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10 CFR 50.90

July 14, 2016
GO2-16-086

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 APPLICATION TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-545, REVISION 3**

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Energy Northwest is submitting a request for an amendment to the Technical Specifications (TS) for Columbia Generating Station (Columbia). The proposed change revises the TS to eliminate Section 5.5.6, "Inservice Test Program." A new defined term, "Inservice Testing Program," is added to the TS Definitions section. This request is consistent with TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR Usage Rule Application to Section 5.5 Testing."

By letter dated December 9, 2014, (Agency wide Documents Access and Management System (ADAMS) Accession No. ML14337A449), the U.S. Nuclear Regulatory Commission (NRC) approved Energy Northwest's request to adopt Code Case OMN-20, "Inservice Test Frequency," under Relief Request RG01 for the fourth 10-year Inservice Testing (IST) interval at Columbia.

Attachment 1 provides a description and assessment of the proposed TS changes. Attachment 2 is a cross-reference between the Columbia TS or Surveillance Requirements (SRs) included in this amendment request versus the standard TS in NUREG-1434 and, in one case NUREG-1433. Attachment 3 provides the existing TS pages marked up to show the proposed changes. Attachment 4 provides revised (clean) TS pages. Attachment 5 provides TS Bases pages marked up to show the associated TS Bases changes and is provided for information only.

Energy Northwest requests approval of the proposed License Amendment one year from the date of submittal, with the amendment being implemented within 60 days of approval.

In accordance with 10 CFR 50.91, a copy of this application with attachments is being provided to the designated Washington State Official.

There are no new commitments being made by this submission.

If you should have any questions regarding this submittal, please contact Ms. L. L. Williams, Licensing Supervisor, at 509-377-8148.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on this 14th day of July, 2016.

Respectfully,

FOR A.L. JAVORIK

A. L. Javorik
Vice President, Engineering

Attachments: As stated

cc: NRC Region IV Administrator
NRC NRR Project Manager
NRC Sr. Resident Inspector - 988C
CD Sonoda - BPA - 1399 (w/o enclosures)
WA Horin - Winston & Strawn (email)
RR Cowley - WDOH (email)
JO Luce - EFSEC (email)
EFSECutc.wa.gov-- EFSEC (email)

DESCRIPTION AND ASSESSMENT

1.0 DESCRIPTION

The proposed change eliminates the Technical Specifications (TS), Section 5.5.6, "Inservice Test (IST) Program," to remove requirements duplicated in American Society of Mechanical Engineers (ASME) Code for Operations and Maintenance of Nuclear Power Plants (OM Code), Case OMN-20, "Inservice Test Frequency." A new defined term, "Inservice Testing Program," is added to TS Section 1.1, "Definitions". The proposed change to the TS is consistent with Technical Specification Task Force (TSTF) TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR Usage Rule Application to Section 5.5 Testing".

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

Energy Northwest has reviewed the model safety evaluation provided in the Federal Register Notice of Availability dated March 28, 2016, to the Technical Specifications Task Force in a letter dated December 11, 2015, (Nuclear Regulatory Commission (NRC) Agency wide Documents Access and Management System (ADAMS) Accession No. ML15314A305). This review included a review of the NRC staff's evaluation, as well as the information provided in TSTF-545, Revision 3. Energy Northwest concluded that the justifications presented in TSTF-545, Revision 3, and the model safety evaluation prepared by the NRC staff, are applicable to Columbia Generating Station (Columbia) and justify this amendment for the incorporation of the changes to Columbia's TS.

Columbia was issued a construction permit on March 19, 1973, and the provisions of 10 CFR 50.55a(f)(2) are applicable.

2.2 Variations

The proposed amendment is consistent with the Standard Technical Specifications (STS) changes described in TSTF-545, Revision 3; however, Energy Northwest proposes non-substantive variations from TSTF-545, as identified below. These variations do not affect the applicability of TSTF-545 or the NRC staff's model safety evaluation to the proposed license amendment.

Attachment 2 provides a cross-reference between the Columbia TS or Surveillance Requirements (SRs) included in this amendment request versus the NUREG-1434 and, in one case NUREG-1433, TS or SRs included in TSTF-545. Attachment 2 includes a listing of the referenced Columbia/TSTF-545 TS or SRs and is provided for information purposes only and is not intended to be a verbatim description of the TS or the SR. This cross-reference highlights the following items.

a. Specifically, Columbia did not renumber the section numbers in TS 5.5. Instead, Section 5.5.6," Inservice Testing Program," is revised as "Deleted”. Therefore, the LCO 3.0.6 discussion of 5.5.11 Safety Function Determination Program (SFDP) is not changed.

b. TSs or SRs included in TSTF-545 and corresponding Columbia TSs or SRs with identical SR numbers. These are not deviations from TSTF-545.

c. SRs included in TSTF-545 and corresponding Columbia SRs with differing SR numbers. These are administrative deviations from TSTF-545 with no impact on the NRC staff's model safety evaluation.

d. SRs included in TSTF-545 that are not contained in the Columbia TS. These are not applicable to Columbia and are identified by use of "NA" in the Columbia Surveillance Requirement Number column. This is an administrative deviation from TSTF-545 with no impact on the NRC staff's model safety evaluation.

e. Columbia plant-specific SR that is not contained in the TSTF-545 markups. This plant-specific SR is identified by use of "NA" in the Equivalent NUREG-1434 (or 1433) Revision 4 Surveillance Requirement Number column of Attachment 2. Energy Northwest has determined that the Columbia plant-specific SR is consistent with the intent of TSTF-545, Revision 3, and with the NRC staff's model safety evaluation. The frequency for this SR in Columbia’s TS is “in accordance with the Inservice Testing Program.” Therefore it is appropriate to include this SR within the scope of this change.

In addition, in some cases for both items a and b above, the wording of the actual Columbia SR is not identical to that of the NUREG-1434 or NUREG-1433 TS SR. However, in these cases, the intent of the Columbia SR is the same as the NUREG-1434 or NUREG-1433 SR. Therefore, these are administrative deviations from TSTF-545 with no impact on the NRC staff's model safety evaluation.

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Analysis

Energy Northwest requests adoption of the Technical Specification (TS) changes described in TSTF-545, "TS Inservice Testing Program Removal and Clarify SR Usage Rule Application to Section 5.5 Testing," which is an approved change to the Improved Standard Technical Specifications (ISTS), into the Columbia Generating Station (Columbia) TS. The proposed change revises the TS Chapter 5, "Administrative Controls," Section 5.5, "Programs and Manuals," to delete the "Inservice Testing (IST) Program" specification. Requirements in the IST Program are removed, as they are duplicative of requirements in the American Society of Mechanical Engineers (ASME) Operations and Maintenance (OM) Code, as clarified by Code Case OMN-20, "Inservice Test Frequency." By letter dated December 9, 2014, (Agency wide Documents Access and Management System (ADAMS)(Accession No. ML14337A449), the Nuclear Regulatory Commission (NRC) approved the adoption of approved Code Case OMN-20, “Inservice Test Frequency,” under Relief Request RG01 for the fourth 10-year Inservice Testing (IST) interval at Columbia. Other requirements in Section 5.5 are eliminated because the NRC has determined their appearance in the TS is contrary to regulations. A new defined term, "Inservice Testing Program," is added, which references the requirements of Title 10 of the Code of Federal Regulations (10 CFR), Part 50, paragraph 50.55a(f). Energy Northwest has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change revises TS Chapter 5, "Administrative Controls," Section 5.5, "Programs and Manuals," by eliminating the "Inservice Testing Program" specification. Most requirements in the Inservice Testing Program are removed, as they are duplicative of requirements in the ASME OM Code, as clarified by Code Case OMN-20, "Inservice Test Frequency," which has been approved for use at Columbia. The remaining requirements in the Section 5.5 IST Program are eliminated because the NRC has determined their inclusion in the TS is contrary to regulations. A new defined term, "Inservice Testing Program," is added to the TS, which references the requirements of 10 CFR 50.55a(f).

   Performance of inservice testing is not an initiator to any accident previously evaluated. As a result, the probability of occurrence of an accident is not significantly affected by the proposed change. Inservice test frequencies under Code Case OMN-20 are equivalent to the current testing period allowed by the TS with the exception that testing frequencies greater than 2 years may be extended by up to 6 months to facilitate test scheduling and consideration of plant operating conditions that may not be suitable for performance of the required testing. The testing frequency extension will not affect the ability of the components to mitigate any accident previously evaluated as the components are required to be operable during the testing period extension. Performance of inservice tests utilizing the allowances in OMN-20 will not significantly affect the reliability of the tested components. As a result, the availability of the affected components, as well as their ability to mitigate the consequences of accidents previously evaluated, is not affected.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?

   Response: No

   The proposed change does not alter the design or configuration of the plant. The proposed change does not involve a physical alteration of the plant; no new or different kind of equipment will be installed. The proposed change does not alter the types of inservice testing performed. In most cases, the frequency of inservice testing is unchanged. However, the frequency of testing would not result in a new or different kind of accident from any previously evaluated since the testing methods are not altered.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed change eliminates some requirements from the TS in lieu of requirements in the ASME Code, as modified by use of Code Case OMN-20. Compliance with the ASME Code is required by 10 CFR 50.55a. The proposed change also allows inservice tests with frequencies greater than 2 years to be extended by 6 months to facilitate test scheduling and consideration of plant operating conditions that may not be suitable for performance of the required testing. The testing frequency extension will not affect the ability of the components to respond to an accident as the components are required to be operable during the testing period extension. The proposed change will eliminate the existing TS SR 3.0.3 allowance to defer performance of missed inservice tests up to the duration of the specified testing frequency, and instead will require an assessment of the missed test on equipment operability. This assessment will consider the effect on a margin of safety (equipment operability). Should the component be inoperable, the Technical Specifications provide actions to ensure that the margin of safety is protected. The proposed change also eliminates a statement that nothing in the ASME Code should be construed to supersede the requirements of any TS. The NRC has determined that statement to be incorrect. However, elimination of the statement will have no effect on plant operation or safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Energy Northwest concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.0 ENVIRONMENTAL EVALUATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

5.0 IMPACT ON PREVIOUS SUBMITTALS

The proposed change affects TS pages that currently have pending amendments. The following TS pages have the associated amendments pending. The marked up pages in this submittal do not include proposed TS changes currently pending before the NRC.

TITLE	PAGES	SUBMITTED	IMPACT
LAR to adopt TSTF-425	3.1.7-2, 3.4.3-1, 3.4.4-2, 3.5.1-4, 3.5.2-3, 3.6.1.3-7 & 8, 3.6.1.6-2, 3.6.2.3-2, and 3.6.4.2-3	March 17, 2015	The LAR to adopt TSTF-425 relocates certain Surveillance Frequencies to a licensee-controlled program. None of the SRs proposed to be modified pursuant to TSTF-425 are included or affected by this request to adopt TSTF-545. Therefore, these two changes are not considered "linked" submittals.
LAR Modify Technical Specification Surveillance Requirement (SR) 3.4.3.1 and SR 3.4.4.1 Safety/Relief Valves (SRVs) Setpoint Lower Tolerance	3.4.3-1 and 3.4.4-2	May 10, 2016	The LAR to revise SR 3.4.3.1 and 3.4.4.1 only affects the Surveillance description and not the Frequency. This request to adopt TSTF-545 only affects the Frequency. Therefore, these two changes are not considered "linked" submittals.

Columbia TS Section or Surveillance Requirement Number	**Equivalent NUREG-1434 Revision 4 Section or Surveillance Requirement Number**	**Columbia TS Wording**	**Section or Surveillance Requirement modified by TSTF-545 (Yes/No)**	**Section or Surveillance Requirement modified in Columbia License Amendment**
1.1 Definitions	1.1 Definitions	NA	Yes	Yes
LCO 3.0.6	LCO 3.0.6	…..evaluations and limitations may be required in accordance with Specification 5.5.11, "Safety Function Determination Program (SFDP)." …	Yes	No Section 5.5.6 was marked deleted versus renumbering section 5.5 so no change to 3.0.6 required
SR 3.1.7.6	SR 3.1.7.7	In accordance with the Inservice Testing Program	Yes	Yes
SR 3.4.3.1	NA	In accordance with the Inservice Testing Program	No	Yes Columbia SR 3.4.3.1 is an additional SR as the Safety Relief Valve (SRV) Limiting Condition for Operation (LCO) is split into 2 LCOs 3.4.3 and 3.4.4. This SR is the same as SR 3.4.4.1, which is included in TSTF-545.
SR 3.4.4.1	SR 3.4.4.1	In accordance with the Inservice Testing Program	Yes	Yes
SR 3.4.6.1	SR 3.4.6.1	In accordance with the Inservice Testing Program	Yes	Yes

Columbia TS Section or Surveillance Requirement Number	**Equivalent NUREG-1434 Revision 4 Section or Surveillance Requirement Number**	**Columbia TS Wording**	**Section or Surveillance Requirement modified by TSTF-545 (Yes/No)**	**Section or Surveillance Requirement modified in Columbia License Amendment**
SR 3.5.1.4	SR 3.5.1.4	In accordance with the Inservice Testing Program	Yes	Yes
SR 3.5.2.5	SR 3.5.2.5	In accordance with the Inservice Testing Program	Yes	Yes
SR 3.6.1.3.5	SR 3.6.1.3.5	In accordance with the Inservice Testing Program	Yes	Yes
SR 3.6.1.3.6	SR 3.6.1.3.7	In accordance with the Inservice Testing Program	Yes	Yes
SR 3.6.1.6.2	SR 3.6.1.7.2 (NUREG-1433)	In accordance with the Inservice Testing Program	No - NUREG lists 92 days or in accordance with the Surveillance Frequency Control Program	Yes – Columbia's Frequency is in accordance with the IST Program. The TS Bases for NUREG-1433 SR 3.6.1.7.2 states that the 92 day frequency is based on the Inservice Testing Program requirements. The markup in TSTF-545 capitalizes this phrase in the TS Bases.

Columbia TS Section or Surveillance Requirement Number	Equivalent NUREG-1434 Revision 4 Section or Surveillance Requirement Number	Columbia TS Wording	Section or Surveillance Requirement modified by TSTF-545 (Yes/No)	Section or Surveillance Requirement modified in Columbia License Amendment
NA	SR 3.6.1.7.2		Yes	No. - 3.6.1.7, RHR Containment Spray System not in Columbia's TSs
SR 3.6.2.3.2	SR 3.6.2.3.2	In accordance with the Inservice Testing Program	Yes	Yes
SR 3.6.4.2.2	SR 3.6.4.2.2	In accordance with the Inservice Testing Program	Yes	Yes
NA	SR 3.6.5.3.4		Yes	No - 3.6.5.3, Drywell Isolation Valves not in Columbia's TS
5.5.6	5.5.7	5.5.6 Inservice Testing Program	Yes	Yes – remainder of section 5.5 was not renumbered
5.5.7 – 5.5.17	5.5.8 – 5.5.17		Yes	No – Columbia did not renumber the 5.5 sections

PROPOSED TECHNICAL SPECIFICATIONS CHANGES (MARK-UPS)

1.1 Definitions

END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine throttle valve limit switch or from when the turbine governor valve hydraulic control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR50.55a(f).
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. LEAKAGE into the drywell 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;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE; and

d. Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.3	Verify continuity of explosive charge.	31 days
SR 3.1.7.4	Verify the concentration of boron in solution is within the limits of Figure 3.1.7-1.	31 days AND Once within 24 hours after water or boron is added to solution AND Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-1
SR 3.1.7.5	Verify each SLC subsystem manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.1.7.6	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1220 psig.	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM
SR 3.1.7.7	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (SRVs) - ≥ 25% RTP

LCO 3.4.3 The safety function of 12 SRVs shall be OPERABLE, with two SRVs in the lowest two lift setpoint groups OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required SRVs inoperable.	A.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the required SRVs are as follows: Number of SRVs / Setpoint (psig) 2 / 1165 ± 34.9 4 / 1175 ± 35.2 4 / 1185 ± 35.5 4 / 1195 ± 35.8 4 / 1205 ± 36.1	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM
SR 3.4.3.2	Verify each required SRV opens when manually actuated.	24 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify the safety function lift setpoints of the required SRVs are as follows: Number of SRVs / Setpoint (psig) 2 / 1165 ± 34.9 4 / 1175 ± 35.2 4 / 1185 ± 35.5 4 / 1195 ± 35.8 4 / 1205 ± 36.1	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM
SR 3.4.4.2	-------------------------------NOTE----------------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify each required SRV opens when manually actuated.	24 months

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. --- Verify equivalent leakage of each RCS PIV is ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at an RCS pressure of 1035 psig. The actual test pressure shall be ≥ 935 psig.	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.1.2	-----------------------------NOTE------------------------------ Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than 48 psig in MODE 3, if capable of being manually realigned and not otherwise inoperable. --	
	Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.1.3	Verify ADS accumulator backup compressed gas system average pressure in the required bottles is $\geq$ 2200 psig.	31 days
SR 3.5.1.4	Verify each ECCS pump develops the specified flow rate with the specified differential pressure between reactor and suction source. SYSTEM / FLOW RATE / DIFFERENTIAL PRESSURE BETWEEN REACTOR AND SUCTION SOURCE LPCS / $\geq$ 6200 gpm / $\geq$ 128 psid LPCI / $\geq$ 7200 gpm / $\geq$ 26 psid HPCS / $\geq$ 6350 gpm / $\geq$ 200 psid	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.3	Verify, for each required ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.2.4	-----------------------------NOTE------------------------------ One low pressure coolant injection (LPCI) subsystem may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned and not otherwise inoperable. --- Verify each required ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.5	Verify each required ECCS pump develops the specified flow rate with the specified differential pressure between reactor and suction source. SYSTEM / FLOW RATE / DIFFERENTIAL PRESSURE BETWEEN REACTOR AND SUCTION SOURCE LPCS / $\geq$ 6200 gpm / $\geq$ 128 psid LPCI / $\geq$ 7200 gpm / $\geq$ 26 psid HPCS / $\geq$ 6350 gpm / $\geq$ 200 psid	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM
SR 3.5.2.6	------------------------------NOTE----------------------------- Vessel injection/spray may be excluded. --- Verify each required ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	24 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.3	------------------------------NOTES----------------------------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. --- Verify each primary containment isolation manual valve and blind flange that is located inside primary containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed.	Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days
SR 3.6.1.3.4	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	31 days
SR 3.6.1.3.5	Verify the isolation time of each power operated, automatic PCIV, except for MSIVs, is within limits.	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify a representative sample of reactor instrument line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify the combined leakage rate for all secondary containment bypass leakage paths is ≤ 0.04% primary containment volume/day when pressurized to ≥ P_a.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	Verify leakage rate through each MSIV is ≤ 16.0 scfh when tested at ≥ 25.0 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.12	Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two or more lines with one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening.	E.1 Restore all vacuum breakers in two lines to OPERABLE status.	1 hour
F. Required Action and associated Completion Time of Condition A, B or E not met.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.6.1	------------------------------NOTES----------------------------- 1. Not required to be met for vacuum breakers that are open during Surveillances. 2. Not required to be met for vacuum breakers open when performing their intended function. --- Verify each vacuum breaker is closed.	14 days
SR 3.6.1.6.2	Perform a functional test of each vacuum breaker.	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM
SR 3.6.1.6.3	Verify the full open setpoint of each vacuum breaker is ≤ 0.5 psid.	24 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.3.1	Verify each RHR suppression pool cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.6.2.3.2	Verify each RHR pump develops a flow rate ≥ 7100 gpm through the associated heat exchanger while operating in the suppression pool cooling mode.	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.2.1	------------------------------NOTES---------------------------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative controls. 2. Not required to be met for SCIVs that are open under administrative controls. --- Verify each secondary containment isolation manual valve and blind flange that is not locked, sealed, or otherwise secured, and is required to be closed during accident conditions is closed.	31 days
SR 3.6.4.2.2	Verify the isolation time of each power operated, automatic SCIV is within limits.	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM
SR 3.6.4.2.3	Verify each automatic SCIV actuates to the isolation position on an actual or simulated automatic isolation signal.	24 months

5.5 Programs and Manuals

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Table 3.9-1, Note 1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Deleted~~Inservice Testing Program~~

~~This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.~~

~~a. Testing Frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:~~

~~ASME OM Code and applicable Addenda terminology for inservice testing activities~~	~~Required Frequencies for performing inservice testing activities~~
~~Weekly~~	~~At least once per 7 days~~
~~Monthly~~	~~At least once per 31 days~~
~~Quarterly or every 3 months~~	~~At least once per 92 days~~
~~Semiannually or every 6 months~~	~~At least once per 184 days~~
~~Every 9 months~~	~~At least once per 276 days~~
~~Yearly or annually~~	~~At least once per 366 days~~
~~Biennially or every 2 years~~	~~At least once per 731 days~~

~~b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;~~

~~c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and~~

~~d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.~~

PROPOSED TECHNICAL SPECIFICATIONS BASES CHANGES
(FOR INFORMATION ONLY)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.6

Demonstrating each SLC System pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1220 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve, and is indicative of overall performance. Such inservice tests confirm component OPERABILITY and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the ~~Inservice Testing Program~~INSERVICE TESTING PROGRAM.

SR 3.1.7.7 and SR 3.1.7.8

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The Surveillance Frequency for SR 3.1.7.7 is controlled under the Surveillance Frequency Control Program.

Demonstrating that all heat traced piping between the boron solution storage tank and the suction valve to the injection pumps is unblocked ensures that there is a functioning flow path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping up to the suction valve is unblocked is to pump from the storage tank to the test tank. Upon completion of this verification, the pump suction piping must be drained and flushed with demineralized water since the suction piping between the pump suction valve and pump suction is not heat traced. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. However, if, in performing SR 3.1.7.1, it is determined that the temperature of the solution in the storage tank has fallen below the specified minimum, SR 3.1.7.8 must be performed once within 24 hours after the solution temperature is restored within the limits of Figure 3.1.7-1.

BASES

APPLICABILITY	With THERMAL POWER ≥ 25% RTP, the specified number of SRVs must be OPERABLE since there is considerable energy in the reactor core and the limiting design basis transients are assumed to occur. The SRVs may be required to provide pressure relief to limit peak reactor pressure. The requirements for SRVs in MODE 1 with THERMAL POWER < 25% RTP and in MODES 2 and 3 are discussed in LCO 3.4.4, "SRVs - < 25% RTP." In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The SRV function is not needed during these conditions.
ACTIONS	A.1 With less than the minimum number of required SRVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure, or core thermal margins may be challenged. If one or more required SRVs are inoperable, the plant must be brought to a MODE or other specified Condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	SR 3.4.3.1 This Surveillance demonstrates that the required SRVs will open at the pressures assumed in the safety analysis of Reference 2. The demonstration of the SRV safety function lift settings is in accordance with the ~~Inservice Testing Program~~INSERVICE TESTING PROGRAM. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

BASES

ACTIONS (continued)

Required Action A.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B.1 and B.2

If two or more required SRVs are inoperable, a transient may result in the violation of the ASME Code limit on reactor pressure. The plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This Surveillance demonstrates that the required SRVs will open at the pressures assumed in the safety analysis of Reference 2. The demonstration of the SRV safety function lift settings is in accordance with the ~~Inservice Testing Program~~INSERVICE TESTING PROGRAM. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

SR 3.4.4.2

A manual actuation of each required SRV is performed to verify that, mechanically, the valve is functioning properly and no blockage exists in the valve discharge line. This can be demonstrated by the response of the turbine governor valves or bypass valves, by a change in the measured steam flow, or any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be

BASES

ACTIONS (continued)

reduces the potential for a LOCA outside the containment. The Completion Times are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

Performance of leakage testing on each RCS PIV is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition. As stated in the LCO section of the Bases, the test pressure may be at a lower pressure than the maximum pressure differential (at the RCS maximum pressure of 1035 psig), provided the observed leakage rate is adjusted in accordance with Reference 4. The actual test pressure shall be ≥ 935 psig. For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

The Frequency required by the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM is within the ASME OM Code Frequency requirement and is based on the need to perform this Surveillance under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

Therefore, this SR is modified by a Note that states the leakage Surveillance is only required to be performed in MODES 1 and 2. Entry into MODE 3 is permitted for leakage testing at high differential pressures with stable conditions not possible in the lower MODES.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

The flow path piping has the potential to develop voids and pockets of entrained air. Maintaining the pump discharge lines of the HPCS System, LPCS System, and LPCI subsystems full of water ensures that the systems will perform properly, injecting their full capacity into the RCS upon demand. This will also prevent a water hammer following an ECCS initiation signal. One acceptable method of ensuring the lines are full is to vent at the high points. The 31 day Frequency is based on operating experience, on the procedural controls governing system operation, and on the gradual nature of void buildup in the ECCS piping.

SR 3.5.1.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves potentially capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency of this SR was derived from the ~~Inservice Testing program~~ INSERVICE TESTING PROGRAM requirements for performing valve testing at least once every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control and because improper valve alignment would only affect a single subsystem. This Frequency has been shown to be acceptable through operating experience.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The pump flow rates are verified against a system pressure difference. For the LPCS and LPCI pumps the pressure difference is equivalent to that between the reactor and the suppression pool air volume. For the HPCS pump it is equivalent to the differential above the suction source (suppression pool or condensate storage tank). Under these conditions the total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during LOCAs. A 92 day Frequency for this Surveillance is in accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM requirements.

SR 3.5.1.5

The ECCS subsystems are required to actuate automatically to perform their design functions. This Surveillance test verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of HPCS, LPCS, and LPCI will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup, and actuation of all automatic valves to their required positions. This Surveillance also ensures that the HPCS System will automatically restart on an RPV low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) trip and that the suction is automatically transferred from the CST to the suppression pool. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlaps this Surveillance to provide complete testing of the assumed safety function.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes vessel injection/spray during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.3.2

Verifying the correct alignment for manual, power operated, and automatic valves in the RCIC flow path provides assurance that the proper flow path will exist for RCIC operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. For the RCIC System, this SR also includes the steam flow path for the turbine and the flow controller position.

The 31 day Frequency of this SR was derived from the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM requirements for performing valve testing at least every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control and because improper valve position would affect only the RCIC System. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.3.3 and SR 3.5.3.4

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow against a system head corresponding to reactor pressure is tested both at the higher and lower operating ranges of the system. The required system head should overcome the RPV pressure and associated discharge line losses. Adequate reactor steam pressure must be available to perform these tests. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these SRs. Adequate reactor steam pressure to perform SR 3.5.3.3 is 935 psig and to perform SR 3.5.3.4 is 150 psig. Adequate steam flow to perform SR 3.5.3.3 is represented by THERMAL POWER $\geq$ 10% RTP and to perform SR 3.5.3.4 is represented by turbine bypass valves $\geq$ 10%

BASES

SURVEILLANCE REQUIREMENTS (continued)

open. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time to satisfactorily perform the Surveillance is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure test has been satisfactorily completed and there is no indication or reason to believe that RCIC is inoperable. Therefore, these SRs are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for the flow tests after the required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SRs.

A 92 day Frequency for SR 3.5.3.3 is consistent with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM requirements. The 24 month Frequency for SR 3.5.3.4 is based on the need to perform this Surveillance under the conditions that apply just prior to or during startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.3.5

The RCIC System is required to actuate automatically to perform its design function. This Surveillance verifies that with a required system initiation signal (actual or simulated) the automatic initiation logic of RCIC will cause the system to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup and actuation of all automatic valves to their required positions. This Surveillance test also ensures that the RCIC System will automatically restart on an RPV low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) trip and that the suction is automatically transferred from the CST to the suppression pool. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.2 overlaps this Surveillance to provide complete testing of the assumed design function.

While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

Two Notes are added to this SR. The first Note allows valves and blind flanges located in high radiation areas to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable since the primary containment is inerted and access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA and personnel safety. Therefore, the probability of misalignment of these isolation devices, once they have been verified to be in their proper position, is low. A second Note is included to clarify that PCIVs that are open under administrative controls are not required to meet the SR during the time that the PCIVs are open. These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

SR 3.6.1.3.4

The traversing incore probe (TIP) shear isolation valves are actuated by explosive charges. Surveillance of explosive charge continuity provides assurance that TIP valves will actuate when required. Other administrative controls, such as those that limit the shelf life and operating life, as applicable, of the explosive charges, must be followed. The 31 day Frequency is based on operating experience that has demonstrated the reliability of the explosive charge continuity.

SR 3.6.1.3.5

Verifying the isolation time of each power operated, automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.6. The isolation time test ensures that each valve will isolate in a time period less than or equal to that assumed in the safety analysis. The Frequency of this SR is in accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.6

Verifying that the full closure isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The full closure isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA and transient analyses. The Frequency of this SR is in accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM.

SR 3.6.1.3.7

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.1, "Primary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.8

This SR requires a demonstration that a representative sample of reactor instrument lines' excess flow check valves (EFCVs) are OPERABLE by verifying that each tested valve actuates to the isolation position on an actual or simulated instrument line break condition. The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). In addition, the EFCVs in the sample are representative of the various plant configurations, models, sizes and operating environments. This ensures that any potentially common problem with a specific type or application of EFCV is detected at the earliest possible time. This SR provides assurance that the reactor instrumentation lines' EFCVs will perform as designed. The excess flow check valves in reactor instrument lines are tested by providing an instrument line break signal with pressure at 85 psig to 1050 psig, and at no more than 212°F, RPV coolant temperature, while the EFCV is being exercised. Testing within this pressure range provides a high degree of assurance that these valves will close during an instrument line break while at normal operating pressure.

BASES

ACTIONS (continued)

F. 1 and F.2

If the vacuum breakers in one or more lines cannot be closed or restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1

Each vacuum breaker is verified to be closed to ensure that a potential breach in the primary containment boundary is not present. This Surveillance is performed by observing local or control room indications of vacuum breaker position or by verifying a differential pressure of $\geq$ 0.5 psid is maintained between the reactor building and suppression chamber. The 14 day Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel, and has been shown to be acceptable through operating experience.

Two Notes are added to this SR. The first Note allows reactor building-to-suppression chamber vacuum breakers opened in conjunction with the performance of a surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers. The second Note is included to clarify that vacuum breakers open due to an actual differential pressure, are not considered as failing this SR.

SR 3.6.1.6.2

Each vacuum breaker must be cycled to ensure that it opens properly to perform its design function and returns to its fully closed position. This ensures that the safety analysis assumptions are valid. The Frequency of this SR is in accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM.

SR 3.6.1.6.3

BASES

ACTIONS (continued)

E.1 and E.2

If the open suppression chamber-to-drywell vacuum breaker cannot be closed within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.7.1

Each vacuum breaker is verified closed (except when the vacuum breaker is performing its intended design function) to ensure that this potential large bypass leakage path is not present. This Surveillance is performed by observing the vacuum breaker position indication or by verifying that a differential pressure of ≥ 0.5 psid between the suppression chamber and drywell is maintained for 1 hour without makeup. The 14 day Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel, and has been shown to be acceptable through operating experience.

A Note is added to this SR which allows suppression chamber-to-drywell vacuum breakers opened in conjunction with the performance of a surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers.

SR 3.6.1.7.2

Each required vacuum breaker must be cycled to ensure that it opens adequately to perform its design function and returns to the fully closed position. This ensures that the safety analysis assumptions are valid. The 31 day Frequency of this SR was developed, based on ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM requirements to perform valve testing at least once every 92 days. A 31 day Frequency was chosen to provide additional assurance that the vacuum breakers are OPERABLE, since they are located in a harsh environment (the suppression chamber airspace). In addition, this functional test is required within 12 hours after a discharge of steam to the suppression chamber from the safety/relief valves.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the system is a manually initiated system. This Frequency has been shown to be acceptable, based on operating experience.

SR 3.6.2.3.2

Verifying each RHR pump develops a flow rate ≥ 7100 gpm, while operating in the suppression pool cooling mode with flow through the associated heat exchanger, ensures that the primary containment peak pressure and temperature can be maintained below the design limits during a DBA (Ref. 2). The normal test of centrifugal pump performance required by the ASME OM Code (Ref. 4) is covered by the requirements of LCO 3.5.1, "ECCS - Operating." Such inservice tests confirm component OPERABILITY, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM.

REFERENCES

1. FSAR, Section 6.2.1.1.3.3.
2. FSAR, Section 6.2.2.3.
3. 10 CFR 50.36(c)(2)(ii).
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
5. NEDC-32988-A, Revision 2, Technical Justification to Support Risk-Informed Modification to Selected Required End States for BWR Plants, December 2002.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.6.3.2.1 Operating each head area return fan for ≥ 15 minutes ensures that each subsystem is OPERABLE and that all associated controls are functioning properly. It also ensures that blockage or fan or motor failure can be detected for corrective action. The 92 day Frequency is consistent with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM Frequencies, operating experience, the known reliability of the fan motors and controls, and the two redundant fans available.
REFERENCES	1. Regulatory Guide 1.7, Revision 1, September 1976. 2. FSAR, Section 6.2.5.2.1. 3. Columbia Generating Station Technical Memo TM-2065, "Requirements for Containment Mixing Fans," Revision 0, July 15, 1994. 4. 10 CFR 50.36(c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

This SR verifies each secondary containment isolation manual valve and blind flange that is not locked, sealed, or otherwise secured, and is required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the secondary containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those SCIVs in secondary containment that are capable of being mispositioned are in the correct position.

Since these SCIVs are readily accessible to personnel during normal unit operation and verification of their position is relatively easy, the 31 day Frequency was chosen to provide added assurance that the SCIVs are in the correct positions. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

Two Notes have been added to this SR. The first Note applies to valves and blind flanges located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these isolation devices, once they have been verified to be in the proper position, is low.

A second Note has been included to clarify that SCIVs that are open under administrative controls are not required to meet the SR during the time the SCIVs are open. These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

SR 3.6.4.2.2

Verifying the isolation time of each power operated, automatic SCIV listed in Licensee Controlled Specification Table 1.6.4.2-1 is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the SCIV will isolate in a time period less than or equal to that assumed in the safety analyses. The Frequency of this SR is in accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM.

PROPOSED TECHNICAL SPECIFICATIONS CLEAN PAGES

1.1 Definitions

END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine throttle valve limit switch or from when the turbine governor valve hydraulic control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR50.55a(f).
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. LEAKAGE into the drywell 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;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE; and

d. Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.3	Verify continuity of explosive charge.	31 days
SR 3.1.7.4	Verify the concentration of boron in solution is within the limits of Figure 3.1.7-1.	31 days <u>AND</u> Once within 24 hours after water or boron is added to solution <u>AND</u> Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-1
SR 3.1.7.5	Verify each SLC subsystem manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.1.7.6	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1220 psig.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.1.7.7	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (SRVs) - ≥ 25% RTP

LCO 3.4.3 The safety function of 12 SRVs shall be OPERABLE, with two SRVs in the lowest two lift setpoint groups OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required SRVs inoperable.	A.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the required SRVs are as follows: Number of SRVs / Setpoint (psig) 2 — 1165 ± 34.9 4 — 1175 ± 35.2 4 — 1185 ± 35.5 4 — 1195 ± 35.8 4 — 1205 ± 36.1	In accordance with the INSERVICE TESTING PROGRAM
SR 3.4.3.2	Verify each required SRV opens when manually actuated.	24 months

Number of SRVs	Setpoint (psig)
2	1165 ± 34.9
4	1175 ± 35.2
4	1185 ± 35.5
4	1195 ± 35.8
4	1205 ± 36.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify the safety function lift setpoints of the required SRVs are as follows: Number of SRVs — Setpoint (psig) 2 — 1165 ± 34.9 4 — 1175 ± 35.2 4 — 1185 ± 35.5 4 — 1195 ± 35.8 4 — 1205 ± 36.1	In accordance with the INSERVICE TESTING PROGRAM
SR 3.4.4.2	-------------------------------NOTE----------------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify each required SRV opens when manually actuated.	24 months

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	-------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. --- Verify equivalent leakage of each RCS PIV is ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at an RCS pressure of 1035 psig. The actual test pressure shall be ≥ 935 psig.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.1.2	-----------------------------NOTE------------------------------ Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than 48 psig in MODE 3, if capable of being manually realigned and not otherwise inoperable. --- Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.1.3	Verify ADS accumulator backup compressed gas system average pressure in the required bottles is $\geq$ 2200 psig.	31 days
SR 3.5.1.4	Verify each ECCS pump develops the specified flow rate with the specified differential pressure between reactor and suction source. <u>SYSTEM</u> — <u>FLOW RATE</u> — <u>DIFFERENTIAL PRESSURE BETWEEN REACTOR AND SUCTION SOURCE</u> LPCS — $\geq$ 6200 gpm — $\geq$ 128 psid LPCI — $\geq$ 7200 gpm — $\geq$ 26 psid HPCS — $\geq$ 6350 gpm — $\geq$ 200 psid	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

<table>
<tr><th colspan="2">SURVEILLANCE</th><th>FREQUENCY</th></tr>
<tr><td>SR 3.5.2.3</td><td>Verify, for each required ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.</td><td>31 days</td></tr>
<tr><td>SR 3.5.2.4</td><td>-----------------------------NOTE------------------------------
One low pressure coolant injection (LPCI) subsystem may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned and not otherwise inoperable.

Verify each required ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.</td><td>

31 days</td></tr>
<tr><td>SR 3.5.2.5</td><td>Verify each required ECCS pump develops the specified flow rate with the specified differential pressure between reactor and suction source.

SYSTEM | FLOW RATE | DIFFERENTIAL PRESSURE BETWEEN REACTOR AND SUCTION SOURCE
LPCS | ≥ 6200 gpm | ≥ 128 psid
LPCI | ≥ 7200 gpm | ≥ 26 psid
HPCS | ≥ 6350 gpm | ≥ 200 psid</td><td>In accordance with the INSERVICE TESTING PROGRAM</td></tr>
<tr><td>SR 3.5.2.6</td><td>------------------------------NOTE-----------------------------
Vessel injection/spray may be excluded.

Verify each required ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.</td><td>

24 months</td></tr>
</table>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.1.3.3	-----------------------------NOTES----------------------------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. -- Verify each primary containment isolation manual valve and blind flange that is located inside primary containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed.	Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days
SR 3.6.1.3.4	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	31 days
SR 3.6.1.3.5	Verify the isolation time of each power operated, automatic PCIV, except for MSIVs, is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.1.3.6	Verify the isolation time of each MSIV is $\geq$ 3 seconds and $\leq$ 5 seconds.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify a representative sample of reactor instrument line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify the combined leakage rate for all secondary containment bypass leakage paths is ≤ 0.04% primary containment volume/day when pressurized to ≥ P_a.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	Verify leakage rate through each MSIV is ≤ 16.0 scfh when tested at ≥ 25.0 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.12	Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two or more lines with one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening.	E.1 Restore all vacuum breakers in two lines to OPERABLE status.	1 hour
F. Required Action and associated Completion Time of Condition A, B or E not met.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.6.1	------------------------------NOTES----------------------------- 1. Not required to be met for vacuum breakers that are open during Surveillances. 2. Not required to be met for vacuum breakers open when performing their intended function. --- Verify each vacuum breaker is closed.	14 days
SR 3.6.1.6.2	Perform a functional test of each vacuum breaker.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.1.6.3	Verify the full open setpoint of each vacuum breaker is ≤ 0.5 psid.	24 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.3.1	Verify each RHR suppression pool cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.6.2.3.2	Verify each RHR pump develops a flow rate ≥ 7100 gpm through the associated heat exchanger while operating in the suppression pool cooling mode.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.2.1	------------------------------NOTES----------------------------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative controls. 2. Not required to be met for SCIVs that are open under administrative controls. --- Verify each secondary containment isolation manual valve and blind flange that is not locked, sealed, or otherwise secured, and is required to be closed during accident conditions is closed.	31 days
SR 3.6.4.2.2	Verify the isolation time of each power operated, automatic SCIV is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.4.2.3	Verify each automatic SCIV actuates to the isolation position on an actual or simulated automatic isolation signal.	24 months

5.5 Programs and Manuals

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Table 3.9-1, Note 1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Deleted