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April 2, 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.

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 specific changes proposed here is necessary before the Technical Specifications can be finalized.

Attachments A through Z 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 reference (b). We understand that the NRC will review each of these proposed changes and inform Commonwealth Edison of their acceptability A few additional changes will be submitted by April 9, 1984.

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

B001 1/16 Limited Dist

cc: Byron Resident Inspector

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ATTACHMENT A

(Definition 1.0)

Circled items noted in this attachment have been previously submitted.

1) Definition 1.30 (Page 1-5) Site Boundary

The phrase ",as defined by exclusion area," has been added before "shall". Also, reference to Figure 5.1-1; page 5-2 has been added.

"As defined by exclusion area" has been added to define what site boundary they are referring to, and (figure 5.1-1; page 5-2) is the drawing that shows the exclusion area.

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PURGE - PURGING

1.24 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.25 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever ~~is greater~~. With one excore detector inoperable, the remaining three detectors ~~shall be used~~ are used for computing the average.

RATED THERMAL POWER

1.26 RATED THERMAL POWER shall be a total core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.28 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all ~~full length~~ rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.30 The SITE BOUNDARY, as defined by exclusion area, shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee. (See figure 5.1-1, page 5-2).

SLAVE RELAY TEST

1.31 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

ATTACHMENT B

(Section 2.0)

Circled items noted in this attachment have been previously submitted.

1) Table 2.2-1 (pg. 2-5) Reactor Trip System Instrumentation Trip Setpoints

   Number 2, a and b; Values for Allowable Value were changed to reflect current PLS values 111.1% and 27.1% respectively.

2) Table 2.2-1 (pg 2-5) Reactor Trip System Instrumentation Trip Setpoint

   Number 5; Values for Allowable Values has been changed to reflect current PLS value of 30.9%.

3) Table 2.2-1 (pg. 2-5) Reactor Trip System Instrumentation Trip Setpoints

   Number 6 and 7; Values for Z "10.0 and 5.15" and Sensor Error "0" were changed to reflect current PLS values. Also, note 5 was added under Sensor error column.

4) Table 2.2-1 (pg. 2-5) Reactor Trip System Instrumentation Trip Setpoints

   Number 9; Values for Total Allowance "5.0", Z "2.21", and Allowable Value "1871" were changed to reflect current PLS values.

5) Table 2.2-1 (pg. 2-6) Reactor Trip System Instrumentation Trip Setpoints

   Number 15; Value for Z "0", has been changed to concur with Data in Westinghouse Statistical Study for Byron Station.

6) Table 2.2-1 (pg. 2-6) Reactor Trip System Instrumentation Trip Setpoints

   Number 16; Values for Trip Setpoint "540" and Allowable Value "N/A" were changed and 16a heading was changed to Emergency Trip Header Pressure to reflect what the trip comes off. Also, item 16b heading was changed for clarity.

7) Table 2.2-1 (pg. 2-6) Reactor Trip System Instrumentation Trip Setpoints

   Loop Design flow - 95,700 gpm has been changed to 94,400 to reflect value from FSAR Table 5.1-1.

8) Table 2.2-1 (pg. 2-7) Reactor Trip System Instrumentation Trip Setpoints

   Number 19b "or" was inserted to reflect that either can be used as an input for RTS interlocks.

   Number 19d "$\geq$ 7.8%" and 19e "$<$12.2" changes to Allowable Values have been made to reflect corrections made by Westinghouse in reference to Westinghouse Statistical Setpoint Study for Byron.

(Section 2.0)

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

"Note 5: The sensor error for temperature is 1.2 and for pressure is 1.0" has been added for clarity.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤111.~~[illegible]~~ 1% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤27.~~[illegible]~~ 1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	≤25% of RTP*	≤~~31%~~ of RTP* 30.9%
6. Source Range, Neutron Flux	17.0	10.0	~~1.29~~ 0	≤10^5 cps	≤1.4 x 10^5 cps
7. Overtemperature ΔT	8.7	~~4.51~~ 5.15	~~1.1~~ 0 SEE NOTE 5	See Note 1	See Note 2
8. Overpower ΔT	4.3	1.3	1.2	See Note 3	See Note 4
9. Pressurizer Pressure-Low	~~3.1~~ 5.0	2.21 ~~0.71~~	1.5	≥1885 psig	≥~~1874~~ 1871 psig
10. Pressurizer Pressure-High	3.1	0.71	1.5	≤2385 psig	≤2396 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	≤92% of instrument span	≤93.8% of instrument span

*RTP = RATED THERMAL POWER

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	2.5	1.77	0.6	≥90% of loop design flow*	≥89.2% of loop design flow*
13. Steam Generator Water Level Low-Low	27.1	~~27.18~~ 18.28	1.5	≥40.8% of narrow range instrument span	≥~~36%~~ 39.1% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	3.3	0	0	≥~~4090~~ 4920 volts - each bus	≥4768 volts - each bus
15. Underfrequency - Reactor Coolant Pumps	14.4	~~13.3~~ 0	0	≥~~57.5~~ 57.0 Hz	≥~~57.6~~ 52.6 Hz
16. Turbine Trip					
a. Emergency Trip Header ~~Low Fluid Oil~~ Pressure	N.A.	N.A.	N.A.	≥~~[illegible]~~ 540 psig	≥~~[illegible]~~ NA 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 = ~~95,700~~ 94,400 gpm

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
19. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	≤10% of RTP*	≤12.2% of RTP*
OR 2) P-13 input	N.A.	N.A.	N.A.	<10% RTP* Turbine Impulse Pressure Equivalent	<12.2% RTP* Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	≤30% of RTP*	≤50.2% of RTP*
d. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A	≤10% of RTP*	≥ <7.8% of RTP*
e. Turbine Impulse Chamber Pressure, P-13	N.A.	N.A.	N.A.	<10% RTP* Turbine Impulse Pressure Equivalent	<12.2 of ~~<11%~~ RTP* Turbine Impulse Pressure Equivalent
20. Reactor Trip Breakers	N.A.	N.A.	N.A	N.A.	N.A.
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

*RTP = RATED THERMAL POWER

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°F),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

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

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

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ATTACHMENT C

(Section 3/4.1)

Circled items noted in this attachment have been previously submitted.

1) Section 3.1.1.3 (pg 3/4 1-4) Reactivity Control Systems - moderator Temperature Coefficient

   Under Action a. deleted the word "above" from sentence to add simplicity. "In lieu of any other report required by Specification 6.7.1, "has been deleted to be consistent with Tech Specs.

2) Section 3.1.2.1/4.1.2.1 (pg 3/4 1-7) Reactivity Control System - Boration Systems

   In Applicability, Mode 4 has been added because Section 3.1.2.1 is referring to Mode 4.

   A note stating "*A maximum of one centrifugal charging pump shall be operable whenever the temperature of one or more of the RCS Cold leg is less than 350°F" because it is applicable to Mode 4.

3) Section 3.1.2.2 (pg 3/4 1-8) Reactivity Control System - Operating

   The "*" was deleted from 3.1.2.2 because it is applicable to Mode 4.

4) Section 3.1.2.2 (pg 3/4 1-8) Reactivity Control System - Operating

   Mode 4 was deleted because not applicable to this section.

5) Section 3.1.2.3 (page 3/4 1-9) Charging Pump - Shutdown

   Add Note
   "* 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."

   This note is added for clarification as it has been in other LCO's.

6) Section 3.1.2.4 (Page 3/4 1-10) Charging Pumps - Operating.

   Deletion of * and the footnote is requested because they are applicable to Mode 4.

7) Section 3.1.2.5 (Page 3/4 1-11) Borated Water Source - Shutdown.

   The suggested change to 2652 from 2650 is necessary in order to be consistent with bases.

   Also 6.7% has been changed to 7.0% and 8.7% has been changed to 9.0% to make monitoring from panel easier for operators.

ATTACHMENT C

(Section 3/4.1) Continued

8) Section 3.1.2.6 (pg 3/4 1-12) Borated Water Sources Item 5(b) added from Section 3.5.4 for consistency discussion.

Also, delete "as required by Specification 3.1.2.2" because it is not applicable below 350°F (Mode 4) and these tanks must be operable in Modes 1 through 4.

9) Section 4.1.2.6 (pg. 3/4 1-13) Reactivity Control System - Operating

Added "C" which states "At least once per 24 hours by verifying the RWST vent path temperature to be greater than or equal to 35°F when the outside air temperature is less than 35°F." This was added to complete Surveillance Requirements.

10) Section 3.1.3.1 (pg. 3/4 1-14/15) Reactivity Control System - Movable Control Assemblies

Deleted the words "full length" from the page because all rods at Byron are of the same length.

11) Section 3.1.3.1 (pg 3/4 1-14 and 1-15) Reactivity Control Systems

Under Actions, 3a. has been switched with 3d. to add clarity and for its Operating Importance.

12) Section 4.1.3.3 (pg. 3/4 1-18) Reactivity Control System - Shutdown.

Add sentence stating "The Digital Position Indication System does not indicate the actual position of the Shutdown rods, between 18 steps and 210 steps withdrawn" from pg. B 3/4 1-3 to Section 4.1.3.3 for System Capability Clarifications.

13) Section 3.1.3.4 (pg. 3/4 1-19) Reactivity Control System - Rod Drop Time

Deleted the word "full length from section 3.1.3.4 to indicate all rods are of one length at Byron.

14) Section 3.1.3.4 (pg. 3/4 1-19) Reactivity Control System - Rod Drop Time

Deleted step (a) from Action because Rod Drop time must be verified before reaching Modes 1 & 2. Therefore (a) will never apply.

ATTACHMENT C

(Section 3/4.1) Continued

15) Section 3.1.3.b (pg. 3/4 1-21) Reactivity Control Systems

In Section 3.1.3.6 the "s" was deleted from Figures and "and 3.1-2" was also deleted to get rid of all references to three loop Operations In Action a. The word "immediately" was inserted at the beginning of the sentence and "within 2 hours" was deleted. The phrase "Immediately initiate reduction in" was added to the sentence upon the deletion of "Reduce" and "within 2 hours" to give clarity to proper operations.

"b" was turned into "c", "c" is now "d" and a new "b" was created stating "Immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing g.t. or equal to 7000 ppm boron or equivalent boration rate, until the req'd control bank position is restored." This was added to be consistent with Surveillance 4.1.1.1.1.b pg 3/4 1-1.

16) Section 4.1.3.b (pg 3/4 1-21) Reactivity Control System.

Deleted the word "Monitor" and placed "Alarm" in its place.

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MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0 $\Delta k/k/^{\circ}F$ for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition, or

b. Less negative than -4.1×10^{-4} $\Delta k/k/^{\circ}F$ for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only#.
Specification 3.1.1.3b. - MODES 1, 2, and 3 only#.

ACTION:

a. ~~With the~~ MTC more positive than the limit of Specification 3.1.1.3a. ~~above,~~ operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 $\Delta k/k/^{\circ}F$ within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. ~~In lieu of any other report required by Specification 6.7.1, a~~ A Special Report is prepared and submitted to the Commission pursuant to Specification 6.7.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.
#See Special Test Exception 3.10.3.

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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.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

~~*Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 350°F.~~

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CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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 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.

* 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.

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CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two~~*~~ charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.4. At least two charging pumps shall be demonstrated OPERABLE by verifying, on recirculation flow, that a differential pressure across each pump of greater than or equal to 2416 psid is developed when tested pursuant to Specification 4.0.5.

DELETE

~~*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.~~

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BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A Boric Acid Storage System with:

1) A minimum contained borated water level of ~~6.7%~~ 7.0% (~~2650~~ 2652 gallons),

2) A minimum boron concentration of 7000 ppm, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water level of ~~8.7%~~ 9.0% (38,740 gallons),

2) A minimum boron concentration of 2000 ppm, and

3) A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

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

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 35°F.

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BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. A Boric Acid Storage System with:

1) A minimum contained borated water level of 40% (15,780 gallons),

2) A minimum boron concentration of 7000 ppm, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water level of 89% (395,000 gallons),

2) A minimum boron concentration of 2000 ppm,

3) A minimum solution temperature of 35°F, and

4) A maximum solution temperature of 100°F.

5) Heat traced portions of the associated flow paths shall be OPERABLE

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

ACTION:

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

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

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

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

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

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 35°F or greater than 100°F.

c. At least once per 24 hours by verifying the RWST vent path temperature to be greater than or equal to 35°F when the outside air temperature is less than 35°F.

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REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All ~~full-length~~ shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more ~~full-length~~ rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one ~~full-length~~ rod inoperable or misaligned from the group step counter ~~demand position~~ by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one ~~full-length~~ rod trippable but inoperable due to causes other than ~~addressed~~ by ACTION a. above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figures 3.1-1 and 3.1-2. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      d a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

*See Special Test Exceptions 3.10.2 and 3.10.3.

ACTION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

a ~~d)~~ The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each ~~full-length~~ rod shall be determined to be within the group demand limit by ~~verifying the~~ individual rod positions at least once ~~per 12 hours~~ except during time intervals when the rod position deviation ~~monitor~~ is inoperable, then verify the group positions at least once per 4 hours. ALARM

4.1.3.1.2 Each ~~full-length~~ rod not fully inserted in the core shall be determined OPERABLE ~~by movement~~ of at least 10 steps in any one direction at least once per 31 days.

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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.

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ROD DROP TIME

LIMITING CONDITION FOR OPERATION

Delete

3.1.3.4 The individual ~~full-length~~ shutdown and control rod drop time from the fully withdrawn position shall be less than or equal to 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 550°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

~~a. With the rod drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.~~

a. ~~b.~~ With the rod drop time within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of ~~full-length~~ rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

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CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*#.

ACTION:

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

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

Immediately initiate reduction in
c. ~~b.~~ ~~Reduce~~ THERMAL POWER ~~within 2 hours~~ to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the above figure ~~s, or~~ 3.1-1.

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

b. Immediately initiate and continue boration at greater than or equal to 30 gpm of a Solution containing g.t. or equal to 7000 ppm boron or equivalent boration rate, until the req'd control bank position is restored.

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exceptions 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.

ATTACHMENT D

(Section 3/4.2)

1) Section 3.2.3 (pg. 3/4 2-8) Power Distribution Limits

A period was added after R and "since - - - - - - $F^{N}_{\Delta H}$" was deleted because it is stated in Bases.

2) Table 3.2-1 (pg. 3/4 2-15) DNB Parameters

Under Limits for PRessurizer PRessure $\geq$ 2220 psig has been changed to $\geq$ 2205 psig because it is the indication provided to Operator.

Also, all other pressures in the Technical Specifications are in gage.

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3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figure 3.2-3 for four loop operation.

Where:

a. $R = \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.3\ (1.0 - P)]}$,

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R. ~~since Figure 3.2-3 includes penalties for undetected feedwater venturi fouling of 0.1% and for measurement uncertainties of 2.1% for flow and 4% for incore measurement of~~ $\cancel{F^N_{\Delta H}}$.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux-High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS	
	Four Loops in Operation	Three Loops in Operation
Reactor Coolant System T_{avg}	$\leq$ 592°F	**
Pressurizer Pressure	~~$\geq$ 2220 psia*~~ 2205 psig*	**

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

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

ATTACHMENT E

(Section 3/4.3)

Circled items noted in this attachment have been previously submitted.

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

Number 16; Emergency Tripheader EHC has been deleted from heading and added in Step A in place of Low Auto Stop Oil to indicate what turbine trip indications come off. Train has also been added after *S 3, 2, 2 to give clarification. Step b Stop has been changed to Throttle to correct the valve type.

2) Table 3.3-2 (pg. 3/4 3-8) Reactor Trip System Instrumentation Response Times.

Number 16; Step(a) Low Auto Stop Oil has been deleted and Emergency Trip header has taken its place to indicate what the turbine trip comes off. Step(b) Stop has been deleted and Throttle added to indicate the correct valve type.

3) Table 4.3-1 (pg. 3/4 3-10) Reactor Trip System Instrumentation Surveillance Requirements

number 16; Same explanation as above.

4) Table 4.3-1 (pg. 3/4 3-12) Table Notations

Sentence number 12 has been rearranged as shown to add clarification and give a more accurate description.

5) Table 3.3-3 (Page 3/4 3-19) Engineered Safety Features Actuation Instrumentation.

"3/stm. gen." has been changed to "4/stm. gen." because there are 4 channels/gen.

6) Table 3.3-4 (Page 3/4 3-22) Engineered Safety Feature Actuation Instrumentation

Change the Allowable Value for 1.c to "$\geq$5.5 psig".

The changes in table 3.3-4, Pg. 3/4 3-22 were made to concur with the values from the PLS (Precautions, Limitations and Setpoints).

7) Table 3.3-4 (pg. 3/4 3-22) Engineered Safety Features Actuation System Instrumentation Trip Setpoints

Number 1(c), (d), (e) has changed values for Allowable value from 3.5, 1839, 160 (psig) to 5.8, 1823, 617 (psig) respectively. 1(d) has changed values for Trip Setpoint from 1850 to 1829 (psig). Number 2(c) has changed value for Allowable value form 22.0 to 21.0 (psig) to concur with Westinghouse Statistical Setpoint Study.

8) Table 3.3-4 (pg. 3/4 3-23) Engineered Safety Features Actuation System Instrumentation Trip Setpoints.

   Number 3(b); Values for Allowable Value has been changed from 22.0 to 21.0 to concur with Westinghouse Statistical Setpoint Study.

9) Table 3.3-4 (pg. 3/4 3-24) Engineered Safety Features Actuation System Instrumentation Trip Setpoints

   Number 4(c), (d) has changes for Allowable Value 12.0, 610 has been changed to 9.2 and 617 respectively. Number 4(c) has changed for Trip Setpoint 10.0 psig to 8.2 psig. Number 5(b) has changes for Trip Setpoint and Allowable Value to concur with Westinghouse Statistical Setpoint Study.

10) Table 3.3-4 (pg. 3/4 3-25) Engineered Safety Features Actuation System Instrumentation Trip Setpoints

   Number 6 (c), (d) has changes for Trip Setpoint and Allowable Value to concur with Westinghouse Statistical Setpoint Study.

11) Table 3.3-4 (pg 3/4 3-27) Engineered Safety Features Actuation System Instrumentation Trip Setpoints.

   Number 9 (a) has changes for Trip Setpoint 1950 psig has been changed to 1930 psig to concur with PLS.

12) Table 3.3-9 (pg 3/4 3-50) Remote Shutdown Monitoring Instrumentation

   Number 4; Under Pressurizer Pressure, Total No. of Channel, the number of Channel was changed from 2 to 1. Because there is only one channel.

13) Table 4.3-6 (pg 3/4 3-51) Remote Shutdown Monitoring Instrumentation

   Number 2; Next to Channel Check, an asterisk has been added to M to show that it is applicable below P-6.

14 Table 3.3-10 (pg 3/4 3-53) Accident Monitoring Instrumentation.

   Number 11 has been deleted because it is not needed in reference to letter.

15) Table 4.3-7 (pg 3/4 3-54) Accident Monitoring Instrumentation Surveillance Requirements.

   Number 11 has been deleted because it is not necessary in reference to letter.

16) Section 4.3.3.9 (pg 3/4 3-60) Radioactive Liquid Effluent Monitoring Instrumentation

The following was added after the word Digital and before Channel; ", and Analog". This is added to give page consolidation.

17) Section 3.3.3.9 (pg 3/4 3-60) Delete page which discusses "Digital Channel Operational Check"

18) Table 4.3-9 (pg 3/4 3-63) Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements. The column was added for Analog Channel Operational Test because we are using Analog Channels not Digital.

19) Table 4.3-8 (pg 3/4 3-64) Table Notations.

Number 5 was added to represent the fact that we are working with Analog Instruments.

20) Table 3.3-13 & Table 4.3-13 (Page 3/4 3-66, 67 1& 3/4 3-70.71)

"Monitor" has been changed to "sampler" in the following tables:

1. Table 3.3-13; lines 1b, 1c, 4b, and 4c.

2. Table 4.3-9; lines 1b, 1c, 4b, and 4c.

The revision is the result of comments made by the NRC during a meeting concerning process radiation monitor calibrations, noting that several other stations have designated particulate and iodine channels as "samplers". This designation lessens the calibration requirements placed on particulate and iodine channels, thus making the current Byron calibration procedures acceptable to the NRC.

21) Table 4.3-9 (pg 3/4 3-72) Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements.

Number 5; 1RE-PR090 to 1RE-PR009 to correct type.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Pressurizer Pressure-Low (Above P-7)	4	2	3	1	6#
10. Pressurizer Pressure-High	4	2	3	1, 2	6#
11. Pressurizer Water Level-High (Above P-7)	3	2	2	1	7#
12. Reactor Coolant Flow-Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	7#
13. Steam Generator Water Level-Low-Low	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2	6#
14. Undervoltage-Reactor Coolant Pumps (Above P-7)	4-1/bus	2	3	1	6#
15. Underfrequency-Reactor Coolant Pumps (Above P-7)	4-1/bus	2	3	1	6#
16. Turbine Trip -~~Emergency Tripheader~~ ~~EHC~~ (Above P-7)					
a. ~~Low Auto-Stop Oil~~ Emergency Trip Header Pressure	3 / Train	2 / Train	2 / Train	1	7#
b. Turbine ~~Stop~~ Throttle Valve Closure	4	4	1	1	~~14#~~ 7 ?

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TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Low Reactor Coolant Flow - Low	
a. Single Loop (Above P-8)	$\leq$ 1 second
b. Two Loops (Above P-7 and below P-8)	$\leq$ 1 second
13. Steam Generator Water Level-Low-Low	$\leq$ 2 seconds
14. Undervoltage-Reactor Coolant Pumps (above P-7)	$\leq$ 1.5 seconds
15. Underfrequency-Reactor Coolant Pumps (above P-7)	$\leq$ 0.6 second
16. Turbine Trip (above P-7)	
a. ~~Low Auto Stop Oil~~ Emergency Trip Header Pressure	N.A.
b. Turbine ~~Stop~~ Throttle Valve Closure	N.A.
17. Safety Injection Input from ESF	N.A.
18. Reactor Coolant Pump Breaker Position Trip (Above P-7)	N.A.
19. Reactor Trip System Interlocks	N.A.
20. Reactor Trip Breakers	N.A.
21. Automatic Trip and Interlock Logic	N.A.

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TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level-Low-Low	S	R	M	N.A.	N.A.	1, 2
14. Undervoltage~~-Reactor~~ Coolant Pumps (Above P-7)	N.A.	R	N.A.	M	N.A.	1
15. Underfrequency~~-Reactor~~ Coolant Pumps (Above P-7)	N.A.	R	N.A.	M	N.A.	1
16. Turbine Trip (Above P-7)						
a. ~~Low Auto Stop Oil~~ Emergency Trip Header Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine ~~Stop~~ throttle Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
18. Reactor Coolant ~~Pump Breaker~~ Position Trip (Above P-7)	N.A.	N.A.	N.A.	R	N.A.	1
19. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	M	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	M (8)	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	M (8)	N.A.	N.A.	1

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TABLE 4.3-1 (Continued)

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TABLE NOTATIONS

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

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

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

(1) If not performed in previous 7 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore axial flux difference above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained, evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

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

(8) With power greater than or equal to the interlock Setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.

(9) Monthly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Monthly surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to an increase of twice the count rate within a 10-minute period.

(10) Setpoint verification is not applicable.

(11) At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.

(12) At least once per 18 months during shutdown verify that on a simulated Boron Dilution Doubling test signal ~~the normal~~ that CVCS ~~discharge~~ valves ~~will~~ 112 D&E close and ~~the centrifugal charging pump suction valves from the RWST will~~ open and 112 B&C open within 30 seconds.

(13) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Reactor Trip, P-4	2	2	2	1, 2, 3	22
c. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	20
d. Steam Generator Water Level, P-14 (High-High)	4 ~~3~~/stm. gen.	2/stm. gen. in any operating stm. gen.	2/stm. gen. in each operating stm. gen.	1, 2, 3	20

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	2.5	0.71	1.5	≤ 5.0 psig	≤ ~~3.6~~ 5.8 psig
d. Pressurizer Pressure-Low	13.0	10.71	1.5	≥ ~~1850~~ 1829 psig	≥ ~~1839~~ 1823 psig
e. Steam Line Pressure-Low (above P-11)	14.2	10.71	1.5	≥ 640 psig	≥ ~~610~~ 617 psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-3	5.0	0.71	1.5	≤ 20.0 psig	≤ ~~22~~ 21.0 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	5.0	0.71	1.5	≤ 20.0 psig	≤ ~~22~~ 21.0 psig
c. Containment Vent 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) Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation Trip Setpoints and Allowable Values.				

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	5.0	0.71	1.5	≤ ~~10.0~~ 8.2 psig	≤ ~~12.0~~ 9.2 psig
d. Steam Line Pressure-Low (Above P-11)	14.2	10.71	1.5	≥ 640 psig	≥ ~~610~~ 617 psig*
e. Steam Line Pressure-Negative Rate-High (Below P-11)	8.0	0.5	0	≤ -100 psi/s	≤ -110.0 psi/s**
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)	5.0	2.18	1.5	≤ ~~82%~~ 81.4 of narrow range instrument span	≤ ~~83%~~ 82.7 of narrow range instrument span

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	30.0	27.18	1.5	≥ ~~41%~~ 40.8 of narrow range instruement span	≥ ~~40%~~ 39.1 of narrow range instrument span
d. Undervoltage-RCP Bus-Start Motor Driven Pump and Diesel-Driven Pump	N.A.	N.A.	N.A.	≥ ~~70% RCP bus voltage (4890 volts)~~ 4920 volts	≥ ~~69% RCP bus voltage (4674 volts)~~ 4768 volts
e. Safety Injection-Start Motor-Driven Pump and Diesel-Driven Pump	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Loss of Power					
a. ESF Bus Undervoltage (Electromechanical Relaying)	N.A.	N.A.	N.A.	2870 volts	2870 volts ± 143 1.8 ± 0.1s time delay
b. ~~Grid Degraded~~ Voltage (Solid State Relaying)	N.A.	N.A.	N.A.	3804 volts 9s time delay	3804 volts ± 76 9 ± 0.9s time delay
9. Engineered Safety Feature Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	$\leq$ ~~1950~~ 1930 psig	no change $\leq$ ~~2050~~ psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
c. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	550°F	$\geq$ 548°F and $\leq$ 552°F
d. Steam Generator Water Level, P-14 (High-High)	See Item 5.b. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.				

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TABLE 3.3-9
REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Intermediate Range Neutron Flux	1PL06J	2	1
2. Source Range Neutron Flux	1PL06J	2	1
3. Reactor Coolant Temperature - Wide Range			
a. Hot Leg	1PL05J	1/loop	1/loop
b. Cold Leg	1PL05J	1/loop	1/loop
4. Pressurizer Pressure	1PL06J	~~2~~ 1	1
5. Pressurizer Level	1PL06J	2	1
6. Steam Generator Pressure	1PL04J/1PL05J	1/stm gen	1/stm gen
7. Steam Generator Level	1PL04J	1/stm gen	1/stm gen
8. RHR Flow Rate	LOCAL	2	1
9. RHR Temperature	LOCAL	2	1
10. Auxiliary Feedwater Flow Rate	1PL04J/1PL05J	1/stm gen	1/stm gen

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Intermediate Range Neutron Flux	M	N.A.
2. Source Range Neutron Flux	M*	N.A.
3. Reactor Coolant Temperature - Wide Range	M	R
4. Pressurizer Pressure	M	R
5. Pressurizer Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Level	M	R
8. RHR Flow Rate	M	R
9. RHR Temperature	M	R
10. Auxiliary Feedwater Flow Rate	M	R

* When Applicable below P-6

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TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2	1
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Narrow Range	1/steam generator	1/steam generator
8. Steam Generator Water Level - Wide Range	1/steam generator	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10. Auxiliary Feedater Flow Rate	2/steam generator	1/steam generator
~~11. Reactor Coolant System Subcooling Margin Monitor~~	~~1~~	~~1~~
12. PORV Position Indicator (Open/Closed)	1/Valve	1/Valve
13. PORV Block Valve Position Indicator (Open/Closed)	1/Valve	1/Valve
14. Safety Valve Position Indicator (Open/Closed)	1/Valve	1/Valve
15. Containment Floor Drain Sump Water Level (Narrow Range)	2	1
16. Containment Water Level (Wide Range)	2	1
17. In Core Thermocouples	4/core quadrant	2/core quadrant
18. Containment High Range Area Radiation	2	1
19. Containment Hydrogen Concentration	2	1
20. Neutron Flux (Power Range)	4	2
21. Auxiliary Building Vent Stack - Wide Range Noble Gas	1/stack	1/stack
22. Main Steam Line Radiation	1/stm line	1/stm line
23. Reactor Vessel Water Level	2	1

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TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. Steam Generator Water Level - Wide Range	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Auxiliary Feedater Flow Rate	M	R
~~11. Reactor Coolant System Subcooling Margin Monitor~~	~~M~~	~~R~~
12. PORV Position Indicator (Open/Closed)	M	R
13. PORV Block Valve Position Indicator (Open/Closed)	M	R
14. Safety Valve Position Indicator (Open/Closed)	M	R
15. Containment Floor Drain Sump Water Level (Narrow Range)	M	R
16. Containment Water Level (Wide Range)	M	R
17. In Core Thermocouples	M	R
18. Containment High Range Area Radiation	M	R*
19. Containment Hydrogen Concentration	S	Q
20. Neutron Flux (Power Range)	M	R
21. Auxiliary Building Vent Stack - Wide Range Noble Gas	M	R
22. Main Steam Line Radiation	M	R
23. Reactor Vessel Water Level	M	R

*CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10R/h and a one point calibration check of the detector below 10R/h with an installed or portable gamma source.

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RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION, or explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.7.1.7 why this inoperability was not corrected within the time specified.

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

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and DIGITAL CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-8.

and ANALOG

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RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION, or explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.7.1.7 why this inoperability was not corrected within the time specified.

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

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and DIGITAL CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-8.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	ANALOG CHANNEL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release					
Liquid Radwaste Effluent Line (0RE-PR001)	D	P	R(3)	Q(1)	N.A.
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release					
a. Essential Service Water ~~Outlet~~ RCFC 1A and 1C Outlet Line (1RE-PR002)~~ and 1RE-PR003)~~	D	M	R(3)	~~Q(2)~~ N.A.	Q(5)
b. Essential Service Water RCFC 1B and 1D Outlet (1RE-PR003)					
c. ~~b.~~ Station Blowdown Line (0RE-PR010)	D	M	R(3)	~~Q(2)~~ N.A.	Q(5)
3. Flow Rate Measurement Devices					
a. Liquid Radwaste Effluent Line (Loop-WX001)	D(4)	N.A.	R	Q	N.A.
b. Station Blowdown Line (Loop-CW032)	D(4)	N.A.	R	Q	N.A.

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TABLE 4.3-8 (Continued)

TABLE NOTATIONS

DIGITAL

(1) The DITIGAL 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) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

(5) The Analog Channel Operational Test shall also demonstrate that control room alarm ~~annunciation~~ 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.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. Plant Vent Monitoring System - Unit 1			
a. Noble Gas Activity Monitor-Providing Alarm			
1) High Range (1RE-PR028D)	1	*	39
2) Low Range (1RE-PR028~~A~~B)	1	*	39
b. Iodine ~~Monitor~~ Sampler (1RE-PR028C)	1	*	40
c. Particulate ~~Monitor~~ Sampler (1RE-PR028~~B~~A)	1	*	40
d. Effluent System Flow Rate Measuring Device (Loop-VA~~109~~019)	1	*	36
e. Sampler Flow Rate Measuring Device ~~(1FI-PR162)~~ (1FT-PR165)	1	*	36
2. Gaseous Waste Management System			
a. Hydrogen Analyzer (0AT-GW8000)	1	**	38
b. Oxygen Analyzer (0AT-GW804 and 0AT-GW8003)	2	**	41

DELETE

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TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
2 ~~1~~. Plant Vent Monitoring System – Unit Two			
a. Noble Gas Activity Monitor-Providing Alarm			
1) High Range (~~1~~2RE-PR028D)	1	*	39
2) Low Range (~~1~~2RE-PR028~~A~~B)	1	*	39
b. Iodine ~~Monitor~~ Sampler (~~1~~2RE-PR028C)	1	*	40
c. Particulate ~~Monitor~~ Sampler (~~1~~2RE-PR028~~B~~A)	1	*	40
d. Effluent System Flow Rate Measuring Device (Loop-VA~~109~~020)	1	*	36
e. Sampler Flow Rate Measuring Device (~~1~~2FI-PR162)	1	*	36
3 ~~2~~. Gaseous Waste Management System			
a. Hydrogen Analyzer (0AT-GW8000)	1	**	38
b. Oxygen Analyzer (0AT-GW004 and 0AT-GW8003)	2	**	41

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4. Gas Decay Tank System			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (0RE-PR002A and 2B)	2	*	35
b. Flow Rate Monitor	1	*	36
c. Sampler Flow Rate Monitor (0FI-PR159)	1	*	36
5. Containment Purge System			
a. Noble Gas Activity Monitor - Providing Alarm (1RE-PR001A)	1	*	37
b. Iodine Sampler ~~Monitor~~ (1RE-PR001C)	1	*	40
c. Particulate Sampler ~~Monitor~~ (1RE-PR001B)	1	*	40
d. System Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Monitor (1FI-PR100)	1	*	36
6. Radioactivity Monitors Providing Alarm and Automatic Closure of Surge Tank Vent Component Cooling Water Line (0RE-PR009 and 1RE-PR009)	2	*	42

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TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Plant Vent Monitoring System - Unit 1					
a. Noble Gas Activity Monitor - Providing Alarm					
1) High Range (1RE-PR028D)	D	M	R(3)	Q(2)	*
2) Low Range (1RE-PR028~~A~~ B)	D	M	R(3)	Q(2)	*
b. Iodine ~~Monitor~~ Sampler (1RE-PR028C)	D	M	R(3)	Q(2)	*
c. Particulate ~~Monitor~~ Sampler (1RE-PR028~~B~~ A)	D	M	R(3)	Q(2)	*
d. Effluent System Flow Rate Measuring Device (~~Loop~~ OFE-VA019)	D	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device (1FI-PR162)	D	N.A.	R	Q	*
2. Gaseous Waste Management System					
DELETE					
a. Hydrogen Analyzer (0AT-GW8000)	D	N.A.	Q(4)	M	**
b. Oxygen Analyzer (0AT-GW004 and 0AT-GW8003)	D	N.A.	Q(5)	M	**

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TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
2 ~~1~~. Plant Vent Monitoring System - Unit Two					
a. Noble Gas Activity Monitor - Providing Alarm					
1) High Range (~~1~~2RE-PR028D)	D	M	R(3)	Q(2)	*
2) Low Range (~~1~~2RE-PR028~~A~~B)	D	M	R(3)	Q(2)	*
b. Iodine ~~Monitor~~ Sampler (~~1~~2RE-PR028C)	D	M	R(3)	Q(2)	*
c. Particulate ~~Monitor~~ Sampler (~~1~~2RE-PR028~~B~~A)	D	M	R(3)	Q(2)	*
d. Effluent System Flow Rate Measuring Device (~~Loop-VA019~~) OFE-VA020	D	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device (~~1~~2FI-PR162)	D	N.A.	R	Q	*
3 ~~2~~. Gaseous Waste Management System					
a. Hydrogen Analyzer (0AT-GW8000)	D	N.A.	Q(4)	M	**
b. Oxygen Analyzer (0AT-GW004 and 0AT-GW8003)	D	N.A.	Q(5)	M	**

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4~~3~~. Gas Decay Tank System					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (ORE-PR002A and 2B)	P	P	R(3)	Q(1)	*
b. Flow Rate Monitor (OFIT - GW001)	P	N.A.	R	Q	*
c. Sampler Flow Rate Monitor (OFI-PR159)	D	N.A.	R	Q	*
5~~4~~. Containment Purge System					
a. Noble Gas Activity Monitor - Providing Alarm (1RE-PR001~~A~~B)	D	P	R(3)	Q(2)	*
b. Iodine ~~Monitor~~ Sampler (1RE-PR001C)	P	P	R(3)	N.A.	*
c. Particulate ~~Monitor~~ Sampler (1RE-PR001~~B~~A)	P	P	R(3)	N.A.	*
d. System Flow Rate Monitor (IFT-VQ093)	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor (1FI-PR100)	D	N.A.	R	Q	*

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TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Radioactivity Monitors Providing Alarm and Automatic Closure of Surge Tank Vent Component Cooling Water Line (ORE-PR009 and 1RE-~~PR090~~) PR009	D	M	R(3)	Q(1)	*

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ATTACHMENT F

(Section 3/4.4)

Circled items noted in this attachment have been previously submitted.

1) Section 3.4.3 (pg 3/4 4-9) Reactor Coolant System.

In Action 3.4.3.a, delete the word "backup" from "backup pressurizer heaters" and "backup heaters."

The work backup is deleted from ACTION to be consistent with what they are referring to in 3.4.3.

2) Section 4.4.4.3 (pg 3/4 4-10) Reactor Coolant System - Relief Valves

Surveillance 4.4.4.3 has been deleted. There is no emergency power supply for the PORV's and the block valves. The backup force to close the valves in an emergency is a supply of nitrogen filled accumulators.

3) Section 4.4.b.1 (pg 3/4 4-18) Reactor Coolant System.

Surveillance Requirement 4.4.6.1.a replaces "ANALOG" with "DIGITAL".

Byron Station will use Digital Channels in the Operational test because this (Digital Channels) happens to be the available instrument.

4) Section 4.4.6.1 (pg 3/4 4-18) Reactor Coolant System Leakage.

A new surveillance has been added to verify the containment floor drain collection sump is filled. The commitment for the surveillance is addressed in an NRC inspection report response.

5) Section 3.4.6.2 (pg 3/4 4-19) Reactor Coolant System - Operational Leakage

Action; Step b, "4 hours" has been changed to "24 hours" to allow time to do corrective actions without causing plant transient.

6) Section 4.4.6.2.1 (pg 3/4 4-20) Reactor Coolant System

4.4.6.2.1.c has a change in tolerance of ±15 to ±20 to be consistent with values previously given.

7) Section 3.4.8, Table 4.4-4 (pg 3/4 4-25, 4-26, 4-28) Reactor Coolant System.

Delete the phrase "of gross radioactivity" for the following:

Section 3.4.8.b
Action 3.4.8.d (two places)
Item 4.a for table 4.4-4

This was deleted because it is an inappropriate wording of the sentence which implies you are measuring "gross radioactivity" in grams.

(0433M)

ATTACHMENT F (Continued)

(Section 3/4.4)

8) Section 3.4.93 (pg 3/4 4-35) Reactor Coolant System.

Alteration of sentence in step a; "a lift setpoint not to exceed the limits of" is inserted after "with" and before "Figure 3.4-4". The words "nominal on" were deleted to avoid having to follow curve exactly.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with at least two groups of pressurizer heaters each having a capacity of at least 150 kW and a water level of less than or equal to 92% (1755 cubic feet).

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

4.4.3.3 The cross-tie for the pressurizer heaters to the ESF power supply shall be demonstrated OPERABLE at least once per 18 months by energizing the heaters.

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3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. Performance of a CHANNEL CALIBRATION, and

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

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

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

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

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

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LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

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

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

c. The containment air pressure instrumentation and reactor containment fan cooler outlets and inlets Dewcell and dry bulb temperature instrumentation.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

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

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

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

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

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OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 500 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within ~~4~~ 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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SURVEILLANCE REQUIREMENTS

4.4.6.2.1 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;

±20

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± ~~[illegible]~~ 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.

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microCuries per gram ~~of gross radioactivity.~~

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

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12-month period. The provisions of Specification 3.0.4 are not applicable;

b. With the total cumulative operating time at a reactor coolant specific activity greater than 1 microCurie per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6-month period, prepare and submit a Special Report to the Commission within 30 days, pursuant to Specification 6.7.2, indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable;

c. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

d. With the specific activity of the reactor coolant greater than $100/\bar{E}$ microCuries per gram ~~of gross radioactivity~~, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

* With T_{avg} greater than or equal to 500°F.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microCuries per gram ~~of gross radioactivity~~, perform the sampling and analysis requirements of Item 4a of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

For this ACTION statement, prepare and submit a Special Report to the Commissioner pursuant to Specification 6.7.2 within 30 days with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, and Chief, Accident Evaluation Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, while limit was exceeded and one ~~example~~ Sample after the radioiodine activity was reduced to less than the limit, including for each isotopic analysis, the date and time of sampling and the radioiodine concentrations,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded 1 microCurie per gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Radioactivity Determination**	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	Once per 14 days	1
3. Radiochemical for Ē Determination***	Once per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 μCi/gram DOSE EQUIVALENT I-131 or 100/Ē μCi/gram ~~of gross radioactivity,~~ and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

LIMITING CONDITION FOR OPERATION

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

a lift setpoint not to exceed the limits of

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

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

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

Mode 4

ACTION:

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

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

c. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.7.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

ATTACHMENT G

(Section 3/4.5)

1) Section 3.5.3 (pg 3/4 5-7) Emergency Core Cooling System

Delete the phrase "or equal to" from the Note, it comes after "than" and before "350°F". This is done to make note describe Mode 4.

2) Section 4.5.3.2 (pg 3/4 5-8) Emergency Core Cooling System

Delete the phrase "less than or equal to" from Section 4.5.3.2. It comes after "is" and before "350°F". This is done to make section be in description of Mode 4.

3) Section 3.5.4 (page 3/4 5-9) Refueling Water Storage Tank.

Delete this page. Information included in Section 3.1.2.6.

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3/4.5.2 ECCS SUBSYSTEMS - T_{avg} < 350°F

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.

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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.

Delete

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3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) and the heat traced portions of the associated flow paths shall be OPERABLE with:

a. A minimum contained borated water level of ~~88.7%~~ 89% (395,000 gallons),

b. A minimum boron concentration of 2000 ppm,

c. A minimum water temperature of 35°F, and

d. A maximum water temperature of 100°F.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 35°F or greater than 100°F.

c. At least once per 24 hours by verifying the RWST vent path temperature to be greater than or equal to 35°F when the outside air temperature is less than 35°F.

ATTACHMENT H

(Section 3/4.6)

Circled items noted in this attachment have been previously submitted.

1) Section 4.6.1.5 (pg 3/4 6-7) Containment Systems.

   Changed numbering System to Reflect the Subject in Question.

   Changed Description Title to put in main control vernacular.

2) Section 3.6.1.5 (pg 3/4 6-11) Containment Systems

   Deleted "provided... time" from 3.6.1.7b because Byron only has one line.

3) Section 4.7.1.7.3/4.6.1.7.4 (pg 3/4 6-12) Containment System.

   Replace "Operable" from 4.6.1.7.3 and insert "to maintain integrity".

   Replace "Operable" from 4.6.1.7.4 and insert "to maintain integrity".

4) Section 4.6.2.3 (pg 3/4 6-15) Containment System

   Deleted after "verifying" and before "flow" and phrase "a cooling water" and its place put "an essential service water" to describe the proper system description.

5) Table 3.6-1 (Pages 3/4 6-18 to 22) Containment Isolation Valves For clarification, "s" has been changed to "sec".

   Section 4.6.1.7.4 (pg 3/4 6-12) Containment System

   The measured leakage rate of 0.05La is more appropriate for the 8-inch valve instead of 0.012a. We cannot determine the basis for 0.01La.

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AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 122°F.

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

ACTION:

With the containment average air temperature greater than 122°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures of the running fans at the following locations and shall be determined at least once per 24 hours:

Location

1 ~~a.~~ RCFC ~~Unit A Return Air Riser,~~ Dry Bulb Inlet Temperature

1 ~~b.~~ RCFC ~~Unit B Return Air Riser,~~ Dry Bulb Inlet Temperature

1 ~~c.~~ RCFC ~~Unit C Return Air Riser, and~~ Dry Bulb Inlet Temperature

1 ~~d.~~ RCFC ~~Unit D Return Air Riser.~~ Dry Bulb Inlet Temperature

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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 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 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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SURVEILLANCE REQUIREMENTS

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

power removed

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

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

4.6.1.7.4 At least once per 3 months, each 8-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated ~~OPERABLE~~ TO MAINTAIN INTEGRITY by verifying that the measured leakage rate is less than ~~0.01~~ 0.05 L_a when pressurized to at least P_a, 43.6 psig.

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CONTAINMENT COOLING SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.6.2.3 Two electrically independent systems of containment cooling fans shall be OPERABLE with one fan to each system.

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

ACTION:

a. With one system of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore the inoperable system of cooling fans to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two systems of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore at least one system of cooling fans to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required systems of cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one system of the above required containment cooling fans inoperable and one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable system of containment cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each system of containment cooling fans shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) Starting each fan system in slow speed from the control room, and verifying that each fan system operates for at least 15 minutes, and

2) Verifying AN ESSENTIAL SERVICE WATER ~~a cooling water~~ flow rate of greater than or equal to 2660 gpm to each cooler.

b. At least once per 18 months by verifying that each fan system starts automatically on a Safety Injection test signal.

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TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

VALVE NO.	FUNCTION	ISOLATION TIME (s) Sec	TYPE OF OPERATOR
1. Phase "A" Isolation			
1CV8100	Chemical and Volume Control	10	Motor
1CV8112	Chemical and Volume Control	10	Motor
1CV8152	Chemical and Volume Control	10	Air Operator with solenoid accessory
1CV8160	Chemical and Volume Control	10	Air Operator with solenoid accessory
1WO056A	Chilled Water	50	Motor
1WO056B	Chilled Water	50	Motor
1WO020A	Chilled Water	50	Motor
1WO006A	Chilled Water	50	Motor
1WO020B	Chilled Water	50	Motor
1WO006B	Chilled Water	50	Motor
1CC9437B	Component Cooling	10	Air Operator with solenoid accessory
1CC9437A	Component Cooling	10	Air Operator with solenoid accessory
1FP010	Fire Protection	12	Air Operator with solenoid accessory
1FP011	Fire Protection	12	Air Operator with solenoid accessory
1IA065	Instrument Air	15	Air Operator with solenoid accessory
1IA066	Instrument Air	15	Air Operator with solenoid accessory
1OG079	Off-gas	60	Motor
1OG080	Off-gas	60	Motor
1OG081	Off-gas	60	Motor
1OG057A	Off-gas	60	Motor
1OG082	Off-gas	60	Motor
1OG083	Off-gas	60	Motor
1OG084	Off-gas	60	Motor
1OG085	Off-gas	60	Motor
1PR001A	Process Radiation	4.5	Air Operator with solenoid accessory
1PR001B	Process Radiation	4.5	Air Operator with solenoid accessory
1PR066	Process Radiation	5	Air Operator with solenoid accessory

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FUNCTION	ISOLATION TIME ~~(s)~~ Sec	TYPE OF OPERATOR
1. Phase "A" Isolation (Continued)			
1PS228A	Process Sampling	~~0.10~~ NA*	Solenoid
1PS229A	Process Sampling	~~0.10~~ NA*	Solenoid
1PS230A	Process Sampling	~~0.12~~ NA*	Solenoid
1PS228B	Process Sampling	~~0.10~~ NA*	Solenoid
1PS229B	Process Sampling	~~0.10~~ NA*	Solenoid
1PS230B	Process Sampling	~~0.12~~ NA*	Solenoid
1PS9354A	Process Sampling	10	Air Operator with solenoid accessory
1PS9354B	Process Sampling	10	Air Operator with solenoid accessory
1PS9355A	Process Sampling	10	Air Operator with solenoid accessory
1PS9355B	Process Sampling	10	Air Operator with solenoid accessory
1PS9356A	Process Sampling	10	Air Operator with solenoid accessory
1PS9356B	Process Sampling	10	Air Operator with solenoid accessory
1PS9357A	Process Sampling	10	Air Operator with solenoid accessory
1PS9357B	Process Sampling	10	Air Operator with solenoid accessory
1RE9157	Reactor and Containment Drains to Radiowaste	10	Air Operator with solenoid accessory
1RE9159A	Reactor and Containment Drains to Radiowaste	10	Air Operator with solenoid accessory
1RE9159B	Reactor and Containment Drains to Radiowaste	10	Air Operator with solenoid accessory
1RE9160A	Reactor and Containment Drains to Radiowaste	10	Air Operator with solenoid accessory
1RE9160B	Reactor and Containment Drains to Radiowaste	10	Air Operator with solenoid accessory
1RE1003	Reactor and Containment Drains to Radiowaste	10	Air Operator with solenoid accessory
1RE9170	Reactor and Containment Drains to Radiowaste	10	Air Operator with solenoid accessory
1RY8025	Reactor Coolant Pressurizer	10	Air Operator with solenoid accessory
1RY8026	Reactor Coolant Pressurizer	10	Air Operator with solenoid accessory
1RY8033	Reactor Coolant Pressurizer	10	Air Operator with solenoid accessory
1RY8028	Reactor Coolant Pressurizer	10	Air Operator with solenoid accessory

* Proper Valve operation will be demonstrated by verifying the valve strokes to its required position.

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FUNCTION	ISOLATION TIME (s) sec	TYPE OF OPERATOR
1. Phase "A" Isolation (Continued)			
1SI8880	Safety Injection	10	Air Operator with solenoid accessory
1SI8964	Safety Injection	10	Air Operator with solenoid accessory
1SI8871	Safety Injection	10	Air Operator with solenoid accessory
1SI8888	Safety Injection	10	Air Operator with solenoid accessory
1SA032	Service Air	4.5	Air Operator with solenoid accessory
1SA033	Service Air	4.5	Air Operator with solenoid accessory
1SD002C	Steam Generator Blowdown	7.5	Air Operator with solenoid accessory
1SD005B	Steam Generator Blowdown	3	Air Operator with solenoid accessory
1SD002D	Steam Generator Blowdown	7.5	Air Operator with solenoid accessory
1SD002A	Steam Generator Blowdown	7.5	Air Operator with solenoid accessory
1SD005A	Steam Generator Blowdown	3	Air Operator with solenoid accessory
1SD002B	Steam Generator Blowdown	7.5	Air Operator with solenoid accessory
1SD002E	Steam Generator Blowdown	7.5	Air Operator with solenoid accessory
1SD005C	Steam Generator Blowdown	3	Air Operator with solenoid accessory
1SD002F	Steam Generator Blowdown	7.5	Air Operator with solenoid accessory
1SD002G	Steam Generator Blowdown	7.5	Air Operator with solenoid accessory
1SD005D	Steam Generator Blowdown	3	Air Operator with solenoid accessory
1SD002H	Steam Generator Blowdown	7.5	Air Operator with solenoid accessory
1RF026	Waste Disposal	15	Air Operator with solenoid accessory
1RF027	Waste Disposal	15	Air Operator with solenoid accessory

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FUNCTION	ISOLATION TIME (s) sec	TYPE OF OPERATOR
2. Feedwater Isolation			
1FW009A	S/G Feedwater	5	Hydraulic Operator
1FW009B	S/G Feedwater	5	Hydraulic Operator
1FW009C	S/G Feedwater	5	Hydraulic Operator
1FW009D	S/G Feedwater	5	Hydraulic Operator
1FW035A	S/G Feedwater	6	Air Operator with solenoid accessory
1FW035B	S/G Feedwater	6	Air Operator with solenoid accessory
1FW035C	S/G Feedwater	6	Air Operator with solenoid accessory
1FW035D	S/G Feedwater	6	Air Operator with solenoid accessory
1FW039A	S/G Feedwater	6	Air Operator with solenoid accessory
1FW039B	S/G Feedwater	6	Air Operator with solenoid accessory
1FW039C	S/G Feedwater	6	Air Operator with solenoid accessory
1FW039D	S/G Feedwater	6	Air Operator with solenoid accessory
1FW043A	S/G Feedwater	6	Air Operator with solenoid accessory
1FW043B	S/G Feedwater	6	Air Operator with solenoid accessory
1FW043C	S/G Feedwater	6	Air Operator with solenoid accessory
1FW043D	S/G Feedwater	6	Air Operator with solenoid accessory

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FUNCTION	ISOLATION TIME (5) sec	TYPE OF OPERATOR
3. Containment Ventilation Isolation			
1VQ005A	Containment Purge	5	Air Operator with solenoid accessory
1VQ005B	Containment Purge	5	Air Operator with solenoid accessory
1VQ005C	Containment Purge	5	Air Operator with solenoid accessory
1VQ003	Containment Purge	5	Air Operator with solenoid accessory
1VQ002A	Containment Purge	5	Hydraulic Operator
1VQ002B	Containment Purge	5	Hydraulic Operator
1VQ004A	Containment Purge	5	Air Operator with solenoid accessory
1VQ004B	Containment Purge	5	Air Operator with solenoid accessory
1VQ001A	Containment Purge	5	Hydraulic Operator
1VQ001B	Containment Purge	5	Hydraulic Operator
4. Phase "B"/Components Isolation			
1CC9414	Component Cooling	10	Motor
1CC9416	Component Cooling	10	Motor
1CC685	Component Cooling	10	Motor
1CC9438	Component Cooling	10	Motor
1CC9413A	Component Cooling	10	Motor
1CC9413B	Component Cooling	10	Motor
5. Safety Injection/Main Steam Isolation			
1MS001D	Main Steam	5	Hydraulic
1MS101D	Main Steam	10	Air Operator with Solenoid Accessory
1MS001B	Main Steam	5	Hydraulic
1MS101B	Main Steam	10	Air Operator with Solenoid Accessory
1MS001A	Main Steam	5	Hydraulic
1MS101A	Main Steam	10	Air Operator with Solenoid Accessory
1MS001C	Main Steam	5	Hydraulic
1MS101C	Main Steam	10	Air Operator with Solenoid Accessory

ATTACHMENT I

(Section 3/4.7)

Circled items noted in this attachment have been previously submitted.

1) Section 4.7.1.2.1 (pg 3/4 7-5) Plant Systems

   Delete 4.7.1.2.1.a.3, from Section 4.7.1.2.1 because (a.2) verifies correct position of system. Number (a.3.) does not address any valves not already addressed in (a.2). There is no auto control referred to in (a.3).

2) Section 4.7.1.2.1 (pg 3/4 7-5) Plant System

   Inserted in [b.2(d)], between "Bus" and "Undervoltage" the number <u>141</u>, to add a more complete description of the Bus desired.

3) Section 3.7.3 (pg 3/4 7-11) Plant Systems

   The Component Cooling Water System loops do not lend themselves to the present LCO. A pump or heat exchanger may service any loop. The specification must be expanded to account for the components.

4) Section 3.7.6/4.7.6 (pg 3/4 7-15) Plant System

   Changed all phrases "Control Room Emergency Air Cleanup Systems" to "Control Room Ventilation Cleanup Systems" to give proper identification of System. In Section 4.7.6.b the word "duct" was inserted after "the" and before "heaters" to give a thorough description of equipment.

5) Section 4.7.6.c (pg 3/4 7-16) Plant System

   In Section 4.7.6.c.1) The word "Pressurization" was deleted from the Section and in its place, after the word "The" and before "System" the word "Makeup", was inserted to give proper description to system being used.

   In Section 4.76.6.c.2) The Criteria 0.2% was changed to 0.175% to Comply with number from Regulatory Guide 1.52.

   In Section 4.7.6.c.3) The word "Pressurization" was deleted from the Section and in its place, after the word "The" and before "System" the word "Makeup", was inserted to five proper description to system being used.

6) Section 4.7.6.d (Page 3/4 7-16) Control Room Emergency Air Cleanup System

   Change to 0.175% from 0.2% is necessary.

   Table 3 of Regulatory Guide 1.52 "Laboratory Tests for Activated Carbon" has a value of .175% for methyl iodine penetration.

7) Section 4.7.6.e (pg 3/4 7-16) Plant System

In Section 4.7.6.e.1) The word "Pressurization" was deleted from the section and in its place, after the word "The" and before "System" the word "Makeup", was inserted to give proper description to system being used.

In Section 4.7.6.e.2) The phrase "Smoke Density High" was deleted and in its place "ESF actuation signal" after the word "a" and before the word "or" to give proper name to the System described in the Section.

8) In Section 4.7.6.e.2) the word "recirculation" was deleted and in its place is inserted the word "makeup" to give proper name description to the mode of operation.

9) Section 4.7.6.e.4 (pg 3/4 7-17) Plant System Operation

In Section 4.7.6.4 the word "that" located after "Verifying" and before "the" was deleted to give correct grammatical format. The "s" was deleted from heater to express the use of a singular heater unit. The word "dissipate" was deleted and in its place the words "power consumption is" to give description of what is actually occurring in process. The words "in accordance with ANSI N510-1975."

After number 4 a number 5 stating "Verify heaters perform in accordance with N510-1975." was added to ensure that heaters are in accordance with N510-1975.

10) Section 4.7.6.f (pg 3/4 7-17) Plant System

In Section 4.7.6.f the words "and bypass leakage" was deleted because bypass leakage testing will not be used at Byron.

The word "Pressurization" was deleted from 4.7.6.f and in its place, after the word "the" and before the word "System", the word "Makeup" was inserted to give correct name description to the System.

11) Section 3.7.10.1 (pg 3/4 7-28) Plant System.

Changed the word "Spray" located after "or" and before "System" to "Foam" to give system proper description.

12) Table 3.7.5 (pg 3/4 7-35) Fire Hose Stations.

The corrections made in Table 3.7-5 were to add clarification to where equipment are located and to give proper name of equipment used at Byron Station.

ATTACHMENT I (Continued)

(Section 3/4.7)

13) Table 3.7-5 (pg 3/4 7-35) Fire Hose Stations.

The corrections made in Table 3.7-5 were to add clarification to where equipment are located and to give proper name of equipment used at Byron Station.

14) Table 3.7-5 (pg 3/4 7-37) Fire Hose Stations.

The corrections made in Table 3.7-5 were to add clarification to where equipment are located and to give proper name of equipment used at Byron Station.

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SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying by flow or position check 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; and

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

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

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

2) Verifying that the motor-driven pump and the direct-driven diesel pump start automatically upon receipt of each of the following test signals:

a) ESF, or

b) Steam Generator Water Level Low-Low from one steam generator, or

c) Undervoltage on Reactor Coolant Pump 6.9 kV Buses (2/4), or

d) ESF Bus 141 Undervoltage (motor-driven pump only).

4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 2 by verifying normal flow to each steam generator.

4.7.1.2.3 The auxiliary feedwater pump diesel shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying the fuel level in its day tank;

b. At least once per 92 days by verifying that a sample of diesel fuel from its day tank, obtained in accordance with ASTM-D270-1975 is within the acceptable limits specified in Table 1 of ASTM-D975-1977 when checked for viscosity, water, and sediment; and

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with its manufacturer's recommendations for this class of service.

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3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

REPLACE WITH THE ATTACHED "AA"

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

2) Each Component Cooling Water System pump starts automatically on a Safety Injection test signal.

LIMITING CONDITION FOR OPERATION

3.7.3 The Component Cooling Water System shall be OPERABLE with:

a. Two safety loops serving the RH pumps and RH heat exchangers.

b. Two component cooling water pumps powered from 4 KV busses 141 and 142.

c. Two component cooling heat exchangers.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

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

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

c. With only one heat exchanger OPERABLE, restore at least two heat exchangers to OPERABLE status within 7 days or be in at least HOT STANDBY within the next

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

SURVEILLANCE REQUIREMENTS

4.7.3.1 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 its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety-related equi[illegible] actuates to its correct position [illegible]ing containment isolation valves) on a SI tes[illegible]

4.7.3.2 At least two [illegible]ompo[illegible]nt 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

requirements for the routine operation of the component cooling system.

b. By verifying at least once per 18 months during shutdown that each component cooling pump starts automatically on a SI test signal. This will include a test of the common component cooling pump while powered from 4KV buses 141 and 142.

4.7.3.3 At least two component cooling heat exchangers shall be verified OPERABLE at least once per 31 days by:

a. Verifying that each ~~valve~~ component cooling heat exchanger inlet and outlet valve is OPERABLE.

b. Verifying that Essential Service Water is available to each component cooling ~~water~~ heat exchanger.

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3/4.7.6 CONTROL ROOM ~~EMERGENCY AIR CLEANUP~~ Ventilation SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 Two independent Control Room ~~Emergency Air Cleanup~~ Ventilation Systems shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION:

MODES 1, 2, 3 and 4:

With one Control Room ~~Emergency Air Cleanup~~ Ventilation System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one Control Room ~~Emergency Air Cleanup~~ Ventilation System inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Room ~~Emergency Air Cleanup~~ Ventilation System in the recirculation mode.

b. With both Control Room Emergency Air Cleanup Ventilation Systems inoperable, or with the OPERABLE Control Room ~~Air Cleanup~~ Ventilation System, required to be in the recirculation mode by ACTION a. not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.7.6 Each Control Room ~~Emergency Air Cleanup~~ Ventilation System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 90°F;

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the duct heaters operating;

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SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6000 cfm ± 10% for the ~~Pressurization~~ MAKEUP System ~~and 51,000 cfm ± 10% for the Recirculation System~~;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl ~~iodine~~ iodide penetration of less than ~~0.2%~~ 0.175%; and

3) Verifying a system flow rate of 6000 cfm ± 10% for the ~~Pressurization~~ MAKEUP System ~~and 51,000 cfm ± 10% for the Recirculation System~~ when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodine penetration of less than ~~0.2%~~ 0.175%;

e. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.0 inches Water Gauge while operating the system at a flow rate of 6000 cfm ± 10% for the ~~Pressurization~~ MAKEUP System ~~and 51,000 cfm ± 10% for the Recirculation System~~,

2) Verifying that on a ~~Smoke Density-High~~ ESF actuation signal or High Radiation-Control Room, Outside Air Intake, or Turbine Building Intake test signal, the system automatically switches into a ~~recirculation~~ MAKEUP mode of operation with flow through the HEPA filters and charcoal adsorber banks,

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SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that the system maintains the control room at a positive nominal pressure of greater than or equal to 1/8 inch Water Gauge relative to ambient pressure in areas served by the system, and

4) Verifying ~~that~~ the heaters POWER CONSUMPTION IS ~~dissipate~~ 27.2 ± 2.7 kW. ~~when tested in accordance with ANSI N510-1975.~~

5) Verify heaters perform in accordance with ANSI N510-1975.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration ~~and bypass leakage~~ testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 6000 cfm ± 10% for the ~~Pressurization~~ MAKEUP System ~~and 51,000 cfm ± 10% for the Recirculation System; and~~

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 6000 cfm ± 10% for the ~~Pressurization~~ MAKEUP System ~~and 51,000 cfm ± 10% for the Recirculation System.~~

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3/4.7.10 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.10.1 The Fire Suppression Water 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~~ Foam System required to be OPERABLE per Specifications 3.7.10.2 and 3.7.10.5.

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 System otherwise inoperable establish a backup Fire Suppression Water System within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.10.1.1 The Fire Suppression Water 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,

TABLE 3.7-5

FIRE HOSE STATIONS

LOCATION		ELEVATION	HOSE RACK REEL	ANGLE VALVE
Aux. Roof				
L-10:	South wall of U-1 safety valve penthouse	481	1	0FP331
L-26:	North wall of U-2 safety valve penthouse	481	2	0FP338
Aux. Bldg.				
S-18:	By dumb waiter	480	233	0FP458
S-15:	By U-1 prefilters (near stairs)	471	176	0FP329
S-21:	By U-2 prefilters (near stairs)	471	177	0FP334
Q-17:	Wall by elevator in upper cable room	469	244 *	0FP469
Q-19:	Wall by stairs in upper cable room	469	252 *	0FP477
L-11:	outside South~~west~~ east corner of ~~upper cable room~~ UCSR A-1	467	240	0FP465
L-14:	By the south~~west~~ east corner door of room ~~1EE3~~ UCSR C-1	467	241 *	0FP466
M-13:	By the north~~east~~ west corner of ~~room 1EE1~~ UCSR A-1	467	242 *	0FP467
Q-13:	In the north~~east~~ west corner of room ~~1EE2~~ UCSR B-1	467	243 *	0FP468
~~P-18~~ Northwest corner of UCSR C-1:	West of the elevator in upper cable room	467	245 *	0FP470
M-18:	North wall of ~~room 1EE-3~~ UCSR C-1	467	246 *	0FP471
M-18:	South wall of ~~room 2EE-3~~ UCSR C-2	467	247 *	0FP472
L-25:	outside North~~west~~ east corner of room ~~2EE-1~~ UCSR A-2	467	248	0FP473
L-22:	in the North~~west~~ east corner of room ~~2EE-3~~ UCSR C-2	467	249 *	0FP474
M-23:	in the South~~east~~ west corner of room 2~~EE-1~~ UCSR A-2	467	250 *	0FP475
P-20:	West wall of room ~~2EE-3~~ UCSR C-2	467	251 *	0FP476
Q-23:	in the Southwest corner of room 2EE-2 UCSR B-2	467	253 *	0FP478
S-21:	By U-2 ~~prefilters~~ VA filters (U-2 side)	464	232	0FP457
S-15:	By ~~U-1 prefilters~~ VA filters (U-1 side)	464	234	0FP459
S-21:	By ~~U-2 HEPA + prefilter room~~ VA filters (U-2 side)	456	231	0FP456
S-15:	By ~~U-1 HEPA + prefilter room~~ VA filters (U-1 side)	456	235	0FP460
L-10:	By Control room refrig. units	387	106	0FP385
L-12:	By blowdown after filters AF	387	107	0FP384
M-18:	By ~~Aux. Feedwater motor driven pump~~ 1A	387	108	0FP383
N-23:	By ~~remote shutdown panels~~ U-1 RSP	387	111	0FP376
Q-15:	By 480V MCC 132X3	387	113	0FP382
V-18:	By ~~Rad monitor~~ letdown heat exchanger	387	114	0FP379

* Fire Hoses that do not Supply the primary means of fire Suppression

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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 1AP74J~~ 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
M-18:	North wall ~~U-1 by door~~	444	238 *	0FP463
L-7:	East Wall ~~Southeast corner~~ of ~~lower cable room~~ LCSR A-1	443	207 *	0FP327
M-10:	In the ~~By east door to room 171~~ Southeast corner of LCSR B-1	443	208 *	0FP327
P-10:	~~By west door to room 171~~ Southwest " " " "	443	209 *	0FP325
M-13:	South wall of ~~room 173~~ LCSR C-1	443	210 *	0FP326
P-15 ~~P-13~~:	West wall of ~~room 174~~ 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~~ west 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
S-21:	East of ~~By~~ U-2 ~~pipe tunnel trolley beam~~ hydrogen recombiner	405	94	0FP348
V-21:	West of ~~By~~ U-2 hydrogen recombiner	405	95	0FP345
V-15:	West of ~~By~~ U-1 hydrogen recombiner ~~control panel~~	405	96	0FP316
S-15:	East of ~~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 tanks	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

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TABLE 3.7-5 (Continued)

FIRE HOSE STATIONS

LOCATION		ELEVATION	HOSE RACK REEL	ANGLE VALVE
Aux. Bldg. (Continued)				
P-11:	By recycle evaporator feed pumps	350	151	0FP381
M-13:	By ~~480V MCC 133X6~~ U-1 Stairs	350	152	0FP370
N-23:	By gas decay tanks	350	154	0FP352
Q-19:	By "B" Aux. Bldg. Equip. drain tank	350	155	0FP365
Q-17:	By "A" Aux. Bldg. Equip. drain tank	350	156	0FP371
Q-13:	By collection sump pumps	350	157	0FP380
S-18:	Between moderating heat exchangers	350	158	0FP354
V-18:	Between chiller units BR	350	161	0FP353
W-15:	By ~~Cont. spray pump A~~ CS pump 1A	350	163	0FP367
M-13:	By leak detection sump	334	165	0FP448
P-18:	By elevator pit	334	166	0FP449
Fuel Hand. Bldg.				
Z-15:	South of decon. area	430	170	0FP389
X-21:	North of spent fuel pool	430	171	0FP386
Z-15:	By 480V MCC 134X6	405	172	0FP388
AA-19:	~~By spent fuel pit heat exchangers~~ Outside FC Pump Room	405	173	0FP387
Cont. #1				
R-17:	By reactor head assembly area	430	62	1FP163
R-2:	By accumulator tank 1C	430	63	1FP154
R-7:	By hatch Equipment	430	64	1FP160
R-12:	By charcoal filter 1A	430	65	1FP157
R-17:	~~Between return risers~~ By South stairs	403	98	1FP164
R-2:	By ~~incore flux mapping~~ RCFC 1C	403	99	1FP155
R-7:	By pressurizer (outside missle shield)	403	100	1FP161
R-12:	By panel 1PL69J	403	101	1FP158
R-12:	By ~~panel 1PL50J~~ PRT	381	143	1FP159
R-17:	By ~~RC fans 1B + 1A~~ South stairs	381	144	1FP162
R-2:	By ~~panel 1PL66J~~ RCFC 1C	381	145	1FP156
R-7:	By panel 1PL52J	381	146	1FP165

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ATTACHMENT J

(Section 3/4.8)

Circled items noted in this attachment have been previously submitted.

1) Section 3.8.1.1 (pg 3/4 8-1) Electrical Power System.

- In sections 3.8.1.1.a and 3.8.1.2.a, change "the Onsite Class 1E Distribution System with:" to "each Onsite Class 1E 4KV BUS with:"

- In sections 3.8.1.1.a.1 and 3.3.1.2.a.1, change the word "circuit" to "line". This clarifies the beginning and end points of the two required circuits.

- In section 3.8.1.1.a.2 and 3.8.1.2.a.2, the sentence should read "Either of the two transformers forming a system auxiliary transformer bank capable of supplying the buses which are normally supplied by both transformers forming the system auxiliary transformer bank." This clarifies that either transformer is an SAT bank and can be used to supply either Clas 1E 4KV bus.

- Items a(1) and (2) are being replaced to provide clarification of the two required offsite power circuits. Also provides clarification that either transformer of a System Auxiliary Transformer bank is adequate.

2) Section 4.8.1.2.e.4.b (pg 3/4 8-4) Electrical Power System

Add "of the loads" following "After energization" on e.4.b. This is done for clarification.

3) Section 3.8.1.2 (pg 3/4 8-9) A.C. Sources

Items a (1) and (2) are being replaced to provide clarification of the required offsite power source. Clarifies that either SAT bank transformer is adequate.

4) Action 3.8.2.1.a (pg 3/4 3-10) A.C. Sources

Add "and/or Charger" after the word "bank" so that the Action refer to a "battery and/or charger bank...". This is added for clarification of the Action.

5) Section 4.8.2.1.2 (pg 3/4 8-11) Electrical Power System

Add "*" after the word "ohm: for Surveillance Requirements 4.8.2.1.2.b.2 and 4.8.2.1.2.c.3. Add the following note at the bottom of the page as follows:

"Obtained by subtracting the normal resistance of the connecting bus bar from the measured cell-to-cell and terminal connection resistance." At Byron, that batteries are made up of 4 rows of battery cells. While the resistance limit can be met on cell-to-cell measurements of adjacent cells, it cannot be met on cell-to-cell measurements between cells in different rows. This is due to the intrinsic resistance of the connecting bus bars, which must be much longer between rows of cells that between adjacent cells. Because the concern here is that the terminal connections might be poor, subtracting the normal resistance of the bus bar itself from the measured value of cell-to-cell resistance will still provide the resistance of the terminal connection.

Surveillance Requirement 4.8.2.1.2.b.3 should read "3) The electrolyte temperature of all connected cells is above 60°F". This new statement is for clarification of what need to be checked.

6) Section 3.8.3.1 (pg 3/4 8-14) Electrical Power System

- Section 3.8.3.1.c should read as follows:

  "c. 120-Volt AC Instrument Busses consisting of:

  1. Instrument Bus 111 energized from its associated inverter connected to DC Bus 111.

  2. Instrument Bus 112 energized from its associated inverter connected to DC Bus 112.

  3. Instrument Bus 113 energized from its associated inverter connected to DC Bus 111.

  3. Instrument Bus 114 energized from its associated inverter connected to DC Bus 112.

  This places the AC Instrument Busses in a form similar to that of parts a and b.

- Change "vital" to "Instrument" in both places in Action 3.8.3.1.b. to reflect station terminology.

- Change Action 3.8.3.1.c to read:

"c. With a maximum of an A.C. inverter inoperable or not connected to its D.C. power supply, its associated A.C. Instrument bus may be powered from its regulating transformer power supply for up to 72 hrs; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours."

Byron's design provides a self-regulating power supply to th A.C. inst. busses in addition to the AC inverter. This self-regulating power supply is energized from an ESF 480-volt bus. The change to Action C will allow us to operate a maximum of one AC INST BUS from the self-regulating power supply for up to 72 hours. This will allow more time to repair a faulty inverter. Note that if the self-regulating power supply were to be lost, Action b. will still apply.

- Delete Note* because Byron does not need to disconnect the inverters from their DC bus in order to place an equalizing charge on the batteriel; therefore this note is not required.

7) Surveillance Requirement 4.8.4.1.a.1 (pg 3/4 8-17)

Chane "7kV" to "6.9kV" to agree with plant equipment and other points of the Technical Specification Basis.

8) Surveillance Requirements 4.8.4.1.a.2 (pg 3/4 8-18)

Chane the phrase "...10% of each type of 480 volt circuit breaker." to "10% of each type of 480-Volt/125-Volt DC circuit breaker." This change is made so that the new sentence agrees with table 3.8-1.

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3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

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

a. Two physically independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System with:

   1) Each system auxiliary transformer energized from an independent transmission circuit, and

   2) One of the two transformers forming a system auxiliary transformer bank.

REPLACE WITH "A"

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

   1) A separate day tank containing a minimum volume of 450 gallons of fuel,

   2) A separate Fuel Oil Storage System containing a minimum volume of 42,000 gallons of fuel, and

   3) A separate fuel transfer pump.

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

ACTION:

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

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

"A"

a. Each Class 1E 4KV bus capable of being powered from:

1) Either transformer making up the associated units normal System Auxiliary Transformer bank AND

2) Either transformer making up the opposite units System Auxiliary Transformer bank (via the unit cross-tie breakers) WITH

Each units System Auxiliary Transformer bank energized from an independent transmission line.

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SURVEILLANCE REQUIREMENTS (Continued)

4) An impurity level of less than 2 mg of insolubles per 100 ml when tested in accordance with ASTM-D2274-70, analysis shall be completed within 7 days after obtaining the sample but may be performed after the addition of new fuel oil; and

5) The other properties specified in Table 1 of ASTM-D975-1977 and Regulatory Guide 1.137, Revision 1, October 1979, Position 2.a., when tested in accordance with ASTM-D975-1977, analysis shall be completed within 14 days after obtaining the sample but may be performed after the addition of new fuel oil.

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

1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2) Verifying the generator capability to reject a load of greater than or equal to 1034 kW ~~(SX Pump)~~ while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 4.5 Hz,

3) Verifying the diesel generator capability to reject a load of 5500 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection,

4) Simulating a loss of ESF bus voltage by itself, and:

a) Verifying de-energization of the ESF busses and load shedding from the ESF busses, and

b) Verifying the diesel starts on the auto-start signal, energizes the ESF busses with permanently connected loads within 10 seconds, energizes the auto-connected safe shutdown loads through the load sequencing timer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After of the loads energization, the steady-state voltage and frequency of the ESF busses shall be maintained at 4160 ± 420 volts and 60 ± 4.5 Hz during this test.

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LIMITING CONDITION FOR OPERATION

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

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System with:

   1) A system auxiliary transformer energized by one circuit of the offsite transmission network, and

   2) One of the two transformers forming system auxiliary transformer bank.

REPLACE WITH "B"

b. One diesel generator with:

   1) A day tank containing a minimum volume of 450 gallons of fuel,

   2) A fuel storage system containing a minimum volume of 42,000 gallons of fuel, and

   3) A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the spent fuel pool, and within 8 hours, depressurize and vent the Reactor Coolant System through at least a (2) square inch vent. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.5)), and 4.8.1.1.3.

"B"

a. One Class 1E 4KV bus capable of being powered from:

1) Either transformer making up the associated units System Auxiliary Transformer bank OR -

2) Either transformer making up the opposite units System Auxiliary Transformer bank (via the unit cross-tie breakers) WITH

The System Auxiliary Transformer bank supplying the 4 KV bus energized from an offsite transmission line.

ELECTRICAL POWER SYSTEMS

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3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-Volt D.C. Bus 111 fed from Battery 111, and its associated full capacity charger, and

b. 125-Volt D.C. Bus 112 fed from Battery 112, and its associated full capacity charger.

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

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery and/or charger bank 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.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Specification 4.8.2.1.2a.1) within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

c. With one 125-Volt bus inoperable, restore the inoperable bus to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD STANDBY within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1.1 Each D.C. bus shall be determined OPERABLE and energized from its battery at least once per 7 days by verifying correct breaker alignment.

4.8.2.1.2 Each 125-volt battery bank and its associated charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The parameters in Table 4.8-2 meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to 125-volts on float charge.

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 105 volts, or battery overcharge with battery terminal voltage above 145 volts, by verifying that:

1) The parameters in Table 4.8-2 meet the Category B limits,

2) There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohm,* and

3) The average electrolyte temperature of ~~at least every sixth~~ all connected cells is above 60°F.

c. At least once per 18 months by verifying that:

1) The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2) The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohm,* and

4) The battery charger will supply a load equal to the manufacturer's rating for at least 8 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for 240 minutes when the battery is subject to a battery service test;

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1.2d.;

f. At least once per 18 months during shutdown, by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

*Obtained by subtracting the normal resistance of the connecting bus bar from the measured cell-to-cell and terminal connection resistance.

ELECTRICAL POWER SYSTEMS

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3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner with tie breakers open (both) between redundant busses within the unit (and between units at the same station):

a. Division 11 A.C. ESF Busses consisting of:
   1) 4kV ~~4160-Volt~~ Bus 141,
   2) 480-Volt Bus 131X, and
   3) 480-Volt Bus 131Z.

b. Division 12 A.C. ESF Busses consisting of:
   1) 4kV ~~4160-Volt~~ Bus 142
   2) 480-Volt Bus 132X, and
   3) 480-Volt Bus 132Z.

Replace with attachment ①

c. 120-Volt A.C. Bus 111 energized from its associated inverter connected to D.C. Bus 111,*

d. 120-Volt A.C. Bus 113 energized from its associated inverter connected to D.C. Bus 111,*

e. 120-Volt A.C. Bus 112 energized from its associated inverter connected to D.C. Bus 112,* and

f. 120-Volt A.C. Bus 114 energized from its associated inverter connected to D.C. Bus 112.*

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

ACTION:

a. With one of the required divisions of A.C. ESF busses not fully energized, reenergize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. instrument ~~vital~~ bus not energized, reenergize the A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one A.C. inverter inoperable or not connected to its D.C. power supply, reenergize the A.C. vital bus from its associated inverter within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Replace with attachment ②

*Two inverters may be disconnected from their D.C. Bus for up to 24 hours as necessary, for the purpose of ~~performing~~ an equalizing charge on their associated battery bank provided: (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. bus.

Delete

Attachment (1)

C. 120-Volt AC Instrument Busses consisting of:

1. Instrument Bus 111 energized from its associated inverter connected to DC Bus 111.

2. Instrument Bus 112 energized from its associated inverter connected to DC Bus 112.

3. Instrument Bus 113 energized from its associated inverter connected to DC Bus 111.

4. Instrument Bus 114 energized from its associated inverter connected to DC Bus 112.

Attachment (2)

C. With a maximum of an A.C. inverter inoperable or not connected to its D.C. power supply, its associated A.C. Instrument bus may be powered from its regulating transformer power supply for up to 72 hrs; 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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3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices given in Table 3.8-1 shall be OPERABLE.

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

ACTION:

With one or more of the above required containment penetration conductor overcurrent protective device(s) inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping the associated circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the circuit breaker to be tripped or the inoperable circuit breaker racked out, or removed, at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices given in Table 3.8-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1) By verifying that the ~~7~~ 6.9 kV circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers, and performing the following:

a) A CHANNEL CALIBRATION of the associated protective relays,

b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed and as specified in Table 3.8-1, and

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SURVEILLANCE REQUIREMENTS (Continued)

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

/ 125-Volt DC

2) By selecting and functionally testing a representative sample of at least 10% of each type of 480-volt circuit breaker. Circuit breakers selected for functional testing shall be selected on a rotating basis. The functional test shall consist of injecting a current input at the specified Setpoint to each selected circuit breaker and verifying that each circuit breaker functions as designed and the response time is less than or equal to the specified value. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested; and

3) By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a nondestructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these functional tests shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each 7 kV circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

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RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE ~~RATE~~

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose ~~rate~~ due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and

b. For Iodine-131 and 133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the dose ~~rate(s)~~ exceeding the above limits, immediately restore the release rate to within the above limit(s).

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

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose ~~rate~~ due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose ~~rate~~ due to Iodine-131 and 133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

ATTACHMENT K

(Section 3/4.9)

1) Section 3.9.1 (pg 3/4 9-1) Refueling Operations.

Deleted "*" and Note at bottom of page because note doesn't say anything that Mode 6 doesn't say.

"Filled portion" has been changed to "unisolated portions" for clarification.

2) Section 4.9.1.3 (pg 3/4 9-1) Refueling Operations.

In section 4.9.1.3 valves 1CV8455, 1CV8464A, 1PW046, and 2PW046 have been changed to 1CV8430, 1CV8441, 1CV8435, and 1CV8439 respectively. This was done to be consistent with Byron FSAR Amendment 41 Page 15.4-24.

3) Section 3.9.4 (pg 3/4 9-4) Refueling Operation

In section 3.9.4.c.2 "Be" was deleted and "Capable" was made to be the first word of sentence.

Section 3.9.4.a was clarified to distinguish between the personnel hatch and equipment hatch.

4) Section 3.9.9/4.9.9 (pg 3/4 9-10) Refueling Operations.

Changed all phrases stating "Containment Ventilation System" to "Containment Purge Isolation System" to reflect the way it is referred to in bases and title page.

Section 3.9.12 (pg 3/4 9-13) Refueling Operations

Section 3.9.12 has been rewritten and an asterisk added to clearly identify the conditions for exhaust ventilation operation.

In Action: a) the words "discharging through" were deleted and in its place the words "taking suction from" were inserted because the system operates that way.

(0433M)

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3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODE 6~~*~~.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. Removing or unbolting the reactor vessel head, and

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

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

4.9.1.3 Valves ~~1CV8455, 1CV8464A, 1PW046, and 2PW046~~ 1CV8430, 1CV8441, 1CV8435 and 1CV8439 shall be verified closed and secured in position by mechanical stops or by removal of air or electrical power at least once per 31 days.

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

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3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

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

a. The personnel hatch closed and held in place by a minimum of four bolts or removed, replace with

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

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

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

2) ~~Be~~ capable of being closed by an OPERABLE automatic containment purge isolation valve.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

The personnel hatch should have a minimum of one door closed at any one time; and the equipment hatch shall be in place and held by a minimum of four bolts; or the equipment hatch may be removed with the requirements of Specification 3.9.12 in effect.

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

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

3/4.9.12 FUEL HANDLING BUILDING EXHAUST VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent Fuel Handling Building Exhaust Ventilation Systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool or during MODE 6 when the equipment hatch is removed or held in place by less than four Bolts or the personnel hatch is open to the Fuel Building.

ACTION:

a. With one Fuel Handling Building Exhaust Ventilation System inoperable, fuel movement within the containment* or storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE Fuel Handling Building Exhaust Ventilation System is capable of being powered from an OPERABLE emergency power source and is in operation and ~~discharging through~~ taking suction from at least one train of HEPA filters and charcoal adsorbers.

b. With no Fuel Handling Building Exhaust Ventilation System OPERABLE, suspend all operations involving movement of fuel within the containment* or storage pool or crane operation with loads over the storage pool until at least one Fuel Handling Building Exhaust Ventilation System is restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel Handling Building Exhaust Ventilation Systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes;

b. At least once per 18 months, or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system, by:

* When the equipment hatch is removed or held closed by less than four bolts or when the personnel hatch is open to the Fuel Building.

ATTACHMENT L

(Section 3/4.10)

Circled items noted in this attachment have previously been submitted.

1) Section 3.10.2 (pg 3/4 10-2) Special Test Exceptions

- In section 3.10.2 and Action 3.10.2; add "[Fq (z)]" after "3.2.2" and "[RCS Flow Rate]" after "3.2.3".

- In Action 3.10.2 add "(Rod Height)" after "3.1.3.1", "(RIL)" after "3.1.3.5" and "3.1.3.6", "(ΔI)" after "3.2.1" and "QPTR" after "3.2.4".

- In 4.10.2.2.a & b, add "(Fxy)" after 4.2.2.2, "[Fq (z)]" after 4.2.2.3 and "[RCS Flow Rate]" after 4.2.3.2.

  The changes were made to be more informative in relation to what the specification number is referring to.

2) Section 3.10.4 (pg 3/4 10-4) Special Test Exceptions

- Add "Reactor Thermal power does not exceed 10% of rated thermal power." to 3.10.4.a.

- Add "For testing under no flow conditions. (10% thermal power)" to APPLICABILITY: in section 3.10.4

- Add "(10% thermal)" to sections 4.10.4.1 and 4.10.4.2.

  To be more informative about subject matter and to add clarification.

3) Section 3.10.5 (pg 3/4 10-5) Special Test Exceptions

Delete "full-length" in section 3.10.5. All rods are the same length at Byron Station.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

[Fq(z)] [RCS FLOW Rate]

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2, below.

APPLICABILITY: MODE 1.

ACTION:

[Fq(z)] [RCS Flow Rate]

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 (QPTR) are suspended, either: (Rod Height) (RIL) (RIL) (ΔI)

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

[Fq(z)] [RCS Flow Rate]

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specfications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.2 and 4.2.2.3, and [FQ(z)]

(Fxy)

b. Specification 4.2.3.2.

[RCS Flow Rate]

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3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

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

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and Reactor Thermal power does not exceed 10% of rated thermal power.

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

APPLICABILITY: During operation below the P-7 Interlock Setpoint for testing under no flow conditions. (10% thermal power)

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during ~~STARTUP and~~ PHYSICS TESTS. (10% Thermal)

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating ~~STARTUP and~~ PHYSICS TESTS. (10% Thermal)

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3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual ~~full-length~~ shutdown and control rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The digital rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication System inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Position Indication "in" Systems shall be determined to be OPERABLE within 24 hours prior to ~~the~~ start ~~of~~ and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

*This requirement is not applicable during the initial calibration of the Position Indication System provided: (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

ATTACHMENT M

(Section 3/4.11)

Circled items noted in this attachment have been previously submitted.

1) Section 3.11.1.1 (pg 3/4 11-1) Radioactive Effluents

A second Action Statement has been added stating Specifications 3.0.3 and 3.0.4 are not applicable. The cause of the radioactive material released to unrestricted areas should limit unit operation.

2) Section 3.11.2.1 (pg 3/4 11-9) Radioactive Effluents

A second Action Statement has been added stating Specifications 3.0.3 and 3.0.4 are not applicable. The cause of the radioactive material released to unrestricted areas should limit unit operation.

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3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2, for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microCurie/ml total activity.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

b. The provisions of specifications 3.0.3 and 3.0.4 are not applicable. x

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analysis shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE ~~RATE~~

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose ~~rate~~ due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and

b. For Iodine-131 and 133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the doses ~~rate(s)~~ exceeding the above limits, immediately restore the release rate to within the above limit(s).

b. The provisions of specifications 3.0.3 and 3.0.4 are not applicable. x

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose ~~rate~~ due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose ~~rate~~ due to Iodine-131 and 133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

ATTACHMENT N

(Section 3/4.12)

10 Table 4.12-1 (Page 3/4 12-11) Table Notations

For clarification, "(s^{-1})" has been changed to "(sec^{-1})" and "(s)" has been changed to "(sec)".

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TABLE 4.12-1 (Continued)

TABLE NOTATIONS

(1) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.7.1.6.

(2) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

(3) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yeild a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (picoCuries per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22 = the number of disintegrations per minute per picoCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide, ($\cancel{s^{-1}}$), and

Sec^{-1}

Δt = the elapsed time between sample collection, or end of the sample collection period, and time of counting ($\cancel{s}$). sec

Typical values of E, V, Y, and Δt should be used in the calculation.

ATTACHMENT O

(Section Bases)

1) Bases 3/4.1.2 (pg B 3/4 1-3) Boration Systems

Add attachment "BB" which was taken from previous Refueling Water Storage Tank bases section which has been deleted.

2) Bases 3/4.5.4 (pg B 3/4 5-2) Refueling Water Storage Tank

This is being deleted and information added into the Boration System Bases on pg B 3/4 1-3.

3) Bases 3/4.7.6 (pg B 3/4 7-4) Plant System.

Changed all phrases "Control Room Emergency Air Cleanup" to "Control Room Ventilation Cleanup" to give proper identification of system.

4) Bases 3/4.9.9 (pg B 3/4 9-2) Refueling Operations

The word "ventilation" was deleted and the words "Purge Isolation" was added to give proper description of System.

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BORATION SYSTEMS (Continued)

With the RCS temperature below 200°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 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% $\Delta k/k$ 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.

Attachment "BB"

Insert to pg B 3/4 1-3

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.

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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 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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ULTIMATE HEAT SINK (Continued)

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

3/4.7.6 CONTROL ROOM ~~EMERGENCY AIR CLEANUP~~ Ventilation SYSTEM

The OPERABILITY of the Control Room ~~Emergency Air Cleanup~~ Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.7.7 NON-ACCESSIBLE AREA EXHAUST FILTER PLENUM VENTILATION SYSTEM

The OPERABILITY of the Non-Accessible Area Exhaust Filter Plenum Ventilation System ensures that radioactive materials leaking from the ECCS equipment within the pump rooms following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the

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3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine and auxiliary hoist ensure that: (1) manipulator cranes will be used for movement of drive rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a drive rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE FACILITY

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool areas ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of RHR capability. With the reactor vessel head removed and at least 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT ~~VENTILATION~~ Purge Isolation SYSTEM

The OPERABILITY of this system ensures that the containment purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

ATTACHMENT P

(Section 5.0)

1) Section 5.6.2 (pg 5-5) Fuel Storage

Deleted "This is 1 foot . . . . 423 feet 10 inches." on section 5.6.2. The normal pool water level will be between 423' 6" (low alarm) and 424' 1 1/2" (high alarm). We cannot specify exact normal level at this time.

(0433M)

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5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

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

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

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

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 423 feet 2 inches. ~~This is 1 foot 8 inches below the normal pool water level of 423 feet 10 inches.~~

CAPACITY

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

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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

ATTACHMENT Q

(Index)

1) (pg IX)

Section 3/4.5.4 has been deleted from text.

2) (pg X)

Section 3/4.7.6 has been changed from "Control Room Emergency Air Cleanup System" to "Control Room Ventilation System" to give proper name to the system.

3) (pg XII)

Section 3/4.9.9 has been changed from "Containment Ventilation System" to "Containment Purge Isolation System" to give proper name to the system.

4) (pg XVI)

Section 3/4.5.4 has been deleted from text.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

ATTACHMENT R

(Index)

1) Index XI, Limiting Conditions for Operation and Surveillance Requirements under <u>Section</u>; 3/4.7.12 Area Temperature Monitoring... 3/4 7-39 has been deleted to be consistent with the detection of pg 3/4 7-39, 3/4 7-40 and the Bases 3/4.7.12, also "Table 3.7-6...3.4 7-40" has been deleted for same.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

ATTACHMENT S

(Section 3/4.3)

Circled items noted in this attachment have been previously submitted.

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

   Channel Calibration requirement #5 for Power Range High Setpoint and Intermediate Range is deleted because there is no Plateau Curve for these detectors, only for Source Range Detectors.

2) Table 4.3-1 (page 3/4 3-11) Reactor Trip System Instrumentation Surveillance Requirement

   Functional Unit 20 Reactor Trip breaker Trip Actuating Device Operational Test "M(7, 11)" has been changed to "R(11)" because "11" is in conflict with "7" as it requires the Trip Actuating Device Operational Test (Independent verification of the Shunt and Undervoltage Trip) each refueling outage. The SER (attached) also requires the surveillance each refueling outage.

   A system modification to allow easy testing of the Shunt and Undervoltage trips is scheduled for completion the first refueling outage.

3) Table 4.3-1 (page 3/4 3-12) Table Notations

   The sentence "For the Intermediate Range and Power Range Neutron Flux Channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1" has been deleted because it no longer applies to Power Range and Intermediate Range.

4) Table 3.3-3 (Page 3/4 3-18) Engineered Safety Features Actuation System Instrumentation

   The suggested change to "2" from "3" is necessary in order to be consistent with auxiliary Feedwater System at the station. 1PSL-AF051 is suction transfer over for pump 1AF01PA, and 1PSL-AF055 is suction transfer over for pump 1AP01PB.

5) Table 3.3-3 (Page 3/4 3-19) Engineered Safety Features Actuation System Instrumentation

   We have 4 channels per steam generator. Therefore "3" has been changed to "4" for Steam Generator Water Level P-14 (High-High) channels.

6) Table 3.3-4 (Page 3/4 3-24) Engineered Safety Features Actuation System Instrumentation

   Nuclear Lead-Lag (NLL) card in the 7300 series process protection system.

   The PLS calls out a setpoint of -100 psig on the bistables with a time constant of 50 seconds on the. Therefore, "/s" has been deleted from 4.e Trip Setpoint and Allowable Value to reflect the PLS data.

   Also, Trip Setpoint for 5.6 Steam Generator Water Level - High-High (P-14) has been changed to 82.4 from 82 to reflect the PLS data.

7) Table 3.3-4 (Page 3/4 3-25) Engineered Safety Features Actuation System Instrumentation Trip Setpoints

The suggested change to "40.8" and "36" from "41" and "40" is necessary in order to be consistent with PLS data.

8) Table 3.3-5 (Page 3/4 3-29) Engineered Safety Features Actuation System Instrumentation Trip Setpoints

Under Containment Pressure - High - 1 a) $12^{(2)}$ has been changed to $12^{(5)}$ and Under Pressurizer Pressure-Low a) $12^{(2)}$ has been changed to $12^{(5)}$. This is to provide Consistency with the FSAR.

9) Table 3.3-5 (Page 3/4 3-30) Engineered Safety Features Actuation System Instrumentation Trip Setpoints

#5a., 57 has been changed to 55. #5b., $75^{(1)}/65^{(2)}$ has been changed to $65^{(1)}/75^{(2)}$ to be consistent with FSAR 6.2-58 #8, 7 has been changed to 5 to be consistent with the FSAR #9, $25^{(2)}$ has been changed to $250^{(2)}$ to correct a typographical error.

10) Table 3.3-5 (Page 3/4 3-31) Engineered Safety Features Actuation System Instrumentation Trip Setpoints

#13, 7 has been changed to 5 to be consistent with FSAR Table 6.2-58

11) Table 3.3-5 (Page 3-32) Table Notation

#6 has been deleted because it is not referred to in Tables 3.3-5.

12) Tables 3.3-6 and 4.3-3 (page 3/4 3-39, 41)

Delete Functional Unit number 3, "Criticality Radiation Level (ORE-AR037/38)" and associated line "1; 2; *; $\leq$ 15m R/h; 28" on table 3.3-6.

Delete Functional Unit number 3, "Criticality - High Radiation Level (ORE-AR037/38)" and associated line "S; R; M; *" on table 4.3-3.

Renumber "4" to "3", "5" to "4", and "6" to "5" in table 3.3-6 and 4.3-3 to indicate proper sequencing after the deletion of the old number 3.

Although the criticality radiation monitor is required in 10 CFR 70.24, Reg. Guide 8.12 Section C.1 allows for an exemption from 10 CFR 70.24 since Commonwealth Edison has determined a potential for criticality cannot exist. The FSAR shows that K_{eff} in the new fuel vault, spent fuel pool and core during fuel movements will be less than the maximum K_{eff} allowed by ANSI 18.2-1973 and ANSI 210-1976.

13) Surveillance Requirement 4.3.3.3.1 (page 3/4 3-43)

Replace the existing description with attachment "CC".

14) Surveillance Requirement 4.3.3.2 (page 3/4 3-43)

The first sentence of Surveillance Requiremen 4.3.3.2 should be rewritten to read: "Upon actuation of the seismic monitoring instruments, the equipment listed in Table 3.3-7 shall be restored to OPERABLE status within 24 hours following the seismic event.

15) Table 3.3-7 (page 3/4 3-44)

Delete all of item 3; Triaxial Seismic Trigger (PC Board).

Renumber item 4, Response-Spectrum Analyzer, to number "3" and renumber item 5, Triaxial Acceleration Sensors to number "4".

16) Table 4.3-4 (page 3/4 3-45)

Delete all of item 3; Triaxial Seismic Trigger (PC Board).

Renumber item 4, Response - Spectrum Analyzer, to number "3" and replace the "N.A." under "CHANNEL CHECK" with "Q".

Change the title of column "CHANNEL CALIBRATION" to "DIGITAL CHANNEL OPERATIONAL TEST".

Renumber item 5, Triaxial Acceleration Sensors to number "4" and:

1) Replace "SA" for 4a-f with "W",

2) Replace "N.A." for 4a-f with "SA",

3) Replace "Q" for 4a-f with "N.A."

Replace "Q" for items 1.a and 1.b with "R".

Replace "N.A." under "ANALOG CHANNEL OPERATIONAL TEST" for 2.a, b and c with "R".

The original STS system writeup did not adequately describe the as installed system at Byron. Changes in the surveillance requirements were made to conform with vendor manual recommendations.

The proposed change to FSAR Section 3.7.4 is attached for information.

18) Table 3.3-9 (page 3/4 3-50) Remote Shutdown Monitoring Instrumentation

Pressurizer pressure Total No. of channels has been changed to "1" from "2" because there is only one channel or loop (1P-0455) feeding to the Remote Shutdown Panel.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
19. Reactor Trip System Interlocks (Continued)						
d. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	M (8)	N.A.	N.A.	1, 2
e. Turbine Impulse Chamber Pressure, P-13	N.A.	R	M (8)	N.A.	N.A.	1, 2
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	R ~~M~~ (7, 11)	N.A.	1, 2, 3*, 4*, 5* X
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*

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The analog process channel testing is performed by introducing dummy input signals into the instrumentation channels and observing the tripping of the appropriate output bistables. The power range nuclear channels are tested by superimposing a test signal on the actual detector signal. To test the logic matrices of the solid-state logic protection system, pulse test signals are used in all possible trip and nontrip logic combinations. The test pulses are of short duration, and the trip logic is not maintained sufficiently long to permit opening of the reactor trip breakers. During logic testing of one train, the other train can initiate any required protective action. To test the reactor trip breakers, bypass breakers are provided. After a bypass breaker is closed, the associated reactor trip breaker can be tripped with a signal from the corresponding logic train.

In addition to providing inputs to the solid-state logic protection system, analog signals of the protection channels are used for nonprotective functions, such as control, remote indication, and computer monitoring. To protect from faults in the nonsafety circuits affecting the protection system, isolation amplifiers are used. The isolation amplifiers are classified as part of the protection system.

7.2.2 Specific Findings

The concerns arising from the staff review of the reactor trip system and their resolution follow.

7.2.2.1 Testing the Reactor Trip Breakers and Manual Trip Switches

The reactor trip breakers are provided with undervoltage and shunt trip coils. Interrupting power to the undervoltage coil or energizing the shunt coil will trip the breaker. The undervoltage coils receive trip signals from both the solid-state logic protection system and the manual trip switches (including the manual reactor trip switches and the safety injection switches). The shunt trip coils receive trip signals from the manual trip switches only. This provides diversity and enhances the separation between the automatic and manual reactor trip systems.

Testing of the undervoltage coil operation is carried out with a trip signal from the solid-state logic protection system. Testing of the manual reactor trip channel does not allow independent verification of the operability of the shunt coil and the undervoltage coil because the operation of a manual trip switch results in a simultaneous trip action by both coils. The staff will include in the station's Technical Specifications a requirement to periodically and independently verify the operability of the undervoltage and shunt trip functions at least once each refueling outage.

7.2.2.2 Protection System Sensors and Cabling in Nonseismic Structures

Protection system trip circuit inputs that are located in the nonseismic turbine building are (1) turbine stop valve closure limit switches, (2) turbine auto stop oil pressure switches, and (3) turbine impulse pressure transducers.

Items 1 and 2 above provide inputs to the reactor trip on turbine trip circuit; Item 3 provides inputs to the P-7 interlock. The reactor trip on turbine trip

TABLE 4.3-1 (Continued)

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TABLE NOTATIONS

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

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

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

(1) If not performed in previous 7 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore axial flux difference above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained, evaluated and compared to manufacturer's data. ~~For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.~~ X

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

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

(8) With power greater than or equal to the interlock Setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.

(9) Monthly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Monthly surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to an increase of twice the count rate within a 10-minute period.

(10) Setpoint verification is not applicable.

(11) At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.

(12) At least once per 18 months during shutdown verify that on a simulated Boron Dilution Doubling test signal ~~the normal~~ CVCS ~~discharge~~ valves ~~will~~ 112D and E close and ~~the centrifugal charging pump suction valves from the RWST will~~ OPEN AND 112B AND C open within 30 seconds.

(13) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater (Continued)					
g. Auxiliary Feed-water Pump Suction Pressure-Low (Transfer to Essential Service Water)	~~3~~ 2	2	2	1, 2, 3	15*
7. Automatic Opening of Containment Sump Suction Isolation Valves					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	21
b. RWST Level - Low-Low	4	2	3	1, 2, 3, 4	16
Coincident With Safety Injection	See Item 1. above for Safety Injection initiating functions and requirements.				
8. Loss of Power					
a. ESF Bus Undervoltage (Electromechanical Relaying)	2/Bus	2/Bus	1/Bus	1, 2, 3, 4	19*
b. Grid Degraded Voltage (Solid State Relaying)	2/Bus	2/Bus	1/Bus	1, 2, 3, 4	19*

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Reactor Trip, P-4	2	2	2	1, 2, 3	22
c. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	20
d. Steam Generator Water Level, P-14 (High-High)	4 ~~3~~/stm. gen.	2/stm. gen. in any operating stm. gen.	2/stm. gen. in each operating stm. gen.	1, 2, 3	20

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	5.0	0.71	1.5	≤ ~~10.0~~ 8.2 psig	≤ ~~12.0~~ 9.2 psig
d. Steam Line Pressure-Low (above P-11)	14.2	10.71	1.5	≥ 640 psig	≥ ~~610~~ 617 psig*
e. Steam Line Pressure-Negative Rate-High (Below P-11)	8.0	0.5	0	≤ -100 psi/s	≤ -110.0 psi/s**
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)	5.0	2.18	1.5	≤ ~~82%~~ 81.4 of narrow range instrument span	≤ ~~83%~~ 82.7 of narrow range instrument span

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level-Low-Low-Start Motor-Driven Pump and Diesel-Driven Pump	30.0	27.18	1.5	40.8 ≥ ~~41%~~ of narrow range instruement span	36 ≥ ~~40%~~ of narrow range instrument span
d. Undervoltage-RCP Bus-Start Motor Driven Pump and Diesel-Driven Pump	N.A.	N.A.	N.A.	4920 VOLTS ~~≥ 70% RCP* bus voltage* (4890 volts)*~~	4763 VOLTS ~~≥ 69% RCP* bus voltage* (4674 volts)*~~
e. Safety Injection-Start Motor-Driven Pump and Diesel-Driven Pump	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

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TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

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

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 22^{(4)}/12^{(5)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 7^{(3)}$
3) Phase "A" Isolation	$\leq 17^{(2)}/27^{(1)}$
4) Containment Vent Isolation	$\leq 25^{(1)}/10^{(2)}$
5) Auxiliary Feedwater	≤ 60
6) Essential Service Water	$\leq 47^{(1)}/32^{(2)}$
7) Containment Cooling Fans	$\leq 55^{(1)}/40^{(2)}$
8) Start Diesel Generator	≤ 10
b. Steam Line Isolation	≤ 5
5. Containment Pressure-High-3	
a. Containment Spray	55 $\leq \cancel{57}^{(1)}/45^{(2)}$
b. Phase "B" Isolation	$\leq \cancel{75}^{(1)}/\cancel{65}^{(2)}$ 65 75
6. Steam Generator Water Level-High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	$\leq \cancel{7}^{(3)}$ $7^{(3)}$
7. Steam Generator Water Level-Low-Low	
a. Motor-Driven Auxiliary Feedwater Pump	≤ 60
b. Diesel-Driven Auxiliary Feedwater Pumps	≤ 60
8. Containment Pressure-High-2	
Steam Line Isolation	$\leq \cancel{7}$ 5
9. RWST Level-Low-Low Coincident with Safety Injection	
Automatic Opening of Containment Sump Suction Isolation Valves	$\leq \cancel{25}^{(2)}/265^{(1)}$ 250

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. Undervoltage RCP Bus	
a. Motor-Driven Auxiliary Feedwater Pump	$\leq$ 60
b. Diesel-Driven Auxiliary Feedwater Pump	$\leq$ 11
11. Division 1 ESF Bus Undervoltage	
Motor-Driven Auxiliary Feedwater Pump	$\leq$ 60
12. Loss of Power	
a. ESF Bus Undervoltage	$\leq$ 10
b. Grid Degraded Voltage	$\leq$ 10
13. Steam Line Pressure - Negative Rate-High	
Steam Line Isolation	$\leq$ ~~7~~ 5
14. Phase "A" Isolation	
Containment Vent Isolation	$\leq$ 5
15. Auxiliary Feedwater Pump Suction Pressure-Low-Low	
Automatic Switchover to ESW	N.A.

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TABLE 3.3-5 (Continued)

TABLE NOTATIONS

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

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

(3) Air operated valves.

(4) Diesel generator starting and sequence loading delay included. RHR pumps not included.

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

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

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 mR/h	30
2. Containment-Fuel Handling (1RE-AR011/12)	1	2	#	≤ 2 mR/h	26
~~3. Criticality-Radiation Level (ORE-AR037/38)~~	~~1~~	~~2~~	~~*~~	~~≤ 15 mR/h~~	~~28~~
3 ~~4~~. Gaseous Radioactivity-RCS Leakage Detection (1RE-PR011A)	N.A.	1	1, 2, 3, 4	N.A.	29
4 ~~5~~. Particulate Radioactivity-RCS Leakage Detection (1RE-PR011B)	N.A.	1	1, 2, 3, 4	N.A.	29
5 ~~6~~. Main Control Room-Outside Air Intake (ORE-PR031/32 and ORE-PR033/34)	1	2 per intake	All	≤ 2 mR/h	27

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TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Fuel Building Fuel Handling (ORE-AR055/56)	S	R	M	**
2. Containment Fuel Handling (ORE-AR011/12)	S	R	M	#
~~3. Criticality High Radiation Level (ORE-AR037/38)~~	~~S~~	~~R~~	~~M~~	~~*~~
3 ~~4~~. Gaseous Radioactivity-RCS Leakage Detection (1RE-PR011A)	S	R	M	1, 2, 3, 4
4 ~~5~~. Particulate Radioactivity-RCS Leakage Detection (1RE-PR011B)	S	R	M	1, 2, 3, 4
5 ~~6~~. Main Control Room-Outside Air Intake (ORE-PR031/32 and ORE-PR033/34)	S	R	M	All

*With fuel in the fuel storage areas or fuel building.
**With irradiated fuel in the fuel storage pool.
#Must satisfy Specification 3.9.9 requirements.

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INSTRUMENTATION

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SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.7.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-4.

Insert "CC"

4.3.3.3.2 ~~Each of the above required seismic monitoring instruments actuated during a seismic event greater than or equal to 0.02 g~~ shall be restored to OPERABLE status within 24 hours ~~and a CHANNEL CALIBRATION performed within 10 days~~ following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.7.2 within 14 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

Upon actuation of the seismic monitoring instruments, the equipment listed in Table 3.3-7

Insert "CC"

"4.3.3.3.1 The seismic monitoring instrumentation shall be determined operable:

a. At least once per 31 days by verifying operable status indications of the seismic monitoring instrumentation.

b. At least once per 92 days by verifying that:

1) The active seismic sensors, the time-history recorder and the playback unit properly processes the equipments internal test signals.

2) The response spectrum analyzer properly executes its diagnostic routine.

c. At least once per 184 days by verifying that the active seismic sensors, the time-history recorder and the playback unit properly record the equipments internal test signals.

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

1) Verifying the electronic calibration of the time-history recorder and the playback unit.

2) Installing fresh magnetic recording plates in the peak recording accelerometers"

The above verification checks will ensure system operability.

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TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Time - History Accelerographs		
a. 12M-451', Aux. Elect. Rm, OPA02J	N.A.	1
b. 7A-703', Byron River Screen House	N.A.	1
2. Triaxial Peak Accelerographs		
a. Cont./Reactor Eq. Accumulators	-2 g to +2 g	1
b. Cont./Reactor piping	-2 g to +2 g	1
c. Aux. Bldg./Cat. I piping	-2 g to +2 g	1
~~3. Triaxial Seismic Trigger (PC Board)~~		
~~a. Aux. Elect. Rm, OPA02J~~	~~-0.02 g to +0.02 g~~	~~1*#~~
~~b. Byron River Screen House~~	~~-0.02 g to +0.02 g~~	~~1#~~
3 ~~4~~. Response-Spectrum Analyzer		
12M-451', Aux Elect Rm, OPA02J	None	1
4 ~~5~~. Triaxial Acceleration Sensors		
a. Cont./10W - 377'	-2 g to +2 g	1
b. Cont/10W - 502'	-2 g to +2 g	1
c. Cont./10X - 426'	-2 g to +2 g	1
d. Free Field/27 + 0.7N, 47 + 71E	-2 g to +2 g	1
e. Aux. Bldg./18N - 426'	-2 g to +2 g	1
f. Byron River Screen House	-2 g to +2 g	1

*With reactor control room indication.
#Is a component of Time-History Accelerograph.

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TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	DIGITAL CHANNEL OPERATIONAL TEST ~~CHANNEL CALIBRATION~~	ANALOG CHANNEL OPERATIONAL TEST
1. Time History Accelerographs			
a. 12M-451', Aux Elect Rm, OPA02J	SA	R	~~Q~~ R
b. 7A-703', RSH	SA	R	~~Q~~ R
2. Triaxial Peak Accelerographs			
a. Cont./Reactor Eq. Accumulators	N.A.	R	N.A. R
b. Cont./Reactor piping	N.A.	R	N.A. R
c. Aux. Bldg./Cat. I piping	N.A.	R	N.A. R
~~3. Triaxial Seismic Trigger (PC Board)~~	~~N.A.~~	~~N.A.~~	~~N.A.~~
3 ~~4~~. Response-Spectrum Analyzer			
12M-451; Aux Elect Rm, OPA02J	~~N.A.~~ Q	N.A.	Q
4 ~~5~~. Triaxial Acceleration Sensors			
a. Cont./10W - 377'	SA Q	N.A. SA	Q N.A.
b. Cont./10W - 502'	SA Q	N.A. SA	Q N.A.
c. Cont./10X - 426'	SA Q	N.A. SA	Q N.A.
d. Free Field/27 + 07N, 47 + 71E	SA Q	N.A. SA	Q N.A.
e. Aux. Bldg./18N -426'	SA Q	N.A. SA	Q N.A.
f. Byron River Screen House	SA Q	N.A. SA	Q N.A.

value is determined by testing programs such as was done for the reactor coolant loop (Reference 1).

3.7.4 Seismic Instrumentation

3.7.4.1 Instrumentation for Earthquakes

Seismic instrumentation is necessary to determine promptly the seismic response of nuclear power plant features important to permit comparison of such response with that used as the design basis.

The seismic instrumentation for Byron/Braidwood utilizes two types of sensor-recorders with a playback capability available in the control room area. The location and function of these seismic devices were selected to provide adequately for the determination of seismic event loads into the structures via computerized analysis programs. System surveillance is maintained during the loss of AC power by actuation of a local alarm. This system also has a backup DC power supply.

3.7.4.2 Location and Description of Instrumentation

3.7.4.2.1 Playback Unit

A primary playback unit with strip chart recorder, indicator lights, and playback system is provided in the control area. The lights indicate whether the system is ~~triggered~~ actuated and whether the operating basis or safe shutdown maximum accelerations are exceeded in any one of the three orthogonal directions in the basement of the containment structure. These directions coincide with the major axes of the analytical model used in the seismic analysis of the plant structure.

3.7.4.2.2 Time-History Accelerograph

Insert

a. The time history accelerograph located in the Auxiliary Electric Room receives inputs from four triaxial ~~accelerometers,~~ sensors each of which measures the absolute acceleration as a function of time in three orthogonal directions; these directions coincide with the major axes of the analytical model of the structure. These ~~accelerometers~~ sensors are placed at the following locations:

1. in the free field at site coordinates 41+00E, 27+00N and 39+00E, 41+00S for Byron and Braidwood Stations, respectively.

2. on the containment building foundation slab at elevation 377 feet and azimuth 145 degrees,

3. on the containment shell wall at elevation 502 feet and azimuth 145 degrees, and

4. on the containment refueling floor at elevation 426 feet.

INSERT "B"

3.7.4.2.3 Peak Accelerographs

A triaxial peak ~~recorder~~ accelerograph which measures the absolute peak acceleration in three orthogonal directions coinciding with the major axes of the analytical model is provided at each of the following locations:

a. on the accumulator tank located at elevation 426 feet in the containment building;

b. on the safety injection piping at elevation 421 feet in the containment building;

c. on the essential service return piping at elevation 346 feet in the auxiliary building.

3.7.4.2.4 Response Spectrum Analyzer

Insert "C"

~~One centralized triaxial response spectrum analyzer is provided~~ INPUTS for calculating the peak acceleration vs. frequency ~~as~~ ARE measured at the two locations listed below:

a. on the base slab of the containment building, elevation 377 feet. This location serves the dual purpose of monitoring the base slab response and the support motion of reactor equipment recorded from the accelerometer through the time-history accelerograph. This recorder is also equipped to provide signals in the control room area in the event preset acceleration limits are exceeded (see Subsection 3.7.6.3).

b. on the floor, elevation 426 feet, in the counting room in the auxiliary building (recorded directly from the accelerometer).

One time-history accelerograph is provided at the foundation level of the river screen house (Byron only). Using the playback unit, the response spectrum can be determined for this location.

These locations are chosen to allow meaningful correlation between the recorded accelerations and those calculated using the analytical model of the structure.

3.7.4.3 Control Room Operator Notification

The centrally located seismic indicating and recording equipment near the main control room is the source of operator information concerning the acknowledgement of an earthquake. An acceleration of 0.02g in any direction activates the seismic switch which turns on the seismic monitors and lights up the seismic alarm lights at the ~~central station~~ SEISMIC WARNING panel.

Using the spectrum analyzer, an operator can observe the triaxial spectral analysis from the containment building slab monitor and the auxiliary building 426 feet elevation monitor. (The latter ~~accelerometer~~ SENSOR is direct-connected to the spectrum analyzer, while the former ~~accelerometer~~ SENSOR is connected to the analyzer through the time-history accelerograph.)

The operator can also use the playback feature of the system to obtain time history and spectral analysis of any of the ~~four accelerometers~~ RECORDED SENSORS listed in Subsection 3.7.4.2.~~1(b)~~2.

Observed values which exceed the OBE acceleration threshold values stored in the spectrum analyzer are marked during the printout, and an alarm light is energized. Further analysis is needed to authenticate structural loads and to evaluate observations via the structural response-seismic model. An observation which exceeds the SSE acceleration threshold is validated in a similar manner with the structural response-seismic model. When evaluated accelerations exceed SSE threshold values, the reactor is shut down. The alarm lights and the recorder data are available simultaneously with the seismic event from the Containment Building slab monitor and the Auxiliary Building 426-foot elevation monitor. Data from all other locations is available after the seismic event, using the playback feature.

3.7.4.4 Comparison of Measured and Predicted Responses

The computer program which evaluates the time-history data computes the maximum response accelerations at various points of the model. The observed response spectra can be compared with the response spectra. Agreement between the observed response spectra and the computed response spectra from the time-history inputs demonstrates the adequacy of the analytical model. The magnitude of actual forces at various structural locations can then be compared to design values to authenticate the capability of the plant to continue operation without undue risk to the health and safety of the public.

Insert "A"

3.7.4.2.2 Time-History Accelerographs

The time-history accelerograph is the master control unit for all control timing signals and system data interface. It continually monitors and has the ability to digitize voltage signals and record on magnetic tape both pre and post seismic event data. Integral to the accelerograph are two triaxial seismic triggers (PC Boards) which provide a signal to actuate a local alarm on the seismic warning panel during a seismic event. These trigger inputs are continually monitored by the accelerograph. There are two time-history accelerographs utilized as part of the seismic instrumentation.

Insert "B"

b. The other time-history accelerograph and sensor is provided at the foundation level of the river screen house (Byron only). The response spectrum can be determined for this location using the playback unit.

Insert "C"

This unit determines the variation in the maximum response of a single degree-of-freedom system versus its natural frequency vibration when subjected to a time-history motion of its base. The response spectrum analyzer computes the response spectrum of the event, compares it to the design response spectra of the plant, and indicates whether the event exceeded the OBE or SSE criteria.

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Intermediate Range Neutron Flux	1PL06J	2	1
2. Source Range Neutron Flux	1PL06J	2	1
3. Reactor Coolant Temperature - Wide Range			
a. Hot Leg	1PL05J	1/loop	1/loop
b. Cold Leg	1PL05J	1/loop	1/loop
4. Pressurizer Pressure	1PL06J	2	1
5. Pressurizer Level	1PL06J	2	1
6. Steam Generator Pressure	1PL04J/1PL05J	1/stm gen	1/stm gen
7. Steam Generator Level	1PL04J	1/stm gen	1/stm gen
8. RHR Flow Rate	LOCAL	2	1
9. RHR Temperature	LOCAL	2	1
10. Auxiliary Feedwater Flow Rate	1PL04J/1PL05J	1/stm gen	1/stm gen

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ATTACHMENT T

(Section 3/4.4)

Table 3.4-1 (page 3/4 4-21) Reactor Coolant System Pressure Isolation Valves

Deletion of valve number "1SI8900A, B, C, D, 1SI8815" and function "CHG/SI Check Valve, CHG/SI Backup Check Valve" is requested because, RCS leakage can not propagate through these check valves for two reasons. First, the motor operated valves, 1SI8801A and B are normally closed and are only open during a safety injection. Second, the Chemical and Volume Control System is always operating in the normal charging mode which is at a higher pressure than the RCS.

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TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
~~1SI8900A,B,C,D~~	~~CHG/SI Check Valve~~
~~1SI8815~~	~~CHG/SI Backup Check Valve~~
1SI8948A,B,C,D	Accumulator Check Valve
1SI8956A,B,C,D	Accumulator Backup Check Valve
1SI8818A,B,C,D	RHR Cold Leg Check Valve
1SI8819A,B,C,D	SI Cold Leg Check Valve
1SI8949A,B,C,D	SI Hot Leg Check Valve
1SI8905A,B,C,D	SI Hot Leg Backup Check Valve
1SI8841A,B	RHR Hot Leg Check Valve

ATTACHMENT U

(Section 3/4.7)

1) Section 3.7.12 (pg 3/4 7-39) Plant System - Area Temperature Monitoring

   Page 3/4 7-39 has been deleted because Byron Station has no equipment installed.

2) Table 3.7-6 (pg 3/4 7-40) Plant System - Area Temperature Monitoring

   Page 3/4 7-40 has been deleted because Byron Station has no equipment installed.

   Safety-related equipment at Byron Station will not be subjected to temperatures in excess of their environmental qualification test temperatures. A program has been undertaken at Commonwealth Edison to ensure that installed equipment at the plant site will be replaced prior to its end-of-life condition as defined by the results of equipment qualification test program. Temperatures utilized during equipment qualification testing were conservative with respect to Byron site.

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3/4.7.12 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.12 The temperature of each area shown in Table 3.7-6 shall be maintained within the limits indicated in Table 3.7-6.

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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TABLE 3.7-6

AREA TEMPERATURE MONITORING

	AREA	TEMPERATURE LIMIT (°F)
1.	Misc. Elec. Equip. and Battery Rooms, ESF Switchgear Rms., Div. 12 Cable Spreading	108
2.	Upper and Lower Cable Spreading Rooms	90
3.	Diesel-Generator Rooms, Diesel Oil Storage Rooms	132
4.	Aux. Bldg. Drain Tank Cubicles, Recycle Holdup Tank Cubicles, Gas Decay Tank Cubicle, Gas Decay Pipe Tunnel, Recycle Evap. Pipe Tunnel, Floor Drain Tank Cubicle, Recycle Holdup Pipe Tunnel, Filter Pipe Tunnels, Demineralizer Cubicles, Positive Displacement Pump Rooms, Spent Fuel Pit Pump Room, Aux. Bldg. Vent Exhaust Filter Cubicle Process Filter Cubicles, Centrifugal Charging Pump Rooms	122
5.	Containment Spray Pump Rooms, RHR Pump Rooms, Safety Injection Pump Rooms, Pentration Areas El. 346, 364	130
6.	Control Room	90
7.	Radwaste Evaporator Cubicles	118
8.	Boric Acid Tank Cubicles	127

deleted intentionally

ATTACHMENT V

(Bases 3/4.7.12)

1) Bases 3/4.7.12 (pg B 3/4 7-8) Plant System - Area Temperature Monitoring Page B 3/4 7-8 has been deleted because Byron Station has no equipment installed. Safety-related equipment at Byron Station will not be subject to temperatures in excess of their environmental qualification test temperatures. A program has been undertaken at Commonwealth Edison to ensure that installed equipment at the plant site will be replaced prior to its end-of-life condition as defined by the results of equipment qualification test program. Temperatures utilized during equipment qualification testing were conservative with respect to Byron site.

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3/4.7.12 AREA TEMPERATURE MONITORING

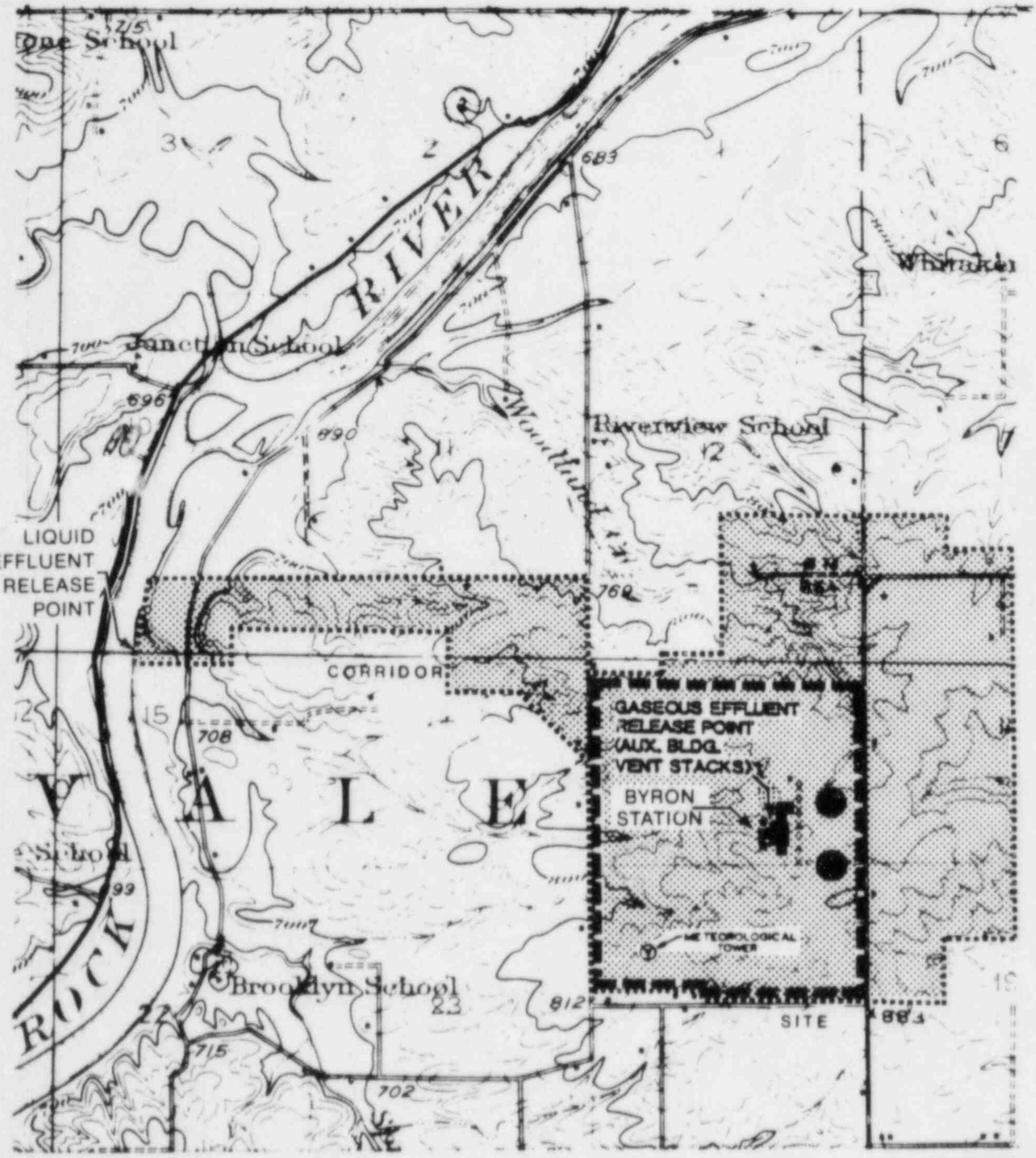
The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY.

ATTACHMENT W

(Section 5.0)

1) Figure 5.1-1 (page 5-2) Exclusion Area and Unrestricted Area For Radioactive Gaseous and Liquid Effluents.

   This figure replaced old Figure 5.1-1 which did not have meteorological tower on the map.



BYRON STATION

FIGURE 5.1-1

EXCLUSION AREA AND UNRESTRICTED AREA FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

ATTACHMENT X

(Section 6.0)

1) Section 6.3.2 (pg 6-11) Onsite.

   Specification 6.2 does not specifically require programs and procedure; however; Section 6.0 does outline required procedures and programs and should be the section referenced.

2) Administrative Control (pg 6-18 - 24)

   These changes were recommended by Commonwealth Edison Technical Services Nuclear Dept. They are consistent with practice for other CECo Nuclear station and are consistent with the changes already made in Section 3/4-12.

(0438M)

OFFSITE (Continued)

h) Instrumentation and Control

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

i) Metallurgy

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

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

ONSITE

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

a. Onsite Review and Investigative Function

The Station Superintendent shall: (1) provide directions for the Review and Investigative Function and appoint the Technical Staff Supervisor, or other comparably qualified individual as the senior participant to provide appropriate directions; (2) approve participants for this function; (3) assure that at least two participants who collectively possess background and qualifications in the subject matter under review are selected to provide comprehensive interdisciplinary review coverage under this function; (4) independently review and approve the findings and recommendations developed by personnel performing the Review and Investigative Function; (5) report all findings of noncompliance with NRC requirements, and provide recommendations to the Division Vice President and General Manager - Nuclear Stations and the Supervisor of the Offsite Review and Investigative Function; and (6) submit to the Offsite Review and Investigative Function for concurrence in a timely manner, those items described in Specification 6.1.G.1.a which have been approved by the Onsite Review and Investigative Function.

b. Responsibility

The responsibilities of the personnel performing this function are:

1) Review of: (1) procedures required by Specification ~~6.2~~ 6.6.1 and 6.6.4 changes thereto, (2) all programs required by Specification ~~6.2~~ and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Plant Superintendent to affect nuclear safety;

2) Review of all proposed tests and experiments that affect nuclear safety;

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REPORTING REQUIREMENTS (Continued)

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.7.1.6 Routine ~~Annual Radiological Environmental Operating~~ Annual Radiological Environmental Operating Reports covering the operation of the Unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activites for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the Land Use Census required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the Radiological Environmental Monitoring Program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program and the corrective actions being taken if the specified program is not being performed as required by Specification 3.12.3; reasons for not conducting the Radiological Environmental Monitoring Program as required by Specification 3.12.1, and discussion of all deviations from the sampling schedule of Table 3.12-1; discussion of environmental sample measurements that exceed the reporting levels of Table 3.12-2 but are not the result of plant effluents, pursuant to ACTION b. of Specification 3.12.1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

Insert "EE"

---

*A single submittal may be made for a multiple unit station.

**One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

*** In lieu of submission with the ~~Semiannual Radioactive Effluent Release~~ Annual Radiological Environmental Operating Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

Insert "EE"

The ~~Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year~~ annual Radiological Environmental Operating Report shall also include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.*** This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the Unit or Station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figures 5.1-3 and 5.1-4) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the ODCM.

The ~~Semiannual Radioactive Effluent Release~~ annual Radiological Environmental Operating Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation." Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The ~~Semiannual Radioactive Effluent Release~~ annual Radiological Environmental Operating Reports shall include any changes made during the reporting period to the PCP ~~and to the ODCM~~, pursuant to Specifications 6.11 and 6.12 respectively, as well as any major changes to Liquid, Gaseous or Solid Radwaste Treatment Systems, pursuant to Specification 6.13. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

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REPORTING REQUIREMENTS (Continued)

Annual Radiological Environmental Operating Report

~~SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT~~*

6.7.1.7 Routine Annual Radiological Environmental Operating ~~Semiannual Radioactive Effluent Release~~ Reports covering the operation of the Unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Annual Radiological Environmental Operating ~~Semiannual Radioactive Effluent Release~~ Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the Unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof. For solid wastes, the format for Table 3 in Appendix B shall be supplemented with three additional categories: class of solid wastes (as defined by 10 CFR Part 61), type of container (e.g., LSA, Type A, Type B, Large Quantity), and SOLIDIFICATION agent or absorbent (e.g., cement, urea formaldehyde).

~~The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.** This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the Unit or Station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figures 5.1-3 and 5.1-4) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the ODCM.~~

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*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate Radwaste Systems, the submittal shall specify the releases of radioactive material from each unit.

~~**In lieu of submission with the Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.~~

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REPORTING REQUIREMENTS (Continued)

~~The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation." Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.~~

Annual Radiological Environmental Operating

The ~~Semiannual Radioactive Effluent Release~~ Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

~~The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PCP and to the ODCM, pursuant to Specifications 6.11 and 6.12 respectively, as well as any major changes to Liquid, Gaseous or Solid Radwaste Treatment Systems, pursuant to Specification 6.13. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.~~

Annual Radiological Environmental Operating Report

The ~~Semiannual Radioactive Effluent Release Reports~~ shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specifications 3.3.3.10 or 3.3.3.11, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORT

6.7.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

RADIAL PEAKING FACTOR LIMIT REPORT

6.7.1.9 The F_{xy} limits for Rated Thermal Power (F_{xy}^{RTP}) shall be provided to the NRC Regional Administrator with a copy to Director of Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555 for all core planes containing Bank "D" control rods and all unrodded core planes and the plot of predicted ($F_q^T \cdot P_{Rel}$) vs Axial Core Height with the limit envelope at least [illegible]0 days prior to each cycle initial criticality unless otherwise approved by the Commission by letter. In addition,

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HIGH RADIATION AREA (Continued)

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

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

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

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified ~~by the Rad/Chem Supervisor or Lead Health Physicist~~ in the RWP.

6.10.2 In addition to the requirements of Specification 6.10.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

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

and conspicuously posted.

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6.11 PROCESS CONTROL PROGRAM (PCP)

6.11.1 The PCP shall be approved by the Commission prior to implementation.

6.11.2 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Annual Radiological Environmental Operating Report ~~Semiannual Radioactive Effluent Release Report~~ for the period in which the change(s) was made. This submittal shall contain:

1) Sufficiently detailed information to totally support the rationale for the change wthout benefit of additional or supplemental information;

2) A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the Onsite Review and Investigative Function.

b. Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.

6.12 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.12.1 The ODCM shall be approved by the Commission prior to implementation.

6.12.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission as controlled revisions within ~~in the Semiannual Radioactive Effluent Release Report for the period in which change(s) was made effective~~. ~~This submittal shall contain:~~ 180 days of the change. Documentation supporting the change shall contain:

1) Sufficiently detailed information to totally support the rationale for the change. without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered, dated and containing the revision number, together with appropriate analyses or evaluations justifying the change(s);

Delete

2) A determination that the change will not reduce the accuracy or reliability of dose calculations or Setpoint determinations; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the Onsite Review and Investigative Function.

b. Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.

1) Bases 3/4.3.3.3 (page B 3/4 3-4)

   Replace Section 3/4.3.3.3 with Insert "JJ".

   The original STS system writeup did not adequately describe the as installed system at Byron. Changes in the bases fully describes the system.

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MOVABLE INCORE DETECTORS (Continued)

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

Insert "JJ"

~~The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.~~

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG 0737, "Clarification of TMI Action Plan Requirements," November 1980.

Bases Insert "JJ" (1 of 2)

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100.

The instrumentation consists of two time-history response spectrum analyzer, a playback unit, three peak recording accelerometers, and six triaxial accelerometers. One time-history recorder and one sensor are located down at the River Screen House. The rest of the equipment, excluding the sensors, is located in the Auxiliary Electrical Room. The remaining sensors are located as follows: three in containment, one in the Auxiliary Building, and one at the free field location 27+07N, 47+71E. The peak recording accelerometers are passive devise which have no interplay on the rest of the system and are located on reactor equipment, reactor piping, and outside containment on Category I piping.

The triaxial accelerometer is based on three orthogonal force-balanced servo-accelerometers which generate a voltage signal upon stimulation. The voltage signals are transmitted to the time-history recorder in the Auxiliary Electrical Room, digitized, and recorded on magnetic tape.

The time-history recorder is the master control unit for all control timing signals and system data interface. It also contains the system triggers used to actuate the system. The master control unit continually monitors two of the sensor inputs, which are processed through the trigger circuits for comparison to the system actuation level. The time-history recorder also has the ability to record both pre and post seismic event data. The other key component in the system is the response spectrum analyzer. This unit determines the variation in the maximum response of a single degree-of-freedom system versus its natural frequency of vibration when either

-cont-

Insert "JJ" (2 of 2)

Bases (cont)

of two designated triaxial accelerometers is subjected to a time-history motion of the accelerometer.

The response spectrum analyzer computes the response spectrum of the event for two sensor locations, compares it to the design response spectra of the plant, and indicates whether the event exceeded the operating basis earthquake criteria or the safe shutdown earthquake criteria.

This instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquake," April 1974

ATTACHMENT Z

These changes delete all reference to three loop oper^tion. This mode of operation is not supported by existing safety analyses.

8376N

1) Index pg (III, X) Safety Limits and Limiting Safety System Settings "During Four Loop Operation" and "Four Loop In Operation" has been deleted from pg X and III respectively to get rid of all indications that Byron has more than one type of Loop Operations.

2) Section 2.1.1 (Page 2-1) Safety Limits

Delete from section 2.1.1 the "s" from "Figures" and "and 2.1-2..... respectively".

3) Figure 2.1-1 (pg 2-2)

Deleted "Four Loops in Operation". Only one type of Operation.

4) Figure 2.1-2 (Page 2-3)

Delete this page for Figure 2.1-2.

5) Section 2.1.1 (Page B2-1) Reactor Core

Delete from paragraph #4, the "s" from "Figures" and "and 2.1-2".

6) Section 3.1.3.4 (Page 3/4 1-19) Rod Drop Time

Delete Action 3.1.3.4.b.

7) Section 3.1.3.6 (Page 3/4 1-21) Control Rod Insertion Limits

Delete from LCO 3.1.3.6 "s" from "Figures" and "and 3.1-2".

8) Figure 3.1-1 (pg 3/4 1-22)

Delete "Four Loop Operation".

9) Figure 3.1-2 (Page 3/4 1-23)

Delete space for Figure 3.1-2.

10) Table 3.2-1 (Page 3/4 2-15) DNB Parameters

Delete "Three Loops in Operation". Delete "Four Loops in Operation".

Delete "**".

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

ATTACHMENT (Continued)

11) Table 3.3-1 (Page 3/4 3-2) Reactor Trip System Instrumentation

Delete from 7. Overtemperature Δ T "b. Three Loop Operation". "Four Loop Operation".

Delete from 8. Overpower Δ T "b. Three Loop Operation". "Four Loop Operation".

Delete the astrics "**" on these lines.

12) Table 3.3-1 (Page 3/4 3-5) Table Notations

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

13) Section 3.7.1.1 (Page 3/4 7-1) Safety Valves

Delete Action 3.7.1.1.7 The Word "four".

Delete Action 3.7.1.1.b.

Change Action 3.7.1.1.c to 3.7.1.1.b.

14) Table 3.7-1 (Page 3/4 7-2)

Delete from Table 3.7-1, the words "During Four Loop Operation", and all of Table 3.7-2.

Delete "II These values left blank pending NRC approval of three loop operation".

15) Section 3/4.7.1.1 (Page B 3/4 7-1) Bases - Safety Valves

Delete "For three loop operation:", and the word "four" from formula.

$$SP = \frac{(X) - (Y)(U)}{X} \times (*)$$

16) Section 3/4 7.1.1 (Page B 3/4 7-2) Bases-Safety Valves

Delete "*" and note with it.

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INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Reference to specific loops should be deleted

BASES

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

reference to specific loop should be deleted.

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

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

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.5.1.

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.5.1.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.5.1.

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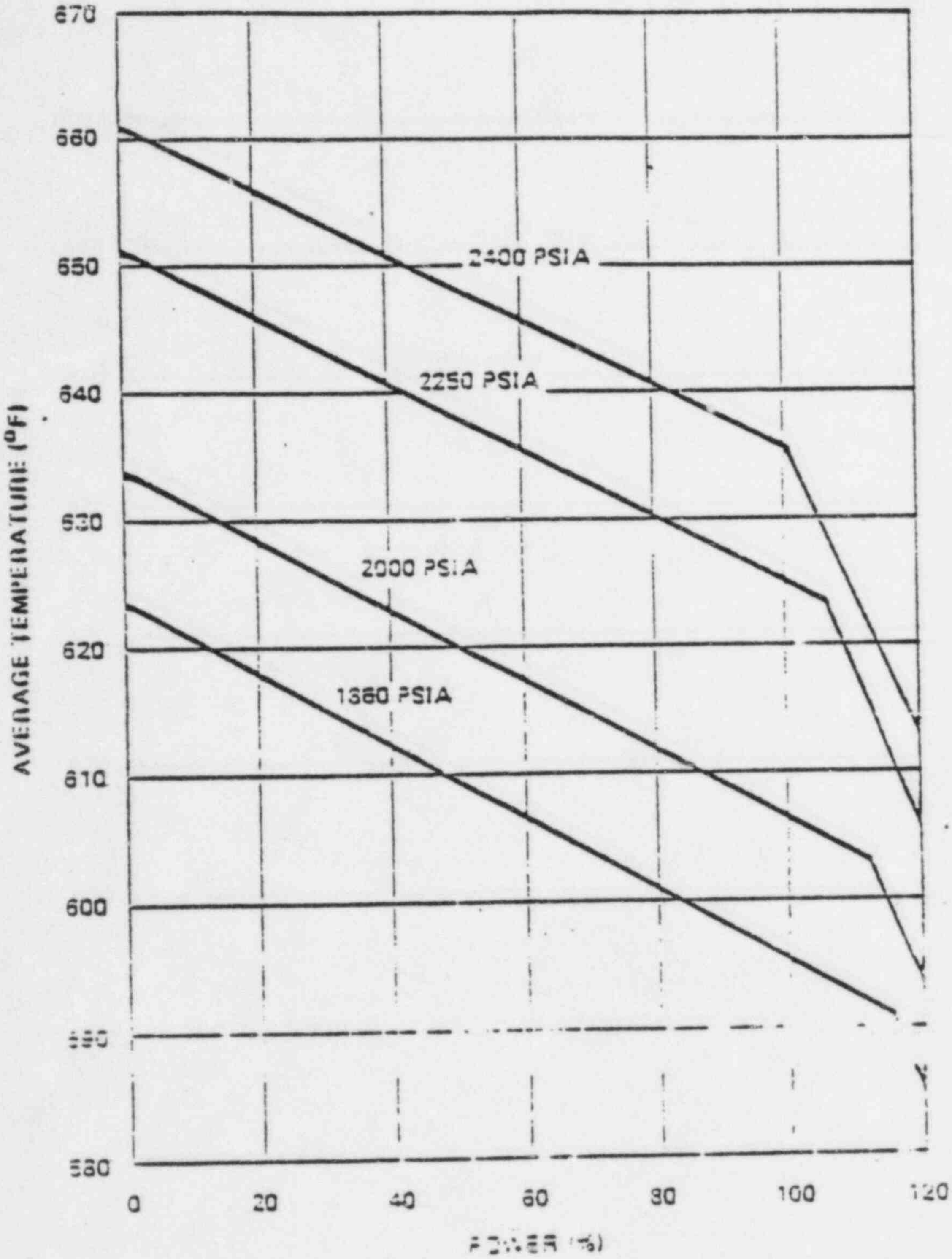

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - ~~FOUR LOOPS IN OPERATION~~

Reference to specific loop should be deleted

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~~Figure 2.1-2 left blank pending NRC approval of three-loop operation.~~

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2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the WRB-1 correlation. The WRB-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The minimum value of the DNBR during steady-state operation, normal operational transients, and anticipated transients is limited to 1.34 for a typical cell and 1.32 for a thimble cell. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve~~s~~ of Figure~~s~~ 2.1-1 ~~and 2.1-2~~ shows the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.34 for a typical cell and 1.32 for a thimble cell, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1 + 0.3\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power

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ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length shutdown and control rod drop time from the fully withdrawn position shall be less than or equal to 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 550°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the rod drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop time within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

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CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*#.

ACTION:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exceptions 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.

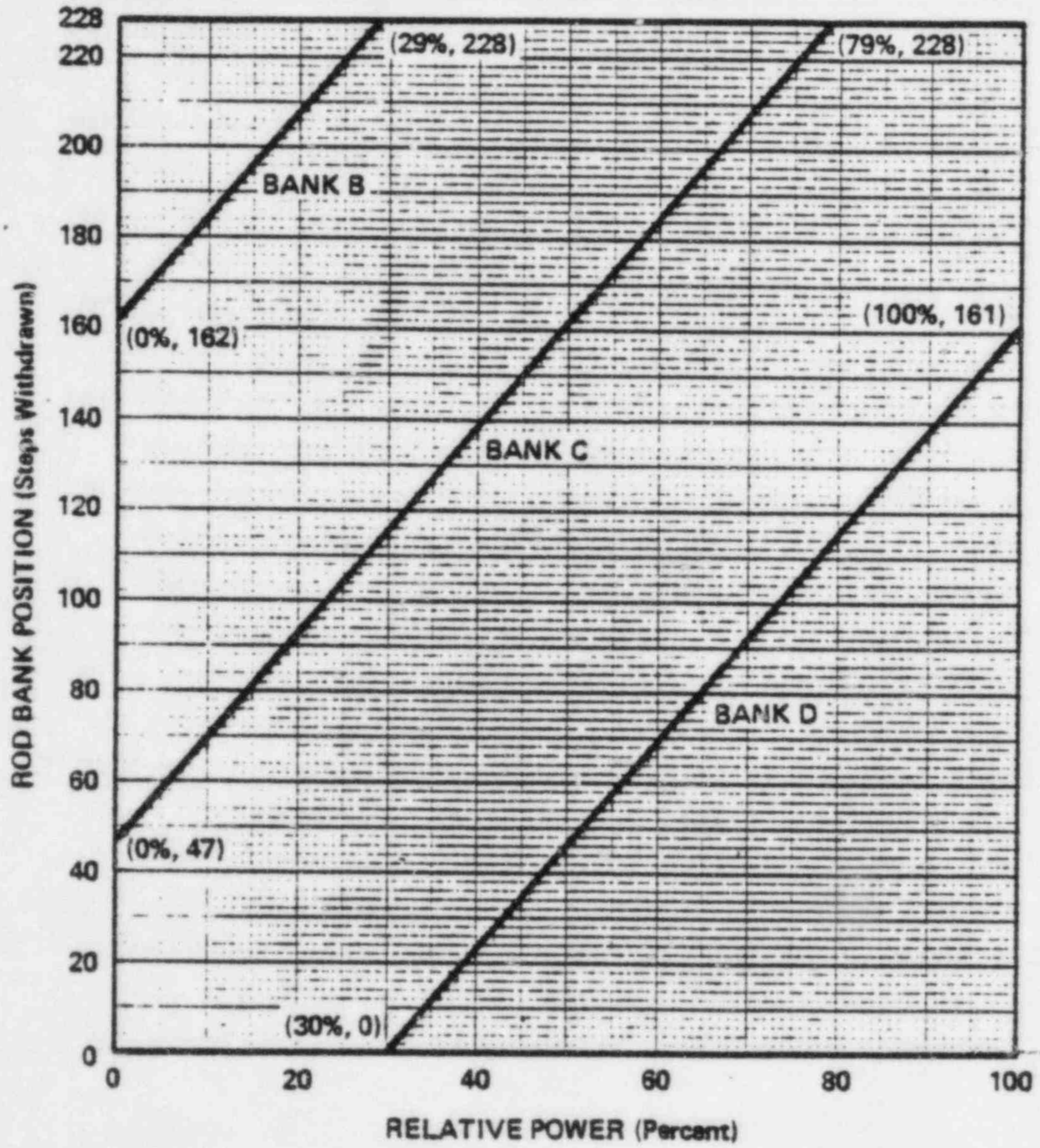

FIGURE 3.1-1

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
~~FOUR LOOP OPERATION~~

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~~Figure 3.1-2 left blank pending NRC approval of three-loop operation~~

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS ~~Four Loops in Operation~~	~~LIMITS~~ ~~Three Loops in Operation~~
Reactor Coolant System T_{avg}	$\leq$ 592°F	~~**~~
Pressurizer Pressure	$\geq$ 2220 psia*	~~**~~

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

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

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TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

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

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TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*With the Reactor Trip System breakers in the closed position and the Control Rod Drive System capable of rod withdrawal.
~~**Values left blank pending NRC approval of three loop operation.~~
#The provisions of Specification 3.0.4 are not applicable.
##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.
###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

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

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

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

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

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

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

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

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

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3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-3.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With four reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~b. With three reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves associated with an operating loop inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

b ~~c~~. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5.

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TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES ~~DURING FOUR LOOP OPERATION~~

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	87
2	65
3	43

TABLE 3.7-2

~~MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING THREE LOOP OPERATION~~

~~MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR*~~	~~MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)~~
~~1~~	~~**~~
~~2~~	~~**~~
~~3~~	~~**~~

~~*At least two safety valves shall be OPERABLE on the non-operating steam generator.~~

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

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3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary Coolant System pressure will be limited to within 110% (1320 psia) of its design pressure of 1200 psia during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 102% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 17.958×10^6 lbs/h which is 119% of the total secondary steam flow of 15.135×10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

~~For four loop operation:~~

$$SP = \frac{(X) - (Y)(V)}{X} \times (109).$$

~~For three loop operation:~~

$$\cancel{SP = \frac{(X) - (Y)(U)}{X} \times (*)}.$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

U = Maximum number of inoperable safety valves per operating steam line,

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SAFETY VALVES (Continued)

109 = Power Range Neutron Flux-High Trip Setpoint for four loop operation,

* = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for three loop operation. This value left blank pending NRC approval of three loop operation,

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

Y = Maximum relieving capacity of any one safety valve in lbs/hour.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

The motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 740 gpm at a pressure of 1450 psig to the entrance of the steam generators. The diesel-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 740 gpm at a pressure of 1450 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the RHR System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm reactor to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

ATTACHMENT Z

These changes delete all reference to three loop operation. This mode of operation is not supported by existing safety analyses.

8376N

Ret to G Tharpe 546-55 40-8027

MAR 2 8 1984

MEETING/TRIP REPORT

Docket No.; 40-8027

Applicant: Sequoyah Fuels Corporation (SFC)
Gore, Oklahoma

Date: March 20-21, 1984

Place: Oak Ridge National Laboratory (ORNL)
Oak Ridge, Tennessee

Attendees: NRC - E. Shum, M. Rhodes
Oak Ridge National Laboratory - Norm Hinkle
John Witherspoon
Richard McLean

Purpose: To discuss and resolve internal staff comments on the draft Environmental Impact Appraisal (EIA) for Sequoyah Fuels Corporation, Sequoyah Facility, Gore, Oklahoma.

Summary: On March 21, 1984, NRC staff met with ORNL staff to discuss questions and comments raised during the internal review of the SFC Draft EIA. The EIA is being prepared by the ORNL Environmental Review Team as part of the renewal action of SFC's Source Material License SUB-1010. The comments will be incorporated into the EIA and another draft will be submitted to the Uranium Process Licensing Section for staff review.

Original Signed By;
E. Y. Shum

E. Y. Shum
Uranium Process Licensing Section
Uranium Fuel Licensing Branch
Division of Fuel Cycle and
Material Safety

MARC J. RHODES

M. J. Rhodes
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