OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-40

1. The U.S. Nuclear Regulatory Commission (the Commission) having previously made the findings set forth in License No. DPR-40 issued August 9, 1973, has now found that:

   A. The application to renew License No. DPR-40 filed by Omaha Public Power District (the licensee) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter 1, and all required notifications to other agencies or bodies have been duly made;

   B Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1), and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this renewed license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for Fort Calhoun Station, Unit No. 1, and that any changes made to the plant's current licensing basis in order to comply with 10 CFR 54.29(a) are in accord with the Act and the Commission's regulations;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   D. There is reasonable assurance: (1) that the activities authorized by this renewed license can be conducted without endangering the health and safety of the public, and (2) that such activities will be conducted in compliance with the rules and regulations of the Commission;

   E. Omaha Public Power District is technically qualified and financially qualified to engage in the activities authorized by this renewed license in accordance with the rules and regulations of the Commission;

   F. Omaha Public Power District has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   G. The issuance of this renewed license will not be inimical to the common defense and security or to the health and safety of the public;

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the Commission concludes that the issuance of Renewed Operating License No. DPR-40 is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by the renewed license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40, and 70, including but not necessarily limited to 10 CFR Sections 30.33, 40.32, 70.23 and 70.31.

2. On the basis of the forgoing findings regarding this facility, Facility Operating License No. DPR-40, issued August 9, 1973, is superseded by Renewed Facility Operating License No. DPR-40, which is hereby issued to the Omaha Public Power District, to read as follows:

A. This renewed license applies to the Fort Calhoun Station, Unit 1, a pressurized water nuclear reactor and associated equipment (the facility), which is owned by the Omaha Public Power District. The facility is located in Washington County, Nebraska, and is described in the Updated Safety Analysis Report as supplemented and amended and the Environmental Report as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Omaha Public Power District:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location in Washington County, Nebraska in accordance with the procedures and limitations set forth in this renewed license;

(2) Pursuant to the Act and 10 CFR Parts 40 and 70, to receive, possess, and use at any time source and special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or instrument calibration or when associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is, subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

Omaha Public Power District is authorized to operate the Fort Calhoun Station, Unit 1, at steady state reactor core power levels not in excess of 1500 megawatts thermal (rate power).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 289 are hereby incorporated in the license. Omaha Public Power District shall operate the facility in accordance with the Technical Specifications.

C. Security and Safeguards Contingency Plans

The Omaha Public Power District shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Fort Calhoun Station Security Plan, Training and Qualification Plan, Safeguards Contingency Plan," submitted by letter dated May 19, 2006.

OPPD shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The OPPD CSP was approved by License Amendment No. 266 and modified by License Amendment No. 284.

D. Fire Protection Program

Omaha Public Power District shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated September 28, 2011 (and supplements dated December 19, 2011, December 22, 2011, March 20, 2012, July 24, 2012, August 24, 2012, September 27, 2012, April 23, 2013, May 21, 2013, July 29, 2013, September 12, 2013, October 11, 2013, November 4, 2013, November 11, 2013, December 18, 2013, January 24, 2014, February 28, 2014, April 10, and June 11, 2014) and as approved in the safety evaluation dated June 16, 2014. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

(1) Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

(a) Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(b) Prior NRC review and approval is not required for individual changes that result in a risk increase less than $1 \times 10^{-7}$/year (yr) for core damage frequency (CDF) and less than $1 \times 10^{-8}$/yr for large early release frequency (LERF). The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(2) Other Changes that May Be Made Without Prior NRC Approval

(a) Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- "Fire Alarm and Detection Systems" (Section 3.8);
- "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
- "Gaseous Fire Suppression Systems" (Section 3.10); and,
- "Passive Fire Protection Features" (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

(b) Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated June 16, 2014, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program

(3) Transition License Conditions

(a) Before achieving full compliance with 10 CFR 50.48(c), as specified by D.(3)(b) and D.(3)(c) below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in D.(2)(b) above.

(b) The licensee shall implement the modifications to its facility, as described in Enclosure 1, Attachment S, Table S-2, "Plant Modifications Committed," of OPPD letter LIC-14-0042 dated April 10, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) by the end of the second refueling outage following issuance of the license amendment. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

(c) The licensee shall implement the items listed in Enclosure 1, Attachment S, Table S-3, "Implementation Items," of OPPD letter LIC-14-0042, dated April 10, 2014, no later than 12 months after issuance of the license amendment.

E. Updated Final Safety Analysis Report

The Omaha Public Power District Updated Final Safety Analysis Report supplement, submitted pursuant to 10 CFR 54.21 (d), describes certain future activities to be completed prior to the period of extended operation. The Omaha Public Power District shall complete these activities no later than August 9, 2013, and shall notify the NRC In writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement, as revised, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71 (e)(4) following issuance of this renewed license. Until that update is complete, the Omaha Public Power District may make changes to the programs and activities described in the supplement without prior Commission approval, provided that the Omaha Public Power District evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

F. Appendix B

The Additional Conditions contained in Appendix B, as revised through Amendment No. 286, are hereby incorporated into this license. Omaha Public Power District shall operate the facility in accordance with the Appendix B Additional Conditions.

Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

4. This renewed license is effective as of the date of issuance and shall expire at midnight on August 9, 2033.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
J. E. Dyer

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments: 1. Appendix A - Technical Specifications
2. Appendix B - Additional Conditions

Date of Issuance: November 4, 2003

APPENDIX A

TO

OPERATING LICENSE NO. DPR-40

RADIOLOGICAL SAFETY TECHNICAL SPECIFICATION

FOR THE

FORT CALHOUN STATION UNIT NO. 1

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

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Amendment No. ~~32,34,43,54,55,57,~~
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# TECHNICAL SPECIFICATIONS

## DEFINITIONS

The following terms are defined for uniform interpretation of these Specifications.

## REACTOR OPERATING CONDITIONS

### Rated Power

A steady state reactor core output of 1500 MWt.

### Reactor Critical

The reactor is considered critical for purposes of administrative control when the neutron flux logarithmic range channel instrumentation indicates greater than $10^{-4}$% of rated power.

### Power Operation Condition (Operating Mode 1)

The reactor is in the power operation condition when it is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

### Hot Standby Condition (Operating Mode 2)

The reactor is considered to be in a hot standby condition if the average temperature of the reactor coolant (Tavg) is greater than 515°F, the reactor is critical, and the neutron flux power range instrumentation indicates less than 2% of rated power.

### Hot Shutdown Condition (Operating Mode 3)

The reactor is in a hot shutdown condition if the average temperature of the reactor coolant (Tavg) is greater than 515°F and the reactor is subcritical by at least the amount defined in Paragraph 2.10.2.

## DEFINITIONS

**REACTOR OPERATING CONDITIONS** (Continued)

Cold Shutdown Condition (Operating Mode 4)

The reactor coolant $T_{cold}$ is less than 210°F and the reactor coolant is ≥ SHUTDOWN BORON CONCENTRATION but < REFUELING BORON CONCENTRATION.

Refueling Shutdown Condition (Operating Mode 5)

The reactor coolant $T_{cold}$ is less than 210°F and the reactor coolant is ≥ REFUELING BORON CONCENTRATION.

Refueling Operation

Any operation involving the shuffling, removal, or replacement of irradiated fuel outside of the reactor pressure vessel. The suspension of any REFUELING OPERATION shall not preclude completion of movement of a component to a safe, conservative position.

The Refueling Boron Concentration

A reactor coolant boron concentration of at least that specified in the CORE OPERATING LIMITS REPORT which corresponds to a shutdown margin of not less than 5% with all CEA's withdrawn.

Shutdown Boron Concentration

The boron concentration required to make the reactor subcritical by the amount defined in Section 2.10.

Refueling Outage or Refueling Shutdown

A plant outage or shutdown to perform refueling operations upon reaching the planned fuel depletion for a specific core.

Plant Operating Cycle

The time period from a REFUELING SHUTDOWN to the next REFUELING SHUTDOWN.

## DEFINITIONS

## REACTOR OPERATING CONDITIONS (Continued)

### Physics Testing

Testing performed under written procedures approved by Plant Operations Review Committee to determine CEA worths and other core nuclear parameters. Deviations from normal operating practice which are necessary to enable some of these tests to be performed are permitted in accordance with: 1) the specific provisions of these technical specifications, 2) authorization under the provisions of 10 CFR 50.59, or 3) other approval of the Commission.

## PROTECTIVE SYSTEMS

### Reactor Trip

The de-energizing of the CEDM magnetic clutch holding coils which releases the CEA's and allows them to drop into the core.

### Instrument Channel

One of four independent measurement channels complete with the sensors, sensor power supply units, amplifiers, and trip modules provided for each safety parameter.

### Reactor Protective System Logic(1)

The system which utilizes relay contact outputs from individual instrument channels to provide the reactor trip signal for de-energizing the magnetic clutch power supplies. The logic system is wired to provide a reactor trip on a 2-of-4 or 2-of-3 basis for any given input parameter.

**PROTECTIVE SYSTEMS** (Continued)

Engineered Safety Feature Logic[2]

The system which utilizes relay contact outputs from individual instrument channels to provide a dual channel signal to independently initiate the actuation of the engineered safety feature equipment. Two logic subsystems, termed A and B, are provided; each subsystem is composed of four channels wired to provide independent safety feature initiation signals on a 2-out-of-4 basis (Containment Radiation High Signal is 1-out-of-2 logic).

Degree of Redundancy

The difference between the number of operable channels and the number of channels which when tripped will cause an automatic system trip.

**INSTRUMENTATION SURVEILLANCE**

Channel Check

A qualitative determination of acceptable operability by observation of channel behavior during normal plant operation. This determination shall where feasible, include comparison of the channel with other independent channels measuring the same variable.

Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including any alarm and/or trip initiating action.

Channel Calibration

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment action, alarms, interlocks or trip, and shall be deemed to include the channel functional test.

## DEFINITIONS

## MISCELLANEOUS DEFINITIONS

Operable - Operability

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power sources, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its specified safety function(s) are also capable of performing their related support function(s).

In Operation

A system or component is IN OPERATION if it is OPERABLE and is performing its design function.

CEA's

All full length shutdown and regulating control rods.

Non-trippable (NT) CEA's

CEA's which are non-trippable.

Containment Integrity

Containment integrity is defined to exist when all of the following are met:

(1) All nonautomatic containment isolation valves which are not required to be open during accident conditions and blind flanges, except for valves that are open under administrative control as permitted by Specification 2.6(1)a, are closed.

(2) The equipment hatch is properly closed and sealed.

(3) The personnel air lock satisfies Specification 2.6(1)b.

(4) All automatic containment Isolation valves are operable, locked closed, or deactivated and secured in their closed position (or isolated by locked closed valves or blind flanges as permitted by a limiting condition for operation).

(5) The uncontrolled containment leakage satisfies Specification 3.5, and

(6) The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is operable.

## DEFINITIONS

### Core Alteration

The movement or manipulation of fuel, sources, reactivity control components, or other components affecting reactivity within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe, conservative position.

### Equivalent Full Power Day (EFPD)

The time interval during power operation when the heat generated by the reactor is equivalent to reactor operation at 100% of rated power for 24 hours.

### Shutdown Margin

Shutdown Margin shall be the amount of reactivity by which:

(1) the reactor is subcritical; or

(2) the instantaneous amount of reactivity by which the reactor would be subcritical from its present condition assuming:

a. All known trippable full length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn, and

b. No change in non-trippable control element assembly position.

### Axial Shape Index

The external AXIAL SHAPE INDEX ($Y_E$) is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The internal AXIAL SHAPE INDEX ($Y_I$) used for the trip and pre-trip signals in the reactor protection system is the above value ($Y_E$) modified by the shape annealing factor, SAF, and a constant, B, to determine the true core axial power distribution for that channel.

$$Y_E = \frac{L-U}{L+U} \qquad Y_I = SAF \times Y_E + B$$

## DEFINITIONS

### Azimuthal Power Tilt - $T_q$

Azimuthal Power Tilt shall be the power asymmetry between azimuthally symmetric fuel assemblies.

### Maximum Radial Peaking Factor ($F_R^T$)

The Maximum Radial Peaking Factor is the maximum ratio of the individual fuel pin power to the core average pin power integrated over the total core height, including tilt. The $F_R^T$ limit is provided in the Core Operating Limits Report.

### Dose Equivalent I-131

That concentration of I-131 (ΦCi/gm) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. In other words,

| | | |
|---|---|---|
| Dose Equivalent I-131 (ΦCi/gm) | = | ΦCi/gm of I-131 |
| | + | 0.0361 x ΦCi/gm of I-132 |
| | + | 0.270 x ΦCi/gm of I-133 |
| | + | 0.0169 x ΦCi/gm of I-134 |
| | + | 0.0838 x ΦCi/gm of I-135 |

## DEFINITIONS

Ē - Average Disintegration Energy

Ē is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration, in MEV, for isotopes, other than iodines, with half lives greater than 15 minutes making up at least 95% of the total non-iodine radioactivity in the coolant.

Offsite Dose Calculation Manual (ODCM)

The document(s) that contain the methodology and parameters used in the calculations of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent radiation monitoring Warn/High (trip) Alarm setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain:

1) The Radiological Effluent Controls and the Radiological Environmental Monitoring Program required by Specification 5.16.

2) Descriptions of the information that should be included in the Annual Radiological Environmental Operating Reports and Annual Radioactive Effluent Release Reports required by Specifications 5.9.4.a and 5.9.4.b.

Unrestricted Area

Any area at or beyond the site boundary access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

Core Operating Limits Report (COLR)

The Core Operating Limits Report (COLR) is a Fort Calhoun Station Unit No. 1 specific document that provides core operating limits for the current operating cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Section 5.9.5. Plant operation within these operating limits is addressed in the individual specifications.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE
   1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal leakoff), that is captured and conducted to collection systems or a sump or collecting tank,
   2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE, or
   3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE),

**DEFINITIONS**

b. Unidentified LEAKAGE
All LEAKAGE (except RCP seal leakoff) that is not identified LEAKAGE, and

c. Pressure Boundary LEAKAGE
LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

RCS Pressure-Temperature Limits Report (PTLR)

The PTLR is a fluence dependent document that provides Limiting Conditions for Operation (LCO) in the form of pressure-temperature (P-T) limits to ensure prevention of brittle fracture. In addition, this document establishes power operated relief valve setpoints which provide low temperature overpressure protection (LTOP) to assure the P-T limits are not exceeded during the most limiting LTOP event. The P-T limits and LTOP criteria in the PTLR are applicable through the effective full power years (EFPYs) specified in the PTLR. NRC approved methodologies are used as the bases for the information provided in the PTLR.

References

(1) USAR, Section 7.2
(2) USAR, Section 7.3

# 1.0 SAFETY LIMITS

## 1.1 Safety Limits (SLs)

### 1.1.1 Reactor Core SLs

Applicability

This specification applies to the limiting combinations of reactor power and reactor coolant system flow, temperature and pressure during operation.

Objective

To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the reactor coolant.

Specifications

(a) The reactor power level shall not exceed the allowable limit for the pressurizer pressure and the cold leg temperatures as shown in Figure 1-1 for 4-pump operation. The safety limit is exceeded if the point defined by the combination of reactor coolant cold leg temperature and power level is at any time above the appropriate pressurizer pressure line.

(b) Peak fuel centerline temperature shall be maintained at < 5081°F, decreasing by 58°F per 10,000 MWD/MTU and adjusted for burnable poison per XN-NF-79-56(P)(A), Revision 1, Supplement 1.

### 1.1.2 Reactor Coolant System Pressure SL

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity to the containment.

Specification

The reactor coolant system pressure shall not exceed 2750 psia when fuel assemblies are located within the reactor vessel.

# 1.0 SAFETY LIMITS

## 1.1 Safety Limits (SLs) (continued)

### Basis

To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the reactor coolant, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB).

At DNB there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperature and the possibility of clad failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of reactor thermal power and reactor coolant flow, temperature and pressure can be related to DNB through a correlation. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients corresponds to a 95% probability at a 95% confidence level that DNB will not occur, which is considered an appropriate margin to DNB for all operating conditions.[1]

The curves of Figure 1-1 represent the loci of points for reactor thermal power (either neutron flux instruments or $\Delta T$ instruments), reactor coolant system pressure, and cold leg temperature for which the minimum DNBR is not less than the minimum DNBR limit. The area of safe operation is below these lines.

SL 1.1.1(b) ensures that fuel centerline temperature remains below the fuel melt temperature 5081°F during normal operating conditions or design anticipated operational occurrences (AOOs) with adjustments for burnup and burnable poison. An adjustment of 58°F per 10,000 MWD/MTU has been established in XN-NF-82-06(P)(A), Revision 1, Supplements 2, 4 and 5 (Ref. 8) and adjustments for burnable poisons are established based on XN-NF-79-56(P)(A), Revision 1, Supplement 1 (Ref. 9).

The reactor core safety limits are based on radial peaks limited by the CEA insertion limits in Section 2.10 and axial shapes within the axial power distribution trip limits in the COLR. The Thermal Margin/Low Pressure trip requirements shall be within the limits provided in the COLR. The Thermal Margin/Low Pressure trip is based on an unrodded integrated total radial peak ($F_R^T$) that is provided in the COLR.

## 1.0 SAFETY LIMITS

### 1.1 Safety Limits (SLs) (continued)

Flow maldistribution effects for operation under less than full reactor coolant flow have been evaluated via model test.[2] The flow model data established the maldistribution factors and hot channel inlet temperature for the thermal analyses that were used to establish the safe operating envelopes presented in Figure 1-1. The reactor protective system is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than the minimum DNBR limit.[1]

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

The reactor coolant system serves as a barrier to prevent radionuclides in the reactor coolant from reaching the containment atmosphere.[3] In the event of a fuel cladding failure, the reactor coolant system is the primary barrier against the release of fission products. Establishing a system pressure limit helps to assure the continued integrity of the reactor coolant system and fuel cladding. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, section III, is 110% of design pressure. The maximum transient pressure allowable in the reactor coolant system piping, valves and fittings under USAS section B31.1 is 120% of design pressure. Thus, the safety limit of 2750 psia (110% of the 2500 psia design pressure) has been established.[4]

The settings and capacity of the main steam safety valves (1000 - 1050 psia)[5], the reactor high-pressure trip (≤ 2400 psia) and the reactor coolant system safety valves (2500-2545 psia)[6] have been established to assure never reaching the reactor coolant system pressure safety limit. The initial hydrostatic test pressure was conducted at 3125 psia (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the nuclear steam supply system (NSSS) pressure does not exceed the safety limit is provided by setting the pressurizer power-operated relief valves, consistent with the reactor high pressure trip, and opening the steam system steam dump and bypass valves upon receipt of a turbine trip signal.[7]

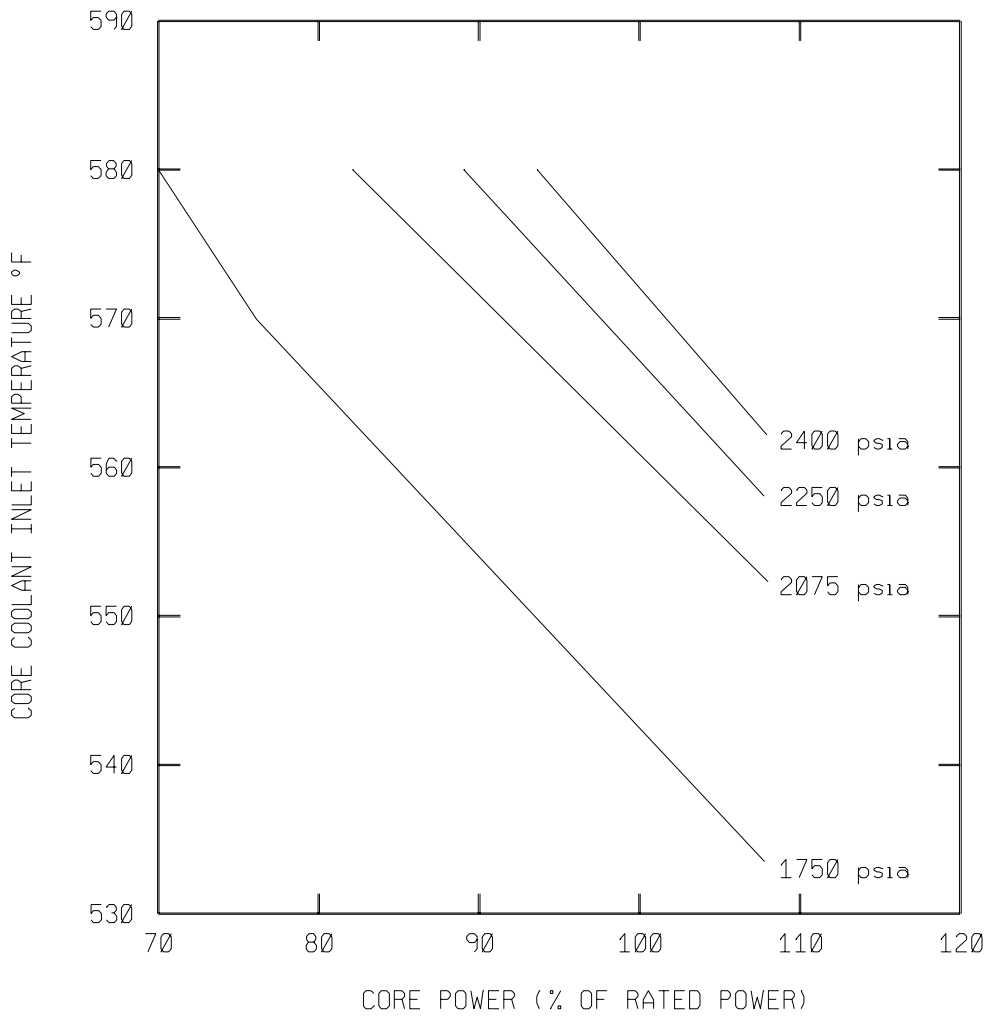
References

(1) USAR, Section 3.6.6
(2) USAR, Section 1.4.6
(3) USAR, Section 4
(4) USAR, Section 4.3.3
(5) USAR, Section 4.3.4
(6) USAR, Section 4.3.9.5
(7) USAR, Section 7.4.5.1
(8) XN-NF-82-06(P)(A), Revision 1, Supplements 2, 4 and 5, “Qualification of Exxon Nuclear Fuel for Extended Burnup,” October 1986
(9) XN-NF-79-56(P)(A), Revision 1, Supplement 1, “Gadolinia Fuel Properties of LWR Fuel Safety Evaluation,” November 1981

1.0 **SAFETY LIMITS**

1.1 Safety Limits (SLs) (continued)

**Figure 1-1**




| Thermal Margin/Low Pressure Safety Limits 4 Pump Operation | Omaha Public Power District Fort Calhoun Station-Unit No. 1 | Figure 1-1 |
|---|---|---|

# 1.0 **SAFETY LIMITS**

## 1.2 Safety Limit Violations

1.2.1 If Safety Limit 1.1.1 is violated, restore compliance and be in at least HOT SHUTDOWN within 1 hour.

1.2.2 If Safety Limit 1.1.2 is violated:

1.2.2.1 In MODE 1 or 2, restore compliance and be in at least HOT SHUTDOWN within 1 hour.

1.2.2.2 In MODES 3, 4, or 5, restore compliance within 5 minutes.

Basis

SL 1.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be operable during MODES 1 and 2 to ensure operation within the reactor core SLs. Applicability is not required in those operating MODES when the reactor is not generating significant thermal power.

If SL 1.1.1 is violated, the requirement to go to HOT SHUTDOWN places the unit in a MODE in which the SL is not applicable. The allowed completion time of 1 hour recognizes the importance of bringing the unit to a MODE where this SL is not applicable and reduces the probability of fuel damage.

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in HOT SHUTDOWN within 1 hour. The allowed completion time of 1 hour provides the operator time to complete the necessary actions to reduce RCS pressure by terminating the cause of the pressure increase, removing mass or energy from the RCS, or a combination of these actions and to establish HOT SHUTDOWN conditions.

If the RCS pressure SL is exceeded in MODES 3, 4, or 5, RCS pressure must be restored within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODES 3, 4 or 5 is potentially more severe than exceeding this SL in MODES 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

# 2.0 LIMITING CONDITIONS FOR OPERATION

## 2.0.1 General Requirements

### Applicability

Applies to the operable status of all systems, subsystems, trains, components, or devices covered by the Limiting Conditions for Operation.

### Objective

To specify corrective measures to be employed for system conditions not covered by or in excess of the Limiting Conditions for Operation.

### Specification

(1) In the event a Limiting Condition for Operation and/or associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least HOT SHUTDOWN within 6 hours, in at least subcritical and < 300°F within the next 6 hours, and in at least COLD SHUTDOWN within the following 30 hours, unless corrective measures are completed that permit operation under the permissible action requirements for the specified time interval as measured from initial discovery or until the reactor is placed in an Operating Mode in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications.

(2) When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.0.1 General Requirements (Continued)

Basis

(1) This specification delineates corrective measures to be taken for circumstances not directly provided for in the system specific specifications and whose occurrence would violate the intent of the specification. For example, Specification 2.3 requires each Low Pressure Safety Injection (LPSI) pump to be operable and provides explicit corrective measures to be followed if one pump is inoperable. Under the terms of Specification 2.0.1(1), if more than one LPSI pump is inoperable, the unit must be placed in at least HOT SHUTDOWN within 6 hours, in at least subcritical and < 300°F within the following 6 hours, and in at least COLD SHUTDOWN within the following 30 hours, unless at least one LPSI pump were restored to operability. It is assumed that the unit is brought to the required mode within the required times by promptly initiating and carrying out the appropriate measures required by the specification.

(2) LCO 2.0.1(2) establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.

If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered.

LCO 2.0.1(2)a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 2.0.1(2)a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72-hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.0.1 General Requirements (Continued)

LCO 2.0.1(2)b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 2.0.1(2)b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12-hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.

LCO 2.0.1(2) requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 2.0.1(2) should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.

# 2.0 LIMITING CONDITIONS FOR OPERATION

## 2.1 Reactor Coolant System

### 2.1.1 Operable Components

Applicability

Applies to the operable status of the reactor coolant system components.

Objective

To specify certain conditions of the reactor coolant system components.

Specifications

Limiting conditions for operation are as follows:

(1) Reactor Critical

All four (4) reactor coolant pumps shall be in operation.

Exceptions

The limitations of this specification may be suspended during the performance of physics tests provided the power level is $\leq 10^{-1}$% of rated power and the flow requirements of Table 2-11 No. 2 are met.

(2) Hot Shutdown or 350°F $\leq T_{cold} \leq$ 515°F

(a) The reactor coolant loops listed below shall be operable:

(i) Reactor coolant loop 1 and at least one associated reactor coolant pump.

(ii) Reactor coolant loop 2 and at least one associated reactor coolant pump.

(b) At least one of the above reactor coolant loops shall be in operation.

Exceptions

All reactor coolant pumps may be de-energized for up to one hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.1 Operable Components (Continued)

(c) If fewer than the above required reactor coolant loops are OPERABLE, the required loops shall be restored to OPERABLE status within 72 hours or the reactor shall be placed in COLD SHUTDOWN within the next 12 hours.

(3) 210°F ≤ $T_{cold}$ ≤ 350°F or $T_{cold}$ < 210°F with fuel in the reactor and all reactor vessel head closure bolts fully tensioned.

(a) At least two (2) of the decay heat removal loops listed below shall be OPERABLE:

(i) Reactor coolant loop 1 and its associated steam generator and at least one associated reactor coolant pump.

(ii) Reactor coolant loop 2 and its associated steam generator and at least one associated reactor coolant pump.

(iii) One shutdown cooling pump, one shutdown cooling heat exchanger, and associated shutdown cooling piping.

(iv) One shutdown cooling pump, in addition to that in (iii) above, one shutdown cooling heat exchanger, in addition to that in (iii) above, and associated shutdown cooling piping.

(b) At least one (1) of the decay heat removal loops listed above shall be IN OPERATION.

(c) With no coolant loop IN OPERATION, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and initiate corrective action to return the required coolant loop to operation in 8 hours.

(d) For the purposes of items a(iii) and a(iv) above, the containment spray pumps can be considered as available shutdown cooling pumps only if both of the following conditions are met:

(i) Reactor Coolant System temperature is less than 120°F.

(ii) The Reactor Coolant System is vented with a vent area greater than or equal to 47 $in^2$.

(4) DELETED

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.1 Operable Components (Continued)

(5) DELETED

(6) Both steam generators shall be filled above the low steam generator water level trip set point and available to remove decay heat whenever the average temperature of the reactor coolant is above 300°F.

(7) Maximum reactor coolant system hydrostatic test pressure shall be 3125 psia. A maximum of 10 cycles of 3125 psia hydrostatic tests are allowed.

(8) Reactor coolant system leak and hydrostatic test shall be conducted within the limitations of the pressure and temperature limit Figure(s) shown in the PTLR.

(9) Maximum secondary hydrostatic test pressure shall not exceed 1250 psia. A minimum measured temperature of 73°F is required. Only 10 cycles are permitted.

(10) Maximum steam generator steam side leak test pressure shall not exceed 1000 psia. A minimum measured temperature of 73°F is required.

(11) Low Temperature Overpressure Protection (LTOP)

(a) The LTOP enable temperature and RCP operations shall be maintained in accordance with the PTLR.

(b) The unit can not be placed on shutdown cooling until the RCS has cooled to an indicated RCS temperature of less than or equal to 350°F.

(c) If no reactor coolant pumps are operating, a non-operating reactor coolant pump shall not be started while $T_c$ is below the LTOP enable temperature stated in the PTLR unless there is a minimum indicated pressurizer steam space of at least 50% by volume.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.1 Operable Components (Continued)

(12) Reactor Coolant System Pressure Isolation Valves

(a) The integrity of all pressure isolation valves listed in Table 2-9 shall be demonstrated, except as specified in (b). Valve leakage shall not exceed the amounts indicated.

(b) In the event that the integrity of any pressure isolation valve specified in Table 2-9 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a nonfunctional valve are in and remain in the mode corresponding to the isolated condition. Manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supply deenergized.

(c) If Specifications (a) and (b) above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

Basis

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation and maintain DNBR above the minimum DNBR limit during all normal operations and anticipated transients.

When Specification 2.1.1(2) is applicable, the reactor coolant pumps (RCPs) are used to provide forced circulation heat removal during heatup and cooldown. Under these conditions, decay heat removal requirements are low enough that a single reactor coolant system (RCS) loop with one RCP is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to provide redundant paths for decay heat removal. Only one RCP needs to be OPERABLE to declare the associated RCS loop OPERABLE. Reactor coolant natural circulation is not normally used but is sufficient for core cooling. However, natural circulation does not provide turbulent flow conditions. Therefore, boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be assured.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.1 Operable Components (Continued)

When Specification 2.1.1(3) is applicable, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be operable. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two shutdown cooling pumps to be OPERABLE.

One of the conditions for which Specification 2.1.1(3) is applicable is when the RCS temperature ($T_{cold}$) is less than 210°F, fuel is in the reactor and all reactor vessel closure bolts are fully tensioned. As soon as a reactor vessel head closure bolt is loosened, Specification 2.1.1(3) no longer applies, and Specification 2.8 is applicable. Specification 2.8 also requires two shutdown cooling loops to be operable if there is less than 23 feet of water above the top of the core.

The restrictions on availability of the containment spray pumps for shutdown cooling service ensure that the SI/CS pumps' suction header piping is not subjected to an unanalyzed condition in this mode. Analysis has determined that the minimum required RCS vent area is 47 $in^2$. This requirement may be met by removal of the pressurizer manway which has a cross-sectional area greater than 47 $in^2$.

When reactor coolant boron concentration is being changed, the process must be uniform throughout the reactor coolant system volume to prevent stratification of reactor coolant at lower boron concentration which could result in a reactivity insertion. Sufficient mixing of the reactor coolant is assured if one low pressure safety injection pump or one reactor coolant pump is in operation. The low pressure safety injection pump will circulate the reactor coolant system volume in less than 35 minutes when operated at rated capacity. The pressurizer volume is relatively inactive; therefore, it will tend to have a boron concentration higher than the rest of the reactor coolant system during a dilution operation. Administrative procedures will provide for use of pressurizer sprays to maintain a nominal spread between the boron concentration in the pressurizer and the reactor coolant system during the addition of boron.[1]

Both steam generators are required to be filled above the low steam generator water level trip set point whenever the temperature of the reactor coolant is greater than the design temperature of the shutdown cooling system to assure a redundant heat removal system for the reactor.

The basis for the LTOP system requirements are documented in the PTLR.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.1 Operable Components (Continued)

The design cyclic transients for the reactor system are given in USAR Section 4.2.2. In addition, the steam generators are designed for additional conditions listed in USAR Section 4.3.4. Flooded and pressurized conditions on the steam side assure minimum tube sheet temperature differential during leak testing. The minimum temperature for pressurizing the steam generator steam side is 70°F; in measuring this temperature, the instrument accuracy must be added to the 70°F; limit to determine the actual measured limit. The measured temperature limit will be 73°F based upon use of an instrument with a maximum inaccuracy of ± 2°F and an additional 1°F safety margin.

References

(1) USAR Section 4.3.7

## TABLE 2-9

## REACTOR COOLANT SYSTEM PRESSURE ISOLATION

| System | Valve No. | Maximum (a)(b) Allowable Leakage |
|---|---|---|
| High-Pressure Safety Injection | | |
| Loop 1A, cold leg | SI-216 | ≤ 5 gpm |
| | SI-201 | ≤ 5 gpm |
| Loop 1B, cold leg | SI-220 | ≤ 5 gpm |
| | SI-204 | ≤ 5 gpm |
| Loop 2A, cold leg | SI-208 | ≤ 5 gpm |
| | SI-195 | ≤ 5 gpm |
| Loop 2B, cold leg | SI-212 | ≤ 5 gpm |
| | SI-198 | ≤ 5 gpm |
| Low-Pressure Safety Injection | | |
| Loop 1A, cold leg | SI-200 | ≤ 5 gpm |
| Loop 1B, cold leg | SI-203 | ≤ 5 gpm |
| Loop 2A, cold leg | SI-194 | ≤ 5 gpm |
| Loop 2B, cold leg | SI-197 | ≤ 5 gpm |

Footnotes:

(a) 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

(b) Minimum test differential pressure shall not be less than 150 psid.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.1 Reactor Coolant System (Continued)

### 2.1.2 Heatup and Cooldown Rate

Applicability

Applies to the temperature change rates and pressure of the Reactor Coolant System (RCS).

Objective

To specify limiting conditions of the reactor coolant system heatup and cooldown rates.

Specification

The combination of RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR and as designated below:

a. Allowable combinations of pressure and temperature ($T_c$) for a specific heatup rate shall be below and to the right of the applicable limit lines as shown on the pressure and temperature (P-T) limit Figure(s) in the PTLR.

b. Allowable combinations of pressure and temperature ($T_c$) for a specific cooldown rate shall be below and to the right of the applicable limit lines as shown on the P-T limit Figure(s) in the PTLR.

c. The heatup rate of the pressurizer shall not exceed 100°F in any one hour period.

d. The cooldown rate of the pressurizer shall not exceed 200°F in any one hour period.

Required Actions

(1) When any of the above limits are exceeded, the following corrective actions shall be taken:

1. Immediately initiate action to restore the temperature or pressure to within the limit.

2. Perform an analysis to determine the effects of the out of limit condition on the fracture toughness properties of the reactor coolant system.

3. Determine that the reactor coolant system remains acceptable for continued operation or be in cold shutdown within 36 hours.

(2) Before the radiation exposure of the reactor vessel exceeds the exposure for which they apply, the P-T limit Figure(s) shown in the PTLR shall be updated in accordance with the following criteria and procedures:

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate (Continued)

(a) The P-T limit Figure(s) are valid for a fast neutron (E ≥ 1 MeV) fluence value and corresponding EFPY as stated in the PTLR.

(b) The limit line on the P-T limit Figure(s) shown in the PTLR shall be updated for a new integrated power period as follows: the total integrated reactor thermal power from startup to the end of the new period shall be converted to an equivalent integrated fast neutron exposure (E≥1 MeV).

(c) The limit lines in the P-T limit Figure(s) shown in the PTLR shall be moved parallel to the temperature axis (horizontal) in the direction of increasing temperature a distance equivalent to the transition temperature shift during the period since the curves were last constructed. The boltup temperature limit line shall remain at 64°F as it is set by the $RT_{NDT}$ of the reactor vessel flange and is not subjected to a fast neutron flux. The lowest service temperature shall remain at 164°F because components related to this temperature are also not subjected to a fast neutron flux.

(d) Technical Specification 2.3(3) shall be reviewed and revised as necessary each time the curves on the P-T limit Figure(s) as shown in the PTLR are revised.

Basis

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to reactor coolant system temperature and pressure changes.[1] These cyclic loads are introduced by normal unit load transients, reactor trips and startup and shutdown operation.

During unit startup and shutdown, the rates of temperature and pressure changes are limited. The design number of cycles for heatup and cooldown is based upon allowable heatup/cooldown rates and cyclic operation. Cycle dependent information such as the pressure-temperature limit curves, low temperature overpressure protection system limits, neutron fluence, and adjusted reference temperatures are contained in the PTLR, which was developed using the methodologies stated in Technical Specification 5.9.6(b) and in the PTLR[2].



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2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate (Continued)

References:

(1) USAR, Section 4.2.2

(2) Technical Data Book IX, Fort Calhoun Station Unit No. 1, RCS Pressure-Temperature Limits Report

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.1 Reactor Coolant System (Continued)

### 2.1.3 Reactor Coolant Radioactivity

Applicability

Applies to the radioactivity of the reactor coolant.

Objective

To ensure that the reactor coolant radioactivity is maintained at a level commensurate with the occupational and public safety.

Specification

(1) The radioactivity of the reactor coolant shall be limited to:

a. $\leq$ 1.0 µCi/gm DOSE EQUIVALENT I-131, and

b. $\leq$ 100/$\bar{E}$ µCi/gm

(2) With the radioactivity of the reactor coolant >1.0 µCi/gm DOSE EQUIVALENT I-131 for more than 100 hours during one continuous time interval or exceeding 60 µCi/gm, be in at least HOT SHUTDOWN with $T_{avg}$ <536°F within 6 hours.

(3) With the radioactivity of the reactor coolant > 100/$\bar{E}$ µCi/gm, be in at least HOT SHUTDOWN with $T_{avg}$ < 536°F within 6 hours.

(4) With the radioactivity of the reactor coolant >1.0 µCi/gm DOSE EQUIVALENT I-131, perform the sampling and analysis requirements of items 1.(a)(2)(ii) and 1.(b)(2)(i) of Table 3-4 until the radioactivity of the reactor coolant is restored to within its limits. Data pursuant to Specification 5.9.4.b for the Annual Report shall be compiled as follows:

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

b. Purification System flow history starting 48 hours prior to the first sample in which the limit was exceeded.

c. The time duration when the radioactivity of the reactor coolant exceeded 1.0 µCi/gm DOSE EQUIVALENT I-131.

d. Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should contain the date and time of sampling and the radioiodine concentrations.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.3 Reactor Coolant Radioactivity (Continued)

e. Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level.

Basis

The limitations on the radioactivity of the reactor coolant ensure that the resulting 2-hour doses at the site boundary will be well within the limits of 10 CFR Part 100 following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM and a concurrent loss of offsite power.

Permitting operation to continue for limited time periods with the reactor coolant's radioactivity levels >1.0 μCi/gm DOSE EQUIVALENT I-131, but ≦ 60 μCi/gm, accommodates possible iodine spiking phenomenon which may occur following changes in thermal power.

Reducing $T_{avg}$ to < 536°F prevents the release of radioactivity should a steam generator tube rupture, since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive radioactivity levels in the reactor coolant will be detected in sufficient time to take appropriate corrective actions(s).

References

USAR, Section 11.1.1.3

USAR, Section 14.14

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.1 Reactor Coolant System (Continued)

### 2.1.4 Reactor Coolant System Leakage Limits

Applicability

Applies to the leakage rates of the reactor coolant system whenever the reactor coolant temperature ($T_{cold}$) is greater than 210 °F.

Objective

To specify limiting conditions of the reactor coolant system leakage rates.

Specifications

To assure safe reactor operation, the following limiting conditions of the reactor coolant system leakage rates must be met:

(1) RCS operational LEAKAGE shall be limited to:

a. No Pressure Boundary LEAKAGE,
b. 1 gpm unidentified LEAKAGE,
c. 10 gpm identified LEAKAGE,
d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

(2) If RCS operational LEAKAGE limits of (1), above, are not met for reasons other than Pressure Boundary LEAKAGE or primary to secondary LEAKAGE, then reduce LEAKAGE to meet limits within 4 hours.

(3) If the Required Action and associated completion time of (2), above, is not met, OR Pressure Boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limits, then be in MODE 3, Hot Shutdown, within 6 hours AND be in MODE 4, Cold Shutdown, within 36 hours.

(4) To determine leakage to the containment, a containment atmosphere radiation monitor (gaseous or particulate) or dew point instrument, and a containment sump level instrument must be operable.

a. With no containment sump level instrument operable, verify that a containment atmosphere radiation monitor is operable, and restore the containment sump level instrument to operable status within 30 days.

b. With no containment atmosphere radiation monitor and no dewpoint instrument operable, restore either a radiation monitor or dewpoint instrument to operable status within 30 days.

c. With only the dewpoint instrument operable, or with no operable instruments, enter Specification 2.0.1 immediately.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.4 Reactor Coolant System Leakage Limits (Continued)

(5) To determine leakage to the secondary system one of the following must be operable:

a. Steam Generator Blow Down Radiation Sample Instrument
b. Condenser Off Gas Radiation Monitor
c. Periodic Secondary Samples Analyzed for Activity

Basis

Background

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

FCS Design Criteria (Ref. 2), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that can not be made 100% leaktight. Leakage from these systems should be detected, located and isolated from the containment atmosphere, if possible, to not interfere with RCS LEAKAGE detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.4 Reactor Coolant System Leakage Limits (Continued)

Applicable Safety Analysis

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The safety analysis assumes a 1 gpm primary to secondary leak as the initial condition. The Technical Specification requirement to limit primary to secondary leakage through any one steam generator to less than 150 gallons per day is significantly less than the initial condition for the safety analysis.

RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.4 Reactor Coolant System Leakage Limits (Continued)

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One SG

The 150 gallon per day operational limit on primary to secondary LEAKAGE through any one SG is based upon guidance in NEI 97-06, Steam Generator Program Guidelines. The Steam Generator Program operational LEAKAGE performance Criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

APPLICABILITY

The potential for RCPB LEAKAGE is greatest when the RCS is pressurized, that is, when the reactor coolant temperature ($T_{cold}$) is greater than 210°F.

In MODES 4 and 5, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

REQUIRED ACTIONS (2).
Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to meet limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

REQUIRED ACTIONS (3)
If any pressure boundary LEAKAGE exists or primary to secondary LEAKAGE is not within limits, or if unidentified or identified LEAKAGE cannot be reduced to meet limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3, Hot Shutdown, within 6 hours and to MODE 4, Cold Shutdown, within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 4, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.4 Reactor Coolant System Leakage Limits (Continued)

References

(1) USAR, Section 11.2.3

(2) USAR, Page G.16-1

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.1 Reactor Coolant System (Continued)

### 2.1.5 Maximum Reactor Coolant Oxygen and Halogens Concentrations

Applicability

Applied to the reactor coolant oxygen and halogens concentrations.

Objective

To specify the maximum oxygen and halogens concentrations of the reactor coolant system for safe reactor operation.

Specifications

Maximum limits of oxygen and halogens concentrations are as follows:

(1) If the concentration of oxygen in the reactor coolant exceeds 0.1 ppm during power operation, corrective action shall be initiated within 8 hours to return oxygen levels to ≤0. 1 ppm.

(2) If the concentration of chloride in the reactor coolant exceeds 0.15 ppm during power operation, corrective action shall be initiated within 8 hours to return chloride levels to ≤0.15 ppm.

(3) If the concentration of fluorides in the reactor coolant exceeds 0.10 ppm following modifications or repair to the reactor coolant system involving welding, corrective action shall be initiated within 8 hours to return fluoride levels to ≤0.10 ppm.

(4) If the oxygen concentration and the chloride or fluoride concentration of the reactor coolant simultaneously exceed the limits given in (1), (2) and (3), respectively, corrective action is to be taken immediately to return the system to within normal operation specifications.

(5) If the concentration limits of oxygen and chloride or fluoride given in (1), (2) and (3) above are not restored within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours thereafter. If the normal operational limits are not restored within an additional 24-hour period, the reactor shall be placed in a cold shutdown condition within 24 hours thereafter.

Basis

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant within the limits specified, the integrity of the reactor coolant system materials in contact with the coolant [1] are protected against potential stress corrosion attack. [2]

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.5 Maximum Reactor Coolant Oxygen and Halogens Concentration (Continued)

If these limits are exceeded, measures can be taken to correct the conditions (e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank) and further, because of the time-dependent nature of any adverse effects arising from the oxygen or halogens concentration in excess of the limits, it is unnecessary to shutdown immediately since the condition can be corrected.

The oxygen and halogen limits specified are at least an order of magnitude below concentrations which could result in damage to the materials found in the reactor coolant system even if maintained for an extended period of time.[3] Thus, the period of 8 hours to initiate corrective action to restore individual concentrations within the limits or 24 hours to restore both concentrations has been established. If the corrective action has not been effective at the end of the 24-hour period, then the reactor will be brought to the hot shutdown condition and the corrective action will continue. If, at the end of a further 24-hour period, the corrective action has not been effective, long-term corrective action could be required and the reactor will be brought to the cold shutdown condition.

References

(1) FSAR, Section 4.3.11

(2) FSAR, Section 4.4.3

(3) Corrosion and Wear Handbook, O.J. DePaul, Editor

## 2.0 LIMITING CONDITIONS FOR OPERATION
### 2.1 Reactor Coolant System (continued)
### 2.1.6 Pressurizer and Main Steam Safety Valves

Applicability
Applies to the status of the pressurizer and main steam safety valves.

Objective
To specify minimum requirements pertaining to the pressurizer and main steam safety valves.

Specifications
To provide adequate overpressure protection for the reactor coolant system and steam system, the following safety valve requirements shall be met:

(1) Two pressurizer safety valves shall be OPERABLE in MODES 1 and 2, with lift settings of 2485 psig +1%/-3% and 2530 psig +1%/-3% respectively.

(2) Whenever there is fuel in the reactor, and the reactor vessel head is installed, a minimum of one OPERABLE safety valve shall be installed on the pressurizer. However, when in at least the COLD SHUTDOWN condition, safety valve nozzles may be open to containment atmosphere during performance of safety valve tests or maintenance to satisfy this specification.

(3) At least four of the five Main Steam Safety Valves (MSSVs) associated with each steam generator shall be OPERABLE in MODES 1 and 2. Lift settings shall be at 985 psig +3/-2%, 1000 psig +3/-2%, 1010 psig +3/-2%, 1025 psig +3/-2%, and 1035 psig +3/-2%.[1]

a. With less than four of the five MSSVs associated with each steam generator OPERABLE, be in at least HOT STANDBY within 6 hours and HOT SHUTDOWN within an additional 6 hours.

(4) Two power-operated relief valves (PORVs) shall be OPERABLE during heatups and cooldowns when the RCS temperature is less than 515°F, and in MODES 4 and 5 whenever the head is on the reactor vessel and the RCS is not vented through a 0.94 square inch or larger vent, to prevent violation of the pressure-temperature limits designated by the P-T limit Figure(s) shown in the PTLR.

a. With one PORV inoperable during heatups and cooldowns when the RCS temperature is less than 515°F, restore the inoperable PORV to OPERABLE within 7 days or be in COLD SHUTDOWN within the next 36 hours and depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the following 36 hours.

b. With both PORVs inoperable during heatups and cooldowns when the RCS temperature is less than 515°F, be in COLD SHUTDOWN within the next 36 hours and depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the following 36 hours.

c. With one PORV inoperable in MODES 4 or 5, within one hour ensure the pressurizer steam space is greater than 50% volume and restore the inoperable PORV to OPERABLE within 7 days. If adequate steam space cannot be established within one hour, then restore the inoperable PORV to OPERABLE within 24 hours. If the PORV cannot be restored in the required time, depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the next 36 hours.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (continued)
2.1.6 Pressurizer and Main Steam Safety Valves (continued)

d. With both PORVs inoperable in Modes 4 or 5, depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the next 36 hours.

(5) Two power-operated relief valves (PORVs) and their associated block valves shall be operable in Modes 1, 2, and 3.

a. With one or both PORV(s) inoperable and capable of being manually cycled, within 1 hour either restore the PORV(s) to operable status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

b. With one PORV inoperable and not capable of being manually cycled, within 1 hour either restore the PORV to operable status or close its associated block valve and remove power from the block valve; restore the PORV to operable status within the following 72 hours or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

---NOTE---
Not applicable when second PORV intentionally made inoperable.
---

c. With both PORVs inoperable and not capable of being manually cycled, within 1 hour close both block valves, remove power from the block valves, and verify LCO 2.5(1), auxiliary feedwater, is met, and restore at least one PORV to OPERABLE status within 8 hours. Otherwise be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

d. With one block valve inoperable, within 1 hour restore the block valve to operable status or place the associated PORV in the closed position. Restore the block valve to operable status within the next 72 hours. Otherwise, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

---NOTE---
Not applicable when second block valve intentionally made inoperable.
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e. With both block valves inoperable, within 1 hour verify LCO 2.5(1), auxiliary feedwater, is met and restore at least one block valve to OPERABLE status within 8 hours. Otherwise, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

Basis

The purpose of the two spring-loaded Pressurizer Safety Valves (PSV's) is to provide Reactor Coolant System (RCS) overpressure protection and thereby ensure that the Safety Limit for RCS pressure (i.e., 2750 psia) is not exceeded for analyzed accidents. The maximum RCS pressure transient for an analyzed accident is associated with a Loss of Load event[2].

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (continued)
2.1.6 Pressurizer and Main Steam Safety Valves (continued)

The power-operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

Action statements (5)b. and c. include the removal of power from a closed block valve to preclude any inadvertent opening of the block valve at a time the PORV may not be closed due to maintenance. However, the applicability requirements of the LCO to operate with the block valve(s) closed with power maintained to the block valve(s) are only intended to permit operation of the plant for a limited period of time not to exceed the next refueling shutdown (Mode 5), so that maintenance can be performed on the PORV(s) to eliminate the seat leakage condition.

To determine the maximum steam flow, the only other pressure relieving system assumed operational is the main steam safety valves. Conservative values for all systems parameters, delay times and core moderator coefficients are assumed. Overpressure protection is provided to portions of the reactor coolant system which are at the highest pressure considering pump head, flow pressure drops and elevation heads.

If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve lift pressure would be less than half of the capacity of one safety valve. This specification, therefore, provides adequate defense against overpressurization when the reactor is subcritical.

Performance of certain calibration and maintenance procedures on safety valves requires removal from the pressurizer. Should a safety valve be removed, either operability of the other safety valve or maintenance of at least one nozzle open to atmosphere will assure that sufficient relief capacity is available. Use of plastic or other similar material to prevent the entry of foreign material into the open nozzle will not be construed to violate the "open to atmosphere" provision, since the presence of this material would not significantly restrict the discharge of reactor coolant.

The total relief capacity of the ten main steam safety valves is $6.606 \times 10^6$ lb/hr. If, following testing, the as found setpoints are outside +/-1% of nominal nameplate values, the valves are set to within the +/-1% tolerance. The main steam safety valves were analyzed for a total loss of main feedwater flow while operating at 1500 MWt[3] to ensure that the peak secondary pressure was less than 1100 psia, the ASME Section III upset pressure limit of 10% greater than the design pressure. At the power of 1500 MWt, sufficient relief valve capacity is available to prevent overpressurization of the steam system on loss-of-load conditions.[4] These analyses are based on a minimum of four-of-five operable main steam safety valves on each main steam header.

The power-operated relief valve low setpoint will be adjusted to provide sufficient margin, when used in conjunction with Technical Specification Sections 2.1.1 and 2.3, to prevent the design basis pressure transients from causing an overpressurization incident. Limitation of this requirement to scheduled cooldown ensures that, should emergency conditions dictate rapid cooldown of the reactor coolant system, inoperability of the low temperature overpressure protection system would not prove to be an inhibiting factor. The effective full flow area of an open PORV is 0.94 $in^2$.

Removal of the reactor vessel head provides sufficient expansion volume to limit any of the design basis pressure transients. Thus, no additional relief capacity is required.

References
(1) Article 9 of the 1968 ASME Boiler and Pressure Vessel Code, Section III
(2) USAR, Section 14.9
(3) USAR, Section 14.10
(4) USAR, Sections 4.3.4, 4.3.9.5

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.7 Pressurizer Operability

Applicability

Applies to the status of the pressurizer and pressurizer heaters.

Objective

To specify minimum requirements pertaining to the pressurizer water volume and availability of heaters for accident conditions.

Specifications

(1) The pressurizer shall be operable with at least 150 KW of pressurizer heaters, and pressurizer inventory shall be maintained in a range of level 40.5% to 69.2%.

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in HOT SHUTDOWN within the following 12 hours. With the pressurizer otherwise inoperable, be in HOT SHUTDOWN within the following 12 hours. This is applicable for Modes 1 and 2.

b. With the pressurizer level outside the above range, either restore the level within the specified limits within 2 hours or be in HOT SHUTDOWN within the following 12 hours. This is applicable for Modes 1 and 2.

Basis

The requirement that 150 KW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT SHUTDOWN. Either diesel will fulfill the minimum requirements of this specification. The level should be maintained above the lower limit to prevent heater cutoff and the upper limit should not be exceeded to prevent going solid or reducing the effectiveness of the pressurizer sprays by immersion during an RCS swell transient.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.1 Reactor Coolant System (Continued)

### 2.1.8 Reactor Coolant System Vents

Applicability

Applies to the status of the reactor coolant gas vent system. This specification is applicable while in modes 1, 2, or 3.

Objective

To ensure capability of venting non-condensible gases from the reactor coolant system, the following gas vent system requirements must be met:

(1) At least one reactor coolant system vent path consisting of at least two valves in series powered from emergency buses shall be OPERABLE and closed at each of the following locations:

a. Reactor vessel head.

b. Pressurizer steam space.

(2) With one of the above reactor coolant system vent paths inoperable, startup and/or power operation may continue provided power is removed from the valve actuators of all the inoperable valves; restore the inoperable vent path to OPERABLE status within 30 days or be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

(3) With both of the above reactor coolant system vent paths inoperable, maintain the inoperable vent path closed with power removed from the valve actuators of all the inoperable valves in the inoperable vent paths and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

Basis

The purpose of this specification is to ensure a method and system is available to remove steam and/or non-condensible gases from the reactor coolant system, which may inhibit core cooling during natural circulation. The Power Operated Relief Valves are not to be considered a vent path for the purpose of this specification.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.2 Chemical and Volume Control System

### 2.2.1 Boric Acid Flow Paths - Shutdown

Applicability

Applies to the operational status of the boric acid flow paths in MODES 4 and 5 when fuel is in the reactor.

Objective

To assure operability of equipment required to add negative reactivity.

Specification

As a minimum, one of the following boric acid flow paths from an OPERABLE borated water source shall be OPERABLE:

a. A flow path from boric acid storage tank CH-11A via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System.

b. A flow path from boric acid storage tank CH-11B via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System.

c. A flow path from both boric acid storage tanks (CH-11A and CH-11B) via either a boric acid transfer pump or gravity feed connection and a charging pump to the Reactor Coolant System.

d. A flow path from the SIRW tank via either a charging pump or a high pressure safety injection pump to the Reactor Coolant System.

Required Actions

(1) With none of the above boric acid flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.2 Chemical and Volume Control System (Continued)

2.2.2 Boric Acid Flow Paths - Operating

Applicability

Applies to the operational status of the boric acid flow paths whenever the reactor coolant temperature ($T_{cold}$) is greater than or equal to 210°F.

Objective

To assure operability of equipment required to add negative reactivity.

Specification

At least two of the following boric acid flow paths from OPERABLE borated water sources shall be OPERABLE:

a. A flow path from boric acid storage tank CH-11A, via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System.

b. A flow path from boric acid storage tank CH-11B, via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System.

c. A flow path from both boric acid storage tanks (CH-11A and CH-11B) via either a boric acid transfer pump or gravity feed connection and a charging pump to the Reactor Coolant System.

d. A flow path from the SIRW tank via a charging pump to the Reactor Coolant System.

Required Actions

(1) With only one of the above required boric acid flow paths to the Reactor Coolant System OPERABLE, restore to at least two OPERABLE boric acid flow paths to the Reactor Coolant System within 72 hours.

(2) With the required actions of (1) not met, or with none of the required boric acid flow paths to the Reactor Coolant System OPERABLE, be in at least HOT SHUTDOWN within 6 hours, in at least subcritical and <300°F within the next 6 hours, and in at least COLD SHUTDOWN within the following 30 hours.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.2 Chemical and Volume Control System (Continued)

### 2.2.3 Charging Pumps - Shutdown

Applicability

Applies to the operational status of charging pumps in MODES 4 and 5 when fuel is in the reactor.

Objective

To assure operability of equipment required to add negative reactivity.

Specification

At least one charging pump or one high pressure safety injection pump in the boric acid flow path required to be OPERABLE pursuant to Specification 2.2.1 shall be OPERABLE.

Required Actions

(1) With no charging pump or high pressure safety injection pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.2 Chemical and Volume Control System (Continued)

### 2.2.4 Charging Pumps - Operating

Applicability

Applies to the operational status of charging pumps whenever the reactor coolant temperature ($T_{cold}$) is greater than or equal to 210°F.

Objective

To assure operability of equipment required to add negative reactivity.

Specification

At least two charging pumps shall be OPERABLE.

Required Actions

(1) With only one charging pump OPERABLE, restore to at least two OPERABLE charging pumps within 72 hours.

(2) With the required actions of (1) not met, or with no charging pumps OPERABLE, be in at least HOT SHUTDOWN within 6 hours, in at least subcritical and <300°F within the next 6 hours, and in at least COLD SHUTDOWN within the following 30 hours.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.2 Chemical and Volume Control System (Continued)

2.2.5 Boric Acid Transfer Pumps - Shutdown

Applicability

Applies to the operational status of the boric acid transfer pumps in MODES 4 and 5 when fuel is in the reactor.

Objective

To assure operability of equipment required to add negative reactivity.

Specification

At least one boric acid transfer pump shall be OPERABLE if the flow path through the boric acid transfer pump in Specification 2.2.1 is OPERABLE.

Required Actions

(1) With no boric acid transfer pump OPERABLE as required to complete the flow path of Specification 2.2.1, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.2 Chemical and Volume Control System (Continued)

### 2.2.6 Boric Acid Transfer Pumps - Operating

Applicability

Applies to the operational status of the boric acid transfer pumps whenever the reactor coolant temperature ($T_{cold}$) is greater than or equal to 210°F.

Objective

To assure operability of equipment required to add negative reactivity.

Specification

At least the boric acid transfer pump(s) in the boric acid flow path(s) required to be OPERABLE pursuant to Specification 2.2.2 shall be OPERABLE if the flow path(s) through the boric acid transfer pump(s) in Specification 2.2.2 is OPERABLE.

Required Actions

(1) With one boric acid transfer pump required to be OPERABLE to complete one of the two boric acid flow paths of Specification 2.2.2 inoperable, restore the boric acid transfer pump to OPERABLE status within 72 hours.

(2) With the required actions of (1) not met, or with two boric acid transfer pumps required to be OPERABLE to complete both of the boric acid flow paths of Specification 2.2.2 inoperable, be in at least HOT SHUTDOWN within 6 hours, in at least subcritical and <300°F within the next 6 hours, and in at least COLD SHUTDOWN within the following 30 hours.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.2 Chemical and Volume Control System (Continued)

### 2.2.7 Borated Water Source - Shutdown

Applicability

Applies to the operational status of borated water sources in MODES 4 and 5 when fuel is in the reactor.

Objective

To assure operability of equipment required to add negative reactivity.

Specification

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

a. Boric acid storage tank CH-11A with the contents of the tank in accordance with the COLR for a SIRW tank boron concentration at REFUELING BORON CONCENTRATION, and with the ambient temperature of the boric acid solution greater than or equal to the solubility temperature of Figure 2-12.

b. Boric acid storage tank CH-11B with the contents of the tank in accordance with the COLR for a SIRW tank boron concentration at REFUELING BORON CONCENTRATION, and with the ambient temperature of the boric acid solution greater than or equal to the solubility temperature of Figure 2-12.

c. Both boric acid storage tanks CH-11A and CH-11B with the combined contents of both tanks in accordance with the COLR for a SIRW tank boron concentration at REFUELING BORON CONCENTRATION, and with the ambient temperature of the boric acid solution greater than or equal to the solubility temperature of Figure 2-12.

d. The SIRW tank with:

   1. A minimum useable borated water volume of 10,000 gallons,

   2. A minimum boron concentration of REFUELING BORON CONCENTRATION, and

   3. A minimum solution temperature of 50°F.

Required Actions

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

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.2 Chemical and Volume Control System (Continued)

2.2.8 Borated Water Sources - Operating

Applicability

Applies to the operational status of borated water sources whenever the reactor coolant temperature ($T_{cold}$) is greater than or equal to 210°F.

Objective

To assure operability of equipment required to add negative reactivity.

Specification

Both of the following borated water sources shall be OPERABLE:

a. At least one boric acid storage tank (CH-11A or CH-11B) with the contents of the tank in accordance with the COLR, or both boric acid storage tanks (CH-11A and CH-11B) with the combined contents of both tanks in accordance with the COLR, and with the ambient temperature of the boric acid solution greater than or equal to the solubility temperature of Figure 2-12.

b. The SIRW tank with:

   1. A minimum useable borated water volume of 25,000 gallons,

   2. A minimum boron concentration of REFUELING BORON CONCENTRATION, and

   3. A minimum solution temperature of 50°F.

Required Actions

(1) With the above required boric acid storage tank(s) inoperable, restore the tank(s) to OPERABLE status within 72 hours.

(2) With the SIRW tank inoperable, be in at least HOT SHUTDOWN within 6 hours, in at least subcritical and <300°F within the next 6 hours, and in at least COLD SHUTDOWN within the following 30 hours.

(3) With the required actions of (1) not met, or with no OPERABLE borated water source, be in at least HOT SHUTDOWN within 6 hours, in at least subcritical and <300°F within the next 6 hours, and in at least COLD SHUTDOWN within the following 30 hours.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.2 Chemical and Volume Control System (Continued)

Basis

The chemical and volume control system provides control of the reactor coolant system boron inventory.[1] This is normally accomplished by using any one of the three charging pumps in series with one of the two boric acid pumps, or taking suction from one or both of the gravity feed valves. An alternate method of boration is to use the charging pumps directly from the SIRW tank. A third method is to depressurize and use the safety injection pumps.

The chemical and volume control system (CVCS) operates in conjunction with the safety injection system to inject concentrated boric acid into the reactor coolant system on receipt of a pressurizer pressure low signal (PPLS) and/or a containment pressure high signal (CPHS). Because this system is not necessary to mitigate the consequences of accidents, as documented in USAR Chapter 14, this system, including charging pumps, is not classified as Engineered Safeguards equipment.[2]

Operability requirements for the CVCS borated water sources, boric acid flow paths, boric acid transfer pumps, and charging pumps ensure that an adequate source of boric acid is available to provide required shutdown margin during a plant cooldown and applicable plant modes. Operator actions have been identified to ensure the ability of the CVCS system to perform its function in the event of an equipment failure.

Borated water sources

The sources of borated water available are: (1) boric acid storage tank CH-11A, boric acid storage tank CH-11B, or the combination of boric acid storage tanks CH-11A and CH-11B; and (2) the SIRW tank. These sources have sufficient boron to maintain required shutdown margin during a plant cooldown.

Whenever the reactor coolant temperature ($T_{cold}$) is greater than or equal to 210°F, two borated water sources must be operable in order to ensure sufficient capacity. For a borated water source to be considered operable, tank volume, boron concentration, and temperature of the contained boric acid solution must be within their respective requirements.

In Modes 4 and 5 when fuel is in the reactor, only one of these sources must be operable. One source is acceptable during these modes on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting core alterations and positive reactivity changes in the event the single source becomes inoperable. If no sources are operable, restore at least one source to operable status.

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.2 Chemical and Volume Control System (Continued)

Basis (Continued)

Boric acid flow paths

Consistent with the requirement to maintain two borated water sources operable when the RCS temperature is greater than or equal to 210°F, a minimum of two boric acid flow paths from operable borated water sources must also be operable. For a flow path to be considered operable, boric acid must be capable of being transported from the operable borated water source to the reactor coolant system. Consistent with the requirements for borated water sources, 72 hours is allowed to restore the system to two operable flow paths.

The flow paths available depend on which sources of borated water are operable. A flow path from a boric acid storage tank may be through the gravity feed connection or a boric acid transfer pump. When one of the operable sources is the combined contents of both boric acid storage tanks, then the flow path from this source requires that a flow path from each tank to the RCS be operable. This flow path can be established by using various combinations of gravity feed connections and/or boric acid transfer pumps. Both tanks could also be aligned to a single boric acid transfer pump since the specification requires, when using this flow path, that a flow path from the SIRW tank be operable. Therefore, two flow paths are available by maintaining the additional flow path from the SIRW tank.

In Modes 4 and 5 when fuel is in the reactor, only one flow path must be operable. One flow path is acceptable during these modes on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting core alterations and positive reactivity changes in the event the single flow path becomes inoperable. If no flow path is operable, restore at least one flow path to operable status.

Boric Acid Transfer Pumps

Boric acid transfer pumps need only be operable if required to complete an operable boric acid flow path.

Whenever the reactor coolant temperature ($T_{cold}$) is greater than or equal to 210°F, two flow paths from operable borated water sources are required to be operable. The flow path from an operable boric acid storage tank may be through the gravity feed connection or a boric acid transfer pump. If the gravity feed connection from the operable boric acid storage tank is inoperable, then a boric acid transfer pump must be operable in order to complete an operable flow path. The specification allows 72 hours to restore one boric acid transfer pump if it is required to complete a flow path. In this situation, the one inoperable pump renders one required flow path inoperable. The specification requires a plant shutdown if two boric acid transfer pumps are inoperable that are required to complete two flow paths. In this situation, the inoperable pumps render both required flow paths inoperable.

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.2 Chemical and Volume Control System (Continued)

Basis (Continued)

In Modes 4 and 5 when fuel is in the reactor, only one flow path must be operable. This is consistent with the number of operable borated water sources required during these modes. If the gravity feed connection from the operable boric acid storage tank is inoperable, then a boric acid transfer pump must be operable in order to complete an operable flow path.

Boric acid transfer pumps are each of sufficient capacity to feed all three charging pumps at their maximum capacity.

Charging Pumps

Whenever the reactor coolant temperature ($T_{cold}$) is greater than or equal to 210°F, two charging pumps must be operable in order to ensure it is possible to inject concentrated boric acid into the reactor coolant system. With only one pump operable, 72 hours is allowed to restore the system to two operable charging pumps. This is consistent with the allowed outage time for the borated water sources and flow paths required during these modes.

In Modes 4 and 5 when fuel is in the reactor, only one charging pump or high pressure safety injection pump must be operable. This is consistent with the number of operable borated water sources and flow paths required during these modes. A pump is required in order to complete an operable flow path to the reactor coolant system. There are additional restrictions on the use of high pressure safety injection pumps contained in Technical Specification 2.3 to ensure that the reactor vessel is not overpressurized.

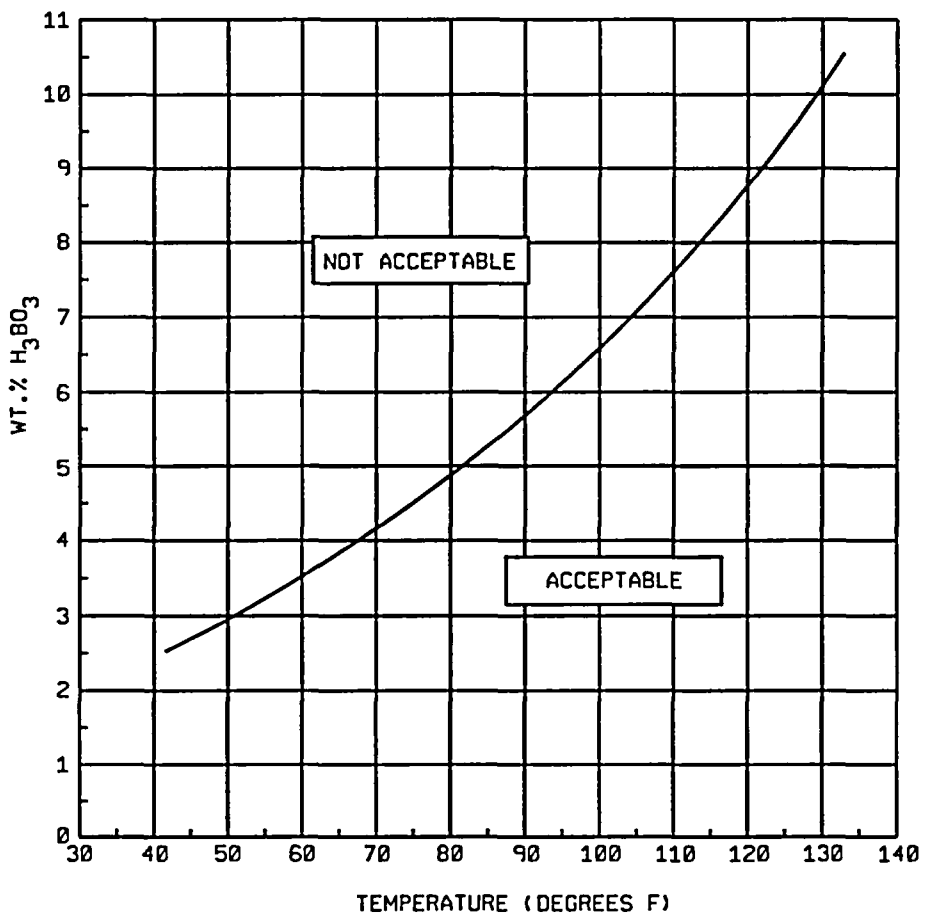
Figure 2-12 contains a 10°F bias to account for temperature measurement uncertainty. An administrative procedure to monitor the temperature of the BASTs and boric acid system piping in the Auxiliary Building ensures that the temperature requirements of Figure 2-12 are met. Should the system temperature be unacceptable for operation at the current boric acid concentration, steps will be taken to reduce the boric acid concentration or raise the temperature of the system such that the concentration is within the acceptable range of Figure 2-12.

The limits on component operability and the time periods for inoperability were selected on the basis of the redundancy indicated above and NUREG-0212 Revision 2. The allowed outage times for the various components are consistent such that a support system has the same allowed outage time as the supported system.

References

(1) USAR Section 9.2
(2) USAR Section 6.1.2.1

Figure 2-12




| | | |
|---|---|---|
| BORIC ACID SOLUBILITY IN WATER | OMAHA PUBLIC POWER DISTRICT<br>FORT CALHOUN STATION-UNIT 1 | FIGURE<br>2-12 |

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.3 Emergency Core Cooling System

Applicability

Applies to the operating status of the emergency core cooling system.

Objective

To assure operability of equipment required to remove decay heat from the core.

Specifications

(1) Minimum Requirements

The reactor shall not be made critical unless all of the following conditions are met:

a. The SIRW tank contains not less than 283,000 gallons of water with a boron concentration of at least the refueling boron concentration at a temperature not less than 50°F.

b. One means of temperature indication (local) of the SIRW tank is operable.

c. All four safety injection tanks are operable and pressurized to at least 240 psig and a maximum of 275 psig with tank level of at least 116.2 inches (67%) and a maximum level of 128.1 inches (74%) with refueling boron concentration.

d. One level and one pressure instrument is operable on each safety injection tank.

e. One low-pressure safety injection train is operable on each associated 4,160 V engineered safety feature bus.

f. One high-pressure safety injection pump is operable on each associated 4,160 V engineered safety feature bus.

g. Both shutdown heat exchangers are operable.

h. Piping and valves shall be operable to provide two flow paths from the SIRW tank to the reactor coolant system.

i. All valves, piping and interlocks associated with the above components and required to function during accident conditions are operable. HCV-2914, 2934, 2974, and 2954 shall have power removed from the motor operators by locking open the circuit breakers in the power supply lines to the valve motor operators. FCV-326 shall be locked open.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System (Continued)

(1) j. One high-pressure safety injection pump is operable on each safety injection refueling water tank-containment sump header.

## 2.0 LIMITING CONDITIONS FOR OPERATION
### 2.3 Emergency Core Cooling System (Continued)

(2) Modification of Minimum Requirements

During power operation, the Minimum Requirements may be modified to allow one of the following conditions to be true at any one time. If the system is not restored to meet the minimum requirements within the time period specified below, the reactor shall be placed in a hot shutdown condition within 12 hours. If the minimum requirements are not met within an additional 48 hours the reactor shall be placed in a cold shutdown condition within 24 hours.

a. One low-pressure safety injection train may be inoperable provided the train is restored to operable status within seven (7) days.

b. One high-pressure safety injection pump may be inoperable provided the pump is restored to operable status within 24 hours.

c. One shutdown heat exchanger may be inoperable for a period of no more than 24 hours.

d. Any valves, interlocks or piping directly associated with one of the above components and required to function during accident conditions shall be deemed to be part of that component and shall meet the same requirements as listed for that component.

e. Any valve, interlock or piping associated with the safety injection and shutdown cooling system which is not covered under d. above but which is required to function during accident conditions may be inoperable for a period of no more than 24 hours.

f. One safety injection tank may be inoperable for reasons other than g. or h. below for a period of no more than 24 hours.

g. Level and/or pressure instrumentation on one safety injection tank may be inoperable for a period of 72 hours.

h. One safety injection tank may be inoperable due to boron concentration not within limits for a period of no more than 72 hours.



Correction letter of 6-3-2004

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System (Continued)

(3) Protection Against Low Temperature Overpressurization

The following limiting conditions shall be applied during scheduled heatups and cooldowns. Disabling of the HPSI pumps need not be required if the RCS is vented through at least a 0.94 square inch or larger vent.

Whenever the reactor coolant system cold leg temperature is below 350°F, at least one (1) HPSI pump shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 320°F, at least two (2) HPSI pumps shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 270°F, all three (3) HPSI pumps shall be disabled.

In the event that no charging pumps are operable when the reactor coolant system cold leg temperature is below 270°F, a single HPSI pump may be made operable and utilized for boric acid injection to the core, with flow rate restricted to no greater than 120 gpm.

(4) Containment Sump Buffering Agent Specification and Volume Requirement

During operating Modes 1 and 2, the containment sump buffering agent baskets shall contain a volume of hydrated sodium tetraborate (NaTB) that is within the area of acceptable operation shown in Figure 2-3.

a. With the above buffering agent requirements not within limits, the buffering agent shall be restored within 72 hours.

b. With Specification 2.3(4)a required action and completion time not met, the plant shall be in hot shutdown within the next 6 hours and cold shutdown within the following 36 hours.

Basis

The normal procedure for starting the reactor is to first heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical. The energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore all engineered safety features and auxiliary cooling systems are required to be fully operable.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.3 Emergency Core Cooling System (Continued)

The USAR Loss of Coolant Accident analysis assumes a minimum SIRW tank inventory of 250,000 gallons has been pumped from the SIRW tank when recirculation begins. Technical Specification 2.3(1) requires that the SIRW tank contains a minimum of 283,000 gallons of usable water. This additional volume over that assumed in the USAR analysis provides sufficient margin to account for the instrument uncertainty. The SIRW tank contains water containing a boron concentration of at least the refueling boron concentration. This is sufficient boron concentration to provide a shutdown margin of 5%, including allowances for uncertainties, with all control rods withdrawn and a new core at a temperature of 68°F.[2]

The limits for the safety injection tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses. The minimum 116.2 inch level corresponds to a volume of 825 $ft^3$ and the maximum 128.1 inch level corresponds to a volume of 895.5 $ft^3$. Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked. Since the system is used for shutdown cooling, the valving will be changed and must be properly aligned prior to start-up of the reactor.

The operable status of the various systems and components is to be demonstrated by periodic tests. A large fraction of these tests will be performed while the reactor is operating in the power range.

If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. For a single component to be inoperable does not negate the ability of the system to perform its function. If it develops that the inoperable component is not repaired within the specified allowable time period, or a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. After a limited time in hot shutdown, if the malfunction(s) is not corrected, the reactor will be placed in the cold shutdown condition utilizing normal shutdown and cooldown procedures. In the cold shutdown condition, release of fission products or damage of the fuel elements is not considered possible.

The plant operating procedures will require immediate action to effect repairs of an inoperable component and therefore in most cases repairs will be completed in less than the specified allowable repair times. The limiting times to repair are intended to assure that operability of the component will be restored promptly and yet allow sufficient time to effect repairs using safe and proper procedures.

The time allowed to repair a safety injection tank is based on the deterministic and probabilistic analyses of Reference (8). The time allowed to repair a LPSI train is based on the deterministic and probabilistic analysis of Reference (9). These analyses concluded that the overall risk impact of the completion times are either risk-beneficial or risk neutral.

The requirement for core cooling in case of postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition reduces the consequences of a loss-of-coolant accident and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 48 hours of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case, the reactor is to be put into the cold shutdown condition.

## 2.0 LIMITING CONDITIONS FOR OPERATION
### 2.3 Emergency Core Cooling System (Continued)

With respect to the core cooling function, there is functional redundancy over most of the range of break sizes.[3][4]

The LOCA analysis confirms adequate core cooling for the break spectrum up to and including the 32 inch double-ended break assuming the safety injection capability which most adversely affects accident consequences and are defined as follows. The entire contents of all four safety injection tanks are assumed to be available for emergency core cooling, but the contents of one of the tanks is assumed to be lost through the reactor coolant system. In addition, of the three high-pressure safety injection pumps and the two low-pressure safety injection pumps, for both large break analysis and small break analysis it is assumed that one high pressure pump and one low pressure pump operate[5]; and also that 25% of their combined discharge rate is lost from the reactor coolant system out of the break. The transient hot spot fuel clad temperatures for the break sizes considered are shown in USAR Section 14.

The restriction on HPSI pump operability at low temperatures, in combination with the PORV setpoints ensure that the reactor vessel pressure-temperature limits would not be exceeded in the case of an inadvertent actuation of the operable HPSI and charging pumps.

Removal of the reactor vessel head, one pressurizer safety valve, or one PORV provides sufficient expansion volume to limit any of the design basis pressure transients. Thus, no additional relief capacity is required.

Technical Specification 2.2(1) specifies that, when fuel is in the reactor, at least one flow path shall be provided for boric acid injection to the core. Should boric acid injection become necessary, and no charging pumps are operable, operation of a single HPSI pump would provide the required flow path. The HPSI pump flow rate must be restricted to that of three charging pumps in order to minimize the consequences of a mass addition transient while at low temperatures.

Hydrated Sodium Tetraborate (NaTB) is required to adjust the pH of the recirculation water to ≥ 7.0 after a loss of coolant accident (LOCA). This pH value is necessary to prevent significant amounts of iodine, released from fuel failures and dissolved in the recirculation water, from converting to a volatile form and evolving into the containment atmosphere. Higher levels of airborne iodine in containment may increase the releases of radionuclides and the consequences of the accident. A pH of ≥ 7.0 is also necessary to prevent stress corrosion cracking (SCC) of austenitic stainless steel components in containment. SCC increases the probability of failure of components.

NaTB is used because of the high humidity in the containment building during normal operation. Since the NaTB is hydrated, it is less likely to absorb large amounts of water from the humid atmosphere and will undergo less physical and chemical change.

Radiation levels in containment following a LOCA may cause the generation of hydrochloric and nitric acids from radiolysis of cable insulation and sump water. NaTB will neutralize these acids.

The required amount of NaTB is represented in a volume quantity converted from the Reference 7 mass quantity using the manufactured density. Verification of this amount during surveillance testing utilizes the measured volume.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.3 Emergency Core Cooling System (Continued)

References

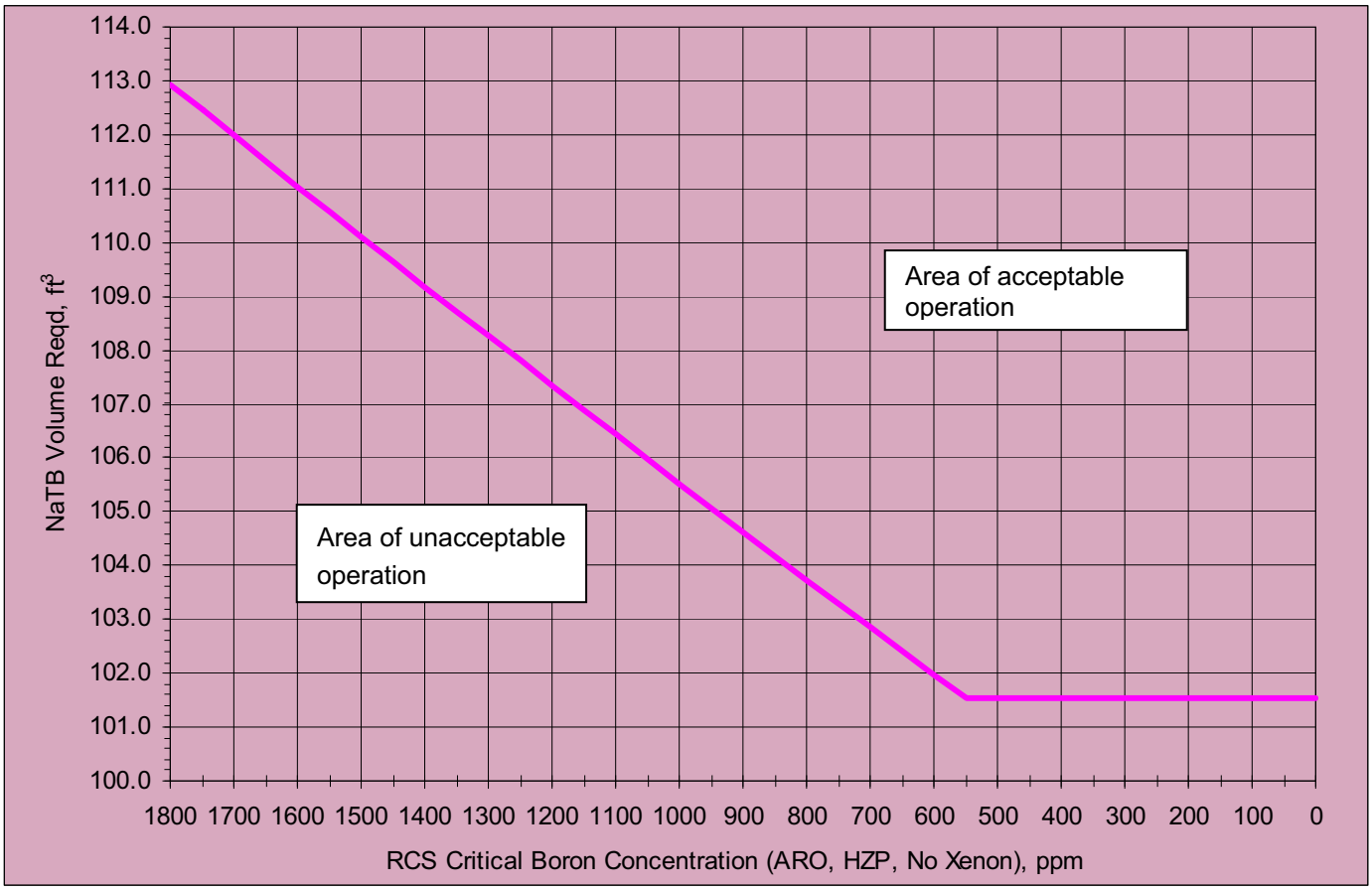
(1) USAR, Section 14.15.1
(2) USAR, Section 6.2.3.1
(3) USAR, Section 14.15.3
(4) USAR, Appendix K
(5) Omaha Public Power District's Submittal, December 1, 1976
(6) Deleted
(7) USAR, Section 4.4.3
(8) CE NPSD-994, "CEOG Joint Applications Report for Safety Injection Tank AOT/SIT Extension," May 1995.
(9) CE NPSD-995, "CEOG Joint Applications Report for Low Pressure Safety Injection System AOT Extension," May 1995.



Correction letter of 6-3-2024

## 2.0 LIMITING CONDITIONS FOR OPERATION
### 2.3 Emergency Core Cooling System (Continued)

Figure 2-3
NaTB Volume Required for RCS Critical Boron Concentration (ARO, HZP, No Xenon)



## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.4 Containment Cooling

Applicability

Applies to the operating status of the containment cooling systems.

Objective

To assure operability of equipment required to remove heat from the containment during normal operating and emergency situations.

Specifications

(1) Minimum Requirements

a. The reactor shall not be made critical, except for low-temperature physics tests, unless all the following are met:

i. The following equipment normally associated with diesel-generator DG-1 (4.16-kV bus 1A3 and associated non-automatically transferring 480-Volt bus sections) is operable, except as noted:[1]

| | |
|---|---|
| Raw water pump | AC-10A |
| Raw water pump | AC-10C |
| Component cooling water pump | AC-3A |
| Component cooling water pump | AC-3C |
| Containment spray pump | SI-3A |
| Containment air cooling and filtering unit | VA-3A |
| Containment air cooling unit | VA-7C |

ii. The following equipment normally associated with diesel-generator DG-2 (4.16-kV 1A4 and associated non-automatically transferable 480 Volt bus sections) is operable, except as noted.[1]

| | |
|---|---|
| Raw water pump | AC-10B |
| Raw water pump | AC-10D |
| Component cooling water pump | AC-3B |
| Containment spray pump | SI-3B |
| Containment air cooling and filtering unit | VA-3B |
| Containment air cooling unit | VA-7D |

iii. Four component cooling heat exchangers shall be operable.

iv. All valves, piping and interlocks associated with the above components and required to function during accident conditions are operable.

(1) Reactor may be made critical with one inoperable raw water pump. LCO action statements shall apply.

## 2.0 **LIMITING CONDITIONS FOR OPERATION**

2.4 Containment Cooling (Continued)

b. During power operation one of the components listed in (1)a.i. or ii. may be inoperable. If the inoperable component is not restored to operability within seven days, the reactor shall be placed in hot shutdown condition within 12 hours. If the inoperable component is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

c. For cases involving Raw Water pump inoperability, if the river water temperature is below 60 degrees Fahrenheit, one Raw Water pump may be inoperable indefinitely without applying any LCO action statement. When the river water temperature is greater than 60 degrees Fahrenheit, an inoperable Raw Water pump shall be restored to operability within 7 days or the reactor shall be placed in a hot shutdown condition within 12 hours. If the inoperable Raw Water pump is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

(2) Modification of Minimum Requirements

a. During power operation, the minimum requirements may be modified to allow a total of two of the components listed in (1)a.i. and ii. to be inoperable at any one time. (This does not include: 1) One Raw Water pump which may be inoperable as described above if the river water temperature is below 60 degrees Fahrenheit or, 2) SI-3A and SI-3B being simultaneously inoperable; or 3) VA-3A and VA-3B, or VA-7C and VA-7D, being simultaneously inoperable. Only two raw water pumps may be out of service during power operations. Either containment spray pump, SI-3A or SI-3B, must be operable during power operations. One train of the containment air cooling and filtering systems, (VA-3A and VA-7C), or (VA-3B and VA-7D), must be operable during power operations.) If the operability of one of the two components is not restored within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. LCO 2.4(1)b. shall be applied if one of the inoperable components is restored within 24 hours. If the operability of both components is not restored within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

b. During power operation one component cooling heat exchanger may be inoperable. If the operability of the heat exchanger is not restored within 14 days, the reactor shall be placed in a hot shutdown condition within 12 hours. If two component cooling heat exchangers are inoperable, the reactor shall be placed in hot shutdown condition within 12 hours. If the inoperable heat exchanger(s) is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

c. Any valves, interlocks and piping directly associated with one of the above components and required to function during accident conditions shall be deemed to be part of that component and shall meet the same requirements as for that component.

d. Any valve, interlock or piping associated with the containment cooling system which is not included in the above paragraph and which is required

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.4 Containment Cooling (Continued)

to function during accident conditions may be inoperable for a period of no more than 24 hours. If operability is not restored within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours.

Basis

A full capacity diesel-generator is connected to each of the two engineered safeguards 4.16-kV buses. Three engineered safeguards 480-Volt double-ended load centers are provided; of the six transformers, three are connected to each of the two 4.16-kV buses. Two load centers are operated as two-bus-section units; the third is provided with a center bus manually transferable to either associated end section. The center bus section supplies HPSI Pump SI-2C, CS Pump SI-3C and Charging Pump CH-1C any of which can thus be supplied from either 4.16-kV bus if required (note that CS pump SI-3C is connected to the center bus, however, this pump is available for manual operation only).

The containment spray pumps will start for a main steam line break (MSLB) but will not start during a loss-of-coolant accident (LOCA). The containment spray pumps initially take coolant from the safety injection and refueling water tank (SIRWT) before the pump suction is switched to the containment recirculation line. For an MSLB, with all safety injection pumps running, the SIRWT supply of water is not exhausted for at least 24 minutes[(2)]. One shutdown cooling heat exchanger is sufficient to satisfy the shutdown cooling requirements during the long-term containment cooling period.[(3)] In addition, in the unlikely event of the component cooling water supply being lost, raw water can be utilized for direct cooling of certain engineered safeguard components.[(4)]

The containment air cooling and filtering system is independent from the containment spray system for the containment cooling function.[(5)] For the limiting LOCA scenario, one of the two containment air cooling trains, comprised of one containment air cooling and filtering unit and one containment air cooling unit, would limit the containment pressure to below the design value without taking credit for the cooling capacity of the containment spray system.[(6)] [(7)] For the limiting MSLB scenario, a heat removal contribution is credited from the air coolers and the containment spray system in the mitigation of containment peak pressure.[(7)]

The cooling equipment provided to limit the containment pressure following a LOCA is divided between the independent power supply systems. The raw water and component cooling water pumps are similarly distributed on the 4.16-kV and 480 Volt buses. In the event of a LOCA, loss of normal power sources and failure of one diesel generator to operate, a minimum of one air cooler train would be connected to the available diesel generator. This would provide adequate containment cooling equipment to limit the containment pressure below the design value. The limiting MSLB event which results in the maximum containment pressure is not affected by the loss of one diesel generator.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.4 Containment Cooling (Continued)

The component cooling system pumps and heat exchanger, the spray pumps and the shutdown heat exchangers are located in the auxiliary building.[9] [10] The raw water pumps are located in the intake structure.[11]

Analyses show that after a high heat load accident such as a large break LOCA or Main Steam Line Break inside containment, three in service component cooling heat exchangers will maintain CCW return temperature in an analyzed range. This assumes all of the containment air cooling units are operating which would create the maximum heat load on the CCW system. In order to ensure that three heat exchangers would be in service after a DBA in conjunction with an assumed single failure, four are required to be operable.

If the river temperature is below 70°F, the single failure of a component cooling heat exchanger's RW valve to open would not raise CCW return temperature to an unanalyzed level. A single failure of a RW valve to auto-open will raise CCW return temperature due to CCW passing through a heat exchanger which has no flow. If the river temperature is greater than or equal to 70°F, a single failure of a heat exchanger's RW valve to open could raise CCW return temperature to an unanalyzed level. Therefore, when the river temperature is greater than or equal to 70°F and RW is not in service to a heat exchanger, the heat exchanger is considered inoperable. Having RW in service eliminates the potential failure of a RW valve to open as a credible single active failure.

When a component cooling heat exchanger is inoperable for maintenance and the river temperature is greater than or equal to 70°F, at least one RW and CCW isolation valve is secured in the closed position. This prevents the valve from opening upon receipt of a Safety Injection Actuation Signal, and therefore prevents CCW passing through a heat exchanger that has no RW flow. If the opening stroke time of a RW or CCW isolation valve on a heat exchanger is such that the valve is considered inoperable, the heat exchanger associated with the inoperable valve is still operable as long as the affected RW or CCW valve is maintained in the open position.

## 2.0 **LIMITING CONDITIONS FOR OPERATION**

2.4 Containment Cooling (Continued)

References

(1) Deleted
(2) USAR, Section 6.2.3.1
(3) USAR, Section 6.2.3.4
(4) USAR, Section 9.8.2
(5) USAR, Section 6.4.5
(6) USAR, Section 6.4.1.2
(7) USAR, Section 14.16.5
(8) Deleted
(9) USAR, Section 9.7
(10) USAR, Section 6.3
(11) USAR, Section 9.8

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.5 Steam and Feedwater Systems

Applicability

When steam generators are relied upon for reactor coolant system heat removal.

Objective

To define certain conditions for the steam and feedwater system necessary to assure adequate decay heat removal.

Specifications

(1) Two AFW trains shall be OPERABLE when $T_{cold}$ is above 300°F.

A. With one steam supply to the turbine driven AFW pump inoperable, or

----------Note----------:
Only applicable if MODE 2 has not been entered following refueling
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the turbine driven AFW pump inoperable following refueling,

restore the affected equipment to OPERABLE status within 7 days and within 8 days from discovery of failure to meet the LCO.

B. With one AFW train inoperable for reasons other than condition A, restore the AFW train to OPERABLE status within 24 hours.

C. If the required action and associated completion times of condition A or B are not met, then the unit shall be placed in MODE 2 in 6 hours, in MODE 3 in the next 6 hours, and less than 300°F without reliance on the steam generators for decay heat removal within the next 18 hours.

D. With both AFW trains inoperable, then initiate actions to restore one AFW train to OPERABLE status immediately. Technical Specification (TS) 2.0.1 and all TS actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status.

(2) The motor driven train is required to be OPERABLE when $T_{cold}$ is below 300°F and the steam generators are relied upon for heat removal. With the motor driven AFW train inoperable, then initiate actions to restore one AFW train to OPERABLE status immediately. Technical Specifications (TS) 2.0.1 and all TS actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.5 Steam and Feedwater Systems

(3) A minimum of 55,000 gallons of water in the emergency feedwater storage tank (EFWST) and a backup water supply to the emergency feedwater storage tank shall be available. With the EFWST inoperable, verify operability of the backup water supply within four hours and once per 12 hours thereafter, and restore the EFWST to OPERABLE status within 24 hours. If these action requirements cannot be satisfied, then the unit shall be placed in at least MODE 3 within 6 hours, and less than 300°F without reliance on the steam generators for decay heat removal within the next 18 hours.

(4) The main steam stop valves are OPERABLE when $T_{cold}$ is above 300°F and capable of closing in four seconds or less under no-flow conditions.

Basis

A reactor shutdown from power requires a removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as long as feedwater to the steam generator is available. Normally, the capability to supply feedwater to the steam generators is provided by operation of the turbine cycle feedwater system. In the unlikely event of complete loss of electrical power to the station, decay heat removal is by steam discharge to the atmosphere via the main steam safety and atmospheric dump valves. Either auxiliary feed pump can supply sufficient feedwater for removal of decay heat from the plant. Technical Specification 2.1.1 establishes when the steam generators are required for heat removal. Each train includes the pump, piping, instruments, and controls to ensure the availability of an OPERABLE flow path capable of taking suction from the EFWST and delivering water to the steam generators.

If one of the two steam supplies to the turbine driven AFW pump is inoperable, or if the turbine driven pump is inoperable while the reactor coolant temperature $T_{cold}$ is above 300°F immediately following refueling, action must be taken to restore the inoperable equipment to OPERABLE status within 7 days. The 7 day completion time is reasonable based on the following reasons:

a. For the inoperability of a steam supply to the turbine driven AFW pump, the 7 day completion time is reasonable since there is a redundant steam supply line for the turbine driven pump.

b. For the inoperability of the turbine driven AFW pump while the reactor coolant temperature $T_{cold}$ is above 300°F immediately subsequent to a refueling, the 7 day completion time is reasonable due to the minimal decay heat levels in this situation.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.5 Steam and Feedwater Systems

Basis (Continued)

c. For both the inoperability of a steam supply line to the turbine driven pump and an inoperable turbine driven AFW pump while the reactor coolant temperature $T_{cold}$ is above 300°F immediately subsequent to a refueling, the 7 day completion time is reasonable due to the availability of the redundant OPERABLE motor driven AFW pump, and due to the low probability of an event requiring the use of the turbine driven AFW pump.

The eight day completion time for 2.5(1)A provides a limit in the maximum time allowed for any combination to be inoperable during any continuous failure to meet the LCO. With one of the required AFW trains inoperable, actions must be taken to restore OPERABLE status within 24 hours. With no AFW trains OPERABLE the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting cooldown with nonsafety grade equipment. In such a condition the unit should not be perturbed by any action, including a power change, that might result in a trip.

The minimum amount of water in the emergency feedwater storage tank is the amount needed for 8 hours of such operation. The tank can be resupplied with water from the raw water system.[1]

A closure time of 4 seconds for the main steam stop valves is considered adequate time and was selected as being consistent with expected response time for instrumentation as detailed in the steam line break analysis.[2][3]

References

(1) USAR, Section 9.4.6

(2) USAR, Section 10.3

(3) USAR, Section 14.12

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.6 Containment System

Applicability

Applies to the reactor containment system.

Objective

To assure the integrity of the reactor containment system.

Specifications

(1) Containment Integrity

a. Containment integrity shall not be violated unless the reactor is in a cold or refueling shutdown condition. Without containment integrity, restore containment integrity within one hour or be in at least hot shutdown within the next 6 hours, in at least subcritical and <300°F within the next 6 hours and in cold shutdown within the following 30 hours. Normally locked or sealed-closed valves (except for PCV-742A/B/C/D) may be opened intermittently under administrative control without constituting a violation of containment integrity.

b. The personnel air lock shall be operable unless the reactor is in a cold or refueling shutdown condition. Both doors shall be closed except when the air lock is being used for normal transit, then at least one air lock door shall be closed. The entire air lock assembly leakage rate shall be in accordance with Specification 5.19.

(i). With one personnel air lock door inoperable.

a. Maintain at least the operable air lock door closed and either restore the inoperable air lock door to operable status within 24 hours or lock the operable air lock door closed. Entry and exit is permissible to perform repairs of the affected air lock components without constituting a violation of containment integrity.
b. Operation may then continue until performance of the next required entire air lock assembly leakage test provided that the operable air lock door is verified to be locked closed at least once per 31 days.
c. Otherwise, be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.
d. Entry into another operational mode or specified condition is allowed if the provisions stated in 2.6(1)b.(i)a. above are met.

(ii). With the personnel air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to operable status within 24 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.6 Containment System (Continued)

c. DELETED

d. The containment isolation valves VA-280 and VA-289 shall be locked closed. Opening of these valves intermittently under administrative control is not allowed.

e. The containment purge isolation valves will be locked closed unless the reactor is in a cold or refueling shutdown condition.

(2) Internal Pressure

The internal pressure shall not exceed 3 psig (except for containment leak rate tests).

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.6 Containment System (Continued)

Basis

The reactor coolant system conditions of cold shutdown assure that no steam will be formed and, hence, there would be no pressure buildup in the containment if the reactor coolant system ruptures. The shutdown margins are selected based on the type of activities that are being carried out.

Regarding internal pressure limitations, the containment design pressure of 60 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 3 psig.[1] The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment. Operation of the purge isolation valves is prevented during normal operations due to the size of the valves (42 inches) and a concern about their ability to close against the differential pressure that could result from a LOCA or MSLB. Specification 2.6(1)a applies when both doors of the PAL are declared inoperable, or the entire air lock assembly leakage exceeds the requirements of Specification 5.19. Specification 2.6(1)b(ii) applies when mechanisms other than a door, such as the inner door equalizing valve, are declared inoperable.

The Hydrogen Purge System is required to be operable in order to control the quantity of combustible gases in containment in a post-LOCA condition.[2] The containment integrity will be protected by ensuring the penetration valves VA-280 and VA-289 are "locked closed" while HCV-881 and HCV-882 are normally closed during power operation. The applicable surveillance testing requirements of Table 3-5 will ensure that the system is capable of performing its design function. The blowers (VA-80A and VA-80B), associated valves, and piping are single failure proof, have been designed as a Seismic Class I System, and are redundant to the VA-82 filter header. VA-80A or VA-80B is capable of providing sufficient hydrogen removal capabilities as required by the USAR to prevent the hydrogen concentration inside of containment from exceeding the 4% flammability limit.[3] Electrical Equipment qualification was not required as the radiation doses in the area of the Hydrogen Purge System equipment were below the minimum requirements.[4]

VA-80A or VA-80B with the associated valves and piping may be inoperable for 30 days. The redundancy of the blowers allows one blower with associated valves and piping to be removed from operation while the other train has the capability to provide 100% hydrogen control.

References

(1) USAR, Section 14.16; Figure 14.16-2
(2) Regulatory Guide 1.7 (1971)
(3) USAR, Section 14.17
(4) Engineering Study 86-10, Calculation 53

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.7 Electrical Systems

Applicability

Applies to the availability of electrical power for the operation of plant components.

Objective

To define those conditions of electrical power availability necessary to provide for safe reactor operation and the continuing availability of engineered safety features.

Specifications

(1) Minimum Requirements

The reactor shall not be heated up or maintained at temperatures above 300°F unless the following electrical systems are operable:

a. Unit auxiliary power transformers T1A-1 or T1A-2 (4,160 V).

b. House service transformers T1A-3 and T1A-4 (4,160 V).

c. 4,160 V engineered safety feature buses 1A3 and 1A4.

d. 4,160 V/480 V Transformers T1B-3A, T1B-3B, T1B-3C, T1B-4A, T1B-4B, T1B-4C.

e. 480 V distribution buses 1B3A, 1B3A-4A, 1B4A, 1B3B, 1B3B-4B, 1B4B, 1B3C, 1B3C-4C, 1B4C.

f. MCC No. 3A1, 3A2, 3B1, 3C1, 3C2, 4A1, 4A2, 4B1, 4C1 and 4C2.

g. 125 V d-c buses No. 1 and 2 (Panels EE-8F and EE-8G).

h. 125 V d-c distribution panels AI-41A and AI-41B.

i. 120V a-c instrument buses A, B, C, and D (Panels AI-40-A, B, C and D).

j. Two (2) 125 V d-c bus No. 1 required inverters: (A and C), or (A and associated swing inverter), or (C and associated swing inverter) AND;

   Two (2) 125 V d-c bus No. 2 required inverters: (B and D), or (B and associated swing inverter), or (D and associated swing inverter).

k. Station batteries No. 1 and 2 (EE-8A and EE-8B) including one battery charger on each 125 V d-c bus No. 1 and 2 (EE-8F and EE-8G).

l. Two emergency diesel generators (DG-1 and DG-2).

m. One diesel fuel oil storage system containing a volume of diesel fuel which is ≥ a 7-day supply.

n. Lubricating oil inventory for each DG is ≥ a 7-day supply.

o. Each required starting air receiver bank pressure is ≥ 190 psig.

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.7 Electrical Systems (Continued)

(2) Modification of Minimum Requirements

The minimum requirements may be modified to the extent that one of the following conditions will be allowed after the reactor coolant has been heated above 300°F. However, the reactor shall not be made critical unless all minimum requirements are met. If any of the provisions of these exceptions are violated, the reactor shall be placed in a hot shutdown condition within the following 12 hours. If the violation is not corrected within an additional 12 hours, the reactor shall be placed in a cold shutdown condition within an additional 24 hours.

a. Both unit auxiliary power transformers T1A-1 and T1A-2 (4.16 kV) may be inoperable for up to 72 hours.

b. Either house service transformer T1A-3 or T1A-4 (4.16kV) may be inoperable for up to 7 days. The NRC Operations Center shall be notified by telephone within 4 hours after transformer inoperability. Additionally, within 24 hours from discovery of either house service transformer inoperability, declare the required feature(s) associated with the inoperable house service transformer inoperable, when its redundant required feature (including the steam driven auxiliary feedwater pump FW-10) is inoperable.

c. Both house service transformers T1A-3 and T1A-4 (4.16kV) may be inoperable for up to 72 hours. The loss of the 161kV incoming line renders both transformers inoperable. The NRC Operations Center shall be notified by telephone within 4 hours after inoperability of both transformers.

## 2.0 LIMITING CONDITIONS FOR OPERATION
### 2.7 Electrical Systems (Continued)

d. Either one of the 4.16kV engineered safeguards buses, 1A3 or 1A4 may be inoperable for up to 8 hours provided the operability of the diesel generator associated with the operable bus is demonstrated immediately.

e. One of each group of 4160 V/480 V Transformers (T1B-3A or 4A), (T1B-3B or 4B), and (T1B-3C or 4C) may be inoperable for up to 8 hours.

f. One of the 480 V distribution buses connected to bus 1A3 or connected to bus 1A4 may be inoperable for up to 8 hours.

g. Either Group of MCC No.'s (3A1, 3A2, 3B1, 3C1, 3C2,) or (4A1, 4A2, 4B1, 4C1, 4C2) may be inoperable for up to 8 hours.

h. One of the four 120V a c instrument buses (A, B, C or D) may be inoperable for 8 hours provided the reactor protective and engineered safeguards systems instrument channels supplied by the remaining three buses are all operable.

i. Two battery chargers may be inoperable for up to 8 hours provided battery charger No. 1 (EE-8C) or No. 2 (EE-8D) is operable.

j. Either one of the emergency diesel generators (DG-1 or DG-2) may be inoperable for up to seven consecutive days. In addition, the cumulative total time of inoperability for both DGs during any calendar month shall not exceed seven days. If one diesel generator is inoperable, within 8 hours (regardless of when the inoperable diesel generator is restored to operability) EITHER:

   (1) Start the other diesel generator to verify operability, OR

   (2) Ensure the absence of common cause for the diesel generator inoperability for the other diesel generator.

   Additionally, within 4 hours from discovery of either diesel generator inoperability, declare the required feature(s) associated with the inoperable diesel generator inoperable, when its redundant required feature (including the steam driven auxiliary feedwater pump FW-10) is inoperable.

k. Not used.

l. Island buses 1B3A-4A, 1B3B-4B, and 1B3C-4C may be inoperable for up to 8 hours.

m. Either one of the 125V d-c buses No. 1 or 2 (Panels EE-8F or EE-8G) may be inoperable for up to 8 hours.

n. Either one of the 125V d-c distribution panels AI-41A or AI-41B may be inoperable for up to 8 hours.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.7 Electrical Systems (Continued)

o. One of the required inverters may be inoperable for up to 24 hours provided the reactor protective and engineered safeguards systems instrument channels supplied by the remaining three required inverters are all operable and the 120V a-c instrument bus associated with the inoperable inverter is powered from its bypass source.

(3) Modification of Minimum Requirements for Diesel Fuel Oil, Diesel Lube Oil, and Starting Air

The minimum requirements may be modified to the extent that any of the following conditions will be allowed after the reactor coolant has been heated above 300°F. However, the reactor shall not be made critical unless all minimum requirements are met.

a. If the inventory of diesel fuel oil in the diesel fuel storage system is less than a 7-day supply, but greater than a 6 day supply, then restore the required inventory within 48 hours.

b. If one or more diesel generators has lube oil inventory < a 7-day supply and > a 6 day supply, then restore the lube oil inventory to within limits within 48 hours.

c. If the total particulates of fuel oil stored in FO-1 or FO-10 is not within limits, then restore fuel oil total particulates to within limits within 7 days.

d. If the properties of new fuel oil stored in FO-1 or FO-10 is not within limits, then restore stored fuel oil properties to within limits within 30 days.

e. If one or more diesel generators has the required starting air receiver bank with pressure < 190 psig and > 150 psig, then restore starting air receiver bank pressure to > 190 psig within 48 hours.

f. If the Required Action and associated Completion Time of a, b, c, d or e are not met or one or more diesel generators have diesel fuel oil, lube oil, or a required starting air subsystem not within limits for reasons other than a, b, c, d, or e, then declare the associated DG inoperable immediately.

## 2.0 **LIMITING CONDITIONS FOR OPERATION**

2.7 Electrical Systems (Continued)

Basis

The electrical system equipment is arranged so that no single failure can inactivate enough engineered safeguards to jeopardize the plant safety. The 480 V safeguards are arranged on nine bus sections. The 4.16 kV safeguards are supplied from two buses.

The normal source of auxiliary power with the plant at power for the safeguards buses is from the house service power transformers being fed from the 161 Kv incoming line with on-site emergency power from either one of two diesel generators and off-site standby power via the unit auxiliary transformers.[1] The loss of the 161kV incoming line renders the house service transformers (T1A-3 and T1A-4) inoperable in that the transformers cannot supply power to the 4.16kV safeguards buses 1A3 and 1A4. Inoperability of the house service transformer(s) or loss of the 161kV incoming line is not reportable pursuant to 10 CFR 50.72 criteria; however, the NRC will be promptly notified of these events via the NRC Operations Center.

The two emergency diesel generators on site do not require offsite power for start up or operation.

Upon loss of normal and standby power sources, the 4.16 Kv buses 1A3 and 1A4 are energized from the diesel generators. Bus load shedding, transfer to the diesel generator and pickup of critical loads are carried out automatically.[2]

When the turbine generator is out of service for an extended period, the generator can be isolated by opening motor operated disconnect switch DS-T1 in the bus between the generator and the main transformer, allowing the main transformer and the unit auxiliary power transformers (T1A-1 and T1A-2) to be returned to service.[3] The auxiliary power transformers are not considered inoperable during these normal plant startup/shutdown realignments.

Minimum requirements are implemented prior to raising the RCS temperature above 300°F to assure availability of engineered safety features.

The time allowed to repair an inoperable inverter is based upon engineering judgement, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. In the event of inverter failure, the load on the inverter is automatically transferred to its safety related bypass source. The associated 120 V a-c instrument bus is considered OPERABLE when it is being powered from its bypass source and during the short time it takes to manually or automatically transfer between sources.

Swing inverter EE-8T is associated with inverters A and C. Swing inverter EE-8U is associated with inverters B and D. A swing inverter can take the place of either of its associated inverters. An inverter is required if it is one from the combination of inverters used to satisfy Specification 2.7(1)j. If at least four safety related inverters are OPERABLE and supplying 120 V a-c instrument buses A, B, C, and D in a combination that satisfies Specification 2.7(1)j, the non-required inverter(s) may be inoperable or out-of-service indefinitely.

## 2.0 LIMITING CONDITIONS FOR OPERATION
### 2.7 Electrical Systems (Continued)

Basis (continued)

Equipment served by 4.16 kV and 480 V auxiliary buses and MCC's is arranged so that loss of an entire 4.16 kV bus does not compromise safety of the plant during DBA conditions. For example, if 4.16 kV bus 1A3 is lost, two raw water pumps, one low pressure safety injection pump, two high pressure safety injection pumps, one auxiliary feedwater pump, two component cooling water pumps, one containment spray pump and two containment air fans are lost. This leaves two raw water pumps, one low pressure safety injection pump, one high pressure safety injection pump, one component cooling water pump, one containment spray pumps and two containment air fans which is more than sufficient to control containment pressure below the design value during the DBA.

Each diesel generator has sufficient capacity to start and run at design load required by engineered safety features equipment. The safety features operated from one diesel generator can adequately cool the core for any loss of coolant accident and also maintain the containment pressure within the design value. TS 2.7(2)j limits a single period of inoperability for one diesel generator (DG) to 7 consecutive days and states that the cumulative total time of inoperability for both DGs during any calendar month shall not exceed 7 days. This is to ensure that a DG is not taken out in excess of 7 consecutive days in 2 months (e.g., 7 days at the end of 1 calendar month followed by up to 7 days at the beginning of the next month). The engine base tank capacity of 550 gallons on each diesel provides 3 hours running time (worst case loading) before transfer of fuel oil from the 18,000 gallon capacity emergency diesel generator fuel oil storage tank FO-1 is mandatory. Two fuel oil transfer pumps per diesel, with each being powered from the associated diesel, are available for transferring fuel oil from FO-1 to the day tanks. The minimum diesel fuel oil inventories available to the diesel generators from the emergency diesel generator fuel oil storage tanks FO-1, FO-10, base tanks, and day tanks are maintained to assure there is a 7-day supply of fuel (Ref. 6).

Engineering calculations have determined that 26,739 gallons of diesel fuel will be required to operate both diesel generators for eight hours and a single diesel generator for the remainder of the required 7-day operating period assuming a loss of offsite power coincident with a design basis accident. If diesel fuel storage tanks FO-1 and FO-10 are maintained at a minimum indicated inventory of 16,000 gallons and 13,000 gallons, respectively, the available onsite storage of fuel will provide 7 days of diesel operating time with approximately 850 gallons of margin. In the event that a 7-day supply of diesel fuel is not available, a 6-day supply will be available if the total combined (indicated) inventory in FO-1 and FO-10 is at least 25,000 gallons.

A minimum amount of diesel fuel oil is reserved in the auxiliary boiler fuel oil storage tank FO-10 for transfer to the emergency diesel generator fuel oil storage tank in the event of an emergency to extend the fuel supply for diesel generator operation to 7 days. Methods of transfer of the fuel oil from this tank to FO-1 have been established and procedures have been developed so that the transfer can be made in a timely manner without adversely impacting diesel generator operation.

Therefore, a minimum diesel fuel oil inventory available to the diesel generators from the total on-site diesel fuel oil storage capacity is maintained to assure the operation of one diesel generator at the required post accident loads for 7 days. The fuel inventory is allowed below the 7 day supply, but above a 6 day supply, for a period of 48 hours.

## 2.0 LIMITING CONDITIONS FOR OPERATION
2.7 Electrical Systems (Continued)

Basis (continued)

This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to initiating a plant shutdown as required by Specification 2.7(3). This period is acceptable based on the remaining capacity (more than 6 days), the fact that procedures are in place to obtain replenishment, and the low probability of an event during this brief period.

Additional supplies of diesel fuel oil are available in the Omaha area and from nearby terminals. Ample facilities exist to assure deliveries to the site within 24 hours.

One battery charger on each battery shall be operating so that the batteries will always be at full charge; this ensures that adequate d-c power will be available for all emergency uses. Each battery has one battery charger permanently connected with a third charger capable of being connected to either battery bus. The chargers are each rated for 400 amperes at 130 volts. Following a DBA the batteries and the chargers will handle all required loads. Each of the reactor protective channels instrumentation channels is supplied by one of the safety-related a-c instrument buses. The removal of one of the safety-related a-c instrument buses is permitted as the 2-of-4 logic may be manually changed to a 2-of-3 logic without compromising safety.

The engineered safeguards instrument channels use safety-related a-c instrument buses (one redundant bus for each channel) and d-c buses (one redundant bus for each logic circuit). The removal of one of the safety-related a-c instrument buses is permitted as the two of four logic automatically becomes a two of three logic.

The requirement in Specification 2.7(2)j, to declare required redundant feature(s) inoperable, is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related components.

Redundant required feature failures consist of inoperable features with a component redundant to the component that has an inoperable DG. The steam driven auxiliary feedwater pump FW-10 is required to be considered a redundant required feature to motor driven auxiliary feedwater pump FW-6, and, is therefore, required to be determined OPERABLE, since there are only two safety-related AFW pumps. With FW-10 and DG-1 INOPERABLE, coincident with a single failure of house service transformer T1A-3, would result in a complete loss of a safety function. With FW-6 and DG-2 INOPERABLE, coincident with a single failure of house service transformer T1A-4, would not result in a complete loss of a safety function since FW-10 would still be OPERABLE.

Redundant required features for an inoperable DG do not include components powered from 125 VDC or 120 VAC sources, since a loss of function would not occur with an inoperable DG coincident with a single failure of its associated house service transformer. Radiation Monitors RM-051, RM-052, and RM-062 are required to be considered redundant features since the monitors are contained on a skid assembly which is powered from 480 VAC.

## 2.0 LIMITING CONDITIONS FOR OPERATION
### 2.7 Electrical Systems (Continued)

Basis (continued)

The time allowed for declaring a redundant required feature(s) inoperable is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This also allows for an exception to the normal beginning for the limiting condition for operation time. In this required action, the time only begins upon discovery that both:

a. An inoperable DG exists and
b. A required feature associated with the other 4160V bus is inoperable.

If at any time during the existence of this Condition (one DG inoperable) a required feature subsequently becomes inoperable, this time begins to be tracked. Discovering one required DG inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE DG, results in starting the time for the required action. Four hours from the discovery of these events existing concurrently, is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

In this modified Condition (one DG inoperable and loss of required component on the opposite DG), the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite electrical distribution system. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, the function has not been lost. The 4-hour allowed time takes into account the operability of the redundant counterpart to the inoperable required feature. Additionally, the 4-hour allowed time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

When a system has installed spare components, the spare component is not required to be OPERABLE to meet required feature operability. As an example, there are three installed 100% capacity high pressure safety injection (HPSI) pumps, one (SI-2B) associated with 4160V bus 1A4, and two (SI-2A and SI-2C) associated with 4160V bus 1A3. Specification 2.3(1) *Minimum Requirements* are that there be one HPSI pump on each associated 4160V bus and each safety injection refueling water tank-containment sump header. This requires that SI-2A OR SI-2C be OPERABLE, not both.

The DG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated DG under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. The onsite storage of 500 gallons in addition to the engine oil sump is sufficient to ensure 7 days of continuous operation. This supply is sufficient supply to allow the operator to replenish lube oil from outside sources. With lube oil inventory < 500 gallons, sufficient lubricating oil to support 7 days of continuous DG operation at full load conditions may not be available. However, the Condition is restricted to lube oil volume reductions that maintain a minimum of 450 gallons, which is at least a 6 day supply. This restriction allows sufficient time to obtain the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.7 Electrical Systems (Continued)

Basis (continued)

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. FCS has a Diesel Fuel Oil Testing Program which includes proper fuel oil quality. This program includes purchasing, receipt testing of new fuel oil, and periodic analyses of the stored fuel oil. FCS is not committed to the fuel analysis portion of Regulatory Guide 1.137 (Ref. 4) or ANSI N195-1976 (Ref. 5); however, these standards were utilized as guidance in the development of the Diesel Fuel Oil testing program. The fuel oil properties governed by these Surveillance Requirements are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level. TS 2.7(3)c is entered as a result of a failure to meet the acceptance criterion of Table 3-5, Item 9c. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between Surveillance intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling, and re-analysis of the DG fuel oil.

With the new fuel oil properties defined in the Bases for Table 3-5, Item 9c not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

Each DG has two starting air subsystems (primary and secondary), each with adequate capacity for five successive start attempts of the DG without recharging the air start receivers. Either subsystem can fulfill the function of starting the DG, however the requirements of TS 3.7(1)a.i must be met for the required starting air subsystem. With starting air receiver bank pressure < 190 psig, sufficient capacity for five successive DG start attempts does not exist. However, as long as the receiver bank pressure is > 150 psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the air receiver bank pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

References

(1) USAR, Section 8.3.1.2
(2) USAR, Section 8.4.1
(3) USAR, Section 8.2.2
(4) Regulatory Guide 1.137
(5) ANSI N195-1976
(6) USAR, Section 8.4.1.2

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.8 Refueling
### 2.8.1 Refueling Shutdown

#### 2.8.1(1) Boron Concentration

Applicability

Applies to Reactor Coolant System boron concentration when fuel is in the reactor and one or more reactor vessel head closure bolts are less than fully tensioned.

Objective

The limit on the boron concentration of the Reactor Coolant System ensures that the reactor remains subcritical when the plant is in REFUELING SHUTDOWN (MODE 5).

Specification

Boron concentration of the Reactor Coolant System shall be greater than or equal to REFUELING BORON CONCENTRATION.

Required Actions

(1) With the boron concentration not within limit, suspend CORE ALTERATIONS immediately, and

(2) Suspend positive reactivity additions immediately, and

(3) Initiate actions to restore boron concentration to within limits immediately.

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.8 Refueling
### 2.8.1 Refueling Shutdown

#### 2.8.1(2) Nuclear Instrumentation

Applicability

Applies to the source range neutron monitors in MODE 5 with fuel in the reactor and with one or more reactor vessel head closure bolts less than fully tensioned.

Objective

To monitor the core reactivity condition and to alert the operator to unexpected changes in core reactivity when the plant is in REFUELING SHUTDOWN (MODE 5).

Specification

Two source range neutron monitors shall be OPERABLE.

Required Actions

(1) With only one source range neutron monitor OPERABLE, suspend CORE ALTERATIONS and positive reactivity additions immediately.

(2) With no source range neutron monitors OPERABLE, suspend CORE ALTERATIONS and positive reactivity additions immediately, and initiate actions to restore one source range neutron monitor to OPERABLE status immediately, and verify RCS boron concentration is greater than or equal to REFUELING BORON CONCENTRATION within 4 hours and once per 12 hours thereafter.

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.8 Refueling
### 2.8.1 Refueling Shutdown

#### 2.8.1(3) Shutdown Cooling System - High Water Level

Applicability
Applies to shutdown cooling requirements in MODE 5 with fuel in the reactor and with one or more reactor vessel head closure bolts less than fully tensioned, and the refueling cavity water level ≥23 ft. above the top of the core.

Objective
To minimize the possibility of a loss of shutdown cooling accident occurring inside containment that could affect public health and safety.

Specification
One OPERABLE Shutdown Cooling loop shall be IN OPERATION except as noted below:

1. The required Shutdown Cooling loop may be removed from operation for ≤ one hour per 8 hour period, provided no operations are permitted that would cause dilution of the RCS boron concentration.

2. The required Shutdown Cooling loop may be inoperable for up to eight hours provided (1) no operations are permitted that would cause dilution of the RCS boron concentration, (2) no CORE ALTERATIONS or REFUELING OPERATIONS are taking place, (3) all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere are closed within 4 hours, and (4) at least one loop is available under administrative controls.

Required Actions

(1) With no Shutdown Cooling loop IN OPERATION (except as allowed by notes 1 or 2 above),

   a. Suspend operations involving a reduction in reactor coolant boron concentration immediately, and

   b. Suspend loading of irradiated fuel assemblies into the reactor core immediately, and

   c. Initiate actions to restore system to operation immediately, and

   d. Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere within 4 hours.

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.8 Refueling
### 2.8.1 Refueling Shutdown

#### 2.8.1(4) Shutdown Cooling System - Low Water Level

Applicability

Applies to shutdown cooling requirements in MODE 5 with fuel in the reactor and with one or more reactor vessel head closure bolts less than fully tensioned, and the refueling cavity water level < 23 ft. above the top of the core.

Objective

To minimize the possibility of a loss of shutdown cooling accident occurring inside of containment that could affect public health and safety.

Specification

Two Shutdown Cooling loops shall be OPERABLE, and one Shutdown Cooling loop shall be IN OPERATION.

Required Actions

(1) With one Shutdown Cooling loop inoperable either,

a. Restore the inoperable Shutdown Cooling loop to OPERABLE status immediately, or

b. Initiate actions to establish at least 23 ft. of water above the top of the core immediately.

(2) With both Shutdown Cooling loops inoperable or one Shutdown Cooling loop not IN OPERATION,

a. Suspend operations involving a reduction in RCS boron concentration immediately, and

b. Initiate actions to restore at least one Shutdown Cooling loop to operation immediately, and

c. Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere within 4 hours.

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.8 Refueling
## 2.8.2 Refueling Operations - Containment

### 2.8.2(1) Containment Penetrations

Applicability

Applies to containment penetrations in MODE 5 during CORE ALTERATIONS and REFUELING OPERATIONS inside containment.

Objective

To minimize the consequences of an accident occurring during CORE ALTERATIONS and REFUELING OPERATIONS inside containment that could affect public health and safety.

Specification

The containment penetrations shall be in the following status:

a. The Equipment Hatch Enclosure (Room 66) doors or the equipment hatch shall be capable of being closed;

b. One door in the Personnel Air Lock shall be capable of being closed; and

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

   1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

   2. capable of being closed.

Note - Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

Required Actions

(1) With one or more containment penetrations not in required status, suspend CORE ALTERATIONS and REFUELING OPERATIONS within containment immediately.

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.8 Refueling
### 2.8.2 Refueling Operations - Containment

#### 2.8.2(2) Refueling Water Level

Applicability

Applies to the refueling water level during CORE ALTERATIONS, and during REFUELING OPERATIONS inside of containment

Objective

To minimize the consequences of a fuel handling accident during CORE ALTERATIONS and REFUELING OPERATIONS inside of the containment that could affect public health and safety.

Specification

The refueling water level shall be ≥23 ft. above the top of the reactor vessel flange.

Required Actions

(1) With the refueling water level not within limits, suspend CORE ALTERATIONS immediately, and

(2) Suspend REFUELING OPERATIONS inside of containment immediately, and

(3) Initiate actions to restore refueling water level to within limits immediately.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling
2.8.2 Refueling Operations - Containment

2.8.2(3) Ventilation Isolation Actuation Signal (VIAS)

Applicability

Applies to operation of the Ventilation Isolation Actuation Signal (VIAS) during CORE ALTERATIONS and REFUELING OPERATIONS inside containment.

Objective

To minimize the consequences of an accident occurring during CORE ALTERATIONS or REFUELING OPERATIONS that could affect public health and safety.

Specification

VIAS, including manual actuation capability, shall be OPERABLE with one gaseous radiation monitor OPERABLE.

Required Actions

(1) Without one radiation monitor OPERABLE, or VIAS manual actuation capability inoperable, immediately suspend CORE ALTERATIONS and REFUELING OPERATIONS.

2.8.2(4) Control Room Ventilation System (CRVS)

Applicability

Applies to operation of the CRVS during CORE ALTERATIONS and REFUELING OPERATIONS inside containment.

Objective

To minimize the consequences of a fuel handling accident to the control room staff.

Specification

The CRVS shall be IN OPERATION and in the Filtered Air mode.

----------Notes----------

1. The control room envelope (CRE) may be opened intermittently under administrative control.
2. Place in toxic gas protection mode immediately if automatic transfer to toxic gas protection mode is not functional.

---

Required Actions

(1) If a CRVS train is not IN OPERATION in Filtered Air mode, immediately place the opposite train IN OPERATION in Filtered Air mode OR immediately suspend CORE ALTERATIONS and REFUELING OPERATIONS.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling
2.8.2 Refueling Operations - Containment

2.8.2(4) Control Room Ventilation System (CRVS) (Continued)

Required Actions (Continued)

(2) If one or more CRVS trains are inoperable due to an inoperable control room envelope (CRE) boundary, immediately suspend CORE ALTERATIONS and REFUELING OPERATIONS.

2.8.3 Refueling Operations - Spent Fuel Pool

2.8.3(1) Spent Fuel Assembly Storage

Applicability

Applies to storage of spent fuel assemblies whenever any irradiated fuel assembly is stored in Region 2 (including peripheral cells) of the spent fuel pool. The provisions of Specification 2.0.1 for Limiting Conditions for Operation are not applicable.

Objective

To minimize the possibility of an accident occurring during REFUELING OPERATIONS that could affect public health and safety.

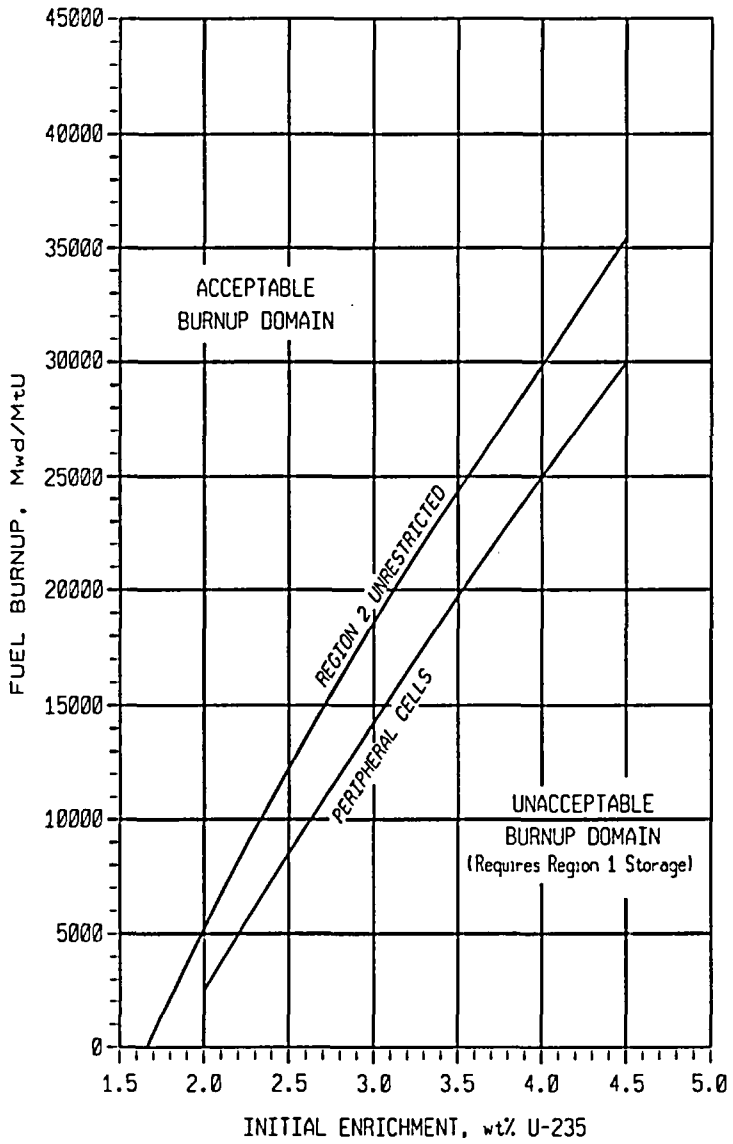
Specification

The combination of initial enrichment and burnup of each spent fuel assembly stored in Region 2 (including peripheral cells) of the spent fuel pool shall be within the acceptable burnup domain of Figure 2-10.

Required Actions

(1) With the requirements of the LCO not met, initiate action to move the noncomplying fuel assembly immediately.

Figure 2-10




LIMITING BURNUP CRITERIA FOR
ACCEPTABLE STORAGE IN REGION 2

NOTES:

1. Any fuel assembly ($\leq$ 4.5% average U-235 enrichment) mechanically coupled with a full length CEA may be located anywhere in Region 2.

2. Peripheral cells are those adjacent to the Spent Fuel Pool wall or the cask laydown area.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling
2.8.3 Refueling Operations - Spent Fuel Pool

2.8.3(2) Spent Fuel Pool Water Level

Applicability

Applies to the water level of the spent fuel pool during REFUELING OPERATIONS in the spent fuel pool. The provisions of Specification 2.0.1 for Limiting Conditions for Operation are not applicable.

Objective

To minimize the consequences of a fuel handling accident during REFUELING OPERATIONS in the spent fuel pool that could affect public health and safety.

Specification

The spent fuel pool water level shall be ≥23 ft. above the top of irradiated fuel assemblies seated in the storage racks.

Required Actions

(1) With the spent fuel pool water level not within limits, suspend REFUELING OPERATIONS in the spent fuel pool immediately.

## 2.0 LIMITING CONDITIONS FOR OPERATION
### 2.8 Refueling
### 2.8.3 Refueling Operations - Spent Fuel Pool

#### 2.8.3(3) Spent Fuel Pool Boron Concentration

Applicability

Applies to the boron concentration of the spent fuel pool when unirradiated fuel assemblies are stored in the spent fuel pool. The provisions of Specification 2.0.1 for Limiting Conditions for Operations are not applicable.

Objective

To minimize the possibility of an accident that could affect public health and safety from occurring when unirradiated fuel assemblies are stored in the spent fuel pool.

Specification

The spent fuel pool boron concentration shall be ≥500 ppm.

Required Actions

(1) With the spent fuel pool boron concentration < 500 ppm, suspend REFUELING OPERATIONS in the spent fuel pool immediately, and

(2) Restore spent fuel pool boron concentration to ≥500 ppm immediately.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling
2.8.3 Refueling Operations - Spent Fuel Pool

2.8.3(4) Spent Fuel Pool Area Ventilation

Applicability

Applies to operation of the ventilation system in the spent fuel pool area during REFUELING OPERATIONS in the spent fuel pool. The provisions of Specification 2.0.1 for Limiting Condition for Operations are not applicable.

Objective

To minimize the consequences of an accident occurring during REFUELING OPERATIONS in the spent fuel pool that could affect public health and safety.

Specification

The spent fuel pool area ventilation system shall be IN OPERATION.

Required Actions

(1) With the spent fuel pool area ventilation system not IN OPERATION, suspend REFUELING OPERATIONS in the spent fuel pool immediately.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling
2.8.3 Refueling Operations - Spent Fuel Pool

2.8.3(5) Control Room Ventilation System (CRVS)

Applicability

Applies to operation of the CRVS during REFUELING OPERATIONS in the spent fuel pool area. The provisions of Specification 2.0.1 for Limiting Conditions for Operation are not applicable.

Objective

To minimize the consequences of a fuel handling accident to the control room staff.

Specification

(1) The CRVS shall be IN OPERATION and in the Filtered Air mode.

(2) A spent fuel pool area radiation monitor shall be IN OPERATION.

----------Notes----------

1. The control room envelope (CRE) may be opened intermittently under administrative control.
2. Place in toxic gas protection mode immediately if automatic transfer to toxic gas protection mode is not functional.

---

Required Actions

(1) If a CRVS train is not IN OPERATION in Filtered Air mode, immediately place the opposite train IN OPERATION in Filtered Air mode OR immediately suspend REFUELING OPERATIONS.

(2) If a spent fuel pool area radiation monitor is not IN OPERATION, immediately suspend REFUELING OPERATIONS.

(3) If one or more CRVS trains are inoperable due to an inoperable control room envelope (CRE) boundary, immediately suspend REFUELING OPERATIONS.

## 2.0 LIMITING CONDITIONS FOR OPERATION
### 2.8 Refueling
### 2.8.3 Refueling Operations - Spent Fuel Pool

#### 2.8.3(6) Spent Fuel Cask Loading

Applicability

Applies to storage of spent fuel assemblies whenever any fuel assembly is located in a spent fuel cask in the spent fuel pool. The provisions of Specification 2.0.1 for Limiting Conditions for Operation are not applicable.

Objective

To minimize the possibility of an accident occurring during REFUELING OPERATIONS that could affect public health and safety.

Specification

(1) The spent fuel pool boron concentration shall be ≥ 800 ppm, and

(2) The combination of initial enrichment and burnup of each spent fuel assembly located in a spent fuel storage cask in the spent fuel pool shall be within the acceptable burnup domain of Figure 2-11.

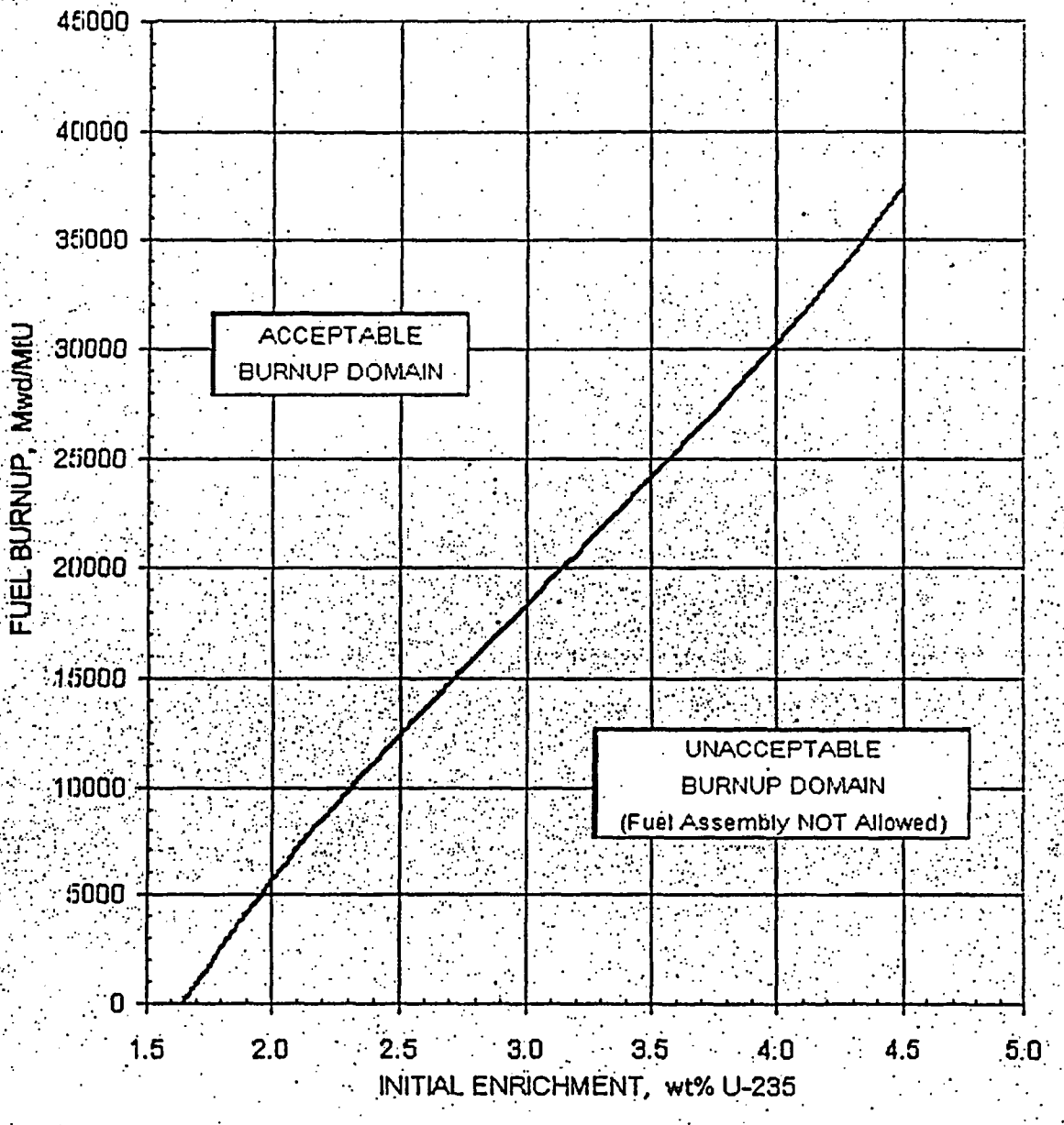
Required Actions

(1) With the spent fuel pool boron concentration <800 ppm, suspend REFUELING OPERATIONS involving spent fuel cask loading immediately, and

(2) Restore spent fuel pool boron concentration to ≥ 800 ppm immediately.

(3) With the requirements of the LCO 2.8.3(6)(2) not met, initiate action to remove the noncomplying fuel assembly from the spent fuel cask immediately.

Figure 2-11




LIMITING BURNUP CRITERIA
FOR
ACCEPTABLE STORAGE IN
SPENT FUEL CASK

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.8 Refueling

### Bases

### 2.8.1 Refueling Shutdown

#### 2.8.1(1) Boron Concentration

The boron concentration of the water filling the reactor refueling cavity (of at least the REFUELING BORON CONCENTRATION) is sufficient to maintain the reactor subcritical by more than 5%, including allowance for uncertainties, in the cold condition with all rods withdrawn. The REFUELING BORON CONCENTRATION is specified in the COLR. Periodic checks of the refueling water boron concentration ensure the proper shutdown margin.

When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe, conservative position.

#### 2.8.1(2) Nuclear Instrumentation

Two OPERABLE source (wide) range neutron monitors are required to provide a signal to ensure that redundant monitoring capability is available to detect changes in core reactivity. With only one source range neutron monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner.

With no source range neutron monitor OPERABLE, there is no means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not being made, the core reactivity condition is stabilized until the monitors are returned to OPERABLE status. This stabilized condition is determined by verifying that the required boron concentration exists. The completion time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. The frequency of once per 12 hours ensures that unplanned changes in boron concentration would be identified. The 12 hour frequency is reasonable, considering the low probability of a change in core reactivity during this period.

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.8 Refueling

Bases (Continued)

### 2.8.1(3) Shutdown Cooling System - High Water Level

The purposes of the SDC system in MODE 5 with fuel in the reactor and with one or more reactor vessel head closure bolt less than fully tensioned, are to remove decay heat and sensible heat from the RCS, provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification.

Inadequate cooling of the reactor coolant could result in boiling of the reactor coolant which could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier.

An OPERABLE SDC loop consists of a SDC pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flowpath and to determine the low end temperature. The flowpath starts in one of the RCS hot legs and is returned to the RCS cold legs. The containment spray pumps can be considered as available shutdown cooling pumps only if the RCS temperature is less than 120°F and the RCS is vented with a vent area greater than or equal to 47 $in^2$. This restrictions ensure that the SI/CS pumps' suction header piping is not subjected to an unanalyzed condition in this mode. Analysis has determined that the minimum required RCS vent area is 47 $in^2$. This requirement may be met by removal of the pressurizer manway which has a cross-sectional area greater than 47 $in^2$.

Specification 2.8.1(3) is modified by an exception that allows the required operating SDC loop to be removed from service for up to 1 hour in each 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, and RCS to SDC isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity. Boron concentration reduction is prohibited because uniform concentration distribution cannot be assured without forced circulation.

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.8 Refueling

Bases (Continued)

2.8.1(3) Shutdown Cooling System - High Water Level (Continued)

Specification 2.8.1(3) is modified by an exception to allow both trains of SDC out of service for up to eight hours provided, in part, that at least one SDC train is available under administrative controls. This allows evolutions such as Engineered Safety Feature testing to be completed when the SDC system is not fully OPERABLE but is considered available since only minor operator actions are required to restore the SDC system to OPERABLE status and place it IN OPERATION. A SDC loop is considered available under administrative controls if there are: (1) approved procedures, (2) a dedicated operator stationed at the controls if they are outside of the control room, and (3) direct communication between the dedicated operator and the control room. Similarly, the SDC system is considered available under administrative controls when an operator is not at the location of the controls provided: (1) procedural guidance is consulted prior to removing SDC from service to determine the time-to-boil, and (2) there is sufficient time for the operator to travel to the local controls and perform the required actions.

With the water level ≥ 23 feet above the top of the core, only one SDC loop is required for decay heat removal. Only one is required because the volume of water above the top of the core provides backup decay heat removal capability. The 23 ft level was selected because it ensures that adequate time is available to initiate emergency procedures to cool the core. For example, assuming the amount of decay heat generated one day after shutdown with an initial reactor coolant temperature of 200°F, this level provides approximately 25 minutes before the reactor coolant would boil. More time is available under conditions more representative of when this specification applies (i.e., when the reactor vessel closure head is removed). For example, five days after shutdown with the initial reactor coolant temperature of 130°F provides more than four hours before the reactor coolant would boil.

If the SDC loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Therefore, actions that reduce boron concentration are required to be suspended immediately. Additionally, suspending any operation that would increase the decay heat load, such as loading a fuel assembly, is a prudent action under this condition. Closing the containment penetrations that provide direct access to the outside environment prevents fission products, if released from a loss of decay heat removal event, from escaping the containment. A completion time of 4 hours is reasonable because most SDC problems can be repaired within 4 hours and because there is a low probability of the cooling boiling in that time.

When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner.

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.8 Refueling

Bases (Continued)

### 2.8.1(4) Shutdown Cooling System - Low Water Level

With the water level < 23 feet above the top of the core, two OPERABLE SDC loops are required since the volume of water above the top of the core will not provide sufficient backup decay heat removal capability.

With one SDC loop inoperable, actions shall be immediately initiated and continued until the SDC loop is restored to OPERABLE status, or until ≥ 23 feet of water is established above the top of the core. With the water level ≥ 23 feet above the top of the core, this Specification is no longer applicable, and Specification 2.8.1(3) is applicable.

An OPERABLE SDC loop consists of a SDC pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flowpath and to determine the low end temperature. The flowpath starts in one of the RCS hot legs and is returned to the RCS cold legs. The containment spray pumps can be considered as available shutdown cooling pumps only if the RCS temperature is less than 120°F and the RCS is vented with a vent area greater than or equal to 47 $in^2$.

With both SDC loops inoperable, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Therefore, actions that reduce boron concentration are required to be suspended immediately. Closing the containment penetrations that provide direct access to the outside environment prevents fission products, if released from a loss of decay heat removal event, from escaping the containment. A completion time of 4 hours is reasonable because most SDC problems can be repaired within 4 hours and because there is a low probability of the coolant boiling in that time.

When "immediately" is used in a completion time, the required action should be pursued without delay and in a controlled manner.

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.8 Refueling

Bases (Continued)

### 2.8.2(1) Containment Penetrations

During CORE ALTERATIONS or REFUELING OPERATIONS inside containment, a release of fission product radioactivity within the containment will be minimized from escaping to the environment when the LCO requirements are met. In MODE 5, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere are less stringent than when the reactor is at power. The LCO does not require CONTAINMENT INTEGRITY. Since there is no potential for containment pressurization as a result of a fuel handling accident, the Appendix J leakage criteria and tests are not required.

For a fuel handling accident in containment, the very conservative assumption that all the rods in a single assembly fail with no credit for containment isolation or atmosphere filtration yields worst 2-hour doses at the exclusion area boundary (EAB) and low population zone (LPZ) that remain well within the limits of 10 CFR 50.67.

During CORE ALTERATIONS or REFUELING OPERATIONS inside of containment, the Equipment Hatch Enclosure (Room 66) doors or the equipment hatch shall be capable of being closed within one hour after a fuel handling accident per administrative controls. Placing administrative controls (closure requirements) on the Equipment Hatch Enclosure (Room 66) doors or the equipment hatch ensures that the release of fission products is minimized (defense in depth).

The Personnel Air Lock (PAL), which is also part of the containment pressure boundary, provides a means for personnel access into containment. The doors are normally interlocked to prevent simultaneously opening when CONTAINMENT INTEGRITY is required. During periods of shutdown when containment closure is not required, the interlock may be disabled and both PAL doors allowed to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or REFUELING OPERATIONS inside containment, CONTAINMENT INTEGRITY is not required, therefore the door interlock mechanism may remain disabled, but one PAL door shall always remain capable of being closed.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere shall be capable of being closed within one hour, per administrative controls, on at least one side. The specification is met when one of the two automatic isolation valves per penetration is OPERABLE, or by closure of a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved (through 10 CFR 50.59 safety evaluation process) and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the other containment penetrations during fuel movements.

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.8 Refueling

Bases (Continued)

### 2.8.2(1) Containment Penetrations (Continued)

The administrative controls to ensure closure of the Equipment Hatch Enclosure (Room 66) doors or equipment hatch, one PAL door, and other penetrations within one hour of a FHA will be implemented in plant procedures. These administrative controls are as follows:

a. the Equipment Hatch Enclosure (Room 66) doors or the equipment hatch and one door in the PAL shall be capable of being closed in less than one hour of a FHA,

b. the Equipment Hatch Enclosure (Room 66) doors or the equipment hatch and one door in the PAL shall not be obstructed unless capability for rapid removal of obstructions is provided (such as quick disconnects for hoses),

c. penetrations providing direct access from the containment atmosphere to the outside atmosphere shall be capable of being closed on one side in less than one hour of a FHA,

d. an individual or individuals shall be designated and available during CORE ALTERATIONS and REFUELING OPERATIONS, capable of closing the Equipment Hatch Enclosure (Room 66) doors or equipment hatch, one door in the PAL, and penetrations that provide direct access from the containment atmosphere to the outside atmosphere.

The required actions shall be completed within one hour after the time of a FHA. Provision of these required actions minimizes the release of fission product radioactivity. The fuel handling accident in containment uses the conservative assumptions that activity is instantaneously released to the reactor coolant cavity water and then released over a two-hour time period from containment to the environment. Implementing closure of containment within one hour from the time of accident minimizes the dose consequences to the EAB and LPZ.

When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner.

### 2.8.2(2) Refueling Water Level

Prior to REFUELING OPERATIONS inside containment, the reactor refueling cavity is filled with approximately 250,000 gallons of borated water. The minimum refueling water level meets the assumption of iodine decontamination factors following a fuel handling accident. When the water level is lower than the required level, CORE ALTERATIONS and REFUELING OPERATIONS inside of containment shall be suspended immediately. This effectively precludes a fuel handling accident from occurring. When "immediately" is used as a completion time, the required action

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.8 Refueling

Bases (Continued)

2.8.2(2) Refueling Water Level (Continued)

should be pursued without delay and in a controlled manner. Suspension of REFUELING OPERATIONS and CORE ALTERNATIONS shall not preclude completion of movement of a component to a safe, conservative position. In addition to suspending REFUELING OPERATIONS and CORE ALTERATIONS, action to restore the refueling water level must be initiated immediately.

Movement of irradiated fuel from the reactor core is not initiated before the reactor core has been subcritical for a minimum of 72 hours if the reactor has been operated at power levels in excess of 2% rated power. The restriction of not moving fuel in the reactor for a period of 72 hours after the power has been removed from the core takes advantage of the decay of the short half-life fission products and allows for any failed fuel to purge itself of fission gases, thus reducing the consequences of a fuel handling accident.

2.8.2(3) Ventilation Isolation Actuation Signal (VIAS)

A Ventilation Isolation Actuation Signal (VIAS) is initiated by a Safety Injection Actuation Signal (SIAS), a Containment Spray Actuation Signal (CSAS) or a Containment Radiation High Signal (CRHS). During CORE ALTERATIONS and REFUELING OPERATIONS only the CRHS is required to respond to a fuel handling or reactivity accident. At least one of the following three radiation monitors (Containment Monitor (RM-051), Containment/Auxiliary Building Stack Swing Monitor (RM-052), Auxiliary Building Stack Radiation Monitor (RM-062) must be OPERABLE and aligned to monitor the containment atmosphere or stack effluents. (Note, the Offsite Dose Calculation Manual may have additional requirements/restrictions concerning operation of these monitors.)

In the event that none of the above radiation monitors are OPERABLE or VIAS manual actuation capability is inoperable, CORE ALTERATIONS and REFUELING OPERATIONS must be suspended thus precluding the possibility of a fuel handling/reactivity accident.

For the fuel handling accident in containment, the very conservative assumption that all the rods in a single assembly fail with no credit taken for containment isolation or atmosphere filtration yields doses at the exclusion area boundary (EAB) and low population zone (LPZ) that remain well within the limits of 10 CFR 50.67.

VIAS initiates closure of the containment pressure relief, air sample, and purge system valves, if open. This action minimizes release of significant radionuclides from the containment to the environment. VIAS also initiates other actions, such as opening of the air supply and exhaust dampers in the safety injection pump rooms in preparation for safety injection pump operation. These other functions are not required to mitigate the consequences of a fuel handling accident, and therefore are not required to be OPERABLE.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling
Bases (Continued)

2.8.2(3) Ventilation Isolation Actuation Signal (VIAS) (Continued)

Requiring one (1) radiation monitor to be OPERABLE and aligned to monitor the containment atmosphere [or stack effluents] is a conservative measure to reduce exposure. Radiation monitoring will assure operators are alerted if a radiological incident occurs in containment to enable implementation of administrative controls as specified in the Bases for 2.8.2(1) "Containment Penetrations." During CORE ALTERATIONS and REFUELING OPERATIONS, the OPERABILITY of the control room ventilation system is addressed by Specification 2.8.2(4). The control room ventilation system is placed in Filtered Air mode as a conservative measure to reduce control room operator exposure. Specification 2.8.2(4) allows the radiological consequences analysis for a fuel handling accident to credit the Filtered Air mode at the time of the accident.

When VIAS is inoperable, CORE ALTERATIONS and REFUELING OPERATIONS in containment are immediately suspended. This effectively precludes a fuel handling accident from occurring. When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner. Suspension of CORE ALTERATIONS and REFUELING OPERATIONS shall not preclude completion of movement of a component to a safe, conservative position.

2.8.2(4) Control Room Ventilation System (CRVS)

Operating the CRVS in the Filtered Air mode is a conservative measure to reduce control room operator exposure. This allows the radiological consequences analysis for a fuel handling accident to credit the Filtered Air mode at the time of the accident. If a CRVS train is not IN OPERATION in Filtered Air mode, the opposite train must immediately be placed IN OPERATION in Filtered Air mode. This action ensures that the remaining train is OPERABLE, and that any active failure will be readily detected. An alternative is to immediately suspend activities (CORE ALTERATIONS and REFUELING OPERATIONS) that could result in a release of radioactivity that might require isolation of the control room envelope (CRE).

Similarly, with one or more CRVS trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities (CORE ALTERATIONS and REFUELING OPERATIONS) that could result in a release of radioactivity that might require isolation of the CRE.

These actions place the unit in a condition that minimizes the accident risk.

When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner. Suspension of CORE ALTERATIONS and REFUELING OPERATIONS shall not preclude completion of movement of a component to a safe, conservative position.

2.8.2(4) Control Room Ventilation System (CRVS) (Continued)

The Specification is modified by two notes. The first note allows the CRE boundary to be opened intermittently under administrative controls. This only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

The second note requires the CRVS to be in toxic gas protection mode if automatic transfer to toxic gas protection mode is not functional. CORE ALTERATIONS and REFUELING OPERATIONS must be suspended immediately. Toxic gas is monitored at the outside air intake duct. Actuation of the system to toxic gas protection mode trips CRVS fans and isolates the outside air dampers. The CRVS is then placed in recirculation mode. In recirculation mode, the filter trains are bypassed.

Fire and smoke detection is provided at the outlet of the recirculation fans to protect against smoke developed from sources in the outside air stream or from sources inside the control room. As in toxic gas protection mode, CRVS fans are tripped and the outside air dampers are isolated.

2.8.3(1) Spent Fuel Assembly Storage

The spent fuel pool is designed for noncriticality by use of neutron absorbing material. The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Figure 2-10, and the accompanying LCO, ensures that the $k_{eff}$ of the spent fuel pool always remains < 0.95 assuming the pool to be flooded with unborated water.

A spent fuel assembly may be transferred directly from the reactor core to the spent fuel pool Region 2 provided an independent verification of assembly burnups has been completed and the assembly burnup meets the acceptance criteria identified in Figure 2-10. When the configuration of fuel assemblies stored in Region 2 (including the peripheral cells) is not in accordance with Figure 2-10, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 2-10. Acceptable fuel assembly burnup is not a prerequisite for Region 1 storage because Region 1 will maintain any type of fuel assembly that the plant is licensed for in a safe, coolable, subcritical geometry.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling
Bases (Continued)

2.8.3(1) Spent Fuel Assembly Storage (Continued)

The provisions of Specification 2.0.1 for Limiting Conditions for Operations are not applicable. If moving fuel assemblies while in MODES 4 or 5, LCO 2.0.1 would not specify any actions. If moving fuel assemblies in MODES 1, 2, or 3, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown. When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner.

2.8.3(2) Spent Fuel Pool Water Level

The minimum water level in the spent fuel pool meets the assumption of iodine decontamination factors following a fuel handling accident. When the water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel pool is immediately suspended. This effectively precludes a fuel handling accident from occurring in the spent fuel pool. Suspension of REFUELING OPERATION shall not preclude completion of movement of a component to a safe, conservative position. The provisions of Specification 2.0.1 for Limiting Conditions for Operations are not applicable. If moving fuel assemblies while in MODES 4 or 5, LCO 2.0.1 would not specify any actions. If moving fuel assemblies in MODES 1, 2, or 3, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown. When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner.

2.8.3(3) Spent Fuel Pool Boron Concentration

The basis for the 500 ppm boron concentration requirement with Boral poisoned storage racks is to maintain the $k_{eff}$ below 0.95 in the event a misloaded unirradiated fuel assembly is located next to a spent fuel assembly. A misloaded unirradiated fuel assembly at maximum enrichment condition, in the absence of soluble poison, may result in exceeding the design effective multiplication factor. Soluble boron in the spent fuel pool water, for which credit is permitted under these conditions, would assure that the effective multiplication factor is maintained substantially less than the design condition.

This LCO applies whenever unirradiated fuel assemblies are stored in the spent fuel pool. The boron concentration is periodically sampled in accordance with Specification 3.2. Sampling is performed prior to movement of unirradiated fuel to the spent fuel pool and periodically when unirradiated fuel is stored in the spent fuel pool.

2.8.3(3) Spent Fuel Pool Boron Concentration (Continued)

The provisions of Specification 2.0.1 for Limiting Conditions for Operations are not applicable. If moving fuel assemblies while in MODES 4 or 5, LCO 2.0.1 would not specify any actions. If moving fuel assemblies in MODES 1, 2, or 3, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner. Suspension of refueling operations shall not preclude completion of movement of a component to a safe, conservative position.

2.8.3(4) Spent Fuel Pool Area Ventilation

The spent fuel pool area ventilation system contains a charcoal filter to prevent release of significant radionuclides to the outside atmosphere. The system does not automatically realign and therefore must be IN OPERATION prior to REFUELING OPERATIONS in the spent fuel pool. When the spent fuel pool area ventilation system is not IN OPERATION, the movement of irradiated fuel assemblies in the spent fuel pool is immediately suspended. This effectively precludes a fuel handling accident from occurring in the spent fuel pool. When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner. Suspension of REFUELING OPERATIONS shall not preclude completion of movement of a component to a safe, conservative position.

The provisions of Specification 2.0.1 for Limiting Conditions for Operations are not applicable. If moving fuel assemblies while in MODES 4 or 5, LCO 2.0.1 would not specify any actions. If moving fuel assemblies in MODES 1, 2, or 3, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

2.8.3(5) Control Room Ventilation System (CRVS)

Operating the CRVS in the Filtered Air mode and requiring a radiation monitor to be IN OPERATION are conservative measures to reduce control room operator exposure. This allows the radiological consequences analysis for a fuel handling accident to credit the Filtered Air mode at the time of the accident.

Radiation monitoring will assure operators are alerted if a radiological incident occurs. This specification can be satisfied by using a permanent spent fuel pool area radiation monitor or a portable area radiation monitor.

## 2.0 **LIMITING CONDITIONS FOR OPERATION**
## 2.8 Refueling

Bases (Continued)

2.8.3(5) Control Room Ventilation System (CRVS) (Continued)

If a CRVS train is not IN OPERATION in Filtered Air mode, the opposite train must immediately be placed IN OPERATION in Filtered Air mode. This action ensures that the remaining train is OPERABLE, and that any active failure will be readily detected. An alternative is to immediately suspend activities (REFUELING OPERATIONS) that could result in a release of radioactivity that might require isolation of the control room envelope (CRE).

Similarly, with one or more CRVS trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities (REFUELING OPERATIONS) that could result in a release of radioactivity that might require isolation of the CRE.

These actions place the unit in a condition that minimizes the accident risk.

When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner. Suspension of REFUELING OPERATIONS shall not preclude completion of movement of a component to a safe, conservative position.

The Specification is modified by two notes. The first note allows the control room envelope (CRE) boundary to be opened intermittently under administrative controls. This only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

The second note requires the CRVS to be in toxic gas protection mode if automatic transfer to toxic gas protection mode is not functional. CORE ALTERATIONS and REFUELING OPERATIONS must be suspended immediately. Toxic gas is monitored at the outside air intake duct. Actuation of the system to toxic gas protection mode trips CRVS fans and isolates the outside air dampers. The CRVS is then placed in recirculation mode. In recirculation mode, the filter trains are bypassed.

Fire and smoke detection is provided at the outlet of the recirculation fans to protect against smoke developed from sources in the outside air stream or from sources inside the control room. As in toxic gas protection mode, CRVS fans are tripped and the outside air dampers are isolated.

## 2.0 **LIMITING CONDITIONS FOR OPERATION**
## 2.8 Refueling

Bases (Continued)

### 2.8.3(6) Spent Fuel Cask Loading

(1) Soluble Boron

The basis for the 800 ppm minimum boron concentration requirement during spent fuel cask loading operations is to maintain the $k_{eff}$ in the cask system less than or equal to 0.95 in the event a mis-loaded unirradiated fuel assembly is located anywhere in the cask with up to 31 other fuel assemblies meeting the burnup and enrichment requirements of LCO 2.8.3(6)(2). This boron concentration also ensures the $k_{eff}$ in the cask system will be less than or equal to 0.95 if an unirradiated fuel assembly is dropped in the space between the spent fuel racks and the cask loading area during cask loading operations next to a spent fuel assembly. A mis-loaded or dropped irradiated fuel assembly at maximum enrichment condition, in the absence of soluble poison, may result in exceeding the design effective multiplication factor. Soluble boron in the spent fuel pool water, for which credit is permitted during spent fuel cask loading operations, assures that the effective multiplication factor is maintained substantially less than the design basis limit.

This LCO applies whenever a fuel assembly is located in a spent fuel cask submerged in the spent fuel pool. The boron concentration is periodically sampled in accordance with Specification 3.2. Sampling is performed prior to movement of fuel into the spent fuel cask and periodically thereafter during cask loading operations, until the cask is removed from the spent fuel pool.

The provisions of Specification 2.0.1 for Limiting Conditions for Operations are not applicable. If moving fuel assemblies while in MODES 4 or 5, LCO 2.0.1 would not specify any actions. If moving fuel assemblies in MODES 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner. Suspension of refueling operations shall not preclude completion of movement of a component to a safe, conservative position.

## 2.0 **LIMITING CONDITIONS FOR OPERATION**
## 2.8 Refueling

Bases (Continued)

2.8.3(6) Spent Fuel Cask Loading (Continued)

(2) Burnup vs. Enrichment

The spent fuel cask is designed for subcriticality by use of neutron absorbing material. The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Figure 2-11, and the accompanying LCO, ensure that the $k_{eff}$ of the spent fuel pool always remains ≤ 0.95 assuming the pool to be flooded with borated water and <1.0 assuming the pool is flooded with unborated water, in accordance with 10 CFR 50.68(b)(4).

A spent fuel assembly may be transferred directly from the spent fuel racks to the spent fuel cask provided an independent verification of assembly burnups has been completed and the assembly burnup meets the acceptance criteria identified in Figure 2-11. If any fuel assembly located in the spent fuel cask is not in accordance with Figure 2-11, immediate action must be taken to make the remove of non-complying fuel assembly from the spent fuel cask and return it to the spent fuel rack.

The provisions of Specification 2.0.1 for Limiting Conditions for Operations are not applicable. If moving fuel assemblies while in MODES 4 or 5, LCO 2.0.1 would not specify any actions. If moving fuel assemblies in MODES 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown. When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner.

References

(1) USAR Section 9.5
(2) USAR Section 9.10
(3) USAR Section 14.18

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.9 Radioactive Waste Disposal System

Applicability

Applies to the transfer of waste gases to the waste gas decay tanks. The provisions of Technical Specification 2.0.1 for Limiting Condition for Operation are not applicable.

Objective

To ensure compliance with General Design Criterion 60 of Appendix A to 10 CFR 50.

Specification

(1) The concentration of hydrogen and oxygen in the waste gas decay tanks shall be limited to below flammability concentrations. With hydrogen and oxygen concentrations above flammability concentrations, restore the concentrations to below flammability limits within 48 hours.

(2) The hydrogen and oxygen monitors shall be monitoring the inservice gas decay tank during the transfer of waste gases to the waste gas decay tank. Whenever the monitors are inoperable, transfer of waste gases to a gas decay tank may continue provided grab samples are taken from the gas decay tank and analyzed:

a. Every eight hours during degassing operations, and
b. Daily during other operations.

Basis

Specification 2.9 ensures that the concentration of potentially explosive gas mixtures entrained in the gas decay tank(s) will be maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits with a measurement program provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

# 2.0 LIMITING CONDITIONS FOR OPERATION

## 2.10 Reactor Core

### 2.10.1 Minimum Conditions for Criticality

Applicability

Applies to the status of the reactor coolant system during reactor criticality.

Objective

To prevent unanticipated power excursions of an unsafe magnitude.

Specifications

(1) The reactor shall not be made critical if the average reactor coolant temperature is below 515°F.

(2) No more than one CEA at a time in a regulating or non-trippable group shall be exercised or withdrawn until after a steam bubble and normal water level as given in operating procedures are established in the pressurizer.

(3) Reactor coolant boron concentration shall not be reduced below that required for the Hot Shutdown Condition until after a steam bubble and normal water level are established in the pressurizer.

Basis

At the beginning of each fuel cycle, the moderator temperature coefficient is expected to be slightly negative at operating temperatures with all CEA's withdrawn. However, variations in cycle core loading and the uncertainty of the calculation are such that it is possible that a slightly positive coefficient could exist.

The moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature. It is, therefore, prudent to restrict the operation of the reactor when reactor coolant temperatures are less than 515°F.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.10 Reactor Core (Continued)

#### 2.10.1 Minimum Conditions for Criticality (Continued)

If the shutdown margin required by the Hot Shutdown Condition is maintained, there is no possibility of an accidental criticality as a result of a change of moderator temperature or a decrease of coolant pressure. Normal water level is established in the pressurizer prior to the withdrawal of CEA or the dilution of boron so as to preclude the possible overpressurization of a solid reactor coolant system.

During physics tests, special operating precautions will be taken. In addition, the strong negative effect of the Doppler coefficient would limit the magnitude of a power excursion resulting from a reduction of moderator density.

## 2.0 LIMITING CONDITIONS FOR OPERATION
### 2.10 Reactor Core (Continued)
### 2.10.2 Reactivity Control Systems and Core Physics Parameters Limits

Applicability

Applies to operation of control element assemblies and monitoring of selected core parameters whenever the reactor is in cold or hot shutdown, hot standby, or power operation conditions.

Objective

To ensure (1) adequate shutdown margin following a reactor trip, (2) the MTC is within the limits of the safety analysis, and (3) control element assembly operation is within the limits of the setpoint and safety analysis.

Specification

(1) Shutdown Margin With $T_{cold}$ >210°F

Whenever the reactor is in hot shutdown, hot standby or power operation conditions, the shutdown margin shall be greater than or equal to the value specified in the COLR. With the shutdown margin less than the value specified in the COLR, initiate and continue boration until the required shutdown margin is achieved.

(2) Shutdown Margin With $T_{cold}$<210°F

Whenever the reactor is in cold shutdown conditions, the shutdown margin shall be ≥3.0% Δk/k. With the shutdown margin <3.0% Δk/k, initiate and continue boration until the required shutdown margin is achieved.

(3) Moderator Temperature Coefficient

The moderator temperature coefficient (MTC) shall be:

a. Less positive than $+0.2 \times 10^{-4}$ Δρ/°F including uncertainties for power levels at or above 80% of rated power.

b. Less positive than $+0.5 \times 10^{-4}$ Δρ/°F including uncertainties for power levels below 80% of rated power.

c. More positive than the value specified in the COLR including uncertainties at rated power.

With the moderator temperature coefficient confirmed outside any one of above limits, change reactivity control parameters to bring the extrapolated MTC value within the above limits within 3 hours or be in at least hot shutdown within 6 hours.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.2 Reactivity Control Systems and Core Physics Parameters Limits (Continued)

Control Element Assemblies

(4) Full Length CEA Position During Power Operation

All full length (shutdown and regulating) CEA's shall be operable with each CEA of a given group positioned within 12 inches (actual position) of all other CEA's in its group. If one or more of the CEA's is inoperable or misaligned, determine the cause and comply with one of the following:

a. If one or more full length CEA's are inoperable due to: 1) being immovable as a result of excessive friction or mechanical interference, or 2) known to be untrippable, determine that the shutdown margin requirement of Specification 2.10.2(1) is satisfied within 1 hour and be in at least hot shutdown within 7 hours.

b. With one full length CEA inoperable due to causes other than addressed in item a. above, and inserted beyond the Long Term Steady State Insertion Limits but within its above specified alignment requirements, power operation may continue for up to 7 EFPD's per occurrence with a total accumulated time of <14 EFPD per fuel cycle.

c. With one full length CEA inoperable due to causes other than addressed in item a. above, but within its above specified alignment requirements and either fully withdrawn or above the Long Term Steady State Insertion Limits if in CEA group 4, power operation may continue.

d. With one or more full length CEA's misaligned from any other CEA's in its group by more than 12 inches but less than 18 inches (actual position) within one hour either:

(i) Restore the misaligned CEA(s) to within 12 inches (actual position) of any other CEA's in its own group (realignment shall be made while maintaining the allowable CEA sequence and CEA insertion limits of the Power Dependent Insertion Limits Figure provided in the COLR; or

(ii) Declare the CEA's inoperable. Power operation may continue provided all of the following conditions are met:

1. The power level shall be reduced to ≤ 70% of the maximum allowable power level for the existing Reactor Coolant Pump combination within an additional one hour; if negative reactivity insertion is required to reduce power, boration shall be used.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.2 Reactivity Control Systems and Core Physics Parameters Limits (Continued)

2. Within one hour after reducing the power as required by 1., above, the remainder of the CEA's in the group with the inoperable CEA shall be aligned to within 12 inches (actual position) of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits for Power Dependent Insertion Limit Figure provided in the COLR, the power level shall be restricted pursuant to the applicable section of b. or c. above during subsequent operation.

or (iii) Be in hot shutdown within the next 6 hours.

e. With one full length CEA misaligned from any other CEA's in its group by 18 inches or more (actual position), reduce power to ≤70% of the maximum allowable power level for the existing Reactor Coolant Pump combination within one hour. If negative reactivity insertion is required to reduce power, boration shall be used. Within one hour after reducing power as required above, either:

(i) Restore the CEA(s) to within 12 inches (actual position) of any other CEA's in its group (realignment shall be made while maintaining the allowable CEA sequence and CEA insertion limits of the Power Dependent Insertion Limit Figure in the COLR); or

(ii) Declare the CEA(s) inoperable and determine that the shutdown margin requirement of Specification 2.10.2(1) is satisfied and align the remainder of the CEA's in the group with the inoperable CEA to within 12 inches (actual position) of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown in the Power Dependent Insertion Limit Figure provided in the COLR. The shutdown margin requirement of Specification 2.10.2(1) shall be determined at least once per shift. The power level shall be restricted pursuant to the applicable section of b. or c. above during subsequent operation; or

(iii) Be in hot shutdown within an additional 5 hours.

f. With more than one full length CEA inoperable or misaligned from any other CEA in its group by 18 inches (actual position) or more, be in at least hot shutdown within 7 hours.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.10 Reactor Core (Continued)

2.10.2 Reactivity Control Systems and Core Physics Parameters Limits (Continued)

(5) Non-trippable CEA Position During Power Operation

All non-trippable CEAs (NTCEA) shall be withdrawn to at least 114 inches (actual position). If one or more NTCEAs becomes misaligned from other NTCEAs by more than 12 inches (actual position) either:

a. Restore the NTCEA to within the specified alignment requirements within one hour, or
b. Be in at least hot shutdown within an additional 6 hours.

(6) Shutdown CEA Insertion Limit During Power Operation

All shutdown CEAs shall be withdrawn to at least 114 inches as a condition for reactor criticality, or with one or more shutdown CEAs withdrawn less than 114 inches, except for surveillance testing, within one hour, either:

a. Withdraw the CEAs to at least 114 inches, or
b. Declare the CEAs inoperable and apply Specification 2.10.2(4).

(7) Regulating CEA Insertion Limits During Hot Standby and Power Operation

The regulating CEA groups shall be positioned within the acceptable operating range for regulating rod position of the Power Dependent Insertion Limits Figure provided in the COLR except during CEA exercises above 114 inches. With all CEAs operable, CEA insertion beyond the Long Term Insertion Limits is restricted to:

1. 4 hours per 24 hour interval,
2. 4 EFPD per 30 EFPD interval, and
3. 14 EFPD per fuel cycle.

a. When the regulating CEA groups are inserted beyond the Transient Insertion Limits within two hours, either:

(i) Restore the regulating CEA groups to above the Transient Insertion Limits, or

(ii) Reduce reactor power to the allowed power of the Power Dependent Insertion Limit Figure of the COLR which permits continued operation above the Transient Insertion Limit using the existing CEA group position.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.10 Reactor Core (Continued)

2.10.2 Reactivity Control Systems and Core Physics Parameters Limits (Continued)

b. When the regulating CEA groups are inserted beyond the Long Term Insertion Limit for a time interval in excess of 4 hours per 24 hour interval, operation may proceed deleting this daily restriction, provided either:

(i) Regulating CEA groups are not inserted below the Short Term Insertion Limit, or

(ii) Regulating CEA groups are not inserted below the Transient Insertion Limit and rates of power increases initiated when the regulating CEA's are inserted below the Short Term Insertion Limit are less than 5%/hour.

c. When the regulating CEA groups are inserted below the Long Term Insertion Limit for time intervals in excess of 4 EFPD per 30 EFPD interval or in excess of 14 EFPD per fuel cycle, either:

(i) Restore the regulating groups to within the Long Term Insertion Limit within two hours, or

(ii) Be in hot shutdown within 6 hours.

(8) CEA Drop Time

The individual full length (shutdown and regulating) CEA drop time, from a fully withdrawn position, shall be ≤ 2.5 seconds from the time the clutch coil is de-energized until the CEA reaches its 90 percent insertion position with:

a. $T_{cold} \geq 515°F$, and
b. All reactor coolant pumps operating.

With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to hot standby or power operation.

(9) Test Exemption

a. CEA Insertion Limits and Misalignment

(i) The insertion limits of Specification 2.10.2 may be suspended during the performance of physics tests provided:

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.2 Reactivity Control Systems and Core Physics Parameters Limits (Continued)

1. The power is restricted to the test power plateau which shall not exceed (85%) of rated power, and

2. The power shall be determined at least once per hour during physics tests.

3. The linear heat rate shall be determined to be within the limits of Specification 2.10.4(1) by monitoring it continuously with the Incore Detector Monitoring System during physics tests above 5% of rated power.

(ii) The alignment requirements of Specification 2.10.2(4) and insertion limits of Specification 2.10.2(7) may be suspended during the performance of physics tests to determine the isothermal temperature coefficient and power coefficient provided:

1. Only the center CEA (CEA 4-1) is misaligned, and

2. The linear heat rate shall be determined to be with the limits of Specification 2.10.4(1) by monitoring it continuously with the Incore Detector Monitoring System during physics test above 5% of rated power.

(iii) With any of the limits of Specification 2.10.4(1) being exceeded while the requirements of Specification 2.10.2(4), (5), (6), and (7) are suspended, either:

1. Reduce power sufficiently to satisfy the requirements of Specification 2.10.4(1), or

2. Be in hot shutdown within 6 hours.

b. Shutdown Margin

(i) The shutdown margin required by 2.10.2(1) may be reduced during physics testing at power levels less than $10^{-1}$% of rated power for measurement of CEA worth and shutdown margin provided that:

1. Reactivity equivalent to at least the highest estimated CEA worth is available from the operable CEA groups withdrawn (assuming the most reactive CEA of the groups withdrawn is stuck in the fully withdrawn position), and

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.2 Reactivity Control Systems and Core Physics Parameters (Continued)

2. The position of each trippable CEA required shall be determined at least once per 2 hours, and

3. Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the Shutdown Margin to less than the limits of Specification 2.10.2(1).

(ii) If the shutdown margin specified in part (i) above is not available, immediately initiate and continue boration until the requirements of 2.10.2(1) are met.

c. Moderator Temperature Coefficient

(i) The moderator temperature coefficient (MTC) requirements of 2.10.2(3) may be suspended during physics tests at less than $10^{-1}$% of rated power.

(ii) If power exceeds $10^{-1}$% of rated power, either:

1. Reduce power to less than $10^{-1}$% of rated power within 15 minutes, or

2. Be in hot shutdown in 2 hours.

Basis
Shutdown Margin

A sufficient shutdown margin ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Shutdown margin requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS $T_{avg}$. The most restrictive condition occurs at EOL, with $T_{avg}$ at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum shutdown margin as specified in the COLR is initially adequate to control the reactivity transient. Accordingly,

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.2 Reactivity Control Systems and Core Physics Parameters Limits (Continued)

the shutdown margin requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With $T_{cold} \leq 210°F$, the reactivity transients resulting from any postulated accident are minimal and a 3.0% Δk/k shutdown margin provides adequate protection.

Control Element Assemblies

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum shutdown margin is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The specifications applicable to one or more CEA's that are determined to be untrippable or stuck, and to one or more misaligned CEA's that cannot be restored to within 12 inches of any other CEA in their group, require a prompt shutdown of the reactor since any of these conditions may be indicative of a possible loss of mechanical functional capability of the CEA system and in the event of any untrippable CEA, the loss of shutdown margin.

For small misalignments (<18 inches absolute) of the CEA's, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCO's and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCO's and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available shutdown margin, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the action statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in power. The one hour time is sufficient to 1) identify causes of a misaligned CEA, 2) take appropriate corrective action to realign the CEA's, and 3) minimize the effects of xenon redistribution.

Overpower margin is provided to protect the core in the event of a large misalignment (≥18 inches) or a CEA. However, this misalignment would cause distortion of the core power distribution. The reactor protective system would not detect the degradation in radial peaking factors and since variations in other systems parameters (e.g., pressure and coolant temperature) may not be sufficient.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.2 Reactivity Control Systems and Core Physics Parameters Limits (Continued)

to cause trips, it is probable that, if the reactor continued to operate in this mode, process variables could be less conservative than those assumed in generating LCO and LSSS setpoints. Therefore, the action statement associated with the large misalignment of a CEA requires a prompt and significant reduction in power prior to attempting realignment of the misaligned CEA.

The action statements applicable to misaligned and inoperable CEA' s include requirements to align the operable CEA's in a given group with the inoperable CEA. Conformance with these alignment requirements brings the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEA's significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors, and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEA's inserted beyond the Long Term Steady State Insertion Limits to preclude such adverse conditions from developing.

CEA Insertion Limits

The Transient Insertion Limits of Specifications 2.10.2(7) are provided to ensure that (1) acceptable power peaking factors are maintained, (2) the minimum available shutdown margin is maintained, and (3) the potential consequences of a CEA ejection accident are limited to acceptable levels. However, long term operation at these insertion limits could have adverse effects on core power distribution during subsequent operation in an unrodded configuration. Since the LSSS setpoints and power distribution LCO's were generated using information based on a core burnup distribution achieved through essentially unrodded long term operation, the CEA insertion limit specification requires that CEA's be nearly fully withdrawn during normal operation. The amount of CEA insertion permitted by the insertion limit of Specification 2.10.2(7) will not have a significant effect upon the unrodded burn-up assumption but will still provide sufficient reactivity control.

Operation up to the Short Term Insertion Limit of Specification 2.10.2(7) is permitted since the degradation in peaking factors due to xenon redistribution effects is negligible for the allowed configurations and time intervals. The effects on the burnup distribution are also negligible. Short term operation in the region between the Short Term and transient limits requires an additional restriction on the rate at which the power may be increased so that the potentially adverse effects of xenon redistribution on peaking factors are annealed out.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.2 Reactivity Control Systems and Core Physics Parameters Limits (Continued)

CEA Drop Time

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the accident analyses. Measurement with $T_{cold} \geq 515°F$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

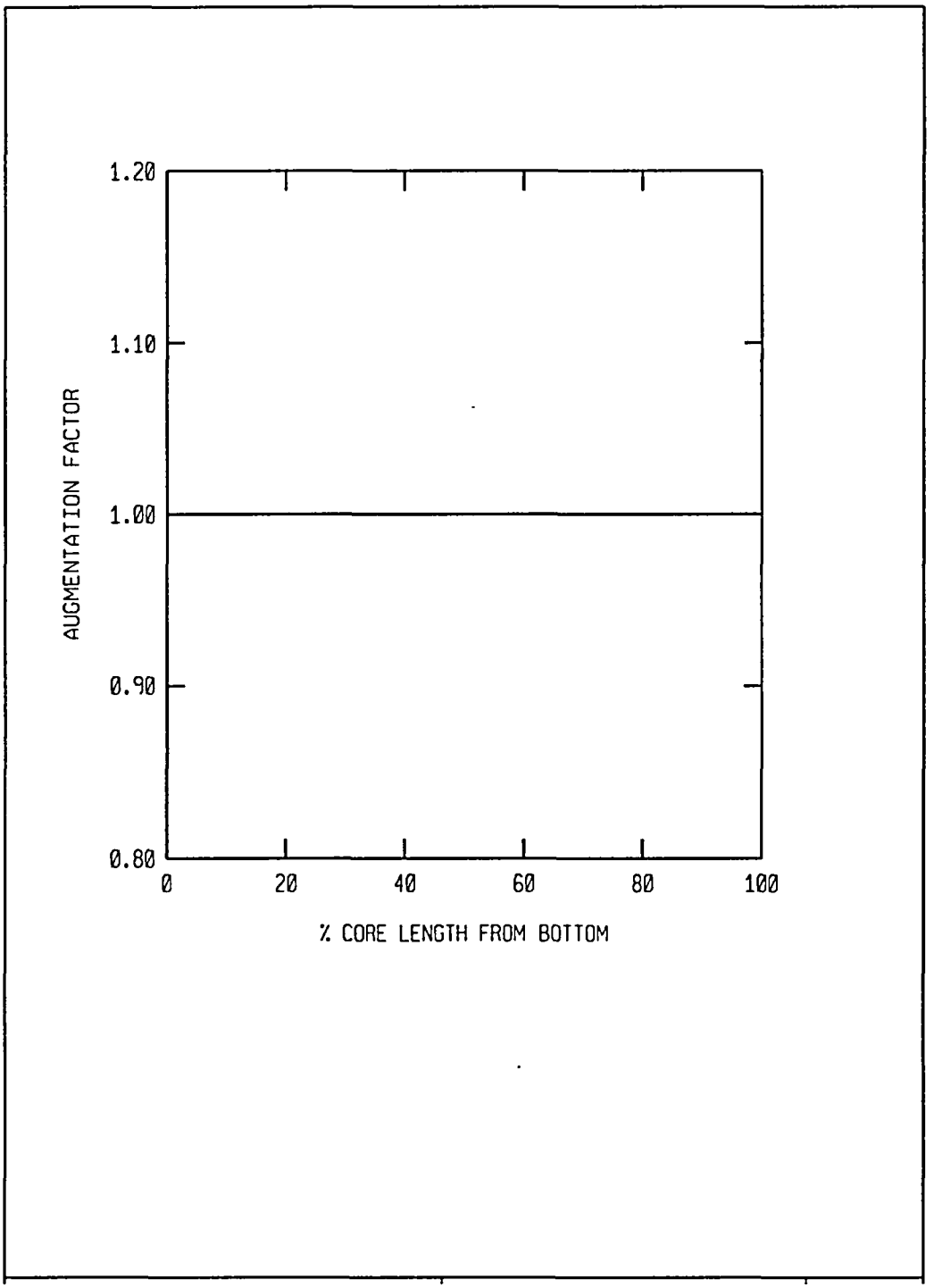
Test Exemptions

The CEA group insertion limits exemption permits individual CEA's to be positioned outside of their normal group heights and insertion limits during performance of such physics tests as those required to measure CEA worth and to determine the reactor stability index and damping factor under xenon oscillation conditions. The center CEA misalignment permits the center CEA to be misaligned during physics tests to determine the isothermal temperature coefficient and power coefficient.

The shutdown margin exemption provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEA's worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

The moderator temperature coefficient (MTC) exemption permits measurement of more positive MTC's at the beginning of cycle and confirmation of CEA bank insertion necessary to maintain the MTC within its limits before entering power operation conditions.

Figure 2-8

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|---|---|
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2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits

Applicability

Applies to power operation conditions.

Objective

To ensure that peak linear heat rates, DNB margins, and radial peaking factors are maintained within acceptable limits during power operation.

Specification

(1) Linear Heat Rate

The linear heat rate shall not exceed the limits of the Allowable Peak Linear Heat Rate vs. Burnup Figure provided in the COLR.

The linear heat rate shall be monitored by the incore detector system in accordance with specifications 2.10.4(1)(a) or 2.10.4(1)(b), or the Axial Shape Index, $Y_I$, shall be maintained within the limits of the Limiting Condition for Operation for Excore Monitoring of LHR Figure provided in the COLR.

(a) When the linear heat rate is continuously monitored by the incore detectors, and the linear heat rate is exceeding its limits as indicated by four or more valid coincident incore detector alarms, either:

(i) Restore the linear heat rate to within its limits within one hour, or

(ii) Be in at least hot standby within the next 6 hours.

(b) When the linear heat rate is continuously monitored by the incore detectors, and the plant computer incore detector alarms become inoperable, operation may be continued for seven days from the date of the last valid core power distribution without reducing power provided each of the following conditions is satisfied:

(i) A core power distribution was obtained utilizing incore detectors within 7 days prior to the incore detector alarm outage and the measured peak linear heat rate was no greater than 90% of the value allowed by (1) above.

(ii) The Axial Shape Index as measured by excore detectors remains within ±.05 of the value obtained at the time of the last measured incore power distribution.

(iii) Power is not increased nor has it been increased since the time of the last incore power distribution.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

(c) When the linear heat rate is continuously monitored by the excore detectors, withdraw the full length CEAs beyond the long term insertion limits of Specification 2.10.2(7) and maintain the Axial Shape Index, $Y_I$ within the limits of Limiting Condition for Operations for the Excore Monitoring of LHR Figure provided in the COLR. If the linear heat rate is exceeding its limits as determined by the Axial Shape Index, $Y_I$, being outside the limits of the Limiting Condition for Operation for Excore Monitoring of LHR Figure provided in the COLR:

(i) Restore the reactor power and Axial Shape Index, $Y_I$, to within the limits of the Limiting Condition for Operations for Excore Monitoring of LHR Figure provided in the COLR within 2 hours, or

(ii) Be in at least hot standby within the next 6 hours.

(d) When calibration of the ex-core detectors has not been accomplished within the previous 30 equivalent full power days, then:

(i) reduce the axial power distribution monitoring trip setpoints as shown in the Axial Power Distribution LSSS for 4 Pump Operation Figure provided in the COLR by 0.03 ASI units; and

(ii) reduce the axial power distribution monitoring Limiting Condition for Operations (LCO for Excore Monitoring of LHR and LCO for DNB Monitoring Figures provided in the COLR) by 0.03 ASI units.

When calibration of the ex-core detectors has not been accomplished within the previous 200 equivalent full power days, the power shall be limited to less than that corresponding to 75% of the peak linear heat rate permitted by Specification 2.10.4(1).

(2) Maximum Radial Peaking Factor ($F_R^T$)

The value of $F_R^T$ shall be within the limit provided in the COLR. $F_R^T$ is determined with all full length CEAs at or above the Long Term Steady State Insertion Limit.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

With $F_R^T \geq$ the limit provided in the COLR within 6 hours:

(a) Reduce power to bring power and $F_R^T$ within the limits of the $F_R^T$, and Core Power Limitations Figure provided in the COLR, withdraw the full length CEA's to or beyond the Long Term Steady State Insertion Limits of Specification 2.10.2(7), and fully withdraw the NTCEA's, or

(b) Be in at least hot standby.

(3) Deleted

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.10 Reactor Core (Continued)

2.10.4 Power Distribution Limits (Continued)

(4) Azimuthal Power Tilt ($T_q$)

When operating above 70% of rated power,

(a) The azimuthal power tilt ($T_q$) shall not exceed 0.10 whenever the CEAs are at or above the Long Term Insertion Limit and $F_R^T$ is being monitored with the incore detectors.

(b) The azimuthal power tilt ($T_q$) shall not exceed 0.03 whenever the provisions of 2.10.4(4)(a) do NOT allow $F_R^T$ to be monitored with the incore detectors. With the indicated azimuthal power tilt determined to be >0.03 but <0.10, correct the power tilt within two hours or determine within the next 6 hours and at least once per subsequent 8 hours, that the maximum radial peaking factor, $F_R^T$, is within the COLR limit, or reduce power to less than 70% of rated power within 8 hours of confirming $T_q$ >0.03.

(c) With the indicated power tilt determined to be ≥ 0.10, power operation may proceed up to 2 hours provided $F_R^T$ does not exceed the limit provided in the COLR, or be in at least hot standby within 6 hours. Subsequent operation for the purpose of measurement to identify the cause of the tilt is allowable provided the power level is restricted to 20% of the maximum allowable thermal power level.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

(5) DNBR Margin During Power Operation Above 15% of Rated Power

(a) The following limits on DNB-related parameters shall be maintained:

| | | |
|---|---|---|
| (i) | Cold Leg Temperature (Core Inlet Temperature) | as specified in the COLR |
| (ii) | Pressurizer Pressure | ≥ 2075 psia[1] |
| (iii) | Reactor Coolant Flow rate | ≥ 202,500 gpm indicated |
| (iv) | Axial Shape Index | as specified in the COLR |

(b) With any of the above parameters exceeding the limit, restore the parameter to within its limit within 2 hours or reduce power to less than 15% of rated power within the next 8 hours.

Basis

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the linear heat rate does not exceed its limit. The Excore Detector Monitoring System performs this function by continuously monitoring the axial shape index (ASI) with the operable quadrant symmetric excore neutron flux detectors. The axial shape index is maintained within the allowable limits of the Limiting Condition for Operation for Excore Monitoring of LHR Figure provided in the COLR. This ASI is adjusted by Specification 2.10.4(1)(c) for the allowed linear heat rate of the Allowable Peak Linear Heat Rate vs. Burnup Figure provided in the COLR and the $F_R^T$ and Core Power Limitations Figure provided in the COLR. In conjunction with the use of the excore monitoring system and in establishing the axial shape index limits, the following assumptions are made: (1) the CEA insertion limits of Specification 2.10.1(6) and long term insertion limits of Specification 2.10.1(7) are satisfied, and (2) the flux peaking augmentation factors are as shown in Figure 2-8.

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(1) Limit not applicable during either a thermal power ramp in excess of 5% of rated thermal power per minute or a thermal power step of greater than 10% of rated thermal power.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

The Incore Detector Monitoring system provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be continuously maintained within the allowable limits of the Allowable Peak Linear Heat Rate vs. Burnup Figure provided in the COLR. The setpoints for these alarms include allowances, set in the conservative directions. If the plant computer fails, the incore detector alarms become inoperable. The provisions of Section 2.10.4(1)(b) are intended to address this situation and assure safe operation of the reactor for up to 7 days.

Calibration of the ex-core detector input to the APD calculator is required to eliminate ASI uncertainties due to instrument drift and axially nonuniform detector exposure. If the recalibration is not performed in the period specified, the prescribed steps will assure safe operation of the reactor.

Total Integrated Radial Peaking Factor ($F_R^T$) and Azimuthal Power Tilt ($T_q$)

The limitation of $T_q$ is provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCO's and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations of $F_R^T$ and $T_q$ are provided to ensure that the assumptions used in the analysis establishing the DNB Margin LCO and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If $F_R^T$ or $T_q$ exceed their basic limitations, operation may continue under the additional restrictions imposed by the action statements since these additional restrictions provide adequate assurance that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCO's and LSSS setpoints remain valid. An azimuthal power tilt >0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The value of $T_q$ that must be used in the equation $F_R^T = F_R(1 + T_q)$ is the measured tilt.

The surveillance requirements for verifying that $F_R^T$ and $T_q$ are within their limits provide assurance that the actual values of $F_R^T$ and $T_q$ do not exceed the assumed values. Verifying $F_R^T$ after each fuel loading prior to exceeding 70% of rated power provides additional assurance that the core was properly loaded.

DNBR Margin During Power Operation Above 15% of Rated Power

The selection of limiting safety system settings and reactor operating limits is such that:

1. No specified acceptable fuel design limits will be exceeded as a result of the design basis anticipated operational occurrences, and

2. The consequences of the design basis postulated accidents will be no more severe than the predicted acceptable consequences of the accident analysis in Section 14.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

In order for these objectives to be met, the reactor must be operated consistent with the operating limits specified for margin to DNB.

The parameter limits given in (5) and the $F_R^T$, and Core Power Limitations Figure provided in the COLR along with the parameter limits on quadrant tilt and control element assembly position (Power Dependent Insertion Limit Figure provided in the COLR) provide a high degree of assurance that the DNB overpower margin will be maintained during steady state operation.

The actions specified assure that the reactor is brought to a safe condition.

The Reactor Coolant System flow rate of 202,500 gallons per minute is the indicated value. It does not include instrumentation uncertainties.

The calorimetric methodology shall be used to measure the Reactor Coolant System flow rate.

AZIMUTHAL POWER TILT

Azimuthal Power Tilt is measured using symmetric in-core or ex-core detectors by assuming that the ratio of the power at any core location in the presence of a tilt to the untitled power at that location is of the form:

$$P_{tilt}(r,\theta)/P_{avg}(r,\theta) - 1 = T_q \bullet g(r) \bullet \cos(\theta - \theta_o)$$

where

| | |
|---|---|
| $P_{tilt}(r,\theta)$ | is the tilted power at radius r and azimuthal angle θ |
| $P_{avg}(r,\theta)$ | is the average or untilted power at that location |
| $T_q$ | is the azimuthal tilt magnitude |
| $g(r)$ | is the radial normalizing factor, normalized to a maximum value of unity |
| $\theta$ | is the azimuthal core location |
| $\theta_o$ | is the azimuthal core location of maximum tilt |

$T_q$ represents the maximum fractional increase in power that can occur anywhere in the core because of tilt. It is the appropriate measured value of tilt to be used when ensuring the validity of the azimuthal tilt assumed by ABB-CE in establishing safety limits.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.12 Control Room Ventilation System

### 2.12.1 Control Room Air Filtration System - Operating

Applicability

Applies to the operational status of the control room air filtration system when the reactor coolant temperature $T_{cold} \geq 210°F$.

Objective

To assure operability of equipment required to filter control room air following a Design Basis Accident.

Specification

Two control room air filtration trains shall be OPERABLE.

----------Note----------
The control room envelope (CRE) boundary may be opened intermittently under administrative control.
---

Required Actions

(1) With one control room air filtration train inoperable for reasons other than (2), restore the inoperable train to OPERABLE status within 7 days.

(2) With one or more control room air filtration trains inoperable due to inoperable CRE boundary:
   a. initiate mitigating actions immediately, AND
   b. verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits, within 24 hours, AND
   c. restore CRE boundary to OPERABLE status within 90 days.

   ----------Note----------
   Not applicable when second control room air filtration train intentionally made inoperable.
   ---

(3) With two control room air filtration trains inoperable for reasons other than (2),
   a. initiate mitigating actions immediately, AND
   b. verify LCO 2.1.3, "Reactor Coolant Activity," is met within 1 hour, AND
   c. restore at least one control room filtration train to OPERABLE status within 24 hours.

(4) With the required actions of (1), (2), or (3) not met, be in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 36 hours.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.12 Control Room Ventilation System

### 2.12.2 Control Room Air Conditioning System

Applicability

Applies to the operational status of the control room air conditioning system when the reactor coolant temperature $T_{cold} \geq 210°F$.

Objective

To assure operability of equipment required to maintain air temperature within the control room following a Design Basis Accident.

Specification

Two control room air conditioning trains shall be OPERABLE.

Required Actions

(1) With one control room air conditioning train inoperable, restore the inoperable train to OPERABLE status within 30 days.

- - - - - - - - - - - - - - - - - - - - - - - - -NOTE - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
Not applicable when second control room air conditioning train intentionally made inoperable.
- - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -

(2) With two control room air conditioning trains inoperable, restore at least one train to OPERABLE status within 24 hours.

(3) With the required actions of (1) or (2) not met, be in HOT SHUTDOWN within 6 hours, and COLD SHUTDOWN within the following 36 hours.

## 2.0 **LIMITING CONDITIONS FOR OPERATION**

### 2.12 Control Room Ventilation System

Bases

### 2.12 Control Room Ventilation System

The control room ventilation system (CRVS) provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. The CRVS contains two independent, redundant control room air filtration trains that filter the air in the control room envelope (CRE), two independent, redundant air conditioning units that circulate and cool the air in the CRE, and a CRE boundary that limits the inleakage of unfiltered air.

The CRE is the area within the confines of the CRE boundary that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

Actuation of the CRVS places the system into either of two separate states of the emergency mode of operation, depending on the initiation signal. Actuation of the system to the emergency radiation state of the emergency mode of operation closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the air within the CRE through the redundant trains of HEPA and charcoal filters. The emergency radiation state also initiates filtered ventilation of the outside air supply to the CRE.

The actions taken in the toxic gas isolation state are similar, except that the signal switches the CRVS to an isolation mode, minimizing outside air entering the CRE through the CRE boundary. Toxic gas is monitored at the outside air intake duct. Actuation of the system to toxic gas protection mode trips CRVS fans and isolates the outside air dampers. The CRVS is then placed in recirculation mode. In recirculation mode, the filter trains are bypassed.

Fire and smoke detection is provided at the outlet of the recirculation fans to protect against smoke developed from sources in the outside air stream or from sources inside the control room. As in toxic gas protection mode, CRVS fans are tripped and the outside air dampers are isolated.

## 2.0 **LIMITING CONDITIONS FOR OPERATION**

### 2.12 Control Room Ventilation System

Bases (Continued)

2.12 Control Room Ventilation System (Continued)

The radiation monitoring system provides an airborne radiation monitor (RM-065), which starts after a ventilation isolation actuation signal (VIAS) to verify control room habitability following a design basis accident. The air entering the CRE is continuously monitored by toxic gas detectors. One detector output above the setpoint will cause actuation of the toxic gas isolation state. The actions of the toxic gas isolation state are more restrictive, and will override the actions of the emergency radiation state.

The CRVS provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release (Ref. 3). The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels (Ref. 4).

The worst case single active failure of a component of the CRVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The CRVS satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

2.12.1 Control Room Air Filtration System - Operating

Each control room air filtration system (CRAFS) train contains a heater and demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, doors, barriers, and instrumentation also form part of the system, as well as demisters that remove water droplets from the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provides back-up in case of failure of the main HEPA filter bank.

The CRAFS is an emergency system, part of which may also operate during normal unit operations in the standby mode of operation. Upon receipt of a VIAS, normal air supply to the CRE is diverted to the filter trains, and the stream of ventilation air is recirculated through the filter trains of the system. The demisters remove any entrained water droplets present to prevent excessive loading of the HEPA filters and charcoal adsorbers. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and adsorbers. Both the demister and heater are important to the effectiveness of the charcoal adsorbers.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.12 Control Room Ventilation System

Bases (Continued)

2.12.1 Control Room Air Filtration System - Operating (Continued)

Outside air is filtered, and then added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air though the CRE boundary from all the surrounding areas adjacent to the CRE boundary.

A single CRAFS train operating at a flow rate of ≤ 1000 cfm will pressurize the CRE to about 0.125 inches water gauge relative to external areas adjacent to the CRE boundary, and provides an air exchange rate in excess of 60% per hour. The CRAFS operation in maintaining the CRE habitable is discussed in USAR, Section 9.10 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. However, the recirculation duct does not require redundant dampers to meet single failure proof criteria. Damper PCV-6682 meets the acceptance criteria for the damper repair option described in Standard Review Plan 6.4, Appendix A. A release of radioactivity requires PCV-6682 to open, should PCV-6682 fail to open, it can be repaired or repositioned open before control room doses exceed the allowable limits of General Design Criterion 19. The CRAFS is designed in accordance with Seismic Category 1 requirements.

The CRAFS is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem total effective dose equivalent (TEDE). The CRAFS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access.

The CRAFS provides airborne radiological protection for the CRE occupants as demonstrated by the CRE occupant dose analyses for the most limiting design basis accident fission product release presented in the USAR, Section 14.15 (Ref. 2).

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.12 Control Room Ventilation System

Bases (Continued)

2.12.1 Control Room Air Filtration System - Operating (Continued)

Two independent and redundant trains of the CRAFS are required to be OPERABLE to ensure that at least one is available if a single active failure disables the other train. Total system failure, such as from a loss of both filtration trains or from an inoperable CRE boundary, could result in exceeding a dose of 5 rem TEDE to the CRE occupants in the event of a large radioactive release.

Each CRAFS train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. A CRAFS train is considered OPERABLE when the associated:

a. Fan is OPERABLE,

b. HEPA filters and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function, and

c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In order for the CRAFS trains to be considered OPERABLE, the CRE boundary must be maintained such that CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

APPLICABILITY

With the reactor coolant temperature $T_{cold} \geq 210°F$, the CRAFS must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.12 Control Room Ventilation System

Bases (Continued)

2.12.1 Control Room Air Filtration System - Operating (Continued)

ACTIONS

(1)

With one CRAFS train inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CRAFS train is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a failure in the OPERABLE CRAFS train could result in loss of CRAFS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining train to provide the required capability.

(2)a, (2)b, and (2)c

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem TEDE), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.12 Control Room Ventilation System

Bases (Continued)

2.12.1 Control Room Air Filtration System - Operating (Continued)

(3)

With reactor coolant temperature $T_{cold} \geq 210°F$, if the inoperable CRAFS or CRE boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least HOT SHUTDOWN within 6 hours, and in COLD SHUTDOWN within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(4)

If both CRAFS trains are inoperable with reactor coolant temperature $T_{cold} \geq 210°F$ for reasons other than an inoperable CRE boundary (i.e., Condition 2), the CRAFS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 2.0.1 must be entered immediately.

2.12.2 Control Room Air Conditioning System

The control room air conditioning system is required to ensure the control room temperature will not exceed equipment OPERABILITY requirements. The reactor protective system panels and the engineered safety features panels were designed for, and the instrumentation was tested at, 120°F. The temperature inside the control cabinets is at most 15°F warmer than the temperature of the control room due to heat produced by the electronic circuitry. Therefore, the temperature of the control room will not affect OPERABILITY of the control cabinets as long as it doesn't exceed 105°F.

During non-emergency operation, the control room temperature may be maintained by using Component Cooling Water (CCW). During design basis accident conditions, the CCW isolation valves to air conditioning units (VA-46A and VA-46B) are automatically closed on a VIAS. This prevents CCW that has been heated by components following a design basis accident from adding heat to the control room. When VIAS is in override, closing these valves maintains the OPERABILITY of the associated air conditioning unit.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.12 Control Room Ventilation System

Bases (Continued)

2.12.2 Control Room Air Conditioning System (Continued)

With the reactor coolant temperature $T_{cold} \geq 210°F$, two trains of the control room air conditioning system are required to be OPERABLE. If one train is inoperable it shall be restored to OPERABLE status within 30 days. In this condition the remaining train is adequate to maintain the control room temperature. With both trains inoperable, the control room air conditioning system may not be capable of performing its intended function and LCO 2.0.1 must be entered immediately.

References

(1) USAR Section 9.10
(2) USAR Section 14.15
(3) USAR Section 14.23
(4) Engineering Analysis (EA)-FC-01-013, “Effects of Secondary Environment Resulting from a Fire Event”

## 2.0 **LIMITING CONDITIONS FOR OPERATION**

### 2.13 Limiting Safety System Settings, Reactor Protective System

Applicability

This specification applies to RPS Limiting Safety System settings and bypasses for instrument channels.

Objective

To provide for automatic protection action in the event that the principal process variables approach a safety limit.

Specification

The reactor protective system trip setting limits and the permissible bypasses for the instrument channels shall be within the Limiting Safety System Setting as stated in Table 2-11.

Basis

The reactor protective system consists of four instrument channels to monitor selected plant conditions which will cause a reactor trip if any of these conditions deviate from a preselected operating range to the degree that a safety limit may be reached.

(1) High Power Level - A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding resulting from some reactivity excursions too rapid to be detected by pressure and temperature measurements (in addition, thermal signals are provided to the high power level trip unit as a backup to the neutron flux signal).

During normal plant operation, with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 109.0% of indicated full power. Adding to this the possible variation in trip point due to calibration and measurement errors, the maximum actual steady-state power at which a trip would be actuated is 112%, which was used for the purpose of safety analysis.[1]

During reactor operation at power levels between 19.1% and 100% of rated power, the Variable High Power Trip (VHPT) will initiate a reactor trip in the event of a reactivity excursion that increases reactor power by 10% or less of rated power. The high power trip setpoint can be set no more than 10% of rated power above the indicated plant power. Operator action is required to increase the set point as plant power is increased. The setpoint is automatically decreased as power decreases.

## 2.0 **LIMITING CONDITIONS FOR OPERATION**

2.13 Limiting Safety System Settings, Reactor Protective System (continued)

During reactor operation at power levels below 19.1% rated power, a reactor trip will occur in the event of a reactivity excursion that results in a power increase up to the lower fixed set point of the VHPT circuit of 19.1% of rated power.[3] During normal power increases below 19.1% reactor trip would be initiated at 19.1% of rated power unless the set point is manually adjusted.

(2) Low Reactor Coolant Flow - A reactor trip is provided to protect the core against DNB should the coolant flow suddenly decrease significantly.

Flow in each of the four coolant loops is determined from a measurement of pressure drop from inlet to outlet of the steam generators. The total flow through the reactor core is measured by summing the loop pressure drops across the steam generators and correlating this pressure sum with the pump calibration flow curves[2].

During four-pump operation, the low flow trip setting of 95% insures that the reactor cannot operate when the flow rate is less than 93% (5) of the assumed minimum indicated value. Refer to Technical Specification 2.10.4 (5)(a)(iii) for the minimum indicated value of reactor coolant flow.

(3) High Pressurizer Pressure - A reactor trip for high pressurizer pressure is provided in conjunction with the reactor and steam system safety valves to prevent reactor coolant system overpressure (Specification 2.1.6). In the event of loss of load without reactor trip, the temperature and pressure of the reactor coolant system would increase due to the reduction in the heat removed from the coolant via the steam generators. The power-operated relief valves are set to operate concurrently with the high pressurizer pressure reactor trip. This setting is below the nominal safety valve setting (2500 psia) to avoid unnecessary operation of the safety valves. This setting is consistent with the trip point assumed in the accident analysis.[1]

(4) Thermal Margin/Low Pressure Trip - The thermal margin/low pressure trip is provided to prevent operation when the DNBR is less than the minimum DNBR limit, including allowance for measurement error. The thermal and hydraulic limits shown in the Thermal Margin/Low Pressure 4-Pump Operation Figure, contained in the COLR, define the limiting values of reactor coolant pressure, reactor inlet temperature, axial shape index, and reactor power level which ensure that the thermal criteria[8] are not exceeded. The low set point of 1750 psia trips the reactor in the unlikely event of a loss-of-coolant accident. The thermal margin/low pressure trip set points shall be set according to the equation given in the COLR for the Thermal Margin/Low Pressure Limit.

## 2.0 **LIMITING CONDITIONS FOR OPERATION**

2.13 Limiting Safety System Settings, Reactor Protective System (continued)

(5) Low Steam Generator Water Level - The low steam generator water level reactor trip protects against the loss of feedwater flow accidents and assures that the design pressure of the reactor coolant system will not be exceeded. The specified set points assure that there will be sufficient water inventory in the steam generator at the time of trip to provide a 12-minute margin before the auxiliary feedwater is required.[9]

The setting listed in Table 2-11 assures that the heat transfer surface (tubes) are covered with water when the reactor is critical.

(6) Low Steam Generator Pressure - A reactor trip on low steam generator secondary pressure is provided to protect against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 500 psia is sufficiently below the full-load operating point of 770 psia so as not to interfere with normal operation but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of ±22 psia in the accident analysis.[1]

(7) Containment High Pressure - A reactor trip on containment high pressure is provided to assure that the reactor is shut down simultaneously with the initiation of the safety injection system. The setting of this trip is identical to that of the containment high pressure signal which indicates safety injection system operation.

(8) Axial Power Distribution - The axial power trip is provided to ensure that excessive axial peaking will not cause fuel damage. The Axial Shape Index is determined from the axially split excore detectors. The set point functions, shown in the COLR ensure that neither a DNBR of less than the minimum DNBR limit nor a fuel centerline temperature greater than the safety limit corresponding to Fuel Centerline Melt (FCM), as determined each fuel cycle and contained in the COLR, will exist as a consequence of axial power maldistributions. Allowances have been made for instrumentation inaccuracies and uncertainties associated with the excore symmetric offset - incore axial peaking relationship. A variance of 5% between Δ T-Power and NI-Power is permitted due to the significant margins to local power density limits before calibration of NI-Power is performed at 30% power.

## 2.0 **LIMITING CONDITIONS FOR OPERATION**

2.13 Limiting Safety System Settings, Reactor Protective System (continued)

(9) Steam Generator Differential Pressure - The Asymmetric Steam Generator Transient Protection Trip Function (ASGTPTF) utilizes a trip on steam generator differential pressure to ensure that neither a DNBR of less than the minimum DNBR limit nor a fuel centerline temperature greater than the safety limit corresponding to FCM, as determined each fuel cycle and contained in the COLR, occurs as a result of the loss of load to one steam generator.

(10) Physics Testing at Low Power - During physics testing at power levels less than $10^{-1}$% of rated power, the tests may require that the reactor be critical. For these tests only the low reactor coolant flow and thermal margin/low pressure trips may be bypassed below $10^{-1}$% of rated power. Written test procedures which are approved by the Plant Review Committee will be in effect during these tests. At reactor power levels less than $10^{-1}$% of rated power the low reactor coolant flow and the thermal margin/low pressure trips are not required to prevent fuel element thermal limits being exceeded. Both of these trips are bypassed using the same bypass switch. The low steam generator pressure trip is not required because the low steam generator pressure will not allow a severe reactor cooldown if a steam line break were to occur during the tests.

References

(1) USAR, Section 14.1
(2) USAR, Section 7.2.3.3
(3) USAR, Section 7.2.3.2
(4) USAR, Section 3.6.6
(5) USAR, Section 14.6.
(6) USAR, Section 14.7
(7) USAR, Section 7.2.3.1
(8) USAR, Section 3.6
(9) USAR, Section 14.10

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.13 Limiting Safety System Settings, Reactor Protective System (continued)

### TABLE 2-11

### RPS LIMITING SAFETY SYSTEM SETTINGS

| No. | Reactor Trip | Trip Setpoints |
|---|---|---|
| 1 | High Power Level (A)<br>4-Pump Operation | ≤ 109.0% of Rated Power |
| 2 | Low Reactor Coolant Flow (B)(F)<br>4-Pump Operation | ≥ 95% of 4 Pump Flow |
| 3 | Low Steam Generator Water Level | ≥ 31.2% of Scale |
| 4 | Low Steam Generator Pressure (C) | ≥ 500 psia |
| 5 | High Pressurizer Pressure | ≤ 2400 psia |
| 6 | Thermal Margin/Low Pressure (B)(F) | 1750 psia to 2400 psia (depending on the reactor coolant temperature as shown in the Thermal Margin/Low Pressure 4 Pump Operation Figure provided in the COLR) |
| 7 | High Containment Pressure (D) | ≤ 5 psig |
| 8 | Axial Power Distribution (E) | (as shown in the Axial Power Distribution LSSS for 4 Pump Operation Figure provided in the COLR) |
| 9 | Steam Generator Differential Pressure | ≤ 135 psid |

A Setpoint cannot be set greater than 10% above measured power whenever reactor power is greater than 10% of rated power.

B May be bypassed below $10^{-4}$% power.

C May be bypassed below 600 psia.

D Bypass allowed for containment leak test.

E Inhibited below 15% power.

F For physics testing at power levels less than $10^{-1}$% of rated power, the low reactor coolant flow and thermal margin/low pressure trips may be bypassed until their reset points are exceeded if automatic bypass removal of $10^{-1}$% of rated power is operable.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.14 Engineered Safety Features System Initiation Instrumentation Settings

Applicability

Applies to the engineered safety features system initiation instrumentation settings.

Objective

To provide for automatic initiation of the engineered safety features in the event that principal process variable limits are exceeded.

Specifications

The engineered safety features system initiation instrumentation setting limits shall be as stated in Table 2-1.

Basis

(1) High Containment Pressure

The basis for the 5 psig setpoint for the high pressure signal is to establish a setting which would be exceeded quickly in the event of a DBA, cover a spectrum of break sizes, and yet be far enough above normal operation maximum internal pressure to prevent spurious initiation.

High containment pressure initiates the steam generator isolation signal which will close the main steam isolation and bypass valves and the main feedwater isolation and bypass valves.

(2) Pressurizer Low Pressure

The pressurizer low pressure safety injection signal is a diverse signal to the high containment pressure safety injection signal. The 1600 psia setting includes an uncertainty of ± 22 psia and is the setting used in the safety analysis.[1]

(3) Containment High Radiation (Air Monitoring)

The containment radiation high signal can be initiated by a containment atmosphere gaseous radiation monitor or an Auxiliary Building Exhaust Stack gaseous radiation monitor[2].

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.14 Engineered Safety Features System Initiation Instrumentation Settings (Continued)

(3) Containment High Radiation (Air Monitoring) (Continued)

Effluent radiation monitor isolation function setpoints will be calculated in accordance with the ODCM. Process radiation monitor setpoints will be calculated in accordance with the applicable Chemistry Manual calibration procedure.

(4) Low Steam Generator Pressure

A signal is provided upon sensing a low pressure in a steam generator to close the main steam isolation valves in order to minimize the temperature reduction in the reactor coolant system with resultant loss of water level and possible addition of reactivity. The setting of 500 psia includes a ± 22 psi uncertainty and was the setting used in the safety analysis.[3]

Closure of the MSIVs (and the bypass valves, along with main feedwater isolation and bypass valves) is accomplished by the steam generator isolation signal which is a logical combination of low steam generator pressure or high containment pressure.

Initiation of containment spray is a logical combination of containment high pressure, pressurizer low/low pressure, and steam generator low pressure.

As part of the AFW actuation logic, a separate signal is provided to terminate flow to a steam generator upon sensing a low pressure in that steam generator if the other steam generator pressure is greater than the pressure setting. This is done to minimize the temperature reduction in the reactor coolant system in the event of a main steam-line break. The setting of 466.7 psia includes a +31.7 psi uncertainty; therefore, a setting of 435 psia was used in the safety analysis.

(5) SIRW Tank Low Level

Level switches are provided on the SIRW tank to actuate the valves in the safety injection pump suction lines in such a manner so as to switch the water supply from the SIRW tank to the containment sump for a recirculation mode of operation after a period of at least 45 minutes in response to a LOCA. The switch-over point of 16 inches above tank bottom is set to prevent the pumps from running dry during the time required to stroke the valves and to hold in reserve approximately 28,000 gallons of water of at least the refueling boron concentration. The USAR loss of coolant accident analysis[4] assumed the recirculation started when the minimum usable volume of 250,000 gallons had been pumped from the tank. Technical Specification 2.3(1) requires that the SIRW tank contain a minimum of 283,000 gallons of usable water. This additional volume over that assumed in the USAR analysis provides sufficient margin to account for the instrument uncertainty.

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.14 Engineered Safety Features System Initiation Instrumentation Settings (Continued)

(6) Low Steam Generator Water Level

As part of the AFW actuation logic, a signal is provided to initiate AFW flow to one or two steam generators upon sensing a low water level in the steam generator(s) if the absolute steam generator pressure criteria are satisfied. This function ensures adequate steam generator water level is maintained in the event of a failure to deliver main feedwater to either steam generator. The setting of 28.2% of wide range tap span includes a +13.2% uncertainty; therefore, a setting of 15% of wide range tap span was used in the safety analysis.

(7) High Steam Generator Delta Pressure

As part of the AFW logic, a high steam generator differential pressure signal is generated to provide AFW to the higher pressure steam generator with a concurrent low level signal if both steam generator pressures are less than 466.7 psia. If the differential pressure between steam generators is less than the setting, neither steam generator is supplied with AFW in the presence of a low level signal. The setting of 119.7 psid includes a -15.3 psi uncertainty; therefore, a setting of 135 psid was used in the AFW safety analysis.

References

(1) USAR, Section 14.1.3

(2) USAR, Section 7.3.2.6

(3) USAR, Section 14.12

(4) USAR, Section 14.15

(5) USAR, Section 7.4.6

(6) USAR, Section 7.5.2.5

(7) USAR, Section 14.4.1

TABLE 2-1

**ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENT SETTING LIMITS**

| | Functional Unit | Channel | Setting Limit |
|---|---|---|---|
| 1. | High Containment Pressure | a. Safety Injection<br>b. Containment Spray (3)<br>c. Containment Isolation<br>d. Containment Air Cooler DBA Mode<br>e. Steam Generator Isolation | ≤ 5 psig |
| 2. | Pressurizer Low/Low Pressure | a. Safety Injection<br>b. Containment Spray (3)<br>c. Containment Isolation<br>d. Containment Air Cooler DBA Mode | ≥ 1600 psia (1) |
| 3. | Containment High Radiation | Containment Ventilation Isolation | In accordance with the applicable Chemistry Manual calibration procedure |
| 4. | Low Steam Generator Pressure | a. Steam Line Isolation<br>b. Auxiliary Feedwater Actuation<br>c. Containment Spray (3) | ≥ 500 psia (2)<br>≥ 466.7 psia<br>≥ 500 psia(2) |
| 5. | SIRW Low Level Switches | Recirculation Actuation | 16 inches +0, -2 in. above tank bottom |

TABLE 2-1 (continued)

**ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENT SETTING LIMITS**

| Functional Unit | Channel | Setting Limit | |
|---|---|---|---|
| 6. 4.16 KV Emergency Bus Low Voltage | a. Loss of Voltage | (2995.2 + 104, -20.8) volts<br>≤ 5.9 seconds[4] | Trip |
| | b. Degraded Voltage | | |
| | i) Bus 1A3 Side | ≥ 3988.8 volts<br>(4.8 ± .5) seconds | Trip |
| | ii) Bus 1A4 Side | ≥ 3990.6 volts<br>(4.8 ± .5) seconds | Trip |
| 7. Low Steam Generator Water Level | Auxiliary Feedwater Actuation | ≥ 28.2% of wide range tap span | |
| 8. High Steam Generator Delta Pressure | Auxiliary Feedwater Actuation | ≤ 119.7 psid | |

(1) May be bypassed below 1700 psia and is automatically reinstated prior to exceeding 1700 psia.
(2) May be bypassed below 600 psia and is automatically reinstated prior to exceeding 600 psia.
(3) Simultaneous containment high pressure, pressurizer low/low pressure, and steam generator low pressure.
(4) Applicable for bus voltage ≤ 2995.2 - 20.8 volts only. (For voltage ≥ (2995.2 - 20.8) volts, time delay shall be > 5.9 seconds).

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.15.1 Instrumentation and Control Systems

Applicability

Applies to plant instrumentation systems.

Objective

To delineate the conditions of the plant instrumentation and control systems necessary to assure reactor safety.

Specifications

The operability, permissible bypass, and Test Maintenance and Inoperable bypass specifications of the plant instrument and control systems shall be in accordance with Tables 2-2 through 2-5.

(1) In the event the number of channels of a particular system in service falls one below the total number of installed channels, the inoperable channel shall be placed in either the bypassed or tripped condition within one hour if the channel is equipped with a bypass switch, and eight hours if jumpers or blocks must be installed in the control circuitry. The inoperable channel may be bypassed for up to 48 hours from time of discovering loss of operability; however, if the inoperability is determined to be the result of malfunctioning RTDs or nuclear detectors supplying signals to the high power level, thermal margin/low pressurizer pressure, and axial power distribution channels, these channels may be bypassed for up to 7 days from time of discovering loss of operability. If the inoperable channel is not restored to OPERABLE status after the allowable time for bypass, it shall be placed in the tripped position or, in the case of malfunctioning RTDs or linear power nuclear detectors, the reactor shall be placed in hot shutdown within 12 hours. If active maintenance and/or surveillance testing is being performed to return a channel to active service or to establish operability, the channel may be bypassed during the period of active maintenance and/or surveillance testing. This specification applies to the high rate trip-wide range log channel when the plant is at or above $10^{-4}$% power and is operating below 15% of rated power.

(2) In the event the number of channels of a particular system in service falls to the limits given in the column entitled "Minimum Operable Channels," one of the inoperable channels must be placed in the tripped position or low level actuation permissive position for the auxiliary feedwater system within one hour, if the channel is equipped with a bypass switch, and within eight hours if jumpers or blocks are required; however, if minimum operable channel conditions for SIRW tank low signal are reached, both inoperable channels must be placed in the bypassed condition within eight hours from time of discovery of loss of operability. If at least one inoperable channel has not been restored to OPERABLE status after 48 hours from time of discovering loss of operability, the reactor shall be placed in a hot shutdown condition within the following 12 hours; however, operation can continue without containment ventilation isolation signals available if the containment ventilation isolation valves are closed.



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## 2.0 LIMITING CONDITIONS FOR OPERATION

2.15.1 Instrumentation and Control Systems (Continued)

If after 24 hours from time of initiating a hot shutdown procedure at least one inoperable engineered safety features or isolation functions channel has not been restored to OPERABLE status, the reactor shall be placed in a cold shutdown condition within the following 24 hours. This specification applied to the high rate trip-wide range log channel when the plant is at or above $10^{-4}$% power and is operating below 15% of rated power.

(3) In the event the number of channels on a particular engineered safety features (ESF) or isolation logic subsystem in service falls below the limits given in the columns entitled "Minimum Operable Channels" or "Minimum Degree of Redundancy," except as conditioned by the column entitled "Permissible Bypass Conditions," sufficient channels shall be restored to OPERABLE status within 48 hours so as to meet the minimum limits or the reactor shall be placed in a hot shutdown condition within the following 12 hours; however, operation can continue without containment ventilation isolation signals available if the ventilation isolation valves are closed. If after 24 hours from time of initiating a hot shutdown procedure sufficient channels have not been restored to OPERABLE status, the reactor shall be placed in a cold shutdown condition within the following 24 hours.

(4) In the event the number of channels of those particular systems in service not described in (3) above falls below the limits given in the columns entitled "Minimum Operable Channels" or "Minimum Degree of Redundancy," except as conditioned by the column entitled "Permissible Bypass Conditions," the reactor shall be placed in a hot shutdown condition within 12 hours. If minimum conditions for engineered safety features or isolation functions are not met within 24 hours from time of discovering loss of operability, the reactor shall be placed in a cold shutdown condition within the following 24 hours. If the number of OPERABLE high rate trip-wide range log channels falls below that given in the column entitled "Minimum Operable Channels" in Table 2-2 and the reactor is at or above $10^{-4}$% power and at or below 15% of rated power, reactor critical operation shall be discontinued and the plant placed in an operational mode allowing repair of the inoperable channels before startup or reactor critical operation may proceed.

If during power operation, the rod block function of the secondary CEA position indication system and rod block circuit are inoperable for more than 24 hours, or the plant computer PDIL alarm, CEA group deviation alarm and the CEA sequencing function are inoperable for more than 48 hours, the CEAs shall be withdrawn and maintained at fully withdrawn and the control rod drive system mode switch shall be maintained in the off position except when manual motion of CEA Group 4 is required to control axial power distribution.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.15 Instrumentation and Control Systems

### 2.15.2 Reactor Protective System (RPS) Logic and Trip Initiation

Applicability

Applies to the operational status of RPS Logic and Trip Initiation channels in MODES 1 and 2; and,

When reactor coolant temperature ($T_{cold}$) is greater than 210°F or MODE 4 with more than one CEA capable of being withdrawn and RCS boron concentration less than REFUELING BORON CONCENTRATION.

Objective

To delineate the conditions of the plant instrumentation and control systems necessary to assure reactor safety.

Specification

Six channels of RPS Logic matrices, four channels of RPS Trip Initiation Logic and two channels of RPS Manual Trip shall be OPERABLE.

Required Actions

(1) With one RPS Logic Matrix channel inoperable, restore the inoperable channel to OPERABLE status within 48 hours.

(2) With one RPS Trip Initiation Logic channel inoperable, de-energize the affected clutch power supply within one hour.

(3) With one RPS Manual Trip channel inoperable, restore the inoperable channel to OPERABLE status prior to entering MODE 2 from MODE 3.

(4) With two RPS Trip Initiation Logic channels affecting the same trip leg inoperable, de-energize the affected clutch power supplies immediately.

(5) With the required actions of (1), (2), or (4) not met, or
with two RPS Manual Trip channels inoperable, or
with two or more RPS Logic Matrices inoperable, or
with two or more RPS Trip Initiation Logic channels inoperable for reasons other than (4):

a. be in HOT SHUTDOWN and verify no more than one CEA is capable of being withdrawn within 6 hours,

OR

b. verify reactor coolant boron concentration is at REFUELING BORON CONCENTRATION within 6 hours.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.15 Instrumentation and Control Systems

### 2.15.3 Alternate Shutdown and Auxiliary Feedwater Panel

Applicability

Applies to the operational status of Alternate Shutdown and Auxiliary Feedwater Panel Functions in MODES 1 and 2.

Objective

To delineate the conditions of the plant instrumentation and control systems necessary to assure reactor safety.

Specification

The Alternate Shutdown and Auxiliary Feedwater Panel Functions/Instrumentation or Control Parameters in Table 2-6 shall be OPERABLE.

Required Actions

(1) With the number of OPERABLE channels or control circuits less than the required number of channels, restore the required number of channels to OPERABLE within seven (7) days.

(2) With the required actions of (1) not met, be in HOT SHUTDOWN within 12 hours.

## 2.0 LIMITING CONDITIONS FOR OPERATION
### 2.15 Instrumentation and Control Systems (Continued)

Basis

During plant operation, the complete instrumentation systems will normally be in service. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor protective system (RPS) and engineered safety features (ESF) system when one or more of the channels are out of service. Reactor safety is provided by RPS, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continued operation with certain instrumentation channels out of service since provisions were made for this in the plant design.

The RPS and most engineered safety feature channels are supplied with sufficient redundancy to provide the capability for channel test at power, except for backup channels such as derived circuits in the ESF logic system.

When one of the four channels is taken out of service for maintenance, RPS logic can be changed to a two-out-of-three coincidence for a reactor trip by bypassing the removed channel. If the bypass is not effected, the out-of-service channel (Power Removed) assumes a tripped condition (except high rate-of-change of power, high power level and high pressurizer pressure),[1] which results in a one-out-of-three channel logic. If in the 2-out-of-4 logic system of the RPS one channel is bypassed and a second channel manually placed in a tripped condition, the resulting logic is 1-out-of-2. At rated power, the minimum OPERABLE high-power level channel is 3 in order to provide adequate power tilt detection. If only 2 channels are OPERABLE, the reactor power level is reduced to 70% rated power which protects the reactor from possibly exceeding design peaking factors due to undetected flux tilts and from exceeding dropped CEA peaking factors.

An RPS Logic matrix channel consists of two matrix power supplies, four matrix relays and their associated contacts as well as all interconnecting wiring. An RPS Trip Initiation Logic channel consists of an M contactor and associated contacts, an interposing relay and all interconnecting wiring. Two RPS Trip Initiation Logic channels associated with the same pair of CEDM clutch power supplies are considered to affect the same trip leg.

Integrated into the trip initiation logic are two RPS Manual Trip channels. Manual Trip #1 operates by directly de-energizing all four M contactors in response to the operation of a manual pushbutton. Manual Trip #2 operates by de-energizing an undervoltage relay which results in the opening of two circuit breakers, CB-AB and CB-CD, which supply power to the CEDM clutch power supplies. Manual Trip channel #1 consists of manual trip pushbutton #1 and interconnecting wiring. Manual Trip channel #2 consists of manual trip pushbutton #2, circuit breakers CB-AB and CB-CD, and associated interconnecting wiring.

With one manual reactor trip channel inoperable, operation may continue until the reactor is shut down for other reasons. No safety analyses assume operation of the Manual trip. Because of this, the Required Action is to restore the inoperable channel to OPERABLE status prior to entering MODE 2 from MODE 3 during the next plant startup.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.15 Instrumentation and Control Systems (Continued)

Basis (Continued)

The ESF logic system is a Class 1 protection system designed to satisfy the criteria of IEEE 279, August 1968. Two functionally redundant ESF logic subsystems "A" and "B" are provided to ensure high reliability and effective in-service testing. These logic subsystems are designed for individual reliability and maximum attainable mutual independence both physically and electrically. Either logic subsystem acting alone can automatically actuate engineered safety features and essential supporting systems.

All Engineered Safety Features are initiated by 2-out-of-4 logic matrices except containment high radiation which operates on a 1-out-of-2 basis. The number of installed channels for Containment Radiation High Signal (CRHS) is two. CRHS isolates the containment pressure relief, air sample and purge system valves.

Entry into Technical Specification 2.15.1(3) is made when conditions have caused one logic subsystem ("A" or "B") to become inoperable but the redundant logic subsystem remains operable. The loss of a prime initiation relay (which renders all 4 channels of a logic subsystem inoperable) is the condition most likely to cause entry into Technical Specification 2.15.1(3). In this situation, the remaining ESF logic subsystem still has the capability to automatically actuate engineered safety features equipment and essential supporting systems. The 48-hour completion time is commensurate with the importance of avoiding the vulnerability of a single failure in the remaining ESF logic subsystem. Technical Specification 2.15.1(3) will not be used upon loss of the common channels that affect both "A" and "B" subsystems prime initiators operability unless the permissible bypass condition is met. Upon exiting TS 2.15.1(3) following the restoration of a prime initiation relay to OPERABLE status, if any channel(s) remain inoperable, the appropriate Limiting Conditions for Operation (LCO) (TS 2.15.1(1) or TS 2.15.1(2) is applicable with the length of inoperability measured from time of discovery of: 1) prime initiation relay inoperable, or 2) channel inoperability, whichever is longer.

The ESF system provides a 2-out-of-4 logic on the signals used to actuate the equipment connected to each of the two emergency diesel generator units.

The rod block system automatically inhibits all CEA motion in the event a LCO on CEA insertion, CEA deviation, CEA overlap or CEA sequencing is approached. The installation of the rod block system ensures that no single failure in the control element drive control system (other than a dropped CEA) can cause the CEAs to move such that the CEA insertion, deviation, sequencing or overlap limits are exceeded. Accordingly, with the rod block system installed, only the dropped CEA event is considered an Anticipated Operational Occurrence (AOO) and factored into the derivation of the Limiting Safety System Settings (LSSS) and LCO. With the rod block function out-of-service, several additional CEA deviation events must be considered as AOOs. Analysis of these incidents indicates that the single CEA withdrawal incident is the most limiting of these events. An analysis of the at-power single CEA withdrawal incident was performed for Fort Calhoun for various initial Group 4 insertions, and it has been concluded that the LCO and LSSS are valid for a Group 4 insertion of less than or equal to 15%.

## 2.0 LIMITING CONDITIONS FOR OPERATION
### 2.15 Instrumentation and Control Systems (Continued)

Basis (Continued)

Operability of the primary CEA position indication system (CEAPIS) channel and the secondary CEAPIS channel is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits of TS 2.10.2. The primary CEAPIS channel utilizes the output of a synchro transmitter geared to the clutch output shaft. CEA position is displayed visually at the main control panel.

The secondary CEAPIS channel utilizes the output of a voltage divider network controlled by a series of reed switches. The reed switches are actuated by a permanent magnet attached to the rack assembly. Position information is supplied to the distributed control system (DCS) flat-panel touch monitors for simultaneous viewing of all CEA group positions.

Limit switches on the regulating CEAs and reed switches on the shutdown CEAs provide an additional means of determining CEA position when the CEAs are fully inserted or fully withdrawn. When the CEAs are fully inserted or fully withdrawn, this indication (displayed on the DCS) can be used in lieu of secondary CEAPIS data to meet the shiftly CHANNEL CHECK of primary CEAPIS. However, as limit switch indication is not fully independent of secondary CEAPIS, primary CEAPIS must be used to verify secondary CEAPIS data.

In MODES 1 and 2, CEA position indication is required to allow verification that the CEAs are positioned and aligned as assumed in the safety analysis. If one CEA position indication channel is inoperable for one or more CEAs, TS 3.1, Table 3-3, Item 4 (CEA position verification) must be performed within 15 minutes following any CEA motion in that group to ensure that the CEAs are positioned as required.

The operability of the Alternate Shutdown Panel (AI-185), including Wide Range Logarithmic Power and Source Range Monitors on AI-212, and Emergency Auxiliary Feedwater Panel (AI-179) instrument and control circuits ensures that sufficient capability is available to permit entry into and maintenance of the Hot Shutdown Mode from locations outside of the Control Room. This capability is required in the event that Control Room habitability is lost due to fire in the cable spreading room or Control Room.

Variances which may exist at startup between the more accurate ΔT-Power and Nuclear Instrumentation Power (NI-Power) are not significant for enabling of the trip functions. By 15% of rated power as measured by the uncalibrated NI Power, the Axial Power Distribution (APD) and Loss of Load (LOL) trip functions are enabled while the High Rate of Change of Power trip is bypassed.

The APD trip function acts to limit the axial power shape to the range assumed in the setpoint analysis. Significant margins to local power density limits exist at 15% power, as well as power levels up to at least 30% (where NI calibration occurs).



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## 2.0 LIMITING CONDITIONS FOR OPERATION

2.15 Instrumentation and Control Systems (Continued)

Basis (Continued)

The LOL trip function acts as an anticipatory trip for the high pressurizer pressure and high power trips in order to limit the severity of a LOL transient. This trip is not credited in the USAR Chapter 14 Safety Analyses and any variance between ΔT-Power and NI-Power has no effect on the safety analysis.

The High Rate of Change of Power trip acts to limit power excursions from low power levels and bypassing of this trip at a high power level is conservative. This trip is not credited in the USAR Chapter 14 Safety Analyses for Mode 1 operation. Any variance between ΔT-Power and NI-Power has no effect on the safety analysis.

Steam generator blowdown isolation ensures that the auxiliary feedwater system performs its design function of maintaining adequate steam generator (SG) water level for decay heat removal once the auxiliary feedwater actuation signal (AFAS) is actuated. The steam generator blowdown isolation function consists of two trains (logic subsystems). Each train closes one SG blowdown isolation valve to each SG. Each SG has redundant (Train A and Train B) blowdown isolation valves. Four clutch power relays initiate closure of the SG blowdown isolation valves with each clutch power relay closing one valve when the reactor trips. Failure of one clutch power relay to initiate SG blowdown isolation or failure of one train will not prevent single valve isolation of SG blowdown flow.

References

(1) USAR, Section 7.2.7.1

## TABLE 2-2

## Instrument Operating Requirements for Reactor Protective System

| No. | Functional Unit | Minimum Operable Channels | Minimum Degree of Redundancy | Permissible Bypass Condition | Test, Maintenance and Inoperable Bypass |
|---|---|---|---|---|---|
| 1 | Not Used | | | | |
| 2 | High Power Level | 2[b][c] | 1[c] | Thermal Power Input Bypassed below $10^{-4}$% of Rated Power[a][d] | (e) |
| 3 | Thermal Margin/Low Pressurizer Pressure | 2[b] | 1 | Below $10^{-4}$% of Rated Power[a][d] | (e) |
| 4 | High Pressurizer Pressure | 2[b] | 1 | None | (e) |
| 5 | Low R.C. Flow | 2[b] | 1 | Below $10^{-4}$% of Rated Power [a][d] | (e) |
| 6 | Low Steam Generator Water Level | 2/Steam Gen[b] | 1/Steam Gen | None | (e) |
| 7 | Low Steam Generator Pressure | 2/Steam Gen[b] | 1/Steam Gen | Below 600 psia[a][d] | (e) |
| 8 | Containment High Pressure | 2[b] | 1 | During Leak Test | (e) |
| 9 | Axial Power Distribution | 2[b][c] | 1[c] | Below 15% of Rated Power[g] | (e) |
| 10 | High Rate Trip-wide Range Log Channels | 2[b] | 1 | Below $10^{-4}$% and above 15% of Rated Power[a][g] | (e) |
| 11 | Loss of Load | 2[b] | 1 | Below 15% of Rated Power[g] | (e) |
| 12 | Steam Generator Differential Pressure | 2[b] | 1 | None | (e) |

a. Bypass automatically removed.

b. Specification 2.15.1(2) is applicable.

**TABLE 2-2**
(Continued)

c. If two channels are inoperable, load shall be reduced to 70% or less of rated power.

d. For low power physics testing this trip may be bypassed up to $10^{-1}$% of rated power.

e. Specification 2.15.1(1) is applicable.

f. Deleted.

g. For each channel, the same bistable automatically activates the Loss of Load and Axial Power Distribution (APD) trips and automatically bypasses the high rate trip at 15% of rated power. Only the APD trip is a Limiting Safety System Setting. Therefore, the bistable is set to actuate within the APD tolerance band.

## TABLE 2-3

## Instrument Operating Requirements for Engineered Safety Features

| No. | Functional Unit | Minimum Operable Channels | Minimum Degree of Redundancy | Permissible Bypass Condition | Test, Maintenance and Inoperable Bypass |
|---|---|---|---|---|---|
| 1 | Safety Injection | | | | |
| A | Manual | 1 | None | None | N/A |
| B | High Containment Pressure | | | | |
| | Logic Subsystem A | 2[a][d][l] | 1 | During Leak | (f) |
| | Logic Subsystem B | 2[a][d][l] | 1 | Test | |
| C | Pressurizer Low/Low Pressure | | | | |
| | Logic Subsystem A | 2[a][d][l] | 1 | Reactor Coolant | (f) |
| | Logic Subsystem B | 2[a][d][l] | 1 | Pressure Less Than 1700 psia[b] | |
| 2 | Containment Spray | | | | |
| A | Manual[m] | 1 | None | None | N/A |
| B | High Containment Pressure | | | | |
| | Logic Subsystem A | 2[a][c][d][l] | 1 | During Leak | (f) |
| | Logic Subsystem B | 2[a][c][d][l] | 1 | Test | |
| C | Pressurizer Low/Low Pressure | | | | |
| | Logic Subsystem A | 2[a][c][d][l] | 1 | Reactor Coolant | (f) |
| | Logic Subsystem B | 2[a][c][d][l] | 1 | Pressure Less Than 1700 psia[b] | |
| D | Steam Generator Low Pressure | | | | |
| | Logic Subsystem A | 2/Steam Gen[a][c][d][l] | 1/Steam Gen | Steam Generator Pressure Less Than 600 psia[n] | (f) |
| | Logic Subsystem B | 2/Steam Gen[a][c][d][l] | 1/Steam Gen | | |
| 3 | Recirculation | | | | |
| A | Manual | 1 | None | None | N/A |
| B | SIRW Tank Low Level | | | | |
| | Logic Subsystem A | 2[a][k][l] | 1 | None | (j) |
| | Logic Subsystem B | 2[a][k][l] | 1 | | |
| 4 | Emergency Off-Site Power Trip | | | | |
| A | Manual | 1[e] | None | None | N/A |
| B | Emergency Bus Low Voltage (Each Bus) | | | | |
| | -Loss of Voltage | 2[d] | 1 | Reactor Coolant | (f) |
| | -Degraded Voltage | 2[a][d] | 1 | Temperature Less Than 300° F | |

**TABLE 2-3**
(Continued)

| No. | Functional Unit | Minimum Operable Channels | Minimum Degree of Redundancy | Permissible Bypass Condition | Test, Maintenance and Inoperable Bypass |
|---|---|---|---|---|---|
| 5 | Auxiliary Feedwater | | | | |
| A | Manual | 1 | None | None | N/A |
| B | Auto. Initiation<br>Logic Subsystem A<br>Logic Subsystem B | | | Operating Modes 3, 4, and 5 | |
| | -Steam Generator Low Level | 2[a][d][l] | 1 | | (h) |
| | -Steam Generator Low Pressure | 3[a][g][l] | 1 | | (i) |
| | -Steam Generator Differential Pressure | 3[a][g][l] | 1 | | (i) |

a Circuits on ESF Logic Subsystems A and B each have 4 channels.

b Auto removal of bypass above 1700 psia.

c Coincident containment high pressure, pressurizer low/low pressure, and steam generator low pressure signals are required for initiation of containment spray.

d If minimum OPERABLE channel conditions are reached, one inoperable channel must be placed in the tripped condition or low level actuation position for auxiliary feedwater system within eight hours from the time of discovery of loss of operability. Specification 2.15.1(2) is applicable.

e Control switch on incoming breaker.

f If one channel becomes inoperable, that channel must be placed in the tripped or bypassed condition within eight hours from time of discovery of loss of operability. Specification 2.15.1(1) is applicable.

g Three channels required because bypass or failure results in auxiliary feedwater actuation block in the affected channel.

h Specification 2.15.1(1) is applicable.

**TABLE 2-3**
(Continued)

i If the channel becomes inoperable, that channel must be placed in the bypassed condition within eight hours from time of discovery of loss of operability. If the channel is not returned to OPERABLE status within 48 hours from time of discovery of loss of operability, one of the eight channels may continue to be placed in the bypassed condition provided the Plant Operations Review Committee has reviewed and documented the judgment concerning prolonged operation in bypass of the inoperable channel. The channel shall be returned to OPERABLE status no later than during the next cold shutdown. If one of the four channels on one steam generator is in prolonged bypass and a channel on the other steam generator becomes inoperable, the second inoperable channel must be placed in bypass within eight hours from time of discovery of loss of operability. If one of the inoperable channels is not returned to OPERABLE status within seven days from the time of discovery of the second loss of operability, the unit must be placed in hot shutdown within the following 12 hours.

j If one channel becomes inoperable, that channel must be placed in the bypassed condition within eight hours from time of discovery of loss of operability. If the channel is not returned to OPERABLE status within 48 hours from time of discovery of loss of operability, one of the eight channels may continue to be placed in the bypassed condition provided the Plant Operations Review Committee has reviewed and documented the judgment concerning prolonged operation in bypass of the inoperable channel. The channel shall be returned to OPERABLE status no later than during the next cold shutdown. If a channel is in prolonged bypass and a channel on the opposite train becomes inoperable, the second inoperable channel must be placed in bypass within eight hours from time of discovery of loss of operability. If one of the inoperable channels is not returned to OPERABLE status within seven days from the time of discovery of the second loss of operability, the unit must be placed in hot shutdown within the following 12 hours.

k Specification 2.15.1(2) is applicable.

l Specification 2.15.1(3) is applicable. If ESF Logic Subsystems A and B are inoperable, enter Specification 2.0.1.

m Steam Generator Low Pressure permissive is required for actuation.

n Auto removal of bypass prior to exceeding 600 psia.

## TABLE 2-4

### Instrument Operating Conditions for Isolation Functions

| No. | Functional Unit | Minimum Operable Channels | Minimum Degree of Redundancy | Permissible Bypass Condition | Test, Maintenance and Inoperable Bypass |
|---|---|---|---|---|---|
| 1 | Containment Isolation | | | | |
| A | Manual | 1 | None | None | N/A |
| B | Containment High Pressure | | | | |
| | Logic Subsystem A | 2[a][e][g] | 1 | During Leak Test | (f) |
| | Logic Subsystem B | 2[a][e][g] | 1 | | |
| C | Pressurizer Low/Low Pressure | | | | |
| | Logic Subsystem A | 2[a][e][g] | 1 | Reactor Coolant Pressure Less Than 1700 psia[b] | (f) |
| | Logic Subsystem B | 2[a][e][g] | 1 | | |
| 2 | Steam Generator Isolation | | | | |
| A | Manual | 1 | None | None | N/A |
| B | Steam Generator Isolation | 1 | None | None | N/A |
| | (i) Steam Generator Low Pressure | | | | |
| | Logic Subsystem A | 2/Steam Gen[a][e][g] | 1/Steam Gen | Steam Generator Pressure Less Than 600 psia[c] | (f) |
| | Logic Subsystem B | 2/Steam Gen[a][e][g] | 1/Steam Gen | | |
| | (ii) Containment High Pressure | | | | |
| | Logic Subsystem A | 2[a][e][g] | 1 | During Leak Test | (f) |
| | Logic Subsystem B | 2[a][e][g] | 1 | | |
| 3 | Ventilation Isolation | | | | |
| A | Manual | 1 | None | None | N/A |
| B | Containment High Radiation | | | | |
| | Logic Subsystem A | 1[d][g] | None | If Containment Relief and Purge Valves are Closed | (f) |
| | Logic Subsystem B | 1[d][g] | None | | |
| 4 | Steam Generator Blowdown Isolation | | | | |
| A | Manual | 1[h] | None | Operating Modes 3, 4, & 5 | N/A |
| B | Reactor Trip Trains A and B | 2[h][i] | None | Operating Modes 3, 4, & 5 OR if at least one valve for each steam generator is closed | (j) |

**TABLE 2-4**
(Continued)

a Circuits on ESF Logic Subsystems A and B each have 4 channels.

b Auto removal of bypass prior to exceeding 1700 psia.

c Auto removal of bypass prior to exceeding 600 psia.

d A and B trains are both actuated by either the Containment or Auxiliary Building Exhaust Stack initiating channels. The number of installed channels for Containment Radiation High Signal is two for purposes of Specification 2.15.1(1).

e If minimum operable channel conditions are reached, one inoperable channel must be placed in the tripped condition within eight hours from the time of discovery of loss of operability. Specification 2.15.1(2) is applicable.

f If one channel becomes inoperable, that channel must be placed in the tripped or bypassed condition within eight hours from the time of discovery of loss of operability. Specification 2.15.1(1) is applicable.

g Specification 2.15.1(3) is applicable. If ESF Logic Subsystems A and B are inoperable, enter Specification 2.0.1.

h. "Minimum Operable Channels" for steam generator blowdown isolation refers to the minimum number of trains (logic subsystems) which are required to be operable to provide manual or automatic SG blowdown isolation.

i. If both trains become inoperable, power operation may continue provided at least one SG blowdown isolation valve for each steam generator is closed OR be in MODE 2 within 6 hours, and in MODE 3 in the next 6 hours. Specifications 2.15.1(1), (2), (3) and (4) are <u>not</u> applicable; TS LCO 2.0.1 is <u>not</u> applicable.

j. If one train becomes inoperable, that train may be placed in the bypassed condition. If the train is not returned to OPERABLE status within 24 hours from time of discovery of loss of operability, operation may continue as long as one SG blowdown isolation valve to each steam generator is closed. If the train is not returned to OPERABLE status within 24 hours from time of discovery, with blowdown not isolated to both SGs, be in MODE 2 within 6 hours, and in MODE 3 in the next 6 hours. Specifications 2.15.1(1), (2), (3) and (4) are <u>not</u> applicable; TS LCO 2.0.1 is <u>not</u> applicable.

## TABLE 2-5

## Instrumentation Operating Requirements for Other Safety Feature Functions

| No. | Functional Unit | Minimum Operable Channels | Minimum Degree of Redundancy | Permissible Bypass Condition |
|---|---|---|---|---|
| 1 | CEA Position Indication Systems | 1[(a)] | None | None |
| 2 | Pressurizer Level | 1 | None | Not Applicable |

**NOTES**:

(a) If one channel of CEA position indication is inoperable for one or more CEAs, requirements of specification 2.15.1 are modified for item 1 to "Perform TS 3.1, Table 3-3. Item 4 within 15 minutes following any CEA motion in that group." Specifications 2.15.1(1), (2), and (3) are not applicable.

## TABLE 2-6

### Alternate Shutdown and Auxiliary Feedwater Panel Functions

| Function/Instrument or Control Parameter | Location | Required Number of Channels |
|---|---|---|
| 1. Reactivity Control | | |
| a. Source Range Power | AI-212 | 1 |
| b. Reactor Wide Range Logarithmic Power | AI-212 | 1 |
| 2. Reactor Coolant System Pressure Control | | |
| a. Pressurizer Wide Range Pressure (0-2500 psia) | AI-179 | 1 |
| 3. Decay Heat Removal via Steam Generators | | |
| a. Reactor Coolant Hot Leg Temperature | AI-185 | 1 (Note 1) |
| b. Reactor Coolant Cold Leg Temperature | AI-185 | 1 (Note 1) |
| c. Steam Generator Pressure | AI-179 | 1 per Steam Generator |
| d. Steam Generator Narrow Range Level | AI-179 | 1 per Steam Generator |
| e. Steam Generator Wide Range Level | AI-179 | 1 per Steam Generator |
| 4. Reactor Coolant System Inventory Controls | | |
| a. Pressurizer Level | AI-185 | 1 |
| b. Volume Control Tank Level | AI-185 | 1 |
| c. Charging Pump CH-1B and its associated controls | AI-185 | 1 |
| d. Charging Isolation Valve Control | AI-185 | 1 |
| 5. Transfer Functions | | |
| a. All Transfer Switches/Lockout Relays | AI-185 | 1 |
| b. All Transfer Switches/Lockout Relays | AI-179 | 1 |

**Note 1**: One reactor coolant hot leg temperature indication and one reactor coolant cold leg temperature indication channel must both be operable on the same steam generator (i.e., RC-2A or RC-2B).

**TABLE 2-6**
(Continued)

**Alternate Shutdown and Auxiliary Feedwater Panel Functions**

| Function/Instrument or Control Parameter | Location | Required Number of Channels |
|---|---|---|
| 6. Auxiliary Feedwater Controls | | |
| a. Steam Generator RC-2A and 2B Auxiliary Feedwater Isolation Inboard and Outboard Valves Control | AI-179 | 1 |
| b. Steam-Driven Pump FW-10 Recirculation Valve Control | AI-179 | 1 |
| c. Steam-Driven Pump FW-10 Steam Isolation Valve Control | AI-179 | 1 |
| d. Steam from Steam Generator RC-2A and RC-2B to FW-10 Steam Isolation Valve Control | AI-179 | 1 |

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.16 River Level

Applicability

At all times.

Objective

To specify maximum and minimum Missouri River levels as measured at the intake structure which must be present to assure safe reactor operation.

Specification

The water level of the Missouri River shall remain less than 1004 feet mean sea level and greater than or equal to 976 feet 9 inches mean sea level as measured at the intake structure.

Required Actions

(1) When the Missouri River level reaches elevation 1004 feet mean sea level, the reactor shall be in a HOT SHUTDOWN condition and in COLD SHUTDOWN within 36 hours following entry into HOT SHUTDOWN; and

(2) If the Missouri River level is less than 976 feet 9 inches mean sea level, the reactor shall be placed in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 36 hours; and,

(3) At Missouri River levels less than 980 feet or greater than 1002 feet mean sea level, a continuous watch will be maintained to monitor river levels to assure no sudden loss of water supply occurs on low river level and provide adequate response time for rising river levels.

Basis

At the Fort Calhoun Station (FCS) site, the probable maximum flood that might occur as a result of runoff from a probable maximum rainstorm over the area below the Gavins Point dam coupled with an assumed outflow of 50,000 cubic feet per second from Gavins Point reservoir is 1009.3 feet. In the unlikely event that the Oahe or Fort Randall dams fail at that time, the Corps of Engineers has estimated that the flood level could be as high as 1014 feet[1].

The intake structure can be protected from these Missouri River floods using removable flood gates on doorways and the screen wash discharge trough. The water level inside the intake cells can be controlled by positioning the exterior sluice gates to restrict the flow into the cells. The position of the exterior sluice gates must be verified by manual actuation, which requires access to the intake structure veranda. Access to the veranda is lost when the east doors to the intake structure are blocked by installing the flood barriers which must be installed prior to a river level of 1004 feet in order to allow egress to the north. This requires the station to be shutdown prior to 1004 feet. The 36-hour allowance to cold shutdown following hot shutdown entry allows for cool down by steaming to atmosphere, if desired. If the station desires to cool down by steaming to the condenser, shutdown cooling must be initiated prior to 1004 feet so that actions verifying sluice gate position can be completed before access to the veranda is lost. A continuous watch will be established at 1002 feet msl to provide adequate response time for rising river levels in accordance with the abnormal operating procedure. The FCS emergency plan will be implemented during these high and low river level conditions to protect the plant. The auxiliary building can be protected to 1014 feet with the installation of removable flood barriers and sandbagging at the 1013 foot elevation of the equipment hatch room (Room 66).

**LIMITING CONDITIONS FOR OPERATION**

2.16 River Level (Continued)

Basis (Continued)

The minimum river level of 976 feet 9 inches provides adequate suction to the raw water (RW) pumps for cooling plant components. The minimum elevation of the RW pump suction is 973 feet 9 inches. An intake cell level of 976 feet 9 inches is required for RW pump minimum submergence level (MSL)[2]. The partial loss of this supply is considered highly unlikely. However, provisions for low water levels during winter and spring ice conditions are considered necessary. When river level is low, head loss from debris and/or ice on the traveling screens and/or trash racks could reduce intake cell levels such that the required RW pump MSL is not achieved. This could lead to pump degradation from the formation of vortices at the free water surface. Thus, when the continuous watch requirement is in effect, in addition to monitoring river level to assure no sudden loss of water supply occurs, the level of the intake cells is monitored.

Intake cell levels are also adversely affected by the flows associated with the non-safety related circulating water (CW) pumps since the large flow rates associated with the CW pumps create significant head losses even with relatively clean intake cell conditions. However, the CW pumps have a much higher MSL requirement (983 feet 0 inches) and would become unstable and trip or be manually shutdown well before intake cell levels decrease to the RW pump MSL. The head loss associated with CW pump flow would then be recovered and intake cell levels would rise.

References

(1) USAR, Section 2.7.1.2
(2) USAR, Section 9.8

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.20 Steam Generator Coolant Radioactivity

Applicability

Applies to the radioactivity of the steam generator coolant.

Objective

To ensure that the steam generator coolant radioactivity is maintained at a level commensurate with the occupational and public safety.

Specification

(1) The radioactivity of the steam generator coolant shall be ≤ 0.10 $\mu$Ci/gm DOSE EQUIVALENT I-131.

(2) With the radioactivity of the steam generator coolant > 0.10 $\mu$Ci/gm DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN within 6 hours.

Basis

The limitations on the steam generator coolant's radioactivity ensure that the resultant off-site doses will be well within the limits of 10 CFR Part 100 in the event of a steam line break. This dose also includes the effects of a coincident 1.0 GPM primary-to-secondary tube leak in the steam generator of the affected steam line and a concurrent loss of off-site power.

References

USAR Section 14.12.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.21 Post-Accident Monitoring Instrumentation

Applicability

Applies to post-accident monitoring instrumentation not included as part of the Reactor Protective System or Engineered Safety Features. This specification is applicable while in modes 1, 2 and 3.

Objective

To assure that instrumentation necessary to monitor plant parameters during post-accident conditions is operable or that backup methods of analysis are available.

Specifications

Post-accident instrumentation shall be operable as provided in Table 2-10. If the required instrumentation is not operable, then the appropriate action specified in Table 2-10 shall be taken.

Basis

Post-accident monitoring instrumentation provides information, during and following an accident, which is considered helpful to the operator in determining the plant condition. It is desirable that this instrumentation be operable at all times during operation of the plant. However, none of the post-accident monitors are required for safe shutdown of the plant nor are any control or safety actions initiated by the monitors.

In general, the post-accident monitors provide wide range capabilities for parameters which are beyond the range of normal protective and control instrumentation. They also provide remote sampling and analysis capability to reduce personnel exposure under post-accident conditions. Because the information necessary to assess the effect of an accident (i.e., core damage) can be obtained from other sources and by manual methods, it is not necessary that the post-accident monitors be operable at all times.

The subcooled margin monitor, the Heated Junction Thermocouple (HJTC) and the Core Exit Thermocouple (CET) comprise the Inadequate Core Cooling Instrumentation (ICCI) System required by Item II.F.2 of NUREG-0737. The function of the ICCI is to enhance the ability of the plant operator to diagnose the approach to, the existence of, and the recovery from inadequate core cooling. Additionally, it aids in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of Generic Letter 83-37. They are not required by accident analysis or to bring the plant to cold shutdown.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.21 Post-Accident Monitoring Instrumentation (Continued)

In the event more than four (4) sensors in a reactor vessel level (HJTC) channel are inoperable, repairs may only be possible during the next refueling outage. This is because the sensors are accessible only after the missile shields and the reactor vessel head cable trays are removed. It is not feasible to repair a channel except during a refueling outage. If only one channel is inoperable, it should be restored to OPERABLE status in a refueling outage as soon as possible. If both channels are inoperable, at least one channel shall be restored to operable status during the next refueling outage.

The core exit thermocouples were installed pursuant to NUREG-0737. There are seven (7) installed per core quadrant, four of which are required by the NUREG. This is clarified via the footnote to Table 2-10.

## TABLE 2-10

## Post-Accident Monitoring Instrumentation Operating Limits

| Instrument | Minimum Operable Channels | Action |
|---|---|---|
| 1. Containment Wide Range Radiation Monitors (RM-091A & B) | 2 | (a) |
| 2. Wide Range Noble Gas Stack Monitor RM-063 (Noble Gas Portion Only) | 1 | (a) |
| 3. Main Steam Line Radiation Monitor (RM-064) | 1 | (a) |
| 4. Not Used | | |
| 5. Containment Water Level | | |
| Narrow Range (LT-599 & LT-600) | 1 | (d) |
| Wide Range (LT-387 & LT-388) | 2 | (b)(c) |
| 6. Containment Wide Range Pressure | 2 | (b)(c) |
| 7. Reactor Coolant System Subcooled Margin Monitor | 2 | (e)(f) |
| 8. Core Exit Thermocouples (i) | 2/Core Quadrant | (g)(h) |
| 9. Reactor Vessel Level (HJTC) (j) | 2 | (k)(l) |

(a) With the number of OPERABLE channels less than required by the minimum channels operable requirements, initiate the pre-planned alternate method of monitoring the appropriate parameter(s) within 72 hours, and

1. either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

2. prepare and submit a special report to the Commission pursuant to specification 5.9.3 within 14 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedules for restoring the system to OPERABLE status.

(b) With one channel inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

**TABLE 2-10 (Continued)**

**Post-Accident Monitoring Instrumentation Operating Limits**

(c) With both channels inoperable, restore at least one channel to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours.

(d) With the number of OPERABLE channels less than required by the minimum channels operable requirements, operation may continue until the next cold shutdown, at which time the required channel(s) shall be made operable.

(e) With the number of OPERABLE channels one less than the minimum channels operable requirement, either

1. restore the inoperable channel(s) to OPERABLE status within 7 days, or
2. initiate an alternate means of monitoring the subcooled margin, or
3. be in at least HOT SHUTDOWN within the next 12 hours.

(f) With both channels inoperable,

1. restore the inoperable channel(s) to OPERABLE status within 48 hours, or
2. initiate an alternate means of monitoring the subcooled margin, or
3. be in at least HOT SHUTDOWN within the next 12 hours.

(g) With the number of OPERABLE Core Exit Thermocouples per core quadrant one less than the minimum operable requirement, either restore the inoperable Core Exit Thermocouple(s) to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

(h) With all Core Exit Thermocouples within a core quadrant inoperable, either restore the inoperable Core Exit Thermocouple(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

(i) With the number of OPERABLE Core Exit Thermocouples less than the four per core quadrant required by NUREG-0737, either restore to at least four OPERABLE Core Exit Thermocouples per core quadrant within seven days of discovery of loss of operability, or prepare and submit a special report to the Commission pursuant to Specification 5.9.3 within 30 days, outlining the actions taken, the cause of the inoperability and the plans for restoring the inoperable Core Exit Thermocouple(s) to OPERABLE status.

(j) A channel is eight sensors in a probe. A channel is OPERABLE if four or more sensors, two or more in the upper four and two or more in the lower four, are OPERABLE.

**TABLE 2-10 (Continued)**

**Post-Accident Monitoring Instrumentation Operating Limits**

(k) With the number of OPERABLE channels one less than the minimum channels operable requirement,

1. either restore the inoperable channel to OPERABLE status within 7 days of discovery of loss of operability if repairs are feasible during power operation (MODE 1), or

2. prepare and submit a special report to the Commission pursuant to Specification 5.9.3 within 30 days of discovery of loss of operability, outlining the action taken, the cause of the inoperability, and the plans for restoring the channel to operable status.

(l) With both channels inoperable, either restore the inoperable channel(s) to OPERABLE status within 48 hours of discovery of loss of operability if repairs are feasible during power operation (MODE 1), or

1. initiate an alternate method of monitoring the reactor vessel inventory, and

2. prepare and submit a special report to the Commission pursuant to Specification 5.9.3, within 30 days of discovery of loss of operability, outlining the action taken, the cause of the inoperability and the plans and schedules for restoring the system to OPERABLE status, and

3. restore the system to OPERABLE status at the next scheduled Refueling Outage.

Not Used

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.23 Steam Generator (SG) Tube Integrity

Applicability

Applies whenever the reactor coolant temperature ($T_{cold}$) is greater than 210°F.

Objective

To ensure that SG tube integrity is maintained.

Specifications

NOTE: Separate Condition entry is allowed for each SG Tube.

(1) The following conditions shall be maintained:

(a) SG tube integrity shall be maintained, and

(b) All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

(2) If the requirements of (1)(b) above are not met for one or more SG tubes, then perform the following:

(a) Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection within 7 days, and

(b) Plug the affected tube(s) in accordance with the Steam Generator Program prior to exceeding 210°F reactor coolant temperature (Tcold) following the next refueling outage or SG tube inspection.

(3) If the Required Action and associated completion time of (2), above, is not met, or if SG tube integrity is not maintained, then be in MODE 3, Hot Shutdown, within 6 hours AND be in MODE 4, Cold Shutdown, within 36 hours.

Basis

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by Technical Specification 2.1.1, "Operable Components."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

## 2.0 LIMITING CONDITIONS FOR OPERATION

2.23 Steam Generator (SG) Tube Integrity (continued)

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.23, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.23, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.23. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in Technical Specification 2.1.4. "Reactor Coolant System Leakage Limits," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes releases of activity occur from the faulted steam generator to the environment via the condenser air ejector and Main Steam Safety Valves (MSSVs) and Atmospheric Dump Valves (ADVs). The release via the condenser air ejector starts at the initiation of the event and continues to the reactor trip, while the release via the MSSVs/ADVs starts at the reactor trip and continues for the duration of the event."

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute or is assumed to increase to 1 gallon per minute as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT 1-131 is assumed to be equal to the Technical Specification 2.1.3, "Reactor Coolant Radioactivity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to- tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.23, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

## 2.0 **LIMITING CONDITIONS FOR OPERATION**

2.23 Steam Generator (SG) Tube Integrity (continued)

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section Ill, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm per SG. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in Technical Specification 2.1.4. "Reactor Coolant System Leakage Limits," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced when $T_{cold}$ is > 210°F.

RCS conditions are far less challenging in MODES 4 and 5 than during MODES 1, 2, and 3. In MODES 4 and 5, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

Specification 2.23(2) applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by Technical Specification 3.17. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program.

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.23 Steam Generator (SG) Tube Integrity (continued)

The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Specification 2.23(3) applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action 2.23(2)b allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to exceeding 210°F reactor coolant temperature ($T_{cold}$) following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

If the Required Actions and associated Completion Times of Technical Specification 2.23(2) are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 4 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

References

1. NEI 97-06, "Steam Generator Program Guidelines."

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 100.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

## 3.0 SURVEILLANCE REQUIREMENTS

3.0.1 Each surveillance requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

3.0.2 The surveillance intervals are defined as follows:

| Notation | Title | Frequency |
|---|---|---|
| S | Shift | At least once per 12 hours |
| D | Daily | At least once per 24 hours |
| W | Weekly | At least once per 7 days |
| BW | Biweekly | At least once per 14 days |
| M | Monthly | At least once per 31 days |
| Q | Quarterly | At least once per 92 days |
| SA | Semiannual | At least once per 184 days |
| A | Annually | At least once per 366 days |
| R | Refueling | At least once per 18 months |
| P | Start up | Prior to Reactor Start up, if not completed in the previous week. |

Exception to these intervals are stated in the individual Specifications.

3.0.3 The provisions of Specifications 3.0.1 and 3.0.2 are applicable to all codes and standards referenced within the Technical Specifications. The requirements of the Technical Specifications shall have precedence over the requirements of the codes and standards referenced within the Technical Specifications.

3.0.4 Surveillance Requirements shall be met during the MODES or other specified conditions in the individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the OPERABILITY requirements for the corresponding Limiting Condition for Operation. Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specifications 3.0.1 and 3.0.2, shall constitute noncompliance with the OPERABILITY requirements for the corresponding Limiting Condition for Operation except as provided in Specification 3.0.5. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. Surveillance Requirements do not have to be performed on inoperable equipment.

3.0.5 If it is discovered that a Surveillance was not performed within its specified surveillance interval, then compliance with the requirement to declare the OPERABILITY requirements for the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

## 3.0 SURVEILLANCE REQUIREMENTS

If the Surveillance is not performed within the delay period, the OPERABILITY requirements for the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION(S) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the OPERABILITY requirements for the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION(S) must be entered.

## 3.0 SURVEILLANCE REQUIREMENTS
## BASIS

Specifications 3.0.1 through 3.0.5 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

> "Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting condition of operation will be met."

Specification 3.0.1 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillance that are not performed during refueling outages. The limitation of Specification 3.0.1 is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

The provisions of Specification 3.0.2 define the surveillance intervals for use in the Technical Specifications. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications. A few surveillance requirements have uncommon intervals. In such a case the surveillance interval shall be performed as defined by the individual specifications.

Specification 3.0.3 extends the testing interval required by codes and standards referenced by the Technical Specifications. This clarification is provided to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities. Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the codes and standards referenced therein. The requirements of regulations take precedence over the TS. Therefore test intervals governed by regulation cannot be extended by the TS. An example of this exception is the Containment Leakage Rate Testing Program.

Specification 3.0.4 establishes the requirement that Surveillances must be met during the MODES or other specified conditions in the Specification for which the requirements of the Limiting Condition for Operation apply, unless otherwise specified in the individual Surveillances. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified surveillance interval, in accordance with Specifications 3.0.1 and 3.0.2, constitutes a failure to meet the OPERABILITY requirements for the corresponding Limiting Condition for Operation.

## 3.0 SURVEILLANCE REQUIREMENTS

**BASIS** (continued)

Systems and components are assumed to be OPERABLE when the associated Surveillances have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when either:

a. The systems or components are known to be inoperable, although still meeting the Surveillances or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated Limiting Condition for Operation are not applicable unless otherwise specified. The Surveillances associated with a special test exception (STE) are only applicable when the STE is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given Surveillance. In this case, the unplanned event may be credited as fulfilling the performance of the Surveillance. This allowance includes those Surveillances whose performance is normally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by ACTIONS, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with Specifications 3.0.1 and 3.0.2, prior to returning equipment to OPERABLE status.

Specification 3.0.5 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified surveillance interval. A delay period of up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with Specification 3.0.1, and not at the time that the specified surveillance interval was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with ACTIONS or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

## 3.0 SURVEILLANCE REQUIREMENTS
**BASIS** (continued)

When a Surveillance with a surveillance interval based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, Specification 3.0.5 allows for the full delay period of up to the specified surveillance interval to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity. Specification 3.0.5 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by ACTIONS.

Failure to comply with specified surveillance intervals for Surveillance Requirements is expected to be an infrequent occurrence. Use of the delay period established by Specification 3.0.5 is a flexibility which is not intended to be used as an operational convenience to extend surveillance intervals. While up to 24 hours or the limit of the specified surveillance interval is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the corrective action program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the allowable outage time limits of the ACTIONS for the applicable Limiting Condition for Operation begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the allowable outage time limits of the ACTIONS for the applicable Limiting Condition for Operation begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the allowable outage time limits of the ACTIONS, restores compliance with Specification 3.0.4.

# 3.0 SURVEILLANCE REQUIREMENTS

## 3.1 Instrumentation and Control

### Applicability

Applies to the reactor protective system and other critical instrumentation and controls.

### Objective

To specify the minimum frequency and type of surveillance to be applied to critical plant instrumentation and controls.

### Specifications

Calibration, testing and checking of instrument channels, reactor protective system and engineered safeguards system logic channels and miscellaneous instrument systems and controls shall be performed as specified in Tables 3-1 to 3-3a.

### Basis

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action and a check supplements this type of built-in surveillance.

If the channels are normally off scale during times when Surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

Based on the District's experience in operation of conventional power plants and on reported nuclear plant experience, a checking frequency of once-per-shift is deemed adequate for reactor and steam system instrumentation. Calibrations are performed to ensure the presentation and acquisition of accurate information.

The power range safety channels are calibrated (heat balance adjustment only) daily against a calorimetric balance standard to account for errors induced by changing rod patterns and core physics parameters.

Other channels, subject only to the "drift" errors, can be expected to remain within acceptable tolerances if recalibration is performed on a refueling frequency.

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.1 Instrumentation and Control (Continued)

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

The minimum calibration frequencies of once-per-day (heat balance adjustment only) for the power range safety channels, and once each refueling shutdown for the process system channels, are considered adequate.

The minimum testing frequency for those instrument channels connected to the Reactor Protective System and Engineered Safety Features is based on ABB/CE probabilistic risk analyses and the accumulation of specific operating history. The quarterly frequency for the channel functional tests for these systems is based on the analyses presented in the NRC approved topical report CEN-327-A, "RPS/ESFAS Extended Test Interval Evaluation," as supplemented, and OPPD's Engineering Analysis EA-FC-93-064, "RPS/ESF Functional Test Drift Analysis."

The low temperature setpoint power operated relief valve (PORV) CHANNEL FUNCTIONAL TEST verifies operability of the actuation circuitry using the installed test switches. PORV actuation could depressurize the reactor coolant system and is not required.

Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.

Calculation of the Reactor Coolant System (RCS) total flow rate by performance of a precision calorimetric heat balance once every 18 months verifies that the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate (Table 3-3, Item 15, Reactor Coolant Flow).

The frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered, Steam Generator tubes plugged or repaired, or other activities, which may have caused an alteration of flow resistance.

This requirement is modified by a footnote that requires the surveillance to be performed within 24 hours after ≥95% reactor thermal power (RTP) following power escalation from a refueling outage. The footnote is necessary to allow measurement of the flow rate at normal operating conditions at power in MODE 1.

## 3.0 **SURVEILLANCE REQUIREMENTS**

### 3.1 Instrumentation and Control (Continued)

The Control Room Envelope (CRE) surveillance requirement (SR) verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Technical Specification (TS) 2.12.1(2) must be entered. TS 2.12.1(2)c allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 1) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 2). These compensatory measures may also be used as mitigating actions as required by TS 2.12.1(2)b. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 3). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

References

1. Regulatory Guide 1.196
2. NEI 99-03, "Control Room Habitability Assessment," June 2001
3. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability." (Adams Accession No. ML040300694).

**TABLE 3-1**

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF REACTOR PROTECTIVE SYSTEM**

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 1. Power Range Safety Channels | a. Check: | S | a. |
| | 1) Neutron Flux | | 1) CHANNEL CHECK |
| | 2) Thermal Power | | 2) CHANNEL CHECK |
| | b. Adjustment | D(3) | b. Channel adjustment to agree with heat balance calculation. |
| | c. Test | Q(1) | c. CHANNEL FUNCTIONAL TEST |
| 2. Wide-Range Logarithmic Neutron Monitors | a. Check | S | a. CHANNEL CHECK |
| | b. Test(2) | P | b. CHANNEL FUNCTIONAL TEST |
| 3. Reactor Coolant Flow | a. Check | S | a. CHANNEL CHECK |
| | b. Test | Q(1) | b. CHANNEL FUNCTIONAL TEST |
| | c. Calibrate | R | c. CHANNEL CALIBRATION |

**TABLE 3-1** (Continued)

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF REACTOR PROTECTIVE SYSTEM**

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 4. Thermal Margin/Low Pressure | a. Check | S | a. |
| | 1) Pressure Setpoint | | 1) CHANNEL CHECK |
| | 2) Pressure Input | | 2) CHANNEL CHECK |
| | b. Test | Q(1) | b. CHANNEL FUNCTIONAL TEST |
| | c. Calibrate: | R | c. |
| | 1) Temperature Input | | 1) CHANNEL CALIBRATION |
| | 2) Pressure Input | | 2) CHANNEL CALIBRATION |
| 5. High-Pressurizer Pressure | a. Check | S | a. CHANNEL CHECK |
| | b. Test | Q(1) | b. CHANNEL FUNCTIONAL TEST |
| | c. Calibrate | R | c. CHANNEL CALIBRATION |

**TABLE 3-1** (Continued)

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF REACTOR PROTECTIVE SYSTEM**

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 6. Steam Generator Level | a. Check | S | a. CHANNEL CHECK |
| | b. Test | Q(1) | b. CHANNEL FUNCTIONAL TEST |
| | c. Calibrate | R(4) | c. CHANNEL CALIBRATION |
| 7. Steam Generator Pressure | a. Check | S | a. CHANNEL CHECK |
| | b. Test | Q(1) | b. CHANNEL FUNCTIONAL TEST |
| | c. Calibrate | R | c. CHANNEL CALIBRATION |
| 8. Containment Pressure | a. Test | Q(1) | a. CHANNEL FUNCTIONAL TEST |
| | b. Calibrate | R | b. CHANNEL CALIBRATION |
| 9. Loss of Load | a. Test | P | a. CHANNEL FUNCTIONAL TEST |
| 10. Manual Trips | a. Test | P | a. CHANNEL FUNCTIONAL TEST |
| 11. Steam Generator Differential Pressure | a. Check | S | a. CHANNEL CHECK |
| | b. Test | Q(1) | b. CHANNEL FUNCTIONAL TEST |
| | c. Calibrate | R | c. CHANNEL CALIBRATION |

**TABLE 3-1** (Continued)

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF REACTOR PROTECTIVE SYSTEM**

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 12. Reactor Protection System Logic Units | a. Test | Q(1) | a. CHANNEL FUNCTIONAL TEST |
| 13. Axial Power Distribution | a. Check: | S | a. |
| | 1) Axial Shape Index Indication | | 1) CHANNEL CHECK |
| | 2) Upper Trip Setpoint Indication | | 2) CHANNEL CHECK |
| | 3) Lower Trip Setpoint Indication | | 3) CHANNEL CHECK |
| | b. Test | Q(1) | b. CHANNEL FUNCTIONAL TEST |
| | c. Calibrate | R | c. CHANNEL CALIBRATION |

**NOTES:**

(1) The quarterly tests will be done on only one of four channels at a time to prevent reactor trip.

(2) Calibrate using built-in simulated signals.

(3) Not required unless the reactor is in the power operating condition and is therefore not required during plant startup and shutdown periods.

(4) During PLANT OPERATING CYCLE 27 only, the maximum allowed surveillance test interval shall not exceed 28 months.

## TABLE 3-2

## MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURES, INSTRUMENTATION AND CONTROLS

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 1. Pressurizer Pressure Low | a. Check | S | a. CHANNEL CHECK |
| | b. Test | Q[1]P[4] | b. CHANNEL FUNCTIONAL TEST |
| | c. Calibrate | R | c. CHANNEL CALIBRATION |
| 2. Pressurizer Low Pressure Blocking Circuit | a. Calibrate | R | a. CHANNEL CALIBRATION |
| 3. Safety Injection Actuation Logic | a. Test | Q | a. CHANNEL FUNCTIONAL TEST (Simulation of PPLS or CPHS 2/4 Logic) |
| | b. Test | R[7] | b. CHANNEL FUNCTIONAL TEST |

**TABLE 3-2** (continued)

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURES, INSTRUMENTATION AND CONTROLS**

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 4. Containment Pressure High Signal | a. Test | Q | a. CHANNEL FUNCTIONAL TEST |
| | b. Calibrate | R | b. CHANNEL CALIBRATION |
| 5. Containment Spray Actuation Logic | a. Test | Q | a. CHANNEL FUNCTIONAL TEST (Simulation of PPLS, CPHS, and SGLS[8] 2/4 Logic) |
| | b. Test | R[7] | b. CHANNEL FUNCTIONAL TEST |
| 6. Containment Radiation High Signal [2] | a. Check | D | a. CHANNEL CHECK |

**TABLE 3-2** (continued)

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURES, INSTRUMENTATION AND CONTROLS**

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 6. (continued) | b. Test | Q | b. CHANNEL FUNCTIONAL TEST |
| | c. Calibrate | R | c. Secondary and Electronic Calibration performed at refueling frequency. Primary calibration performed with exposure to radioactive sources only when required by the secondary and electronic calibration. |
| 7. Manual Safety Injection Actuation | a. Test | R | a. CHANNEL FUNCTIONAL TEST |
| 8. Manual Containment Isolation Actuation | a. Check | R(9) | a. Observe isolation valves closure. |
| | b. Test | R(9) | b. CHANNEL FUNCTIONAL TEST |
| 9. Manual Containment Spray Actuation | a. Test | R | a. CHANNEL FUNCTIONAL TEST |
| 10. Automatic Load Sequencers | a. Test | Q | a. CHANNEL FUNCTIONAL TEST |
| 11. Diesel Testing | See Technical Specification 3.7 | | |

**TABLE 3-2** (continued)

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURES, INSTRUMENTATION AND CONTROLS**

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 12. Diesel Fuel Transfer Pump | a. Test | M | a. Pump run to refill day tank. |
| 13. SIRW Tank Low Level Signal | a. Check | S | a. CHANNEL CHECK |
| | b. Test | Q | b. CHANNEL FUNCTIONAL TEST |
| | c. Calibrate | R | c. CHANNEL CALIBRATION |
| 14. Safety Injection Tank Level and Pressure | a. Check | S(5) | a. Verify that level and pressure are within limits. |

**TABLE 3-2** (continued)

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURES, INSTRUMENTATION AND CONTROLS**

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 14. (continued) | b. Calibrate | R | b. CHANNEL CALIBRATION |
| 15. Boric Acid Tank Level | a. Check | W | a. Verify that level is within limits. |
| 16. Boric Acid Tank Temperature | a. Check | W | a. Verify that temperature is within limits. |
| 17. Steam Generator Low Pressure Signal (SGLS) | a. Check | S | a. CHANNEL CHECK |
| | b. Test | Q(3) | b. CHANNEL FUNCTIONAL TEST |
| | c. Calibrate | R | c. CHANNEL CALIBRATION |

**TABLE 3-2** (continued)

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURES, INSTRUMENTATION AND CONTROLS**

| Channel Description | Surveillance Function | | Frequency | Surveillance Method | |
|---|---|---|---|---|---|
| 18. SIRW Tank Temperature | a. | Check | D[6] | a. | Verify that temperature is within limits. |
| | b. | Test | R | b. | Measure temperature of SIRW tank with standard laboratory instruments. |
| 19. Manual Recirculation Actuation | a. | Test | R[10] | a. | CHANNEL FUNCTIONAL TEST |
| 20. Recirculation Actuation Logic | a. | Test | Q | a. | CHANNEL FUNCTIONAL TEST |
| | b. | Test | R[7][10] | b. | CHANNEL FUNCTIONAL TEST |
| 21. 4.16 KV Emergency Bus Low Voltage (Loss of Voltage and Degraded Voltage) Actuation Logic | a. | Check | S | a. | Verify voltage readings are above alarm initiation on degraded voltage level - supervisory lights "on". |
| | b. | Test | Q | b. | CHANNEL FUNCTIONAL TEST (Undervoltage relay) |
| | c. | Calibrate | R | c. | CHANNEL CALIBRATION |
| 22. Manual Emergency Off-site Power Low Trip Actuation | a. | Test | R | a. | CHANNEL FUNCTIONAL TEST |

**TABLE 3-2** (continued)

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURES, INSTRUMENTATION AND CONTROLS**

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 23. Auxiliary Feedwater | a. Check: | S | a. |
| | 1) Steam Generator Water Level Low (Wide Range) | | 1) CHANNEL CHECK |
| | 2) Steam Generator Pressure Low | | 2) CHANNEL CHECK |
| | b. Test: | QR(7) | b. |
| | 1) Actuation Logic | | 1) CHANNEL FUNCTIONAL TEST |
| | c. Calibrate: | R | c. |
| | 1) Steam Generator Water Level Low (Wide Range)(10) | | 1) CHANNEL CALIBRATION |
| | 2) Steam Generator Pressure Low | | 2) CHANNEL CALIBRATION |
| | 3) Steam Generator Differential Pressure High | | 3) CHANNEL CALIBRATION |
| 24. Manual Auxiliary Feedwater Actuation | a. Test | R | a. CHANNEL FUNCTIONAL TEST |
| 25. Manual Steam Generator Blowdown Isolation | a. Test | R | a. CHANNEL FUNCTIONAL TEST |
| 26. Automatic Steam Generator Blowdown Isolation | a. Test | R | a. CHANNEL FUNCTIONAL TEST |

**NOTES**: (1) Not required unless pressurizer pressure is above 1700 psia.
(2) CRHS monitors are the containment atmosphere gaseous radiation monitor and the Auxiliary Building Exhaust Stack gaseous radiation monitor.
(3) Not required unless steam generator pressure is above 600 psia.
(4) QP - Quarterly during designated modes and prior to taking the reactor critical if not completed within the previous 92 days (not applicable to a fast trip recovery).
(5) Not required to be done on a SIT with inoperable level and/or pressure instrumentation.
(6) Not required when outside ambient air temperature is greater than 50°F and less than 105°F.
(7) Tests backup channels such as derived circuits and equipment that cannot be tested when the plant is at power.
(8) SGLS is required for containment spray pump actuation only. SGLS lockout relays are not actuated for this test.
(9) During PLANT OPERATING CYCLE 27 only, the maximum allowed surveillance test interval for valves PCV-1849A&B shall not exceed 28 months.
(10) During PLANT OPERATING CYCLE 27 only, the maximum allowed surveillance test interval shall not exceed 28 months.

**TABLE 3-3**

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS**

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 1. Primary CEA Position Indication System | a. Check | S | a. CHANNEL CHECK |
| | b. Test | M | b. Test of power dependent insertion limits, deviation, and sequence monitoring systems. |
| | c. Calibrate | R | c. Physically measured CEDM position used to verify system accuracy. Calibrate CEA position interlocks. |
| 2. Secondary CEA Position Indication System | a. Check | S | a. Comparison of output data with primary CEAPIS. |
| | b. Test | M | b. Test of power dependent insertion limit, deviation, out-of-sequence, and overlap monitoring systems. |
| | c. Calibrate | R | c. Calibrate secondary CEA position indication system and CEA interlock alarms. |
| 3. Area and Post-Accident Radiation Monitors[1] | a. Check | D | a. CHANNEL CHECK |
| | b. Test | Q | b. CHANNEL FUNCTIONAL TEST |
| | c. Calibrate | R | c. Secondary and Electronic calibration performed at refueling frequency. Primary calibration with exposure to radioactive sources only when required by the secondary and electronic calibration. RM-091 A/B - Calibration by electronic signal substitution is acceptable for all range decades above 10 R/hr. Calibration for at least one decade below 10 R/hr. shall be by means of calibrated radiation source. |

[1]Post Accident Radiation Monitors are: RM-063, RM-064, and RM-091A/B. Area Radiation Monitors are: RM-070 thru RM-082, RM-084 thru RM-089, and RM-095 thru RM-098.

**TABLE 3-3** (Continued)

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS**

| Channel Description | Surveillance Function | | Frequency | Surveillance Method |
|---|---|---|---|---|
| 4. CEA Position | a. | Check | S | a. Verify the position of each CEA to be within 12 inches of all other CEAs in the group. |
| 5. Primary to Secondary Leak-Rate Detection Radiation Monitors (RM-054A/B, RM-057) | a. | Check | D | a. CHANNEL CHECK |
| | b. | Test | Q | b. CHANNEL FUNCTIONAL TEST |
| | c. | Calibrate | R | c. Secondary and Electronic calibration performed at refueling frequency. Primary Calibration performed with exposure to radioactive sources only when required by the secondary and electronic calibration. |
| 6. Pressurizer Level | a. | Check | S | a. Verify that level is within limits. |
| | b. | Check | M | b. CHANNEL CHECK |
| | c. | Calibrate | R | c. CHANNEL CALIBRATION |
| 7. CEA Drive System Interlocks | a. | Test | R | a. Verify proper operation of all CEDM system interlocks, using simulated signals where necessary. |
| | b. | Test | P | b. If haven't been checked for three months and plant is shutdown. |

**TABLE 3-3** (Continued)

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS**

| Channel Description | Surveillance Function | | Frequency | Surveillance Method |
|---|---|---|---|---|
| 8. Dropped CEA Indication | a. | Test | R | a. Insert a negative rate of change power signal to all four Power Range Safety Channels to test alarm. |
| | b. | Test | R | b. Insert CEA's below lower electrical limit to test dropped CEA alarm. |
| 9. Calorimetric Instrumentation | a. | Calibrate | R | a. CHANNEL CALIBRATION |
| 10. Control Room Ventilation System | a. | Test | R | a. Check damper operation for DBA mode. |
| | b. | Test | In accordance with CRE Habitability Program | b. Perform required control room envelope (CRE) unfiltered air inleakage testing in accordance with the CRE Habitability Program. |
| 11. Containment Humidity Detector | a. | Test | R | a. CHANNEL FUNCTIONAL TEST |
| 12. Interlocks-Isolation Valves on Shutdown Cooling Line | a. | Test | R | a. CHANNEL FUNCTIONAL TEST |
| 13. Control Room Air Conditioning System | a. | Test | R | a. Verify each train has the capability to remove the assumed heat load through combination of testing and calculations. |

**TABLE 3-3** (Continued)

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS**

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 14. Not Used | | | |
| 15. Reactor Coolant System Flow | a. Check | R[1] | a. Calculation of reactor coolant flow rate. |
| 16. Pressurizer Pressure | a. Check | S | a. CHANNEL CHECK |
| 17. Reactor Coolant Inlet Temperature | a. Check | S | a. CHANNEL CHECK |
| 18. Low-Temperature Set-point Power-Operated Relief Valves | a. Test | PM | a. CHANNEL FUNCTIONAL TEST (excluding actuation) |
| | b. Calibrate | R | b. CHANNEL CALIBRATION |

[1] Required to be performed within 24 hours after ≥ 95.00% reactor thermal power following power escalation.

**TABLE 3-3** (Continued)

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS**

| Channel Description | Surveillance Function | | Frequency | Surveillance Method |
|---|---|---|---|---|
| 19. Auxiliary Feedwater Flow | a. | Check | M | CHANNEL CHECK |
| | b. | Calibrate | R | CHANNEL CALIBRATION |
| 20. Subcooled Margin Monitor | a. | Check | M | CHANNEL CHECK |
| | b. | Calibrate | R | CHANNEL CALIBRATION |
| 21. PORV Operation | a. | Test | R | CHANNEL FUNCTIONAL TEST |
| 22. PORV Block Valve Operation and Position Indication | a. | Check | Q | Cycle valve. Valve is exempt from testing when it has been closed to comply with LCO Action Statement 2.1.6(5)a. |
| | b. | Calibrate | R | Check valve stroke against limit switch position. |
| 23. Not Used | | | | |
| 24. Not Used | | | | |

**TABLE 3-3** (Continued)

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS**

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 25. Containment Purge Isolation Valves (PCV-742A, B, C, & D) | a. Check | M | a. Verify valve position using control room indication. |
| 26. Not Used | | | |
| 27. Containment Water Level Narrow Range (LT-599 & LT-600) | a. Check | M | a. CHANNEL CHECK |
| | b. Calibrate | R | b. CHANNEL CALIBRATION |
| Wide Range (LT-387 & LT-388) | a. Check | M | a. CHANNEL CHECK |
| | b. Calibrate | R | b. CHANNEL CALIBRATION |
| 28. Containment Wide Range Pressure Indication | a. Check | M | a. CHANNEL CHECK |
| | b. Calibrate | R | b. CHANNEL CALIBRATION |
| 29. Not Used | | | |

**TABLE 3-3** (Continued)

**MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS**

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 30. Core Exit Thermo-couple | a. Check | M | a. CHANNEL CHECK |
| | b. Calibrate | R | b. CHANNEL CALIBRATION |
| 31. Heated Junction Thermocouple (YE-116A and YE-116B) | a. Check | M | a. CHANNEL CHECK |
| | b. Calibrate | R | b. CHANNEL CALIBRATION |

PM - Prior to scheduled cold leg cooldown below 300°F; monthly whenever temperature remains below 300°F and reactor vessel head is installed.

**TABLE 3-3A**
**MINIMUM FREQUENCY FOR CHECKS, CALIBRATIONS AND FUNCTIONAL TESTING OF ALTERNATE SHUTDOWN PANELS (AI-185 AND AI-212) AND EMERGENCY AUXILIARY FEEDWATER PANEL (AI-179) INSTRUMENTATION AND CONTROL CIRCUITS**

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 1. WIDE RANGE LOGARITHMIC POWER AND SOURCE RANGE MONITORS (AI-212) | a. CHECK | M | a. CHANNEL CHECK |
| | b. CALIBRATE | R | b. CHANNEL CALIBRATION |
| 2. REACTOR COOLANT COLD LEG TEMPERATURE (AI-185) | a. CHECK | M | a. CHANNEL CHECK |
| | b. CALIBRATE | R | b. CHANNEL CALIBRATION |
| 3. REACTOR COOLANT HOT LEG TEMPERATURE (AI-185) | a. CHECK | M | a. CHANNEL CHECK |
| | b. CALIBRATE | R | b. CHANNEL CALIBRATION |
| 4. PRESSURIZER LEVEL (AI-185) | a. CHECK | M | a. CHANNEL CHECK |
| | b. CALIBRATE | R | b. CHANNEL CALIBRATION |
| 5. VOLUME CONTROL TANK LEVEL (AI-185) | a. CHECK | M | a. CHANNEL CHECK |
| | b. CALIBRATE | R | b. CHANNEL CALIBRATION |
| 6. ASP CONTROL CIRCUITS (AI-185) | a. TEST | R | a. CHANNEL FUNCTIONAL TEST |

**TABLE 3-3A** (Continued)

**MINIMUM FREQUENCY FOR CHECKS, CALIBRATIONS AND FUNCTIONAL TESTING OF ALTERNATE SHUTDOWN PANELS (AI-185 AND AI-212) AND EMERGENCY AUXILIARY FEEDWATER PANEL (AI-179) INSTRUMENTATION AND CONTROL CIRCUITS**

| Channel Description | Surveillance Function | Frequency | Surveillance Method |
|---|---|---|---|
| 7. STEAM GENERATOR LEVEL, WIDE RANGE (AI-179) | a. CHECK | M | a. CHANNEL CHECK |
| | b. CALIBRATE | R[1] | b. CHANNEL CALIBRATION |
| 8. STEAM GENERATOR LEVEL, NARROW RANGE (AI-179) | a. CHECK | M | a. CHANNEL CHECK |
| | b. CALIBRATE | R[1] | b. CHANNEL CALIBRATION |
| 9. STEAM GENERATOR PRESSURE (AI-179) | a. CHECK | M | a. CHANNEL CHECK |
| | b. CALIBRATE | R | b. CHANNEL CALIBRATION |
| 10. PRESSURIZER PRESSURE (AI-179) | a. CHECK | M | a. CHANNEL CHECK |
| | b. CALIBRATE | R | b. CHANNEL CALIBRATION |
| 11. EAFW CONTROL CIRCUITS (AI-179) | a. TEST | R | a. CHANNEL FUNCTIONAL TEST |

[1] During PLANT OPERATING CYCLE 27 only, the maximum allowed surveillance test interval shall not exceed 28 months.

# 3.0 **SURVEILLANCE REQUIREMENTS**

## 3.2 Equipment and Sampling Tests

### Applicability

Applies to plant equipment and conditions related to safety.

### Objective

To specify the minimum frequency and type of surveillance to be applied to critical plant equipment and conditions.

### Specifications

Equipment and sampling tests shall be conducted as specified in Tables 3-4 and 3-5.

### Basis

The equipment testing and system sampling frequencies specified in Tables 3-4 and 3-5 are considered adequate, based upon experience, to maintain the status of the equipment and systems so as to assure safe operation. Thus, those systems where changes might occur relatively rapidly are sampled frequently and those static systems not subject to changes are sampled less frequently.

The control room air filtration system (CRAFS) consists of redundant high efficiency particulate air filters (HEPA) and charcoal adsorbers. HEPA filters are installed before and after the charcoal adsorbers. The charcoal adsorbers are installed to reduce the potential intake of iodine to the control room. The in-place test results will confirm system integrity and performance. The laboratory carbon sample test results should indicate methyl iodide removal efficiency of at least 99.825 percent for expected accident conditions.

CRAFS standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Monthly heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. Each CRAFS train must be operated for ≥ 10 continuous hours with the heaters energized. The monthly Frequency is based on the known reliability of the equipment, and the two train redundancy available.

Each CRAFS train is verified to start and operate on an automatic and manual actuation signal. The Frequency of 18 months is based on industry operating experience and is consistent with the typical refueling cycle.

## 3.0 **SURVEILLANCE REQUIREMENTS**

### 3.2 Equipment and Sampling Tests (continued)

The spent fuel storage-decontamination areas air treatment system is designed to filter the building atmosphere to the auxiliary building vent during refueling operations. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. In-place testing is performed to confirm the integrity of the filter system. The charcoal adsorbers are periodically sampled to insure capability for the removal of radioactive iodine.

The Safety Injection (SI) pump room air treatment system consists of charcoal adsorbers which are installed in normally bypassed ducts. This system is designed to reduce the potential release of radioiodine in SI pump rooms during the recirculation period following a DBA. The in-place and laboratory testing of charcoal adsorbers will assure system integrity and performance.

Pressure drops across the combined HEPA filters and charcoal adsorbers, of less than 9 inches of water for the control room filters (VA-64A & VA-64B) and of less than 6 inches of water for each of the other air treatment systems will indicate that the filters and adsorbers are not clogged by amounts of foreign matter that would interfere with performance to established levels.

The hydrogen purge system provides the control of combustible gases (hydrogen) in containment for a post-LOCA environment. The surveillance tests provide assurance that the system is operable and capable of performing its design function. VA-80A or VA-80B is capable of controlling the expected hydrogen generation (67 SCFM) associated with 1) Zirconium - water reactions, 2) radiolytic decomposition of sump water and 3) corrosion of metals within containment. The system should have a minimum of one blower with associated valves and piping (VA-80A or VA-80B) available at all times to meet the guidelines of Regulatory Guide 1.7 (1971).

If significant painting, fire or chemical release occurs such that the HEPA filters or charcoal adsorbers could become contaminated from the fumes, chemicals or foreign materials, testing will be performed to confirm system performance.

Demonstration of the automatic and/or manual initiation capability will assure the system's availability.

Verifying Reactor Coolant System (RCS) leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary (RCPB) is maintained. Pressure boundary leakage would at first appear as unidentified leakage and can only be positively identified by inspection. Unidentified leakage is determined by performance of an RCS water inventory balance. Identified leakage is then determined by isolation and/or inspection. Since Primary to Secondary Leakage of 150 gallons per day cannot be measured accurately by an RCS water inventory balance, note "***" for line item 8a on Table 3-5 states that the Reactor Coolant System Leakage surveillance is not applicable to Primary to Secondary Leakage. Primary to secondary leakage is measured by performance of effluent monitoring within the secondary stem and feedwater systems.

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.2 Equipment and Sampling Tests (continued)

The RCS water inventory balance must be performed with the reactor at stable operating conditions (stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump (RCP) seal leakoff flows). Therefore, a note is added allowing that this surveillance requirement is not required to be performed until after establishing normal operating temperature and pressure.

Stable operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful. For RCS operational leakage determined by water inventory balance, stable operation is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal leakoff flows. The water inventory balance should be performed as soon as practical after stable conditions are met.

During Plant startup, a visual leak check is performed at normal system pressure prior to entering MODE 3. This verification is performed to ensure no RCPB leaks exist.

Table 3-5, Item 9a, ensures that there is an adequate inventory of fuel oil in the storage tanks to operate both diesel generators for eight hours and a single diesel generator for the remainder of the required 7-day operating period. If diesel fuel storage tanks FO-1 and FO-10 are maintained at a minimum indicated inventory of 16,000 gallons and 13,000 gallons respectively, the available onsite capacity will provide 7 days of diesel operating time with approximately 850 gallons of margin.

Table 3-5, Item 9b ensures that sufficient lube oil inventory is available to support at least 7 days of full load operation for each DG. The 500 gallon requirement is based on the DG manufacturer consumption values for the run time of the DG. Implicit in this Surveillance Requirement is the requirement to assure the capability to transfer the lube oil from its storage location to the DG, since the DG lube oil sump does not hold adequate inventory for 7 days of full load operation without the level reaching the manufacturer recommended minimum level. A 31 day Surveillance interval is adequate to ensure that a sufficient lube oil supply is onsite, since DG starts and run time are closely monitored by the unit staff.

For Table 3-5, Item 9c, the tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between receipt of new fuel and conducting the tests to exceed 31 days. The tests, limits, and applicable ASTM Standards are as follows:

a. Sample the new fuel oil in accordance with ASTM D4057 (Ref. 2),

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.2 Equipment and Sampling Tests (continued)

b. Verify in accordance with the tests specified in ASTM D975 (Ref. 2) that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89, or an API gravity at 60°F of ≥ 27° and ≤ 39° when tested in accordance with ASTM D287 (Ref. 2), a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point ≥ 125°F, and

c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176 or a water and sediment content within limits when tested in accordance with ASTM D2709 (Ref. 2).

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO concern since the fuel oil is not added to the storage tanks. Within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975 (Ref. 3) are met for new fuel oil when tested in accordance with ASTM D975 (Ref. 2), except that the analysis for sulfur may be performed in accordance with ASTM D2622 (Ref. 2). The 31-day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs. Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure. Particulate concentrations should be determined in accordance with ASTM 6217 (Ref. 2) with the exception that the filters specified in the ASTM method may have a nominal pore size of up to 3 microns. This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. For those designs in which the total stored fuel oil volume is contained in two or more interconnected tanks, each tank must be considered and tested separately. The Surveillance interval of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Surveillance intervals.

Table 3-5, Item 9d ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of five engine start cycles without recharging. A start cycle is defined as the cranking time required to accelerate the DG to firing speed. The pressure specified in this Surveillance Requirement is intended to reflect the lowest value at which the five starts can be accomplished. The 31-day Surveillance interval takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.2 Equipment and Sampling Tests (continued)

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks once every 92 days per Table 3-5, Item 9e, eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, and contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance interval is established to ensure excessive water does not accumulate in the fuel oil system, which meets the intent of Regulatory Guide 1.137 (Ref. 4). This Surveillance Requirement is for preventative maintenance. The presence of water does not necessarily represent failure of this Surveillance Requirement provided the accumulated water is removed during performance of the Surveillance

Table 3-5, Item 8b verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this surveillance requirement is not met, compliance with LCO 3.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a footnote which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of daily is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

Table 3-5, Item 25 verifies adequate measurements are taken to ensure that facility protective actions will be taken (and power operation will be terminated) in the event of high and/or low river level conditions. The high river level limit of less than 1004 feet mean sea level is based on the maximum elevation at which facility flood control measures provide protection to safety related equipment (i.e., due to restricted access/egress to the intake structure veranda once the flood barriers are installed prior to river level reaching 1004 feet msl). A continuous watch will be established at 1002 feet mean sea level to provide adequate response time for rising river levels in accordance with the abnormal operating procedure. The river level surveillance requirement specified also ensures sufficient net positive suction head is available for operating the RW pumps.

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.2 Equipment and Sampling Tests (continued)

The minimum river level of 976 feet 9 inches provides adequate suction to the RW pumps for cooling plant components. The surveillance frequency of "Daily" is a reasonable interval and models guidance provided in NUREG-0212, Revision 2, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors," Section 4.7.6. This surveillance requirement verifies that the Missouri River water level is maintained at a level greater than or equal to 976 feet 9 inches mean sea level. A continuous watch is established to monitor the river level when the river level reaches 980 feet mean sea level to assure no sudden loss of water supply occurs.

Table 3-5, Item 26 verifies the proper position of stops on high pressure safety injection system valves. The valves have stops to position them properly so that flow is restricted to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. The refueling frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned transients if the Surveillances were performed with the reactor at power.

References

1) USAR, Section 9.10
2) ASTM D4057, ASTM D975, ASTM D4176, ASTM D2622, ASTM D287, ASTM 6217, ASTM D2709
3) ASTM D975, Table 1
4) Regulatory Guide 1.137
5) EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

**TABLE 3-4**

**MINIMUM FREQUENCIES FOR SAMPLING TESTS**

| | Type of Measurement and Analysis | Sample and Analysis Frequency |
|---|---|---|
| 1. Reactor Coolant | | |
| (a) Power Operation (Operating Mode 1) | (1) Gross Radioactivity (Gamma emitters) | 1 per 3 days |
| | (2) Isotopic Analysis for DOSE EQUIVALENT I-131 | (i) 1 per 14 days |
| | | (ii) 1 per 8 hours[1] whenever the radioactivity exceeds 1.0 μCi/gm DOSE EQUIVALENT I-131. |
| | | (iii) 1 sample between 2-8 hours following a thermal power change exceeding 15% of the rated thermal power within a 1-hour period. |
| | (3) E Determination | 1 per 6 months[2] |
| | (4) Dissolved oxygen and chloride | 1 per 3 days |
| (b) Hot Standby (Operating Mode 2) | (1) Gross Radioactivity (Gamma emitters) | 1 per 3 days |
| Hot Shutdown (Operating Mode 3) | (2) Isotopic Analysis for DOSE EQUIVALENT I-131 | (i) 1 per 8 hours[1] whenever the radioactivity exceeds 1.0 μCi/gm DOSE EQUIVALENT I-131. |
| | | (ii) 1 sample between 2-8 hours following a thermal power change exceeding 15% of the rated thermal power change exceeding 15% of the rated thermal power within a 1-hour period. |
| | (3) Dissolved oxygen and chloride | 1 per 3 days |

**TABLE 3-4** (Continued)

**MINIMUM FREQUENCIES FOR SAMPLING TESTS**

| | Type of Measurement and Analysis | Sample and Analysis Frequency |
|---|---|---|
| 1. Reactor Coolant (Continued) | | |
| (c) Cold Shutdown (Operating Mode 4) | (1) Chloride | 1 per 3 days |
| (d) Refueling Shutdown (Operating Mode 5) | (1) Chloride<br>(2) Boron Concentration | 1 per 3 days(3)<br>1 per 3 days(3) |
| (e) Refueling Operation | (1) Chloride<br>(2) Boron Concentration | 1 per 3 days(3)<br>1 per 3 days(3) |
| 2. SIRW Tank | Boron Concentration | M |
| 3. Concentrated Boric Acid Tanks | Boron Concentration | W |
| 4. SI Tanks | Boron Concentration | M |
| 5. Spent Fuel Pool | Boron Concentration | See Footnote 4 below |
| 6. Steam Generator Blowdown (Operating Modes 1 and 2) | Isotopic Analysis for Dose Equivalent I-131 | W(5) |

(1) Until the radioactivity of the reactor coolant is restored to ≤1 μCi/gm DOSE EQUIVALENT I-131.

(2) Sample to be taken after a minimum of 2 EFPD and 20 days of power operation have elapsed since reactor was subcritical for 48 hours or longer.

(3) Boron and chloride sampling/analyses are not required when the core has been off-loaded. Reinitiate boron and chloride sampling/analyses prior to reloading fuel into the cavity to assure adequate shutdown margin and allowable chloride levels are met.

(4) Prior to placing unirradiated fuel assemblies in the spent fuel pool or placing fuel assemblies in a spent fuel cask in the spent fuel pool, and weekly when unirradiated fuel assemblies are stored in the spent fuel pool, or every 48 hours when fuel assemblies are in a spent fuel storage cask in the spent fuel pool.

(5) When Steam Generator Dose Equivalent I-131 exceeds 50 percent of the limits in Specification 2.20, the sampling and analysis frequency shall be increased to a minimum of 5 times per week. When Steam Generator Dose Equivalent I-131 exceeds 75 percent of this limit, the sampling and analysis frequency shall be increased to a minimum of once per day.

## TABLE 3-5
## MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

| | | Test | Frequency | USAR Section Reference |
|---|---|---|---|---|
| 1. | Control Element Assemblies | Drop times of all full-length CEA's | Prior to reactor criticality after each removal of the reactor vessel closure head | 7.5.3 |
| 2. | Control Element Assemblies | Partial movement of all CEA's (Minimum of 6 in) | Q | 7 |
| 3. | Pressurizer Safety Valves | Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be 2485 psig ±1% and 2530 psig ±1% respectively. | In accordance with the Inservice Testing Program[1] | 7 |
| 4. | Main Steam Safety Valves | Set Point | R | 4 |
| 5. | DELETED | | | |
| 6. | DELETED | | | |
| 7. | DELETED | | | |
| 8a. | Reactor Coolant System Leakage[3] | Evaluate | D[2] | 4 |
| 8b. | Primary to Secondary Leakage[4] | Continuous process radiation monitors or radiochemical grab sampling | D[2] | 4 |
| 9a | Diesel Fuel Supply | Verify diesel fuel inventory ≥ a 7 day supply of fuel. | M | 8.4 |
| 9b. | Diesel Lubricating Oil | Verify lubricating oil inventory ≥ a 7 day supply. | M | 8.4 |
| 9c. | Diesel Fuel Oil Properties | Test Properties | In accordance with the Diesel Fuel Oil Testing Program | 8.4 |
| 9d. | Required Diesel Generator Air Start Receiver Bank Pressure | Air Pressure | M | 8.4 |

**TABLE 3-5**
**MINIMUM FREQUENCIES FOR EQUIPMENT TESTS**

| | | Test | Frequency | USAR Section Reference |
|---|---|---|---|---|
| 9e. | Check for and Remove Accumulated Water from Each Fuel Oil Storage Tank | Check for Water and Remove | Q | 8.4 |
| 10a. | Charcoal and HEPA Filters for Control Room Air Filteration System (CRAFS) | 1. In-Place Testing[5]<br>Charcoal adsorbers and HEPA filter banks shall be leak tested and show ≥99.95% Freon (R-11 or R-112) and cold DOP particulates removal, respectively. | On a refueling frequency or every 720 hours of system operation or after each complete or partial replacement of the charcoal adsorber/HEPA filter banks, or after any major structural maintenance on the system housing or following significant painting, fire or chemical releases in a ventilation zone communicating with the system. | 9.10 |
| | | 2. Laboratory Testing[5]<br>Verify, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows methyliodide penetration less than 0.175% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%. | On a refueling frequency or every 720 hours of system operation or after any structural maintenance on the HEPA filter or charcoal adsorber housing or following significant painting, fire or chemical release in a ventilation zone communicating with the system. | |

**TABLE 3-5**
**MINIMUM FREQUENCIES FOR EQUIPMENT TESTS**

| | Test | Frequency | USAR Section Reference |
|---|---|---|---|
| 10a. (continued) | 3. Overall System Operation | | |
| | a. Each train shall be operated. | 15 continuous minutes every month with heaters operating. | |
| | b. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 9 inches of water at system design flow rate. | R | |
| | c. Fan shall be shown to operate within ± 10% design flow. | R | |
| | 4. Automatic and manual initiation of each train shall be demonstrated. | R | |
| 10b. Charcoal Adsorbers for Spent Fuel Storage Pool Area | 1. In-Place Testing[5] Charcoal adsorbers shall be leak tested and shall show ≥99% Freon (R-11 or R-112) removal. | On a refueling frequency or every 720 hours of system operation, or after each complete or partial replacement of the charcoal adsorber bank, or after any major structural maintenance on the system housing or following significant painting, fire or chemical release in a ventilation zone communicating with the system. | 6.2<br>9.10 |
| | 2. Laboratory Testing Verify, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows methyliodide penetration less than 10% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 95%. | On a refueling frequency or every 720 hours of system operation or after any structural maintenance on the HEPA filter or charcoal adsorber housing or following significant painting, fire or chemical release in a ventilation zone communicating with the system. | |

**TABLE 3-5**
**MINIMUM FREQUENCIES FOR EQUIPMENT TESTS**

| | | Test | Frequency | USAR Section Reference |
|---|---|---|---|---|
| 10b. | (continued) | 3. Overall System Operation<br>a. Operation of each circuit shall be demonstrated. | 15 continuous minutes every month. | |
| | | b. Volume flow rate through charcoal filter shall be shown to be between 4500 and 12,000 cfm. | R | |
| | | 4. Manual initiation of the system shall be demonstrated. | R | |
| 10c. | Charcoal Adsorbers for S.I. Pump Room | 1. In-Place Testing[5]<br>Charcoal adsorbers shall be leak tested and shall show ≥99% Freon (R-11 or R-112) removal. | On a refueling frequency or every 720 hours of system operation, or after each complete or partial replacement of the charcoal adsorber bank, or after any major structural maintenance on the system housing or following significant painting, fire or chemical release in any ventilation zone communicating with the system. | 9.10<br>6.2 |
| | | 2. Laboratory Testing<br>Verify, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows methyliodide penetration less than 10% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 95%. | On a refueling frequency or following 720 hours of system operation or after any structural maintenance on the HEPA filter or charcoal adsorber housing or following significant painting, fire or chemical release in a ventilation zone communicating with the system. | |
| | | 3. Overall System Operation<br>a. Operation of each circuit shall be demonstrated. | 15 continuous minutes every month. | |
| | | b. Volume flow rate shall be shown to be between 3000 and 6000 cfm. | R | |

**TABLE 3-5**
**MINIMUM FREQUENCIES FOR EQUIPMENT TESTS**

| | | Test | Frequency | USAR Section Reference |
|---|---|---|---|---|
| 10c. | (continued) | 4. Automatic and/or manual initiation of the system shall be demonstrated. | R | |
| 11. | Containment Ventilation System Fusible Linked Dampers | 1. Demonstrate damper action.<br>2. Test a spare fusible link. | 1 year, 2 years, 5 years, and every 5 years thereafter. | 9.10 |
| 12. | Diesel Generator Under-Voltage Relays | Calibrate | R | 8.4.3 |
| 13. | Motor Operated Safety Injection Loop Valve Motor Starters (HCV-311, 314, 317, 320, 327, 329, 331, 333, 312, 315, 318, 321) | Verify the contactor pickup value at ≤85% of 460 V. | R | |
| 14. | Pressurizer Heaters | Verify control circuits operation for post-accident heater use. | R | |
| 15. | Spent Fuel Pool Racks | Test neutron poison samples for dimensional change, weight, neutron attenuation change and specific gravity change. | 1, 2, 4, 7, and 10 years after installation, and every 5 years thereafter. | |
| 16. | Reactor Coolant Gas Vent System | 1. Verify all manual isolation valves in each vent path are in the open position. | During each refueling outage just prior to plant start-up. | |
| | | 2. Cycle each automatic valve in the vent path through at least one complete cycle of full travel from the control room. Verification of valve cycling may be determined by observation of position indicating lights. | R | |
| | | 3. Verify flow through the reactor coolant vent system vent paths. | R | |

**TABLE 3-5**
**MINIMUM FREQUENCIES FOR EQUIPMENT TESTS**

| | | Test | Frequency | USAR Section Reference |
|---|---|---|---|---|
| 17. | DELETED | | | |
| 18. | Shutdown Cooling | 1. Verify required shutdown cooling loops are OPERABLE and one shutdown cooling loop is IN OPERATION. | S (when shutdown cooling is required by TS 2.8). | |
| | | 2. Verify correct breaker alignment and indicated power is available to the required shutdown cooling pump that is not IN OPERATION. | W (when shutdown cooling is required by TS 2.8). | |
| 19. | Refueling Water Level | Verify refueling water level is ≥ 23 ft. above the top of the reactor vessel flange. | Prior to commencing, and daily during CORE ALTERATIONS and/or REFUELING OPERATIONS inside containment. | |
| 20. | Spent Fuel Pool Level | Verify spent fuel pool water level is ≥ 23 ft. above the top of irradiated fuel assemblies seated in the storage racks. | Prior to commencing, and weekly during REFUELING OPERATIONS in the the spent fuel pool. | |
| 21. | Containment Penetrations | Verify each required containment penetration is in the required status. | Prior to commencing, and weekly during CORE ALTERATIONS and/or REFUELING OPERATIONS in containment. | |
| 22. | Spent Fuel Assembly Storage | Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-10. | Prior to storing the fuel assembly in Region 2 (including peripheral cells). | |
| 23. | P-T Limit Curve | Verify RCS Pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified by the P-T limit Figure(s) shown in the PTLR. | This test is only required during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. While these operations are occurring, this test shall be performed every 30 minutes. | |
| 24. | Spent Fuel Cask Loading | Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-11. | Prior to placing the fuel assembly in a spent fuel cask in the spent fuel pool. | |

**TABLE 3-5**
**MINIMUM FREQUENCIES FOR EQUIPMENT TESTS**

| | Test | Frequency | USAR Section Reference |
|---|---|---|---|
| 25. River Level | Verify water level is within limits by measurement at least once per 24 hours, when the water level is less than 1004 feet and greater than or equal to 976 feet 9 inches above mean sea levels. | D | 9.8 |
| 26. HPSI Throttle Valves | Verify, for each HPSI throttle valve listed below, each position stop is in the correct position.<br><br>HCV-311 HCV-312<br>HCV-314 HCV-315<br>HCV-317 HCV-318<br>HCV-330 HCV-321 | R | |

[1] The provisions of Technical Specification 3.0.1 and 3.0.5 do not apply.
[2] Whenever the system is at or above operating temperature and pressure.
[3] Not applicable to primary to secondary LEAKAGE.
[4] Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG. This surveillance is not required to be performed until 12 hours after establishment of steady state operation.
[5] Tests shall be performed in accordance with applicable section(s) of ANSI N510-1980.

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.3 Reactor Coolant System and Other Components Subject to ASME XI Boiler & Pressure Vessel Code Inspection and Testing Surveillance

Applicability

Applies to in-service surveillance of primary system components and other components subject to inspection and testing according to ASME XI Boiler & Pressure Vessel Code.

Objective

To ensure the integrity of the reactor coolant system and other components subject to inspection and testing according to ASME XI Boiler & Pressure Vessel Code.

Specifications

(1) Surveillance of the ASME Code Class 1, 2 and 3 systems, except the steam generator tubes inspection, should be covered by ASME XI Boiler & Pressure Vessel Code.

a. Not Used

b. Surveillance of the reactor coolant pump flywheels shall be performed as indicated in Table 3-6.

c. A surveillance program to monitor radiation-induced changes in the mechanical and impact properties of the reactor vessel materials shall be maintained in accordance with 10 CFR Part 50 Appendix H.[1] Examinations results shall be used to update the PTLR.

(2) Surveillance of Reactor Coolant System Pressure Isolation Valves

a. Periodic leakage testing* on each valve listed in Table 2-9 shall be accomplished prior to entering the power operation mode every time the plant is placed in the cold shutdown

* To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.3 Reactor Coolant System and Other Components Subject to ASME XI Boiler & Pressure Vessel Code Inspection and Testing Surveillance (Continued)

condition for refueling, each time the plant is placed in a cold shutdown condition for 7 days if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.

b. Whenever the integrity of a pressure isolation valve listed in Table 2-9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure line shall be recorded daily.

Basis

Undetected prolonged leakage of borated reactor coolant onto carbon steel sets up an unusual corrosion mechanism. Detection of this leakage at an early stage can best be accommodated directly after an outage and before startup. The inspection program specified in Specification 3.3(1) places major emphasis on the areas of highest stress concentration as determined by general design evaluation and experience with similar systems. The inspections will rely on non-destructive analysis methods utilizing up-to-date analyzing equipment and trained personnel. Volumetric inspection of the reactor pressure vessel is to be performed completely from the outside diameter. The testing techniques and acceptance criteria of Section XI of the ASME B&PV Code will be utilized, except where specific relief is granted by the Commission.

Reference

(1) USAR, Section 4.5.3

## TABLE 3-6

## REACTOR COOLANT PUMP SURVEILLANCE

| | REQUIREMENT | METHOD | FREQUENCY |
|---|---|---|---|
| 1.1 | Reactor Coolant Pump Flywheels | Visual inspection of upper surface of top disc and bottom surface of bottom disc; volumetric inspection from circumference of all disc segments. | When motor is disassembled for maintenance purposes. |

Not Used

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.5 Containment Test

Applicability

Applies to containment leakage and structural integrity.

Objective

To verify that the:

(1) Locked closed manual containment isolation valves are closed and locked,
(2) potential leakage from containment is within acceptable limits, and
(3) structural performance of all important components in the containment prestressing system is acceptable.

Specifications

(1) Prior to the reactor going critical after a refueling outage, and at least once per 31 days thereafter, an administrative check will be made to confirm that all "locked closed" manual containment isolation valves, except for valves that are open under administrative control as permitted by Specification 2.6(1)a, are closed and locked. Valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position shall be verified closed during each cold shutdown except that such verification need not be performed more often than once per 92 days.

(2) Containment Integrated Leakage Rate Test (Type A Tests)

Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program.

(3) Containment Penetrations Leak Rate Tests (Type B Tests)

Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program for the following penetrations:

(i) Equipment Hatch
(ii) Personnel Air Lock
(iii) Mechanical Penetrations M-1 through M-99
(iv) Fuel Transfer Tube (Mechanical Penetration M-100)
(v) Electrical Penetrations:

| | | | | |
|---|---|---|---|---|
| A-1 | B-9 | D-6 | F-2 | E-HCV-383-3A |
| A-2 | B-10 | D-7 | F-4 | E-HCV-383-3B |
| A-4 | B-11 | D-8 | F-5 | E-HCV-383-4A |
| A-5 | C-1 | D-9 | F-6 | E-HCV-383-4B |
| A-6 | C-2 | D-10 | F-7 | |
| A-7 | C-4 | D-11 | F-8 | |
| A-8 | C-5 | E-1 | F-9 | |
| A-9 | C-6 | E-2 | F-10 | |

## 3.0 **SURVEILLANCE REQUIREMENTS**

3.5 Containment Tests (Continued)

| | | | |
|---|---|---|---|
| A-10 | C-7 | E-4 | F-11 |
| A-11 | C-8 | E-5 | G-1 |
| B-1 | C-9 | E-6 | G-2 |
| B-2 | C-10 | E-7 | G-3 |
| B-4 | C-11 | E-8 | G-4 |
| B-5 | D-1 | E-9 | H-1 |
| B-6 | D-2 | E-10 | H-2 |
| B-7 | D-4 | E-11 | H-3 |
| B-8 | D-5 | F-1 | H-4 |

(4) Containment Isolation Valves Leak Rate Tests (Type C Tests)

Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program for the following penetrations:

| | | | |
|---|---|---|---|
| M-2 | M-31 | M-52 | IA-3092 |
| M-7 | M-38 | M-53 | IA-3093 |
| M-8 | M-39 | M-57 | IA-3094 |
| M-11 | M-40 | M-58 | |
| M-14 | M-42 | M-69 | |
| M-15 | M-43 | M-73 | |
| M-18 | M-44 | M-74 | |
| M-19 | M-45 | M-79 | |
| M-20 | M-46 | M-80 | |
| M-22 | M-47 | M-86 | |
| M-24 | M-48 | M-87 | |
| M-25 | M-50 | M-88 | |
| M-30 | M-51 | M-89 | |
| | | M-HCV-383-3 | |
| | | M-HCV-383-4 | |

## 3.0 SURVEILLANCE REQUIREMENTS
### 3.5 Containment Tests (Continued)

Basis

The containment is designed for an accident pressure of 60 psig.[2] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a maximum temperature of about 120°F. With these initial conditions the temperature of the steam-air mixture at the peak accident pressure of 60 psig is 288°F.

Prior to initial operation, the containment was strength-tested at 69 psig and then was leak tested. The design objective of the pre-operational leakage rate test has been established as 0.1% by weight for 24 hours at 60 psig. This leakage rate is consistent with the construction of the containment, which is equipped with independent leak-testable penetrations and contains channels over all inaccessible containment liner welds, which were independently leak-tested during construction.

Safety analyses have been performed on the basis of a leakage rate of 0.1% of the free volume per day of the first 24 hours following the maximum hypothetical accident. With this leakage rate, a reactor power level of 1500 MWt, and with minimum containment engineered safety systems for iodine removal in operation (one air cooling and filtering unit), the public exposure would be well below 10 CFR Part 100 values in the event of the maximum hypothetical accident.[3] The performance of an integrated leakage rate test and performance of local leak rate testing of individual penetrations at periodic intervals during plant life provides a current assessment of potential leakage from the containment.

The reduced pressure (5 psig) test on the PAL is a conservative method of testing and provides adequate indication of any potential containment leakage path. The test is conducted by pressurizing between two resilient seals on each door. The test pressure tends to unseat the resilient seals which is opposite to the accident pressure that tends to seat the resilient seals. A periodic test ensures the overall PAL integrity at 60 psig.

The integrated leakage rate test (Type A test) can only be performed during refueling shutdowns.

## 3.0 SURVEILLANCE REQUIREMENTS
### 3.5 Containment Tests (continued)

The frequency of periodic integrated leakage rate tests is based on several major considerations: (1) There is a low probability of leaks in the liner because of the test of leak-tightness of the welds during erection and conformance of the complete containment to a low leak rate at 60 psig during pre-operational testing, which is consistent with 0.1% leakage at design basis accident conditions and absence of any significant stresses in the liner during reactor operation. (2) Periodic testing is conducted at full accident pressure, on those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves). A low value (0.60 $L_a$) of total leakage is specified as acceptable from penetrations and isolation valves. (3) The tendon stress surveillance program provides assurance that an important part of the structural integrity of the containment is maintained. (4) A review of leakage rates obtained during past containment integrated leakage rate testing is conducted to set appropriate frequency of performance. (5) Visual inspection of the containment structure is conducted every other refueling and prior to each Integrated Leakage Rate Test.

As left leakage prior to the first startup after performing a required leakage test is required to be $< 0.6\ L_a$ for combined Type B and C leakage, and $< 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis.

Integrity tests of the purge isolation valves are established to identify excessive degradation of the resilient seats of these valves. Simultaneous testing of redundant purge valves from a leak test connection accessible from outside containment provides adequate testing. The testing method is identical to the Type C purge isolation valve test performed in accordance with 10 CFR Part 50, Appendix J. For leakages found to be greater than 18,000 SCCM, repairs shall be initiated to ensure these valves meet the acceptance criteria.

References

(1) USAR, Section 5.9

(2) USAR, Section 5.1.1

(3) USAR, Section 14.15

## 3.0 **SURVEILLANCE REQUIREMENTS**

### 3.6 Safety Injection and Containment Cooling Systems Tests

Applicability

Applies to the safety injection system, the containment spray system, the containment cooling system and air filtration system inside the containment.

Objective

To verify that the subject systems will respond promptly and perform their intended functions, if required.

Specifications

(1) Safety Injection System

System tests shall be performed on a refueling frequency. A test safety feature actuation signal will be applied to initiate operation of the system. The safety injection and shutdown cooling system pump motors may be de-energized for this portion of the test.

A second overlapping test will be considered satisfactory if control board indication and visual observations indicate all components have received the safety feature actuation signal in the proper sequence and timing (i.e., the appropriate pump breakers shall have opened and closed, and all valves shall have completed their travel).

(2) Containment Spray System

a. System tests shall be performed on a refueling frequency. The test shall be performed with the isolation valves in the spray supply lines at the containment blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

b. At least every ten years the spray nozzles shall be verified to be open.

c. The test will be considered satisfactory if:

(i) Visual observations indicate that at least 264 nozzles per spray header have operated satisfactorily.

(ii) No more than one nozzle per spray header is missing.

d. Representative samples of Hydrated Sodium Tetraborate (NaTB) that have been exposed to the same environmental conditions as that in the mesh baskets shall be tested on a refueling frequency by:

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.6 Safety Injection and Containment Cooling Systems Tests (continued)

(i) Verifying that the containment sump buffering agent baskets contain a volume of granular NaTB that is within the area of acceptable operation of Figure 2-3.

(ii) Verifying that a sample from the buffering agent baskets provides adequate pH upward adjustment of the recirculation water.

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.6 Safety Injection and Containment Cooling Systems Tests (Continued)

(3) Containment Recirculating Air Cooling and Filtering System

a. The emergency mode dampers will be verified to be in their accident positions and the automatic valve, fan, and fusible link automatic damper operation will be checked for operability on a refueling surveillance interval.

b. Each fan required to function during accident conditions will be exercised monthly.

c. Each air filtering circuit will be operated at least 15 continuous minutes every month.

d. A visual examination of the HEPA and charcoal filters will be made on a refueling surveillance interval to ensure that leak paths do not exist.

e. Measurement of pressure drop across the HEPA filter bank shall be performed on a refueling surveillance interval to verify a pressure drop of less than 2 inches of water at system design flow. Measurement of pressure drop across the combined HEPA and charcoal adsorber banks shall be performed on a refueling surveillance interval to verify a pressure drop of less than 2.5 inches of water at system design flow.

f. The Containment Recirculating Air Cooling and Filtering Unit HEPA filters will be replaced at an interval not to exceed 10 years. The provisions of Technical Specification 3.0.1 do not apply.

g. Fans shall be shown to operate within +/-10% design flow on a refueling surveillance interval.

h. Containment Recirculating Air Cooling and Filtering Unit relief ports shall be exercised to verify operability on a refueling surveillance interval.

3.0 **SURVEILLANCE REQUIREMENTS**
3.6 Safety Injection and Containment Cooling Systems Tests (Continued)

Basis

The safety injection system and the containment cooling system are principal plant safeguards that are not operated during normal reactor operation.

Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes containment isolation and a containment spray system test requires the system to be temporarily disabled. The method of assuring operability of these systems is, therefore, to combine systems tests to be performed during refueling shutdowns in addition to more frequent component tests which can be performed during reactor operation.

The refueling shutdown tests demonstrate proper automatic operation of the safety injection and containment spray systems. A test signal is applied to initiate automatic action and verification made that the components receive the safety injection actuation signals in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.[(1) (2)]

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked daily and the initiating circuits are tested monthly. In addition, the active components (pumps and valves) are to be tested every three months to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The test interval of three months is based on the judgement that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), yet more frequent tests would result in increased wear over a long period of time. Verification

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.6 Safety Injection and Containment Cooling Systems Tests (Continued)

that the spray piping and nozzles are open will be made initially by a smoke test or other suitably sensitive method, and at appropriate intervals thereafter. A single containment spray header flow rate of 1885 gpm of atomized spray is required to provide the containment response[3] specified in Section 2.4 of the Technical Specification: To achieve the 1885 gpm flow rate, no greater than ten (10) spray nozzles may be inoperable of which no more than one may be missing. Since the material is all stainless steel, normally in a dry condition, with no plugging mechanism available, retesting at appropriate intervals is considered to be more than adequate.

Other systems that are also important to the emergency cooling function are the SI tanks, the component cooling system, the raw water system and the containment air coolers. The SI tanks are a passive safeguard. In accordance with the specifications, the water volume and pressure in the SI tanks are checked periodically. The other systems mentioned operate when the reactor is in operation and are continuously monitored for satisfactory performance.

The in-containment air treatment system is designed to filter the containment building atmosphere during accident conditions. Both in-containment air treatment systems are designed to automatically start upon accident signals. Should one system fail to start, the redundant system is designed to start automatically. Each of the two systems has 100 percent capacity.[4]

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter.

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.6 Safety Injection and Containment Cooling Systems Tests (Continued)

Operation of the system for 10 hours every month will demonstrate operability of the filters and adsorbers system and remove excessive moisture build-up on the adsorbers.

Demonstration of the automatic initiation capability will assure system availability.

Determination of the volume of buffering agent in containment must be performed due to the possibility of leaking valves and components in the containment building that could cause dissolution of the buffering agent during normal operation.

A refueling frequency shall be utilized to visually determine that the volume of buffering agent contained in the buffering agent baskets is within the area of acceptable operation based on the buffering agent volume required by Figure 2-3. A measured value or the Technical Data Book (TDB) II, "Reactivity Curves" may be used to obtain a hot zero power (HZP) critical boron concentration (CBC). The "as found" volume of buffering agent must be within the area of acceptable operation of Figure 2-3 using this HZP CBC value. Prior to exiting the refueling outage, visual buffering agent volume determination is performed to ensure that the "as-left" volume of buffering agent contained in the baskets is ≥ 112.9 $ft^3$. This requirement ensures that there is an adequate quantity of buffering agent to adjust the pH of the post-LOCA sump solution to a value ≥ 7.0 for HZP CBC up to 1800 ppm.

Testing must be performed to ensure the solubility and buffering ability of the NaTB after exposure to the containment environment. A representative sample of 1.39 to 1.42 grams of NaTB from one of the baskets in containment is submerged in 0.99 – 1.01 liters of water at a boron concentration of 2436 – 2456 ppm (equivalent to a RCS boron concentration of 1800 ppm - Figure 2-3) using boric acid. At a standard temperature of 115 – 125°F, without agitation, the solution must be left to stand for 4 hours. The liquid is then decanted and mixed, the temperature is adjusted to 75 – 79°F and the pH measured. At this point, the pH must be ≥ 7.0. The representative sample weight is based on the minimum required NaTB weight of 5418 pounds, less the quantity required to account for acidic radiolysis products (758 pounds), and maximum possible post-LOCA sump volume of 398,445 gallons, normalized to a 1.0 liter sample. At a manufactured density of 48.0 $lb_m/ft^3$, 5418 pounds corresponds to the minimum volume of 112.9 $ft^3$.

For dissolution testing, the boron concentration of the test water is representative of the maximum possible boron concentration corresponding to the maximum possible post-LOCA sump volume. The post-LOCA sump volume originates from the Reactor Coolant System (RCS), the Safety Injection Refueling Water Tank (SIRWT), the Safety Injection Tanks (SITs) and the Boric Acid Storage Tanks (BASTs). The maximum post-LOCA sump boron concentration is based on a cumulative boron concentration in the RCS, SIRWT, SITs and BASTs of 2446 ppm. The cumulative boron concentration is based on a maximum RCS HZP CBC with no Xenon at Beginning of Cycle conditions, SIRWT and SIT boron concentrations at maximum allowed values of 2,350 ppm and maximum BAST concentration of 4.5 % wt. Agitation of the test solution is prohibited since an adequate standard for the agitation intensity cannot be specified. The test time of 4 hours is necessary to allow time for the dissolved NaTB to naturally diffuse through the sample solution. In the post-LOCA containment sump, rapid mixing would occur, significantly decreasing the actual amount of time before the required pH is achieved. This would ensure achieving a pH ≥ 7.0 by the onset of recirculation after a LOCA.

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.6 Safety injection and Containment Cooling Systems Tests (Continued)

References

(1) USAR, Section 6.2
(2) USAR, Section 6.3
(3) USAR, Section 14.16
(4) USAR, Section 6.4

## 3.0 SURVEILLANCE REQUIREMENTS
### 3.7 Emergency Power System Periodic Tests

Applicability

Applies to periodic testing and surveillance requirements of the emergency power system.

Objective

To verify that the emergency power system will respond promptly and properly when required.

Specifications

The following tests and surveillance shall be performed as stated:

(1) Diesel generators:

a. Each diesel engine shall be started at least once per 31 days on a staggered basis. The engine shall be run with all protective devices operable. The test shall verify that:

i. The diesel starts and accelerates to idle speed. Following a warm-up period as recommended by the manufacturer, the diesel generator will be accelerated to rated speed and voltage.

However, at least once per 184 days in these surveillance tests, the diesel generator shall demonstrate that it can be started and accelerated to rated speed and voltage in less than or equal to 10 seconds without a prior warm-up.

The signal initiated to start the diesel shall be varied from one test to another to verify all manual and auto start circuits.(1)

ii. With the diesel running at rated speed and voltage, the generator shall be synchronized with the 4.16 KV bus and the diesel breaker manually closed from the electrical control board. The generator shall then be loaded to at least the continuous(2) KW rating and run for at least 60 minutes before being off-loaded and the diesel breaker tripped.

b. The auto-start initiating circuit for each diesel shall be tested prior to each plant startup if not done during the previous week.

c. Tests shall be conducted during each refueling outage to demonstrate the satisfactory overall automatic operation of each diesel system. This test shall be conducted by:(*)

(*) During PLANT OPERATING CYCLE 27 only, the maximum allowed surveillance test interval shall not exceed 28 months.

## 3.0 SURVEILLANCE REQUIREMENTS

3.7 Emergency Power System Periodic Tests (Continued)

i. Initiation of a simulated auto-start signal to verify that the diesel starts.

ii. Initiation of a simulated simultaneous loss of 4.16 KV supplies to bus 1A3 (1A4). Proper operation will be verified by observation of:

(1) De-energization of bus 1A3 (1A4).
(2) Load shedding from bus (both 4160 V and 480 V).
(3) Energization of bus 1A3 (1A4).
(4) Automatic sequence start of emergency load, and
(5) Operation of ≥ 5 minutes while its generator is loaded with the emergency load.

iii. Verification that emergency loads do not exceed the 2000-HR KW rating of the engine.[2]

d. Manual control of diesel generators and breakers shall also be verified during refueling shutdowns.

e. The fuel oil transfer pumps shall be verified to be operable each month.

(2) Station Batteries

a. Every month the voltage of each cell (to the nearest 0.01 volt), the specific gravity, and temperature of a pilot cell in each battery shall be measured and recorded.[3][4]

b. Every three months the specific gravity of each cell, the temperature reading of every fifth cell, and the amount of water added shall be measured and recorded. During the first refueling outage and every third refueling outage thereafter the batteries shall be subjected to a rated load discharge test.

c. At monthly intervals the third battery charger, which is capable of being connected to either of the two D.C. distribution buses, shall be paralleled in turn to each D.C. bus. In each case, load shall be transferred to this reserve battery charger by switching out the normal charger. The reserve charger shall be run on load for 30 minutes on each bus and the system shall finally be returned to normal.

## 3.0 **SURVEILLANCE REQUIREMENTS**

3.7 Emergency Power System Periodic Tests (Continued)

d. During refueling shutdowns the correct function of all D.C. emergency transfer switches shall be demonstrated by manual transfer of normal D.C. supply breakers at the 125 volt D.C. distribution panels.

(3) Emergency Lighting

The correct functioning of the emergency lighting system required for plant safe shutdown shall be verified at least once each year.

(4) 13.8 kV Transmission Line

The 13.8 kV transmission line will be energized and loaded to minimum shutdown requirements on a refueling frequency.

(5) Required Safety Related Inverters

The correct inverter output (voltage, frequency, and alignment to required 120 V a-c instrument buses) shall be verified weekly.

Basis

The emergency power system provides power requirements for the engineered safety features in the event of a DBA. Each of the two diesel generators is capable of supplying minimum required safety feature equipment from independent buses. This redundancy is a factor in establishing testing intervals. The monthly tests specified will demonstrate operability and load capacity of each diesel generator. These tests are conducted to meet the objectives of NRC Generic Letter 84-15 regarding the issue of reductions in cold fast starts. For this reason, the test verifying a 10 second start will be conducted from ambient conditions once per 184 days for each diesel. Other monthly tests will allow for manufacturer's recommended warm-up to reduce the mechanical stress and wear on the diesel engines. The fuel supply and various controls are continuously monitored and alarmed for off-normal conditions. Automatic starting on loss of off-site power and automatic load shedding, diesel connection, and loading will be verified on a refueling frequency. At the same intervals, capability will be verified for manual emergency control of these functions from the diesel and switch-gear rooms.

Considering system redundancy, the specified testing intervals for the station batteries should be adequate to detect and correct any malfunction before it can result in system malfunction. Batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails.

## 3.0 **SURVEILLANCE REQUIREMENTS**

### 3.7 Emergency Power System Periodic Tests (Continued)

Surveillance of the required safety related inverters verifies that they are functioning properly with the 120 V a-c buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the 120 V a-c buses. The weekly frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

References

(1) USAR, Section 7.3.4.2
(2) USAR, Section 8.4.1
(3) USAR, Section 8.3.4
(4) USAR, Section 8.4.2

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.8 Main Steam Isolation Valves

Applicability

Applies to periodic testing of the main steam isolation valves.

Objective

To verify the ability of the main steam isolation valves to close upon signal.

Specifications

The operation of the main steam isolation valves shall be tested during each refueling outage to demonstrate a closure time of four seconds or less under no-flow conditions.[1]

Basis

The main steam isolation valves serve to limit an excessive reactor coolant system cooldown rate and resultant reactivity insertion following a main steam break incident. Their ability to close upon signal will be verified at each scheduled refueling outage.

References

(1) USAR, Section 10.3

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.9 Auxiliary Feedwater System

Applicability

Applies to periodic testing requirements of the turbine-driven and motor-driven auxiliary feedwater pumps.

Objective

To verify the operability of the auxiliary feedwater (AFW) system and its ability to respond properly when required.

Specifications

(1) The position of valves necessary to ensure auxiliary feedwater flow to the steam generators shall be verified by a monthly inspection. Anytime maintenance is performed on the auxiliary feedwater system which alters valve alignments, an operator shall check that the affected AFW system valves are properly aligned, to ensure AFW flow to the steam generators, and a second operator shall independently verify proper valve alignment.

(2) The operability of the motor-driven auxiliary feedwater pump and the steam turbine-driven auxiliary feedwater pump shall be confirmed at least monthly.

(3) The operability of auxiliary feedwater pumps' steam generator level regulating valves HCV-1107A, HCV-1107B, HCV-1108A, HCV-1108B, and auxiliary feedwater cross-tie valve HCV-1384 shall be confirmed in accordance with the Inservice Testing Program.

(4) Verify the proper alignment of the required AFW flow paths by verifying flow from the emergency feedwater storage tank to each steam generator prior to raising the reactor coolant temperature ($T_c$) above 300°F whenever the unit has been in MODE 4 and/or 5 for greater than 30 days.

(5) On a refueling frequency:

a. Verify that each automatic valve in the flow path actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

b. Verify that each auxiliary feedwater pump starts as designed automatically upon receipt of each auxiliary feedwater actuation test signal.

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.9 Auxiliary Feedwater System (Continued)

Basis

The basic safety related function of the auxiliary feedwater (AFW) system is to provide an alternate source of feedwater to the steam generators in the event of a loss of main feedwater. The auxiliary feedwater system is designed so that a single active failure coincident with a loss of offsite power will not prevent a safe plant shutdown.

Specification 3.9(1)

The valve position verifications performed monthly will confirm the availability of an auxiliary feedwater flow path to the steam generators. Following AFW system maintenance, the affected valves are verified to be in the correct position.

Specification 3.9(2)

The testing of the auxiliary feedwater pumps every month will verify their operability by recirculating water to the emergency feedwater storage tank.

Proper functioning of the AFW steam turbine admission valve and the starting of the feedwater pump will demonstrate the integrity of the steam driven pump. Verification of correct operation will be made both from instrumentation within the main control room and direct visual observation of the pumps.

Specification 3.9(3)

Operating AFW system cross-tie valve HCV-1384 and regulating valves HCV-1107A, HCV-1107B, HCV-1108A and HCV-1108B one at a time in accordance with the Inservice Testing Program will confirm operability of the valves.

Specification 3.9(4)

This surveillance ensures that the AFW system is properly aligned by verifying flow from the emergency feedwater storage tank to each steam generator. The required flow paths are through valve HCV-1384 and valves HCV-1105/HCV-1106 using FW-6. The surveillance is performed prior to raising the reactor coolant temperature ($T_c$) above 300°F whenever the reactor coolant temperature has been below 210°F for greater than 30 days. OPERABILITY of the AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW system during a subsequent shutdown.

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.9 Auxiliary Feedwater System (Continued)

Basis (Continued)

Specification 3.9(5)

This surveillance ensures that the AFW pumps will start, and automatic valves will reposition, in the event of any accident or transient that generates an AFAS. The surveillance demonstrates that AFW pumps FW-6 and FW-10 start, and valves HCV-1107A/B and HCV-1108A/B actuate to their correct position automatically, on a test signal. The test is accomplished by utilizing overlapping tests to verify that the instrumentation, valves and motor driven pump are OPERABLE with the steam turbine-driven pump tested separately to verify its OPERABILITY. The refueling frequency is based on the need to perform this surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance were performed with the reactor at power. This frequency is acceptable, based on the design reliability and operating experience of the equipment.

References

(1) USAR, Section 9.4

(2) Technical Specification 2.5

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.10 Reactor Core Parameters

Applicability

Applies to reactor core parameters that affect shutdown margin, MTC, linear heat rate and DNB margin.

Objective

To require evaluation of reactor core parameters.

Specification

(1) Shutdown Margin

a. The shutdown margin shall be determined:

1. By verifying that the CEA group withdrawal is above the Transient Insertion Limits of Specification 2.10.2 whenever the reactor is in hot standby or power operation conditions at least once per shift, or

2. By considering the following factors whenever the reactor is in hot or cold shutdown at least once per day.

(i) Reactor coolant system boron concentration;

(ii) CEA position;

(iii) Reactor coolant system temperature;

(iv) Fuel burnup;

(v) Xenon concentration; and

(vi) Samarium concentration.

b. The overall core reactivity balance shall be compared to predicted values to demonstrate agreement with $\pm$ 1.0% $\Delta k/k$ at least once per 31 EFPD. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a cycle burnup of 2000 MWD/MTU after each refueling.

## 3.0 SURVEILLANCE REQUIREMENTS
### 3.10 Reactor Core Parameters (Continued)

(2) Moderator Temperature Coefficient

The MTC shall be determined at the following frequencies and power conditions during each fuel cycle:

1. Prior to initial operation above 5% of rated power, after each fuel loading.
2. At any power level within 500 MWD/T of initial operation after each refueling.
3. At any power level within ± 14 EFPD of reaching a rated power equilibrium boron concentration of 300 ppm.

(3) Regulating CEA Insertion Limits

a. The position of each regulating CEA group shall be determined to be above the Transient Insertion Limits at least once per shift.

b. The accumulated times during which the regulating CEA groups are inserted beyond the Steady State Insertion Limits but above the Transient Insertion Limits shall be determined once per day.

(4) Linear Heat Rate Monitoring Systems

a. The incore detector monitoring system may be used for monitoring the core power distribution provided that at least once per 31 days of accumulated power operation the incore detector alarms generated by the plant computer are verified to be valid and satisfy the requirements of the core distribution map.

b. The excore detector monitoring system may be used for monitoring the core power distribution by:

1. Verifying at least once per 31 days of accumulated power operation that the axial shape index, $Y_I$, monitoring limit setpoints are maintained within the allowable limits of the Limiting Condition for Operations for Excore LHR Monitoring Figure provided in the COLR, as adjusted by Specification 2.10.4(1).

(5) Maximum Radial Peaking Factor ($F_R^T$)

$F_R^T$ shall be determined to be within the COLR limit at the following intervals:

a. After each refueling and prior to operation above 70 percent of rated power.

b. At least once per 31 EFPDs of accumulated power operation.

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.10 Reactor Core Parameters (Continued)

(6) Azimuthal Power Tilt (Tq)

Whenever the core power is above 70% of rated power, the azimuthal power tilt shall be determined to be within its limits by calculating the tilt at least once every day using either:

a. The excore detectors with at least four safety channels operable, or

b. The incore detectors with at least two strings of three rhodium detectors per full core height quadrant operable.

(7) DNB Parameters

a. The cold leg temperature, pressurizer pressure, and axial shape index shall be verified to be within the limits of Section 2.10.4(5) at least once per shift.

b. The reactor vessel coolant total flow rate shall be determined to be within its limit by measurement at least once per month.

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.12 Radioactive Waste Disposal System

Applicability

Applies to the instrumentation used to determine hydrogen and oxygen concentrations in the waste gas decay tanks.

Objective

To ensure the concentrations of hydrogen and oxygen in the gaseous radioactive waste system are maintained below their flammability concentrations as required by Specification 2.9.

Specifications

The hydrogen and oxygen monitoring system for the waste gas decay tanks shall have a:

a. daily channel check (when in service)
b. monthly cross comparison with a grab sample
c. quarterly channel calibration using a gas mixture with concentrations in the range of interest

Basis

The specification ensures that instrumentation used to determine the concentration of potentially explosive gas mixtures entrained in the gas decay tank(s) will be maintained in an operable condition. Maintaining the instrumentation used to determine hydrogen and oxygen concentration with a surveillance program provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR 50.

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.16 Residual Heat Removal System Integrity Testing

Applicability

Applies to determination of the integrity of the residual heat removal (RHR) systems and associated components.

Objective

To verify that the leakage from the residual heat removal system components is within acceptable limits.

Specifications

(1) a. The portion of the shutdown cooling system that is outside the containment, and the piping between the containment spray pump suction and discharge isolation valves, shall be examined for leakage at a pressure no less than 300 psig. This shall be performed on a refueling frequency.

b. Piping from valves HCV-383-3 and HCV-383-4 to the suction isolation valves of the low pressure safety injection pumps and containment spray pumps and to the high pressure safety injection pumps shall be examined for leakage at a pressure no less that 82 psig. This shall be performed at the testing frequency specified in (1)a. above.[1]

c. The portion of the high pressure safety injection (HPSI) system that is located outside the containment and downstream of the HPSI pumps shall be examined for leakage when subjected to the discharge pressure of a HPSI pump operating in the minimum recirculation mode. This test shall be performed at the frequency specified in (1)a. above. The leakage contribution from this section shall be the observed leakage from this piping at the test pressure multiplied by the square root of the ratio 1500/P, where P is the test discharge pressure (in psig) of the operating HPSI pump.

d. An internal leakage test shall be performed on a refueling frequency. The test shall measure and quantify the leakage to the safety injection refueling water tank (SIRWT) from applicable water leakage paths.

e. Visual inspection of the system's components shall be performed at the frequency specified in (1)a. above to uncover any significant external leakage to atmosphere (including leakage from valve stems, flanges, and pump seals). The leakage shall be measured by collection and weighing or by any other equivalent method.

(2) a. The sum of leakages from section (1)a, (1)b, (1)c, and (1)d above shall not exceed 3800 cc/hour.

b. Repairs shall be made as required to maintain leakage within the acceptable limits.

[1] During PLANT OPERATING CYCLE 27 only, the maximum allowed surveillance test interval shall not exceed 28 months.

## 3.0 SURVEILLANCE REQUIREMENTS

### 3.16 Residual Heat Removal System Integrity Testing (Continued)

Basis

The limiting leakage to atmosphere from the RHR system (3800 cc/hour) is based upon a plant specific leak rate analysis for RHR system components operating after a design basis accident.

The test pressures for sections 3.16(1)a and 3.16(1)b, and the pressure correction factors in sections 3.16(1)c give adequate margins over the highest pressures within the lines after a design basis accident.[1]

A RHR system leakage of 3800 cc/hr will limit off-site exposures due to leakage to insignificant levels relative to those calculated for direct leakage from the containment in the design basis accident. The safety injection system pump rooms are equipped with individual charcoal filters which are placed into operation by means of switches in the control room. The radiation detectors in the auxiliary building exhaust duct are used to detect high radiation level. The 3800 cc/hr leak rate is sufficiently high to allow for reasonable leakage through the pump seals are valve packings, and yet small enough to be readily handled by the pumps and radioactive waste system. Leakage to the safety injection system pump room sumps will be returned to the spent regenerant tanks.[2] Additional makeup water to the containment sump inventory can be readily accommodated via the charging pumps from either the safety injection refueling water tank (SIRWT) or the concentrated boric acid storage tanks.

The anlaysis for the loss of coolant accident assumed a total (internal and external) leakage from all RHR systems sources of 3800 cc/hour. The internal leakage would leak back into the water remaining in the SIRWT.

References

(1) USAR, Section 9.3
(2) USAR, Section 6.2

## 3.0 **SURVEILLANCE REQUIREMENTS**

### 3.17 Steam Generator (SG) Tube Integrity

Applicability

Applies to in-service surveillance of steam generator tubes.

Objective

To ensure the integrity of the steam generator tubes.

Specifications

Each steam generator shall be demonstrated OPERABLE by performance of the following:

(1) Verify SG Tube Integrity in accordance with the Steam Generator Program.

(2) Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to exceeding 210°F reactor coolant temperature (Tcold).

(3) Reporting Requirements

A report shall be submitted within 180 days after exceeding 210°F reactor coolant temperature (Tcold) following completion of an inspection performed in accordance with the Specification 5.23, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

3.0 **SURVEILLANCE REQUIREMENTS**

3.17 Steam Generator (SG) Tube Integrity (Continued)

Basis

During shutdown periods the SGs are inspected as required by this Surveillance Requirement (SR) and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.17(1). The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.23 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 5.23 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to exceeding 210°F reactor coolant temperature ($T_{cold}$) following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

References

1. NEI 97-06, "Steam Generator Program Guidelines."

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 100.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

# 4.0 DESIGN FEATURES

## 4.1 Site

The site for Fort Calhoun Station Unit No. 1 is in Washington County, Nebraska, on the west bank of the Missouri River and approximately nineteen miles north, northwest of the city of Omaha, Nebraska. The exclusion area, as defined in 10 CFR Part 100, Section 100.3(a), consists of approximately 1242 acres. The exclusion area boundary extent includes approximately 660 acres in Washington County, Nebraska, owned by the Omaha Public Power District (OPPD), and 582 acres in Harrison County, Iowa, on the east bank of the river directly opposite the facility, on which the District retains perpetual easement rights. The minimum exclusion area boundary point is located approximately at the 187.0 degree radial from the outer wall of the containment building and at a distance of 910 meters.

## 4.2 Reactor Core

### 4.2.1 Fuel Assemblies

The reactor shall contain 133 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO®, or M5 clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide ($UO_2$) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

### 4.2.2 Control Element Assemblies

The reactor core shall contain 49 control element assemblies (CEAs). The control material shall be silver indium cadmium, boron carbide, or hafnium metal as approved by the NRC.

## 4.3 Fuel Storage

### 4.3.1 Criticality

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

a. Fuel assemblies having a maximum U-235 enrichment of 4.5 weight percent,
b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.5 of the USAR,

## 4.0 DESIGN FEATURES (Continued)

c. A nominal 8.6 inch center to center distance between fuel assemblies placed in Region 2, the high density fuel storage racks,

d. A nominal 9.8 inches (East-West) by 10.3 inches (North South) center to center distances between fuel assemblies placed in Region 1, the low-density fuel storage racks,

e. New or partially spent fuel assemblies with a discharge burnup in the "acceptable domain" of Figure 2-10 for "Region 2 Unrestricted" may be allowed unrestricted storage in any of the Region 2 fuel storage racks in compliance with Reference (1).

f. Partially spent fuel assemblies with a discharge burnup between the "acceptable domain" and "Peripheral Cells" of Figure 2-10 may be allowed unrestricted storage in the peripheral cells of the Region 2 fuel storage racks in compliance with Reference (1).

g. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable domain" of Figure 2-10 will be stored in Region 1 in compliance with Reference (1).

4.3.1.2 The new fuel storage rack is designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent,

b. A nominal 16 inch center to center distance between fuel assemblies placed in the storage rack.

4.3.1.3 The spent fuel casks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.5 weight percent,

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.5 of the USAR,

c. $k_{eff} \leq 0.95$ if fully flooded with borated water $\geq 800$ ppm, which includes an allowance for uncertainties as described in Section 9.5 of the USAR,

d. A nominal 9.075-inch center-to-center distance between fuel assemblies placed in the spent fuel cask,

e. Spent fuel assemblies with a combination of discharge burnup and initial average assembly enrichment in the "acceptable" range of Figure 2-11.

## 4.0 **DESIGN FEATURES** (Continued)

### 4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 23 ft.

### 4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1083 fuel assemblies.

References:

(1) Letter from R. Wharton (NRC) to T. Patterson (OPPD), Amendment 174 to Facility Operating License No. DPR-40, (TAC NO. M94789) Dated July 30, 1996, NRC-96-0126.

(2) Ft. Calhoun USAR, Reference 9.5-1

# 5.0 ADMINISTRATIVE CONTROLS

## 5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

## 5.2 Organization

5.2.1 Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the USAR.

b. The plant manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The corporate officer with responsibility for overall plant nuclear safety shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Plant Staff

The plant staff organization shall be as described in Chapter 12 of the USAR and shall function as follows:

a. The minimum number and type of licensed and unlicensed operating personnel required onsite for each shift shall be as shown in Table 5.2-1.

## 5.0 ADMINISTRATIVE CONTROLS

5.2 Organization (Continued)

b. An Operator or Technician qualified in Radiation Protection Procedures shall be onsite when fuel is in the reactor.

c. All core alterations shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator limited to fuel handling who has no other concurrent responsibilities during the operation.

d. Fire protection program responsibilities are assigned to those positions and/or groups designated by asterisks in USAR 12.1-1 through 12.1-4 according to the procedures specified in Section 5.8 of the Technical Specifications.

e. The Manager - Shift Operations, the Shift Managers, and the Unit Supervisors shall hold a senior reactor operator license. The Licensed Operators shall hold a reactor operator license.

5.3 Facility Staff Qualification

5.3.1 Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, with the exception of the Manager - Radiation Protection (MRP) and the Shift Technical Advisor (STA), the senior reactor operator licensees, and the reactor operator licensees, who shall meet the requirements set forth in Regulatory Guide 1.8, Revision 3, dated May 2000, entitled "Qualification and Training of Personnel for Nuclear Power Plants."

## TABLE 5.2-1

## MINIMUM SHIFT CREW COMPOSITION[ii]

| License Category | Core Alteration | Cold Shutdown or Refueling Shutdown | Operating or Hot Shutdown Modes |
|---|---|---|---|
| Senior Operator License | 2[i] | 1 | 2[iii] |
| Operator License | 2 | 1 | 2[iv] |
| Non-Licensed | (As required) | 1 | 2 |
| Shift Technical Advisor | None | None | 1 |

(i) This includes the individual with Senior Operator License supervising Core Alterations.

(ii) Shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 5.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewmember being late or absent.

(iii) At least one of these individuals must be in the control room at all times.

(iv) At least one of these individuals (or the second senior licensed operator, if both senior licensed operators are in the control room) must be present at the controls at all times.

## 5.4 Training

5.4.1 A retraining and replacement training program for the plant staff shall be maintained under the direction of the Manager - Training and shall meet or exceed the requirements of Section 6 of ANSI/ANS 3.1-1993, as modified by Regulatory Guide 1.8, Revision 3, dated May 2000 and 10 CFR Part 55.

## 5.5 Not Used

## 5.6 Not Used

## 5.0 ADMINISTRATIVE CONTROLS

5.7 Not used.

5.8 Procedures

5.8.1 Written procedures and administrative policies shall be established, implemented and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33; and

c. Not used.

d. All programs specified in Specification 5.11 through 5.24.

5.8.2 Temporary changes to procedures of 5.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant supervisory staff, at least one of whom holds a Senior Reactor Operator's License.

## 5.0 ADMINISTRATIVE CONTROLS

5.8 Procedures (Continued)

c. The change is documented, reviewed by a qualified reviewer and approved by either the plant manager or the department head designated by Administrative Controls Standing Orders as the responsible department head for that procedure within 14 days of implementation.

5.8.3 Written procedures shall be implemented which govern the selection of fuel assemblies to be placed in Region 2 of the spent fuel racks (Technical Specification 2.8). These procedures shall require an independent verification of initial enrichment requirements and fuel burnup calculations for a fuel bundle to assure the "acceptance" criteria for placement in Region 2 are met. This independent verification shall be performed by individuals or groups other than those who performed the initial acceptance criteria assessment, but who may be from the same organization.

5.9 Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the appropriate NRC Regional Office unless otherwise noted.

5.9.1 Not Used

## 5.0 ADMINISTRATIVE CONTROLS

5.9 Reporting Requirements (Continued)

5.9.2 Not Used

5.9.3 Special Reports

Special reports shall be submitted to the appropriate NRC Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification where appropriate:

a. In-service inspection report, reference 3.3.
b. Tendon surveillance, reference 5.21.
c. DELETED
d. DELETED
e. DELETED
f. DELETED
g. Materials radiation surveillance specimens reports, reference 3.3.
h. DELETED
i. Post-accident monitoring instrumentation, reference 2.21.
j. DELETED

## 5.0 **ADMINISTRATIVE CONTROLS**

5.9 Reporting Requirements (Continued)

5.9.4 Unique Reporting Requirements

a. Annual Radioactive Effluent Release Report

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year of operation shall be submitted before May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be 1) consistent with the objectives outlined in the ODCM and PCP, and 2) in conformance with 10 CFR 50.36a. and Section IV.B.1 of Appendix I to 10 CFR 50.

b. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Section IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR 50.

c. Not Used

5.9.5 Core Operating Limits Report (COLR)

a. Core Operating Limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

| | |
|---|---|
| 2.2.7 | Borated Water Sources - Shutdown |
| 2.2.8 | Borated Water Sources - Operating |
| 2.10.2 | Reactivity Control Systems and Core Physics Parameters Limits |
| 2.10.4 | Power Distribution Limits |
| 2.13 | RPS Limiting Safety System Settings, Table 2-11, Items 6, 8, and 9 |

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

5.0 **ADMINISTRATIVE CONTROLS**
5.9 Reporting Requirements (Continued)

1. OPPD-NA-8301, "Reload Core Analysis Methodology Overview" approved version as specified in the COLR.

2. OPPD-NA-8302, "Neutronics Design Methods and Verification", approved version as specified in the COLR.

3. OPPD-NA-8303, "Transient and Accident Methods and Verification", approved version as specified in the COLR.

4. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Report," April 1995 (Westinghouse Proprietary) as approved in the Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 178 to Facility Operating License No. DPR-40, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Docket No. 50-285, dated October 25, 1996.

5. XN-75-32(P)(A) Supplements 1, 2, 3, & 4, "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.

6. XN-NF-82-06(P)(A) and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

7. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

8. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.

9. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

10. BAW-10240(P)(A), "Incorporation of M5™ Properties in Framatome ANP Approved Methods," Framatome ANP, Inc., approved version as specified in the COLR.

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulics limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as shutdown margin (SDM), transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

## 5.0 ADMINISTRATIVE CONTROLS

5.9.6 Reactor Coolant System (RCS) Pressure - Temperature Limits Report (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature overpressure protection, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for Technical Specifications 2.1.1 and 2.1.2.

b. The analytical methods used in the PTLR shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. CE NPSD-683-A, Revision 6, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," April 2001.
2. WCAP-15443, Revision 0, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel," July 2000.
3. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Number 199 to Facility Operating License DPR-40 Omaha Public Power District Fort Calhoun Station, Unit Number 1, dated June 7, 2001.
4. CEN-636, Revision 2, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials, dated July 2000.
5. FC06876, Revision 0, "Performance of Low Temperature Overpressure Protection System Analyses Using RELAP5: Methodology Paper."
6. FC06877, "Low Temperature Overpressure Protection (LTOP) Analysis, Revision 1."
7. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Number 207 to Facility Operating License Number DPR-40 Omaha Public Power District Fort Calhoun Station, Unit Number 1, dated April 22, 2002.
8. Letter LTR-CI-01-25, Revision 0 from Westinghouse Electric Company (S.T. Byrne) to OPPD (J. Jensen), "Assessment of Extended Beltline Limit for Fort Calhoun Station Reactor Pressure Vessel," dated December 18, 2001.
9. WCAP-15741, Revision 0, "Reactor Vessel Surveillance Program Withdrawal Schedule Modifications," dated September 2001.
10. Letter from NRC (A. B. Wang) to Omaha Public Power District (R. T. Ridenoure), Fort Calhoun Station - Unit 1, Exemption from the Requirements of Appendix G to 10 CFR Part 50 (TAC No. MB8237), dated July 30, 2003.
11. Letter from Information Systems Laboratories (William Arcieri) to OPPD (J. Jensen), "WCA-09-2002: Transmittal of RELAP5/MOD3.2d," dated August 2, 2002.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period (i.e., the number of EFPY used in the P-T limit/LTOP analysis) and for any revision or supplement thereto.

## 5.0 ADMINISTRATIVE CONTROLS

### 5.10 Record Retention

5.10.1 Records shall be retained as described in the Quality Assurance Program.

### 5.11 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

5.11.1 In lieu of the "control device" required by paragraph 20.1601(a) of 10 CFR Part 20, and as an alternative method allowed under ' 20.1601(c), each high radiation area (as defined in ' 20.1601) in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by required issuance of a Radiation Work Permit.* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Manager-Radiation Protection (MRP) in the Radiation Work Permit.

5.11.2 The requirements of 5.11.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr** but less than 500 rads/hr*** (Locked High Radiation Area). In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or the MRP (or designee) with the following exception:

a. In lieu of the above, for accessible localized Locked High Radiation Areas located in large areas such as containment, where no lockable enclosure exists in the immediate vicinity to control access to the Locked High Radiation Area and no such enclosure can be readily constructed, then the Locked High Radiation Area shall be:

*Radiation Protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.
**At 30 centimeters (12 inches) from the radiation source or from any surface penetrated by the radiation.
***At 1 meter from the radiation source or from any surface penetrated by the radiation.

## 5.0 ADMINISTRATIVE CONTROLS

### 5.11 Radiation Protection Program (Continued)

i. roped off such that an individual at the rope boundary is exposed to 1000 mrem/hr or less,
ii conspicuously posted, and
iii a flashing light shall be activated as a warning device.

### 5.12 Environmental Qualification

Deleted

### 5.13 Secondary Water Chemistry

A secondary water chemistry monitoring program to inhibit steam generator tube degradation shall be implemented. This program shall be described in the station chemistry manual and shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

2. Identification of the procedures used to measure the values of the critical parameters;

3. Identification of process sampling points;

4. Procedures for the recording and management of data;

5. Procedures defining corrective actions for off control point chemistry conditions; and

6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

### 5.14 Systems Integrity

A program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels shall be implemented. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

## 5.0 ADMINISTRATIVE CONTROLS

### 5.15 Post-Accident Radiological Sampling and Monitoring

The following programs shall be implemented and maintained to ensure the capability to accurately monitor and/or sample and analyze radiological effluents and concentrations in a post-accident condition:

1. A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. (Any space which will require occupancy to permit an operator to aid in mitigation of, or recovery from, an accident is designated as vital.)

2. A program which will ensure the capability to obtain and analyze radioactive iodines and particulates in plant gaseous effluents.

These programs shall include the following:

1. Training of personnel.

2. Procedures for monitoring and/or sampling and analysis.

3. Provisions for maintenance of sampling and analysis equipment.

### 5.16 Radiological Effluents and Environmental Monitoring Programs

The following programs shall be established, implemented, and maintained.

#### 5.16.1 Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for control of radioactive effluents and for maintaining the doses to individuals in unrestricted areas from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the operability of radioactive liquid and gaseous radiation monitoring instrumentation including operability tests and setpoint determination in accordance with the methodology in the ODCM.

b. Limitations on the concentration of radioactive material, other than dissolved or entrained noble gases, released in liquid effluents to unrestricted areas conforming to ten times 10 CFR 20.1001-20.2401, Appendix B, Table 2, Column 2. For dissolved or entrained noble gases, the concentration shall be limited to 2.0 E-04 μCi/ml total activity.

## 5.0 ADMINISTRATIVE CONTROLS

5.16 Radiological Effluents and Environmental Monitoring Programs (Continued)

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM.

d. Limitations on the annual and quarterly doses or dose commitment to individuals in unrestricted areas from radioactive materials in liquid effluents released to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

e. Determination of cumulative doses from radioactive effluents for the current calendar quarter and current calendar year in accordance with the ODCM on a quarterly basis.

f. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity in plant effluents.

g. Limitations on the concentration resulting from radioactive material, other than noble gases, released in gaseous effluents to unrestricted areas conforming to ten times 10 CFR 20.1001-20.2401, Appendix B, Table 2, Column 1. For noble gases, the concentration shall be limited to five times 10 CFR 20.1001-20.2401, Appendix B, Table 2, Column 1.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

i. Limitations on the annual and quarterly doses to an individual beyond the site boundary from Iodine-131, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

j. Limitations on the annual dose or dose commitment to an individual beyond the site boundary due to releases or radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

5.16.2 Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

## 5.0 ADMINISTRATIVE CONTROLS

5.16 Radiological Effluents and Environmental Monitoring Programs (Continued)

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

b. A Land Use Census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required by the results of this census.

c. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

5.17 Offsite Dose Calculation Manual (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program. This documentation shall contain:

1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the Plant Operations Review Committee and the approval of the plant manager.

c. Temporary changes to the ODCM may be made in accordance with Technical Specification 5.8.2.

d. Shall be submitted to the Nuclear Regulatory Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed and shall indicate the date (e.g., month/year) the change was implemented.

5.18 DELETED

## 5.0 ADMINISTRATIVE CONTROLS

### 5.19 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995," as modified by the following exceptions:

(1) If the Personnel Air Lock (PAL) is opened during periods when containment integrity is not required, the PAL door seals shall be tested at the end of such periods and the entire PAL shall be tested within 14 days after RCS temperature $T_{cold}$ > 210EF.

(2) Type A tests may be deferred for penetrations of the steel pressure retaining boundary where the nominal diameter does not exceed one inch.

(3) Elapsed time between consecutive Type A tests used to determine performance shall be at least 24 months or refueling interval.

b. The containment design accident pressure ($P_a$) is 60 psig.

## 5.0 ADMINISTRATIVE CONTROLS

5.19 Containment Leakage Rate Testing Program (Continued)

c. The maximum allowable primary containment leakage rate, $L_a$, at $P_a$, shall be 0.1% of containment air weight per day.

d. Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is ≤ 1.0 $L_a$. During unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 $L_a$ Maximum Pathway Leakage Rate (MXPLR) for Type B and C tests and ≤ 0.75 $L_a$ for Type A tests.

2. Personnel Air Lock testing acceptance criteria are:

(i) Overall Personnel Air Lock leakage is ≤ 0.1 $L_a$ when tested at ≥ $P_a$.

(ii) For each PAL door, seal leakage rate is ≤ 0.01 $L_a$ when pressurized to ≥ 5.0 psig.

e. Containment Purge Valve (PCV-742A/B/C/D) testing acceptance criterion is:

For each Containment Purge Valve, leakage rate is < 18,000 SCCM when tested at ≥ $P_a$.

f. If at any time when containment integrity is required and the total Type B and C measured leakage rate exceeds 0.60 $L_a$ Minimum Pathway Leakage Rate (MNLPR), repairs shall be initiated immediately. If repairs and retesting fail to demonstrate conformance to this acceptance criteria within 48 hours, then containment shall be declared inoperable.

g. The provisions of Specification 3.0.1 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

h. The provisions of Specifications 3.0.4 and 3.0.5 are applicable to the Containment Leakage Rate Testing Program.

5.20 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license or

2. A change to the USAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

## 5.0 ADMINISTRATIVE CONTROLS

### 5.20 Technical Specifications (TS) Bases Control Program (Continued)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR.

d. Proposed changes that meet the criteria of 5.20.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

### 5.21 Containment Tendon Testing Program

This program provides controls for monitoring any tendon degradation in prestressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Containment Tendon Testing Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 3, 1990.

The provisions of TS 3.0.1 and TS 3.0.5 are applicable to the Containment Tendon Testing Program inspection frequencies.

If the acceptance criteria are not met, an immediate investigation shall be made to determine the cause(s) and extent of the non-conformance to the criteria, and the results shall be reported to the Commission within 90 days via a special report in accordance with Technical Specification 5.9.3.

### 5.22 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An API gravity or an absolute specific gravity within limits,
   2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
   3. A clear and bright appearance with proper color, or a water and sediment content within limits;

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil, and

c. Total particulate concentration of the fuel oil is $\leq$ 10 mg/l when tested every 31 days.

The provisions of TS 3.0.1 and TS 3.0.5 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

## 5.0 **ADMINISTRATIVE CONTROLS**

### 5.23 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary- to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 2.1.4, "Reactor Coolant System Operational Leakage."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

## 5.0 **ADMINISTRATIVE CONTROLS**

5.23 Steam Generator (SG) Program (continued)

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

## 5.0 **ADMINISTRATIVE CONTROLS**

### 5.24 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Ventilation System (CRVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by the CRVS, operating within the tolerance for design flow rate, at a Frequency of 18 months. The results shall be trended and used as part of an 18 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.1 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered leakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

## 6.0 INTERIM SPECIAL TECHNICAL SPECIFICATIONS

6.1 Limits on Reactor Coolant Pump Operation - Deleted, Am. No. 56

6.2 Use of Spent Fuel Shipping Cask - Deleted, Am. No. 57

6.3 Auxiliary Feedwater Automatic Initiation Setpoint - Deleted, Am. No. 65

6.4 Operation With Less than 75% of Incore Detector Strings Operable - Deleted - Am. No. 73

# Appendix B

## Additional Conditions

## Renewed Facility Operating License No. DPR-40

Omaha Public Power District shall comply with the following conditions on the schedules noted below:

| Amendment Number | Additional Conditions | Implementation Date |
|---|---|---|