WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J Garrett
Vice President, Engineering

February 1, 2006

ET 06-0001

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

Subject: Docket No. 50-482: Revision to Technical Specification 5.5.8, "Inservice Testing Program" to Adopt Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-479, Revision 0

Gentlemen:

Pursuant to 10 CFR 50.90, Wolf Creek Nuclear Operating Corporation (WCNOC) hereby requests an amendment to Facility Operating License Number NPF-42 for the Wolf Creek Generating Station. The license amendment request (LAR) proposes to revise Technical Specification (TS) 5.5.8, "Inservice Testing Program."

This LAR proposes to revise TS 5.5.8 for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3. The proposed changes are consistent with Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-479, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a." Revision 3.1 of NUREG-1431, "Standard Technical Specifications Westinghouse Plants," incorporated TSTF-479, Revision 0. The NRC issued a letter to the TSTF on December 6, 2005, indicating the acceptability of TSTF-479, Rev. 0.

Attachments I through V provide the Evaluation, Markup of Technical Specification pages, Retyped Technical Specification pages, Proposed TS Bases Changes (for information only), and Summary of Regulatory Commitments. Final TS Bases changes will be implemented pursuant to TS 5.5.14, "Technical Specification Bases Control Program," at the time the amendment is implemented.

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment. The amendment application was reviewed by the WCNOC Plant Safety Review Committee.

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In accordance with 10 CFR 50.91, a copy of this amendment application is being provided to the designated Kansas State official.

WCNOC requests approval of this proposed License Amendment by October 4, 2006. The changes proposed are not required to address an immediate safety concern. It is anticipated that the license amendment, as approved, will be effective upon issuance, to be implemented within 90 days from the date of issuance. If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Kevin Moles at (620) 364-4126.

Very truly yours,

Terry J. Garrett

TJG/rlt

Attachments:
- I - Evaluation
- II - Markup of Technical Specification pages
- III - Retyped Technical Specification pages
- IV - Proposed TS Bases Changes (for information only)
- V - Summary of Regulatory Commitments

cc: T. A. Conley, (KDHE), w/a
J. N. Donohew (NRC), w/a
W. B. Jones (NRC), w/a
B. S. Mallett (NRC), w/a
Senior Resident Inspector (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Terry J. Garrett, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By [signature]
Terry J. Garrett
Vice President Engineering

SUBSCRIBED and sworn to before me this 1 day of Feb., 2006.

OFFICIAL SEAL
RHONDA L. TIEMEYER
MY COMMISSION EXPIRES
January 11, 2010

Rhonda L. Tiemeyer
Notary Public

Expiration Date January 11, 2010

EVALUATION

1.0 DESCRIPTION

The proposed license amendment revises the requirements in Technical Specification (TS) 5.5.8, "Inservice Testing Program," to update references to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI as the source of requirements for the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes delete reference to Section XI of the Code and incorporate reference to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and addresses the applicability of Surveillance Requirement (SR) 3.0.2 to other normal and accelerated Frequencies specified in the Inservice Testing (IST) Program. The proposed changes are consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-479, "Changes to Reflect Revision of 10 CFR 50.55a," Rev. 0.

The proposed changes are consistent with the implementation of the Wolf Creek Generating Station (WCGS) Third 10-Year Interval IST Program in accordance with the requirements of 10 CFR 50.55a(f). The Third 10-Year Interval started on September 4, 2005.

2.0 PROPOSED CHANGE

TS 5.5.8, "Inservice Testing Program," is revised to indicate that the Inservice Testing Program shall include testing Frequencies applicable to the ASME OM Code.

TS 5.5.8b. is revised to indicate that there may be some non-standard Frequencies utilized in the IST Program in which the provisions of SR 3.0.2 are applicable. Specifically, TS 5.5.8b. is revised to state:

> "The provisions of SR 3.0.2 are applicable to the above required Frequencies and other normal and accelerated Frequencies specified in the Inservice Testing Program for performing inservice testing activities;"

Proposed revisions to the TS Bases are also included in this application. The changes to the affected TS Bases pages will be incorporated in accordance with TS 5.5.14, "Technical Specifications (TS) Bases Control Program."

3.0 BACKGROUND

In 1990, the ASME published the initial edition of the ASME OM Code that provides rules for inservice testing of pumps and valves. The ASME OM Code replaced Section XI of the Boiler and Pressure Vessel Code for inservice testing of pumps and valves. The 1995 edition of the ASME OM Code was incorporated by reference into 10 CFR 50.55a(b) on September 22, 1999. Since 10 CFR 50.55a(f)(4)(ii) requires that inservice testing during successive 10-year intervals comply with the requirements of the latest edition and addenda of the Code incorporated into 10 CFR 50.55a(b), TS 5.5.8 must be revised to reference the ASME OM Code.

4.0 TECHNICAL ANALYSIS

Section XI of the ASME Codes has been revised on a continuing basis over the years to provide updated requirements for the inservice inspection and inservice testing of components. Until 1990, the ASME Code requirements addressing the inservice testing of pumps and valves were contained in Section XI, Subsections IWP (pumps) and IWV (valves). In 1990, the ASME published the initial edition of the OM Code that provides the rules for the inservice testing of pumps and valves. Since the establishment of the 1990 Edition of the OM Code, the rules for inservice testing are no longer being updated in Section XI. As identified in NRC SECY-99-017 dated January 13, 1999, the NRC has generally considered the evolution of the ASME Code to result in a net improvement in the measures for inspecting piping and components and testing pumps and valves.

By final rule issued on September 22, 1999 (64 FR 51370) the NRC amended 10 CFR 50.55a(f)(4)(ii) to require licensees to update their IST Program to the latest approved edition of the ASME OM Code incorporated by reference into 10 CFR 50.55a(b). TS 5.5.8 currently reference the ASME Boiler and Pressure Vessel Code, Section XI, as the source of the IST Program requirements for ASME Code 1, 2, and 3 components. The Code of record for the ongoing Third 10-Year IST Program interval is the 1998 Edition including the OMa-1999 and OMb-2000 Addenda of the ASME OM Code. The proposed changes to TS 5.5.8 are necessary for consistency with the IST requirements of 10 CFR 50.55a.

Additionally, TS 5.5.8 is revised to indicate the provisions of SR 3.0.2 are applicable to other IST Frequencies that are not specifically listed in the testing Frequencies that are identified in TS 5.5.8. The IST Program may have Frequencies for testing that are based on risk or other factors and do not conform to the standard testing Frequencies specified in TS 5.5.8. The Frequency of the Surveillance may be determined through a mix of risk informed and performance based means in accordance with the IST Program. Application of SR 3.0.2 to other IST Frequencies is consistent with the guidance in NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," April 1995. Response 6.2-2 in Appendix G of NUREG-1482 indicates that the 25% tolerance applies to increased Frequencies in the same manner that it applies to regular Frequencies. This response would indicate that the 25% tolerance specified in SR 3.0.2 is applicable to any IST Frequency.

5.0 REGULATORY ANALYSIS

5.1 <u>No Significant Hazards Consideration</u>

Wolf Creek Nuclear Operating Corporation (WCNOC) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," Part 50.92(c), 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 5.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f) regarding the inservice testing of pumps and valves. 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 does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. They do not involve the addition or removal of any equipment, or any design changes to the facility. Therefore, the proposed change does not represent 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 TS 5.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f) regarding the inservice testing of pumps and valves. 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 does not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure. Therefore, this proposed change does not create the possibility of an accident of a different kind than previously evaluated.

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

Response: No

The proposed change revises TS 5.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f) regarding the inservice testing of pumps and valves. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves. The safety function of the affected pumps and valves will be maintained. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

5.2 Applicable Regulatory Requirements/Criteria

NRC regulation, 10 CFR 50.55a, defines the requirements for applying industry codes to each licensed nuclear powered facility. The regulations require that during successive 120-month intervals, programs be developed utilizing the latest edition and addenda incorporated into paragraph (b) of 10 CFR 50.55a on the date 12 months prior to the date of issuance of the operating license subject to the limitations and modifications identified in paragraph (b).

There are no changes being proposed such that compliance with any of the regulatory requirements above would come into question. The evaluations documented above confirm that WCGS will continue to comply with all applicable regulatory requirements.

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 CONSIDERATION

WCNOC has evaluated the proposed amendment for environmental considerations. The review has determined that the proposed amendment would change requirements with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, and would change an inspection or surveillance requirement. However, 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 amendments meet 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

7.1 References

1. ASME Operation and Maintenance Code for Operation and Maintenance of Nuclear Power Plants, 1998 Edition including the OMa-1999 and OMb-2000 Addenda.

2. 10 CFR 50.55a, Codes and standards.

3. SECY-99-017, "Proposed Amendment to 10 CFR 50.55a," January 13, 1999.

4. NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," April 1995.

5. Federal Register Notice: Industry Codes and Standards; Amended Requirements, published September 22, 1999 (64 FR 51370).

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

7.2 Precedents

A similar change was approved for the Susquehanna Steam Electric Station, Units 1 and 2, in Amendment No. 228 and Amendment No. 204 on December 7, 2005. However, these amendments did not address the application of SR 3.0.2 to other IST Frequencies.

ATTACHMENT II

MARKUP OF TECHNICAL SPECIFICATION PAGES

5.5 Programs and Manuals

5.5.7 Reactor Coolant Pump Flywheel Inspection Program (continued)

radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code)

5.5.8 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 ~~specified in Section XI of the ASME Boiler and Pressure Vessel Code~~ and applicable Addenda as follows:

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

OM

and other normal and accelerated Frequencies specified in the Inservice Testing Program

ATTACHMENT III

RETYPED TECHNICAL SPECIFICATION PAGES

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

k. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the USAR, Section 3.9(N), cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Containment Tendon Surveillance Program

This program provides controls for monitoring tendon performance, including the effectiveness of the tendon corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial plant operation as well as periodic testing thereafter. The Containment Tendon Surveillance Program, and its inspection frequencies and acceptance criteria, shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an exemption or relief has been authorized by the NRC.

The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

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

In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer

(continued)

5.5 Programs and Manuals

5.5.7 Reactor Coolant Pump Flywheel Inspection Program (continued)

radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

5.5.8 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 Operation 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 in the Inservice Testing Program for performing inservice testing activities;

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

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

(continued)

ATTACHMENT IV

PROPOSED TS BASES CHANGES (FOR INFORMATION ONLY)

BASES

ACTIONS

B.1 and B.2 (continued)

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 368°F, overpressure protection is provided by the LTOP 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.

Addition to the RCS of borated water with a concentration greater than or equal to the minimum required RWST concentration shall not be considered a positive reactivity change. Cooldown of the RCS for restoration of OPERABILITY of a pressurizer code safety valve, with a negative moderator temperature coefficient, shall not be considered a positive reactivity change provided the RCS is borated to the COLD SHUTDOWN, xenon-free condition per Specification 3.1.1 (Ref. 5).

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 (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint tolerance is ± 2% for OPERABILITY, however, the valves shall be set at 2460 psig ± 1% per Ref. 1 to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.

2. USAR, Chapter 15.

3. WCAP-7769, Rev. 1, June 1972.

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

5. NRC letter (W. Reckley to N. Carns) dated November 22, 1993: "Wolf Creek Generating Station - Positive Reactivity Addition; Technical Specification Bases Change."

Code for Operation and Maintenance of Nuclear Power Plants.

BASES

ACTIONS (continued)

G.1 and G.2

If the Required Actions of Condition F are not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 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. In MODES, 3 (with any RCS cold leg temperature ≤ 368°F), 4, 5, and 6 (with the reactor vessel head on) automatic PORV OPERABILITY may be required. See LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed. The basis for the Frequency of 92 days is the ASME Code ~~Section XI~~ (Ref. 4).

The Note modifies this SR by stating that it is not required to be performed with the block valve closed, in accordance with the Required Actions of this LCO. Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable.

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. Operating experience has shown that these valves usually pass the Surveillance when performed at the required Inservice Testing Program frequency. The Frequency is acceptable from a reliability standpoint.

REFERENCES

1. USAR, Figure 7.2-1 (Sheet 11) and 7.6-4 (Sheets 1-3).
2. Regulatory Guide 1.32, February 1977.
3. USAR, Section 15.2.
4. ASME ~~Boiler and Pressure Vessel Code, Section XI.~~ Code for Operation and Maintenance of Nuclear Power Plants.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 (continued)

result in an injection into the RCS. This may be accomplished by placing the pump control switch in pull to lock and closing at least one valve in the discharge flow path, or by closing at least one valve in the discharge flow path and removing power from the valve operator, or by closing at least one manual valve in the discharge flow path under administrative control. Providing pumps are rendered incapable of injecting into the RCS, they may be energized for purposes such as testing or for filling accumulators.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SR 3.4.12.4

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.

The RHR suction isolation valves are verified to be opened every 72 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction isolation valves remain open.

The ASME Code ~~Section XI~~ (Ref. 8), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

SR 3.4.12.5

The RCS vent of ≥ 2.0 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that is not locked, sealed, or otherwise secured in the open position.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.8 (continued)

A Note has been added indicating that this SR is not required to be performed until 12 hours after decreasing RCS cold leg temperature to ≤ 368°F.

SR 3.4.12.9

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 18 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ASME, Boiler and Pressure Vessel Code, Section III.
4. USAR, Chapter 15.
5. 10 CFR 50, Section 50.46.
6. 10 CFR 50, Appendix K.
7. Generic Letter 90-06.
8. ASME ~~Boiler and Pressure Vessel Code, Section XI.~~ Code for Operation and Maintenance of Nuclear Power Plants.
9. USAR, Section 5.2.2.10.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

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 within the frequency allowed by the American Society of Mechanical Engineers (ASME) Code Section XI (Ref. 6), 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.

Test pressures less than 2235 psig but greater than 150 psig are allowed for valves where higher pressures could tend to diminish leakage channel opening. Observed leakage shall be adjusted for actual pressure to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one half power.

In addition, testing must be performed once after the check 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 check valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a check 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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

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 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.

SR 3.4.14.2

The RHR suction isolation valve interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be < 425 psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The 18 month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. This SR is not required to be performed when the RHR suction isolation valves are open to satisfy LCO 3.4.12.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. ASME ~~Boiler and Pressure Vessel Code, Section XI.~~ Code for Operation and Maintenance of Nuclear Power Plants.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.4 (continued)

problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. The following ECCS pumps are required to develop the indicated differential pressure on recirculation flow:

Centrifugal Charging Pump	≥ 2400 psid
Safety Injection Pump	≥ 1445 psid
RHR Pump	≥ 165 psid

This verifies both that the measured performance is within an acceptable tolerance of the original pump 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 applicable portions of the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5 and SR 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 on an actual or simulated RWST Level Low-Low 1 Automatic Transfer signal coincident with an 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 controls. 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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.4 (continued)

required by Section XI of the ASME Code (Ref. 5). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested 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 abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

This test ensures that each pump develops a differential pressure of greater than or equal to 219 psid at a nominal flow of 300 gpm when on recirculation (Ref. 6).

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment High-3 pressure signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. 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 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.

The surveillance of containment sump isolation valves is also required by SR 3.5.2.5. A single surveillance may be used to satisfy both requirements.

SR 3.6.6.7

This SR requires verification that each containment cooling train actuates upon receipt of an actual or simulated safety injection signal. Upon actuation, each fan in the train starts in slow speed or, if operating, shifts to slow speed and the cooling water flow rate increases to ≥ 2000 gpm to each cooler train. The 18 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the 18 month Frequency.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.8

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41. GDC 42, and GDC 43, and GDC 50.
2. 10 CFR 50, Appendix K.
3. USAR, Section 6.2.1.
4. USAR, Section 6.2.2.
5. ASME ~~Boiler and Pressure Vessel Code, Section XI.~~ Code for Operation and Maintenance of Nuclear Power Plants.
6. Performance Improvement Request 2002-0945.

BASES

ACTIONS

B.1 and B.2 (continued)

operating experience in resetting all channels of protective function and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

A sensitivity study (Ref. 7) was performed to analyze the loss of load/turbine trip event initiated from power levels based on Table 3.7.1-1 and assuming both beginning of life and end of life reactivity feedback conditions. The results of all cases studied showed that the secondary system peak pressure was maintained below 110% of the secondary system design pressure limit.

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," provides sufficient protection.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

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 unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit conditions from full power conditions in an orderly manner and without challenging unit 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 Code ~~Section XI~~ (Ref. 5), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 6). According to Reference 6, the following tests are required:

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria.

The 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. The ASME 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. USAR, Section 10.3.2.

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

3. USAR, Section 15.2.

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

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

6. ANSI/ASME OM-1-1987.

7. AN-94-017 Rev. 0, "RETRAN-02 MSSV Analysis for ITIP 2625," M. L. Howard, May 1994.

Code for Operation and Maintenance of Nuclear Power Plants.

BASES

ACTIONS

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

A.1 and A.2

With one MFIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 4 hours. When these valves are closed, they are performing their required safety function.

The 4 hour Completion Time takes into account the redundancy afforded by the dual-redundant actuators on the MFIVs and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 4 hour Completion Time is reasonable, based on operating experience.

Inoperable MFIVs that are closed must be verified on a periodic basis that they are closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed.

B.1 and B.2

If the MFIV(s) cannot be restored to OPERABLE status, or closed, within the associated Completion Time, 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 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV is ≤ 5 seconds on an actual or simulated main feedwater isolation actuation signal from each actuator train. The MFIV closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. This is consistent with Regulatory Guide 1.22 (Ref. ~~4~~)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 (continued)

The Frequency for this SR is in accordance with the Inservice Testing Program. Operating experience has shown that these components usually pass the Surveillance when performed at the Inservice Testing Program Frequency. This test is conducted in MODE 3 with the unit at nominal operating temperature and pressure, as discussed in Reference 2. 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.

SR 3.7.3.2

This SR verifies that each MFIV can close on an actual or simulated actuation signal. The manual close hand switch in the control room provides an acceptable actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. 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

The frequency of MFIV testing 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.

REFERENCES

1. USAR, Section 10.4.7.
2. ASME ~~Boiler and Pressure Vessel Code, Section XI.~~ Code for Operations and Maintenance of Nuclear Power Plants.
~~3. USAR, Table 7.3-14.~~
3. ~~4.~~ Regulatory Guide 1.22, Rev. 0.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to manual vent/drain valves, and 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 being mispositioned are in the correct position.

The 31 day Frequency, based on engineering judgment, is consistent with procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a Note indicating that the SR is not required to be performed for the AFW flow control valves until the AFW System is placed in standby or THERMAL POWER is above 10% RTP.

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 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 Code ~~Section XI~~ (Ref. 2) (only required at 3 month intervals) satisfies this requirement. The test Frequency in accordance with the Inservice Testing Program results in testing each pump once every 3 months, as required by Reference 2.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.4 (continued)

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

SR 3.7.5.5

This SR verifies that the AFW is properly aligned by verifying the flow paths from the CST to each steam generator prior to entering MODE 2 after more than 30 days in MODE 5 or 6. 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 judgement 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. USAR, Section 10.4.9.
2. ASME ~~Boiler and Pressure Vessel Code, Section XI.~~
3. NRC letter (C. Poslusny to O. Maynard) dated December 16, 1998: "Wolf Creek Generating Station - Technical Specification Bases Change, Auxiliary Feedwater System."
4. Performance Improvement Request 2002-0945.

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LIST OF COMMITMENTS

The following table identifies those actions committed to by WCNOC in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Kevin Moles at (620) 364-4126.

COMMITMENT	Due Date/Event
The proposed changes to the WCGS Technical Specifications will be implemented within 90 days of NRC approval.	Within 90 days of NRC approval