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1CAN031604

10 CFR 50.90
10 CFR 50.55a

March 25, 2016

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

SUBJECT: Application to Revise Technical Specifications to Adopt TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR Usage Rule Application to Section 5.5 Testing," and to Request an Alternative to the ASME Code
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) is submitting a request for an amendment to the Technical Specifications (TS) for Arkansas Nuclear One, Unit 1. The proposed change revises the TSs to eliminate Section 5.5, "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 & Clarify SR Usage Rule Application to Section 5.5 Testing."

Pursuant to 10 CFR 50.55a(z), the application also proposes an alternative to the testing frequencies in the American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code, by adoption of approved Code Case OMN-20, "Inservice Test Frequency," for the current 10 year Inservice Testing (IST) interval.

Attachment 1 provides a description and assessment of the proposed 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.

Approval of the proposed amendment and relief request is requested by October 1, 2016. Once approved, the amendment shall be implemented within 90 days.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Arkansas state official.

No new commitments have been identified in this letter.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 25, 2016.

If you have any questions or require additional information, please contact Stephenie Pyle at 479-858-4704.

Sincerely,

ORIGINAL SIGNED BY JEREMY G. BROWNING

JGB/dbb

Attachments:

1. Description and Assessment of Technical Specification Changes
2. Proposed Technical Specification Changes (Mark-Up)
3. Revised Technical Specification Pages
4. Proposed Technical Specification Bases Changes (Mark-Up) – Information Only
5. Description and Assessment of the Proposed Alternative to the ASME Code

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Attachment 1 to

1CAN031604

Description and Assessment of Technical Specification Changes

Description and Assessment of Technical Specification Changes

1.0 DESCRIPTION

The proposed change eliminates the Technical Specifications (TS), Section 5.5, "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 & Clarify SR Usage Rule Application to Section 5.5 Testing."

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

Entergy Operations, Inc. (Entergy), has reviewed the model safety evaluation (SE) addressed to the Technical Specifications Task Force in a letter dated December 11, 2015 (ADAMS Accession No. ML15317A071). This review included a review of the NRC staff's evaluation, as well as the information provided in TSTF-545. Entergy concluded that the justifications presented in TSTF-545, and the model safety evaluation prepared by the NRC staff are applicable to Arkansas Nuclear One, Unit 1 (ANO-1) and justify this amendment for the incorporation of the changes to the ANO-1 TS.

ANO-1 was issued a construction permit on December 6, 1968, and the provisions of 10 CFR 50.55a(f)(1) are applicable.

2.2 Variations

No technical variances are proposed in this amendment request. The following bulleted items identify administrative-type variations. These differences to not result in any technical conflict with the intent of TSTF-545 or the associated model SE. Therefore, Entergy has concluded the following variations are both acceptable and permit NRC review of the proposed adoption of TSTF-545 under the Consolidated Line Item Improvement Process (CLIIP).

- Some minor formatting clean-up is performed, where needed, to adjust available space on a page or in a table cell. With reference to TS 3.7.1, Table 3.7.1-1 is moved from TS Page 3.7.1-3 to Page 3.7.1-2 and the former is deleted in its entirety.

- In a few examples, the ANO-1 Surveillance Requirement (SR) numbering does not match the numbering included in the TSTF-545 markup pages; however, Entergy verified the SRs are equivalent. TSTF-545 contains markup pages based on Revision 4 of NUREG 1430, "Standard Technical Specifications – Babcock & Wilcox Plants," while the ANO-1 TSs are based on Revision 1 of NUREG 1430. With reference to SR 3.7.2.1, the IST reference also appears in the SR column of the table and, therefore, is capitalized consistent with the intent of TSTF-545. NUREG 1430, Revision 4, contains the IST reference in the Completion Time column only.

- Entergy performed a search of the entire ANO-1 TSs for the key phrase "inservice testing program" and "IST". In addition to that already described above, the TS Bases may have included other examples of this phrase, all of which are now capitalized as a defined term, consistent with TSTF-545. Also, the following reference (or similar) is added to applicable TS Bases (note that the TS Bases markups included in Attachment 4 are submitted for information only).

  ASME OM Code - 2001 Edition through 2003 Addenda and Code Case OMN-20 (Inservice Test Frequency).

- TSTF-545 deletes the IST program TS 5.5.8 and re-numbers all subsequent TS programs. This also impacts several TS Bases references. Entergy proposes to retain the TS 5.5.8 reference, now shown as "DELETED", and not change the subsequent TS program numbers. These program numbers are referenced in a wealth of station procedures. By maintaining the current program numbering, excessive administrative burden is avoided. Based on this approach, several TSTF-545 TS Bases markup pages associated with the TSTF-545 program numbering are not included in Attachment 4 of this application.

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Analysis

Entergy Operations, Inc. (Entergy), requests adoption of the Technical Specification (TS) changes described in TSTF-545, "TS Inservice Testing Program Removal & Clarify SR Usage Rule Application to Section 5.5 Testing," which is an approved change to the Improved Standard Technical Specifications (ISTS), into the Arkansas Nuclear One, Unit 1 (ANO-1) TS. The proposed change revises the TS Chapter 5, "Administrative Controls," Section 5.5, "Programs and Manuals," to delete the "Inservice Testing Program" specification. Requirements in the Inservice Testing (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). Entergy 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 IST 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 accident 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 Surveillance Requirement (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, Entergy 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 PRECEDENCE

Because TSTF-545, Revision 3, was approved in December 2015, time has not permitted licensees to apply, and gain approval for, adoption of TSTF-545. Therefore, other than TSTF 545 itself, no other precedence exists at this time.

Attachment 2

1CAN031604

Proposed Technical Specification Changes (mark-up)

1.1 Definition (continued)

DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."
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, such as that from pump seals or valve packing (except RCP seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection and leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required pressurizer safety valve inoperable in MODE 3 or MODE 4 with RCS temperature > 259 °F.	C.1 Be in MODE 4 with RCS temperature ≤ 259 °F.	18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each required pressurizer safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~. Following testing, as-left lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.14.1	---------------------------NOTE-------------------------------- Not required to be performed in MODES 3 and 4. --- Verify leakage from each RCS pressure isolation check valve, or pair of check valves, as applicable, is less than or equal to an equivalent of the Allowable Leakage Limit identified below at a differential test pressure ≥ 150 psid. Pressure Isolation Check Valve(s) — Allowable Leakage Limit DH-14A — ≤ 5 gpm DH-13A and DH-17 — ≤ 5 gpm total DH-14B — ≤ 5 gpm DH-13B and DH-18 — ≤ 5 gpm total	In accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~ AND Once prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months
SR 3.4.14.2	Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual high RCS pressure signal.	18 months
SR 3.4.14.3	Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual high RCS pressure signal: a. ≤ 340 psig for one valve; and b. ≤ 400 psig for the other valve.	18 months
SR 3.4.14.4	Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual Core Flood Tank isolation valve “not closed” signal.	18 months
SR 3.4.14.5	Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual Core Flood Tank isolation valve “not closed” signal.	18 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) temperature > 350 °F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Reduce RCS temperature to ≤ 350 °F.	6 hours 12 hours
C. Less than 100% of the ECCS flow equivalent to a single OPERABLE train available.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify each ECCS 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 3.5.2.2	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each reactor building purge isolation valve is closed.	31 days
SR 3.6.3.2	------------------------------NOTE---------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. --	
	Verify each reactor building isolation manual valve and blind flange that is located outside the reactor building and not locked, sealed, or otherwise secured, and is required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.	31 days
SR 3.6.3.3	------------------------------NOTE---------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. --	
	Verify each reactor building isolation manual valve and blind flange that is located inside the reactor building and not locked, sealed, or otherwise secured, and required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.4	Verify the isolation time of each automatic power operated reactor building isolation valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~
SR 3.6.3.5	Verify each automatic reactor building isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify each reactor building spray manual, power operated, and automatic valve in each required flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.5.2	Operate each required reactor building cooling train fan unit for ≥ 15 minutes.	31 days
SR 3.6.5.3	Verify each required reactor building cooling train cooling water flow rate is ≥ 1200 gpm.	31 days
SR 3.6.5.4	Verify each required reactor building spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~
SR 3.6.5.5	Verify each automatic reactor building spray valve in each required flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.6.5.6	Verify each required reactor building spray pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.5.7	Verify each required reactor building cooling train starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.5.8	Verify each spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. --- Verify each required MSSV lift setpoint in accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~. Following testing, as-left lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~

Moved up from Page 3.7.1-3

Table 3.7.1-1
Allowable Power Level and RPS Nuclear Overpower Trip
Allowable Value versus OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVS OPERABLE (PER SG)	MAXIMUM ALLOWABLE POWER LEVEL (% RTP)	RPS NUCLEAR OVERPOWER TRIP ALLOWABLE VALUE (% RTP)
6	85.7	89.9
5	71.4	74.9
4	57.1	59.9
3	42.8	44.9
2	28.5	29.9

~~Table 3.7.1-1~~
~~Allowable Power Level and RPS Nuclear Overpower Trip~~
~~Allowable Value versus OPERABLE Main Steam Safety Valves~~

~~MINIMUM NUMBER OF MSSVS OPERABLE (PER SG)~~	~~MAXIMUM ALLOWABLE POWER LEVEL (% RTP)~~	~~RPS NUCLEAR OVERPOWER TRIP ALLOWABLE VALUE (% RTP)~~
~~6~~	~~85.7~~	~~89.9~~
~~5~~	~~71.4~~	~~74.9~~
~~4~~	~~57.1~~	~~59.9~~
~~3~~	~~42.8~~	~~44.9~~
~~2~~	~~28.5~~	~~29.9~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	-------------------------------NOTE---------------------------- Only required to be performed in MODES 1 and 2. --- Verify isolation time of each MSIV is within the limits specified in the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~.	In accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~
SR 3.7.2.2	-------------------------------NOTE---------------------------- 1. Only required to be performed in MODES 1 and 2. 2. Not required to be met when SG pressure is < 750 psig. --- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	18 months

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One Startup Feedwater Control Valve in one or more flow paths inoperable.	D.1 Close or isolate Startup Feedwater Control Valve. <u>AND</u> D.2 Verify Startup Feedwater Control Valve is closed or isolated.	72 hours Once per 7 days
E. Two valves in the same flow path inoperable for one or more flow paths.	E.1 Isolate affected flow path.	8 hours
F. Required Action and associated Completion Time not met.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	-------------------------------NOTE---------------------------- Only required to be performed in MODES 1 and 2. --- Verify the isolation time of each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve is within the limits provided in the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~.	In accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 18 hours
D. Two EFW trains inoperable in MODE 1, 2, or 3.	D.1 -------------NOTE---------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW train is restored to OPERABLE status. ----------------------------------- Initiate action to restore one EFW train to OPERABLE status.	 Immediately
E. Required EFW train inoperable in MODE 4.	E.1 Initiate action to restore EFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.5.1	Verify each EFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	------------------------------NOTE----------------------------- Not required to be performed for the turbine driven EFW pump, until 24 hours after reaching ≥ 750 psig in the steam generators. --- Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	 In accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel. Surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that during 10 year intervals all four reactor coolant pump flywheels will be examined. Such examinations will be performed to the extent possible through the access ports, i.e., those areas of the flywheel accessible without motor disassembly. The surface and volumetric examination may be accomplished by Acoustic Emission Examination as an initial examination method. Should the results of the Acoustic Emission Examination indicate that additional examination is necessary to ensure the structural integrity of the flywheel, then other appropriate NDE methods will be performed on the area of concern.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program inspection frequencies.

5.5.8 ~~Inservice Testing Program~~DELETED

~~This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:~~

~~a. Testing frequencies applicable to the ASME Code For Operation and Maintenance (OM) of Nuclear Power Plants and applicable Addenda as follows:~~

~~ASME OM Code terminology for inservice testing activities~~	~~Required Frequencies for performing inservice testing activities~~
~~Monthly~~	~~At least once per 31 days~~
~~Every 6 weeks~~	~~At least once per 42 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~~

Attachment 3

1CAN031604

Revised Technical Specification Pages

1.1 Definition (continued)

DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."
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. <u>Identified LEAKAGE</u>

1. LEAKAGE, such as that from pump seals or valve packing (except RCP seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. <u>Unidentified LEAKAGE</u>

All LEAKAGE (except RCP seal water injection and leakoff) that is not identified LEAKAGE;

c. <u>Pressure Boundary LEAKAGE</u>

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required pressurizer safety valve inoperable in MODE 3 or MODE 4 with RCS temperature > 259 °F.	C.1 Be in MODE 4 with RCS temperature ≤ 259 °F.	18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each required pressurizer safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, as-left lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	----------------------------NOTE--------------------------------- Not required to be performed in MODES 3 and 4. --- Verify leakage from each RCS pressure isolation check valve, or pair of check valves, as applicable, is less than or equal to an equivalent of the Allowable Leakage Limit identified below at a differential test pressure ≥ 150 psid. Pressure Isolation Check Valve(s) — Allowable Leakage Limit DH-14A — ≤ 5 gpm DH-13A and DH-17 — ≤ 5 gpm total DH-14B — ≤ 5 gpm DH-13B and DH-18 — ≤ 5 gpm total	In accordance with the INSERVICE TESTING PROGRAM AND Once prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months
SR 3.4.14.2	Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual high RCS pressure signal.	18 months
SR 3.4.14.3	Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual high RCS pressure signal: c. ≤ 340 psig for one valve; and d. ≤ 400 psig for the other valve.	18 months
SR 3.4.14.4	Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual Core Flood Tank isolation valve “not closed” signal.	18 months
SR 3.4.14.5	Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual Core Flood Tank isolation valve “not closed” signal.	18 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) temperature > 350 °F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Reduce RCS temperature to ≤ 350 °F.	6 hours 12 hours
C. Less than 100% of the ECCS flow equivalent to a single OPERABLE train available.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify each ECCS 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 3.5.2.2	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each reactor building purge isolation valve is closed.	31 days
SR 3.6.3.2	------------------------------NOTE---------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---	
	Verify each reactor building isolation manual valve and blind flange that is located outside the reactor building and not locked, sealed, or otherwise secured, and is required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.	31 days
SR 3.6.3.3	------------------------------NOTE---------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---	
	Verify each reactor building isolation manual valve and blind flange that is located inside the reactor building and not locked, sealed, or otherwise secured, and required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.4	Verify the isolation time of each automatic power operated reactor building isolation valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.3.5	Verify each automatic reactor building isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify each reactor building spray manual, power operated, and automatic valve in each required flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.5.2	Operate each required reactor building cooling train fan unit for $\geq$ 15 minutes.	31 days
SR 3.6.5.3	Verify each required reactor building cooling train cooling water flow rate is $\geq$ 1200 gpm.	31 days
SR 3.6.5.4	Verify each required reactor building spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.5.5	Verify each automatic reactor building spray valve in each required flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.6.5.6	Verify each required reactor building spray pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.5.7	Verify each required reactor building cooling train starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.5.8	Verify each spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	--------------------------------NOTE---------------------------- Only required to be performed in MODES 1 and 2. --- Verify each required MSSV lift setpoint in accordance with the INSERVICE TESTING PROGRAM. Following testing, as-left lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

Table 3.7.1-1
Allowable Power Level and RPS Nuclear Overpower Trip
Allowable Value versus OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVS OPERABLE (PER SG)	MAXIMUM ALLOWABLE POWER LEVEL (% RTP)	RPS NUCLEAR OVERPOWER TRIP ALLOWABLE VALUE (% RTP)
6	85.7	89.9
5	71.4	74.9
4	57.1	59.9
3	42.8	44.9
2	28.5	29.9

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	--------------------------------NOTE----------------------------- Only required to be performed in MODES 1 and 2. --- Verify isolation time of each MSIV is within the limits specified in the INSERVICE TESTING PROGRAM.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.2.2	--------------------------------NOTE----------------------------- 1. Only required to be performed in MODES 1 and 2. 2. Not required to be met when SG pressure is < 750 psig. --- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	18 months

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One Startup Feedwater Control Valve in one or more flow paths inoperable.	D.1 Close or isolate Startup Feedwater Control Valve.	72 hours
	AND	
	D.2 Verify Startup Feedwater Control Valve is closed or isolated.	Once per 7 days
E. Two valves in the same flow path inoperable for one or more flow paths.	E.1 Isolate affected flow path.	8 hours
F. Required Action and associated Completion Time not met.	F.1 Be in MODE 3.	6 hours
	AND	
	F.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	-------------------------------NOTE---------------------------- Only required to be performed in MODES 1 and 2. --- Verify the isolation time of each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve is within the limits provided in the INSERVICE TESTING PROGRAM.	In accordance with the INSERVICE TESTING PROGRAM

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 18 hours
D. Two EFW trains inoperable in MODE 1, 2, or 3.	D.1 -------------NOTE---------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW train is restored to OPERABLE status. ----------------------------------- Initiate action to restore one EFW train to OPERABLE status.	 Immediately
E. Required EFW train inoperable in MODE 4.	E.1 Initiate action to restore EFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each EFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	-------------------------------NOTE------------------------------ Not required to be performed for the turbine driven EFW pump, until 24 hours after reaching ≥ 750 psig in the steam generators. --- Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	 In accordance with the INSERVICE TESTING PROGRAM

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel. Surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that during 10 year intervals all four reactor coolant pump flywheels will be examined. Such examinations will be performed to the extent possible through the access ports, i.e., those areas of the flywheel accessible without motor disassembly. The surface and volumetric examination may be accomplished by Acoustic Emission Examination as an initial examination method. Should the results of the Acoustic Emission Examination indicate that additional examination is necessary to ensure the structural integrity of the flywheel, then other appropriate NDE methods will be performed on the area of concern.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program inspection frequencies.

5.5.8 DELETED

Attachment 4

1CAN031604

Proposed Technical Specification Bases Changes (Mark-Up) – Information Only

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~. Pressurizer safety valves are to be tested in accordance with the requirements of the ASME OM Code (Ref. 6), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is + 1%, - 3% for OPERABILITY (Ref. 7); however, the valves are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. SAR, Section 4.2.4.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1968.
3. SAR, Section 4.3.8.
4. SAR, Section 4.3.11.4.
5. 10 CFR 50.36, Technical Specifications.
6. ASME, Boiler and Pressure Vessel Code, Section XI.
7. ASME OM Code - 2001 Edition through 2003 Addenda and Code Case OMN-20 (Inservice Test Frequency).

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on RCS pressure isolation check valve(s) is required to verify that leakage is below the specified limit and to identify leaking valve(s). The leakage limit of 5 gpm maximum applies to each isolation check valve which is closest to the reactor vessel in the DHR System injection lines (DH-14A and DH-14B) and to each parallel pair of check valves which protect an individual low pressure injection line (total for DH-13A and DH-17, and total for DH-13B and DH-18). Leakage testing requires a stable pressure condition. Reference 5 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to account for the maximum pressure differential by assuming leakage is directly proportional to the square root of the pressure differential.

If the in series PIVs are not separately 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 in series valves would be lost.

Testing is to be performed on a Frequency consistent with 10 CFR 50.55a(f) (Ref. 6) as contained in the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~, and allowed by the ASME OM Code (Ref. 5). This Frequency is based on the need to perform such surveillances under conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the unit at power.

The leakage surveillance is to be performed at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complimentary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.14.2, SR 3.4.14.3, SR 3.4.14.4, and SR 3.4.14.5

Verifying that the DHR autoclosure interlocks are OPERABLE ensures that RCS pressure will not over pressurize the DHR system. The interlock(s) that prevent the valves from being opened and that close the valves are designed to protect the DHR System from gross overpressurization. Although the specified values include certain process measurement uncertainties, additional allowances for instrument uncertainty are contained in the implementing procedures. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and on the potential for an unplanned transient if the Surveillance was performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

REFERENCES

1. "Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves," issued April 20, 1981.

2. NUREG-75/014, Reactor Safety Study, Appendix V, October 1975.

3. NUREG-0677, The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes, May 1980.

4. 10 CFR 50.36.

5. ASME OM Code 2001 Edition through 2003 Addenda and Code Case OMN-20 (Inservice Test Frequency).

6. 10 CFR 50.55a(f).

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

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 actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an inoperable valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.2

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME OM Code (Ref. 7). This testing confirms component OPERABILITY, trends performance, and detects incipient failures by indicating abnormal performance. SRs are specified in the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~, which encompasses the ASME OM Code.

SR 3.5.2.3

This SR demonstrates that each automatic ECCS valve actuates to the required position on an actual or simulated ESAS signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the ESAS testing, and equipment performance is monitored as part of the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~.

SR 3.5.2.4

The intent of this SR is to verify that the ECCS pumps are capable of automatically starting on an ESAS signal. Because of the system design configuration and the limitations imposed on pump operation during the unit conditions when this test would be conducted, this verification must be conducted through a series of sequential, overlapping or total steps in order to demonstrate functionality. SR 3.5.2.4 demonstrates that each ECCS pump would be capable of starting by verifying that its breaker closes on receipt of an actual or simulated ESAS signal. SR 3.5.2.4 works in conjunction with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~ (SR 3.5.2.2) which periodically verifies the ability of the pumps to start and operate within limits, and the ESAS actuation logic testing which periodically verifies the ability of the ESAS to sense, process and generate an actuation signal.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.4 (continued)

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

SR 3.5.2.5

Periodic inspections of the reactor building sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance during a unit outage. Operating experience has shown this Frequency to be acceptable to detect abnormal degradation.

REFERENCES

1. SAR, Section 6.
2. Letter from A. C. Thadani (NRC) to P. S. Walsh (BWOG) dated March 9, 1993.
3. 10 CFR 50.46.
4. SAR, Section 14.2.2.5.2.
5. 10 CFR 50.36.
6. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
7. ASME OM Code 2001 Edition through 2003 Addenda and Code Case OMN-20 (Inservice Test Frequency).
8. Condition Report CR-ANO-1-2009-0997.

LCO

Reactor Building isolation valves form a part of the reactor building boundary. The reactor building isolation valve safety function is related to minimizing the loss of reactor coolant inventory and establishing the reactor building boundary during a DBA.

The automatic power operated reactor building isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 24-inch purge valves must be maintained closed. The valves covered by this LCO are listed in the SAR (Ref. 4). Their associated stroke times are contained in the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~. The normally closed manual reactor building isolation valves are considered OPERABLE when the valves are closed, blind flanges are in place, or open under administrative controls. These passive isolation valves/devices are listed in Reference 4.

The reactor building isolation valve leakage rates are addressed by LCO 3.6.1, "Reactor Building," as Type C testing.

This LCO provides assurance that the reactor building isolation valves will perform their designated safety functions to minimize the loss of reactor coolant inventory and establish the reactor building boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3 and 4, the reactor building isolation valves OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. In MODES 5 and 6, the probability and consequences of events are reduced due to the pressure and temperature limitations of these MODES. The requirements for reactor building isolation valves during MODE 5 and 6, primarily related to movement of irradiated fuel in the reactor building, are addressed in LCO 3.9.3, "Reactor Building Penetrations."

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, except for purge valve penetration flow paths, to be unisolated intermittently under administrative controls, consistent with Generic Letter (GL) 91-08 (Ref. 5). These administrative controls consist of stationing a dedicated individual at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for reactor building isolation is indicated. Due to ALARA concerns, it is permissible for this dedicated individual to be stationed in a nearby lower dose area provided the intent of rapidly isolating the penetration is retained. Due to the size of the reactor building purge line penetration and the fact that those penetrations exhaust directly from the reactor building atmosphere to the environment, the penetration flow paths containing these valves may not be opened under administrative controls.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.2 (continued)

The SR specifies that the reactor building isolation valves open under administrative controls are not required to meet the SR during the time the valves are open. 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.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these reactor building isolation valves, once they have been verified to be in the proper position, is low.

SR 3.6.3.3

This SR requires verification that each reactor building isolation manual valve and blind flange that is located inside the reactor building and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the reactor building boundary is within design limits. For reactor building isolation valves inside reactor building, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these reactor building isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that reactor building isolation valves open under administrative controls are not required to meet the SR during the time they are open. 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.

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these reactor building isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.4

Verifying that the isolation time of each automatic power operated reactor building isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period consistent with the industry standards for sizing valve operators. The isolation time and Frequency of this SR are in accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.4

Verifying that each required reactor building spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow rate are within ± 10 % of a point on the pump head curve. Flow and differential pressure are measured during normal tests of centrifugal pump performance required by the ASME OM Code (Ref. 5). Since the Reactor Building Spray System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms the discharge pressure and flow rate are within ± 10 % of a point on the pump head curve and is indicative of overall pump performance. Such inservice tests confirm component OPERABILITY, trend performance, and may detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~.

SR 3.6.5.5 and SR 3.6.5.6

These SRs require verification that each automatic reactor building spray valve actuates to its correct position and that each reactor building spray pump starts upon receipt of an actual or simulated actuation signal. The SRs are considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal properly. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. During testing of the spray pump, the reactor building isolation valve in the spray line is closed with its breaker open to prevent spraying the reactor building. After spray pump performance is verified, the pump is stopped. Its breaker is racked down to prevent restart. Power is then restored to the reactor building isolation valve for valve testing. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.5.7

This SR requires verification by control board indication that each required reactor building cooling train actuates upon receipt of an actual or simulated actuation signal. The 18 month Frequency has been shown to be acceptable through operating experience. See SR 3.6.5.5 and SR 3.6.5.6, above, for further discussion of the basis for the 18 month Frequency.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.8

This surveillance ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Confirmation that the spray nozzles are unobstructed may be obtained by such means as foreign materials exclusion (FME) controls during maintenance, a visual inspection of the affected portions of the system, by an air or smoke flow test following maintenance involving opening portions of the system downstream of the containment isolation valves, or by draining/flushing the filled portions of the system inside containment, as appropriate. Maintenance that could result in nozzle blockage is generally a result of a loss of FME control. If loss of FME control occurs, an inspection or flush of the affected portions of the system should be adequate to confirm that the spray nozzles are unobstructed since water flow would be required to transport any debris to the spray nozzles.

REFERENCES

1. SAR, Section 1.4.

2. SAR, Chapter 6.

3. SAR, Chapter 14.

4. 10 CFR 50.36.

5. ASME OM Code 2001 Edition through 2003 Addenda and Code Case OMN-20 (Inservice Test Frequency).

6. Condition Report CR-ANO-1-2009-0997.

LCO

The MSSVs are provided to prevent overpressurization as discussed in the Applicable Safety Analysis section of these Bases. The LCO requires fourteen MSSVs (seven on each main steam line) to be OPERABLE to ensure compliance with the ASME Code following DBAs initiated at full power. Operation with less than the required complement of MSSVs requires a limitation on unit THERMAL POWER and adjustment of the Reactor Protection System (RPS) nuclear overpower trip setpoint. The minimum number of OPERABLE MSSVs per steam generator for various power levels and the associated maximum allowable nuclear overpower trip setpoint are identified in Table 3.7.1-1. This effectively limits the Main Steam System steam flow while the MSSV relieving capacity is reduced due to valve inoperability. To be OPERABLE, lift setpoints must remain within limits, according to SR 3.7.1.1.

The safety function of the MSSVs is to open, relieve steam generator overpressure, and reseat when pressure has been reduced.

OPERABILITY of the MSSVs requires periodic surveillance testing in accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~.

With all MSSVs OPERABLE, at least one MSSV per steam generator is set at 1050 psig nominal, while the remaining MSSVs per steam generator are set at varied pressures up to and including 1100 psig nominal. The lift settings correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform the design safety function.

The LCO is modified by a Note that allows all but one MSSV on each main steam header to be gagged and the setpoints for the two (one on each header) OPERABLE MSSVs to be reset for the duration of hydrotesting in MODE 3. This is necessary to allow the hydrotest pressure to be attained.

APPLICABILITY

In MODES 1, 2, and 3, the MSSVs are required to be OPERABLE to prevent overpressurization of the main steam system.

In MODES 4 and 5, there is no credible transient requiring the MSSVs.

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized. There is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of MSSV lift setpoints in accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~. The safety and relief valve tests are performed in accordance with ASME OM Code (Ref. 6) and include the following for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

The ASME OM Code requires the testing of all valves every 5 years, with a minimum of 20% of the valves tested every 24 months. Reference 6 provides the activities and frequencies necessary to satisfy the requirements and allows an as-found Owner- established setpoint tolerance per the table below:

Owner-Established Allowable MSSV Setpoint Tolerance

MSSV Nominal Setpoint	Allowable Tolerance
1050	+ 3.0% / - 3.0%*
1060	+ 3.0% / - 3.9%*
1070	+ 3.0% / - 4.8%*
1090	+ 3.0% / - 6.5%*
1100	+ 3.0% / - 7.4%*

* See Reference 7 for Allowable Negative Setpoint Tolerance

Although not required by the IST Program, the valves are reset to ± 1% during the Surveillance to allow for drift.

If during testing one or more valves are found to be outside a lift setting tolerance band of ± 1%, but within a lift setting tolerance band per the Owner-Established Allowable MSSV Setpoint Tolerance, the valves remain OPERABLE. If found outside the lift setting tolerance band of the Owner-Established Allowable MSSV Setpoint Tolerance, the valve is inoperable. If the valve is restored to within the Owner-Established Allowable MSSV Setpoint Tolerance, the valve may be considered OPERABLE. However, the valve(s) must be reset to within a lift setting tolerance band of ± 1% prior to test completion; otherwise, the valve(s) should be considered inoperable. Resetting the valve to within a lift setting tolerance band of ± 1% is required to account for drift that may occur between test intervals.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.1.1 (continued)

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. SAR, Section 10.3.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.

3. SAR, Chapter 14.

4. AREVA/FANP Document 77-5018959-01, “ANO-1 Overpressure Protection Report.”

5. 10 CFR 50.36.

6. ASME OM Code –2001 through 2003 Addenda and Code Case OMN-20 (Inservice Test Frequency).

7. EC-26246.

ACTIONS (continued)

B.1

If the Required Action and associated Completion Time of Condition A are not met, the unit must be placed in MODE 3 within the next 12 hours. The Completion Time is reasonable, based on operating experience, to reach MODE 3.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODE 3, the inoperable MSIV(s) may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The main steam and feedwater systems do not provide a direct path from the reactor building atmosphere to the environment. Therefore, the Completion Time is reasonable, and provides for diagnosis and repair of many MSIV problems, thereby avoiding unnecessary shutdown.

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable in view of MSIV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position.

D.1 and D.2

If the Required Actions and associated Completion Times of Condition C are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 4 within 24 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from MODE 3 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is as specified in theINSERVICE TESTING PROGRAM ~~Inservice Testing Program~~. The MSIV isolation time is assumed in the accident and reactor building analyses. This Surveillance is normally performed prior to returning the unit to power operation, e.g., during MODE 3, following a refueling outage, because the MSIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power.

The Frequency for this SR is in accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve is as specified in the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~.

The MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve isolation time is assumed in the accident and reactor building analyses. This Surveillance is normally performed prior to returning the unit to power operation, e.g., during MODE 3, following a refueling outage. The MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves are not tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR.

The Frequency for this SR is in accordance with the INSERVICE TESTING PROGRAM~~Inservice Testing Program~~.

SR 3.7.3.2

This SR verifies that each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage.

The Frequency for this SR is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

This SR is modified by two Notes. The first Note allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was established.

SR 3.7.3.2 is also modified by a second Note which indicates that the automatic closure capability is not required to be met when the SG pressure is < 750 psig. At < 750 psig, the main steam line isolation Function of EFIC may be disabled to prevent automatic actuation on low SG pressure during a unit shutdown.

ACTIONS (continued)

E.1

In MODE 4, either the steam generator loops or the DHR loops can be used to provide heat removal, which is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With the required EFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the EFW water and steam supply flow paths provides assurance that the proper flow paths exist for EFW operation. Correct alignment for automatic valves may be other than the post-accident position provided the valve is otherwise OPERABLE. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since those valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance 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.

The 31 day Frequency is based on the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each EFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that EFW pump performance has not degraded below the established acceptance criteria during the cycle. Flow and differential head are indicators of pump performance required by the ASME OM Code (Ref. 5). Because it is undesirable to introduce cold EFW into the steam generators while they are operating, this test may be performed on a test flow path.

This test is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing as discussed in the ASME OM Code (Ref. 5) and the INSERVICE TESTING PROGRAM satisfies this requirement.

This SR is modified by a Note indicating that the SR may be deferred until suitable test conditions are established. This deferral is required because there may be insufficient steam pressure to perform the test.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.6

This SR ensures that the EFW flowpath to each steam generator is open and that water reaches the steam generators from the EFW system. This test is performed during shutdown to minimize thermal cycles to the emergency feedwater nozzles on the steam generator due to the lower temperature of the emergency feedwater. The motor-driven EFW pump is specified because of its availability at the low steam generator pressure conditions that exist in the shutdown condition. The 18-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. SAR, Section 7.1.4.

2. SAR, Section 10.4.8.

3. NRC Letter dated January 12, 1981, (1CNA018103).

4. 10 CFR 50.36.

5. ASME OM Code 2001 Edition through 2003 Addenda and Code Case OMN-20 (Inservice Test Frequency).

Attachment 5

1CAN031604

Description and Assessment of the Proposed Alternative to the ASME Code

DESCRIPTION AND ASSESSMENT OF THE PROPOSED ALTERNATIVE TO THE ASME CODE

Request in Accordance with 10 CFR 50.55a(z)(2)

Alternative Due To Hardship Without a Compensating Increase in Quality and Safety

1.0 DESCRIPTION

The request is to adopt a proposed alternative to the American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code by adoption of approved Code Case OMN-20, "Inservice Test Frequency."

2.0 ASSESSMENT

Technical Evaluation of the Proposed Alternative to the OM Code

Section IST of Division 1 of the OM Code, which is incorporated by reference in 10 CFR 50.55a(a), specifies component test frequencies based either on elapsed time periods (e.g., quarterly, 2 years) or on the occurrence of a plant condition or event (e.g., cold shutdown, refueling outage).

ASME Code Case OMN-20, “Inservice Test Frequency,” has been approved for use by the ASME OM committee as an alternative to the test frequencies for pumps and valves specified in ASME OM Division: 1 Section IST 2009 Edition through OMa-2011 Addenda, and all earlier editions and addenda of ASME OM Code.

Code Case OMN-20 is not referenced in the latest revision of Regulatory Guide 1.192 (August 2014) as an acceptable OM Code Case to comply with 10 CFR 50.55a(f) requirements as allowed by 10 CFR 50.55a(b)(6). The proposed alternative is to use Code Case OMN-20 to extend or reduce the IST frequency requirements for the fourth 10-year IST interval or until OMN-20 is incorporated into the next revision of Regulatory Guide 1.192.

ASME Code Components Affected

The Code Case applies to pumps and valves specified in ASME OM Division: 1 Section IST 2009 Edition through OMa-2011 Addenda and all earlier editions and addenda of ASME OM Code. Frequency extensions may also be applied to accelerated test frequencies (e.g., pumps in Alert Range) as specified in OMN-20.

For pumps and valves with test periods of 2 years of less, the test frequency allowed by OMN-20 and the current Technical Specification (TS) Inservice Testing Program (as modified by SR 3.0.2 and EGM 2012-001) are the same. For pumps and valves with test frequencies greater than 2 years, OMN-20 allows the test frequency to be extended by 6 months. The current TS Inservice Testing (IST) Program does not allow extension of test frequencies that are greater than 2 years.

Applicable Code Edition and Addenda

ASME Code Case OMN-20 applies to ASME OM Division: 1 Section IST 2009 Edition through OMa-2011Addenda and all earlier editions and addenda of ASME OM Code.

The Arkansas Nuclear One, Unit 1 (ANO-1) Code Edition and Addenda that are applicable to the program interval are the ASME OM Code 2001 Edition through 2003 Addenda (reference Entergy Letter CNRO-2007-00044 dated November 30, 2007, "Inservice Testing Plan," ADAMS Accession No. ML073410350). The ANO-1 current interval ends November 30, 2017.

Applicable Code Requirement

This request is made in accordance with 10 CFR 50.55a(z)(2), and proposes an alternative to the requirements of 10 CFR 50.55a(f), which requires pumps and valves to meet the test requirements set forth in specific documents incorporated by reference in 10 CFR 50.55a(a). ASME Code Case OMN-20 applies to Division 1, Section IST of the ASME OM Code and associated addenda incorporated by reference in 10 CFR 50.55a(a).

Reason for Request

The IST Program controls specified in Section 5.5 of TS provide: a) a table specifying certain IST frequencies; b) an allowance to apply SR 3.0.2 to inservice tests required by the OM Code and with frequencies of two years or less; c) an allowance to apply SR 3.0.3 to inservice tests required by the OM Code; and d) a statement that, "Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS." In Regulatory Issue Summary (RIS) 2012-10, "NRC Staff Position on Applying Surveillance Requirement 3.0.2 and 3.0.3 to Administrative Controls Program Tests," and Enforcement Guidance Memorandum (EGM) 2012-001, "Dispositioning Noncompliance with Administrative Controls Technical Specifications Programmatic Requirements that Extend Test Frequencies and Allow Performance of Missed Tests," the NRC stated that items b, c, and d of the TS IST Program were inappropriately added to the TS and may not be applied (although the EGM allows licensees to continue to apply those paragraphs pending a generic resolution of the issue).

In RIS 2012-10 and EGM 2012-001, the NRC stated that the current TS allowance to apply Surveillance Requirements (SR) 3.0.2 and SR 3.0.3 to the IST Program would no longer be permitted. In response, OMN-20, which provides allowances similar to SR 3.0.2, was approved and is proposed to be used as an alternative to the test periods specified in the OM code. The proposed alternative substitutes an approved Code Case for the existing TS requirements that the NRC has determined are not legally acceptable as a TS allowance. This proposed alternative provides an equivalent level of safety as the existing TS allowance, while maintaining consistency with 10 CFR 50.55a and the ASME OM Code.

Proposed Alternative and Basis for Use

The proposed alternative is OMN-20, "Inservice Test Frequency," which addresses testing periods for pumps and valves specified in ASME OM Division 1, Section IST, 2009 Edition through OMa-2011 Addenda, and all earlier editions and addenda of ASME OM Code.

This request is being made in accordance with 10 CFR 50.55a(z)(2), in that the existing requirements are considered a hardship without a compensating increase in quality and safety for the following reasons:

1) For IST testing periods up to and including 2 years, Code Case OMN-20 provides an allowance to extend the IST testing periods by up to 25%. The period extension is to facilitate test scheduling and considers plant operating conditions that may not be suitable for performance of the required testing (e.g., performance of the test would cause an unacceptable increase in the plant risk profile due to transient conditions or other ongoing surveillance, test or maintenance activities). Period extensions are not intended to be used repeatedly merely as an operational convenience to extend test intervals beyond those specified. The test period extension and the statements regarding the appropriate use of the period extension are equivalent to the existing TS SR 3.0.2 allowance and the statements regarding its use in the SR 3.0.2 Bases. Use of the SR 3.0.2 period extension has been a practice in the nuclear industry for many decades and elimination of this allowance would place a hardship on Entergy Operations, Inc. (Entergy), when there is no evidence that the period extensions affect component reliability.

2) For IST testing periods of greater than 2 years, OMN-20 allows an extension of up to 6 months. The ASME OM Committee determined that such an extension is appropriate. The 6-month extension will have a minimal impact on component reliability considering that the most probable result of performing any inservice test is satisfactory verification of the test acceptance criteria. As such, pumps and valves will continue to be adequately assessed for operational readiness when tested in accordance with the requirements specified in 10 CFR 50.55a(f) with the frequency extensions allowed by Code Case OMN-20.

3) As stated in EGM 2012-001, if an inservice test is not performed within its frequency, SR 3.0.3 will not be applied. The effect of a missed inservice test on the operability of TS equipment will be assessed under the licensee's Operability Determination Program.

Duration of Proposed Alternative

The proposed alternative is requested for the current 10-year IST interval or until Code Case OMN-20 is incorporated into a future revision of Regulatory Guide 1.192, referenced by a future revision of 10 CFR 50.55a, whichever occurs first.

Precedents

The NRC approved the use of OMN-20 for North Anna on March 27, 2014 (NRC ADAMS Accession Number ML14084A407).