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September 19, 1984

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

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

Reference (a): August 27, 1984 letter from B. H. Youngblood to D. L. Farrar.

Dear Mr. Denton:

This is to provide proposed changes to the final draft version of the Byron 1 Technical Specifications that was distributed in reference (a). NRC review of the specific changes proposed here is necessary before the Technical Specifications can be finalized.

Attachments 1 through 19 to this letter contain marked-up pages of various sections of the Technical Specifications. The justification for the changes is provided in each attachment. We understand that the NRC will review each of these proposed changes and inform Commonwealth Edison of their acceptability.

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

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

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator

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cc: Byron Resident Inspector

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ATTACHMENT 1

Byron Station proposes to modify Surveillance Requirement 4.7.7.d.3, on the Non-Accessible Area Exhaust Filter Plenum Ventilation System (page 3/4 7-19) as indicated on the attached copy.

Justification

The deletion of "greater than or equal to 1/4 in. water gauge" is requested because sections 6.5.1.1.2b, 9.4.5.1.1a (items 1 and 7) and 9.4.5.1.2l4 of the FSAR as well as section 9.4.3 of the February 1982 SER states that these rooms are to be maintained at a negative pressure. No specific value for negative pressure is provided in these references. However, Table 3.11-2 of the FSAR indicates that the pressure in water gauge should be maintained in a range typically between -0.25 to 0.0. Current wording in the Technical Specifications is more restrictive than what is assumed in the FSAR.

Byron Station in their Pre-Operational Test 2.84.11, Auxiliary Building Ventilation Test, will verify the ECCS Equipment Rooms to be at a negative pressure of at least 1/8 in. (1/4 in ± 1/8 in) water gauge relative to the adjacent clean areas in the normal operating mode and in the accident mode (LOCA coincident with LOEP) with only the booster fans operating. The equipment rooms include:

- 1A and 1B Safety Injection Pump Rooms
- 1A and 1B Centrifugal Charging Pump Rooms
- 1A and 1B Containment Spray Pump Rooms
- 1A and 1B Residual Heat Removal (RHR) Pump Rooms
- 1A and 1B RHR Heat Exchanger Rooms

It is also requested that the ECCS Equipment Rooms negative pressure be verified relative to the adjacent clean areas rather than the outside atmosphere. Sections 9.4.5.1.1a3 and 9.4.5.1.2a6 of the FSAR as well as section 9.4.3 of the SER state that the system controls radioactivity by supplying air from the clean areas to areas with greater potential for contamination. Maintaining a negative pressure in the ECCS Equipment Rooms ensures that air flows from the adjacent clean areas into the potentially contaminated equipment rooms to control the spread of contamination. The system design does not include installed instruments which compare the pressure differential between the ECCS Equipment Rooms and the outside atmosphere. To make this measurement, temporary tubing would have to be installed from the ECCS Equipment Rooms to the outside atmosphere and therefore is not a recommended solution. The current system design allows the ECCS Equipment Room pressure to be compared relative to the adjacent clean areas and this is why the change has been requested.

SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying a system flow rate of 66,900 cfm ± 10% through the exhaust filters plenum during operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, when the average of the methyl iodide penetration for the three samples is less than 7.1%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks of less than 6.0 inches Water Gauge while operating the exhaust filters plenum at a flow rate of 66,900 cfm ± 10%, and

2) Verifying that the exaust filters plenum starts on manual initiation or Safety Injection test signal.

3) Verifying that the system maintains the ECCS equipment rooms at a negative pressure ~~of greater than or equal to 1/4 in. water gauge~~ relative to the ~~outside atmosphere~~ adjacent clean areas during system operation.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the exhaust filters plenum satisfies the in-place penetration testing acceptance criteria of less than 1.0% in accordance with ANSI N510-1975 for a DOP test aerosol while operating at a flow rate of 66,900 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the exhaust filters plenum satisfies the in-place penetration testing acceptance criteria of less than 1.0% remove greater than in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 66,900 cfm ± 10%.

ATTACHMENT 2

Byron Station proposes to modify Surveillance Requirement 4.3.4.2, on Turbine Overspeed Protection (page 3/4 3-75) as indicated on the attached copy.

Justification

This change is requested based on an Operation and Maintenance Memo 041 issued by Westinghouse. Following a Westinghouse review of testing frequency and performance data from turbine and component incidents records and a 1982 survey of utilities operating Westinghouse nuclear turbines, it was concluded that there was no significant difference in failure rates between valves tested weekly and those tested monthly. Westinghouse also noted that a monthly versus a weekly valve testing frequency may be beneficial because it reduces the time a plant is operating in a "transient state". Westinghouse recommended that the throttle, governor, interceptor and reheat stop valves be tested monthly.

With the requested change, Surveillance Requirements 4.3.4.2 a and c would be performed on the same frequency and therefore were combined. It is also requested that the words "During turbine operation" be added since if the unit were shutdown for any period of time it would be unnecessary to verify every month that the Turbine Overspeed Protection System was OPERABLE.

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3/4.3.4 TURBINE OVERSPEED PROTECTION

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LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one throttle valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam line(s) or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:

During turbine operation 31 direct observation of the movement of the

a. ^ ~~a~~At least once per ~~7~~ days by ~~cycling each of the following~~ valves below through ~~at least~~ one complete cycle from the running position:

1) Four high pressure turbine throttle valves,
2) Four high pressure turbine governor valves,
3) Six turbine reheat stop valves,
4) Six turbine reheat intercept valves, and

by cycling

b. Within 7 days prior to entering MODE 3 from MODE 4, ^each of the 12 extraction steam nonreturn check valves ~~shall be cycled~~ from the closed position.

~~c. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position,~~

During turbine operation

of

c ~~d.~~ ^ ~~a~~At least once per 31 days by direct observation, ~~verify~~ freedom of movement of ^ the 12 extraction steam nonreturn check valve weight arms.

each of

d ~~e.~~ At least once per 18 months by performance of a CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems, ~~and~~

e ~~f.~~ At least once per 40 months by disassembling at least one of each of the valves given in Specifications 4.3.4.2a. and b. above, and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

ATTACHMENT 3

Byron Station is requesting a change to Technical Specification 3.9.4, "Containment Building Penetrations" as shown on the attached copy.

This change is requested by Byron Station for the following reasons:

1. The removal of the equipment hatch during refueling outages will reduce occupational exposure by eliminating the wait time for passage through the personnel hatch and thereby augment ALARA.

2. It will reduce the outage time by eliminating unnecessary movement of the equipment hatch and provide ready and convenient access to containment.

3. It will reduce maintenance on the equipment hatch by reducing the frequency of removal and installation.

Justification

A fuel handling accident in the Fuel Handling Building has been addressed in Section 15.7 of FSAR, and additional information presented in response to Question 311.13. Therefore, the only potential accident affected by the proposed change to the Technical Specification would be a fuel handling accident in containment.

This accident was addressed in response to question 311.2, where it was shown that the containment radiation monitoring system would cause the containment purge isolation valves to close before any significant radiation would be released from containment.

Since the fuel handling accident in containment assumes the same scenario as a fuel handling accident in the Fuel Handling Building, the same radioactive release would occur in containment as was assumed for a fuel handling accident in the Fuel Handling Building. However, with the equipment hatch removed, this volume of radioactive gas would now be diluted by the volume of air in both containment and the Fuel Handling Building, thus reducing the concentration below that assumed for a fuel handling accident in the Fuel Handling Building. Thus the consequences of a fuel handling accident in the Fuel Handling Building envelope the consequences of a fuel handling accident in containment with the equipment hatch removed.

No analysis has been done to confirm that a release of radioactive gas in containment would activate the radiation monitoring system in the Fuel Handling Building, and thereby route the released activity through the Fuel Handling Building emergency exhaust system. Therefore, when the equipment hatch is removed, and fuel is being handled in containment, the Fuel Handling Building emergency exhaust system must be operating.

In summary, the proposed change will not involve an increase in the probability or consequences of accidents previously considered, and does not reduce any previously considered safety margin. Therefore, there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. The proposed change will, however, allow Byron Station to reduce occupational exposure and reduce the time required during a refueling outage.

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3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

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

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

INSERT →

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

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

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.4 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.

INSERT:

b. The equipment hatch may be removed, provided the Fuel Handling Building Exhaust Ventilation System is operating in the emergency exhaust mode.

ATTACHMENT 4

Byron Station proposes to modify Technical Specification 4.6.4.1, "Hydrogen Monitors" as indicated on the attached copy.

Justification

In surveillance requirement 4.6.4.1 a) and b), the term "nominal" is ambiguous and does not indicate the variation about the one and four volume percent H_2 that is acceptable when purchasing gas samples. In addition, the current surveillance procedure used at Byron Station to calibrate these hydrogen monitors requires five gas samples, with hydrogen contents which vary from zero percent H_2 to greater than 20 percent H_2, balance N_2. This procedure provides a more accurate calibration than can be obtained with two gas samples which only vary in hydrogen content from one to four volume percent. The proposed change to the Technical Specification will allow the more accurate procedure to be used, while allowing some flexibility in the purchase of gas samples.

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3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen monitors shall be OPERABLE.*

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

ATTACHMENT 5

Byron Station proposes to change Table 3.7-6 of Technical Specification 3.7.12, "Area Temperature Monitoring" by deleting certain areas previously included.

Justification

The APPLICABILITY statement of this specification requires temperature monitoring only on components and systems which are required to be OPERABLE. Byron Station has conducted a review of all areas previously identified in Table 3.7-6 and deleted those areas which do not contain components or systems required to be OPERABLE as defined in Technical Specifications.

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3/4.7.12 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.7-6

AREA TEMPERATURE MONITORING

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

ATTACHMENT 6

Byron Station proposes to modify item 6b on Table 3.3-1 Reactor Trip System Instrumentation (pg 3/4 3-2), Table Notations (pg 3/4 3-5) and Action Statement 5 (pg 3/4 3-6), as indicated on the attached copies.

Justification

This change is requested to accurately reflect the operation of the Boron Dilution Protection System. The change includes an action statement if two channels of the source range neutron flux monitors are not operable in Modes 3, 4 and 5 and allows for the block of the Boron Dilution Protection System when rods are being withdrawn.

In Mode 3, Byron procedure BGP 100-2 allows the operator to manually block both trains of the Boron Dilution Protection System (BDPS) prior to withdrawing the shutdown banks. Current wording in the Technical Specifications does not allow the BDPS to be blocked which could result in actuation of the system when a Reactor Startup is commenced. Therefore, double asterisks have been added to Modes 2, 3, 4 and 5 to allow for the block of the Boron Dilution Protection System when rods are being withdrawn. Block switches are provided on the main control board to perform this function. By design, this action disables only the automatic switchover of the charging pump suction from the VCT to the RWST on a flux doubling signal. The Reactor Trip and all alarms generated from the source range channels are neither bypessed nor blocked.

In addition, there is currently no action statement for loss of both Source Range Channels. In Mode 5 the provisions of LCO 3.0.3 do not apply. Therefore, it is proposed that Action Statement 5 on page 3/4 3-6 be modified to have an item a. which addresses the situation of the number of operable channels <u>one</u> less than the minimum channels operable requirement and item b was added to address the situation of the number of operable channels <u>two</u> less than the minimum channels operable requirement. In the event of the loss of both source range channels which results in the loss of both trains of the BDPS, Action 5b ensures the Reactor trip breakers are opened as soon as possible, the valves in the path of the boron dilution are secured closed and the Shutdown Margin is verified.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2#
b. Low Setpoint	4	2	3	1###, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3**, 4**, 5**	5
7. Overtemperature ΔT	4	2	3	1, 2	6#
8. Overpower ΔT	4	2	3	1, 2	6#
9. Pressurizer Pressure-Low (Above P-7)	4	2	3	1	6#

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TABLE 3.3-1 (Continued)

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TABLE NOTATIONS

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

#The provisions of Specification 3.0.4 are not applicable.

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

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

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

ACTION STATEMENTS

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

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

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

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

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

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

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

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

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

ACTION 5 - With the number of OPERABLE channels: a. One ~~one~~ less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status within 48 hours or within the next hour open the reactor trip breakers, suspend all operations involving positive reactivity changes, and verify valves 1CV-111B, 1CV-8428, 1CV-8439, 1CV-8441 and 1CV-8435 are closed and secured in position.
b. See Insert "AA"

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

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

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

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

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

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

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers within the next hour.

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

Insert "AA"

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

ATTACHMENT 7

Byron Station proposes to change item 7b on Table 4.3-2 (page 3/4 3-38) of Technical Specification 3.3.2 as shown on the attached copy.

Justification

There is no provision in the current design for testing the automatic opening of the Containment Sump Suction isolation valves on a RWST level low-low signal coincident with a Safety Injection Signal by means of an ACTUATION LOGIC TEST. The testing of this function should be performed under the ANALOG CHANNEL OPERATIONAL TEST. This testing would verify the operability of the trip function and the proper setpoint. This change makes Table 4.3-2 consistent with NUREG-0452 Rev. 4 Standard Technical Specifications.

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. Automatic Opening (Continued)								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. RWST Level-Low-Low Coincident With Safety Injection	N.A.	R	~~N.A.~~ M	N.A.	~~M~~ N.A.	N.A.	N.A.	1, 2, 3, 4
	See Item 1. above for all Safety Injection Surveillance Requirements							
8. Loss of Power								
a. ESF Bus Undervoltage	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Grid Degraded Voltage	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Feature Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
c. Low-Low T_{avg}, P-12	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Generator Water Level, P-14 (High-High)	S	R	M	N.A.	M(1)	M(1)	Q	1, 2, 3

TABLE NOTATION

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

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ATTACHMENT 8

Byron Station proposes to change Table 4.11-2, Radioactive Gaseous Waste Sampling and Analysis Program (page 3/4 11-10) as shown on the attached copy.

Justification

The requirement for monthly analysis of tritium in the Containment Purge should be deleted from item 2 of Table 4.11-2 because any containment purge effluent released from the plant exits via the Auxiliary Building vent stack where sampling and analysis of tritium will be provided in accordance with item 3 of Table 4.11-2. Since at Byron Station containment purge is not directly connected with the outside atmosphere it would be redundant to sample tritium in the containment purge and then in the Auxiliary Building vent stack. In addition, the containment purge monitor at Byron Station is not designed to detect tritium.

The requirement of continuous sampling of Containment Purge should also be deleted from item 4 of Table 4.11-2 because continuous sampling is performed by the Auxiliary Building vent stack monitors which includes contributions from containment purge.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)[1] (μCi/ml)
1. Waste Gas Decay Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[2]	1×10^{-4}
2. Containment Purge	P Each PURGE[3] Grab Sample	P Each PURGE[3]	Principal Gamma Emitters[2]	1×10^{-4}
		~~M~~	~~H-3~~	~~1×10^{-6}~~
3. Auxiliary Bldg Vent Stack (Units 1 and 2)	M[4][5] Grab Sample	M	Principal Gamma Emitters[2]	1×10^{-4}
			H-3	1×10^{-6}
4. All Release Types as listed in ~~2.e~~ ~~9 and~~ 3. above.	Continuous[6]	W[7] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[6]	W[7] Particulate Sample	Principal Gamma Emitters[2]	1×10^{-11}
	Continuous[6]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[6]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

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ATTACHMENT 9

Byron Station proposes to modify paragraph 6.2.2.c (page 6-1) as indicated on the attached copy.

Justification

It is proposed that the unit organization include two Radiation Chemistry Technicians on site when fuel is in the reactor. This change is consistent with what is currently required in the Byron General Site Emergency Plan (GSEP). Therefore, for consistency we request that this change be made.

(0548M)

6.1 RESPONSIBILITY

6.1.1 The Superintendent, Byron Station, shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Engineer (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Division Vice President and General Manager-Nuclear Stations shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown in Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1; and

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

c. Two ~~A~~ Radiation Chemistry Technicians,* qualified in radiation protection procedures, shall be on site when fuel is in the reactor;

d. ALL CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. A site Fire Brigade of at least five members* shall be maintained onsite at all times. The Fire Brigade shall not include the Shift Engineer, and the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

*The Radiation Chemistry Technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the requi ed positions.

ATTACHMENT 10

Byron Station proposes to change Technical Specification 5.3.1 (page 5-4) as shown on the attached copy.

Justification

The initial core loading contains some fuel assemblies with enrichments greater than 3.10 but less than 3.20 weight percent U^{235}.

The actual enrichment of future reload cores has not been determined. The upper limit on the enrichment of reload cores is set at 4.0 weight percent U^{235} by the value used in the criticality analysis for new fuel storage (FSAR Section 9.1).

(0548M)

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5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1619 grams uranium. The initial core loading shall have a maximum enrichment of ~~3.10~~ less than 3.20 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of ~~3.50~~ 4.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

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

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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

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

b. For a pressure of 2485 psig, and

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

VOLUME

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

5.5 METEOROLOGICAL TOWER LOCATION

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

ATTACHMENT 11

Byron Station proposes to modify Table 3.8-2, Motor-Operated Valves Thermal Overload Protection Devices (pages 3/4 8-30 to 3/4 8-33) as indicated on the attached copies.

Justification

Based on discussions with the NRC, it is our understanding that Table 3.8-2 of the Technical Specifications should contain all safety related motor-operated valves with thermal overload protection devices. As such, Table 3.8-2 has been revised to include all safety related motor-operated valves with thermal overload protection devices.

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TABLE 3.8-2

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION DEVICES

VALVE NUMBER	FUNCTION
1RC8001A	RC Loop 1A Hot Leg Stop Valve
1RC8001B	RC Loop 1B Hot Leg Stop Valve
1RC8001C	RC Loop 1C Hot Leg Stop Valve
1RC8001D	RC Loop 1D Hot Leg Stop Valve
1OG081	H2 Recomb Suction Cnmt. Isol. Valve
1CC9438	CC Wtr from RC Pumps Thermal Bar Isol. Valve
1CC9416	CC Wtr from RCPS Isol. Valve
1OG057A	H2 Recomb Cnmt. Isol. Valve Disch. "H"
1RC8003A	RC Loop 1A Bypass Leg Stop Valve
1RC8003D	RC Loop 1D Bypass Leg Stop Valve
1RH8701A	RC Loop 1A to RHR Pump Isol. Valve
1RH8702A	RC Loop 1C to RHR Pump Isol. Valve
1SI8808A	Accum. 1A Disch. Isol. Valve
1SI8808D	Accum. 1D Disch. Isol. Valve
1RY8000A	Pzr. Relief Isol. Valve 1A
1WO056B	Chilled Water Cnmt. Isol. Valve
1RC8002A	RC Loop 1A Cold Leg Stop Valve
1RC8002B	RC Loop 1B Cold Leg Stop Valve
1RC8002C	RC Loop 1C Cold Leg Stop Valve
1RC8002D	RC Loop 1D Cold Leg Stop Valve
1RC8003B	RC Loop 1B Bypass Leg Stop Valve
1RC8003C	RC Loop 1C Bypass Leg Stop Valve
1RY8000B	Pzr. Relief Valve 1B
1OG080	H2 Recomb Suct. Cnmt. Isol. Valve
1WO056A	Chilled Water Cnmt. Isol. Valve
1OG079	H2 Recomb. Disch. Cnmt. Isol. Valve
1CV8112	RC Pump Seal Water Return Isol. Valve
1RH8701B	RC Loop 1A to RHR Pump Isol. Valve
1RH8702B	RC Loop 1C to RHR Pump Isol. Valve
1SI8808B	Accum. 1B Disch. Isol. Valve
1SI8808C	Accum. 1C Disch. Isol. Valve
1CC9414	CC Water from React. Chg. Pumps Isol. Valve 1B
0OG059	Unit 1 Suct Isol Vlv H_2 Recomb
0OG060	Unit 1 Discharge Isol Vlv H_2 Recombiner
0OG061	Unit Discharge Xtie for H_2 Recombiner
0OG062	Unit Xtie on Discharge of H_2 Recombiner
0OG063	Unit Suction Xtie for H_2 Recombiner
0OG064	Unit Suction Xtie for H_2 Recombiners
0OG065	OB H_2 Analyzer Inlet Isol Vlv

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION DEVICES

	VALVE NUMBER	FUNCTION
	00G066	OB H_2 Recomb Disch Isol Vlv
	10G057A	OA H_2 Recomb ~~Cnmt~~. Isol. Valve Disch.
	10G079	H_2 Recomb Disch. Cnmt. Isol. Valve
	10G080	H_2 Recomb Suct. Cnmt. Isol. Valve
	10G081	H_2 Recomb Suction Cnmt. Isol. Valve
	10G082	OA H_2 Recomb Disch Cnmt Isol Vlv
	10G083	OA H_2 Recomb Disch Cnmt Isol Vlv
	10G084	OA H_2 Recomb Cnmt Outlet Isol Vlv
	10G085	H_2 Recomb Cnmt Outlet Isol Vlv
Insert A		
	1AF013A	AF Mtr Drv Pmp Disch Hdr Dwst Isol Vlv
	1AF013B	AF Mtr Drv Pmp Dsch Hdr Dwst Isol Vlv
	1AF013C	AF Mtr Drv Pp Disch Hdr Dwst Isol Vlv
	1AF013D	AF Mtr Drv Pp Disch Hdr Dwst Isol Vlv
	1AF013E	AF Dsl Drv Pm Dsch Hdr Dwst Isol Vlv
	1AF013F	AF Dsl Drv Pp Dsch Hdr Dwst Isol Vlv
	1AF013G	AF Dsl Drv Pp Dsch Hdr Dwst Isol Vlv
	1AF013H	AF Dsl Drv Pp Dsch Hdr Dwst Isol Vlv
Insert B		
	1CC685	RCP Thermal Barrier Outlet Hdr Cnmt Isol Vlv
Insert C →	1CC9413A	RCP CC Supply Dwst CNMT Isol
	1CC9413B	RCPs CC Supply Upst CNMT Isol
Insert D →	1CC9414	CC Water from ~~React. Chg. Pumps~~ RCPs Isol. Valve ~~1B~~
	1CC9416	CC Wtr from RCPS Isol. Valve
Insert E →	1CC9438	CC Wtr from RC Pumps Thermal Bar Isol. Valve
	1CS001A	1A CS Pp Suct from RWST ~~364~~
	1CS001B	1B CS Pp Suction from RWST ~~364~~
	1CS007A	CS Pp 1A Disch Line Dwst Isol Vlv
	1CS007B	CS Pp 1B Disch Line Downstream Isol Vlv
	1CS009A	1A Pump Suction from 1A Recirc Sump
	1CS009B	1B CS Cont Recirc Sump B Suct Isol Vlv to CS
	1CS019A	CS Eductor 1A Suction Conn Isol Vlv
	1CS019B	CS Eductor 1B Suction Conn Isol Vlv
Insert F		
	1CV112D	MOV RWST to Chg Pp Suct Hdr
	1CV112E	MOV RWST to Chg Pp Suct Hdr
Insert G →	1CV8100	MOV RCP Seal Leakoff Hdr Isol
	1CV8105	MOV Chrg Pps Disch Hdr Isol Vlv
	1CV8106	MOV Chrg Pps Disch Hdr Isol Vlv
	1CV8109	MOV PD Chrg. Pp Miniflow Recirc. Vlv
	1CV8110	MOV A & B Chg. pp Recirc Downstream Isol
	1CV8111	MOV A & B Chg Pp Recirc Upstream Isol
	1CV8112	RC Pump Seal Water Return Isol. Valve

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION DEVICES

VALVE NUMBER	FUNCTION
1CV8355A	MOV RCP 1A Seal Inj Inlet to containment Isol
1CV8355B	MOV RCP 1B Seal Inj Inlet Isol
1CV8355C	MOV RCP 1C Seal Inj Isol
1CV8355D	MOV RCP 1D Seal Inj Isol
1CV8804A	MOV RHR Sys X-Tie Vlv to Chrgng Pump Suction Hdr A.B.
1RC8001A	RC Loop 1A Hot Leg Stop Valve
1RC8001B	RC Loop 1B Hot Leg Stop Valve
1RC8001C	RC Loop 1C Hot Leg Stop Valve
1RC8001D	RC Loop 1D Hot Leg Stop Valve
1RC8002A	RC Loop 1A Cold Leg Stop Valve
1RC8002B	RC Loop 1B Cold Leg Stop Valve
1RC8002C	RC Loop 1C Cold Leg Stop Valve
1RC8002D	RC Loop 1D Cold Leg Stop Valve
1RC8003A	RC Loop 1A Bypass Leg Stop Valve
1RC8003B	RC Loop 1B Bypass Leg Stop Valve
1RC8003C	RC Loop 1C Bypass Leg Stop Valve
1RC8003D	RC Loop 1D Bypass Leg Stop Valve
1RH610	RH PP 1RH01PB Recirc, Line Isol.
1RH611	RH PP 1RH01PB Recirc, Line Isol.
1RH8701A	RC Loop 1A to RHR Pump Isol. Valve
1RH8702A	RC Loop 1C to RHR Pump Isol. Valve
1RH8701B	RC Loop 1A to RHR Pump Isol. Valve
1RH8702B	RC Loop 1C to RHR Pump Isol. Valve
1RH8716A	RH HX 1RH02AA Dwnstrm Isol Vlv
1RH8616B	RH HX 1RH02AB Dwnstrm Isol Valve
1RY8000A	Prz. Relief Isol. Valve 1A
1RY8000B	Prz. Relief Valve 1B Isol.
1SI8801A	SI Charging Pump Disch Isol Vlv
1SI8801B	SI Charging Pump Disch Isol Vlv
1SI8802A	SI PP 1A Disch Line Dwst Cont Isol Vlv
1SI8802B	SI PP 1B Disch Line Dwst Isol Vlv
1SI8804B	SI Pump 1B Suct X-tie from RHR HX
1SI8806	SI Pumps Upstream Suction Isol
1SI8807A	SI to Chg PP Suction Crosstie Isol Vlv
1SI8807B	SI to Chg PP Suction Crosstie Isol Vlv
1SI8808A	Accum. 1A Disch. Isol. Valve
1SI8808B	Accum. 1B Disch. Isol. Valve
1SI8808C	Accum. 1C Disch. Isol. Valve
1SI8808D	Accum. 1D Disch. Isol. Valve

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION DEVICES

VALVE NUMBER	FUNCTION
1SI8809A	SI RX HX 1A Dsch Line Dwst Isol Vlv
1SI8809B	SI RX HX 1B Dsch Line Dwst Isol Vlv
1SI8811A	SI Cnmt Sump A Outlet Isol Vlv
1SI8811B	SI Cnmt Sump B Outlet Isol Vlv A
1SI8812A	SI Rwst to RH Pp 1B Outlet Isol Vlv
1SI8812B	SI Rwst to RH Pp 1B Outlet Isol Vlv
1SI8813	SI Pumps 1A-1B Recirc Line Dwst Isol
1SI8814	SI Pump 1A Recirc Line Isol Vlv
1SI8835	SI Pumps X-tie Disch Isol Vlv
1SI8840	SI RHR XH Disch Line Upstrm Cont Pen Isl Vlv
I 1ST8821A	SI PP 1A Disch Line X-tie Isol Vlv
I 1ST8821B	SI Pump 1B Disch Line X-tie Isol Vlv
I 1ST8920	SI Pump 1B Recirc Line Isol Vlv
I 1ST8923A	SI PP 1A Suction Isol Vlv
I 1ST8923B	SI Pump 1B Suct Isol Valve
1SI8924	SI Pump 1A Suction X-tie Dwnstrm Isol Vlv
1SX016B	RCFC B&D Sx Supply MOV
1SX016A	RCFC A&C SX Supply MOV
1SX027A	RCFC A&C Return C
1SX027B	RCFC B&D SX Return MOV
~~1W0006A~~	~~Chilled Wtr Coils 1A & 1C Supply Isol Vlv~~
~~1W0006B~~	~~Chilled Wtr Coils 1B & 1D Supply Isol Vlv~~
~~1W0020A~~	~~Chilled Wtr Coils 1A & 1C Return Isol Vlv~~
~~1W0020B~~	~~Chilled Wtr Coils 1B & 1D Return Isol Vlv~~
~~1W0023A~~	~~Chiller 1W001CA Oil Cooler Return Vlv~~
~~1W0023B~~	~~Chiller 1W001CB Oil Cooler Return Vlv~~
~~1W0056A~~	~~Chilled Water Cnmt. Isol. Valve~~
~~1W0056B~~	~~Chilled Water Cnmt. Isol. Valve~~

Insert H

Insert A

1AFC06A	1A AF Pp SX Suct Isol Vlv
1AFC06B	1B AF Pp SX Suct Dwst Isol Vlv

Insert B

1AFC17A	1A AF Pp SX Suct Upst Isol Vlv
1AFC17B	1B AF Pp SX Suct Upst Isol Vlv

Insert C

1CC9412A	CC to RH HX 1A Isol Vlv
1CC9412B	CC to RH HX 1B Isol Vlv

Insert D

1CC 9415	Unit 1 Serv. Loop Isol Vlv

Insert E

1CC9473A	Disch Hdr X-tie Isol Vlv
1CC9473B	Disch Hdr X-tie Isol Vlv

Insert F

1CV112B MOV VCT Outlet Upstm Isol VCT Vlv

1CV112C MOV VCT Outlet Dwnstm Isol VCT Vlv

Insert G

1CV8104 MOV Emerg Boration Vlv

Insert H

VALVE NUMBER	FUNCTION
0SX007	CC HX Outlet Vlv
0SX063A	SX to Cont Rm Refrig Cdsr 0A
0SX063B	SX to Cont Rm Refrig Cdsr 0B
0SX146	CC HX "0" return Vlv to Unit 1 MDCT
0SX147	CC HX "0" return Vlv to Unit 2 MDCT
0SX157A	SX M/U Pp 0A Supply Fill to MDCT ~~MOV 0SX2AA SX Tower Vlv~~
0SX157B	SX M/U Pp 0B Supply to MDCT 0B MOV
0SX158A	SX M/U Pp 0A Supply Fill to MDCT MOV
0SX158B	SX M/U Pp 0B Supply to MDCT 0B MOV
0SX162A	MDCT 0A Bypass to basin MOV
0SX162B	MDCT 0B Bypass to basin MOV
0SX162C	MDCT 0A Bypass to basin MOV
0SX162D	MDCT 0B Bypass to basin MOV
0SX163A	MDCT 0A Riser Isol Vlv MOV
0SX163B	MDCT 0A Riser Isol Vlv MOV
0SX163C	MDCT 0A Riser Isol Vlv MOV
0SX163D	MDCT 0A Riser Isol Vlv MOV
0SX163E	MDCT 0B Riser Isol Vlv MOV
0SX163F	MDCT 0B Riser Isol Vlv MOV
0SX163G	MDCT 0B Riser Isol Vlv MOV
0SX163H	MDCT 0B Riser Isol Vlv MOV

VALVE NUMBER	FUNCTION
1SX001A	1A SX Pp Suct Vlv MOV
1SX001B	1B SX Pp Suct. Vlv MOV
1SX004	U-1 SX Supply to U-1 CCW HX MOV
1SX005	1B SX Pp Supply to 0 CCW HX MOV
1SX007	CC HX Outlet Vlv
1SX010	U-1 Trn A return Vlv AB
1SX011	TrnA TrnB Unit 1 return X-tie Vlv AB
1SX033	1A SX Pp Disch X-tie MOV
1SX034	1B SX Pp Disch X-tie MOV
1SX136	Unit 1 Trn B return Vlv AB
1SX150A	Sx strn drn to waste treatment bldg MOV
1SX150B	Sx strn drn to TR bldg MOV

ATTACHMENT 12

Byron Station proposes to change Technical Specification 5.6.3 (page 5-5) as shown on the attached copy.

Justification

The spent fuel storage pool is designed to store 1050 spent fuel assemblies and 10 failed fuel assemblies for a total of 1060 fuel assemblies. This change is consistent with the FSAR.

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5.6 FUEL STORAGE

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CRITICALITY

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

DRAINAGE

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

CAPACITY

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

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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

ATTACHMENT 13

Byron Station proposes to modify paragraph 6.12.2 (page 6-24) of the Technical Specification as indicated on the attached copy.

Justification

Byron Station will completely barricade and conspicuously post all entrances into individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as in containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area. These areas will be temporarily established as required and Byron Station does not have a system for installing flashing lights as a warning device. There is an engineering concern with providing a suitable power supply in such an adverse environment and with the installation of the flashing lights in containment.

In addition, the requirements of the Code of Federal Regulations are met for entry into a high radiation area and there are no additional requirements for the situation described above. It is felt that barricading and posting of the areas would provide adequate control and it is not necessary to provide an additional warning device. Therefore, it is requested that the requirement for a flashing light be deleted.

HIGH RADIATION AREA (Continued)

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

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

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

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

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

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

ATTACHMENT 14

Byron Station proposes to modify pages 6-2, 6-6, 6-7, 6-8, 6-9, 6-10, 6-12 and 6-13 as indicated in the attached copies.

Justification

In section 6.2.3 on page 6-2 the name of the Independent Safety Engineering Group (ISEG) has been changed to the Onsite Nuclear Safety Group (ONSG). This change is as a result of a General Company Order issued by the Chairman and President of Commonwealth Edison Company.

It is also requested that records of activity by the ONSG be forwarded "quarterly" rather than "each calendar month". Byron Station intends to document weekly reports and these will be compiled and issued quarterly. It is felt that issuing quarterly reports will be more efficient than issuing monthly reports. Quarterly reports are consistent with the program currently established by the Office of Nuclear Safety.

In section 6.4.1 on page 6-6 ISEG is changed to the Office of Nuclear Safety. Within the Commonwealth Edison organization the responsibility for collecting and documenting operational experience feedback (OPEX) resides with the headquarters Office of Nuclear Safety.

Since Commonwealth Edison operates several nuclear units, the headquarters Office of Nuclear Safety performs several functions such as the OPEX function to augment the on-site groups. Those experiences determined by the onsite group as needing feedback are forwarded to the group at the headquarters office for formal issuance as an OPEX.

In section 6.5.1 on page 6-7 the Supervisor of the Offsite Review and Investigative Function is appointed by the Manager of Nuclear Safety and not by the Executive Vice President. This change is requested to reflect Commonwealth Edisons organizational responsibilities.

The changes are made on page 6-9 to reflect the current organizational and functional responsibility at Commonwealth Edison.

The changes proposed on pages 6-8, 6-10, 6-12 and 6-13 are to correct typographical errors or to maintain consistency with the functional titles used throughout Section 6.

Several of the changes proposed are also to maintain consistency with terminology between other Commonwealth Edison operating units and the Byron Station.

UNIT STAFF (Continued)

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

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

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

FUNCTION

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

COMPOSITION

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

RESPONSIBILITIES

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

RECORDS

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

6.2.4 SHIFT TECHNICAL ADVISOR

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

*Not responsible for sign-off function.

6.2.4 SHIFT TECHNICAL ADVISOR (Continued)

To assure capability for performance of all STA functions:

(1) The shift foreman (SRO) shall participate in the SCRE shift relief turnover.

(2) During the shift, the shift engineer and the shift foreman (SRO) shall be made aware of any significant changes in plant status in a timely manner by the SCRE.

(3) During the shift, the shift engineer and the shift foreman (SRO) shall remain abreast of the current plant status. The shift foreman (SRO) shall return to the control room two or three times per shift, where practicable, to confer with the SCRE regarding plant status. Where not practicable to return to the control room, the shift foreman (SRO) shall periodically check with the SCRE for a plant status update. The shift foreman (SRO) shall not abandon duties original to reactor operation, unless specifically ordered by the shift engineer.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971. The Rad/Chem Supervisor or Lead Health Physicist, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, for a Radiation Protection Manager. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Production Training Department and shall meet or exceed the requirements and recommendations of Section 5 of ANSI/ANS 3.1-1978 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience identified by the ~~ISEG~~. Office of Nuclear Safety.

6.5 REVIEW INVESTIGATION AND AUDIT

The Review and Investigative Function and the Audit Function of activities affecting quality during facility operations shall be constituted and have the responsibilities and authorities outlined below.

6.5 REVIEW INVESTIGATION AND AUDIT (Continued)

OFFSITE

Manager of Nuclear Safety

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

a. Offsite Review and Investigative Function

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

ive

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

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

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

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

OFFSITE (Continued)

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

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

7) All REPORTABLE EVENTS;

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

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

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

b. Audit Function

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

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

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

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

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

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

OFFSITE (Continued)

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

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

7) Onsite and offsite reviews;

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

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

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

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

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

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

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

c. Authority

Manager of / and the / report to the Chairman and President

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

OFFSITE (Continued)

d. Records

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

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

e. Procedures

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

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

2) Use of committees and consultants,

3) Review and approval,

4) Detailed listing of items to be reviewed,

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

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

f. Personnel

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

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

OFFSITE (Continued)

h) Instrumentation and Control

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

i) Metallurgy

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

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

ONSITE

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

a. Onsite Review and Investigative Function

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

b. Responsibility

The responsibilities of the personnel performing this function are:

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

2) Review of all proposed tests and experiments that affect nuclear safety;

ONSITE (Continued)

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

4) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety;

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

6) Review of all REPORTABLE EVENTS;

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

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

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

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

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

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

c. Authority

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

ATTACHMENT 15

Byron Station proposes to modify surveillance requirement 4.5.1.1c as shown on the attached copy.

Justification

Verifying that the breaker is open accomplishes the same objective without requiring the operator to remove fuses.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 70 gallons by verifying the boron concentration of the accumulator solution,

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

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

ATTACHMENT 16

Byron Station proposes to change item 6f of Technical Specification Table 3.3-3, "Engineered Safety Features Actuation System Instrumentation", as indicated on the attached copy.

Justification

The current version of the Specification, which identifies ACTION statement 18, requires a plant shutdown of the number of OPERABLE channels is one less than the MINIMUM CHANNELS OPERABLE, and the inoperable channel cannot be restored to OPERABLE status within 48 hours. With the proposed change, which identified ACTION statement 19, an inoperable channel may be placed in the tripped condition and STARTUP and/or POWER OPERATION may continue. Since there are only two channels available, when one is placed in the trip condition, the plant protection is conservative since a trip signal from the remaining channel will meet the two channels to trip requirement. The proposed changed provides flexability for plant operation without reducing plant protection.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater					
a. Manual Initiation	2	1	2	1, 2, 3	22
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Stm. Gen. Water Level-Low-Low					
1) Start Motor-Driven Pump	4/stm. gen.	2/stm. gen. in any operating stm gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	19*
2) Start Diesel-Driven Pump	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	19*
d. Undervoltage - RCP Bus-Start Motor-Driven Pump and Diesel-Driven Pump	4-1/bus	2	3	1, 2	19*
e. Safety Injection - Start Motor-Driven Pump and Diesel-Driven Pump	See Item 1. above for all Safety Injection initiating functions and requirements.				
f. Division 11 ESF Bus Undervoltage-Start Motor-Driven Pump (Start as part of DG sequencing)	2	2	2 1	1, 2, 3	18 19

FINAL DRAFT

ATTACHMENT 17

Byron Station proposes to modify Technical Specification Sections 4.7.1.2.3(b) (page 3/4 7-5), 4.7.5.3(b) (page 3/4 7-14), 4.7.10.1.2(b) (page 3/4 7-30) and 4.8.1.1.2(d) and (e) (pages 3/4 8-3 and 3/4 8-4) as shown on the attached copies.

Justification

The proposed changes to the diesel oil sampling requirements as shown on the attached sheets are in accordance with previous discussions and transmittals to the NRC.

SURVEILLANCE REQUIREMENTS (Continued)

valve (manual, power operated or automatic

2) Verifying by flow or position check that each ~~non-automatic~~ ~~valve~~ 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 actuates 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 drain sample of diesel fuel from its day tank, obtained in accordance with ASTM-D4057-1981 ~~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.

FINAL DRAFT

SURVEILLANCE REQUIREMENTS (Continued)

4.7.5.3 The essential service water make-up pump shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that:

1) The fuel storage tank level is at least 16%,

2) The diesel starts from ambient conditions on a simulated low basin level test signal and operates for at least 30 minutes, and

3) Each valve (manual, power-operated, or automatic) in the flow path is in its correct position.

b. At least once per 92 days by verifying that a drain sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM ~~-D270-1975~~, D4057-198 is within the acceptable limits specific in Table 1 of ASTM-D975-1977 when checked for viscosity, water, and sediment; and

c. At least once per 18 months by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service and by cycling each testable valve in the flow path through at least one complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that a drain sample of diesel fuel from the fuel oil day tank, obtained in accordance with ASTM ~~D270-1975~~ D4057-1981, 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 procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.10.1.3 The fire pump diesel starting 24-volt battery bank shall be demonstrated OPERABLE:

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

   1) The electrolyte level of each battery is above the plates, and

   2) The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery, and

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

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

   2) The battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anticorrosion material.

SURVEILLANCE REQUIREMENTS (Continued)

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1) Verifying the fuel level in the day tank,

2) Verifying the fuel level in the fuel storage tank,

3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day tank,

4) Verifying the diesel starts from ambient condition and accelerates to at least 600 rpm in less than or equal to 10 seconds.* The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds* after the start signal. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss of ESF bus voltage by itself, or

c) Simulated loss of ESF bus voltage in conjunction with an ESF actuation test signal, or

d) An ESF actuation test signal by itself.

5) Verifying the generator is synchronized, loaded to greater than or equal to 5500 kW in less than or equal to 60 seconds,* operates with a load greater than or equal to 5500 kW for at least 60 minutes, and

6) Verifying the diesel generator is aligned to provide standby power to the associated ESF busses.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tanks;

c. At least once per 92 days by checking for and removing accumulated water from the fuel oil storage tanks;

d. ~~At least once per 92 days and from new fuel oil prior to its addition to the storage tanks by verifying that a sample obtained in accordance with ASTM-D270-1975 meets the following minimum requirements in accordance with the tests specified in ASTM-D975-1977:~~

Insert "5"

~~1) A water and sediment content of less than or equal to 0.05 volume percent;~~

~~2) A kinematic viscosity of 40°C of greater than or equal to 1.3 centistokes, but less than or equal to 4.1 centistokes;~~

*The diesel generator start (10 sec) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts for the purpose of this surveillance testing may be preceded by an engine prelube period and/or other warmup procedures recommended by the manufacturer so that mechanical stress and wear on the diesel engine is minimized.

To: CAL MOOR
NRC
4 SHEETS

Attachment 5
To pg 3/4 8-3

4.8.1.1.2

d. By sampling new fuel oil in accordance with ASTM-D4057 prior to addition to storage tanks and:

1) By verifying in accordance with the tests specified in ASTM-D975-81 prior to addition to the storage tanks that the sample has:

a) An API Gravity of within 0.3 degrees at 60/60 F, or a specific gravity of within 0.0016 at 60/60 F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60 F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees;

b) A kinematic viscosity at 40 F of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if the gravity was not determined by comparision with the supplier's certification;

c) A flash point equal to or greater than 125 F; and

d) A clear and bright appearance with proper color when tested in accordance with ASTM-D4176-82.

OTHER ~~VISCOSITY, WATER AND SEDIMENT~~

2) By verifying within 30 days of obtaining the sample that the ~~other~~ properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that the analysis for sulfur may be performed in accordance with ASTM-D1552-79 or ASTM-D2622-82.

From Calloway

e. ~~At least once every 92 days by obtaining a drain sample in accordance with ASTM-D4057-81 and verifying that the properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that the analysis for sulfur may be performed in accordance with ASTM-D1552-79 or ASTM-D2622-82.~~ VISCOSIT WATER AND SEDIMEN

e. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78. ~~Method A;*~~

t
n10

3) ~~A specific gravity as specified by the manufacturer at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity at 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees;~~

f. 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 while maintaining voltage at 4160 $\pm$ 420 volts and frequency at 60 $\pm$ 4.5 Hz, (transient state), 60 $\pm$ 1.2 Hz (steady state).

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 energization, the steady-state voltage and frequency of the ESF busses shall be maintained at 4160 $\pm$ 420 volts and 60 $\pm$ 1.2 Hz during this test.

ATTACHMENT 18

Byron Station proposes to modify Technical Specification 3.4.1.2 and 3.10.4 as shown on the attached copies:

<u>Justification</u>

The change to specification 3.4.1.2 is merely a note at the bottom of the page referencing specification 3.10.4.

The change to specification 3.10 4 maintains the requirement of specification 3.4.1.2, that three reactor coolant loops must be OPERABLE in mode 3, but deletes the requirement that two reactor coolant loops must be operating. This change is an extension to mode 3 of the special test exception of specification 3.4.1.1 which allows the reactor coolant pumps to be turned off in modes 1 and 2 during startup and PHYSICS TESTS. Allowing the reactor coolant pumps to be off in mode 3 during rod drop tests presents no greater concern than allowing all reactor coolant pumps to be off in modes 1 and 2 during startup and PHYSICS TESTS.

These changes are requested to allow the hot no-flow rod drop time measurements to be performed in a timely manner without the requirement to start the reactor coolant pumps every hour which would result in unnecessary cycling of the RCP's.

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LIMITING CONDITION FOR OPERATION

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

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

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

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

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

APPLICABILITY: MODE 3.**

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side narrow range water level to be greater than or equal to 41% at least once per 12 hours.

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

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

** See Special Test Exception 3.10.4

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3/4.10.4 REACTOR COOLANT LOOPS

FINAL DRAFT

LIMITING CONDITION FOR OPERATION

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

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

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

2~~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.

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

APPLICABILITY: During operation below the P-7 Interlock Setpoint, or performance of rod drop time measurements in mode 3.

ACTION:

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

b. With less than the above required reactor coolant loops OPERABLE during the performance of rod drop time measurements in mode 3, immediately initiate corrective action to restore the required loops to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

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

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

ATTACHMENT 19

Byron Station proposes to modify the Bases section of Limiting Safety System Settings as shown on the attached copy.

Justification

The proposed changes correct inaccuracies in the P-6 description and provide a more complete description of the P-10 interlock.

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

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

an automatic

the manual block that

and Neutron Flux doubling

Source Range

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

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

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

P-13 Provides input to P-7.

provides an automatic backup function to

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