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April 9, 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.

Dear Mr. Denton:

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

Attachments A through 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) and (c). 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 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 paragraph beginning "The minimum value of the DNBR. . . for all operating conditions." and replace with "The DNB design basis is as follows: There must be at least 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent confidence that the minimum DNBR for the limiting rods is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analysis using values of input parameters without uncertainties."

2. Section 2.2.1 (pg. B2-5) Reactor Trip System Instrumentation Setpoints

In the first paragraph, delete "1.30" and replace with "the DNBR limit".

Per discussions between Westinghouse and CECo, the above changes are being made in order to better clarify the DNBR limit as it applies to Byron Station.

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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 curve~~s~~ of Figure~~s~~ 2.1-1 ~~and 2.1-2~~ shows the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.34 for a typical cell and 1.32 for a thimble cell, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of ~~1.55~~ 1.49 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:

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

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

Insert "A"

The DNB design basis is as follows: there must be be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 confidence that the minimum DNBR for the limiting rods is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analysis using values of input parameters without uncertainties.

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Power Range, Neutron Flux, High Rates (Continued)

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than ~~1.30.~~ the DNBR limit.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide core protection during reactor STARTUP to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

ATTACHMENT B

(Section 3/4.3)

Circled items noted in this attachment have been previously submitted.

1. Table 3.3.13 and Table 4.3-9 (pg. 3/4 3-67, 71) Radioactive Gaseous Effluent Monitoring Instrumentation

   Deletion of 4b. Flow Rate Monitor, 4c. Sampler Flow Rate Monitor, 5d. System Flow Rate Monitor and 5e. Sampler Flow Rate Monitor are requested because these monitors are not needed to comply with the Technical Specifications. The only monitors required are vent stack monitors. These instruments are not required to be operable for Noble Gas Activity Monitors.

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

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

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TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4 ~~3.~~ Gas Decay Tank System					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (0RE-PR002A and 2B)	P Delete	P	R(3)	Q(1)	*
b. ~~Flow Rate Monitor~~ (0FIT-GW001)	P	N.A.	R	Q	*
c. ~~Sampler Flow Rate~~ Monitor (0FI-PR159)	D	N.A.	R	Q	*
5 ~~4.~~ Containment Purge System					
a. Noble Gas Activity Monitor - Providing Alarm (1RE-PR001~~A~~B)	D	P	R(3)	Q(2)	*
b. Iodine ~~Monitor~~ Sampler (1RE-PR001C)	P	P	R(3)	N.A.	*
c. Particulate ~~Monitor~~ Sampler (1RE-PR001~~B~~A)	P	P	R(3)	N.A.	*
d. ~~System Flow Rate~~ Monitor (1FT-VQ043)	D	N.A.	R	Q	*
e. ~~Sampler Flow Rate~~ Monitor (1FI-PR100)	D	N.A.	R	Q	*

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ATTACHMENT C

(Section 3/4.4)

Circled items noted in this attachment have been previously submitted.

1. Section 3/4.4.6 (pg. 3/4 4-18) Action Statement

   Delete the existing "Action" statement and replace with the following:

   a. With "a" of the above required Leakage Detection System inoperable, grab samples of the containment atmosphere must be obtained and analyzed for gaseous and particulate radioactivity at least once per 24 hours.

   b. With only two of the above required Leakage Detection Systems Operable, operation may continue for up to 30 days; otherwise, be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours."

   This change is for clarification of the existing statement.

2. Section 3/4.4.6 (pg. 3/4 4-18) Reactor Coolant System Leakage

   In Surveillance 4.4.6.1.d delete "channel check" and "and Analog Channel Operational Test". These two surveillances do not apply to the containment air pressure and reactor fan cooler outlet and inlet temperature - performance.

3. Section 3/4.4.10 (pg. 3/4 4-38) Action a.

   Delete the words "more than 50°F above the minimum temperature required by NDT considerations" and add "above 200°F."

   The addition of this 200°F limit would prevent the plant from entering Mode 5 with a Class 1 component not conforming to ASME Section XI.

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3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

ACTION:

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

Insert "XX"

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

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

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

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

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

INSERT "XX"

a. With "a" of the above required Leakage Detection System inoperable, grab samples of the containment atmosphere must be obtained and analyzed for gaseous and particulate radioactivity at least once per 24 hours.

b. With only two of the above required Leakage Detection Systems Operable, operation may continue for up to 30 days; otherwise, be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

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3/4.4.10 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 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.

ATTACHMENT D

(Section 3/4.5)

Circled items noted in this attachment have been previously submitted.

1. Section 3/4.5.1 (pg. 3/4 5-1) Accumulators

   A contained borated water level of 34% has been changed to 64%. This agrees with the gallons quoted.

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

   Add an "asterisk" following the word "inoperable" on the second line in the paragraph. Also, add the following note and asterisk to the bottom of the page.

   "* It may be desirable to operate the SI pumps for testing and also for filling the accumulator tanks while in mode 4. It is permissible to operate the SI pumps if the appropriate valves are locked closed to prevent pressurization of the RCS. To run the A-Pump, valves SI8802A and SI8821A must be locked closed. The B-Pump may be used by locking closed valves SI8835, SI8802A and SI8802B with valves SI8821A and SI8821B open. When the accumulators are being filled block valves SI8808A, SI8808B, SI8808C and SI8808D must also be locked closed."

   Although this section requires that both safety injection pumps be demonstrated inoperable in Mode 4, it may be desirable to operate the SI pumps for testing and also for filling the accumulator tanks while in Mode 4. The above justifies the operation of the SI pumps.

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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water level of between ~~34%~~ 64% (6995 gallons) and 66% (7217 gallons),

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

d. A nitrogen cover-pressure of between ~~600~~ 617 and ~~650~~ 662 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

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

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

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SURVEILLANCE REQUIREMENTS

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

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

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

ATTACHMENT E

(Section 3/4.8)

1. Section 4.8.1.1.2.d (pg. 3/4 8-3) A.C. Sources Surveillance

   Delete "obtained in accordance with ASTM-D270-1975"

   Reference to ASTM-D270-1975 has been deleted. The tank does not have sample taps located at the top, middle and bottom as specified in ASTM-D270-1975. Therefore, this part of the surveillance cannot be met with the present tank design.

2. Section 3/4.8.4 (pgs. 3/4 8-17 through 8-33) Electrical Equipment Protective Devices.

   <u>Justification</u>

   The TRIP SETPOINT and RESPONSE TIME columns are being deleted from the technical specifications Table 3.8-1 because the values shown are not representative of the data which should be conveyed to meet the intent of the technical specification Section 4.8.4.1. The data necessary to meet the intent of Section 4.8.4.1 consists of a definition of the acceptable functional criteria for all of the overcurrent protection devices such that they conform with the overcurrent protection design to maintain containment penetration integrity.

   The information necessary to describe the acceptable functional criteria cannot be presented by a single trip setpoint and response time. Due to the nature of overcurrent protection devices, the operability limits can best be defined by a curve.

   CECo therefore contends that the deletion of the TRIP SETPOINT and RESPONSE TIME information will be less ambigious than trying to express either a single data point or a technical description for the designed response of the various overcurrent protective devices.

   CECo will perform the testing of the overcurrent protection devices in compliance with approved engineering procedures and vendor recommendations in order to prove that the devices remain within their operable limits as described in FSAR Q.40.13.

   Also in Table 3.8-1 the information concerning the 125V D.C. feed through a penetration has been deleted. Upon investigation it was found that the overcurrent capacity of the D.C. feed cable to the penetration, is far less than the fault current capacity of the penetration. If a fault condition appeared on this circuit and the protective device failed to operate, the feed cable to the penetration for this circuit would rail before the integrity of the penetration was impaired. On this basis we propose the deletion of the 125VDC Pnl 114 circuit listed on page 3/4.8-33.

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 $\pm$ 420 volts and 60 $\pm$ 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:

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;

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;

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3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 18 months:

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

a) A CHANNEL CALIBRATION of the associated protective relays,

b) An integrated system functional test which includes simulated automatic actuation of the system ~~and verifying that each relay and associated circuit breakers and control circuits function as designed and as specified in Table 3.8-1, and~~ to demonstrate that the overall penetration protection design remains within operable limits.

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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 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.

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	DEVICE	TRIP SETPOINT (Amperes)	RESPONSE TIME (Sec/Cycle)
1. 6.9 kV Switchgear			
1RC01PA-RCPA Bus 157 Cub 1	Primary	Long time - 1440x2.1 Inst. - 7680	11.5 N.A.
Bus 157 Norm. Feed ACB 1577	Backup	Long time - 4800x2 Gr. - 200	0.9 0.3
Bus 157 Emerg. Feed ACB 1572	Backup	Long time - 4800x2 Inst. - 7680	0.9 N.A.
1RC01PB-RCPB Bus 156 Cub 2	Primary	Long time - 1400x2.1 Inst. - 7680	11.5 N.A.
Bus 156 Norm. Feed ACB 1566	Backup	Long time - 4800x2 Gr. - 200	0.7 0.3
Bus 156 Emerg. Feed ACB 1562	Backup	Long time - 4800x2 Gr. - 200	0.7 0.3
1RC01PC RCPC Bus 158 Cub 5	Primary	Long time - 1440x2.1 Inst. - 7680	11.5 N.A.
Bus 158 Norm. Feed ACB 1582	Backup	Long time - 4800x2 Gr. - 200	0.7 0.3
Bus 158 Emerg. Feed ACB 1587	Backup	Long time - 4800x2 Gr. - 200	0.7 0.3

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	DEVICE	TRIP SETPOINT (Amperes)	RESPONSE TIME (Sec/Cycle)
1. 6.9 kV Switchgear (Continued)			
1RC01PD - RCPD Bus 159 Cub 5	Primary	Long time - 1440x2.1 Inst. - 7680	11.5 N.A.
Bus 159 Norm. Feed ACB 1591	Backup	Long time - 4800x2 Gr. - 200	0.7 0.3
Bus 159 Emerg. Feed ACB 1597	Backup	Long time - 4800x2 Gr. - 200	0.7 0.3
2. 480V Switchgear			
1RY03EA - Pzr. Htr. Backup Group A	Primary	MCCB - 100	N.A.
Compt. A1-A6, B1	Backup	MCCB - 100	N.A.
1RY03EB - Pzr. Htr. Backup Group B	Primary	MCCB - 100	N.A.
Compt. B1-B6, A1	Backup	MCCB - 100	N.A.
1RY03EC - Pzr. Htr. Backup Group C	Primary	MCCB - 100	N.A.
Compt. A1-A6, B1	Backup	MCCB - 100	N.A.
1RY03ED - Pzr. Htr. Backup Group D	Primary	MCCB - 100	N.A.
Compt. B1-B6, A1	Backup	MCCB - 100	N.A.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	DEVICE	TRIP SETPOINT (Amperes)	RESPONSE TIME (Sec/Cycle)
3. 480V A.C. Ckt. Bkrs.			
1VP01CA - RCFC Fan 1A Low Speed Feed Bkr Swgr 131X Cub 4C	Primary	Long time - 450 Inst. - 4,500	20-32 N.A.
Hi Speed Feed Bkr Swgr 131X Cub 5C	Primary	Long time - 900 Inst. - 7,500	20-32 N.A.
Bus 131X Norm. Feed 141 Swgr., Cub 14, ACB 1415	Backup	Long time - 960 Inst. - 3,960	3.4 N.A.
1VP01CC - RCFC Fan 1C Low Speed Feed Bkr Swgr 131X Cub 4C	Primary	Long time - 450 Inst. - 4,500	20-32 N.A.
Hi Speed Feed Bkr Swgr 131X Cub 5C	Primary	Long time - 900 Inst. - 7,500	20-32 N.A.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	DEVICE	TRIP SETPOINT (Amperes)	RESPONSE TIME (Sec/Cycle)
3. 480V A.C. Ckt. Bkrs. (Continued)			
1VP01CB - RCFC Fan 1B			
Low Speed Feed Bkr Swgr 132X Cub 4C	Primary	Long time - 450 Inst. - 4,500	20-32 N.A.
Hi Speed Feed Bkr Swgr 132X Cub 5C	Primary	Long time - 900 Inst. - 7,500	20-32 N.A.
Bus 132X Norm. Feed 142 Swgr., Cub 14, ACB 1423	Backup	Long time - 960 Inst. - 3,960	3.4 N.A.
1VP01CD - RCFC Fan 1D			
Low Speed Feed Bkr Swgr 132X Cub 2C	Primary	Long time - 450 Inst. - 4,500	20-32 N.A.
Hi Speed Feed Bkr Swgr 132X Cub 3C	Primary	Long time - 900 Inst. - 7,500	20-32 N.A.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	DEVICE	TRIP SETPOINT (Amperes)	RESPONSE TIME (Sec/Cycle)
4. 480V Molded Case Ckt. Bkts. (MCCB)			
	MCC 133x4		
1RC01PA-A Cub B1	Primary Backup	15 15	N.A. N.A.
1RC01PA-B Cub B2	Primary Backup	40 40	N.A. N.A.
1HC22G Cub B3	Primary Backup	15 15	N.A. N.A.
1FH036 Cub B4	Primary Backup	15 15	N.A. N.A.
1VP05CA Cub C1	Primary Backup	30 30	N.A. N.A.
1RF03P Cub C2	Primary Backup	30 30	N.A. N.A.
1RC01PD-A Cub D1	Primary Backup	15 15	N.A. N.A.
1RC01PD-B Cub D2	Primary Backup	40 40	N.A. N.A.
1RF02PB Cub D4	Primary Backup	30 30	N.A. N.A.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	DEVICE	TRIP SETPOINT (Amperes)	RESPONSE TIME (Sec/Cycle)
4. 480V Molded Case Ckt. Bkts. (MCCB) (Continued)			
	MCC 133x4		
1RF01P	Primary	15	N.A.
Cub D5	Backup	15	N.A.
1RE01PA	Primary	40	N.A.
Cub D6	Backup	40	N.A.
1VP02CA	Primary	40	N.A.
Cub E1	Backup	40	N.A.
1VP04CA	Primary	125	N.A.
Cub E2	Backup	125	N.A.
1VP04CC	Primary	125	N.A.
Cub F1	Backup	125	N.A.
1EW11EA	Primary	125	N.A.
Cub F3	Backup	125	N.A.
1EW11EB	Primary	125	N.A.
Cub F3	Backup	125	N.A.
1EW11EC	Primary	125	N.A.
Cub F3	Backup	125	N.A.
1IC02EA	Primary	20	N.A.
Cub F5	Backup	20	N.A.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	DEVICE	TRIP SETPOINT (Amperes)	RESPONSE TIME (Sec/Cycle)
4. 480V Molded Case Ckt. Bkts. (MCCB) (Continued)			
	MCC 133x4		
1IC02EB	Primary	20	N.A.
Cub G1	Backup	20	N.A.
1IC02EC	Primary	20	N.A.
Cub G2	Backup	20	N.A.
1IC02EF	Primary	30	N.A.
Cub A1	Backup	30	N.A.
1IC02EE	Primary	30	N.A.
Cub A2	Backup	30	N.A.
1IC02ED	Primary	30	N.A.
Cub A3	Backup	30	N.A.
1FH02J	Primary	15	N.A.
Cub G1	Backup	15	N.A.
1FH03J	Primary	15	N.A.
Cub G2	Backup	15	N.A.
1RC01PB-B	Primary	40	N.A.
Cub B1	Backup	40	N.A.
1RE01PB	Primary	70	N.A.
Cub B3	Backup	70	N.A.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	DEVICE	TRIP SETPOINT (Amperes)	RESPONSE TIME (Sec/Cycle)
4. 480V Molded Case Ckt. Bkts. (MCCB) (Continued)			
	MCC 134x5		
1RC01PC-A	Primary	15	N.A.
Cub C1	Backup	15	N.A.
1RC01PC-B	Primary	40	N.A.
Cub C2	Backup	40	N.A.
1VP05CB	Primary	30	N.A.
Cub D1	Backup	30	N.A.
1RC01PB-A	Primary	15	N.A.
Cub C3	Backup	15	N.A.
1HC656-A	Primary	40	N.A.
Cub D3	Backup	40	N.A.
1VP02CB	Primary	40	N.A.
Cub F1	Backup	40	N.A.
1RC01R-A	Primary	15	N.A.
Cub F2	Backup	15	N.A.
1RF02PA	Primary	30	N.A.
Cub G3	Backup	30	B.A.
1EW12EA	Primary	125	N.A.
Cub F3	Backup	125	N.A.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	DEVICE	~~TRIP SETPOINT (Amperes)~~	~~RESPONSE TIME (Sec/Cycle)~~
4. 480V Molded Case Ckt. Bkts. (MCCB) (Continued)			
	MCC 134x5		
1EW12EB	Primary	125	N.A.
Cub F3	Backup	125	N.A.
1EW12EC	Primary	125	N.A.
Cub F3	Backup	125	N.A.
1VP04CB	Primary	125	N.A.
Cub F4	Backup	125	N.A.
1VP04CD	Primary	125	N.A.
Cub F5	Backup	125	N.A.
1SI8808C	Primary	70	N.A.
Cub A2	Backup	70	N.A.
1SI8808B	Primary	70	N.A.
Cub A3	Backup	70	N.A.
1RH8702B	Primary	15	N.A.
Cub B1	Backup	15	N.A.
1RH8701B	Primary	15	N.A.
Cub B3	Backup	15	N.A.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	DEVICE	~~TRIP SETPOINT (Amperes)~~	~~RESPONSE TIME (Sec/Cycle)~~
480V Molded Case Ckt. Bkts. (MCCB) (Continued)			
	MCC 132x2		
1CV8112	Primary	5	N.A.
Cub B4	Backup	5	N.A.
1OG079	Primary	15	N.A.
Cub C1	Backup	15	N.A.
1WO056A	Primary	5	N.A.
Cub C2	Backup	5	N.A.
1OG080	Primary	15	N.A.
Cub C3	Backup	15	N.A.
1RY8000B	Primary	15	N.A.
Cub C4	Backup	15	N.A.
1RY8003C	Primary	15	N.A.
Cub C5	Backup	15	N.A.
1IP06E	Primary	20	N.A.
Cub E1	Backup	20	N.A.
1RC8003B	Primary	15	N.A.
Cub D4	Backup	15	N.A.
1LL43J	Primary	70	N.A.
Cub E2	Backup	70	N.A.
1RC8002A	Primary	40	N.A.
Cub G1	Backup	40	N.A.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	DEVICE	~~TRIP SETPOINT (Amperes)~~	~~RESPONSE TIME (Sec/Cycle)~~
4. 480V Molded Case Ckt. Bkts. (MCCB) (Continued)			
	MCC 132x2		
1RC8002B Cub G2	Primary	40	N.A.
	Backup	40	N.A.
1RC8002C Cub G3	Primary	40	N.A.
	Backup	40	N.A.
1RC8002D Cub G4	Primary	40	N.A.
	Backup	40	N.A.
	MCC 131x2A		
1SI8808D Cub A2	Primary	70	N.A.
1AP25E-A MCC 131x2 Cub B2	Backup	125	N.A.
1SI8808A Cub A3	Primary	70	N.A.
1AP25E-A MCC 131x2 Cub B2	Backup	125	N.A.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	DEVICE	~~TRIP SETPOINT (Amperes)~~	~~RESPONSE TIME (Sec/Cycle)~~
4. 480V Molded Case Ckt. Bkts. (MCCB) (Continued)			
	MCC 131x2		
1RC8001A Cub	Primary	40	N.A.
	Backup	40	N.A.
1RC8001B Cub	Primary	40	N.A.
	Backup	40	N.A.
1RC8001C Cub	Primary	40	N.A.
	Backup	40	N.A.
1RC8001D Cub	Primary	40	N.A.
	Backup	40	N.A.
1RH8701A Cub	Primary	15	N.A.
	Backup	15	N.A.
1RH8702A Cub	Primary	15	N.A.
	Backup	15	N.A.
1LL42J Cub	Primary	70	N.A.
	Backup	70	N.A.
1VQ001A Cub	Primary	10	N.A.
	Backup	10	N.A.
1VQ002A Cub	Primary	10	N.A.
	Backup	10	N.A.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	DEVICE	~~TRIP SETPOINT (Amperes)~~	~~RESPONSE TIME (Sec/Cycle)~~
4. 480V Molded Case Ckt. Bkts. (MCCB) (Continued)			
	MCC 131x2		
1RC8003D Cub	Primary Backup	15 15	N.A. N.A.
1RC8003A Cub	Primary Backup	15 15	N.A. N.A.
1OG057A Cub	Primary Backup	15 15	N.A. N.A.
1CC9416 Cub	Primary Backup	15 15	N.A. N.A.
1CC9438 Cub	Primary Backup	15 15	N.A. N.A.
1OG081 Cub	Primary Backup	15 15	N.A. N.A.
	MCC 133x6		
1HC016 - Cub B2 Cub B1	Primary Backup	125 125	N.A. N.A.
1LL04E - Cub C3 Cub C1	Primary Backup	225 225	N.A. N.A.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	DEVICE	~~TRIP SETPOINT (Amperes)~~	~~RESPONSE TIME (Sec/Cycle)~~
480V Molded Case Ckt. Bkts. (MCCB) (Continued)			
		~~MCC 133x6~~	
1VP03CA	Primary	125	N.A.
Cub A3	Backup	125	N.A.
1VP03CD	Primary	125	N.A.
Cub C4	Backup	125	N.A.
	MCC 132x5		
1CC9414	Primary	5	N.A.
Cub B4	Backup	N.A.	N.A.
	MCC 134x7		
1LL05E	Primary	225	N.A.
Cub	Backup	225	N.A.
1VP03CB	Primary	125	N.A.
Cub	Backup	125	N.A.
1VP03CC	Primary	125	N.A.
Cub	Backup	125	N.A.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	DEVICE	~~TRIP SETPOINT (Amperes)~~	~~RESPONSE TIME (Sec/Cycle)~~
4. 480V Molded Case Ckt. Bkts. (MCCB) (Continued)			
	MCC 131x2B		
1W0056B	Primary	5	N.A.
Cub A1	Backup	5	N.A.
1RY8000A	Primary	10	N.A.
Cub A5	Backup	10	N.A.
	~~125 VDC Pnl 114 Sec. E~~		
~~1LL52J~~	~~Primary~~	70	N.A.
~~Cub 152~~	~~Backup~~	70	N.A.
5. 260 VAC RCD Power (53 rods, 5 panels)			
Stationary Gripper	Primary	10 - Fuse	N.A.
Coils (all panels)	Backup	10 - Fuse	N.A.
Lift Coils	Primary	50 - Fuse	N.A.
(all panels)	Backup	50 - Fuse	N.A.
Movable Gripper	Primary	10 - Fuse	N.A.
Coils (all panels)	Backup	10 - Fuse	N.A.

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ATTACHMENT F

(Section 3/4.9)

1. Section 3/4.9.6 (pg. 3/4 9-6) Refueling Machine

   In step 3.9.6.a.1; "minimum" capacity is changed to "rated" capacity and "3250" pounds to "2850" pounds.

   In step 3.9.6.b.1; "minimum" capacity is changed to "rated" capacity and "3000" pounds to "2500" pounds.

   Typically, a crane has a "rated" capacity and not a "minimum" capacity, thus the necessary change. The rated capacity of the manipulator crane was changed per Westinghouse letter CAW 6825. The rated capacity of the auxiliary hoist was changed to prevent the handling of heavy loads.

2. Section 4.9.6.1 (pg. 3/4 9-6) Surveillance Requirements

   Replace "3250" pounds with "3563" pounds. The manipulator crane should be tested at 125% of the rated capacity of 2850 pounds.

3. Section 4.9.6.2 (pg. 3/4 9-6) Surveillance Requirement

   Replace "3000" pounds with "2500" pounds. This was changed to reflect change in rated capacity in 3.9.6.b.1.

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3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

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

a. The refueling machine 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 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 G

(Section 3/4.11)

1. Section 3.11.1.4 (pg. 3/4 11-8) Liquid Holdup Tanks

The curie limit for the Primary Water Storage Tank (PWST) is based upon the maximum concentration in the PWST that, if released from the tank, will result in concentations not greater than the 10CFR20 MPC limits for unrestricted areas. For Byron, two cases are evaluated with the most conservative value chosen as the curie limit. Case 1 is the nearest potable water supply (a well 1960 feet East-South-East of the Aux. Bldg.). From the FSAR, Section 2.4.13.3 a dilution factor of 2200 is given with a transient time of 30.5 years.

Case 2 is the nearest surface water body (a spring 3630 feet Northwest of the Aux. Bldg.). From the FSAR, Section 2.4.13.1.1 a dilution factor of 655 is given with a transient time of 64.12 years.

The major source of activity input to the PWST is from the Recycle Evaporator Monitor Tank. To obtain the estimated activity in this tank, the Recycle Holdup Tank inventory (FSAR, Table 2.4-20) was used. Each isotopes concentration was divided by its evaporator decon factor (FSAR, Table 12.2-34) to obtain the Recycle Monitor Tank activity. This activity is assumed to be equivalent to the PWST activity.

Because of the transient times involved (as specified above) Cs-137 is the only isotope that has a long enough half-life to still be present in appreciable amounts after transient; thus, Cs-137 is the limiting isotope.

Isotope	Recycle Holdup Tank uCi/ml	Decon Factor	Recycle Monitor Tank uCi/ml
Rb-88	3.7E-2	10000	3.7E-6
RB-89	2.1E-3	10000	2.1E-7
Mo-99	5.3E-2	10000	5.3E-6
I-131	2.5E-2	1000	2.5E-5
I-132	2.8E-2	1000	2.8E-5
I-133	4.0E-2	1000	4.0E-5
I-134	5.6E-3	1000	5.6E-6
I-135	2.2E-2	1000	2.2E-5
Cs-134	2.3E-2	10000	2.3E-6
Cs-136	2.8E-2	10000	2.8E-14
Cs-137	1.5E-2	10000	1.5E-6
Cs-138	9.8E-3	10000	9.8E-7
Ba-137M	1.4E-2	10000	1.4E-6
			1.36E-4 uCi/ml

$$\% \ Cs137 = \frac{1.5E-6}{1.36E-4} = 1.1\%$$

CASE 1 - Potable Water

Dilution factor = 2200

MPC_{CS137} = 2E-5 uCi/ml

PWST volume = 1.89E9 ml

(2200)(2E-5 uCi/ml)(1.89E9 ml) ÷ 10^6 uCi/Ci = 83 Ci Cs137 in PWST

CASE 2 - Surface Water

Dilution factor = 655

(655)(2E-5 uCi/ml)(1.89E9 ml) ÷ 10^6 uCi/Ci = 25 Ci Cs137 in PWST

Case 2 is limiting

25 Ci Cs-137/.011 = 2270 Ci Limit in PWST
$\cong$ 2,000 Ci

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LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material, excluding tritium and dissolved or entrained noble gases contained in any outside tanks shall be limited to the following:

a. Primary Water Storage Tank $\leq$ 2000 Curies, and

b. Outside Temporary Tank $\leq$ 10 Curies.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.7.1.7.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM THE APPLICANT

ATTACHMENT H

(Bases Section 3/4)

1. Section 3/4.2 (pg. B 3/4 2-1) Power Distribution Limits

   In the first paragraph, third line, insert "design" between the words "minimum" and "DNBR".

2. Section 3/4.2 (pg. B 3/4 2-4)

   Delete the paragraph beginning "Fuel Rod Bowing reduces. . . partially offset rod bow penalties." and insert "Fuel rod bowing reduces the value of DNB ratio. The Safety Analysis for Byron/Braidwood cores maintained sufficient margin between the safety analysis limit DNBR's and the design limit DNBR's to accommodate full flow and low flow penalties identified in WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation" which is applicable to 17 x 17 Optimized Fuel Assembly analysis utilizing the WRB-1 DNB Correlation.

3. Section 3/4.2.5 (pg. B 3/4 2-6) DNB Parameters

   First paragraph, fifth line, delete the words "a minimum" and replace with "design". Also on that same line delete the words "of 1.30".

4. Section 3/4.4.1 (pg. B 3/4 4-1) RC Loop and Coolant Circulation

   First paragraph, second line, delete the words "above 1.30" and replace with "above the applicable Safety Analysis DNBR".

   Per discussions between Westinghouse and CECo, the above 4 changes are being made in order to better clarify the DNBR limit as it applies to Byron Station.

5. Section 3/4.4.9 (pg. B 3/4 4-10) Cooldown

   This has been added to be consistent with the change that was made on page 3/4 4-32.

   Insert the following paragraph as the last paragraph on Pg 3/4 4-10: "The notch in the cooldown curve of Figure 3.4-3 is due to the added constraint on the vessel closure flange given in Appendix G of 10 CFR 50. This constraint requires that, at pressures greater than 20% of the preservice system hydrostatic test pressure, the flange regions that are highly stressed by the bolt preload must exceed the RT_{NDT} of the material by at least 120°F. In the case of Byron 1, the flange RT_{NDT} + 120°F impinges on the cooldown curves and therefore the notch is required.

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BASES

DESIGN

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to 1.34 for a typical cell and 1.32 for a thimble cell during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power; and

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

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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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Insert "B"

Fuel rod bowing reduces the value of DNB ratio. The Safety Analysis for Byron/Braidwood cores maintained sufficient margin between the safety analysis limit DNBR's and the design limit DNBR's to accomodate full flow and low flow penalties identified in WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation" which is applicable to 17x17 Optimized Fuel Assembly analysis utilizing the WRB-1 DNB Correlation.

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PRESSURE/TEMPERATURE LIMITS (Continued)

K_{IR} = constant provided by the code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

Add Paragraph "A"

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3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR ~~above 1.30~~ during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour.

ABOVE THE APPLICABLE SAFETY ANALYSIS DNBR

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

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump with one or more RCS cold legs less than or equal to 380°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

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QUADRANT POWER TILT RATIO (Continued)

not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

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

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain ~~a minimum~~ DESIGN DNBR ~~of 1.30~~ throughout each analyzed transient.

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

Paragraph "A"

The notch in the cooldown curve of Fig 3.4-3 is due to the added constraint on the vessel closure flange given in App. G of 10 CFR 50. This constraint requires that, at pressures greater than 20% of the preservice system hydrostatic test pressure, the flange regions that are highly stressed by the bolt preload must exceed the RT_{NDT} of the material by at least 120 °F. In the case of Byron 1, the flange RT_{NDT} +120 °F impinges on the cooldown curve, and therefore the notch is required.

ATTACHMENT I

(Section 6.0)

1. Section 6.3.2.c (pg. 6-12) Authority

   The word "pases" has been changed to "phases" for typographical error.

(0441M)

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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 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 - 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

Phases

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 ~~pases~~ 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 J

(Section 3/4.2)

1) Figure 3.2-3 (pg. 3/4 2-9) RCS Total Flow Rate Versus R.

The new graph has been supplied by Westinghouse

46 1320

K•E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO.

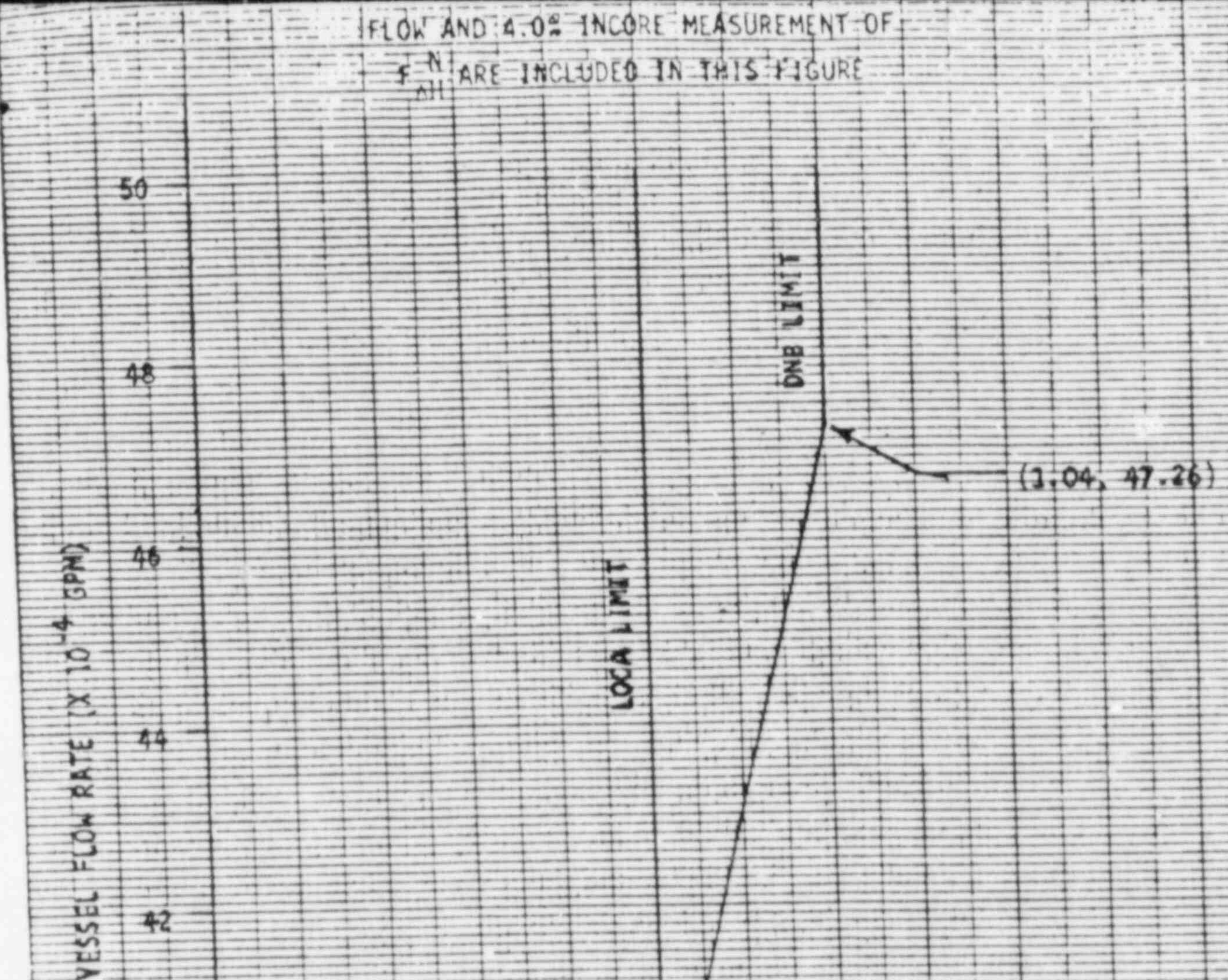

FIGURE 3.2-3. N-LOOP OPERATION FLOW VS. $F^N_{\Delta H}$ LIMIT-BYRON/BRAIDWOOD

New Page 3/4 2-~~10~~ 9

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

RCS Total Flow Rate vs. R