AmerenUE
Callaway Plant

PO Box 620
Fulton, MO 65251

December 28, 2007

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop P1-137
Washington, DC 20555-0001

ULNRC-05462

Ladies and Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
FACILITY OPERATING LICENSE NPF-30
REQUEST TO REVISE TECHNICAL SPECIFICATION (TS)
5.5.8, "INSERVICE TESTING PROGRAM"
(LICENSE AMENDMENT REQUEST OL-1284)

Pursuant to 10 CFR 50.90, Union Electric Company (AmerenUE) hereby requests amendment of the facility Operating License (NPF-30) for the Callaway plant.

The proposed amendment would revise Technical Specification (TS) Administrative Controls 5.5.8, "Inservice Testing Program," to indicate that the Inservice Testing Program shall include testing frequencies applicable to the ASME Code for Operation and Maintenance (ASME OM Code), and to indicate that there may be some non-standard Frequencies specified as 2 years or less in the Inservice Testing Program to which the provisions of SR 3.0.2 are applicable. The proposed changes are consistent with NRC-approved Technical Specification Task Force (TSTF) Travelers TSTF-479, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a," and TSTF-497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less." AmerenUE has determined that this License Amendment Request (LAR) does not involve a significant hazard consideration as determined per 10 CFR 50.92.

AmerenUE is submitting this LAR in conjunction with an industry consortium of six plants as a result of a mutual agreement known as Strategic Teaming and Resource Sharing (STARS). The STARS group consists of the six plants operated by Union Electric Company (AmerenUE), TXU Generation Company LP, Pacific Gas and

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Electric Company, Arizona Public Service Company, STP Nuclear Operating Company, and Wolf Creek Nuclear Operating Corporation.

Diablo Canyon is the lead STARS plant for this effort. The LAR for Diablo Canyon was submitted on December 29, 2006. The LAR was subsequently approved, and license amendments were issued for Diablo Canyon Units 1 and 2 on June 25, 2007. This LAR for Callaway is intended to be submitted in conjunction with other STARS plant LARs, with the goal of reducing the amount of NRC resources required to evaluate and approve these requests.

AmerenUE's technical and regulatory evaluation of the proposed TS changes, the TS changes themselves, and the TS Bases changes (for information only) are provided as attachments to this letter. The changes proposed in this LAR are not required to address any immediate safety concern. Further, this letter does not contain any new commitments.

AmerenUE requests approval of this LAR no later than December 31, 2008. Once approved, it is anticipated that the amendment will be implemented within 90 days from the date of issuance. If you have any questions or require additional information, please contact Mr. Scott Maglio at (573) 676-8719.

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

Sincerely,

Luke H. Graessle
Manager, Regulatory Affairs

Executed on: 12-28-07

Attachments:
1. Licensee Evaluation
2. TS Page Markups
3. Retyped TS Pages
4. Changes to TS Bases (for information only)

cc:

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LICENSEE EVALUATION

Subject: Application for license amendment to revise TS 5.5.8, "Inservice Testing Program," to include testing frequencies applicable to the ASME OM Code, and to indicate that there may be some non-standard IST Frequencies to which the provisions of SR 3.0.2 are applicable.

1.0 DESCRIPTION

2.0 PROPOSED CHANGES

3.0 BACKGROUND

4.0 TECHNICAL EVALUATION

5.0 REGULATORY EVALUATION

5.1 No Significant Hazards Consideration

5.2 Applicable Regulatory Requirements/Criteria

6.0 ENVIRONMENTAL CONSIDERATION

7.0 REFERENCES

1.0 DESCRIPTION

This evaluation supports a request to amend Operating License NPF-30 for Callaway Plant Unit 1 to update references to the source of requirements for the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves and to address the applicability of Surveillance Requirement (SR) 3.0.2 to some non-standard pump and valve testing frequencies.

The proposed changes to the Technical Specifications are consistent with NRC-approved Technical Specification Task Force (TSTF) Travelers TSTF-479, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a," (Reference 1) and TSTF-497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less" (Reference 2).

2.0 PROPOSED CHANGES

The proposed change would revise 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) (Reference 3). The changes also address the applicability of SR 3.0.2 to some non-standard frequencies specified as 2 years or less in the Inservice Testing (IST) Program.

The proposed changes are consistent with the implementation of the Callaway Plant Third 10-Year Interval IST Program in accordance with the requirements of 10 CFR 50.55a(f). The Third 10-Year Interval started on December 19, 2005.

The proposed TS changes are as follows:

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

- TS 5.5.8b is revised to indicate that there may be some nonstandard frequencies utilized in the IST Program to 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 to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities."

Various sections of the TS Bases are also revised for consistency with the requirements of 10 CFR 50.55a(f)(4), i.e., to ensure consistent reference to the ASME Code for Operation and Maintenance of Nuclear Power Plants. 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). 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 EVALUATION

The purposes of the Inservice Testing Program are to assess the operational readiness of pumps and valves, to detect degradation that might affect component OPERABILITY, and to maintain safety margins with provisions for increased surveillance and corrective action. NRC regulation, 10 CFR 50.55a, defines the requirements for applying industry codes to each licensed nuclear powered facility. Section XI of the ASME Code 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 (Reference 4), 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 (Reference 5), the NRC amended 10 CFR 50.55a(b) to reference the latest approved edition of the ASME OM Code. 10 CFR 50.55a(f)(4)(ii) requires inservice tests to comply with the requirements of the latest edition and addenda of Code, as referenced in 10 CFR 50.55a(b), 12 months before the start of a new 10-year (120-month) interval. TS 5.5.8 currently references 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 2001 Edition including the OMa-2002 and OMb-2003 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, the proposed changes to TS 5.5.8 indicate that the provisions of SR 3.0.2 are applicable to other IST Frequencies that are not specifically listed in the testing Frequencies 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 specified as 2 years or less is consistent with the guidance in NUREG-1482, paragraph 3.1.3 (Reference 6). This response would indicate that the 25% tolerance specified in SR 3.0.2 is applicable to any IST Frequency specified as 2 years or less.

5.0 REGULATORY EVALUATION

The proposed changes revise 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 address the applicability of Surveillance Requirement (SR) 3.0.2 to other normal and accelerated frequencies specified as 2 years or less in the Inservice Testing (IST) Program.

5.1 No Significant Hazards Consideration

AmerenUE 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 amendment 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)(4) 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, nor does it 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 amendment 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)(4) 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 amendment 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)(4) 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 functions of the affected pumps and valves will be maintained. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

Based on the above, AmerenUE concludes that the proposed amendment does not involve a 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 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 start of each 120-month inspection interval 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 AmerenUE 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

AmerenUE 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 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

7.1 References

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

2. Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less."

3. ASME Code for Operation and Maintenance of Nuclear Power Plants, 2001 Edition including the OMa-2002 and OMb-2003 Addenda.

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

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

6. NUREG-1482, Revision 1, "Guidelines for Inservice Testing at Nuclear Power Plants," January 2005.

7. Letter dated December 6, 2005, from USNRC to Technical Specifications Task Force

8. Letter dated October 4, 2006, from USNRC to Technical Specifications Task Force

9. Wolf Creek Generating Station Amendment No. 172 dated November 15, 2006

7.2 Precedent

NRC accepted TSTF-479 in December, 2005 (Reference 7) and accepted TSTF-497 in October, 2006 (Reference 8).

A similar change was approved for Wolf Creek Generating Station in Amendment No. 172 on November 15, 2006 (Reference 9). However, even though TSTF-497 was not approved at the time of this license amendment application, this amendment addresses the application of SR 3.0.2 to other IST Frequencies with the same wording that is now included in the approved TSTF-497.

As noted in the cover letter, license amendments (Nos. 196 and 197) incorporating changes similar to those proposed for Callaway were issued to Diablo Canyon Units 1 and 2 on June 25, 2007.

ATTACHMENT 2

MARKUP OF TECHNICAL SPECIFICATIONS

Attachment 2 to
ULNRC-05462

5.5 Programs and Manuals (continued)

5.5.8 Inservice Testing Program

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

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 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

OM

and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program

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

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the

(continued)

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ATTACHMENT 3

RETYPED TECHNICAL SPECIFICATIONS

Attachment 3 to
ULNRC-05462

5.5 Programs and Manuals (continued)

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 to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

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

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

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing

(continued)

Attachment 3 to
ULNRC-05462

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of thetubing during a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All inservice steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown, and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 (3DP) against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm total for all four steam generators.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

(continued)

Attachment 3 to
ULNRC-05462

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

(continued)

ATTACHMENT 4

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES
(for information only)

1. Add Bases Insert 1 to the following pages:

   Page B 3.4.10-4,
   Page B 3.4.11-8,
   Page B 3.4.12-13,
   Page B 3.4.14-7,
   Page B 3.6.6-10,
   Page B 3.7.1-5,
   Page B 3.7.3-10, and
   Page B 3.7.5-11.

Bases Insert 1 ASME Code for Operation and Maintenance of Nuclear Power Plant

Attachment 4 to
ULNRC-05462

BASES

ACTIONS (continued)

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 with any RCS cold leg temperature ≤ 275°F 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 275°F, overpressure protection is provided by the COMS. 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 (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ±2% 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.

2. FSAR, Chapter 15.

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

Insert 1

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

Attachment 4 to
ULNRC-05462

BASES

ACTIONS (continued)

G.1 and G.2

If the Required Actions of Condition F are not met, 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 1, 2, 3, and 4 (with any RCS cold leg temperature ≤ 275°F), 5, and 6 (with the reactor vessel head on), automatic PORV OPERABILITY is required. See LCO 3.4.12 for requirements in MODES 4, 5, and 6.

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. FSAR Section 15.5.1.
2. Regulatory Guide 1.32, February 1977.
3. FSAR, Section 15.2.
4. ASME, Boiler and Pressure Vessel Code, Section XI.

Insert 1

Attachment 4 to
ULNRC-05462

BASES

SURVEILLANCE REQUIREMENTS

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

incapable of injecting into the RCS such that a single failure or single action will not 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 controls.

The Frequency of 12 hours is sufficient, considering administrative controls and 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 vent pathway that is not locked, sealed, or otherwise secured in the open position, or

b. Once every 31 days for a valve that is locked, sealed, or otherwise secured in the open position. A removed pressurizer safety valve or open manway also fits this category.

(continued)

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SURVEILLANCE REQUIREMENTS

SR 3.4.12.8 (continued)

The 12 hour allowance considers the unlikelihood of a low temperature overpressure event during this time.

A Note has been added indicating that this SR is not required to be performed until 12 hours after decreasing any RCS cold leg temperature to ≤ 275°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. FSAR, 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.
9. FSAR Section 5.2.2.10.
10. FSAR Section 6.3.2.
11. FSAR Section 7.6.4.
12. Amendment No. 124 to Facility Operating License NPF-30 dated April 2, 1998.

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ACTIONS (continued)

B.1 and B.2

If leakage cannot be reduced or the system isolated, 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 MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. 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.

C.1

The inoperability of the RHR suction isolation valve interlock could allow inadvertent opening of the valves at RCS pressures in excess of the RHR system design pressure. If the RHR suction isolation valve interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one deactivated remote manual valve within 4 hours. This Action accomplishes the purpose of the interlock.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV used to satisfy Required Action A.1 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

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. 7) as contained in the Inservice Testing Program, 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

(continued)

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SURVEILLANCE REQUIREMENTS

SR 3.4.14.2 (continued)

are used to satisfy LCO 3.4.12. 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.

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. (Insert 1)
7. 10 CFR 50.55a(g).

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SURVEILLANCE REQUIREMENTS

entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water by venting RHR and SI pump casings and accessible ECCS discharge piping high point vents ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. In conjunction with or in lieu of venting, Ultrasonic Testing (UT) may be performed to verify ECCS pumps and associated piping are full of water. The design of the centrifugal charging pump is such that significant noncondensible gases do not collect in the pump. Therefore, it is unnecessary to require periodic pump casing venting to ensure the centrifugal charging pumps will remain OPERABLE. Accessible high point vents are those that can be reached without hazard or high radiation dose to personnel. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.

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 Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. The ECCS pumps are required to develop the following differential pressures on recirculation flow: 1) centrifugal charging pumps ≥ 2400 psid; 2) safety injection pumps ≥ 1445 psid; and 3) RHR pumps ≥ 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 or 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

SR 3.5.2.5 and SR 3.5.2.6 (continued)

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SURVEILLANCE REQUIREMENTS

SR 3.6.6.1 (continued)

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

SR 3.6.6.2

Operating each containment cooling train fan unit for ≥ 15 minutes ensures that all fan units are OPERABLE. It also ensures that abnormal conditions or degradation of the fan unit can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the fan units and controls, the two train redundancy available, and the low probability of significant degradation of the containment cooling train occurring between surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.6.3

Verifying that each containment cooling train ESW cooling flow rate is ≥ 2200 gpm provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 3). The Frequency was developed considering the known reliability of the Essential Service Water System, the two train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6.4

Verifying 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 pressure are normal tests of centrifugal pump performance 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 ensures that each pump develops a discharge pressure of greater than or equal to 250 psig. 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.

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REFERENCES (continued)

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, GDC 43 and GDC 50.
2. 10 CFR 50, Appendix K.
3. FSAR, Section 6.2.1.
4. FSAR, Section 6.2.2.
5. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

Insert 1

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BASES (Continued)

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 (Ref. 5).

(Ref. 5)

The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a +3%/-1% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings pressure shall 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, if necessary, to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR, Section 10.3.2, Main Steam Supply System - System Description.

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

3. FSAR, Section 15.2, Decrease in Heat Removal by the Secondary System.

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

Insert 1

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

6. Westinghouse Letter SCP-99-129, dated July 7, 1999.

7. WCAP-16265-P, "Callaway Replacement Steam Generator Program NSSS Licensing Report," September 2004.

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BASES (Continued)

REFERENCES

1. FSAR, Section 10.3, Main Steam Supply System.

2. FSAR, Section 6.2, Containment Systems.

3. FSAR, Section 15.1.5, Steam System Piping Failure.

4. 10 CFR 100.11.

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

5. ~~6.~~ FSAR 6.2.1.4.3.3, Containment Pressure - Temperature Results.

6. ~~11.~~ Amendment 172 to Facility Operating License No. NPF-30, (NRC Safety Evaluation included), Callaway Unit 1, dated June 16, 2006.

7. ~~8.~~ Westinghouse Letter, SCP-07-26, dated March 6, 2007.

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REFERENCES (continued)

2. Westinghouse Letter, SCP-05-027, Revision 2, dated September 9,2005.

3. FSAR, Table 7.3-14, NSSS Instrument Operating Conditions for Isolation Functions.

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

5. FSAR Figure 7.2-1, Sheets 13 and 14.

6. FSAR Figure 10.4-6, Sheets 1 and 2.

7. WCAP-16265-P, dated August 2004.

Insert 1

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

9. FSAR Section 9.3.1, Compressed Air System

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SURVEILLANCE REQUIREMENT

SR 3.7.5.1 (continued)

In order for the TDAFP and MDAFPs to be OPERABLE while the AFW system is in automatic control or above 10% RTP, the discharge flow control valves (ALHV0005, 6, 7, 8, 9, 10, 11, and 12) shall be in the full open position. The TDAFP and MDAFPs remain OPERABLE with the discharge flow control valves throttled to maintain steam generator levels during plant heatup, cooldown, or if started due to an Auxiliary Feedwater Actuation Signal (AFAS) or manually started in anticipation of an AFAS.

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

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. 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.

The required differential pressure for the AFW pumps when tested in accordance with the Inservice Testing Program is:

a. The acceptance criteria for the MDAFPs have been calculated using a limiting performance curve. The acceptance criteria, given as a table below, have been determined based on the Loss of Normal Feedwater (LONF) or Loss of Non-emergency AC Power (LOAC) events.

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SR 3.7.5.3 (continued)

and that valves ALHV0030, 31, 32, 33, 34, 35 and 36 actuate to the required position upon receipt of an Auxiliary Feedwater Pump suction Pressure-Low signal.

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an AFAS by demonstrating that each AFW pump starts automatically on an actual or simulated auxiliary feedwater actuation signal. 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.

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 system 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. FSAR, Section 10.4.9, Auxiliary Feedwater System.
2. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~ Insert 1
3. FSAR, Section 9.3.1, Compressed Air System.

(continued)