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June 20, 1984

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

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

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

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

(c): April 2, 1984 letter from T. R. Tramm to H. R. Denton.

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

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

Dear Mr. Denton:

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

Attachments A through J to this letter contain marked-up pages of various sections of the Technical Specifications. A summary explanation of the changes is provided for each attachment. Justifications are provided where appropriate.

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

B001
1/15

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

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

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator

lm

cc: Byron Resident Inspector

8838N

ATTACHMENT A
(Bases Section 2.0)

Circled items noted in this attachment have been previously submitted.

1) Section 2.1.1 (pg. B2-1) Reactor Core

Delete "and a reference cosine with a peak of 1.55 for axial power shape."

Delete the previous change which changed 1.55 to 1.49. The number is now 1.55.

This change is based on WCAP 10315 - "Nuclear Design and Core Physics Characteristics of the Byron Unit 1 Nuclear Power Plant Cycle 1".

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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. Insert "A"

The curves of Figures 2.1-1 ~~and 2.1-2~~ show 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~~ ~~1.49~~ 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: ×

$$F^N_{\Delta H} = \cancel{1.55}\ \cancel{1.49}\ 1.55\ [1+ 0.3\ (1-P)]$$ ×

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

ATTACHMENT B
(Section 3/4.2)

Circled items noted in this attachment have been previously submitted.

1) Section 3/4.2.1 (pg. 3/4 2-1) Axial Flux Difference.

In Sections 3.2.1.a and 3.2.1.b, delete the number "3000" and replace wih "5000". Also, in Section 3.2.1.b delete the words "+3%, -12%" and replace with "+3%, -9% for the initial cycle and +3%, -12% each subsequent cycle".

This change is required for initial PWR cores and has been recommended by Westinghouse. Subsequent cycles will revert to 3000MWD/MTU and +3% to -12%.

2) Section 3/4.2.3 (pg. 3/4 2-8) RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor

(pg. 3/4 2-8)

Delete the original LCO and replace with "3.2.3 Indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows:

a. RCS Total Flowrate $\geq$ 386,000 GPM

b. $F^N_{\Delta H} \leq 1.55\ [1.0 + 0.3\ (1.0-P)]$

where:

Measured values of $F^N_{\Delta H}$ are obtained by using the movable incore detectors to obtain a power distribution map., and

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

Delete "the combination of".

Delete "and R" and replace with "and $F^N_{\Delta H}$".

Delete "shown on Figure 3.2-3".

-(pg. 3/4 2-9)

Delete Figure 3.2-3

-(pg. 3/4 2-10)

Item b delete "comparison" and insert "determination".

Delete the words "the combination of R" and insert "$F^N_{\Delta}H$".

Item c delete "the combination of R" and insert "$F^N_{\Delta}H$".

Delete "comparison" and insert "determination".

Delete the words "within the region of".

Delete the words "operation shown on Figure 3.2-3."

Section 4.2.3.2 delete "the combination of".

Delete "R" and replace with "$F^N_{\Delta}H$".

Delete the words "within the region of" and also "operation of Figure 3.2-3."

Section 4.2.3.3 delete the words "within the region of", "operation of Figure 3.2-3" and "the most recently obtained value of R obtained per Specification 4.2.3.2 is assumed to exist." Following the words "12 hours when" insert "outside the above limits".

With the elimination of Rod Bow Penalty, all references to R can be deleted. The proposed changes define $F^N_{\Delta}H$ directly without using a ratio.

Since there is no rod bow penalty, Figure 3.2-3 (pg. 3/4 2-9) could be simplified and described by a rectangle. The proposed change to the LCO Section 3.2.3 incorporates the requirements of Figure 3.2-3; therefore, the figure we've previously submitted can be deleted.

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

+3%, -9% for the initial cycle and
+3%, -12% each subsequent cycle

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the following target band (flux difference units) about the target flux difference:

a. ± 5% for core average accumulated burnup of less than or equal to ~~3000~~ 5000 MWD/MTU, and

b. ~~+ 3%, -12%~~ for core average accumulated burnup of greater than ~~3000~~ 5000 MWD/MTU.

X

The indicated AFD may deviate outside the above required target band at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1 and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the above required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1 above 15% of RATED THERMAL POWER*.

ACTION:

a. With the indicated AFD outside of the above required target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes, either:
   1. Restore the indicated AFD to within the above required target band limits, or
   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits of Figure 3.2-1 and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:
   1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and
   2. The Power Range Neutron Flux# High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours. X

*See Special Test Exception 3.10.2.

#Surveillance testing of the Power Range Neutron Flux channel may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

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.~~

Replace with Insert A attached.

~~Where:~~

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

~~b. $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, 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 $F^N_{\Delta H}$.~~

APPLICABILITY: MODE 1.

ACTION:

With ~~the combination of~~ RCS total flow rate ~~and R~~ or $F^N_{\Delta H}$ 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~~ and $F^N_{\Delta H}$ 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.

Insert A

2.3 Indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows for four loop operation.

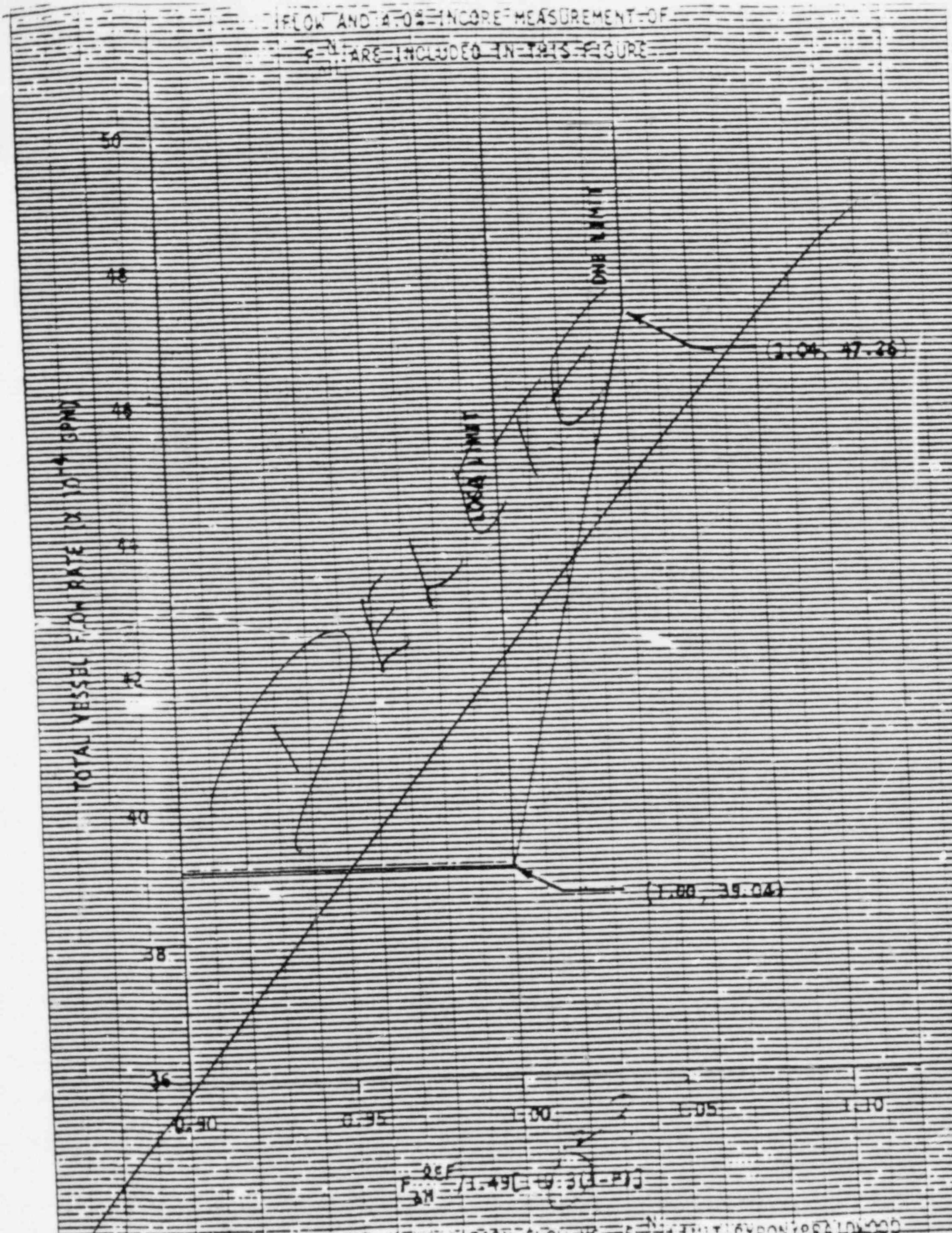
a. RCS Total Flowrate $\geq$ 386,000 GPM

b. $F^N_{\Delta H} \leq 1.55\,[1.0 + 0.3(1.0 - P)]$

where:

Measured values of $F^N_{\Delta H}$ are obtained by using the movable incore detectors to obtain a power distribution map, and

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}.$$



New Page 3/4 2-9

Figure 3.2-3 N-Loop Operation vs. $F^N_{\Delta H}$ Limit

RCS Total Flow Rate vs. R

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate ~~comparison~~ determination that ~~the combination of R~~ $F^N_{\Delta H}$ and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours; and

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that ~~the combination of R~~ $F^N_{\Delta H}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate ~~comparison~~ determination, to be ~~within the region of~~ acceptable ~~operation shown on Figure 3.2-3~~ prior to exceeding the following THERMAL POWER levels:

   1. A nominal 50% of RATED THERMAL POWER,

   2. A nominal 75% of RATED THERMAL POWER, and

   3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 ~~The combination of~~ indicated RCS total flow rate and ~~R~~ $F^N_{\Delta H}$ shall be determined to be ~~within the region of~~ acceptable~~ operation of Figure 3.2-3~~:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be ~~within the region of~~ acceptable ~~operation of Figure 3.2-3~~ at least once per 12 hours when ~~the most recently obtained value of R, obtained per Specification 4.2.3.2, is assumed to exist.~~ OUTSIDE THE ABOVE LIMITS

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months.

ATTACHMENT C
(Section 3/4.4)

Circled items in this attachment have been previously submitted.

1) Section 3/4.4.2.1 and 3/4.4.2.2 (pg. 3/4 4-7,8) Safety Valves and Shutdown

The starred (*) item of Technical Specification 3.4.2.1 and 3.4.2.2 needs to be deleted based upon the FSAR response to NRC Q212.116 which is attached and describes the approved ASME Section XI method of testing. The station will bench test the valves using nitrogen at the ambient temperature. This practice is consistent with the majority of the Nuclear Power Industry. Bench testing of relief valves in this manner is consistent with ASME Section XI requirements.

2) Section 3/4.4.10 (pg. 3/4 4-38) Structural Integrity

Delete the words "each reactor coolant pump flywheel....Revision 1, August 1975." Also, insert the following:

"4.4.10.b

1. In-place ultrasonic volumetric examination of the areas of higher stress concentration at the bore and key ways will be performed each 40 month period during refueling or maintenance shutdowns coinciding with the inservice inspection schedule as required by Section XI of the ASME Code.

2. Visual examination of all exposed surfaces will be performed each 40 month period and a surface examination of the bore and keyway surfaces will be performed whenever the flywheels are removed for maintenance purposes, but not more frequently than once each 10 year interval."

The requirements for examination procedures and acceptance criteria as described in the Regulatory Guide 1.14, Revision 1, August 1975 will be followed.

The proposed change is consistent with our response to Question 251.7 of the FSAR.

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3/4.4.2 SAFETY VALVES

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.1 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5.

~~*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.~~

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SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.2 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%. ~~*~~

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling ~~MODE~~ mode.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

~~*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.~~

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REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature ~~more than 50°F above the minimum temperature required by NDT considerations.~~ above 200°F.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.A In addition to the requirements of Specification 4.0.5, ~~each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.~~

4.4.10.b Each reactor coolant pump flywheel shall be inspected as follows:

1) In-place ultrasonic volumetric examination of the areas of higher stress concentration at the bore and key ways will be performed each 40 month period during refueling or maintenance shutdowns coinciding with the in service inspection schedule as required by Section XI of the ASME Code.

2) Visual examination of all exposed surfaces will be performed each 40 month period and a surface examination of the bore and keyway surfaces will be performed whenever the flywheels are removed for maintenance purposes, but not more frequently than once each 10 year interval.

QUESTION 212.116

"Recent operating experience has indicated that relief valve setpoints may be temperature sensitive. Discuss this effect on Byron/Braidwood relief valves."

RESPONSE

Changes to relief valve setpoints due to temperature variations are understood and have been considered. Temperature changes affect the spring rate of the valve spring, reducing the setpoint as the temperature increases. Normal ambient air temperature variations do not significantly affect the setpoint. However, when a cold valve relieves hot fluid, the setpoint can be reduced. This effect has been considered in the design of the valves and fluid systems.

ATTACHMENT D
(Section 3/4.6)

1) Section 3.6.4.2 (pg. 3/4 6-24) Electric Hydrogen Recombiners.

The words "power setting" and "power" are changed to "temperature controller" and "setting" as there is no power setting on the recombiners installed at Byron, only temperature controllers.

CONTAINMENT SYSTEMS

ELECTRIC HYDROGEN RECOMBINERS

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LIMITING CONDITION FOR OPERATION

3.6.4.2 Two independent Hydrogen Recombiner Systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2

ACTION:

With one Hydrogen Recombiner System inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each Hydrogen Recombiner System shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying, during a Recombiner System functional test that the minimum heater sheath temperature increases to greater than or equal to 1200°F within 90 minutes. Upon reaching 1200°F, increase the ~~power setting~~ to maximum ~~power~~ for 2 minutes and verify that the power meter reads greater than or equal to 38 kW, and
temperature controller setting

b. At least once per 18 months by:

1) Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,

2) Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiners enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and

3) Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

ATTACHMENT E
(Section 3/4.7)

Circled items noted in this attachment have been previously submitted.

1) Section 3/4.7.10.1(a) (pg. 3/4 7-28) Fire Suppression Water System.

-Delete the words "each with a capacity of 2500 gpm".

Section 3/4.7.10.1(b)

-Delete the words "Deluge or".

Section 4.7.10.1.1 F(1) (pg. 3/4 7-29)

Delete "pump develops at least 2500 gpm at a system head of 347 feet (150 psig)"

Insert "fire suppression discharge performance with a capacity of 3750 ± 10% gpm at 98 ± 10% psig.

The proposed change incorporates results of an NRC audit requesting Technical Specifciations to be updated to reflect NFPA requirements. The 150% rated capacity is mentioned in NFPA-20-Centrifugal Fire Pumps (See attachment).

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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~~ System required to be OPERABLE per Specifications 3.7.10.2 and 3.7.10.5.

FOAM

APPLICABILITY: At all times.

ACTION:

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

b. With the Fire Suppression Water 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,

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SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 6 months by performance of a system ring header flush,

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

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

   1) Verifying that each fire suppression discharge performance with ~~pump develops at least 2500 gpm at a system head of 347 feet (150 psig),~~ a capacity of 3750 ± 10% gpm at 98 ± 10% psig.

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

   3) Verifying that each fire suppression pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 125 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 8, Section 16 of the Fire Protection Handbook, 15th Edition, published by the National Fire Protection Association.

4.7.10.1.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

   1) The fuel storage tank contains at least 325 gallons of fuel, and

   2) The diesel starts from ambient conditions and operates for at least 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel oil 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 procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

A-2-1.4 Water sources containing salt or other materials deleterious to the fire protection systems should be avoided.

A-2-3.1 A centrifugal fire pump should be selected in the range of operation from 90 percent to 150 percent of its rated capacity. The performance of the pump when applied at capacities over 140 percent of rated capacity may be adversely affected by the suction conditions. Application of the pump at capacities less than 90 percent of the rated capacity is not recommended.

The selection and application of the fire pump should not be confused with pump operating conditions. With proper suction conditions, the pump can operate at any point on its characteristic curve from shutoff to 150 percent of its rated capacity.

A-2-7 Some locations or installations may not require a pump house. When a pump room or pump house is required, it should be of ample size and located to permit short and properly arranged piping. The suction piping should receive first consideration. The pump house should preferably be a detached building of noncombustible construction. A one-story pump room with a combustible roof, either detached or well cut off from an adjoining one-story building, is acceptable if sprinklered. When a detached building is not feasible, the pump room should be so located and constructed as to protect the pump unit and controls from falling floors or machinery, and from fire that might drive away the pump operator or damage the pump unit or controls. Access to the pump room should be provided from outside the building. Where the use of brick or reinforced concrete is not feasible, metal lath and plaster is recommended for the construction of the pump room. The pump room or pump house should not be used for storage purposes. Vertical shaft turbine-type pumps may require a removable panel in the pump house roof to permit the pump to be removed for inspection or repair.

A-2-7.1 Impairment. A fire pump which is inoperative for any reason at any time constitutes an impairment to the fire protection system. It should be returned to service without delay.

A-2-7.6 Pump rooms and pump houses should be dry and free of condensate. Some heat may be required to accomplish this.

A-2-8.1 The exterior of aboveground steel piping should be kept painted.

A-2-9.1 The exterior of steel suction piping should be kept painted.

ATTACHMENT F
(Section 3/4.8)

Circle items noted in this attachment have been previously submitted.

1) Section 3/4.8.3 (pg. 3/4 8-14) Onsite Power Distribution

   Delete "(both) between redundant busses within the unit (and".

   These tie breakers do not exist at Byron Station.

2) Section 3/4.8.4 (pg. 3/4 8-18) Electrical Equipment Protective Devices

   Delete the previous request to add "125 VOLT-DC". At this voltage level the cable would fail before damage to the penetration could occur.

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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.

DC

1, 2

ATTACHMENT G
(Section 3/4.9)

Circled items noted in this attachment have been previously submitted.

1) Section 3/4.9.6 (pg. 3/4 9-6) Manipulator Crane.

-Item b(1), previously a change was submitted to change the rated capacity of the auxiliary hoist from 3000 pounds to 2500 pounds. The actual number for the auxiliary hoist rated capacity is 2000 pounds per Manipulator Crane Stearns Roger Incorporated vendor manual.

-Delete "Refueling Machine" and insert "Manipulator Crane". This is correct Byron Station nomenclature.

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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.

2) By selecting and functionally testing a representative sample of at least 10% of each type of 480-volt ~~125 VOLT-DC~~ 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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3/4.9.6 ~~REFUELING MACHINE~~ Manipulator Crane

LIMITING CONDITION FOR OPERATION

3.9.6 The ~~refueling machine~~ manipulator crane shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The ~~refueling machine~~ manipulator crane used for movement of fuel assemblies having:

1) A ~~minimum~~ rated capacity of ~~3250~~ 2850 pounds, and

2) An overload cutoff limit less than or equal to 2850 pounds.

b. The auxiliary hoist used for latching and unlatching drive rods having:

1) A ~~minimum~~ rated capacity of ~~3000~~ ~~2500~~ 2000 pounds, and

2) A load indicator which shall be used to prevent lifting loads in excess of 1000 pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisified, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least ~~3250~~ 3563 pounds and demonstrating an automatic load cutoff when the crane load exceeds 2850 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor vessel shall be demonstrated OPERABLE within 100 hours ~~prior~~ to the start of such operations by performing a load test of at least ~~3000~~ 2500 pounds.

ATTACHMENT H
(Bases Section 3/4)

Circled items noted in this attachment have been previously submitted.

1) Section 3/4.2.2 (pg. B3/4 2-4) Heat Flux Hot Channel Factor, and RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor.

   Delete the following sentences: "As noted on Figure 3.2-3, RCS Flow rate....will not be below the design DNBR value." Also, "R as calculated in Specification 3.2.3....is the maximum "as measured" value allowed".

2) Section 3/4.2.3 (pg B3/4 2-5).

   -Delete the sentence "When RCS flow rate....with the limits of Figure 3.2.3".

   -Insert the words "following initial plant startup" between the words "Therefore" and "a penalty".

   -Delete the words "is included in Figure 3.2-3" and replace with "as well as a measurement error of 2.1% have been included in the limiting value of 386,000 gpm for RCS total flow rate. The thermal design flow of 377,600 gpm increased for measurement error and penalized for feedwater venturi fouling is 385,915 gpm."

3) Section 3/4.7.10 (pg. B3/4 7-7).

   Add the words "Pump capacity is based on NFPA 20 1984 which calls for 150% flow at 65% discharge pressure." after the second paragraph.

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

c. The control rod insertion limits of Specification 3.1.3.6 are maintained, and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. ~~As noted on Figure 3.2-3, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value.~~ The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

~~R as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.~~

INSERT "B"

Fuel rod bowing reduces the value of DNB ratio. Credit is available to partially offset this reduction. This credit comes from a generic design margin which totals 9.1% when the analysis is performed with the approved interim methods. The margin used to partially offset rod bow penalties is 9.1%. This margin breaks down as follows:

1)	Design limit DNBR	(1.6)%
2)	Grid spacing Ks	(2.9)%
3)	Thermal Diffusion Coefficient	(1.2)%
4)	DNBR multiplies	(1.7)%
5)	Pitch Reduction	(1.7)%

The margin used to partially offset rod bow penalities is (5.9)% with the remaining (3.2)% used to trade off against measured flow which may be as much as (2)% lower than thermal design flow plus uncertainties. The penalties applied to $F^N_{\Delta H}$ to account for rod bow as a function of burnup are consistent with those described in Mr. John F. Stolz's (NRC) letter to T. M. Anderson (Westinghouse) dated April 5, 1979 with the difference being due to the amount of margin each unit uses to partially offset rod bow penalities.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

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BASES

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

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per Specification 6.7.1.9 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

~~When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-3.~~ Measurement errors of 2.1% for RCS total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

as well as a measurement error of 2.1% have been included in the limiting value of 386,000 gpm for RCS total flow rate. The thermal design flow of 377,600 gpm increased for measurement error and penalized for feedwater venturi fouling is 385,915 gpm.

following initial plant startup,

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the results to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi ~~is included in Figure 3.2-3.~~ Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The ~~12~~-hour periodic surveillance of indicated RCS flow is sufficient to detect [redacted] flow degradation which could lead to operation outside the acceptable region of operation. ~~shown on Figure 3.2-3.~~

3/4.2.4 QUADRANT POWER TILT RATIO

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

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

The 2 hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such ACTION does

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3/4.7.10 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the Fire Suppression Systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The Fire Suppression System consists of the water system, spray, and/or sprinklers, CO_2, Halon, and fire hose stations. The collective capability of the Fire Suppression Systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility Fire Protection Program.

In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire-fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire-fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression. Pump capacity is based on NFPA 20, 1984 which calls for 150% flow at 65% discharge pressure.

The Surveillance Requirements provide assurance that the minimum OPERABILITY requirements of the Fire Suppression Systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying either the weight or the level of the tanks. Level measurements are made by either a U.L. or F.M. approved method.

In the event the Fire Suppression Water System becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

3/4.7.11 FIRE RATED ASSEMBLIES

The functional integrity of the fire rated assemblies and barrier penetrations ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. These design features minimize the possibility of a single fire rapidly involving several areas of the facility prior to detection of and the extinguishing of the fire. The fire barrier penetrations are a passive element in the facility fire protection program and are subject to periodic inspections.

Fire barrier penetrations, including cable penetration barriers, fire doors and dampers are considered functional when the visually observed condition is the same as the as-designed condition. For those fire barrier penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.

During periods of time when a barrier is not functional, either: (1) a continuous fire watch is required to be maintained in the vicinity of the affected barrier, or (2) the fire detectors on at least one side of the affected barrier must be verified OPERABLE and an hourly fire watch patrol established, until the barrier is restored to functional status.

ATTACHMENT I
(Section 5.0)

1) Section 5.3.1 (pg. 5-4) Fuel Assemblies.

Delete the number "1748" and replace with "1594". This new number represents the maximum total gram weight of the uranium in each Optimized Fuel Assembly (OFA) rod as recommended by Westinghouse.

2) Section 5.3.2 (pg. 5-4) Control Rod Asemblies.

Delete the words "full-length and no part-length" and "full-length" from this paragraph. Byron Station does not utilize part-length control rods and it is therefore unnecessary to reference full-length or part-length when discussing control rod assemblies.

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5.3 REACTOR CORE

FUEL ASSEMBLIES

1594

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of ~~1748~~ grams uranium. The initial core loading shall have a maximum enrichment of 3.10 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.50 weight percent U-235.

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 587.4°F.

5.5 METEOROLOGICAL TOWER LOCATION

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

ATTACHMENT J
(Section 6.0)

Circled items noted in this attachment have been previously submitted.

1). Section 6.7.1.9 (pg. 6-20) Radial Peaking Factor Limit Report.

Delete the following words from this section:

a. "NRC Regional Administrator with a copy to"

b. "Attention: Chief, Core Performance Branch"

c. "Washington, D.C. 20555"

d. "at least 60 days"

Also, insert the words "with the reload analysis submitted or technical specificaton change" between the words "criticality" and "unless otherwise approved".

2). Section 6.7.1.9 (pg. 6-21) Radial Peaking Factor Limit Report.

Remove the words "submittal or an.... Factor Limit Report" and insert "reviewed by the commission". Also, delete the words "60 days".

The above Section 6.7.1.9 changes are consistent with the reporting method used by the office stations at Commonwealth Edison.

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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.

The ~~Semiannual Radioactive Effluent Release~~ Annual Radiological Environmental Operating 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.

The ~~Semiannual Radioactive Effluent Release Reports~~ Annual Radiological Environmental Operating Report 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^{RTP}_{xy}) 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 60 days~~ prior to each cycle initial criticality, unless otherwise approved by the Commission by letter. In addition.
with the reload analysis submittal or technical specification change.

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RADIAL PEAKING FACTOR LIMIT REPORT (Continued)

In the event that the limit should change requiring a new review by the Commission. ~~submittal or an amended submittal to the Peaking Factor Limit Report,~~ it shall be submitted ~~60 days~~ prior to the date the limit would become effective unless otherwise approved by the Commission by letter. Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.7.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

6.8 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.8.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.8.2 The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories;