Docket No.: STN 50-454

OCT 26 1984

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

Subject: Byron Station Unit 1 - Technical Specifications

By our letter dated August 27, 1984, we provided you with Byron Station Unit 1 Technical Specifications in final draft form. Several typographical errors, omissions and clarifications were noted by our letter of September 19, 1984, and replacement pages were enclosed. Because many additional pages had been changed, we enclosed a complete set of Byron Technical Specifications with our letter of October 12, 1984 for your review and certification that those technical specifications accurately reflected the plant, the FSAR and the Safety Evaluation Report. Several additional typographical errors, omissions and clarifications were noted by our letter of October 19, 1984.

Since October 19, 1984, we have completed our review of the Technical Specifications for the ultimate heat sink. The enclosure includes page changes to incorporate change to the Technical Specifications for the ultimate heat sink and to incorporate additional corrections and clarifications.

We believe that these Technical Specifications are complete and not subject to further change prior to a decision on issuance of an operating license.

If this changes any of the statements in your letter dated October 23, 1984, concerning technical specifications, advise us in writing.

Sincerely,

ORIGINAL SIGNED BY:

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Enclosures: As stated

cc: See next page

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## TABLE 2.2-1

## REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (SE) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 1. Manual Reactor Trip | N.A. | N.A. | N.A. | N.A. | N.A. |
| 2. Power Range, Neutron Flux | | | | | |
| a. High Setpoint | 7.5 | 4.56 | 0 | ≤109% of RTP* | ≤111.1% of RTP* |
| b. Low Setpoint | 8.3 | 4.56 | 0 | ≤25% of RTP* | ≤27.1% of RTP* |
| 3. Power Range, Neutron Flux, High Positive Rate | 1.6 | 0.5 | 0 | ≤5% of RTP* with a time constant ≥2 seconds | ≤6.3% of RTP* with a time constant ≥2 seconds |
| 4. Power Range, Neutron Flux, High Negative Rate | 1.6 | 0.5 | 0 | ≤5% of RTP* with a time constant ≥2 seconds | ≤6.3% of RTP* with a time constant ≥2 seconds |
| 5. Intermediate Range, Neutron Flux | 17.0 | 8.4 | 0 | ≤25% of RTP* | ≤30.9% of RTP* |
| 6. Source Range, Neutron Flux | 17.0 | 10.0 | 0 | ≤$10^5$ cps | ≤$1.4 \times 10^5$ cps |
| 7. Overtemperature ΔT | 27.7 | 5.38 | See Note 5 | See Note 1 | See Note 2 |
| 8. Overpower ΔT | 4.3 | 1.3 | 1.2 | See Note 3 | See Note 4 |
| 9. Pressurizer Pressure-Low | 5.0 | 2.21 | 1.5 | ≥1885 psig | ≥1871 psig |
| 10. Pressurizer Pressure-High | 3.1 | 0.71 | 1.5 | ≤2385 psig | ≤2396 psig |
| 11. Pressurizer Water Level-High | 5.0 | 2.18 | 1.5 | ≤92% of instrument span | ≤93.8% of instrument span |

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

| | | |
|---|---|---|
| $\tau_6$ | = | Time constant utilized in the measured $T_{avg}$ lag compensator, $\tau_6$ = 0 s, |
| T' | $\leq$ | 588.4°F (Nominal $T_{avg}$ at RATED THERMAL POWER), |
| $K_3$ | = | 0.00134, |
| P | = | Pressurizer pressure, psig, |
| P' | = | 2235 psig (Nominal RCS operating pressure), |
| S | = | Laplace transform operator, $s^{-1}$, |

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

(i) for $q_t - q_b$ between -∞% and +10%, $f_1(\Delta I) = 0$, where $q_t$ and $q_b$ are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds +10%, the ΔT Trip Setpoint shall be automatically reduced by 2.0% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.9% of ΔT span.

## 2.1 SAFETY LIMITS

## BASES

### 2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the WRB-1 correlation. The WRB-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNB design basis is as follows: there must 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.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum design 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 1.49. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

REACTOR CORE (Continued)

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

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 $f_1$ ($\Delta I$) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature $\Delta T$ trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

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

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

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 30% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely on decreasing power between P-8 and P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. For undervoltage, the delay is set so that the time required for a signal to cause a reactor trip after the Undervoltage Trip Setpoint is reached shall not exceed 1.5 seconds. For underfrequency, the delay is set so that the time required for a signal to cause a reactor trip after the Underfrequency Trip Setpoint is reached shall not exceed 0.6 second. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A flow path from the Boric Acid Storage System via a boric acid transfer pump and a centrifugal charging pump to the Reactor Coolant System if the Boric Acid Storage System is OPERABLE as given in Specification 3.1.2.5a. for MODES 5 and 6 or as given in Specification 3.1.2.6a. for MODE 4; or

b. The flow path from the refueling water storage tank via a centrifugal charging pump to the Reactor Coolant System if the refueling water storage tank is OPERABLE as given in Specification 3.1.2.5b. for MODES 5 and 6 or as given in Specification 3.1.2.6b. for MODE 4.

APPLICABILITY: MODES 4*, 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

## CHARGING PUMP - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 4*, 5, and 6.

ACTION:

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

### SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, on recirculation flow, that a differential pressure across the pump of greater than or equal to 2396 psid is developed when tested pursuant to Specification 4.0.5.

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

---

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

**An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

## CHARGING PUMPS - OPERATING

### LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

### SURVEILLANCE REQUIREMENTS

4.1.2.4. At least two charging pumps shall be demonstrated OPERABLE by verifying, on recirculation flow, that a differential pressure across each pump of greater than or equal to 2396 psid is developed when tested pursuant to Specification 4.0.5.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_{xy}$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER;

b. Increasing the measured $F_{xy}$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties;

c. Comparing the $F_{xy}$ computed ($F_{xy}^C$) obtained in Specification 4.2.2.2b., above, to:

1) The $F_{xy}$ limits for RATED THERMAL POWER ($F_{xy}^{RTP}$) for the appropriate measured core planes given in Specifications 4.2.2.2e. and f., below, and

2) The relationship:

$$F_{xy}^L = F_{xy}^{RTP} [1+0.2(1-P)]$$

Where $F_{xy}^L$ is the limit for fractional THERMAL POWER operation expressed as a function of $F_{xy}^{RTP}$ and P is the fraction of RATED THERMAL POWER at which $F_{xy}$ was measured.

d. Remeasuring $F_{xy}$ according to the following schedule:

1. When $F_{xy}^C$ is greater than the $F_{xy}^{RTP}$ limit for the appropriate measured core plane but less than the $F_{xy}^L$ relationship, additional power distribution maps shall be taken and $F_{xy}^C$ compared to $F_{xy}^{RTP}$ and $F_{xy}^L$:

a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which $F_{xy}^C$ was last determined, or

b) At least once per 31 EFPD, whichever occurs first.

TABLE 3.3-1 (Continued)
ACTION STATEMENTS (Continued)

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

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status within 48 hours or within the next hour open the reactor trip breakers, suspend all operations involving positive reactivity changes, and verify valves 1CV-111B, 1CV-8428, 1CV-8439, 1CV-8441 and 1CV-8435 are closed and secured in position. With no channels OPERABLE verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, and take the actions stated above within 1 hour and verify compliance at least once per 12 hours therafter.

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

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

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

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

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

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

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

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

## TABLE 3.3-5

## ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 1. Manual Initiation | |
| a. Safety Injection (ECCS) | N.A. |
| b. Containment Spray | N.A. |
| c. Phase "A" Isolation | N.A. |
| d. Phase "B" Isolation | N.A. |
| e. Containment Vent Isolation | N.A. |
| f. Steam Line Isolation | N.A. |
| g. Feedwater Isolation | N.A. |
| h. Auxiliary Feedwater | N.A. |
| i. Essential Service Water | N.A. |
| j Containment Cooling Fans | N.A. |
| k. Start Diesel Generator | N.A. |
| l. Control Room Isolation | N.A. |
| m. Turbine Trip | N.A. |
| 2. Containment Pressure-High-1 | |
| a. Safety Injection (ECCS) | $\leq 27^{(1)}/12^{(5)}$ |
| 1) Reactor Trip | $\leq 2$ |
| 2) Feedwater Isolation | $\leq 7^{(3)}$ |
| 3) Phase "A" Isolation | $\leq 2^{(6)}$ |
| 4) Containment Vent Isolation | $\leq 7$ |
| 5) Auxiliary Feedwater | $\leq 60$ |
| 6) Essential Service Water | $\leq 42^{(1)}$ |
| 7) Containment Cooling Fans | $\leq 40^{(1)}$ |
| 8) Start Diesel Generator | $\leq 12$ |
| 9) Control Room Isolation | N.A. |
| 10) Turbine Trip. | N.A. |
| 3. Pressurizer Pressure-Low | |
| a. Safety Injection (ECCS) | $\leq 27^{(1)}/12^{(5)}$ |
| 1) Reactor Trip | $\leq 2$ |
| 2) Feedwater Isolation | $\leq 7^{(3)}$ |
| 3) Phase "A" Isolation | $\leq 2^{(6)}$ |
| 4) Containment Vent Isolation | $\leq 7$ |

## TABLE NOTATIONS

*With new fuel or irradiated fuel in the fuel storage areas or fuel building.

**Trip Setpoint is to be established such that the actual submersion dose rate would not exceed 10 mR/hr in the containment building. For containment purge or vent the Setpoint value may be increased up to twice the maximum concentration activity in the containment determined by the sample analysis performed prior to each release in accordance with Table 4.11-2 provided the value does not exceed 10% of the equivalent limits of Specification 3.11.2.1.a in accordance with the methodology and parameters in the ODCM.

## ACTION STATEMENTS

ACTION 26 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge valves are maintained closed.

ACTION 27 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the Control Room Ventilation System and initiate operation of the Control Room Make-up System.

ACTION 28 - Must satisfy the ACTION requirement for Specification 3.4.6.1.

ACTION 29 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, ACTION a. of Specification 3.9.12* must be satisfied. With both channels inoperable, provide an appropriate portable continuous monitor with the same Alarm Setpoint in the fuel pool area with one Fuel Handling Building Exhaust filter plenum in operation. Otherwise satisfy ACTION b. of Specification 3.9.12.*

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*Satisfaction of Specification 3.9.12 ACTIONS are not required prior to the initial entry into MODE 1 when there is no irradiated fuel in the storage pool.

## TABLE 3.3-9

## REMOTE SHUTDOWN MONITORING INSTRUMENTATION

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

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

| CAPSULE NUMBER | VESSEL LOCATION | LEAD FACTOR | WITHDRAWAL TIME (EFPY)* |
|---|---|---|---|
| U | 58.5° | 4.05 | 1st Refueling |
| X | 238.5° | 4.05 | 6 |
| V | 61° | 3.37 | 10 |
| Y | 241° | 3.37 | 15 |
| W | 121.5° | 4.05 | Standby |
| Z | 301.5° | 4.05 | Standby |

*Withdrawal time may be modified to coincide with those refueling outages or reactor shutdowns most closely approaching the withdrawal schedule.

## 3/4.4.11 REACTOR COOLANT SYSTEM VENTS

### LIMITING CONDITION FOR OPERATION

3.4.11 At least one reactor vessel head vent path consisting of two valves in series powered from emergency buses shall be OPERABLE and closed.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

### SURVEILLANCE REQUIREMENTS

4.4.11 Each reactor vessel head vent path shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying all manual isolation valves in each vent path are locked in the open position.

b. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

c. Verifying flow through the reactor vessel head vent paths during venting operations at COLD SHUTDOWN or REFUELING.

## 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 31% and 63%,

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

d. A nitrogen cover-pressure of between 602 and 647 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.

SURVEILLANCE REQUIREMENTS (Continued)

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1) Verifying automatic isolation and interlock action of the RHR System from the Reactor Coolant System by ensuring that:

a) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 360 psig the interlocks prevent the valves from being opened, and

b) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 662 psig the interlocks will cause the valves to automatically close.

2) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

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

1) Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal and on a RWST Level-Low-Low test signal, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1) Centrifugal charging pump $\geq$ 2396 psid,

2) Safety Injection pump $\geq$ 1412 psid, and

3) RHR pump $\geq$ 181 psid.

## TABLE 5.7-1

## COMPONENT CYCLIC OR TRANSIENT LIMITS

| COMPONENT | CYCLIC OR TRANSIENT LIMIT | DESIGN CYCLE OR TRANSIENT |
|---|---|---|
| Reactor Coolant System | 200 heatup cycles at ≤ 100°F/h and 200 cooldown cycles at ≤ 100°F/h. | Heatup cycle - $T_{avg}$ from ≤ 200°F to ≥ 550°F. Cooldown cycle - $T_{avg}$ from ≥ 550°F to ≤ 200°F. |
| | 200 pressurizer cooldown cycles at ≤ 200°F/h. | Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 100°F. |
| | 80 loss of load cycles, without immediate Turbine or Reactor trip. | ≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER. |
| | 40 cycles of loss-of-offsite A.C. electrical power. | Loss-of-offsite A.C. electrical ESF Electrical System. |
| | 80 cycles of loss of flow in one reactor coolant loop. | Loss of only one reactor coolant pump. |
| | 400 Reactor trip cycles. | 100% to 0% of RATED THERMAL POWER. |
| | 10 auxiliary spray actuation cycles. | Spray water temperature differential > 320°F. |
| | 50 leak tests. | Pressurized to ≥ 2485 psig. |
| | 5 hydrostatic pressure tests. | Pressurized to ≥ 3100 psig. |
| Secondary Coolant System | 1 large steam line break. | Break in a > 6-inch steam line. |
| | 5 hydrostatic pressure tests. | Pressurized to ≥ 1350 psig. |

## 3/4.5.3 ECCS SUBSYSTEMS - $T_{avg}$ < 350°F

### LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,*

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

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

APPLICABILITY: MODE 4.

ACTION:

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

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

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission within 90 days, pursuant to Specification 6.9.2, describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

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

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1) Less than or equal to $L_a$, 0.10% by weight of the containment air per 24 hours at $P_a$, 44.4 psig, or

2) Less than or equal to $L_t$, 0.07% by weight of the containment air per 24 hours at $P_t$, 22.2 psig.

b. A combined leakage rate of less than 0.60 $L_a$ for all penetrations and valves subject to Type B and C tests, when pressurized to $P_a$.

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

ACTION:

With either the measured overall integrated containment leakage rate exceeding 0.75 $L_a$ or 0.75 $L_t$, as applicable, or the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 $L_a$, restore the overall integrated leakage rate to less than 0.75 $L_a$ or less than 0.75 $L_t$, as applicable, and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 $L_a$ prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than $P_a$, 44.4 psig, or $P_t$, 22.2 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

## TABLE 3.6-1

## CONTAINMENT ISOLATION VALVES

| PENETRATION | VALVE NO. | FUNCTION | MAXIMUM ISOLATION TIME (SEC) |
|---|---|---|---|
| 1. Phase "A" Isolation | | | |
| 28 | 1CV8100 | RCP Seal Water Return | 10 |
| 28 | 1CV8112 | RCP Seal Water Return | 10 |
| 41 | 1CV8152 | RCS Letdown | 10 |
| 41 | 1CV8160 | RCS Letdown | 10 |
| 5 | 1WØ020A | Chilled Water | 50 |
| 5 | 1WØ056A | Chilled Water | 50 |
| 6 | 1WØ006A | Chilled Water | 50 |
| 8 | 1WØ020B | Chilled Water | 50 |
| 8 | 1WØ056B | Chilled Water | 50 |
| 10 | 1WØ006B | Chilled Water | 50 |
| 22 | 1CC9437B* | Excess Ltdn HX Return | 10 |
| 48 | 1CC9437A* | Excess Ltdn HX Supply | 10 |
| 34 | 1FP010* | Fire Protection | 12 |
| 39 | 1IA065 | Instrument Air | 15 |
| 39 | 1IA066 | Instrument Air | 15 |
| 13 | 1ØG079 | Hydrogen Recombiner | 60 |
| 13 | 1ØG080 | Hydrogen Recombiner | 60 |
| 13 | 1ØG082 | Hydrogen Recombiner | 60 |
| 13 | 1ØG084 | Hydrogen Recombiner | 60 |
| 23 | 1ØG081 | Hydrogen Recombiner | 60 |
| 23 | 1ØG085 | Hydrogen Recombiner | 60 |
| 69 | 1ØG057A | Hydrogen Recombiner | 60 |
| 69 | 1ØG083 | Hydrogen Recombiner | 60 |
| 56 | 1SA032 | Service Air | 4.5 |
| 56 | 1SA033 | Service Air | 4.5 |
| 80 | 1SD002C | Steam Generator Blowdown | 7.5 |
| 80 | 1SD005B | Steam Generator Blowdown | 3.0 |
| 81 | 1SD002D | Steam Generator Blowdown | 7.5 |
| 82 | 1SD002A | Steam Generator Blowdown | 7.5 |
| 82 | 1SD005A | Steam Generator Blowdown | 3.0 |
| 83 | 1SD002B | Steam Generator Blowdown | 7.5 |
| 88 | 1SD002E | Steam Generator Blowdown | 7.5 |
| 88 | 1SD005C | Steam Generator Blowdown | 3.0 |
| 89 | 1SD002F | Steam Generator Blowdown | 7.5 |
| 90 | 1SD002G | Steam Generator Blowdown | 7.5 |
| 90 | 1SD005D | Steam Generator Blowdown | 3.0 |
| 91 | 1SD002H | Steam Generator Blowdown | 7.5 |

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| PENETRATION | VALVE NO. | FUNCTION | MAXIMUM ISOLATION TIME (SEC) |
|---|---|---|---|
| 1. Phase "A" Isolation (Continued) | | | |
| 52 | 1PR001A | Process Radiation | 4.5 |
| 52 | 1PR001B | Process Radiation | 4.5 |
| 52 | 1PR066 | Process Radiation | 5.0 |
| 12 | 1PS228A | Hydrogen Monitor | N/A** |
| 12 | 1PS229A | Hydrogen Monitor | N/A** |
| 12 | 1PS230A | Hydrogen Monitor | N/A** |
| 12 | 1PS228B | Hydrogen Monitor | N/A** |
| 12 | 1PS229B | Hydrogen Monitor | N/A** |
| 12 | 1PS230B | Hydrogen Monitor | N/A** |
| 70 | 1PS9354A | Primary Process Sampling | 10 |
| 70 | 1PS9354B | Primary Process Sampling | 10 |
| 70 | 1PS9355A | Primary Process Sampling | 10 |
| 70 | 1PS9355B | Primary Process Sampling | 10 |
| 70 | 1PS9356A | Primary Process Sampling | 10 |
| 70 | 1PS9356B | Primary Process Sampling | 10 |
| 70 | 1PS9357A | Primary Process Sampling | 10 |
| 70 | 1PS9357B | Primary Process Sampling | 10 |
| 11 | 1RE9170 | Reactor Bldg Equip Drains | 10 |
| 11 | 1RE1003 | Reactor Bldg Equip Drains | 10 |
| 65 | 1RE9157 | Reactor Bldg Equip Drains | 10 |
| 65 | 1RE9159A | Reactor Bldg Equip Drains | 10 |
| 65 | 1RE9159B | Reactor Bldg Equip Drains | 10 |
| 65 | 1RE9160A | Reactor Bldg Equip Drains | 10 |
| 65 | 1RE9160B | Reactor Bldg Equip Drains | 10 |
| 27 | 1RY8025 | PRT Nitrogen | 10 |
| 27 | 1RY8026 | PRT Nitrogen | 10 |
| 27 | 1RY8033 | PRT Nitrogen | 10 |
| 44 | 1RY8028 | PRT Make-up | 10 |
| 55 | 1SI8964 | Accumulator Fill | 10 |
| 55 | 1SI8880 | Nitrogen Supply to Accumulator | 10 |
| 55 | 1SI8871 | Accumulator Fill | 10 |
| 55 | 1SI8888 | Hot Leg Safety Injection | 10 |
| 47 | 1RF026 | Reactor Building Floor Drains | 15 |
| 47 | 1RF027 | Reactor Building Floor Drains | 15 |

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| PENETRATION | VALVE NO. | FUNCTION | MAXIMUM ISOLATION TIME (SEC) |
|---|---|---|---|
| 2. Phase "B" Isolation | | | |
| 21 | 1CC9414 | RCP Mtr Brng Return | 10 |
| 21 | 1CC9416 | RCP Mtr Brng Return | 10 |
| 24 | 1CC685 | RCP Thermal Barrier Return | 10 |
| 24 | 1CC9438 | RCP Thermal Barrier Return | 10 |
| 25 | 1CC9413A | RCP Cooling Wtr Supply | 10 |
| 3. Safety Injection | | | |
| 71 | 1CV8105* | CVCS Charging | 10 |
| 71 | 1CV8106* | CVCS Charging | 10 |
| 7 | 1SX016B* | Essential Service Water | N/A |
| 9 | 1SX027B* | Essential Service Water | N/A |
| 14 | 1SX027A* | Essential Service Water | N/A |
| 15 | 1SX016A* | Essential Service Water | N/A |
| 26 | 1SI8801A* | Cold Leg Safety Injection | N/A |
| 26 | 1SI8801B* | Cold Leg Safety Injection | N/A |
| 92 | 1SI8811A* | Containment Recirc. Sump | N/A |
| 93 | 1SI8811B* | Containment Recirc. Sump | N/A |
| 4. Containment Ventilation Isolation | | | |
| 94 | 1VQ003 | Mini-Flow Purge Exhaust | 5 |
| 94 | 1VQ005A | Mini-Flow Purge Exhaust | 5 |
| 94 | 1VQ005B | Mini-Flow Purge Exhaust | 5 |
| 94 | 1VQ005C | Mini-Flow Purge Exhaust | 5 |
| 95 | 1VQ002A | Purge Exhaust | 5 |
| 95 | 1VQ002B | Purge Exhaust | 5 |
| 96 | 1VQ004A | Mini-Flow Purge Supply | 5 |
| 96 | 1VQ004B | Mini-Flow Purge Supply | 5 |
| 97 | 1VQ001A | Purge Supply | 5 |
| 97 | 1VQ001B | Purge Supply | 5 |
| 5. Containment Spray Actuation | | | |
| 1 | 1CS007A | Containment Spray | 30 |
| 16 | 1CS007B | Containment Spray | 30 |
| 6. Main Steam Isolation | | | |
| 77 | 1MS101D* | Main Steam | 10.0 |
| 78 | 1MS101A* | Main Steam | 10.0 |

## TABLE 3.6-1 (Continued)

## CONTAINMENT ISOLATION VALVES

| PENETRATION | VALVE NO. | FUNCTION | MAXIMUM ISOLATION TIME (SEC) |
|---|---|---|---|
| 6. Main Steam Isolation (Continued) | | | |
| 85 | 1MS101B* | Main Steam | 10.0 |
| 86 | 1MS101C* | Main Steam | 10.0 |
| 7. Feedwater Isolation | | | |
| 76 | 1FW009D* | Main Feedwater | 5.0 |
| 76 | 1FW043D* | Main Feedwater | 6.0 |
| 79 | 1FW009A* | Main Feedwater | 5.0 |
| 79 | 1FW043A* | Main Feedwater | 6.0 |
| 84 | 1FW009B* | Main Feedwater | 5.0 |
| 84 | 1FW043B* | Main Feedwater | 6.0 |
| 87 | 1FW009C* | Main Feedwater | 5.0 |
| 87 | 1FW043C* | Main Feedwater | 6.0 |
| 99 | 1FW035D* | Main Feedwater | 6.0 |
| 99 | 1FW039D* | Main Feedwater | 6.0 |
| 100 | 1FW035A* | Main Feedwater | 6.0 |
| 100 | 1FW039A* | Main Feedwater | 6.0 |
| 101 | 1FW035B* | Main Feedwater | 6.0 |
| 101 | 1FW039B* | Main Feedwater | 6.0 |
| 102 | 1FW035C* | Main Feedwater | 6.0 |
| 102 | 1FW039C* | Main Feedwater | 6.0 |
| 8. Remote Manual | | | |
| 68 | 1RH8701A* | RH Suction | N/A |
| 68 | 1RH8701B* | RH Suction | N/A |
| 75 | 1RH8702A* | RH Suction | N/A |
| 75 | 1RH8702B* | RH Suction | N/A |
| 59 | 1SI8881* | Hot Leg Safety Injection | N/A |
| 73 | 1SI8824* | Hot Leg Safety Injection | N/A |
| 66 | 1SI8825* | Hot Leg RH Injection | N/A |
| 60 | 1SI8823* | Cold Leg Safety Injection | N/A |
| 50 | 1SI8890A* | Cold Leg RH Injection | N/A |
| 51 | 1SI8890B* | Cold Leg RH Injection | N/A |
| 26 | 1SI8843* | Cold Leg Safety Injection | N/A |
| 33 | 1CV8355A* | RCP Seal Injection | N/A |
| 33 | 1CV8355D* | RCP Seal Injection | N/A |
| 53 | 1CV8355B* | RCP Seal Injection | N/A |
| 53 | 1CV8355C* | RCP Seal Injection | N/A |

## TABLE 3.6-1 (Continued)

## CONTAINMENT ISOLATION VALVES

| PENETRATION | VALVE NO. | FUNCTION | MAXIMUM ISOLATION TIME (SEC) |
|---|---|---|---|
| 8. Remote Manual (Continued) | | | |
| 59 | 1SI8802A* | Hot Leg Safety Injection | N/A |
| 73 | 1SI8802B* | Hot Leg Safety Injection | N/A |
| 60 | 1SI8835* | Hot Leg Safety Injection | N/A |
| 50 | 1SI8809A* | RH Cold Leg Injection | N/A |
| 51 | 1SI8809B* | RH Cold Leg Injection | N/A |
| 66 | 1SI8840* | Hot Leg Safety Injection - | N/A |
| 100 | 1AF013A* | Feedwater | N/A |
| 100 | 1AF013E* | Feedwater | N/A |
| 101 | 1AF013B* | Feedwater | N/A |
| 101 | 1AF013F* | Feedwater | N/A |
| 102 | 1AF013C* | Feedwater | N/A |
| 102 | 1AF013G* | Feedwater | N/A |
| 99 | 1AF013D* | Feedwater | N/A |
| 99 | 1AF013H* | Feedwater | N/A |
| 9. Manual | | | |
| 37 | 1CV8346* | RCS Loop Fill | N/A |
| I3 | 1VQ016 | Instrument Penetration | N/A |
| I3 | 1VQ017 | Instrument Penetration | N/A |
| I3 | 1VQ018 | Instrument Penetration | N/A |
| I3 | 1VQ019 | Instrument Penetration | N/A |
| I5 | 1RY075 | Instrument Penetration | N/A |
| 30 | 1WM190 | Make-Up Demin | N/A |
| 57 | 1FC009 | Spent Fuel Pool Cleaning | N/A |
| 57 | 1FC010 | Spent Fuel Pool Cleaning | N/A |
| 32 | 1FC011 | Spent Fuel Pool Cleaning | N/A |
| 32 | 1FC012 | Spent Fuel Pool Cleaning | N/A |
| 77 | 1MS021D* | Main Steam | N/A |
| 78 | 1MS021A* | Main Steam | N/A |
| 85 | 1MS021B* | Main Steam | N/A |
| 86 | 1MS021C* | Main Steam | N/A |
| AL | 1PR002E | Process Radiation | N/A |
| AL | 1PR033A | Process Radiation | N/A |
| AL | 1PR033B | Process Radiation | N/A |
| AL | 1PR002F | Process Radiation | N/A |
| AL | 1PR033C | Process Radiation | N/A |
| AL | 1PR033D | Process Radiation | N/A |

## TABLE 3.6-1 (Continued)

## CONTAINMENT ISOLATION VALVES

| PENETRATION | VALVE NO. | FUNCTION | MAXIMUM ISOLATION TIME (SEC) |
|---|---|---|---|
| 9. Manual (Continued) | | | |
| 99 | 1FW015D* | Feedwater | N/A |
| 100 | 1FW015A* | Feedwater | N/A |
| 101 | 1FW015B* | Feedwater | N/A |
| 102 | 1FW015C* | Feedwater | N/A |
| 10. Check | | | |
| 28 | 1CV8113 | RCP Seal Water Return | N/A |
| 37 | 1CV8348* | RCS Loop Fill | N/A |
| 6 | 1WØ007A | Chilled Water | N/A |
| 10 | 1WØ007B | Chilled Water | N/A |
| 21 | 1CC9534 | RCP Mtr Brng Return | N/A |
| 24 | 1CC9518 | RCP Thermal Barrier Return | N/A |
| 25 | 1CC9486 | RCP Cooling Wtr Supply | N/A |
| 1 | 1CS008A | Containment Spray | N/A |
| 16 | 1CS008B | Containment Spray | N/A |
| 39 | 1IA091 | Instrument Air | N/A |
| 30 | 1WM191 | Make-Up Demin | N/A |
| 52 | 1PR032 | Process Radiation | N/A |
| AL | 1PR002G | Process Radiation | N/A |
| AL | 1PR002H | Process Radiation | N/A |
| 12 | 1PS231A | Hydrogen Monitor | N/A |
| 12 | 1PS231B | Hydrogen Monitor | N/A |
| 27 | 1RY8047 | PRT Nitrogen | N/A |
| 44 | 1RY8046 | PRT Make-Up | N/A |
| 26 | 1SI8815* | Safety Injection | N/A |
| 50 | 1SI8818A* | Safety Injection | N/A |
| 50 | 1SI8818D* | Safety Injection | N/A |
| 51 | 1SI8818B* | Safety Injection | N/A |
| 51 | 1SI8818C* | Safety Injection | N/A |
| 59 | 1SI8905A* | Safety Injection | N/A |
| 59 | 1SI8805D* | Safety Injection | N/A |
| 60 | 1SI8819A* | Safety Injection | N/A |
| 60 | 1SI8819B* | Safety Injection | N/A |

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| PENETRATION | VALVE NO. | FUNCTION | MAXIMUM ISOLATION TIME (SEC) |
|---|---|---|---|
| 10. Check (Continued) | | | |
| 60 | 1SI8819C* | Safety Injection | N/A |
| 60 | 1SI8819D* | Safety Injection | N/A |
| 66 | 1SI8841A* | Safety Injection | N/A |
| 66 | 1SI8841B* | Safety Injection | N/A |
| 73 | 1SI8905B* | Safety Injection | N/A |
| 73 | 1SI8905C* | Safety Injection | N/A |
| 55 | 1SI8968* | Safety Injection | N/A |
| 34 | 1FP345* | Fire Protection | N/A |
| 33 | 1CV8368A* | RCP Seal Injection | N/A |
| 33 | 1CV8368D* | RCP Seal Injection | N/A |
| 53 | 1CV8368B* | RCP Seal Injection | N/A |
| 53 | 1CV8368C* | RCP Seal Injection | N/A |
| 11. S/G Safeties/PORVs | | | |
| 77 | 1MS013D* | Main Steam | N/A |
| 77 | 1MS014D* | Main Steam | N/A |
| 77 | 1MS015D* | Main Steam | N/A |
| 77 | 1MS016D* | Main Steam | N/A |
| 77 | 1MS017D* | Main Steam | N/A |
| 78 | 1MS013A* | Main Steam | N/A |
| 78 | 1MS014A* | Main Steam | N/A |
| 78 | 1MS015A* | Main Steam | N/A |
| 78 | 1MS016A* | Main Steam | N/A |
| 78 | 1MS017A* | Main Steam | N/A |
| 85 | 1MS013B* | Main Steam | N/A |
| 85 | 1MS014B* | Main Steam | N/A |
| 85 | 1MS015B* | Main Steam | N/A |
| 85 | 1MS016B* | Main Steam | N/A |
| 85 | 1MS017B* | Main Steam | N/A |
| 86 | 1MS013C* | Main Steam | N/A |
| 86 | 1MS014C* | Main Steam | N/A |
| 86 | 1MS015C* | Main Steam | N/A |
| 86 | 1MS016C* | Main Steam | N/A |
| 86 | 1MS017C* | Main Steam | N/A |
| 77 | 1MS018D* | Main Steam | 20 |
| 78 | 1MS018A* | Main Steam | 20 |
| 85 | 1MS018B* | Main Steam | 20 |
| 86 | 1MS018C* | Main Steam | 20 |

*Not subject to Type C leakage tests.

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

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5* Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3:

With one MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. The provisions of Specification 3.0.4 are not applicable. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

*Not applicable prior to initial criticality on Cycle 1, provided the RCS boron concentration is greater than or equal to 1900 ppm and only 1 MSIV is open at a time.

## 3/4.7.5 ULTIMATE HEAT SINK

### LIMITING CONDITION FOR OPERATION

3.7.5 Two independent ultimate heat sinks (UHS) cooling towers shall be OPERABLE, each with

a. A minimum water level in the UHS cooling tower basin of 873.75 ft msl (86% of total volume),

b. Four OPERABLE cooling tower fans (0A, 0B, 0E, 0F for U1),

c. One OPERABLE essential service water makeup pump per train,

d. An essential service water pump discharge temperature of less than or equal to 80°F,

e. A minimum Rock River water level at or above 670.6 feet mean sea level, USGS datum, at the river screenhouse, and

f. Two OPERABLE deep wells with:

   (1) Rock River water level forecast by National Weather Service to exceed 702.0 feet msl, or

   (2) Rock River water level at or below 670.6 ft msl, or

   (3) Tornado watch issued by National Weather Service that includes Byron site area.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

a. With a water level of less than 873.75 ft msl (86% of total volume) in either UHS cooling tower basin, restore the water level to 873.75 ft msl in each UHS cooling tower basin within 6 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the fans listed above inoperable, restore the listed fans to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

c. With one essential service water makeup pump inoperable, restore the essential service water makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

LIMITING CONDITION FOR OPERATION (Continued)

d. With the essential service water pump discharge water temperature not meeting the above requirement, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With the minimum Rock River water level not meeting the above requirement, notify the NRC within 1 hour in accordance with the procedure of §50.72 of actions or contingencies to ensure an adequate supply of cooling water to the Byron Station for a minimum of 30 days, verify the Rock River flow within one hour, and:

(1) If Rock River flow is less than 700 cfs be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours, or

(2) If Rock River flow is equal to or greater than 700 cfs continue verification procedure every 12 hours or until Rock River water level exceeds 670.6 ft msl, or

(3) If Rock River level is equal to or less than 664.7 ft msl be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours

f. With one deep well inoperable and:

(1) The Rock River water level predicted, through NWS flood forecasts, to exceed 702 ft msl, or

(2) The Rock River water level at or below 670.6 ft msl, or

(3) A tornado watch issued by the NWS that includes the area for the Byron Station.

Notify the NRC within 1 hour in accordance with the procedure of §50.72 of actions or contingencies to ensure an adequate supply of cooling water to the Byron Station for a minimum of 30 days and restore both wells to OPERABLE status before the Rock River water level exceeds 702 ft msl or the minimum Rock River level or flow falls below 664.7 ft or 700 cfs, respectively, or within 72 hours, whichever occurs first, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 The UHS shall be determined OPERABLE at least once per:

a. 24 hours by verifying the water level in each UHS cooling tower basin to be greater than or equal to 873.5 feet msl. (86% of total volume)

SURVEILLANCE REQUIREMENTS (Continued)

b. 24 hours by verifying the essential service water pump discharge water temperature is within its limit.

c. 24 hours by verifying that the Rock River water level is within its limits.

d. 31 days by starting from the control room each U.S cooling tower fan not already in operation and operating each of these fans for at least 15 minutes.

e. 31 days by

1. verifying that the fuel supply for each diesel powered essential service water makeup pump is at least 36% of the fuel supply tank volume.

2. starting the diesel from ambient conditions on a simulated low basin level test signal and operating the diesel powered pump for 30 minutes.

3. verifying that each valve (manual, power operated, or automatic) in the flow path is in its correct position.

4. by starting each deep well pump and operating it for 15 minutes and verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position.

f. each time the National Weather Service forecasts a flood condition which would create a Rock River level greater than or equal to 702.0 ft msl at the river screenhouse, or the measured river level is less than or equal to 670.6 ft msl, by starting each deep well pump and operating it for 15 minutes and verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position and verifying that each pump will provide at least a 550 gpm flow rate.

g. 92 days by verifying that a drain sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM D-4057-1981, is within the acceptable limits specified in Table 1 of ASTM-D975-1977 when checked for viscosity, water, and sediment.

h. 18 months by subjecting each diesel that powers an essential service water makeup pump to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service and by cycling each testable valve in the flow path through at least one complete cycle of full travel.

i. 18 months by verifying each deep well pump will provide at least 550 gpm flow rate.

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

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample from the Emergency Makeup System obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175% when tested at a temperature of 30°C and a relative humidity of 70%; and

3) Verifying a system flow rate of 6000 cfm ± 10% for the Emergency Makeup System and 51,000 cfm ±10% for the Recirculation System when tested in accordance with ANSI N510-1980.

d. After every 720 hours of Emergency Makeup System operation by verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175% when tested at a temperature of 30°C and a relative humidity of 70%;

e. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.0 inches Water Gauge while operating the Emergency Makeup System at a flow rate of 6000 cfm ± 10%;

SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying that on a Safety Injection or High Radiation-Control Room Outside Air Intake test signal, the system automatically switches into a makeup mode of control room ventilation with flow through the Emergency Makeup System HEPA filters and charcoal adsorber banks;

3)* Verifying that the Emergency Makeup System maintains the control room at a positive nominal pressure of greater than or equal to 1/8 inch Water Gauge relative to ambient pressure in areas adjacent to the control room area when operating an Emergency Makeup System at a flowrate of 6,000 cfm ±10%;

4) Verifying that the heaters dissipate 27.2 ± 2.7 kW when tested in accordance with ANSI N510-1980.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the Emergency Makeup System at a flow rate of 6000 cfm ± 10%; and

g. After each complete or partial replacement of a charcoal adsorber bank in the Emergency Makeup System by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 6000 cfm ± 10%.

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*Up to 5% power (Cycle 1), this surveillance requirement is:

3) Verifying that one Makeup System maintains the control room at a positive nominal pressure of greater than or equal to 1/8 inch Water Gauge relative to ambient pressure in areas adjacent to this Control Room area prior to initial criticality. However, in the interim, this system will be operating such that the Control Room is maintained at a positive pressure with respect to all adjacent areas.

3) Verifying a system flow rate of 66,900 cfm ± 10% through the train and 22,300 cfm ±10% per bank through the exhaust filter plenum during operation when tested in accordance with ANSI N510-1980; and

4) Verifying that with the system operating at a flow rate of 66,900 cfm ±10% through the train and 22,300 cfm ±10% per bank and exhausting through the HEPA filter and charcoal adsorbers, the total bypass flow of the system and the damper leakage is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake and the damper leakage rate is determined by either direct measurements or pressure decay measurements at a test pressure of 2 inches of water and the auxiliary building exhaust fans are operating at their rated flow.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained from each bank of adsorbers of the train in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, when the average for a methyl iodide penetration of less than 1% when tested at a temperature of 30°C and a relative humidity of 70%.

d. At least once per 18 months by:

1) Verifying for each filter bank of the train that the pressure drop across the combined HEPA filters and charcoal adsorber banks of less than 6.0 inches Water Gauge while operating the exhaust filter plenum at a flow rate of 66,900 cfm ± 10% through the train and 22,300 cfm ±10% per bank;

2) Verifying that the exhaust filter plenum starts on manual initiation or Safety Injection test signal; and

3) Verifying that the system maintains the ECCS equipment rooms at a negative pressure of greater than or equal to 1/4 in. Water Gauge relative to the outside atmosphere during system operation while operating at a flow rate of 66,900 cfm ±10% through the train and 22,300 cfm ±10% per bank.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the exhaust filter plenum satisfies the in-place penetration testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a DOP test aerosol while operating at a flow rate of 66,900 cfm ± 10% through the train and 22,300 cfm ±10% per bank; and

SURVEILLANCE REQUIREMENTS (Continued)

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the exhaust filter plenum satisfies the in-place penetration testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 66,900 cfm ± 10% through the train and 22,300 cfm ±10% per bank.

TABLE 3.7-5 (Continued)

FIRE HOSE STATIONS

| LOCATION | ELEVATION | HOSE RACK REEL | ANGLE VALVE |
|---|---|---|---|
| Cont. #1 (Continued) | | | |
| R-7: By equipment hatch | 430 | 64 | 1FP160 |
| R-12: By charcoal filter 1A | 430 | 65 | 1FP157 |
| R-17: By south stairs | 403 | 98 | 1FP164 |
| R-2: By RCFC 1C | 403 | 99 | 1FP155 |
| R-7: By pressurizer (outside missile shield) | 403 | 100 | 1FP161 |
| R-12: By panel 1PL69J | 403 | 101 | 1FP158 |
| R-12: By PRT | 381 | 143 | 1FP159 |
| R-17: By south stairs | 381 | 144 | 1FP162 |
| R-2: By RCFC 1C | 381 | 145 | 1FP156 |
| R-7: By panel 1PL52J | 381 | 146 | 1FP165 |
| Turbine Bldg. | | | |
| K-14: By the control room | 451 | 16 | 1FP194 |

7) Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator loading shall be equivalent to the 2-hour rating of 6050 kW* and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 5500 kW. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2f.6)b);**

8) Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 5935 kW;

9) Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

10) Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation and (2) automatically energizing the emergency loads with offsite power;

11) Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines;

12) Verifying that the automatic LOCA and Shutdown sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval; and

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*Instantaneous loads of 6050 kW (+0, -150) are acceptable as equivalent to the 2-hour rating provided voltage and frequency requirements and cooling system functioning requirements are verified to be within design limits at 6050 kW.

**If Specification 4.8.1.1.2f.6)b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at 5500 kW for 1 hour or until operating temperature has stabilized.

A.C. SOURCES

SHUTDOWN

## LIMITING CONDITION FOR OPERATION

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

a. One class 1E 4160 volt bus capable of being powered from:

   1) Either transformer of the associated units System Auxiliary Transformer bank, or

   2) Either transformer of the other units System Auxiliary Transformer bank, with

   The System Auxiliary Transformer bank supplying the 4160 volt bus energized from an off-site transmission circuit.

b. One diesel generator with:

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

   2) A fuel storage system containing a minimum volume of 44,000 gallons of fuel, and

   3) A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

## SURVEILLANCE REQUIREMENTS

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

## 3/4.8.2 D.C. SOURCES

### OPERATING

### LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

a. With one of the required battery banks and/or chargers inoperable, restore the inoperable battery bank and/or battery bus to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the normal full capacity charger inoperable: 1) restore the affected battery and/or battery bus to operable status with the opposite units full capacity charger within 2 hours or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours, and 2) restore the normal full capacity charger to operable status within 24 hours or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

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

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

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

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

2) The total battery terminal voltage is greater than or equal to 126 volts on float charge.

*This specification is only applicable prior to Unit 2 operation in MODE 4.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 145 volts, by verifying that:

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

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

   3) The average electrolyte temperature of all connected cells is above 60°F.

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

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

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

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

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

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

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

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

*Obtained by subtracting the normal resistance of: 1) the cross room rack connector ($400 \times 10^{-6}$ ohm, typical) and 2) the bi-level rack connector ($50 \times 10^{-6}$ ohm, typical); from the measured cell-to-cell connection resistance.

## 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 6.9 kV and the 4.16 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 to demonstrate that the overall penetration protection design remains within operable limits.

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. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturers data to ensure that it is less than or equal to a value specified by the manufacturer. 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 6.9 kV and 4.16 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 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

| PROTECTIVE DEVICE NUMBER AND LOCATION | DEVICE |
|---|---|
| 4. 480V Molded Case Ckt. Bkts. (MCCB) (Continued) | |
| | MCC 134X5 |
| 1FH02J Cub G1 | Primary Backup |
| 1FH03J Cub G2 | Primary Backup |
| 1RC01PB-B Cub B1 | Primary Backup |
| 1RE01PB Cub B3 | Primary Backup |
| 1RC01PC-A Cub C1 | Primary Backup |
| 1RC01PC-B Cub C2 | Primary Backup |
| 1VP05CB Cub J1 | Primary Backup |
| 1RC01PB-A Cub C3 | Primary Backup |
| 1HC65G-A Cub D3 | Primary Backup |
| 1VP02CB Cub F1 | Primary Backup |
| 1RC01R-A Cub F2 A&B | Primary Backup |
| 1RF02PA Cub G3 | Primary Backup |
| 1EW12EA,B,C Cub F3 A&B | Primary Backup |

TABLE 3.3-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

| PROTECTIVE DEVICE NUMBER AND LOCATION | DEVICE |
|---|---|
| 4. 480V Molded Case Ckt. Bkts. (MCCB) (Continued) | |
| | MCC 134X5 |
| 1VP04CB Cub F4 | Primary<br>Backup |
| 1VP04CD Cub F5 | Primary<br>Backup |
| | MCC 132X2A |
| 1SI8808C Cub A2 | Primary |
| MCC 132X2 Cub B2 | Backup |
| 1SI8808B Cub A3 | Primary |
| MCC 132X2 Cub B2 | Backup |
| | MCC 132X2 |
| 1RH8702B Cub B1 | Primary<br>Backup |
| 1RH8701B Cub B3 | Primary<br>Backup |
| 1CV8112 Cub B4 | Primary<br>Backup |
| 1OG079 Cub C1 | Primary<br>Backup |
| 1WO056A Cub C2 | Primary<br>Backup |
| 1OG080 Cub C3 | Primary<br>Backup |

## TABLE 3.8-2 (Continued)

## MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION DEVICES

| VALVE NUMBER | FUNCTION |
|---|---|
| 1SI8804B | SI Pump 1B Suct X-tie from RHR HX |
| 1SI8806 | SI Pumps Upstream Suction Isol |
| 1SI8807A | SI to Chg PP Suction Crosstie Isol Vlv |
| 1SI8807B | SI to Chg PP Suction Crosstie Isol Vlv |
| 1SI8808A | Accum. 1A Disch. Isol. Valve |
| 1SI8808B | Accum. 1B Disch. Isol. Valve |
| 1SI8808C | Accum. 1C Disch. Isol. Valve |
| 1SI8808D | Accum. 1D Disch. Isol. Valve |
| 1SI8809A | SI RX HX 1A Dsch Line Dwst Isol Vlv |
| 1SI8809B | SI RX HX 1B Dsch Line Dwst Isol Vlv |
| 1SI8811A | SI Cnmt Sump A Outlet Isol Vlv |
| 1SI8811B | SI Cnmt Sump B Outlet Isol Vlv |
| 1SI8812A | SI Rwst to RH Pp 1A Outlet Isol Vlv |
| 1SI8812B | SI Rwst to RH Pp 1B Outlet Isol Vlv |
| 1SI8813 | SI Pumps 1A-1B Recirc Line Dwst Isol |
| 1SI8814 | SI Pump 1A Recirc Line Isol Vlv |
| 1SI8835 | SI Pumps X-tie Disch Isol Vlv |
| 1SI8840 | SI RHR HX Disch Line Upstrm Cont Pen Isl Vlv |
| 1SI8821A | SI PP 1A Disch Line X-tie Isol Vlv |
| 1SI8821B | SI Pump 1B Disch Line X-tie Isol Vlv |
| 1SI8920 | SI Pump 1B Recirc Line Isol Vlv |
| 1SI8923A | SI PP 1A Suction Isol Vlv |
| 1SI8923B | SI Pump 1B Suct Isol Valve |
| 1SI8924 | SI Pump 1A Suction X-tie Dwnstrm Isol Vlv |
| 1SX016B | RCFC B&D Sx Supply MOV |
| 1SX016A | RCFC A&C SX Supply MOV |
| 1SX027A | RCFC A&C Return |
| 1SX027B | RCFC B&D SX Return MOV |
| 0SX007 | CC HX Outlet Vlv |
| 0SX063A | SX to Cont Rm Refrig Cdsr OA |
| 0SX063B | SX to Cont Rm Refrig Cdsr OB |
| 0SX146 | CC Hx "0" return Vlv to Unit 1 MDCT |
| 0SX157A | SX M/U Pp OA Supply Fill to MDCT |
| 0SX157B | SX M/U Pp OB Supply to MDCT OB MOV |
| 0SX158A | SX M/U Pp OA Supply Fill to MDCT MOV |
| 0SX158B | SX M/U Pp OB Supply to MDCT OB MOV |
| 0SX162A | MDCT OA Bypass to basin MOV |
| 0SX162B | MDCT OB Bypass to basin MOV |

REFUELING OPERATIONS

## 3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

### LIMITING CONDITION FOR OPERATION

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

a. The personnel hatch should have a minimum of one door closed at any one time and the equipment hatch shall be in place and held by a minimum of four bolts or the equipment hatch removed pursuant to Surveillance Requirement 4.9.4.2,

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

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

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

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

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

ACTION:

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

### SURVEILLANCE REQUIREMENTS

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

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

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

*Not applicable prior to initial criticality.

SURVEILLANCE REQUIREMENTS (Continued)

4.9.4.2 Verifying that the fuel handling building exhaust filter plenums maintain the fuel building at a negative pressure of greater than or equal to 1/4 inch water gauge relative to the outside atmosphere with the equipment hatch removed,

a. Prior to CORE ALTERATIONS or movement of irradiated fuel and

b. At least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel.

Verification of negative pressure will be performed with systems in the normal REFUELING MODE.

## 3/4.9.5 COMMUNICATIONS

### LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the containment refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the containment refueling station cannot be maintained, suspend all CORE ALTERATIONS.

### SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the containment refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

## 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 capacity equal to or greater than 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 capacity equal to or greater than 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 refueling machine and/or hoist OPERABILITY not satisfied, suspend use of any inoperable refueling machine 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 refueling machine 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 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 2500 pounds.

## 3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE FACILITY

### LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage facility.

APPLICABILITY: With fuel assemblies in the spent fuel storage facility.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

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

### SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of 2000 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.*

*Physical stops are not required until October 31, 1985.

## 3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

## HIGH WATER LEVEL

### LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective ACTION to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

### SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

*The RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

## LOW WATER LEVEL

### LIMITING CONDITION FOR OPERATION

3.9.8.2 Two residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

### SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

*Prior to initial criticality, the RHR loop may be removed from operation for up to 1 hour per 2 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

## 3/4.9.9 CONTAINMENT PURGE ISOLATION SYSTEM

### LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Purge Isolation System shall be OPERABLE.

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

ACTION:

a. With the Containment Purge Isolation System inoperable, close each of the purge valves providing direct access from the containment atmosphere to the outside atmosphere.

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

### SURVEILLANCE REQUIREMENTS

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

## 3/4.9.10 WATER LEVEL - REACTOR VESSEL

### LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

### SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

## 3/4.9.11 WATER LEVEL - STORAGE POOL

### LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

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

### SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

## 3/4.9.12 FUEL HANDLING BUILDING EXHAUST FILTER PLENUMS

### LIMITING CONDITION FOR OPERATION

3.9.12* Two independent Fuel Handling Building Exhaust Filter Plenums shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool

ACTION:

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

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

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

### SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel Handling Building Exhaust Filter Plenums shall be demonstrated OPERABLE:

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

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

*Not applicable prior to MODE 1 because of the low fission product inventory available at or below the limiting power level of 5%.

SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that the Fuel Handling Building Exhaust Filter Plenum satisfies the in-place penetration testing acceptance criteria of less than 1% when using the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the flow rate is 21,000 cfm ± 10%;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, and by showing a methyl iodide penetration of less than 10% when tested at a temperature of 30°C and a relative humidity of 95%.

3) Verifying a flow rate of 21,000 cfm ± 10% through the Fuel Handling Building Exhaust Filter Plenum during operation when tested in accordance with ANSI N510-1980; and

4) Verifying that with the system operating at a flow rate of 21,000 cfm ±10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the system and the leakage is less than or equal to 1% when the system is tested by injecting cold DOP at the system intake and the damper leakage rate is determined by either direct measurements or pressure decay measurements at a test pressure of 2 inches of water and the auxiliary building exhaust fans are operating at their rated flow.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 10% when tested at a temperature of 30°C and a relative humidity of 95%.

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the exhaust filter plenum at a flow rate of 21,000 cfm ± 10%;

2) Verifying that on a Safety Injection or a High Radiation test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks; and

SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that the Fuel Handling Building Exhaust Filter Plenum maintains the fuel building at a negative pressure of greater than or equal to 1/4 inch Water Gauge relative to the outside atmosphere during operation.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the Fuel Handling Building Exhaust Filter Plenum satisfies the in-place penetration testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 fo a DOP test aerosol while operating the system at a flow rate of 21,000 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the Fuel Handling Building Exhaust Filter Plenum satisfies the in-place penetration testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 21,000 cfm ± 10%.

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of the following requirements may be suspended:

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

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

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

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

APPLICABILITY: During operation below the P-7 Interlock Setpoint or performance of hot rod drop time measurements.

ACTION:

a. With the THERMAL POWER greater than the P-7 Interlock Setpoint during the performance of startup and PHYSICS TESTS, immediately open the Reactor trip breakers.

b. With less than the above required reactor coolant loops OPERABLE during performance of hot rod drop time measurements, immediately open the reactor trip breakers and comply with the provisions of the ACTION statements of Specification 3.4.1.2.

SURVEILLANCE REQUIREMENTS

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

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

4.10.4.3 At least the above required reactor coolant loops shall be determined OPERABLE within 4 hours prior to initiation of the hot rod drop time measurements and at least once per 4 hours during the hot rod drop time measurements by verifying correct breaker alignments and indicated power availability and by verifying secondary side narrow range water level to be greater than or equal to 41%.

TABLE 4.11-1 (Continued)

TABLE NOTATIONS

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

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

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the lower limit of detection (microCuries per unit mass or volume),

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

E = the counting efficiency (counts per disintegration),

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

$2.22 \times 10^6$ = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

$\lambda$ = the radioactive decay constant for the particular radionuclide ($sec^{-1}$), and

$\Delta t$ = the elapsed time between the midpoint of sample collection and the time of counting (sec).

Typical values of E, V, Y, and $\Delta t$ should be used in the calculation.

It should be recognized that the LLD is defined as a before the fact limit representing the capability of a measurement system and not as an after the fact limit for a particular measurement.

(2) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed by a method described in the ODCM to assure representative sampling.

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.

A rod bow penalty is not applied to the final value of $F^N_{\Delta H}$ for the following reason:

$F^N_{\Delta H}$ will be maintained within its limits provided the Conditions a. through d. above are maintained. The combination of the RCS flow requirement (399,000 gpm) and the requirement on $F^N_{\Delta H}$ guarantee that the DNBR used in the safety analysis will be met.

Fuel rod bowing does reduce the value of the DNBR. However, predictions with the methods described in WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," July 1979 for the 17x17 Optimized Fuel Assemblies indicate that the fuel rod bow reduction on DNBR will be less than 3% at 33,000 MWD/MTU assembly average burnup. At higher burnups, the decrease in fissionable isotopes and the buildup of fission product inventory more than compensate for the rod bow reduction in DNBR.

There is a 11% margin available between the 1.32 and 1.34 design DNBR limits and the 1.47 and 1.49 safety analysis DNBR limit. Use of the 3% fuel rod bow DNBR margin reduction still leaves a 8% margin in DNBR between design limits and safety analysis limits.

The RCS flow requirement is based on the loop flow rate of 97,600 gpm which is used in the Improved Thermal Design Procedure described in FSAR 4.4.1 and 15.0.3. This design value is then increased by 2.2% for measurement uncertainties. The measurement error for RCS total flow rate is based on 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 results from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% has been included in the 2.2% measurement uncertainty of the RCS flow rate. 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.

Surveillance Requirement 4.2.3.4 provides adequate monitoring to detect possible flow reductions due to any rapid core crud buildup.

Surveillance Requirement 4.2.3.5 specifies that the measurement instrumentation shall be calibrated within seven days prior to the performance of the calorimetric flow measurement. This requirement is due to the fact that the drift effects of this instrumentation are not included in the flow measurement uncertainty analysis. This requirement does not apply for the instrumentation whose drift effects have been included in the uncertainty analysis.

## Engineered Safety Features Actuation System Interlocks

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on $T_{avg}$ below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressure P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure and low steamline pressure and automatically blocks steamline isolation on negative steamline pressure rate. On decreasing pressure; P-11 allows the manual block of Safety Injection low pressurizer pressure and low steamline pressure and allows steamline isolation on negative steamline pressure rate to become active upon manual block of low steamline pressure SI.

P-12 On increasing reactor coolant loop temperature, P-12 automatically provides an arming signal to the Steam Dump System. On decreasing reactor coolant loop temperature, P-12 automatically removes the arming signal from the Steam Dump System.

P-14 An increasing steam generator water level, P-14 automatically trips all feedwater isolation valves and inhibits feedwater control valve modulation.

## 3/4.3.3 MONITORING INSTRUMENTATION

### 3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel reaches its Setpoint and (2) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the system sends actuation signals to initiate alarms and automatic actuation of Emergency Exhaust or Ventilation Systems. The radiation monitor Setpoints given in the requirements are assumed to be values established above normal background radiation levels for the particular area. Radiation monitors ORE-AR055 and 56 serve a dual purpose for plant operations as criticality and fuel handling accident sensors. Although these monitors are designed primarily to detect fuel handling accident releases, they are capable of detecting an inadvertent criticality incident. The Setpoint given in the requirement is established for the fuel handling building isolation function but is also adequate for an inadvertent criticality.

Fire detectors that are used to actuate Fire Suppression Systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of operable fire detectors must be greater.

The loss of detection capability for Fire Suppression Systems, actuated by fire detectors represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

## 3/4.3.3.8 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the Reactor Coolant System and avoid or mitigate damage to Reactor Coolant System components. The allowable out-of-service times and Surveillance Requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

## 3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

## 3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20.

SPECIFIC ACTIVITY (Continued)

take corrective ACTION. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomenon. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

## 3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

    a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

    b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200° F/hr respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F, and

5. System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the 1973 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel and Code.

PRESSURE/TEMPERATURE LIMITS (Continued)

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, $RT_{NDT}$, at the end of 32 effective full power years of service life. The 32 EFPY service life period is chosen such that the limiting $RT_{NDT}$ at the 1/4T location in the core region is greater than the $RT_{NDT}$ of the limiting unirradiated material. The selection of such a limiting $RT_{NDT}$ assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial $RT_{NDT}$; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the $RT_{NDT}$. Therefore, an adjusted reference temperature, based upon the fluence, copper content and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of $\Delta RT_{NDT}$ computed by either Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials" or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in $RT_{NDT}$ at the end of 32 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of $\Delta RT_{NDT}$ determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict the future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the $\Delta RT_{NDT}$ determined from the surveillance capsule exceeds the calculated $\Delta RT_{NDT}$ for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.

PRESSURE/TEMPERATURE LIMITS (Continued)

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, $RT_{NDT}$, is used and this includes the radiation-induced shift, $\Delta RT_{NDT}$, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, $K_I$, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, $K_{IR}$, for the metal temperature at that time. $K_{IR}$ is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The $K_{IR}$ curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

Where: $K_{IR}$ is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature $RT_{NDT}$ Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where: $K_{IM}$ = the stress intensity factor caused by membrane (pressure) stress,

$K_{It}$ = the stress intensity factor caused by the thermal gradients,

PRESSURE/TEMPERATURE LIMITS (Continued)

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

3/4.4.11 REACTOR VESSEL HEAD VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of a reactor vessel head vent path ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System Vent Systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

CONTAINMENT VENTILATION SYSTEM (Continued)

be exceeded in the event of an accident during containment purging operation. Operation with one line open will be limited to 1000 hours during a calendar year.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 $L_a$ leakage limit of Specification 3.6.1.2.b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valv s are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Cooling System are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses. A spray additive tank level of between 78.6% and 90.3% ensures a volume of greater than or equal to 4000 gallons but less than or equal to 4540 gallons.

ULTIMATE HEAT SINK (Continued)

A redundant makeup system using deep wells as a water source is designed to withstand design basis tornado events and river flood events. The second redundant system is the essential service water makeup system that uses the Rock River as a water source. It is designed to withstand all other design basis natural phenomena events and combinations of events except for seismic events during low Rock River flow rates. Since the deep well makeup system is not designed to withstand seismic events, no engineered safety feature is then available for providing makeup.

Thus, provision of the design basis ultimate heat sink function for 30 days is dependent upon successful implementation of plant procedures to provide makeup from alternative sources. To assure that the requirements of the Commission's General Design Criterion 44, "Cooling Water" are met, a technical specification requiring Commission notification of how the procedures will be implemented for the particular contingency is necessary.

Each essential service water makeup pump is powered by a diesel engine with a fuel supply adequate for approximately 3 days of operation. Achievement of the design basis 30-day operation is dependent upon successful implementation of plant procedures to replenish the fuel supply following design basis events

With water in the cooling tower basin at an initial temperature less than or equal to 80°F, shutdown can be achieved, for meteorological conditions following a design basis tornado, without operation of the cooling tower fans and without the temperature of the water discharged from the essential service water pump exceeding 110°F, the maximum acceptable temperature for components and systems cooled by the essential cooling water system. Achievement of the design function of the UHS during more severe meteorological conditions, or following a design basis LOCA, requires operation of one cooling tower fan to maintain the discharge temperature less than or equal to 110°F. Plant procedures ensure that the fans are started and controlled by operator action to maintain a discharge temperature of less than 110°F.

3/4.7.6 CONTROL ROOM VENTILATION SYSTEM

The OPERABILITY of the Control Room Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

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

The OPERABILITY of the Non-Accessible Area Exhaust Filter Plenum Ventilation System ensures that radioactive materials leaking from the ECCS equipment within the pump rooms following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

### 3/4.7.8 SNUBBERS

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

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of systems affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Onsite Review and Investigative Function. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures on a given system and is determined by the number of inoperable snubbers found during an inspection of each system. In order to establish the inspection frequency for each type of snubber on a safety-related system, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber on that system could cause the system to be unprotected and to result in failure during an initiating event. Inspections performed before that interval has

SEALED SOURCE CONTAMINATION (Continued)

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

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, foam, 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. 3750 gpm pump capacity is based on NFPA 20, 1981 standards which call for 150% of rated capacity (2500 gpm) at 65% of 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.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 FUEL HANDLING BUILDING EXHAUST FILTER PLENUM

The limitations on the Fuel Handling Building Exhaust Filter Plenum ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing.