Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

JUN 16 2006

WBN-TS-06-04

10 CFR 50.90

U. S. Nuclear Regulatory Commission
Mail Stop: OFWN P1-35
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket No. 50-390
Tennessee Valley Authority	)	

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - PROPOSED TECHNICAL SPECIFICATION CHANGE WBN-TS-06-04 - AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) CODE TO SUPPORT SECOND INTERVAL OF INSERVICE TESTING PROGRAM

Pursuant to 10 CFR 50.90, TVA is submitting a request for a TS change (WBN-TS-06-04) to license NPF-90 for WBN Unit 1. The proposed TS change removes "applicable supports" from the Inservice Testing (IST) Program and revises the IST Program for pumps and valves to meet the requirements of the latest Edition and Addenda of the ASME Code approved by the NRC for use on the date 12-months prior to the start of the 10-year Inservice Testing Interval. For WBN Unit 1, the second 10-year Inservice Testing Interval will begin December 27, 2006. The ASME Code that was approved in the 10 CFR 50.55a(f)(4) for use on December 27, 2005, was ASME Operations and Maintenance (OM) Code, 2001 Edition, with Addenda through 2003. The proposed change provides consistency with the requirements in 10 CFR 50.55a(f)(4) by replacing the reference to ASME Boiler and Pressure Vessel Code, Section XI, with ASME OM Code. This proposed change is based on Technical Specification Task Force (TSTF) 479, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a." TSTF 279-A, Revision 0, "Remove 'applicable supports' from Inservice Testing Program," was approved by NRC and incorporated into Revision 2 of NUREG-1431, "Standard Technical Specification Westinghouse Plants."

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Enclosure 1 provides a description of the proposed change and confirmation of applicability. Enclosure 2 provides the existing TS pages marked-up to show the proposed change. Enclosure 3 provides the applicable TS Bases pages associated with the TS change provided for information.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the TS change qualifies for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosures to the Tennessee State Department of Public Health.

TVA request approval of this TS change to support a December 27, 2006 implementation date of WBN's second 10-year Inservice Testing Interval. Implementation of TVA's proposed TS change is requested to occur in conjunction with the December 27, 2006 Inservice Testing Program implementation date.

There are no regulatory commitments associated with this submittal. If you have any questions concerning this matter, please call me at (423) 365-1824.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 16th day of June 2006.

Sincerely,

P. L. Pace
Manager, Site Licensing
and Industry Affairs

Enclosures:
1. TVA Evaluation of the Proposed Changes
2. Proposed Technical Specification Changes (mark-up)
3. Proposed Technical Specification Changes (Revised)
4. Changes to Technical Specifications Bases pages

cc: See page 3

JUN 16 2006

Enclosures
cc (Enclosures):

NRC Resident Inspector
Watts Bar Nuclear Plant
1260 Nuclear Plant Road
Spring City, Tennessee 37381

Mr. D. V. Pickett, Senior Project Manager
U.S. Nuclear Regulatory Commission
MS 08G9a
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852-2738

U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303

Mr. Lawrence E. Nanny, Director
Division of Radiological Health
3rd Floor
L & C Annex
401 Church Street
Nashville, Tennessee 37243

ENCLOSURE

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
DOCKET NO. 50-390

PROPOSED LICENSE AMENDMENT REQUEST WBN-TS-06-04
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-90 for WBN Unit 1. The proposed changes would revise the Operating License to delete "applicable supports" from the pump and valve testing program and reference the American Society of Mechanical Engineers Code for Operation and Maintenance of Nuclear Power Plant (ASME OM Code), in the Technical Specifications Administrative Programs instead of ASME Boiler and Pressure Vessel Code, Section XI as the source document for requirements for inservice testing of pumps and valves classified as ASME Code Class 1, 2, and 3. WBN will update the Second Inservice Testing Interval Program to the requirements of the latest Code endorsed by 10 CFR 50.55a(f), (Reference 1). 10 CFR 50.55a now endorses the ASME OM Code (Reference 2) rather than ASME Section XI for inservice testing requirements of pumps and valves. Therefore, it is necessary to revise the technical specifications to recognize the ASME OM Code instead of the ASME Section XI Code. This proposed change is consistent with Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-479, Revision 0, (Reference 3) and TSTF-279-A, Revision 0 (Reference 4). TVA is not proposing any variations or deviations from the technical specification changes described in TSTF-279, Revision 0 and TSTF-479-A, Revision 0. In addition to the TSTFs, TVA has added provisions in Section 5.7.2.11, Item b, to only apply Surveillance Requirement (SR) 3.0.2 to those Inservice Testing Frequencies of two years or less.

2.0 PROPOSED CHANGE

Technical Specification

The following changes are proposed for the WBN Technical Specifications:

- Section 5.7.2.11 - Delete "including applicable supports" in the first sentence.

- Section 5.7.2.11.a - Replace the reference to Section XI of the ASME of the ASME Boiler and Pressure Vessel Code with a reference to the ASME OM Code.

- Section 5.7.2.11.b - Applies SR 3.0.2 to the non-standard Frequencies of two years or less.

- Section 5.7.2.11.d - Replace the reference to Section XI of the ASME of the ASME Boiler and Pressure Vessel Code with a reference to the ASME OM Code.

In summary, these changes reflect changes in the source of Inservice Testing requirements. When WBN begins its Second Inservice Interval, it will be required to implement the inservice testing requirements of the Code edition and addenda referenced in 10 CFR 50.55a twelve months prior to the start of the next 10-Year Interval. The Code has been revised to require inservice testing of pumps and valves to be performed in accordance with the ASME OM Code instead of Section XI of the ASME Boiler and Pressure Vessel Code. This change in 10 CFR 50.55a reflects an organization change in the ASME Code structure. This is essentially an administrative change to update the references for Inservice Testing Program to be consistent with the requirements referenced in 10 CFR 50.55a(f)(4). The reference to "applicable supports" is being deleted from the Inservice Testing Program as these supports are already addressed in the Inservice Inspection Program. These TSTF changes are consistent with the Standard Technical Specification with the exception of the Frequency of two years or less added in Section 5.7.2.11.b.

Technical Specification Bases

The following changes to the Bases of the WBN Technical Specifications are provided for information only. These changes are required in order to support and implement the proposed change to the Technical Specifications:

- Bases Section 3.4.10 - Revise SR 3.4.10.1 to replace "Section XI of the ASME Code" with "the ASME OM Code," and replace Reference 4 with ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

- Bases Section 3.4.11 - Revise SR 3.4.11.1 to replace "Section XI of the ASME Code" with "the ASME OM Code," and replace Reference 3 with ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

- Bases Section 3.4.12 - Replace Reference 8 with ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

- Bases Section 3.4.14 - Revise SR 3.4.14.1 to replace "American Society of Mechanical Engineers (ASME) Code, Section XI" with "the ASME OM Code," and replace Reference 7 with ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

- Bases Section 3.5.2 - Revise SR 3.5.2.4 to replace "Section XI of the ASME Code" with "the ASME OM Code."

- Bases Section 3.6.6 - Revise SR 3.6.6.2 to replace "Section XI of the ASME Code" with "the ASME OM Code," and replace Reference 4 with ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

- Bases Section 3.7.1 - Revise SR 3.7.1.1 to replace "ASME Code, Section XI," ANSI/ASME OM-1-1987, and Reference 4 with "the ASME OM Code." Also Delete Reference 5.

- Bases Section 3.7.2 - Replace Reference 5 with ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

- Bases Section 3.7.3 - Revise SR 3.7.3.1 to replace "ASME Code Section XI" with "the ASME OM Code," and replace Reference 2 with ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

- Bases Section 3.7.5 - Revise SR 3.7.5.2 to replace "Section XI of the ASME Code" with "the ASME OM Code," and replace Reference 2 with ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

3.0 BACKGROUND

ASME historically included requirements for the inservice testing of pumps and valves in Section XI of the Boiler and Pressure Vessel Code. Over the course of several years, these requirements transitioned into national standards and then into a separate, stand alone code entitled the "Code for Operation and Maintenance of Nuclear Power Plants," or the ASME OM Code. For a period of time, ASME maintained a reference to the OM Standards and later the OM Code in Section XI. However, that reference no longer exists.

As required by 10 CFR 50.55a, WBN must update the Inservice Testing Program to meet the requirements of the Code currently endorsed in 10 CFR 50.55a(f)(4) for the Second Inservice Interval. The Code in 10 CFR 50.55a has been updated to reflect the ASME OM Code as the requirements document for inservice testing. Therefore, inservice testing during the Second Inservice Interval at WBN must be performed in accordance with the requirements of ASME OM Code as required in 10 CFR 50.55a(f)(4)(ii). This change to the WBN Technical Specifications is necessary to support this update.

The proposed changes are consistent with the Standard Technical Specification Change Traveler, TSTF-479, Revision 0. NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," Revision 3.1 incorporated TSTF-479, Revision 0. The NRC issued a letter to the TSTF on December 6, 2005, indicating the acceptability of TSTF-479, Revision 0. TVA is aware that since that time, NRC has expressed a concern that frequency extensions may be applied to Frequencies greater than two years and has requested that the TSTF be revised to apply the provisions of SR 3.0.2 to Frequencies of two years or less only. Based on NRC's position for applying SR 3.0.2, TVA also included the two years or less in Section 5.7.2.11.b

The ASME Section XI requirements for Inservice Testing of pumps and valves are discussed in the Updated Final Safety Analysis Report (UFSAR) Sections 3.9.6, 5.5.12, 6.2.6, and 6.3.4 and sections of the Technical Requirements Manual. The current Inservice Testing Program is based upon the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition. The appropriate sections will be revised to change the reference from ASME Section XI to the new ASME OM Code for pump and valve testing when the Second Inservice Testing Interval is implemented on December 27, 2006.

In February 1992, NRC revised the Improved Standard Technical Specification to include the words "applicable supports," due to concerns related to the relocation of the Snubber Program from the Technical Specification. However, the Snubber Program is under the Inservice Inspection Program and not the Inservice Testing Program. In a letter dated July 16, 1998, NRC approved TSTF-279-A, Revision 0 for the deletion of "applicable supports" from the Inservice Testing Program which was subsequently incorporated into Revision 2 of NUREG 1431. This change revises the WBN Technical Specification to be consistent with the Standard Technical Specification.

4.0 <u>TECHNICAL ANALYSIS</u>

ASME has in the last several years, transitioned the requirements for inservice testing of pumps and valves out of Section XI and into a separate, stand alone code entitled the "Code for Operation and Maintenance of Nuclear Power Plants," [ASME OM Code]. The ASME OM Code has been endorsed by the NRC in 10 CFR 50.55a and is the code that will be required for inservice testing of pumps and valves during the WBN Second Inservice Interval. This technical specification change is necessary to support implementation of the Second Inservice Interval requirements.

The testing requirements of the new ASME OM Code are not a direct one for one replacement for the testing requirements of the previous ASME Section XI; however, the differences reflect the evolving and improving nature of a National Standard. Improvements have been incorporated that permit additional flexibility in application while still maintaining a sound technical basis to support continued acceptability of the components being tested.

Although different, the testing requirements are similar and reflect the same type testing. That is, valves are still stroke timed; remote position indicators are still verified to be accurate; seat leakage measurements of critical valves are still performed; relief valves still have their setpoints and seat leakages verified; pumps are still tested for hydraulic performance and mechanical condition; and check valves are verified to open and close properly. These testing requirements implement the purpose of the Inservice Testing Program which is to ensure operability of pumps and valves, detect degradation of the pumps and valves, and to maintain an adequate safety margin.

Incorporation of the ASME OM Code in lieu of ASME Section XI as the source document for requirements for inservice testing of pumps and valves does not alter or change the analytical methods used in the safety analyses or UFSAR accident analyses. Neither does it alter the nature or type of accidents requiring consideration. Limiting values and acceptance criteria used to judge the continued acceptability of the components tested are not changed by the incorporation of the later code. This is considered an administrative change to update to the latest code for testing pumps and valves as required in 10 CFR 50.55a(f)(4)(ii), and therefore, is acceptable from a nuclear safety standpoint.

To be consistent with NUREG-1431, the words "including applicable supports" are being deleted from Technical Specification 5.7.2.11. The reference to "applicable supports" in the pump and valve testing program is inappropriate as the supports are currently addressed in the Inservice Inspection Program. The ASME Program must be applied to the applicable Code Class 1, 2, and 3 components, therefore, NRC approved the removal of supports from the Inservice Testing Program in a letter dated July 16, 1998. This change was subsequently incorporated into NUREG-1431, Revision 2. TVA's Inservice Inspection Program for supports is addressed in the WBN Technical Requirements Manual.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

The American Society of Mechanical Engineers (ASME) has removed the testing requirements for pumps and valves out of ASME Section XI into a stand alone code entitled "ASME OM Code." TVA will be required to update to the latest Code requirements in 10 CFR 50.55a in accordance with 10 CFR 50.55a(f)(4)(ii) at the beginning of the Second Inservice Testing Interval. This proposed technical specification is an administrative change which updates the reference from ASME Section XI to ASME OM Code for the Second Ten-Year Testing Interval for pumps and valves. The proposed change implements the Technical Specification Task Force (TSTF)-479, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a.

The reference to "applicable supports" is also been deleted from the Inservice Testing Program in Technical specification 5.7.2.11. The proposed change implements TSTF-279-A, Revision 0, "Remove "applicable supports" from Inservice Testing Program."

TVA 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 Technical Specification Section 5.7.2.11 for WBN Unit 1 to conform to the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, 2, and 3.

ASME has in the last several years, transitioned the requirements for inservice testing of pumps and valves out of ASME Section XI and into a separate, stand alone code entitled the "Code for Operation and Maintenance of Nuclear Power Plants," [ASME OM Code]. The ASME OM Code has been endorsed by the NRC in 10 CFR 50.55a and is the Code that will be required for inservice testing of pumps and valves during the WBN Second Inservice Interval. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves. The proposed change also deletes the reference to supports from the Inservice Testing Program as supports are already inspected under the Inservice Inspection Program.

The proposed changes do not involve any hardware changes, nor do the changes affect the probability of any event initiators. There will be no change to normal plant operating parameters, accident mitigation capabilities or accident analysis assumptions or inputs. Therefore, the proposed changes do 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 revises the technical specification to delete the reference to "applicable supports" from the Inservice Testing Program and to incorporate the latest Code requirements in 10 CFR 50.55a(f)(4) for Code Class 1, 2, and 3 pumps and valves for WBN's next ten year interval. The testing requirements are similar and reflect the same type testing. Valves are still stroke timed; remote position indicators are still verified to be accurate; seat leakage measurements of critical valves are still performed; relief valves still have their setpoints and seat leakages verified; pumps are still tested for hydraulic performance and mechanical condition; check valves are verified to open and close properly; and supports are still inspected under the appropriate inspection program.

The proposed changes do not involve a modification to the physical configuration of the plant or change methods governing normal plant operation. No test methods are added or deleted. 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 revises the TS for consistency with the Standard Technical Specification and `with the requirements in 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, 2, and 3. This change incorporates revisions to the ASME Code that result in a net improvement in the measures of testing. Incorporation of the ASME OM Code does not alter the limiting values and acceptance criteria used to judge the continued acceptability of components tested by the Inservice Testing Program. Deletion of the reference to supports in the Inservice Testing Program does not alter the support inspection program as the program is currently under the Inservice Inspection Program. Since these limits are not altered, the margin of safety is not altered. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, TVA concludes that the proposed amendment 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.

5.2 Applicable Regulatory Requirements/Criteria

NRC regulation, 10 CFR 50.55a defines the requirements for applying industry codes to each licensed nuclear power facility. The regulation in 10 CFR 50.55a(f)(4)(ii) requires that the Inservice Tests Program to verify operational readiness of pumps and valves whose function is required for safety, conducted during successive 120-month intervals, must comply with the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b), 12 months before the start of the 120-month interval. 10 CFR 50.55a(f)(5)(ii) states that if a revised inservice test program for a facility conflicts with the technical specification for the facility, the licensee shall apply to the Commission for amendment of the technical specification to conform the technical specification to the revised program.

The inappropriate reference to supports in the Inservice Testing Program is being deleted. This change is consistent with TSTF-279-A, Revision 0 which was approved by NRC in a letter dated July 16, 1998 and Revision 2 of NUREG-1431.

TVA is in the process of updating WBN's Inservice Testing Program for the Second 10-Year Interval, scheduled to begin December 27, 2006. TVA has identified updates to WBN's Inservice Testing Program that conflict with the requirements in the technical specifications. Therefore, in accordance with the provisions in 10 CFR 50.55a(f)(5)(ii), TVA is submitting the enclosed license amendment.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0. ENVIRONMENTAL IMPACT CONSIDERATION

A review has determined that the proposed amendment would not 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 not change an inspection or surveillance requirement. The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment 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 amendment.

7.0 REFERENCES

1. 10 CFR 50.55a(f), "Inservice Testing Requirements."

2. American Society Of Mechanical Engineers (ASME) Operation and Maintenance of Nuclear Power Plants, 2001 Edition through 2003 Addenda.

3. Technical Specification Task Force (TSTF)-479, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a."

4. Technical Specification Task Force (TSTF)-279-A, Revision 0, "Remove "applicable supports" from the Inservice Testing Program."

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
PROPOSED LICENSE AMENDMENT REQUEST WBN-TS-06-04

PROPOSED TECHNICAL SPECIFICATIONS CHANGES (MARK-UP)

I. Affected Page Listing

5.0-14

II. Marked Pages

See Attached.

5.7 Procedures, Programs, and Manuals (continued)

5.7.2.10 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulation Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

5.7.2.11 Inservice Testing Program

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

a. Testing frequencies ~~specified in Section XI of the ASME Boiler and Pressure Vessel Code~~ **applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code)** and applicable Addenda as follows:

~~ASME Boiler and Pressure Vessel Code~~ **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 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 Boiler and Pressure Vessel Code~~ **ASME OM Code** shall be construed to supersede the requirements of any TS.

(continued)

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
PROPOSED LICENSE AMENDMENT REQUEST WBN-TS-06-04

PROPOSED TECHNICAL SPECIFICATIONS CHANGES (REVISED)

I. Affected Page Listing

5.0-14

II. Marked Pages

See Attached.

5.7 Procedures, Programs, and Manuals (continued)

5.7.2.10 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulation Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

5.7.2.11 Inservice Testing Program

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 Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

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

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and 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)

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
PROPOSED LICENSE AMENDMENT REQUEST WBN-TS-06-04

PROPOSED TECHNICAL SPECIFICATIONS BASES CHANGES (MARK-UPS)

I. Affected Page Listing

B 3.4-49
B 3.4-50
B 3.4-57
B 3.4-73
B 3.4-85
B 3.4-86
B 3.5-18
B 3.6-40
B 3.6-42
B 3.7-5
B 3.7-6
B 3.7-12
B 3.7-18
B 3.7-19
B 3.7-31
B 3.7-33

II. Marked Pages

See Attached.

BASES

ACTIONS

A.1 (continued)

coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 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. With any RCS cold leg temperatures at or below 350°F, overpressure protection is provided by the COMS System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of~~**Section XI of the ASME Code**~~ **the ASME OM Code** (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 3% for OPERABILITY, however, the valves are reset to ± 1% during the surveillance to allow for drift.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, NB 7000, 1971 Edition through Summer 1973.

(continued)

BASES

REFERENCES (continued)

2. Watts Bar FSAR, Section 15.0, "Safety Analyses."

3. WCAP-7769, Rev. 1, "Topical Report on Overpressure Protection for Westinghouse Pressurized Water Reactors," June 1972.

4. ~~ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."~~ **ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants"**

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed. The basis for the Frequency of 92 days is the ASME OM Code, ~~Section XI~~ (Ref. 3). If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an inoperable PORV that is incapable of being manually cycled, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status.

The Note modifies this SR by stating that it is not required to be met with the block valve closed, in accordance with the Required Action of this LCO.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

REFERENCES

1. Regulatory Guide 1.32, "Criteria for Safety Related Electric Power Systems for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1977.

2. Watts Bar FSAR, Section 15.2, "Condition II - Faults of Moderate Frequency."

3. **ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."** ~~ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."~~

BASES

REFERENCES (continued)

7. Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability, and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light Water Reactors,' pursuant to 10 CFR 50.44(f)."

8. **ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."** ~~Boiler and Pressure Vessel Code, Section XI.~~

9. Letter WAT-D-9448, "Tennessee Valley Authority Watts Bar Nuclear Plant Units 1 & 2 Revised COMS PORV Setpoints", August 27, 1994.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within the frequency allowed by the American Society of Mechanical Engineers (ASME) OM Code, ~~Section XI~~ (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met 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.

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 complementary 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.14.1 (continued) performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Section 50.2, "Definitions--Reactor Coolant Pressure Boundary."
2. Title 10, Code of Federal Regulations, Part 50, Section 50.55a, "Codes and Standards," Subsection (c), "Reactor Coolant Pressure Boundary."
3. Title 10, Code of Federal Regulations, Part 50, Appendix A, Section V, "Reactor Containment," General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."
4. U.S. Nuclear Regulatory Commission (NRC), "Reactor Safety Study--An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Appendix V, WASH-1400 (NUREG-75/014), October 1975.
5. U.S. NRC, "The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes," NUREG-0677, May 1980.
6. Watts Bar FSAR, Section 3.9, "Mechanical Systems and Components" (Table 3.9-17).
7. **ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."** ~~Boiler and Pressure Vessel Code, Section XI.~~
8. Title 10, Code of Federal Regulations, Part 50, Section 50.55a, "Codes and Standards," Subsection (g), "Inservice Inspection Requirements."

SURVEILLANCE REQUIREMENTS

SR 3.5.2.3 (continued)

*For the accessible locations, UT may be substituted to demonstrate the piping is full of water. An accessible ECCS high point is defined as one that:

1) Has a vent connection installed.
2) The high point can be vented with the dose received remaining within ALARA expectations. ALARA for venting ECCS high point vents is considered to not be within ALARA expectations when the planned, intended collective dose for the activity is unjustifiably higher than industry norm, or the licensee's past experience, for this (or similar) work activity.
3) The high point can be vented with industrial safety expectations remaining within the industry norm.

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by ~~Section XI of~~ the ASME OM Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pumps baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, ~~which encompasses Section XI~~ of the ASME Code. ~~Section XI of the~~ ASME OM Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5 and 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances 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 ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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. The extended interval to reach MODE 5 allows additional time and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path exists for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned, are in the correct position.

SR 3.6.6.2

Verifying that each containment 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. Flow and differential head are normal tests of centrifugal pump performance required by ~~Section XI of~~ the ASME OM Code (Ref. 4). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design 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.

(continued)

BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criterion (GDC) 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal System," GDC 40, "Testing of Containment Heat Removal Systems, and GDC 50, "Containment Design Basis."
2. Watts Bar FSAR, Section 6.2, "Containment Systems."
3. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."
4. ASME **OM Code, "Code for Operation and Maintenance of Nuclear Power Plants,"** ~~Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components,"~~ American Society of Mechanical Engineers, New York.

BASES

ACTIONS (continued)

B.1 and B.2

In the case of multiple inoperable MSSVs on one or more steam generators, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. The 4 hour Completion Time for Required Action B.1 is consistent with A.1. An additional 32 hours is allowed in Required Action B.2 to reduce the setpoints. The Completion Time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined using a conservative heat balance calculation as described above (Action A.1) and in the attachment to Reference 6. The values in Specification 3.7.1 include an allowance for instrument and channel uncertainties to the allowable RTP obtained with this algorithm.

Required Action B.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation," provide sufficient protection.

C.1 and C.2

If the Required Actions are not completed within the associated Completion Time, or if one or more steam generators have ≥ 4 inoperable MSSVs, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 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 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME **OM** Code, ~~Section XI~~ (Ref. 4), requires that safety and relief valve tests be

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

performed as follows: ~~in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required:~~

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria;

The **ASME OM Code** ~~ANSI/ASME Standard~~ requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. Additional test frequency requirements apply during the initial five year period. ~~as discussed in Reference 5.~~ The ASME OM Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2 correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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. Watts Bar FSAR, Section 10.3, "Main Steam Supply System."

2. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section III, Article NC-7000, "Overpressure Protection," Class 2 Components.

3. Watts Bar FSAR, Section 15.2, "Condition II - Faults of Moderate Frequency," and Section 15.4, "Condition IV - Limiting Faults."

4. American Society of Mechanical Engineers, (ASME) **OM Code, "Code for Operation and Maintenance of Nuclear Power Plants,"** ~~Boiler and Pressure Vessel Code, Section XI.~~

~~5. ANSI/ASME OM-1-1987, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices."~~

5.~~6.~~ NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

The Frequency is in accordance with the Inservice Testing Program or 18 months. The 18 month Frequency for valve closure time 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, the Frequency is acceptable from a reliability standpoint.

This test is conducted in MODE 3 with the unit at operating temperature and pressure, as discussed in Reference 5 exercising requirements. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

REFERENCES

1. Watts Bar FSAR, Section 10.3, "Main Steam Supply System."
2. Watts Bar FSAR, Section 6.2, "Containment Systems."
3. Watts Bar FSAR, Section 15.4.2.1, "Major Rupture of a Main Steam Line."
4. 10 CFR 100.11.
5. American Society of Mechanical Engineers, **OM Code, "Code for Operation and Maintenance of Nuclear Power Plants,"** . ~~Boiler and Pressure Vessel Code, Section XI.~~

BASES

ACTIONS

E.1 (continued)

conditions, at least one valve in the flow path must be restored to OPERABLE status within 8 hours. The Completion Time of 8 hours is consistent with Condition D.

F.1 and F.2

If the MFIV(s) and MFRV(s) and the associated bypass valve(s) cannot be restored to OPERABLE status, or the MFIV(s) or MFRV(s) closed, or isolated within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 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 3.7.3.1

This SR verifies that the closure time of each MFIV, MFRV, and associated bypass valves is ≤ 6.5 seconds on an actual or simulated actuation signal. The MFIV and MFRV closure times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME OM Code, ~~Section XI~~ (Ref. 2), quarterly stroke requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the Inservice Testing Program or 18 months. The 18 month Frequency for valve closure is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 10.4.7, "Condensate and Feedwater Systems."

2. American Society of Mechanical Engineers, **OM Code, "Code for Operation and Maintenance of Nuclear Power Plants," .** ~~Boiler and Pressure Vessel Code, Section XI.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by ~~Section XI of~~ the ASME **OM** Code (Ref. 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and 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 discussed in the ASME **OM** Code~~, Section XI~~ (Ref. 2) (only required at 3 month intervals) satisfies this requirement. The 31 day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 2.

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

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment. This SR is modified by a Note that states that the SR is not required in MODE 4. MODE 4 does not require automatic activation of the AFW because there is a sufficient time frame for operator action. This is based on the fact that even at 0% power (MODE 3) there is approximately a 10 minute trip delay before actuation of the AFW system to allow for operator action. In MODE 4 the heat removal requirements would be less providing more time for operator action.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.5

This SR verifies that the AFW is properly aligned by verifying the flow through the flow paths from the CST to each steam generator prior to entering MODE 2 after initial fuel loading and prior to subsequent entry into MODE 2 whenever the unit has been in any combination of MODES 5 or 6 for greater than 30 days. Operability of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned.

REFERENCES

1. Watts Bar FSAR, Section 10.4.9, "Auxiliary Feedwater System."

2. American Society of Mechanical Engineers, **OM Code, "Code for Operation and Maintenance of Nuclear Power Plants," .** ~~**Boiler and Pressure Vessel Code, Section XI.**~~