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See Proposed Change To Tech Specs.

SUBJECT: Application for amend to License NPF-21,revising TS to relocate component lists in accordance w/GL 91-08 & selected implementation of Commission policy statement of TS improvements.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

P.O. Box 968 • 3000 George Washington Way • Richland, Washington 99352-0968 • (509) 372-5000

October 21, 1993
G02-93-256

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: **NUCLEAR PLANT NO. 2, OPERATING LICENSE NPF-21 REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATIONS TO RELOCATE COMPONENT LISTS IN ACCORDANCE WITH GENERIC LETTER 91-08 AND SELECTED IMPLEMENTATION OF THE COMMISSION'S POLICY STATEMENT ON TECHNICAL SPECIFICATIONS IMPROVEMENTS**

In accordance with the Code of Federal Regulations, Title 10, Parts 50.90 and 2.101, the Supply System hereby submits a request for amendment to the WNP-2 Technical Specifications. Specifically, the Supply System requests revision of the WNP-2 Technical Specifications consistent with the guidance of Generic Letter 91-08. These changes will implement the relocation of component lists from the Technical Specifications. Additionally, several of these component specific lists are part of Technical Specifications that do not meet the criteria for inclusion in the Technical Specifications as defined in the Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors as noticed in the Federal Register on July 22, 1993. These Technical Specification requirements are also not included in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," dated September 28, 1992. The identified Technical Specifications, as well as the component specific lists, are being relocated to a new plant procedure which will be controlled pursuant to the requirements of 10CFR50.59 and Technical Specification 6.8.1.

The relocation of these Limiting Conditions For Operation (LCOs) is made in accordance with the Commission policy that licensees may adopt portions of the improved Standard Technical Specifications (STS) without fully implementing all Technical Specification improvements as stated in the supplementary information provided with the Final Policy Statement in the Federal Register. The Commission also stated that LCOs which do not meet any of the four criteria cited below may be proposed for removal from the Technical Specifications and relocation to licensee-controlled documents, such as the FSAR. The criteria may be applied to either standard or custom Technical Specifications. The identified LCOs are proposed for relocation based on failure to meet the four criteria for inclusion in Technical Specifications.

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'11

The specific outline for Technical Specifications required to meet 10 CFR 50.36a is provided by Generic Letter 91-08. The guidelines provided by Generic Letter 91-08 and the Policy Statement are followed in this change request in that the component specific lists, and the relocated LCOs, will be included in a plant procedure as recommended in Generic Letter 91-08.

This Technical Specification amendment request includes the following proposed relocations from the Technical Specifications to a Supply System controlled plant procedure:

1) Relocation of Table 3.4.3.2-2, Reactor Coolant System Interface Valves Leakage Pressure Monitors, and the associated Action statement 3.4.3.2.d and Surveillance Requirement 4.4.3.2.3. This relocation is performed in accordance with the Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors. Relocation of Table 3.4.3.2-2 also meets the relocation guidance of Generic Letter 91-08. This change also deletes the note at the bottom of page 3/4 4-11 since this note was only applicable until the start of the first refueling outage.

2) Relocation of Table 3.6.3-1, Primary Containment Isolation Valves, to a plant procedure in accordance with the guidance of Generic Letter 91-08. This change includes deletion of references to this table. It also includes the addition to the Bases section the expectations for opening under administrative control locked or sealed containment isolation valves. These expectations have been modified from the wording included in Generic Letter 91-08 to provide additional clarification relative to plant specific details.

3) Relocation of Technical Specification 3.8.4.1, A.C. Circuits Inside Primary Containment, in accordance with the Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors. The listing of components in the Limiting Condition For Operation section of this Specification meets the purpose of Generic Letter 91-08 and thus could be relocated in accordance with the guidance of the Generic Letter.

4) Relocation of Technical Specification 3.8.4.2, Primary Containment Penetration Conductor Overcurrent Protective Devices, including Table 3.8.4.2-1, in accordance with the Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors. Relocation of Table 3.8.4.2-1 also meets the guidance for relocation per Generic Letter 91-08.

5) Relocation of Technical Specification 3.8.4.3, Motor-Operated Valves Thermal Overload Protection, including Table 3.8.4.3-1, in accordance with the Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors. Relocation of Table 3.8.4.3-1 also meets the guidance for relocation per Generic Letter 91-08.

The Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors contains four criteria against which Technical Specifications are to be evaluated for inclusion in the Improved Technical Specifications. Meeting any of the four criteria results in inclusion in the Improved Technical Specifications. These four criteria are:

> Criterion 1 - Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> Criterion 2 - A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or prevents a challenge to the integrity of a fission product barrier.
>
> Criterion 3 - A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient analysis that either assumes the failure of or prevents a challenge to the integrity of a fission product barrier.
>
> Criterion 4 - A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

As detailed below, the Supply System has evaluated each of the proposed changes identified above per the requirements of 10 CFR 50.92 and determined they do not represent an unreviewed safety question or a significant hazard.

With respect to the proposed relocation of Table 3.6.3-1, Primary Containment Isolation Valves, this change is made in accordance with the guidance provided in Generic Letter 91-08. The Supply System has evaluated this proposed change per the requirements of 10 CFR 50.92 and determined it does not represent an unreviewed safety question or a significant hazards consideration because it does not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The relocation of Table 3.6.3-1 from the Technical Specifications to a licensee controlled document is administrative in nature. The surveillance and operability requirements remain in the Technical Specifications. The Technical Specification restrictions and actions are not being revised or relaxed by this change. The accident analyses that rely on these components are not affected. Plant procedures will only be revised where necessary to reflect the relocation of the Table. Changes to the relocated valve information will be controlled and made in accordance with the administrative controls required by Technical Specification 6.8.1 and 10 CFR 50.59. Since this change does not affect the content, control, or adherence to the Technical Specification, the probability or consequences of previously evaluated accidents are not impacted.

2) Create the possibility of a new or different kind of accident from any previously evaluated. The relocation of information from Table 3.6.3-1 to a licensee controlled document serves to consolidate information on affected components. This information will be controlled under appropriate administrative requirements. This relocation is made in accordance with the guidance of Generic Letter 91-08. The proposed revision does not involve a change in the manner in which these valves will be operated, maintained, or tested. No components or systems are physically added, removed, or modified as a result of the proposed change. Therefore, a new or different kind of accident as a result of this change is not credible.

3) Involve a significant reduction in a margin of safety. The margin of safety associated with these valves is unaffected by this proposed change since the applicable operability and surveillance requirements are not being revised except administratively as recommended by the Generic Letter. No technical changes to valve operation or maintenance will result due to this change. The incorporation of this information into a new plant procedure will ensure changes are made in accordance with the controls of Technical Specification 6.8.1 and 10 CFR 50.59.

The relocation of Table 3.4.3.2-2, Reactor Coolant System Interface Valves Leakage Pressure Monitors, and the associated Action statement 3.4.3.2.d and Surveillance Requirement 4.4.3.2.3, and the relocation of Technical Specifications 3.8.4.1, A.C. Circuits Inside Primary Containment, 3.8.4.2, Primary Containment Penetration Conductor Overcurrent Protective Devices, and Technical Specification 3.8.4.3, Motor-Operated Valves Thermal Overload Protection, have been evaluated by the Supply System against the requirements of 10 CFR 50.92. It has been determined that these changes do not represent a significant hazards consideration since they do not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes are consistent with the Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors. Specifically, the Specifications being proposed for relocation to Supply System controlled documents do not meet any of the four criteria specified in the Policy Statement for equipment to be included in the Technical Specifications.

   Regarding the Reactor Coolant System Interface Valves Leakage Pressure Monitors, these instruments do not meet Criterion 1 since the Policy Statement specifically excludes "instrumentation to identify the source of actual leakage." These Pressure Monitors are designed to identify the source of actual leakage.

The relocation of these Technical Specification requirements to a new plant procedure does not change the requirements. The required testing and associated Actions for out of service equipment will continue to be met. The current restrictions and actions are not being revised or relaxed by this change. The accident analyses that rely on these components are not affected. Relocation results in licensee control of future changes under the requirements of 10 CFR 50.59. Removal of the note at the bottom of page 3/4 4-11 is administrative in nature since this note was applicable only until the first refueling outage. Thus, there is no significant increase in the probability or consequences of an accident previously evaluated as a result of these changes.

2) Create the possibility of a new or different kind of accident from any previously evaluated. This relocated information will be controlled under appropriate administrative requirements. This relocation is made in accordance with the guidance of the Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors. The proposed revisions do not involve a change in the manner in which the equipment will be operated, maintained, or tested, nor in the actions that will be taken should the equipment be out of service or incapable of performing their intended safety functions. No components or systems are physically added, removed, or modified as a result of the proposed change. Therefore, this change will not result in the possibility of a new or different type of accident than those previously evaluated.

3) Involve a significant reduction in a margin of safety. The requested changes are not the result of a physical change to the plant or the manner in which the plant will be operated. The accident analyses for the plant as described in the FSAR are not affected by this proposed change. The operation, maintenance, and testing of equipment is not affected. Therefore, the margin of safety for the plant is not significantly reduced as a result of these changes.

As discussed above, the Supply System considers that the proposed changes do not involve a significant hazards consideration, nor is there a potential for a change in the types or increase in the amount of any effluents that may be released offsite, nor do they involve an increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, per 10 CFR 51.22(b), an environmental assessment of these changes is not required.

The approved amendment will be implemented within thirty days of receipt. Numerous procedural changes will be required to delete the reference to the Technical Specification requirements that will be relocated by this request. Since the Technical Specification requirement relocation does not result in a change in requirements, the resulting reference changes in the procedures will be treated as administrative in nature and will be made at the next

available opportunity. Special procedure revisions to delete reference to the relocated Technical Specifications will thus not be made. As stated above, the specified testing will continue to be performed at the current periodicity with any future changes to these requirements made per the requirements of 10CFR50.59.

This Technical Specification change has been reviewed and approved by the WNP-2 Plant Operations Committee and the Supply System Corporate Nuclear Safety Review Board (CNSRB). In accordance with 10 CFR 50.91, the State of Washington has been provided a copy of this letter.

Very truly yours,

J.V. Parrish,
Assistant Managing Director, Operations

DAS/ds
Attachments

cc: BH Faulkenberry - NRC RV
NS Reynolds - Winston & Strawn
RG Waldo - EFSEC
JW Clifford - NRR
DL Williams - BPA
NRC Site Inspector - 901A

[illegible]

STATE OF WASHINGTON)
)
COUNTY OF BENTON)

Subject: Amendment to Tech Specs to Relocate Component List

I, J. V. PARRISH, being duly sworn, subscribe to and say that I am the Assistant Managing Director, Operations for the WASHINGTON PUBLIC POWER SUPPLY SYSTEM, the applicant herein; that I have the full authority to execute this oath; that I have reviewed the foregoing; and that to the best of my knowledge, information, and belief the statements made in it are true.

DATE 20 Oct, 1993

J. V. Parrish, Assistant Managing Director
Operations

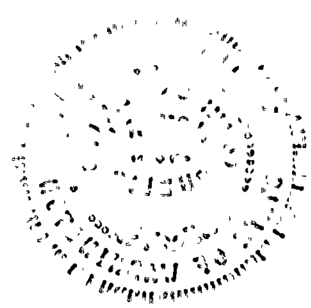
On this date personally appeared before me J. V. PARRISH, to me known to be the individual who executed the foregoing instrument, and acknowledged that he signed the same as his free act and deed for the uses and purposes herein mentioned.

GIVEN under my hand and seal this 20th day of October 1993.

Traci M. Hardy
Notary Public in and for the
STATE OF WASHINGTON

Residing at Kennewick, Washington

My Commission Expires August 9, 1995



INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIST OF TABLES (Continued)

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LIST OF TABLES (Continued)

OPERABLE - OPERABILITY

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

OPERATIONAL CONDITION - CONDITION

1.29 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.30 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation as (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

Change added b[illegible] 2-93-180 d[illegible] July 9, 1993

1.31 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall, or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.32 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position. except ~~as provided in Table 3.6.3-1 of~~ for valves that are open under administrative control as permitted by Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows, or O-rings, is OPERABLE.

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TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TABLE NOTATIONS

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes the diesel generator starting and sequence loading delays assumed in the accident analysis.

(b) Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

*Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

**Isolation system instrumentation response time for associated valves except MSIVs.

#Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time ~~shown in Table 3.6.3-1 and 3.6.5.2-1 for valves~~ in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

##This response time does not include the 45-second time delay.

insert

for each power operated or automatic primary containment isolation valve and secondary containment ventilation system automatic isolation valve (Table 3.6.5.2-1)

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REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 24-hour or less period.

d. 25 gpm total leakage averaged over any 24-hour period.

e. 1 gpm leakage at a reactor coolant system pressure of 950 ± 10 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b. and/or d. above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed (manual or deactivated automatic) (or check*) valve, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Change requested per GO2-93-076 dated April 1, 1993

DELETED

d. ~~With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-1 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

e. With any reactor coolant system UNIDENTIFIED LEAKAGE increase greater than 2 gpm within any 24-hour or less period, identify the source of leakage increase as not service sensitive Type 304 or 316 austenitic stainless steel within 4 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*Which has been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the primary containment atmospheric particulate and gaseous radioactivity at least once per 12 hours.

b. Monitoring the primary containment sump flow rate at least once per shift, not to exceed 12 hours.

c. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months.

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

~~4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints per Table 3.4.3.2-2 by performance of a:~~

~~a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and~~

~~b. CHANNEL CALIBRATION at least once per 18 months.~~

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TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	SYSTEM
HPCS-V-4	HPCS
HPCS-V-5	HPCS
LPCS-V-5	LPCS
LPCS-V-6	LPCS
*RCIC-V-66	RCIC
*RCIC-V-13	RCIC
RHR-V-8	RHR
RHR-V-9/209	RHR
*RHR-V-23	RHR
RHR-V-41A, B, C	RHR
RHR-V-42A, B, C	RHR
RHR-V-50A/123A, 50B/123B	RHR
RHR-V-53A, B	RHR

delete stars

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES LEAKAGE PRESSURE MONITORS

INSTRUMENT NUMBER	FUNCTION	ALARM SETPOINT (psig)
HPCS-PIS-3 (E22-N003)	HPCS Pump Suction Pressure High	≤ 80
LPCS-PIS-5 (E21-N005)	LPCS Pump Discharge Pressure High	≤ 442
RCIC-PS-21 (E51-N021)	RCIC Pump Suction Pressure High	≤ 91
RHR-PIS-22A, B, C (E12-N022)	RHR Pump Discharge Pressure to RPV High	≤ 475
RHR-PS-18 (E12-N018)	RHR Pump Shutdown Cooling Suction Pressure High	≤ 168

Delete Table

*The 18 month leakage test for these valves is deferred until the first refueling outage.

Delete

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3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

deleted by G02-93-180 dated July 9, 1993

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at P_a, 34.7 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

b. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except ~~as provided in Table 3.6.3-1 of~~ for valves that are open under administrative control as permitted by Specification 3.6.3.

c. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

---

*See Special Test Exception 3.10.1.

**Except valves, blind flanges, and deactivated atuomatic valves which are within the primary containment or other areas administratively controlled to prohibit access for reasons of personnel safety (i.e., radiation and temperature) and are locked, sealed, or otherwise secured in the closed position (1½ inch and smaller valves connected to vents, drains or test connections must be closed but need not be sealed). Valves inside containment shall be verified closed following primary containment de-inerting, but verification is not required more often than once per 92 days. Valves in other administratively controlled areas shall be verified closed during each COLD SHUTDOWN, but verification is not required more often than once per 31 days.

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PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.50 percent by weight of the containment air per 24 hours at P_a, 34.7 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and all valves ~~listed in Table 3.6.3-1~~, (except for main steam line isolation valves*, and valves which are hydrostatically leak tested ~~per Table 3.6.3-1~~,) subject to Type B and C tests when pressurized to P_a, 34.7 psig.

c. *Less than or equal to 11.5 scf per hour for any one main steam line isolation valve when tested at P_t, 25.0 psig.

d. A combined leakage rate of less than or equal to 1 gpm times the total number of ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a, 38.2 psig.

deleted by G02-93-180 dated July 9, 1993

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a, or

b. The measured combined leakage rate for all penetrations and all valves ~~listed in Table 3.6.3-1~~, (except for main steam line isolation valves*, and valves which are hydrostatically leak tested) ~~per Table 3.6.3-1~~, subject to Type B and C tests exceeding 0.60 L_a, or

c. The measured leakage rate exceeding 11.5 scf per hour for any one main steam line isolation valve, or

d. The measured combined leakage rate for all ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves,

restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 L_a, and

b. The combined leakage rate for all penetrations and all valves, ~~listed in Table 3.6.3-1~~ (except for main steamline isolation valves* and valves which are hydrostatically leak tested) ~~per Table 3.6.3-1~~, subject to Type B and C tests to less than or equal to 0.60 L_a, and

*Exemption to Appendix J of 10 CFR Part 50.

SURVEILLANCE REQUIREMENTS (Continued)

d. Type B and C tests shall be conducted with gas at P_a, 34.7 psig,* at intervals no greater than 24*** months except for tests involving:

1. Air Locks

2. Main steam line isolation valves,

3. Valves pressurized with fluid from a seal system,

4. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and

5. Purge supply and exhaust isolation valves with resilient seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

deleted by G02-93-180 dated July 9, 1993

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

g. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a, 38.2 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

h. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

i. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirements 4.6.1.8.2 and 4.6.1.8.3.

j. The provisions of Specification 4.0.2 are not applicable to 24-month or 40 ± 10-month surveillance intervals.

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*Unless ~~a hydrostatic test is required per Table 3.6.3-1.~~ hydrostatically tested.

***For those tests conducted during refueling outages, the 24-month interval may be exceeded by no more than 3 months.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

Each

3.6.3 ~~The~~ primary containment isolation valve~~s~~ and ~~the~~ reactor instrumentation line excess flow check valve~~s~~ ~~shown in Table 3.6.3-1~~ shall be OPERABLE. ~~with isolation times less than or equal to those shown in Table 3.6.3-1.~~

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more ~~of the~~ primary containment isolation valves ~~, shown in Table 3.6.3-1~~ inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

1. Restore the inoperable valve(s) to OPERABLE status, or
2. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or
3. Isolate each affected penetration by use of at least one closed manual valve or blind flange*, and
4. The provisions of Specification 3.0.4 are not applicable provided that the affected penetration is isolated in accordance with ACTION a.2. or a.3. above, and provided that the associated system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are performed.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one or more ~~of the~~ reactor instrumentation line excess flow check valves ~~shown in Table 3.6.3-1~~ inoperable, operation may continue and the provisions of Specifications 3.0.3 and 3.0.4 are not applicable provided that within 4 hours either;

1. The inoperable valve is returned to OPERABLE status, or
2. The instrument line is isolated and the associated instrument is declared inoperable.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

---

*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve ~~shown in Table 3.6.3-1~~ shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve ~~shown in Table 3.6.3-1~~ shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power-operated or automatic isolation valve ~~shown in Table 3.6.3-1~~ shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation line excess flow check valve ~~shown in Table 3.6.3-1~~ shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow at greater than a 10 psid differential pressure in hydraulic service and 15 psid differential pressure in pneumatic service.

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing the explosive squib from at least one explosive valve, such that each explosive squib in each explosive valve will be tested at least once per 90 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No explosive squib shall remain in use beyond the expiration of its shelf-life and operating life, as applicable.

Page 3/4 6-21 through 3/4 6-33 DELETED
.Next Page is 3/4 6-34

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

DELETE

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
a. Automatic Isolation Valves		
Main Steam Isolation Valves	1	5*
MS-V-22A,B,C,D(b)		
MS-V-28A,B,C,D(b)		
Main Steam Line Drains	1	
MS-V-16		25
MS-V-19		25
MS-V-67A,B,C,D(b)		15
Reactor Recirc. Cooling Sample Valves	2	5
RRC-V-19		
RRC-V-20		
Containment Purge Exhaust & Supply#	3	
CEP-V-1A,2A,3A,4A		4
CEP-V-1B,2B,3B,4B		4
CSP-V-1		4
CSP-V-2		4
CSP-V-3		4
CSP-V-4		4
CSP-V-93		4
CSP-V-96		4
CSP-V-97		4
CSP-V-98		4

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
a. Automatic Isolation Valves (Continued)		
Equipment Drain (Radioactive)	4	15
EDR-V-19 EDR-V-20		
Floor Drain (Radioactive)	4	15
FDR-V-3 FDR-V-4		
Fuel Pool Cooling/Suppression Pool Cleanup	4	35
FPC-V-149 FPC-V-153(f) FPC-V-154(f) FPC-V-156		
Reactor Recirculation Hydraulic Control(e)	4	15
HY-V-17A,B HY-V-18A,B HY-V-19A,B HY-V-20A,B HY-V-33A,B HY-V-34A,B HY-V-35A,B HY-V-36A,B		
Traversing Incore Probe	4	5
TIP-V-1,2,3,4,5		
TIP-V-15		

DELETE

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
a. Automatic Isolation Valves (Continued)		
Reactor Closed Cooling	4	60
RCC-V-5		
RCC-V-21		
RCC-V-40		
RCC-V-104		
Radiation Monitoring Supply & Return	4	5
PI-VX-250		
PI-VX-251		
PI-VX-253		
PI-VX-256		
PI-VX-257		
PI-VX-259		
Residual Heat Removal		
RHR-V-123A,B(g)	5	15
RHR-V-8(g)(k)	6	40
RHR-V-9(g)	6	40
RHR-V-23(g)	6	90
RHR-V-53A,B(g)	6	40
RHR-V-24A,B(c)	10	270
RHR-V-21	10	270
RHR-V-27A,B(c)	10	36
Reactor Water Cleanup System	7	
RWCU-V-1(d)		30(j)
RWCU-V-4		21(j)

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
a. Automatic Isolation Valves (Continued)		
Reactor Core Isolation Cooling		
RCIC-V-8	8	13(j)
RCIC-V-63	8	16(j)
RCIC-V-76	8	22
Low Pressure Core Spray		
LPCS-V-12	10	180
High Pressure Core Spray		
HPCS-V-23	11	180
b. Excess Flow Check Valves(e)		
Containment Atmosphere		N.A.
PI-EFC-X29d		
PI-EFC-X29f		
PI-EFC-X30a		
PI-EFC-X30f		
PI-EFC-X42c		
PI-EFC-X42f		
PI-EFC-X61c		
PI-EFC-X62b		
PI-EFC-X69f		
PI-EFC-X78a		

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
b. Excess Flow Check Valves (e) (Continued)		
Containment Atmosphere (Continued)		N.A.
PI-EFC-X66		
PI-EFC-X67		
PI-EFC-X82b		
PI-EFC-X84a		
PI-EFC-X86A,B		
PI-EFC-X87A,B		
PI-EFC-X119		
Reactor Pressure Vessel		N.A.
PI-EFC-X18A,B,C,D		
PI-EFC-X37e,f		
PI-EFC-X38a,b,c,d,e,f		
PI-EFC-X39a,b,d,e		
PI-EFC-X40c,d		
PI-EFC-X41c,d		
PI-EFC-X42a,b		
PI-EFC-X44Aa,Ab,Ac,Ad,Ae,Af,Ag,Ah,Aj,Ak,Al,Am		
PI-EFC-X44Ba,Bb,Bc,Bd,Be,Bf,Bg,Bh,Bj,Bk,Bl,Bm		
PI-EFC-X61a,b		
PI-EFC-X62c,d		
PI-EFC-X69a,b,e		
PI-EFC-X70a,b,c,d,e,f		
PI-EFC-X71a,b,c,d,e,f		
PI-EFC-X72a		
PI-EFC-X73a		
PI-EFC-X74a,b,e,f		

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
b. Excess Flow Check Valves (e) (Continued)		
Reactor Pressure Vessel (Continued)		N.A.
PI-EFC-X75a,b,c,d,e,f PI-EFC-X78b,c,f PI-EFC-X79a,b PI-EFC-X106 PI-EFC-X107 PI-EFC-X108 PI-EFC-X109 PI-EFC-X110 PI-EFC-X111 PI-EFC-X112 PI-EFC-X113 PI-EFC-X114 PI-EFC-X115	DELETE	
Other		N.A.
PI-EFC-X40e,f PI-EFC-X41e,f		
c. Manual Containment Isolation Valves		
Demineralized Water		N.A.
DW-V-156 DW-V-157		
Containment Air System		N.A.
CAS-VX-82e CAS-V-730		

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
c. Manual Containment Isolation Valves (Continued)		
Service Air		N.A.
SA-V-109		
Residual Heat Removal		N.A.
RHR-V-11A,B RHR-V-120 RHR-V-121 RHR-V-124A,B RHR-V-125A,B		
Reactor Core Isolation Cooling		N.A.
RCIC-V-64 RCIC-V-742(g)(b)		
Air Supply to Testable Check Valves		N.A.

Air Supply	Check Valve
PI-VX-42d PI-VX-216	RHR-V-50A
PI-VX-69c PI-VX-221	RHR-V-50B
PI-VX-61f PI-VX-219	RHR-V-41A
PI-VX-54Bf PI-VX-218	RHR-V-41B.

DELETE

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

DELETE

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
c. Manual Containment Isolation Valves (Continued)		
Air Supply to Testable Check Valves (Continued)		N.A.
PI-VX-62f RHR-V-41C		
PI-VX-220		
LPCS-V-66 LPCS-V-6		
LPCS-V-67		
HPCS-V-65 HPCS-V-5		
HPCS-V-68		
RCIC-V-184 RCIC-V-66		
RCIC-V-740		
d. Other Containment Isolation Valves		
Main Steam Leakage Control(b)		N.A.
MSLC-V-3A,B,C,D		
Reactor Feedwater/RWCU Return		N.A.
RFW-V-10A,B		
RFW-V-32A,B		
RFW-V-65A,B		
RWCU-V-40		
High Pressure Core Spray		N.A.
HPCS-V-4(g)(b)		
HPCS-V-5(g)(b)		
HPCS-V-12		
HPCS-V-15(f)(b)		

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
d. Other Containment Isolation Valves (Continued)		
High Pressure Core Spray (Continued)		N.A.
HPCS-RV-14(e)(h)		
HPCS-RV-35(e)(h)		
Low Pressure Core Spray		N.A.
LPCS-V-1(f)(b)		
LPCS-V-5(g)(b)		
LPCS-V-6(g)(b)		
LPCS-RV-18(e)(h)		
LPCS-RV-31(e)(h)		
LPCS-FCV-11		
Standby Liquid Control		N.A.
SLC-V-7		
SLC-V-4A,B		
Reactor Core Isolation Cooling		N.A.
RCIC-V-13(g)(b)		
RCIC-V-19		
RCIC-V-28		
RCIC-V-31(f)(b)		
RCIC-V-40		
RCIC-V-66(g)(b)		
RCIC-V-68		
RCIC-V-69		

DELETE

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
d. Other Containment Isolation Valves (Continued)		
Residual Heat Removal/Low Pressure Injection		N.A.
RHR-V-4A,B,C(f)(b)		
RHR-V-16A,B		
RHR-V-17A,B		
RHR-V-41A,B(g)(b)		
RHR-V-42A,B,C(g)(b)		
RHR-V-50A,B(g)(b)		
RHR-V-73A,B		
RHR-V-134A,B(c)		
RHR-V-209(g)(b)		
RHR-RV-1A,B(e)(h)		
RHR-RV-5(e)(h)		
RHR-RV-25A,B,C(e)(h)		
RHR-RV-30(e)(h)		
RHR-RV-36(e)(h)		
RHR-RV-88A,B,C(e)(h)		
RHR-FCV-64A,B,C		
Containment Atmosphere Control(c)(i) (H_2 Recombiner)		N.A.
CAC-V-2		
CAC-FCV-2A,B		
CAC-V-15		
CAC-FCV-1A,B		
CAC-V-11		

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

DELETE

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
d. Other Containment Isolation Valves (Continued)		
Containment Atmosphere Control(c)(i) (H_2 Recombiner) (Continued)		N.A.
CAC-V-6		
CAC-V-4		
CAC-FCV-4A,B		
CAC-V-13		
CAC-V-17		
CAC-FCV-3A,B		
CAC-V-8		
CSP-V-5		
CSP-V-6		
CSP-V-7		
Containment Purge System		N.A.
CSP-V-8		
CSP-V-9		
CSP-V-10		
Reactor Recirculation (Seal Injection)		N.A.
RRC-V-13A,B		
RRC-V-16A,B		
Containment Instrument Air		N.A.
CIA-V-20		
CIA-V-21		

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
d. Other Containment Isolation Valves (Continued)		
Containment Instrument Air (Continued)		N.A.
CIA-V-30A,B		
CIA-V-31A,B		
Post-Accident Sampling System(c)		N.A.
PSR-V-X73-1		
PSR-V-X73-2		
PSR-V-X77A1		
PSR-V-X77A2		
PSR-V-X77A3		
PSR-V-X77A4		
PSR-V-X80-1		
PSR-V-X80-2		
PSR-V-X82-1		
PSR-V-X82-2		
PSR-V-X82-7		
PSR-V-X82-8		
PSR-V-X83-1		
PSR-V-X83-2		
PSR-V-X84-1		
PSR-V-X84-2		
PSR-V-X88-1		
PSR-V-X88-2		

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PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
d. Other Containment Isolation Valves (Continued)		
Radiation Monitoring		N.A.
PI-EFCX-72f		
PI-EFCX-73e		
Transversing Incore Probe System		N.A.
TIP-V-6		
TIP-V-7,8,9,10,11(e)		

DELETE

TABLE NOTATIONS

*But greater than 3 seconds.
#Provisions of Technical Specification 3.0.4 are not applicable.

(a) See Technical Specification 3.3.2 for the isolation signal(s) which operate each group.
(b) Valve leakage not included in sum of Type B and C tests.
(c) May be opened on an intermittent basis under administrative control.
(d) Not closed by SLC actuation signal.
(e) Not subject to Type C Leak Rate Test.
(f) Hydraulic leak test at 38.2 psig.
(g) Not subject to Type C test. Test per Technical Specification 4.4.3.2.2
(h) Tested as part of Type A test.
(i) May be tested as part of Type A test. If so tested, Type C test results may be excluded from sum of other Type B and C tests.
(j) Reflects closure times for containment isolation only.
(k) During operational conditions 1, 2 & 3 the requirement for automatic isolation does not apply to RHR-V-8. Except that RHR-V-8 may be opened in operational conditions 2 & 3 provided control is returned to the control room, with the interlocks reestablished, and reactor pressure is less than 135 psig.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

A.C. CIRCUITS INSIDE PRIMARY CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.4.1 At least the following A.C. circuits inside primary containment shall be deenergized*:

a. Circuits supplied by breakers 2AR and 8AR, MCC E-MC-8C.

b. Circuits supplied by panel E-LP-6BAG.

c. Circuits supplied by panel E-LP-3DAG.

d. Circuits supplied by breakers in cubicles 2BL, 1D, and 2CR of MC-3DA.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) in the specified panel(s) within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the above required A.C. circuits shall be determined to be deenergized at least once per 24 hours** by verifying that the associated circuit breakers are in the tripped condition.

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*Except during entry into the drywell.
**Except at least once per 31 days if locked, sealed, or otherwise secured in the tripped condition.

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PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 All primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.2-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.2-1 inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system and:

   1. For 6.9 kV circuit breakers, de-energize the 6.9 kV circuit(s) by tripping the associated redundant circuit breaker(s) within 72 hours and verify the redundant circuit breaker to be tripped at least once per 7 days thereafter.

   2. For 480 volt circuit breakers, remove the inoperable circuit breaker(s) from service by removing the fuses within 72 hours and verify the fuses associated with the inoperable breaker(s) to be removed at least once per 7 days thereafter.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable to overcurrent devices in 6.9 kV circuits which have their redundant circuit breakers tripped or to 480 volt circuits which have the fuses associated with the inoperable circuit breaker removed.

SURVEILLANCE REQUIREMENTS

4.8.4.2 Each of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.2-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

   1. By verifying that the medium voltage, 6.9 kV, circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level and performing:

      a) A CHANNEL CALIBRATION of the associated protective relays, and

      b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and overcurrent control circuits function as designed.

      c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

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SURVEILLANCE REQUIREMENTS (Continued)

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value equal to 300% of the pickup of the longtime delay trip element and 150% of the pickup of the short time delay trip element, and verifying that the circuit breaker operates within the time delay bandwidth for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current equal to ±20% of the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

DELETE

TABLE 3.8.4.2-1

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

EQUIPMENT	PRIMARY PROTECTION		BACKUP PROTECTION	
a. 6900V Circuit Breakers				
RRC-P-1A	E-CB-RRA (Relay)		E-CB-S5 (Relay)	E-CB-N2/5 (Relay)
RRC-P-1B	E-CB-RRB (Relay		E-CB-S6 (Relay)	E-CB-N2/6 (Relay)
b. 480VAC Fused Disconnects				
MS-V-16	MC-8B-A	Fused	MC-8B	Fused
RWCU-V-1	MC-8B-A	Fused	MC-8B	Fused
RHR-V-9	MC-8B-A	Fused	MC-8B	Fused
RCIC-V-63	MC-8B-A	Fused	MC-8B	Fused
RCC-V-40	MC-8B-A	Fused	MC-8B	Fused
RHR-V-123B	MC-8B-A	Fused	MC-8B	Fused
RCIC-V-76	MC-8B-A	Fused	MC-8B	Fused
RHR-V-123A	MC-8B-A	Fused	MC-8B	Fused

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ELECTRICAL POWER SYSTEMS

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.3 The thermal overload protection of each valve shown in Table 3.8.4.3-1 shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves inoperable, continuously bypass the inoperable thermal overload within 8 hours or declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.3 The thermal overload protection for the above required valves shall be demonstrated OPERABLE at least once per 18 months and following maintenance on the motor starter by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of all thermal overloads for the above required valves.

DELETE

TABLE 3.8.4.3-1

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

DELETE

	VALVE NUMBER	SYSTEM(S) AFFECTED
a.	CAC-V-2 CAC-V-4 CAC-V-6 CAC-V-8 CAC-V-11 CAC-V-13 CAC-V-15 CAC-V-17	Containment Atmospheric Control System
b.	CIA-V-20 CIA-V-30A CIA-V-30B	Containment Instrument Air System
c.	FPC-V-149 FPC-V-153 FPC-V-154 FPC-V-156 FPC-V-172 FPC-V-173 FPC-V-175 FPC-V-181A FPC-V-181B FPC-V-184	Fuel Pool Cooling System
d.	HPCS-V-1 HPCS-V-4 HPCS-V-10 HPCS-V-11 HPCS-V-12 HPCS-V-15 HPCS-V-23	High Pressure Core Spray System
e.	LPCS-V-1 LPCS-V-5 LPCS-FCV-11 LPCS-V-12	Low Pressure Core Spray System
f.	MS-V-1 MS-V-2 MS-V-5 MS-V-16 MS-V-19 MS-V-20 MS-V-67A MS-V-67B MS-V-67C MS-V-67D MS-V-146	Main Steam System
g.	MSLC-V-1A MSLC-V-1B MSLC-V-1C MSLC-V-1D MSLC-V-2A MSLC-V-2B MSLC-V-2C MSLC-V-2D MSLC-V-3A MSLC-V-3B MSLC-V-3C MSLC-V-3D MSLC-V-4 MSLC-V-5 MSLC-V-9 ** MSLC-V-10	Main Steam Isolation Valve Leakage Control System
h.	RCC-V-5 RCC-V-21 RCC-V-40 RCC-V-104 RCC-V-129 RCC-V-130 RCC-V-131	Reactor Closed Cooling Water System
i.	RCIC-V-1 RCIC-V-8 RCIC-V-10 RCIC-V-13 RCIC-V-19 RCIC-V-22 RCIC-V-31	Reactor Core Isolation Cooling System

TABLE 3.8.4.3-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

	VALVE NUMBER	SYSTEM(S) AFFECTED		VALVE NUMBER	SYSTEM(S) AFFECTED
i.	RCIC-V-45	Reactor Core Isolation Cooling System		RHR-V-42C	
	RCIC-V-46			RHR-V-47A	
	RCIC-V-59			RHR-V-47B	
	RCIC-V-63			RHR-V-48A	
	RCIC-V-68			RHR-V-48B	
	RCIC-V-69			RHR-V-49	
	RCIC-V-76			RHR-V-53A	
	RCIC-V-110			RHR-V-53B	
	RCIC-V-113			RHR-V-64A	
				RHR-V-64B	
				RHR-V-64C	
j.	RFW-V-65A	Reactor Feedwater System		RHR-V-68A	
	RFW-V-65B			RHR-V-68B	
				RHR-V-73A	
				RHR-V-73B	
k.	RHR-V-3A	Residual Heat Removal System		RHR-V-74A	
	RHR-V-3B			RHR-V-74B	
	RHR-V-4A			RHR-V-115	
	RHR-V-4B			RHR-V-116	
	RHR-V-4C			RHR-V-123A	
	RHR-V-6A			RHR-V-123B	
	RHR-V-6B			RHR-V-134A	
	RHR-V-8			RHR-V-134B	
	RHR-V-9				
	RHR-V-16A		l.	RRC-V-16A	Reactor Recirculation System
	RHR-V-16B			RRC-V-16B	
	RHR-V-17A				
	RHR-V-17B				
	RHR-V-21				
	RHR-V-23		m.	RWCU-V-1	Reactor Water Cleanup System
	RHR-V-24A			RWCU-V-4	
	RHR-V-24B			RWCU-V-40	
	RHR-V-27A				
	RHR-V-27B				
	RHR-V-40				
	RHR-V-42A				
	RHR-V-42B				

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TABLE 3.8.4.3-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

	VALVE NUMBER	SYSTEM(S) AFFECTED
n.	SGT-V-1A SGT-V-1B SGT-V-3A1 SGT-V-3A2 SGT-V-3B1 SGT-V-3B2 SGT-V-4A1 SGT-V-4A2 SGT-V-4B1 SGT-V-4B2 SGT-V-5A1 SGT-V-5A2 SGT-V-5B1 SGT-V-5B2	Standby Gas Treatment System
o.	AS-V-68A AS-V-68B	Auxiliary Steam System
p.	SW-V-2A SW-V-2B SW-V-4A SW-V-4B SW-V-4C SW-V-12A SW-V-12B SW-V-24A SW-V-24B SW-V-24C SW-V-29 SW-V-44 SW-V-54 SW-V-75A SW-V-75B SW-V-90 SW-V-187A SW-V-187B SW-V-188A SW-V-188B	Standby Service Water System

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ELECTRICAL POWER SYSTEMS

REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.4 Two RPS electric power monitoring channels for each inservice RPS MG set or alternate source shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electric power monitoring channel for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring channel to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

b. With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring channel to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.4 The above specified RPS power monitoring channels instrumentation shall be determined OPERABLE:

a. By performance of a CHANNEL FUNCTIONAL TEST each time the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed within the previous 6 months.

b. At least once per 18 months by demonstrating the OPERABILITY of overvoltage, undervoltage and underfrequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

   1. Overvoltage $\leq$ 132 VAC,

   2. Undervoltage $\geq$ 108 VAC, and

   3. Underfrequency $\geq$ 57 Hz.

DEPRESSURIZATION SYSTEMS (Continued)

changed by G02-93-180 dated: July 9, 1993

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 90°F results in a water temperature of approximately 135°F immediately following blowdown which is below the 200°F used for complete condensation via quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus, there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak bulk temperature of the suppression pool is maintained below 200°F during any period of relief valve operation with sonic conditions at the discharge exit for quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety/relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety/relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety/relief valve to assure mixing and uniformity of energy insertion to the pool.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within ~~the~~ their respective time limits ~~specified~~ ensures for those isolation valves designed to close automatically that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

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BASES

Insert new wording From attached page

3/4.6.4 VACUUM RELIEF

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell and between the reactor building and suppression chamber. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are nine pairs of valves to provide redundancy and capacity so that operation may continue indefinitely with no more than two pairs of vacuum breakers inoperable in the closed position.

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system once per 18 months, along with the surveillance of the doors, hatches, dampers, and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Continuous operation of the system with the heaters OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

The OPERABILITY of the systems required for the detection and control of hydrogen gas ensures that these systems will be available to maintain the hydrogen concentration within the primary containment below its flammable limit during post-LOCA conditions. Either drywell and suppression chamber hydrogen recombiner system is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. The hydrogen control system is consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," September 1976.

This section shall be moved to a new page B 3/4 6-6 on the reverse side of this page

PRIMARY CONTAINMENT ISOLATION VALVES (Continued)

The opening of locked or sealed closed (i.e. manual) 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 valves, (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.

Editing Note: The above information shall be inserted at the top of page B 3/4 6-5.

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8.2.1-1 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Delete

~~Primary containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers by periodic surveillance.~~

~~The surveillance requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturers brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.~~

~~The bypassing of the motor-operated valve thermal overload protection continuously or during accident conditions ensures that the thermal overload protection will not prevent safety-related valves from performing their function. The surveillance requirements for demonstrating the bypassing of the thermal overload protection continuously and during accident conditions are in accordance with Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.~~

Insert

The RPS electric power monitoring system isolates the RPS bus from the motor-generator set or alternate power source in the event of overvoltage, undervoltage, or underfrequency. This system protects the RPS components against unacceptable voltage or frequency conditions. Isolation of the RPS power supplies is the fail-safe condition.