip on the loss-of-load)".

The sizing of the Pressurizer safety valves, as stated in the BYRON overpressure Protection Report, is based upon a loss of load event assuming no reactor trip or power-operated relief valves.

The Loss of Load Analysis in FSAR Section 15.2 is the limiting transient with respect to overpressure. This analysis verifies that the safety valve capacity (already sized) is sufficient to maintain reactor coolant system pressure below 110% of design pressure (2750 psia). In this analysis, no credit is taken for a reactor trip from the second reactor trip setpoint signal which is either overtemperature ΔT or high pressurizer pressure.

ATTACHMENT J (CONTINUED)
(Section Bases 3/4)

4) Section 3/4.4.9 Pressure/Temperature Limits (pg B 3/4 4-15)

- Change "HPSI" to "centrifugal charging". This is for clarification.

The following should be added at the end of the last paragraph on pg B 3/4 4-14.

"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 overshoots will not exceed the nominal 10EFPY Appendix G reactor vessel NDT limits."

This is being added in response to an NRC question concerning the PORV Setpoint curve.

5) Section 3/4.5.3 (pg B 3/4 5-2) ECCS Subsystems

Change "350°F" to "330°F" in the first paragraph.

6) Section 3/4.6.1.5 Air Temperature (pg B 3/4 6-2)

Delete the word "listed" between the words "made at all" and "locations" and replace with "of the listed running fan". Also, delete the word "prior" and change "determining" to "determine". This change is made for consistency to the LCO Surveillance Requirements.

7) Section 3/4.8 (pg B 3/4 8-1)

Insert the following as the last paragraph: "The reporting requirement of 10CFR50.36 do not apply during the first eight hours in the Action Statement."

This requirement is necessary to complete the Pre-Operational Test Program on Unit 2. The Unit 2 Auxiliary Power Test will require breaker interlocks to be checked and this will affect the Unit 2 Diesel Generator output breaker. Diesel Generator retesting may also be required. This exception to the normal reporting method will only remain in affect until fuel load on Unit 2.

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MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -4.1×10^{-4} Δk/k/°F. The MTC value of -3.2×10^{-4} Δk/k/°F represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -4.1×10^{-4} Δk/k/°F.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 550°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, (4) the reactor vessel is above its minimum RT_{NDT} temperature, and (5) the plant is above the cooldown steam dump permissive, P-12.

~~Insert "SS"~~

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above ~~200°F~~ 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% Δk/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 15,780 gallons of 7000-ppm borated water from the boric acid storage tanks or 70,450 gallons of 2000-ppm borated water from the refueling water storage tank.

BORATION SYSTEMS (Continued)

With the RCS temperature below ~~200°F~~ 350°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below ~~350°F~~ 330°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2,652 gallons of 7000-ppm borated water from the boric acid storage tanks or 11,840 gallons of 2000-ppm borated water from the refueling water storage tank (RWST).

Insert attached "BB"

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. The Digital Position Indication System does not indicate the actual position of the shutdown rods between 18 steps and 210 steps withdrawn.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

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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~~ ABOVE THE APPLICABLE SAFETY ANALYSIS 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 ~~1~~ 6 hours.

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

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 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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3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip ~~until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load)~~ and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

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BASES

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 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 380°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.

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ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and Safety Injection pumps except the required OPERABLE charging pump to be inoperable below ~~350°F~~ 330°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The Surveillance Requirements for leakage testing of ECCS check valves ensures that a failure of one valve will not cause an intersystem LOCA.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

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3/4.6.1.5 AIR TEMPERATURE

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The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, ~~prior~~ to determin~~ing~~e the average air temperature.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 43.9 psig in the event of a cold leg double-ended break accident. The measurement of containment tendon lift-off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

The Surveillance Requirements for demonstrating the containment's structural integrity are ~~in compliance~~ with the recommendations of proposed Rev 3 to Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," ~~January 1976~~ April 1979 and proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979.

The required Special Reports from any engineering evaluation of containment abnormalties shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerances on cracking, the results of the engineering evaluation and the corrective actions taken.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 48-inch containment purge supply and exhaust isolation valves are required to be sealed closed during plant operations since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during plant operation ensures that excessive quantities of radioactive material will not be released via the Containment Purge System. To provide assurance that the 48-inch containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

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BASES

3/4.8.1, 3/4.8.2 AND 3/4.8.3 A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for: (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE, and that the diesel-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods, and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979.

Insert attachment "JJ"

Attachment "JJ"

The reporting requirements of 10CFR50.36 do not apply during the first eight hours in the Action Statement. This requirement is necessary to complete the Pre-Operational test program on the Unit 2 plant. The Unit 2 Auxilliary Power test will require breaker interlocks to be checked and this will affect the Unit 2 Diesel Generator output breaker. Diesel Generator retesting may also be required. This exception to the normal reporting method will only remain in affect until fuel load on Unit 2.

ATTACHMENT K
(Section 6.0)

1) Table 6.2-1 (pg 6-4)

- Replace the present table with

Position	Number of Individuals Required to Fill Position	
	Modes 1, 2, 3, & 4	Modes 5 & 6
SE	1	1
SF	1	None
RO	2	1
AO	2	1
SCRE	1	None

or, whenever a SCRE (SRO/STA) is not included in the shift crew composition, the minimum shift crew shall be as follows:

Position	Number of Individuals Required to Fill Position	
	Modes 1, 2, 3, & 4	Modes 5 & 6
SE	1	1
SF	1	None
RO	2	1
AO	2	1
STA	1	None

SE - Shift Supervisor (Shift Engineer) with a Senior Operator License on Unit 1

SF - Shift Foreman with a Senior Operator license on Unit 1

RO - Individual with a Reactor Operator license on Unit 1

AO - Auxiliary Operator

SCRE - Station Control Room Engineer with a Senior Reactor Operators License on Unit 1.

STA - Shift Technical Advisor.

ATTACHMENT K
(Section 6.0)

- Delete "(other than the Shift Technical Advisor)" which appears twice in the second paragraph.

  This change is consistent with the LaSalle Technical Specification minimum shift crew composition.

2) Section 6.2.4 (pg. 6-5)

   Delete "ONSG" as last word of paragraph and replace with "Office of Nuclear Safety" as per Operating Experience Feedback BAP 1260-1.

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TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

Replace with insert "A"

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SS	1	1
SRO	1	None
RO	2	1
AO	4	1
STA	1	None

SSE - Shift Supervisor (Shift Engineer) with a Senior Operator license on Unit 1
SRO - Individual with a Senior Operator license on Unit 1
RO - Individual with an Operator license on Unit 1
AO - Auxiliary Operator
STA - Shift Technical Advisor

The Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the Unit is in MODE 1, 2, 3 or 4, an individual ~~(other than the Shift Technical Advisor)~~ with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the Unit is in MODE 5 or 6, an individual with a valid Operator license ~~(other than the Shift Technical Advisor)~~ shall be designated to assume the control room command function.

Insert "A"

Position	Number of Individuals Required to Fill Positions	
	Modes 1, 2, 3 & 4	Modes 5 & 6
SE	1	1
SF	1	None
RO	2	1
AO	2	1
SCRE	1	None

or, whenever a SCRE (SRO/STA) is not included in the shift crew composition, the minimum shift crew shall be as follows:

Position	Number of Individuals Required to Fill Positions	
	Modes 1, 2, 3 & 4	Modes 5 & 6
SE	1	1
SF	1	None
RO	2	1
AO	2	1
STA	1	None

Notes:

SE - Shift Supervisor (Shift Engineer) with a Senior Reactor Operators License on Unit 1

SF - Shift Foreman with a Senior Reactor Operators License on unit 1

RO - Individual with a Reactor Operators License on Unit 1

AO - Auxilliary Operator

SCRE - Station Control Room Engineer with a Senior Reactor Operators License on Unit 1

STA - Shift Technical Advisor

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ORGANIZATION (Continued)

e. The Onsite Nuclear Safety Group (ONSG) shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving unit safety. The ONSG shall be composed of at least four, dedicated, full-time engineers of multi-disciplines located on site and shall be augmented on a part-time basis by personnel from other parts of the Commonwealth Edison Company organization to provide expertise not represented in the group. The ONSG shall be responsible for maintaining surveillance of Unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical. The ONSG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving unit safety to the Director, Nuclear Safety.

f. The Station Control Room Engineer** (SCRE) serves as the lead control room licensed Senior Operator during normal operations and as the Shift Technical Advisor (STA) during abnormal operating and accident conditions. In the event of abnormal operating or accident conditions, the SCRE will relinquish his job as control room licensed Senior Operator to a Shift Supervisor and will assume the role of STA. Then he shall provide technical support to the shift in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the unit.

6.2.3 Qualifications of the station management and operating staff shall meet minimum acceptable levels as described in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971. The Rad/Chem Supervisor or Lead Health Physicist shall meet the requirements of Radation Protection Manager of Regulatory Guide 1.8, September, 1975. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

6.2.4 The content and quality of retraining and replacement training programs for the unit staff shall be maintained under the direction of the Production Training Department. The Station Training Department, under the functional control of the Production Training Department, shall have responsibilities for training activities at the station. These training programs shall meet or exceed the requirements and recommendations of Section 5 of ANSI/ANS 3.1-1978 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience identified by the ~~ONSG.~~

Office of Nuclear Safety

*Not responsible for sign-off feature.

**The position of SCRE can be filled by a STA provided a licensed Senior Operator is in the control room.

OPERATING EXPERIENCE FEEDBACK

A. STATEMENT OF APPLICABILITY

The purpose of this procedure is to outline the method to provide for effective and efficient feedback of operating experience to plant personnel. This procedure has the objective to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other cognizant personnel, and is incorporated into training and re-training programs, and evaluated for possible modifications or plant procedures.

B. REFERENCES

1. NUREG-0737, item I.C.5.
2. CECo. Operating Experience Review Program

C. MAIN BODY

1. Operating experience assessment and dissemination is the responsibility of the Administrative and Support Services Assistant Superintendent. If applicable, the Training Supervisor is responsible for incorporation of this information, once obtained, into appropriate training and re-training programs. The Operating Assistant Superintendent is responsible for ensuring dissemination of information to Operating Department personnel.

2. Operating experience information is received at the Station from many sources; among which are:

   a. Operating Experience Assessment Report Transmittals (OPEX).

   b. Deviation Reports.

   c. Off-Site Reviews.

   d. NRC Bulletins, Circulars, Information Notices, and correspondence.

   e. LER Monthly Reports.

   f. W Technical bulletins and vendor advisories.

3. Information pertaining to the above items is handled as follows:

   a. OPEX are received at the Station from the Office of Nuclear Safety. These include NRC Power Reactor Events, ~~INPO/NSAC Significant~~ Operating Experience Reports, NSAC Annual Screening and Evaluation of License Event Reports, Note pad items and DVR's from CECo stations. Upon receipt, the Administrative and Support Services Assistant Superintendent reviews the report and routes it to the Licensing Coordinator who logs the report, coordinates writing any required response, and obtains station review approval. The original copy is routed to the Training Supervisor for dissemination to the SCRE and License personnel indicated on the routing cover sheet. The Training Supervisor is responsible for maintaining a record of the routing completions.

APPROVED
OCT 07 1983
B. O. S. R.

ATTACHMENT L
(T_{avg} Setpoint Change)

1) Table 2.2-1 (pg 2-9)

   In NOTE 1, change "587.7" to "588.4" for the T^{1} setpoint.

2) Table 2.2-1 (pg 2-11)

   In NOTE 3, change "587.7" to "588.4" for the T^{11} setpoint.

3) Section 5.4.2 (pg 5-4)

   Change "587.7" to "588.4"

   The above changes to T_{avg} are being made to reflect the PLS setpoints.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s,

T' ≤ ~~587.7~~ 588.4°F (Nominal T_{avg} at RATED THERMAL POWER),

K_3 = 0.00134,

P = Pressurizer pressure, psig,

P' = 2235 psig (Nominal RCS operating pressure),

S = Laplace transform operator, s^{-1},

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) for $q_t - q_b$ between -42% and -6%, $f_1(\Delta I)$ = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -42%, the ΔT Trip Setpoint shall be automatically reduced by 2.86% of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds -6%, the ΔT Trip Setpoint shall be automatically reduced by 1.86% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~2.5%~~ 3.3 ΔT ~~instrument~~ span.

X

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.00170/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$,

T = As defined in Note 1,

T'' = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ ~~587.7~~ 588.4°F),

S = As defined in Note 3, and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~(2)~~% ΔT ~~instrument~~ span.
2.6

X

NOTE 5: THE SENSOR ERROR FOR TEMPERATURE IS 1.2 and FOR PRESSURE IS 1.0.

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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 ~~1748~~ 1594 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 weight percent U-235. x

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. x x

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 ~~587.4~~ 588.4°F.

5.5 METEOROLOGICAL TOWER LOCATION

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

ATTACHMENT M
(Radiation Monitoring Instrumentation)

1) Table 3.3-6 (pg 3/4 3-39)

Change the alarm/trip setpoint for the Fuel Building - Fuel Handling unit from "$\leq$2 mR/h" to "$\leq$2.5mR/h".

2) Table 3.3-6 (pg 3/4 3-39)

Change the alarm/trip setpoint for the Containment - Fuel Handling unit from $\leq$2mR/h" to "$\leq$100mR/h".

3) Table 3.3-6 (pg 3/4 3-39)

Change the alarm/trip setpoint for the Main Control Room - Outside Air Intake from "$\leq$2mR/h" to "particulate 1.0 E-9μCi/cc
gas 9.5 E-7μCi/cc
iodine 1.12 E-9μCi/cc".

The above setpoints have been calculated according to ANSI/ANS-8.3 - 1979, Appendix B.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Fuel Building-Fuel Handling (ORE-AR055/56)	1	2	**	2.5 ≤ ~~2~~ mR/h	30
2. Containment-Fuel Handling (1RE-AR011/12)	1	2	#	100 ≤ ~~2~~ mR/h	26
3. Criticality-Radiation Level (ORE-AR037/38)	1	2	*	≤ 15 mR/h	28
4. Gaseous Radioactivity-RCS Leakage Detection (1RE-PR011A)	N.A.	1	1, 2, 3, 4	N.A.	29
5. Particulate Radioactivity-RCS Leakage Detection (1RE-PR011B)	N.A.	1	1, 2, 3, 4	N.A.	29
6. Main Control Room-Outside Air Intake (ORE-PR031/32 and ORE-PR033/34)	1	2 per intake	All	≤ ~~2 mR/h~~ particulate 1.0E-9 μCi/cc gas 9.5E-7 μCi/cc iodine 1.12E-9 μCi/cc	27

ATTACHMENT N
(Setpoint Changes)

1) Table 2.2-1 (pg 2-6)

The Z value for Underfrequency - Reactor Coolant Pumps should be "13.3" not "0".

This change is to be conistent with the Setpoint Methodology.

2) Table 3.3-3 (pg 3/4 3-15)

Item 3c should read

"Containment Vent Isolation

1)	High Radiation	See Table 3.3-6 Item 2 for all High Radiation initiating functions and requirements.					
2)	Automatic Actuation Logic and Actuation Relays	2	1	2	1,2,3 4	17	
3)	Manual Phase "A" Isolation	See Item 3.a.1 for all manual Phase "A" Isolation initiating functions and requirements.					
4)	Manual Phase "B" Isolation	See Item 3.b.1 for all manual Phase "B" Isolation initiating functions and requirements.					
5)	Safety Injection	See Item 1 above for all Safety Injection initiating function and requirements."					

The above change is to reflect the signals that cause a Containment Vent Isolation at Byron.

3) Table 3.3-4 (pg 3/4 3-22)

Change item 1c Total allowance from 2.5 to 3.0
Change item 1d Total allowance from 13.0 to 16.1.
Change item 1d Z from 10.71 to 14.41.
Change item 1e Total allowance from 14.2 to 21.2.
Change item 1e Z from 10.71 to 14.81.
Change item 2c Total allowance from 5.0 to 8.0.

The above changes are made to be consistent with the Setpoint Methodology.

ATTACHMENT N (Continued)
(Setpoint Changes)

4) Table 3.3-4 (pg 3/4 3-23)

Item 3c should read

"Containment Vent Isolation

1)	High Radiation	See Table 3.3-6 Item 2 for the High Radiation trip setpoint				
2)	Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	N.A.
3)	Manual Phase A Isolation	See Item 3.a.1 above for all manual Phase "A" Isolation Trip Setpoints and Allowable Values.				
4)	Manual Phase B Isolation	See Item 3.b.1 above for all manual Phase "B" Isolation Trip Setpoints and Allowable Values.				
5)	Safety Injection	See Item 1 above for all Safety Injection Trip Setpoints and Allowable Values".				

The above change is to reflect the signals that cause a Containment Isolation.

5) Table 9.3-4 (pg 3-24)

Change item 4.c Total Allowance from 5.0 to 7.7.
Change item 4.d Total Allowance from 14.2 to 21.2.
Change item 4.d Z from 10.71 to 14.81.
Change item 4.e Allowable Value from $\leq$-110.0 psi to $\leq$-111.5 psi.
Change item 5.b Total Allowance from 5.0 to 6.0.
Change item 5.b Z from 2.18 to 4.28.

These changes are necessary to concur with the updated version of the Westinghouse Setpoint Study.

ATTACHMENT N (Continued)
(Setpoint Changes)

6) Table 9.3-4 (pg 3-25)

Change item 6.c Total Allowance from 30.0 to 27.1.
Change item 6.c Z from 27.18 to 18.28.
Change item 6.c Allowable Value from 36 to 39.1.

These changes are necessary to concur with the updated version of the Westinghouse Setpoint Study.

7) Table 4.3-2 (ph 3-34)

Change item 3.c Containment Vent Isolation to read as follows:

"1)	High Radiation	See Table 4.3-3 Item 2 for all High Radiation Surveillance Requirements.							
2)	Automatic Actuation Logic and Acutation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1,2,3,4
3)	Manual Phase "A" Isolation	See Item 3.a.1 above for all manual Phase "A" Isolation Surveillance Requirements.							
4)	Manual Phase "B" Isolation	See Item 3.b.1 above for all manual Phase "B" Isolation Surveillance Requirements.							
5)	Safety Injection	See Item 1 above for all Safety Injection Surveillance Requirements."							

The above change is to reflect the signals that cause a Containment Vent Isolation at Byron.

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*~~ 39.1%
13. Steam Generator Water Level Low-Low	27.1	~~27.18~~ 18.28	1.5	>40.8% of narrow range instrument span	>36% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	3.3	0	0	4920 ~~>4898~~ volts - each bus	>4768 volts - each bus
15. Underfrequency - Reactor Coolant Pumps	14.4	13.3 ~~13.0~~	0	57.0 ~~>57.5~~ Hz	57.6 ~~>57.6~~ Hz
16. Turbine Trip					
a. Emergency Trip Header ~~Low Fluid Oil~~ Pressure	N.A.	N.A	N.A.	540 >~~X~~ psig	NA >~~X~~ psig
* b. Turbine ~~Stop~~ Throttle Valve Closure	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

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
b. Phase "B" Isolation					
1) Manual Initiation	2 pair	1 pair	2 pair	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure-High-3	4	2	3	1, 2, 3	16
c. Containment Vent Isolation					
1) High Radiation ~~Manual Initiation~~	See Table 3.3-6 Item 2 for all High Radiation initiating functions and requirements ~~2~~	~~1~~	~~2~~	~~1, 2, 3, 4~~	~~17~~
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	17
3) Manual Phase "A" Isolation	See Item 3.a.1 for all manual Phase "A" Isolation initiating functions and requirements.				
4) Manual Phase "B" Isolation	See Item 3.b.1 for all manual Phase "B" Isolation initiating functions and requirements.				
5) Safety Injection	See Item 1. above for all Safety Injection ~~initiating~~ initiating function and requirements.				

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TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
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.	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	3.0 ~~2.5~~	0.71	1.5	≤ 5.0 psig	5.8 ≤ ~~3.6~~ psig
d. Pressurizer Pressure-Low	16.1 ~~13.0~~	14.41 ~~10.71~~	1.5	1829 ≥ ~~1850~~ psig	1823 ≥ ~~1839~~ psig
e. Steam Line Pressure-Low (above P-11)	21.2 ~~14.2~~	14.81 ~~10.71~~	1.5	≥ 640 psig	617 ≥ 610 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 ~~5.0~~	0.71	1.5	≤ 20.0 psig	21 ≤ ~~22.0~~ psig

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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
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
b. Phase "B" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Containment Pressure-High-3	8.0 ~~6.0~~	0.71	1.5	≤ 20.0 psig	≤ 21 ~~22.0~~ psig
c. Containment Vent Isolation					
1) High Radiation ~~Manual Initiation~~	See Table 3.3-6 Item 2 for the ~~all~~ High Radiation trip ~~actuation~~ setpoint ~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) manual Phase "A" Isolation	See Item 3.a.1 above for all manual Phase "A" Isolation Trip Setpoints and Allowable Values.				
4) manual Phase "B" Isolation	See Item 3.b.1 above for all manual Phase "B" Isolation Trip Setpoints and Allowable Values.				
5) Safety Injection	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
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 ~~14.2~~	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 ~~5.0~~	4.28 ~~2.18~~	1.5	81.4 ≤ ~~82%~~ of narrow range instrument span	82.7 ≤ ~~83%~~ of narrow range instrument span

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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
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2, 3, 4
2) Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Containment Vent Isolation	See Table 4.3-3 Item 2 for all High Radiation Surveillance Requirements							
1) High Radiation ~~Manual Initiation~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~R~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3, 4~~
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Manual Phase "A" Isolation	See Item 3.a.1 above for all manual Phase "A" Isolation Surveillance Requirements.							
4) Manual Phase "B" Isolation	See Item 3.b.1 above for all manual Phase "B" Isolation Surveillance Requirements							
5) Safety Injection	See Item 1 above for all Safety Injection Surveillance Requirements							

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 instrument span	39.1 ~~36~~ 39.1 ≥ ~~40%~~ of narrow range instrument span
d. Undervoltage-RCP Bus-Start Motor Driven Pump and Diesel-Driven Pump (6.9KV buses)	N.A.	N.A.	N.A.	≥ 4920 volts ~~70% RCP Bus voltage (4890 volts)~~ 4879	≥ 4768 volts ~~69% RCP Bus voltage (4674 volts)~~ 4623
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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