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50.90

NLS2016046
August 26, 2016

Attention: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Subject: Application to Revise Technical Specifications to Adopt TSTF-545, Revision 3, "TS Inservice Testing Program Removal and Clarify Surveillance Requirement Usage Rule Application to Section 5.5 Testing"
Cooper Nuclear Station, Docket No. 50-298, License No. DPR-46

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Nebraska Public Power District (NPPD) is submitting a request for an amendment to the Technical Specifications (TS) for Cooper Nuclear Station (CNS). The proposed change revises the TS to eliminate Section 5.5.6, "Inservice Testing 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 and Clarify Surveillance Requirement Usage Rule Application to Section 5.5 Testing."

The use of Code Case OMN-20, "Inservice Test Frequency," was previously approved for use at CNS in a Nuclear Regulatory Commission (NRC) safety evaluation dated February 12, 2016 (ML16014A174).

NPPD requests NRC approval of the proposed TS change and issuance of the requested license amendment by September 10, 2017. The amendment will be implemented within 60 days of issuance of the amendment.

Attachment 1 provides a description and assessment of the proposed TS changes. Attachment 2 provides the existing TS pages marked up to show the proposed changes. Attachment 3 provides revised (clean) TS pages. Attachment 4 provides TS Bases pages marked up to show the associated TS Bases changes and is provided for information only.

This proposed TS change has been reviewed by the necessary safety review committees (Station Operations Review Committee and Safety Review and Audit Board). Amendments to the CNS Renewed Facility Operating License through Amendment 256 dated July 25, 2016, have been incorporated into this request. This request is submitted under oath pursuant to 10 CFR 50.30(b).

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NRR

By copy of this letter and its attachments, the appropriate State of Nebraska official is notified in accordance with 10 CFR 50.91(b)(1). Copies to the NRC Region IV office and the CNS Resident Inspector are also being provided in accordance with 10 CFR 50.4(b)(1).

This letter contains no regulatory commitments.

Should you have any questions concerning this matter, please contact Jim Shaw, Licensing Manager, at (402) 825-2788.

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

Executed On: 8/26/16
Date

Sincerely,

Oscar A. Limpias
Vice President - Nuclear and
Chief Nuclear Officer

/dv

Attachments:
1. Description and Assessment of Technical Specifications Changes
2. Proposed Technical Specifications Changes (Mark-up)
3. Revised Technical Specifications Pages
4. Proposed Technical Specifications Bases Changes (Mark-up) - Information Only

cc: Regional Administrator w/ attachments
USNRC - Region IV

Cooper Project Manager w/ attachments
USNRC - NRR Plant Licensing Branch IV-2

Senior Resident Inspector w/ attachments
USNRC - CNS

Nebraska Health and Human Services w/ attachments
Department of Regulation and Licensure

NPG Distribution w/o attachments

CNS Records w/ attachments

Attachment 1

Description and Assessment of Technical Specifications Changes

Cooper Nuclear Station, Docket No. 50-298, License No. DPR-46

Revised Technical Specification Pages

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1.0 Description

2.0 Assessment

2.1 Applicability of Published Safety Evaluation
2.2 Variations

3.0 Regulatory Analysis

3.1 No Significant Hazards Consideration Analysis

4.0 Environmental Evaluation

1.0 DESCRIPTION

The proposed change eliminates Technical Specifications (TS), Section 5.5.6, "Inservice Testing (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 TSTF-545, Revision 3, "TS Inservice Testing Program Removal and Clarify Surveillance Requirement Usage Rule Application to Section 5.5 Testing."

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

Nebraska Public Power District (NPPD) 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 (ML15314A365 and ML15314A305). This review included a review of the Nuclear Regulatory Commission (NRC) staff's evaluation, as well as the information provided in TSTF-545. NPPD concluded that the justifications presented in TSTF-545, and the model safety evaluation prepared by the NRC staff are applicable to Cooper Nuclear Station (CNS) and justify this amendment for the incorporation of the changes to the CNS TS.

CNS was issued a construction permit on June 4, 1968, and the provisions of 10 CFR 50.55a(f)(1) are applicable.

In a letter to the NRC, dated March 19, 2015 (ML15084A221), NPPD submitted requests for relief to certain ASME OM code requirements for the CNS fifth 10-year IST program interval. Relief Request RG-01 requested use of Code Case OMN-20 as an alternative to the frequencies of the ASME OM Code. Relief Request RG-01 was approved by the NRC in a safety evaluation dated February 12, 2016 (ML16014A174).

2.2 Variations

No technical variations are proposed in this amendment request. The following proposed variations are administrative and do not affect the applicability of TSTF-545 or the NRC staff's model safety evaluation dated December 11, 2015. The CNS TS 1) in some cases utilize a different section title or Surveillance Requirement numbering system and, 2) do not include all the specifications shown on the applicable Standard Technical Specifications (STS), NUREG 1433, pages.

- Since the CNS TS do not include SR 3.4.5.1, Reactor Coolant System Pressure Isolation Valve Leakage, as shown on the General Electric BWR/4 STS pages in TSTF-545, there will be no corresponding change to the CNS TS.

- General Electric BWR/4 STS SR 3.5.1.7 is numbered SR 3.5.1.6 in the CNS TS.
- General Electric BWR/4 STS SR 3.5.2.5 is numbered SR 3.5.2.4 in the CNS TS.
- General Electric BWR/4 STS SR 3.6.1.3.6 is numbered SR 3.6.1.3.5 in the CNS TS.
- General Electric BWR/4 STS SR 3.6.1.3.8 is numbered SR 3.6.1.3.6 in the CNS TS.
- General Electric BWR/4 STS section 3.6.2.4 is titled Residual Heat Removal (RHR) Suppression Pool Spray. The equivalent section in the CNS TS is numbered 3.6.1.9 and titled RHR Containment Spray.
- General Electric BWR/4 STS SR 3.6.2.4.2 is numbered SR 3.6.1.9.2 in the CNS TS.
- General Electric BWR/4 STS IST Program is section 5.5.7. The equivalent CNS TS section is 5.5.6.
- TSTF-545 deletes the IST program TS 5.5.6 and re-numbers all subsequent TS programs. NPPD proposes to retain the TS 5.5.6 reference, now shown as "DELETED," and not change the subsequent TS program numbers.

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Analysis

Nebraska Public Power District (NPPD) requests adoption of the Technical Specifications (TS) changes described in TSTF-545, "TS Inservice Testing Program Removal and Clarify Surveillance Requirement Usage Rule Application to Section 5.5 Testing," which is an approved change to the Improved Standard Technical Specifications, into the Cooper Nuclear Station 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." Other requirements in Section 5.5 are eliminated because the Nuclear Regulatory Commission (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). NPPD has evaluated whether or not a significant hazards consideration is involved with the proposed amendment 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." 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, NPPD 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.

Attachment 2

Proposed Technical Specifications Changes (Mark-up)

Cooper Nuclear Station, Docket No. 50-298, License No. DPR-46

Revised Pages

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1.1-4
1.1-5
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3.6-13
3.6-14
3.6-26
3.6-32
3.6-39
5.0-10

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)

I-133, I-134, and I-135 actually present. The DOSE EQUIVALENT I-131 concentration is calculated as follows: DOSE EQUIVALENT I-131 = (I-131) + 0.0060 (I-132) + 0.17 (I-133) + 0.0010 (I-134) + 0.029 (I-135). The dose conversion factors used for this calculation are those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.

INSERVICE TESTING PROGRAM

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;

The INSERVICE TESTIG PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE;

d. Pressure Boundary LEAKAGE

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

LINEAR HEAT GENERATION RATE (LHGR)

The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit,

(continued)

1.1 Definitions

LOGIC SYTEM FUNCTIONAL TEST (continued)	from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.7.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2419 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time segment from the time the sensor contacts actuate to the time the scram solenoid valves deenergize.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that: a. The reactor is xenon free;

(continued)

1.1 Definitions

SHUTDOWN MARGIN (SDM) (continued)	b.	The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and
	c.	All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.	
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.	
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components:	
	a.	The time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established; and
	b.	The time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve.
	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.	

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each SLC subsystem manual 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 3.1.7.7	Verify each pump develops a flow rate ≥ 38.2 gpm at a discharge pressure ≥ 1300 psig.	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction is unblocked.	24 months AND Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the SRVs and SVs are as follows: Number of SRVs — Setpoint (psig) 2 — 1080 ± 32.4 3 — 1090 ± 32.7 3 — 1100 ± 33.0 Number of SVs — Setpoint (psig) 3 — 1240 ± 37.2 Following testing, lift settings shall be within ± 1%.	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM
SR 3.4.3.2	----------NOTE---------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ---------- Verify each SRV opens when manually actuated.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.6	Verify the following ECCS pumps develop the specified flow rate against a system head corresponding to the specified reactor pressure. SYSTEM / FLOW RATE / NO. OF PUMPS / SYSTEM HEAD CORRESPONDING TO A REACTOR PRESSURE OF Core Spray: ≥ 4720 gpm, 1, ≥ 113 psig LPCI: ≥ 15,000 gpm, 2, ≥ 20 psig	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM
SR 3.5.1.7	----------NOTE---------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ---------- Verify, with reactor pressure ≤ 1020 and ≥ 920 psig, the HPCI pump can develop a flow rate ≥ 4250 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.1.8	----------NOTE---------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ---------- Verify, with reactor pressure ≤ 165 psig, the HPCI pump can develop a flow rate ≥ 4250 gpm against a system head corresponding to reactor pressure.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.4	Verify each required ECCS pump develops the specified flow rate against a system head corresponding to the specified reactor pressure. SYSTEM — FLOW RATE — NO. OF PUMPS — SYSTEM HEAD CORRESPONDING TO A REACTOR PRESSURE OF CS — ≥ 4720 gpm — 1 — ≥ 113 psig LPCI — ≥ 7700 gpm — 1 — ≥ 20 psig	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM
SR 3.5.2.5	----------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 (continued)

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 manual isolation 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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
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
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 instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break.	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 leakage rate through each Main Steam line is ≤ 106 scfh when tested at ≥ 29 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.9.1	Verify each RHR containment 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 or can be aligned to the correct position.	31 days
SR 3.6.1.9.2	Verify each required RHR pump develops a flow rate of > 7700 gpm through the associated heat exchanger while operating in the suppression pool cooling mode.	In accordance with the ~~Inservice Testing Program~~ INSERVICE TESTING PROGRAM
SR 3.6.1.9.3	Verify each spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage

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 > 7700 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 means. 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 actuation signal.	24 months

5.5 Programs and Manuals (continued)

5.5.6 ~~Inservice Testing Program~~ (Deleted)

~~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 Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda are 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.~~

(continued)

Attachment 3

Revised Technical Specifications Pages

Cooper Nuclear Station, Docket No. 50-298, License No. DPR-46

Revised Pages

1.1-3
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1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	I-133, I-134, and I-135 actually present. The DOSE EQUIVALENT I-131 concentration is calculated as follows: DOSE EQUIVALENT I-131 = (I-131) + 0.0060 (I-132) + 0.17 (I-133) + 0.0010 (I-134) + 0.029 (I-135). The dose conversion factors used for this calculation are those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).
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; d. Pressure Boundary LEAKAGE LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.
LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

(continued)

1.1 Definitions

LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.7.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2419 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time segment from the time the sensor contacts actuate to the time the scram solenoid valves deenergize.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that:

(continued)

1.1 Definitions

SHUTDOWN MARGIN (SDM) (continued)	a. The reactor is xenon free; b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components: a. The time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established; and b. The time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve. 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.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each SLC subsystem manual 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 3.1.7.7	Verify each pump develops a flow rate ≥ 38.2 gpm at a discharge pressure ≥ 1300 psig.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction is unblocked.	24 months AND Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the SRVs and SVs are as follows: Number of SRVs — Setpoint (psig) 2 — 1080 ± 32.4 3 — 1090 ± 32.7 3 — 1100 ± 33.0 Number of SVs — Setpoint (psig) 3 — 1240 ± 37.2 Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.4.3.2	----------NOTE---------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ---------- Verify each SRV opens when manually actuated.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.6	Verify the following ECCS pumps develop the specified flow rate against a system head corresponding to the specified reactor pressure. SYSTEM / FLOW RATE / NO. OF PUMPS / SYSTEM HEAD CORRESPONDING TO A REACTOR PRESSURE OF Core Spray: ≥ 4720 gpm, 1, ≥ 113 psig LPCI: ≥ 15,000 gpm, 2, ≥ 20 psig	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.1.7	----------NOTE---------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ---------- Verify, with reactor pressure ≤ 1020 and ≥ 920 psig, the HPCI pump can develop a flow rate ≥ 4250 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.1.8	----------NOTE---------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ---------- Verify, with reactor pressure ≤ 165 psig, the HPCI pump can develop a flow rate ≥ 4250 gpm against a system head corresponding to reactor pressure.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.4	Verify each required ECCS pump develops the specified flow rate against a system head corresponding to the specified reactor pressure. SYSTEM / FLOW RATE / NO. OF PUMPS / SYSTEM HEAD CORRESPONDING TO A REACTOR PRESSURE OF CS / ≥ 4720 gpm / 1 / ≥ 113 psig LPCI / ≥ 7700 gpm / 1 / ≥ 20 psig	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.2.5	----------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 (continued)

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 manual isolation 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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
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
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 instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break.	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 leakage rate through each Main Steam line is ≤ 106 scfh when tested at ≥ 29 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.9.1	Verify each RHR containment 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 or can be aligned to the correct position.	31 days
SR 3.6.1.9.2	Verify each required RHR pump develops a flow rate of > 7700 gpm through the associated heat exchanger while operating in the suppression pool cooling mode.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.1.9.3	Verify each spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage

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 > 7700 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 means. 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 actuation signal.	24 months

5.5 Programs and Manuals (continued)

5.5.6 (Deleted)

(continued)

Attachment 4

Proposed Technical Specifications Bases Changes (Mark-up) - Information Only

Cooper Nuclear Station, Docket No. 50-298, License No. DPR-46

Revised Pages

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B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.10 and apply at all times, unless otherwise stated. ← SR 3.0.2 and SR 3.0.3 apply in Chapter 5 when invoked by a Chapter 5 Specification.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a Special Operations LCO are only applicable when the Special Operations LCO is used as an allowable exception to the requirements of a Specification.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 3.1.7.8 and SR 3.1.7.9

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 pump and explosive valve tested should be alternated such that both complete flow paths are tested every 48 months at alternating 24 month intervals. 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 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 these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all heat traced piping between the boron solution storage tank and the suction inlet 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 is unblocked is to manually initiate the system, except the explosive valves, and pump from the storage tank to the test tank. Upon completion of this verification, the pump suction piping must be flushed with demineralized water to ensure piping between the storage tank and pump suction is unblocked. The 24 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. This is especially true in light of the temperature verification of this piping

BASES

APPLICABILITY

In MODES 1, 2, and 3, 7 of 8 SRVs and 3 SVs must be OPERABLE, since considerable energy may be in the reactor core and the limiting design basis transients are assumed to occur in these MODES. The SRVs and SVs may be required to provide pressure relief to limit peak reactor pressure.

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 and SV function is not needed during these conditions.

ACTIONS

A.1 and A.2

With the safety function of one or more of the required SRVs or SVs inoperable, a transient may result in the violation of the ASME Code limit on reactor pressure. If the safety function of one or more of the required SRVs or SVs is inoperable, 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 3.4.3.1

This Surveillance requires that the SRVs and SVs will open at the pressures assumed in the safety analysis of Reference 3. The demonstration of the SRV and SV safety function lift settings must be performed during shutdown, since this is a bench test, to be done in accordance with the ~~Inservice Testing Program~~. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The SRV setpoint is ± 3% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

INSERVICE TESTING PROGRAM

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.2 (continued)

OPERABILITY. Also, 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 HPCI System, this SR also includes the steam flow path for the turbine and the flow controller position.

INSERVICE TESTING PROGRAM

The 31 day Frequency of this SR was derived from the ~~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 position would only affect a single subsystem. This Frequency has been shown to be acceptable through operating experience.

In Mode 3 with reactor steam dome pressure less than the actual shutdown cooling permissive pressure, the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Therefore, this SR is modified by a Note that allows LPCI subsystems to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. Alignment and operation for decay heat removal includes when the required RHR pump is not operating or when the system is realigned from or to the RHR shutdown cooling mode. At the low pressures and decay heat loads associated with operation in MODE 3 with reactor steam dome pressure less than the shutdown cooling permissive pressure, a reduced complement of low pressure ECCS subsystems should provide the required cooling, thereby allowing operation of RHR shutdown cooling, when necessary.

~~(continued)~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

the LPCI subsystem. Acceptable methods of de-energizing the valve include de-energizing breaker control power, racking out the breaker or removing the breaker.

The specified Frequency is once during reactor startup before THERMAL POWER is > 25% RTP. However, this SR is modified by a Note that states the Surveillance is only required to be performed if the last performance was more than 31 days ago. Therefore, implementation of ~~this Note~~ requires this test to be performed during reactor startup before exceeding 25% RTP. ~~Verification~~ during reactor startup prior to reaching > 25% RTP is an exception to the normal ~~Inservice Testing Program~~ generic valve cycling Frequency of 92 days, but is considered acceptable due to the demonstrated reliability of these valves. If the valve is inoperable and in the open position, the associated LPCI subsystem must be declared inoperable.

INSERVICE TESTING PROGRAM

SR 3.5.1.6, SR 3.5.1.7, and SR 3.5.1.8

The performance requirements of the low pressure ECCS pumps are determined through application of the 10 CFR 50, Appendix K criteria (Ref. 7). This periodic Surveillance is performed (in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants requirements for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of Reference 8. The pump flow rates are verified against a system head equivalent to the RPV pressure expected during a LOCA. 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 a LOCA.

The flow tests for the HPCI 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 at both the higher and lower operating ranges of the system. The required system head

BASES

SURVEILLANCE REQUIREMENTS (continued)

should overcome the RPV pressure and associated discharge line losses. Adequate reactor 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 HPCI System diverts steam flow. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these tests. Adequate reactor steam pressure must be $\geq$ 920 psig to perform SR 3.5.1.7 and $\geq$ 145 psig to perform SR 3.5.1.8. Adequate steam flow is represented by turbine bypass valves at least 30% open, or total steam flow $\geq 10^6$ lb/hr. Reactor startup is allowed prior to performing the low pressure Surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance test 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 HPCI is inoperable.

Therefore, SR 3.5.1.7 and SR 3.5.1.8 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 required pressure and flow are reached are sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SRs. For SR 3.5.1.8, while adequate pressure can be reached prior to the required Applicability for HPCI, the 12 hour allowance of the Note would not apply until entering the Applicability (>150 psig) with adequate steam flow.

INSERVICE TESTING PROGRAM

The Frequency for SR 3.5.1.6 and SR 3.5.1.7 is in accordance with the ~~Inservice Testing Program~~ requirements. The 24 month Frequency for SR 3.5.1.8 is based on the need to perform the Surveillance under the conditions that apply just prior to or during a 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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.

INSERVICE TESTING PROGRAM

The 31 day Frequency of this SR was derived from the ~~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 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 920 psig and 145 psig to perform SR 3.5.3.4. Adequate steam flow is represented by turbine bypass valves at least 30% open, or total steam flow $\geq 10^6$ lb/hr. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time allowed 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 Surveillance 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

BASES

SURVEILLANCE REQUIREMENTS (continued)

adequate to perform the test. The 12 hours allowed for the flow tests after the required pressure and flow are reached are sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SRs. For SR 3.5.3.4, while adequate pressure can be reached prior to the required Applicability for RCIC, the 12 hour allowance of the Note would not apply until entering the Applicability (>150 psig) with adequate steam flow.

INSERVICE TESTING PROGRAM

A 92 day Frequency for SR 3.5.3.3 is consistent with the ~~Inservice Testing Program~~ requirements. The 24 month Frequency for SR 3.5.3.4 is based on the need to perform the Surveillance under conditions that apply just prior to or during a 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 in order to verify its design function satisfactorily. This Surveillance verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of the RCIC System will cause the system to operate as designed, including actuation of the system throughout its emergency operating sequence; that is, automatic pump startup and actuation of all automatic valves to their required positions. This test also ensures 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 ECST 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.

The 24 month Frequency is based on the need to perform some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures 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.

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.6.1.3.3 (continued)~~

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 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 the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the requirements of the ~~Inservice Testing Program.~~

INSERVICE TESTING PROGRAM

SR 3.6.1.3.6

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The 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. This ensures that the

~~(continued)~~

SURVEILLANCE REQUIREMENTS (continued)

calculated radiological consequences of these events remain within 10 CFR 100 limits. The Frequency of this SR is in accordance with the requirements of 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 was developed considering it is prudent that this Surveillance be performed only during a unit outage since isolation of penetrations would disrupt the normal operation of many critical components. 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 instrumentation line excess flow check valves (EFCVs) are OPERABLE by verifying that each valve actuates to the isolation position on an actual or simulated instrument line break. The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). This SR provides assurance that the instrumentation line EFCVs will perform so that predicted radiological consequences will not be exceeded during the postulated instrument line break event. The 24 month Frequency is based on the need to perform the 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.

The nominal 10 year interval is based on other performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.7.2

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 92 day Frequency of this SR was developed based upon ~~Inservice Testing Program~~ requirements to perform valve testing at least once every 92 days.

INSERVICE TESTING PROGRAM

SR 3.6.1.7.3

Demonstration of vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker full open differential pressure of ≤ 0.5 psid is valid. The 24 month Frequency is based on the need to perform some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures were performed with the reactor at power. For this unit, the 24 month Frequency has been shown to be acceptable, based on operating experience, and is further justified because of other Surveillances performed at shorter Frequencies that convey the proper functioning status of each vacuum breaker.

REFERENCES

1. Bodega Bay Preliminary Hazards Summary Report, Appendix I, Docket 50-205, December 28, 1962.

2. USAR, Section V-2.3.6.

3. 10 CFR 50.36(c)(2)(ii).

BASES

ACTIONS (continued)

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, 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 3.6.1.8.1

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 performing a leak test that confirms that the bypass area between the drywell and suppression chamber is less than or equivalent to a one inch diameter hole. If the bypass test fails, not only must the vacuum breaker(s) be considered open and the appropriate Conditions and Required Actions of this LCO be entered, but also the appropriate Conditions and Required Actions of LCO 3.6.1.1, Primary Containment, must be entered. 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 3.6.1.8.2

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

~~(continued)~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.9.2

Verifying each required RHR pump develops a flow rate > 7700 gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that pump performance has not degraded during the cycle. It is tested in the pool cooling mode to demonstrate pump OPERABILITY without spraying down equipment in the drywell. Flow is a normal test of centrifugal pump performance required by the ASME Code~~, Section XI~~ (Ref. 4). This test confirms one point on the pump performance curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the ~~Inservice Testing Program.~~

INSERVICE TESTING PROGRAM

SR 3.6.1.9.3

This Surveillance is performed following maintenance which could result in nozzle blockage by introduction of air to verify that the spray nozzles are not obstructed and that flow will be provided when required. The Frequency is adequate to detect degradation in performance due to the passive nozzle design and its normally dry state and has been shown to be acceptable through operating experience.

REFERENCES

1. USAR, Chapter XIV, Section 6.3.

2. USAR, Chapter V, Section 2.

3. EE 01-035, EQ Temperature Profile in Containment based on Small Steam Line Break and DBA-LOCA Analysis.

4. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.2.3.2

Verifying that each RHR pump develops a flow rate ≥ 7700 gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that pump performance has not degraded during the cycle. Flow is a normal test of centrifugal pump performance required by ASME Code (Ref. 4). This test confirms one point on the pump design curve, and the results are indicative of overall performance. Such inservice ~~inspections~~ tests confirm component OPERABILITY, trend performance, 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. USAR, Section XIV-6.
2. 10 CFR 36(c)(2)(ii).
3. NEDC 94-034B, C & D
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 that the isolation time of each power operated automatic SCIV 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 isolation time and Frequency of this SR are in accordance with the ~~Inservice Testing Program~~.

INSERVICE TESTING PROGRAM

SR 3.6.4.2.3

Verifying that each automatic SCIV closes on a secondary containment isolation signal is required to minimize leakage of radioactive material from secondary containment following a DBA or other accidents. This SR ensures that each automatic SCIV will actuate to the isolation position on a secondary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary 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 these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section V-3.0.
2. USAR, Section XIV-6.0.
3. USAR, Section XIV-6.3.