10 CFR 50.90

A subsidiary of Pinnacle West Capital Corporation

Palo Verde Nuclear Generating Station

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102-06178-DCM/RJR
April 29, 2010

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3 Docket Nos. STN 50-528, 50-529, and 50-530 Request for Amendment to Technical Specification 5.5.8, Inservice Testing Program

Pursuant to 10 CFR 50.90, Arizona Public Service Company (APS) hereby requests to amend PVNGS Operating License Nos. NPF-41, NPF-51, and NPF-74 for PVNGS Units 1, 2, and 3, respectively. The proposed amendment implements Technical Specification Task Force (TSTF) Change Travelers TSTF-479-A, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a," as modified by TSTF-497-A, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less."

These proposed changes will:

- Revise the Technical Specifications (TS) to align with the requirements of 10 CFR 50.55a," Codes and Standards," paragraph (f), "Inservice Testing (IST) Requirements,"
- Add a provision in the applicable TS section to only apply the extension allowance of Surveillance Requirement (SR) 3.0.2 to the frequency table listed in the TS as part of the IST Program and to normal and accelerated IST frequencies of two years or less, as applicable, and
- Remove a reference to component supports for consistency with the Standard Technical Specifications. Component supports are part of the Inservice Inspection Program at PVNGS.

These proposed changes will align the PVNGS specification wording to that of NUREG-1432, Revision 3.1, Standard Technical Specifications Combustion Engineering Plants and APS requests approval of the proposed amendment within one calendar year of the submittal date. Once approved, the amendment shall be implemented within 90 days.

A047
NRR

In accordance with the PVNGS Quality Assurance Program, the Plant Review Board and the Offsite Safety Review Committee have reviewed and concurred with this proposed amendment. By copy of this letter, this submittal is being forwarded to the Arizona Radiation Regulatory Agency (ARRA) pursuant to 10 CFR 50.91(b)(1).

No commitments are being made to the NRC by this Letter. Should you need further information regarding this amendment request, please contact Russell A. Stroud, Licensing Section Leader at (623) 393-5111.

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

Executed on 4/29/10
(Date)

Sincerely,

D.C. Mims

DCM/RAS/RJR/gat

Enclosure: Evaluation of the Proposed Change

cc:	E. E. Collins Jr.	NRC Region IV Regional Administrator
	J. R. Hall	NRC NRR Project Manager
	L. K. Gibson	NRC NRR Project Manager
	R. I. Treadway	NRC Senior Resident Inspector for PVNGS
	A. V. Godwin	Arizona Radiation Regulatory Agency (ARRA)
	T. Morales	Arizona Radiation Regulatory Agency (ARRA)

ENCLOSURE

Evaluation of the Proposed Change

Subject: These proposed changes will revise the Technical Specifications (TS) to align with the requirements of 10 CFR 50.55a," Codes and Standards," paragraph (f), "Inservice Testing (IST) Requirements," and will align the PVNGS specification wording to that of NUREG-1432, Revision 3.1, Standard Technical Specifications Combustion Engineering Plants.

ATTACHMENTS:

1. Technical Specification Page Markups
2. Retyped Technical Specification Page
3. Technical Specification Bases Pages Markups (For Information Only)

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating Licenses NPF-41, NPF-51, and NPF-74, for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, respectively, and would revise the requirements in Technical Specification (TS) 5.5.8, "Inservice Testing Program."

The proposed changes would revise the Operating Licenses to update references to the source of requirements for the Inservice Testing (IST) of the American Society of Mechanical Engineers (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 TS are consistent with NRC-approved Technical Specification Task Force (TSTF) Travelers TSTF-479-A, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a," (Reference 6.1) as modified by TSTF-497-A, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less" (Reference 6.2). There is one editorial change (not part of either TSTF) to remove the reference to supports for consistency with the Standard Technical Specifications. Supports are included in the Inservice Inspection (ISI) Program.

These proposed changes will align the PVNGS specification wording to that of NUREG-1432, Revision 3.1, Standard Technical Specifications Combustion Engineering Plants.

The third IST interval dates for Units 1, 2, and 3 are January 15, 2008 – January 14, 2018.

2.0 DETAILED DESCRIPTION

In 1990, the ASME published the initial edition of the ASME Code for Operation and Maintenance of Nuclear Power Plants (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 IST 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 IST, during successive 10-year intervals, comply with the requirements of the latest edition and addenda of the ASME Code (Reference 6.3) incorporated into 10 CFR 50.55a(b), TS 5.5.8 must be revised to reference the ASME OM Code.

The proposed TS changes are as follows:

TS 5.5.8, "Inservice Testing Program," is revised to:

1. Change the reference from ASME Boiler and Pressure Vessel Code to ASME OM Code.

2. Indicate that the IST Program shall have testing Frequencies applicable to the ASME OM Code and that there may be some non-standard 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."

3. The editorial change is to remove the statement "including applicable supports" from the first sentence of TS 5.5.8. The inspection of supports is performed as part of the ISI Program.

Various sections of the TS Bases are also being revised for consistency with the requirements of 10 CFR 50.55a(f)(4), i.e., to ensure consistent reference to the ASME OM Code. *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 TECHNICAL EVALUATION

The purpose of the IST Program is 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. Section XI of the ASME Code has been revised on a continuing basis over the years to provide updated requirements for the ISI and IST of components. Until 1990, the ASME Code requirements addressing the IST 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 provided the rules for the IST of pumps and valves. Since the establishment of the 1990 Edition of the OM Code, the rules for IST are no longer being updated in Section XI. As identified in NRC SECY-99-017 (Reference 6.4), 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. 10 CFR 50.55a, defines the requirements for applying industry codes to each licensed nuclear powered facility.

By final rule issued on September 22, 1999 (Reference 6.5), 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 IST 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 will 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 two years or less is consistent with the guidance in NUREG-1482, paragraph 3.1.3 (Reference 6.6).

The editorial change (not part of either TSTF) is to remove the statement "including applicable supports" from the first sentence of TS 5.5.8. The inspection of supports is performed as part of the ISI Program.

These proposed changes will align the PVNGS specification wording to that of NUREG-1432, Revision 3.1, Standard Technical Specifications Combustion Engineering Plants.

4.0 REGULATORY EVALUATION

4.1 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 APS will continue to comply with all applicable regulatory requirements. The proposed editorial change is to remove the statement "including applicable supports" from the first sentence of TS 5.5.8 because the inspection of supports is performed as part of the ISI Program.

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. These proposed changes will align the PVNGS specification wording to that of NUREG-1432, Revision 3.1, Standard Technical Specifications Combustion Engineering Plants.

4.2 Precedent

The submittals associated with these precedents listed below are the same as the APS submittal with the exception that they did not contain the proposed editorial change that APS is requesting. This is because the TS 5.5.8 version contained in these submittals did not contain the statement "including applicable supports" as part of the IST Program.

- Wolf Creek Generating Station: approved November 15, 2006, Agencywide Documents Access and Management System (ADAMS) Accession No. ML062980233

- Diablo Canyon Power Plant: approved June 25, 2007, ADAMS Accession No. ML070990057

- Callaway Plant: approved November 24, 2008, ADAMS Accession No. ML082460241

4.3 No Significant Hazards Consideration Determination

The proposed change 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, incorporate reference to the ASME Code for Operation and Maintenance of Nuclear Power Plants (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.

The proposed editorial change is to remove the statement "including applicable supports" from the first sentence of TS 5.5.8 because the inspection of supports is performed as part of the Inservice Inspection (ISI) Program.

APS 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 changes revise TS 5.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the IST of pumps and valves and eliminates a statement regarding the testing of supports. The proposed changes incorporate revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves and the editorial change eliminates confusion as to the testing program for supports and will align the PVNGS specification wording to that of NUREG-1432, Revision 3.1, Standard Technical Specifications Combustion Engineering Plants. The proposed changes do 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 changes revise TS 5.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the IST of pumps and valves and eliminates a statement regarding the testing of supports. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves and the editorial change eliminates confusion as to the testing program for supports and aligns wording to that of the standard specification.

The proposed changes do 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 changes 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 changes revise 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 and eliminates a statement regarding the testing of supports. The proposed changes incorporate revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves and the editorial change eliminates confusion as to the testing program for supports and aligns wording to that of the standard specification. 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, APS 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.

4.4 Conclusions

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.

5.0 ENVIRONMENTAL CONSIDERATION

APS has evaluated the proposed amendment for environmental considerations. The review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed 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.

6.0 REFERENCES

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

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

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

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

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

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

ATTACHMENT 1

Technical Specification Page Markups

Page 5.5-5

5.5 Programs and Manuals

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of regulatory position c.4.b of Regulatory Guide 1.14, Revision 0, October 1971.

5.5.8 Inservice Testing Program

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

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

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

(continued)

ATTACHMENT 2

Retyped Technical Specification Page

Page 5.5-5

5.5 Programs and Manuals

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of regulatory position c.4.b of Regulatory Guide 1.14, Revision 0, October 1971.

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

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

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

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

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

(continued)

ATTACHMENT 3

Technical Specification Bases Pages Markups
(For Information Only)

Page B3.4.7-4
B3.4.8-2
B3.4.10-4
B3.4.11-5
B3.4.11-6
B3.4.13-10
B3.4.13-11
B3.4.15-5
B3.4.15-7
B3.5.3-9
B3.5.3-10
B3.6.6-7
B3.6.6-9
B3.7.1-5
B3.7.1-6
B3.7.2-9
B3.7.5-9
B3.7.5-11
B3.8.3-5
B3.8.3-10
B3.9.4-2
B3.9.5-2

BASES

LCO (continued)

Note 5 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of SDC trains from operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the SDC trains.

An OPERABLE SDC train is composed of an OPERABLE SDC pump (CS or LPSI) capable of providing flow to the SDC heat exchanger for heat removal.

SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow ~~(current Section XI)~~, if required. A SG can perform as a heat sink when it is OPERABLE and has the minimum water level specified in SR 3.4.7.2.

The RCS loops may not be considered filled until two conditions needed for operation of the steam generators are met. First, the RCS must be intact. This means that all removable portions of the primary pressure boundary (e.g., manways, safety valves) are securely fastened. Nozzle dams are removed. All manual drain and vent valves are closed, and any open system penetrations (e.g., letdown, reactor head vents) are capable of remote closure from the control room. An intact primary allows the system to be pressurized as needed to achieve the subcooling margin necessary to establish natural circulation cooling. When the RCS is not intact as described, a loss of SDC flow results in blowdown of coolant through boundary openings that also could prevent adequate natural circulation between the core and steam generators. Secondly, the concentration of dissolved or otherwise entrained gases in the coolant must be limited or other controls established so that gases coming out of solution in the SG U-tubes will not adversely affect natural circulation. With these conditions met, the SGs are a functional method of RCS heat removal upon loss of the operating SDC train. The ability to feed and steam SGs at all times is not required when RCS temperature is less than 210°F because significant loss of SG inventory through boiling will not occur during time anticipated to take corrective action. The required SG level provides sufficient time to either restore the SDC train or implement a method for feeding and steaming the SGs (using non-class components if necessary).

(continued)

BASES

LCO (continued)

Note 1 permits all SDC pumps to be de-energized ≤ 1 hour per 8 hour period. The circumstances for stopping both SDC pumps are to be limited to situations when the outage time is short and the core outlet temperature is maintained > 10°F below saturation temperature. The 10 degrees F is considered the actual value of the necessary difference between RCS core outlet temperature and the saturation temperature associated with RCS pressure to be maintained during the time the pumps would be de-energized. The instrument error associated with determining this difference is less than 10 degrees F. (There are no special restrictions for instrumentation use.) Therefore, the indicated value of the difference between RCS core outlet temperature and the saturation temperature associated with RCS pressure must be greater than or equal to 20 degrees F in order to use the provisions of the Note allowing the pumps to be de-energized. The Note prohibits boron dilution or draining operations when SDC forced flow is stopped.

Note 2 allows one SDC train to be inoperable for a period of 2 hours provided that the other train is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable train during the only time when these tests are safe and possible.

An OPERABLE SDC train is composed of an OPERABLE SDC pump (CS or LPSI) capable of providing flow to the SDC heat exchanger for heat removal. SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow ~~(current Section XI)~~, if required.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the SDC System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops-MODES 1 and 2";
LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level" (MODE 6).

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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 four 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 **OM** Code (Ref. ~~1~~**3**), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is +3%, - 1% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift (Ref. 2). The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III~~, Section XI~~.
2. PVNGS Operating License Amendment Nos. 75, 61, and 47 for Units 1, 2, and 3, respectively, and associated NRC Safety Evaluation dated May 16, 1994.
3. **ASME Code for Operation and Maintenance of Nuclear Power Plants.**

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

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 **OM** Code (Ref. ~~1~~2), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is +3%, -1% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift (Ref. 3). The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

SR 3.4.11.2

SR 3.4.11.2 requires that the required Shutdown Cooling System suction line relief valve is OPERABLE by verifying its open pathway condition either:

a. Once every 12 hours for a valve that is unlocked, not sealed, or otherwise not secured open in the vent pathway, or

b. Once every 31 days for a valve that is locked, sealed or otherwise secured open in the vent pathway.

The SR has been modified by a Note that requires performance only if a Shutdown Cooling System suction line relief valve is being used for overpressure protection. The Frequencies consider operating experience with mispositioning of unlocked and locked pathway vent valves.

SR 3.4.11.3

SRs are specified in the Inservice Testing Program. Shutdown Cooling System suction line relief valves are to be tested in accordance with the requirements of ~~Section XI of~~ the ASME **OM** Code (Ref. 2), which provides the activities and the Frequency necessary to satisfy the SRs. The Shutdown Cooling System suction line relief valve setpoint is 467 psig.

(continued)

BASES (continued)

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III~~,~~ ~~Section XI~~.
2. ~~ASME, Boiler and Pressure Vessel Code, Section XI~~**ASME Code for Operations and Maintenance of Nuclear Power Plants**.
3. PVNGS Operating License Amendment Nos. 75, 61, and 47 for Units 1, 2, and 3 respectively, and associated NRC Safety Evaluation dated May 16, 1994.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 and 3.4.13.2 (continued)

b. Once every 31 days for a vent pathway that is locked, sealed, or otherwise secured open.

For an RCS vent to meet the specified flow capacity, it requires removing all pressurizer safety valves, or similarly establishing a vent by opening the pressurizer manway (Ref. 11). The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open. The passive vent arrangement must only be open (vent pathway exists) to be OPERABLE. These Surveillances need only be performed if the vent or the Shutdown Cooling System suction line relief valves are being used to satisfy the requirements of this LCO. The Frequencies consider operating experience with mispositioning of unlocked and locked pathway vent valves, and passive pathway obstructions.

SR 3.4.13.3

SRs are specified in the Inservice Testing Program. Shutdown Cooling System suction line relief valves are to be tested in accordance with the requirements of ~~Section XI of~~ the ASME **OM** Code (Ref. 10), which provides the activities and the Frequency necessary to satisfy the SRs. The Shutdown Cooling System suction line relief valve set point is 467 psig.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. UFSAR, Section 15.
4. 10 CFR 50.46.
5. 10 CFR 50, Appendix K.
6. Generic Letter 90-06.
7. UFSAR, Section 5.2.

(continued)

REFERENCES (continued)

8. Pressure Transient Analyses

   a. V-PSAC-009 (3876 MWt w/Original Steam Generators)

   b. MN725-00118 (Unit 2, 4070 MWt w/Replacement Steam Generators)

   c. MN725-00562 (Units 31, 4070 MWt w/Replacement Steam Generators)

9. Mass Input Pressure Transient in Water Solid RCS

   a. V-PSAC-010 (3876 MWt w/Original Steam Generators)

   b. MN725-00117 (Unit 2, 4070 MWt w/Replacement Steam Generators)

   c. MN725-01495 (Units 31,4070 MWt w/Replacement Steam Generators)

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

11. 13-COO-93-016, Sensitivity Study on Pressurizer Vent Paths vs. Days Post Shutdown.

12. PVNGS Calculation 13-N001-6.02-652-2.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1 (continued)

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

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

The SDC PIVs excepted in two of the three FREQUENCIES are UV-651, UV-652, UV-653, and UV-654, due to position indication of the valves in the control room.

Although not explicitly required by SR 3.4.15.1, performance of leakage testing to verify leakage is below the specified limit must be performed prior to returning a valve to service following maintenance, repair or replacement work on the valve in order to demonstrate operability.

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.

(continued)

BASES (continued)

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. UFSAR, Section 3.9.6.2
7. ~~ASME, Boiler and Pressure Vessel Code, Section XI~~ASME Code for Operation and Maintenance of Nuclear Power Plants.
8. 10 CFR 50.55a(g).
9. T.S. LCO 3.4.13 (LTOP)
10. UFSAR Section 7.6.2.2.1, (4.10).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.3.3

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by ~~Section XI of~~ the ASME **OM** Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the unit safety analysis. SRs are specified in the Inservice Testing Program, which encompasses ~~Section XI of~~ the ASME **OM** Code **(Ref. 7)**. The frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.5.3.4, SR 3.5.3.5, and SR 3.5.3.6

These SRs demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SIAS and on an RAS, that each ECCS pump starts on receipt of an actual or simulated SIAS, and that the LPSI pumps stop on receipt of an actual or simulated RAS. 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 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 the Engineered Safety Feature Actuation System (ESFAS) testing, and equipment performance is monitored as part of the Inservice Testing Program.

The following valve actuations must be verified at least once per 18 months:

> on an actual or simulated recirculation actuation signal, the containment sump isolation valves open, and the HPSI, LPSI and CS minimum bypass recirculation flow line isolation valves and combined SI mini flow valve close.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.3.7

Realignment of valves in the flow path on an SIAS is necessary for proper ECCS performance. The safety injection valves have stops to position them properly so that flow is restricted to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. The 18 month Frequency is based on current industry practice. These valves are also monitored in accordance with the requirements of 10 CFR 50.65 (Ref. 5).

SR 3.5.3.8

Periodic inspection of the containment sump ensures that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during an outage, on the need to have access to the location, and on the potential for unplanned transients if the Surveillance were performed with the reactor at power. This Frequency is sufficient to detect abnormal degradation and is confirmed by operating experience.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. UFSAR, Chapter 6.
4. NRC Memorandum to V. Stello, Jr., from R. L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
5. 10 CFR 50.65.
6. Combustion Engineering Owners Group Joint Applications Report for Low Pressure Safety Injection System AOT Extension, CE NPSD-995, dated May 1995, as submitted to NRC in APS letter no. 102-03392, dated June 13, 1995, with updates described in letter no. 102-04250 dated February 26, 1999. Also see TS amendment no. 124 dated February 1, 2000.
7. **ASME Code for Operation and Maintenance of Nuclear Power Plants.**

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.2

Verifying that the containment spray header piping is full of water to the 113 ft level minimizes the time required to fill the header. This ensures that spray flow will be admitted to the containment atmosphere within the time frame assumed in the containment analysis. The analyses shows that the header may be filled with unborated water which helps to reduce boron plate out due to evaporation. The 31 day Frequency is based on the static nature of the fill header and the low probability of a significant degradation of water level in the piping occurring between surveillances. The value of 113 ft is an indicated value which accounts for instrument uncertainty.

SR 3.6.6.3

Verifying that each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by ~~Section XI of~~ the ASME **OM** Code (Ref. 6). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow (either full flow or miniflow as conditions permit). This test is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

(continued)

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. UFSAR, Section 6.2.
3. UFSAR, Section 6.5.
4. UFSAR, Section 7.3.
5. UFSAR, Section 3.1.34
6. ~~ASME, Boiler and Pressure Vessel Code, Section XI~~ASME Code for Operation and Maintenance of Nuclear Power Plants.
7. 10 CFR 50.44.
8. Regulatory Guide 1.7, Revision 0.

BASES

ACTIONS (continued)

D.1

When more than eight required MSSVs per steam generator are inoperable, 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 setpoints in accordance with the Inservice Testing Program. The ASME **OM** Code, Section XI (Ref. 4), requires ~~that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5,~~ the following tests ~~are required~~ for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

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

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

The ASME **OM Code**~~Standard~~ requires that all valves be tested every 5 years, and a minimum of 20% of the valves tested every 24 months. The ASME **OM** Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.1.1 (continued)

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This is to allow testing of the MSSVs at hot conditions. 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. UFSAR, Section 5.2.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. UFSAR, Section 15.2.
4. ~~ASME, Boiler and Pressure Vessel Code, Section XI, Subsection IWV~~**ASME Code for Operation and Maintenance of Nuclear Power Plants**.
~~5. ANSI/ASME OM-1-1987.~~

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.1 (continued)

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

REFERENCES

1. UFSAR, Section 10.3.
2. CESSAR, Section 6.2.
3. UFSAR, Section 15.1.5.
4. 10 CFR 100.11.
5. ~~ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400~~**ASME Code for Operation and Maintenance of Nuclear Power Plants**.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.2 (continued)

normal tests of pump performance required by ~~Section XI of~~ the ASME **OM** Code (Ref. 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing may be performed on recirculation flow. This test confirms one point on the pump design curve and can be indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing, discussed in the ASME **OM** Code, ~~Section XI~~ (Ref. 2), at 3 month intervals satisfies this requirement.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. Normal operating pressure is established in the steam generators when RCS temperature reaches 532°F, this corresponds to a P_{sat} of 900 psia. This deferral is required because there is an insufficient steam pressure to perform the test.

SR 3.7.5.3

This SR ensures that AFW can be delivered to the appropriate steam generator, in the event of any accident or transient that generates an AFAS signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This SR is not required for the non-essential train since there are no automatic valves which receive an AFAS. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions have been established. Normal operating pressure is established in the steam generators when RCS temperature reaches 532°F, this corresponds to a P_{sat} of 900 psia. This deferral is required because there is an insufficient steam pressure to perform the test.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.5 (continued)

To further ensure AFW System alignment, the OPERABILITY of the essential AFW flow paths is verified following extended outages to determine that no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned by requiring a verification of minimum flow capacity of 650 gpm at pressures corresponding to 1270 psia at the entrance to the steam generators. (This SR is not required for the non-essential AFW pump since it is normally used for startup and shutdown.)

REFERENCES

1. UFSAR, Section 10.4.9.
2. ~~ASME, Boiler and Pressure Vessel Code, Section XI, Subsection IWP~~**ASME Code for Operation and Maintenance of Nuclear Power Plants**.

ACTIONS (continued)

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.3 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

E.1

Each DG is OPERABLE with one air receiver capable of delivering an operating pressure of ≥ 230 psig indicated. Although there are two independent and redundant starting air receivers per DG, only one starting air receiver is required for DG OPERABILITY. Each receiver is sized to accomplish 5 DG starts from its normal operating pressure of 250 psig, and each will start the DG in ≤ 10 seconds with a minimum pressure of 185 psig indicated. If the required starting air receiver is < 230 psig and ≥ 185 psig indicated, the starting air system is degraded and a period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This 48-hour period is acceptable based on the minimum starting air capacity (≥ 185 psig indicated), the fact that the DG start must be accomplished on the first attempt (there are no sequential starts in emergency mode), and the low probability of an event during this brief period. Calculation 13-JC-DG-203 (Ref. ~~9~~8) supports the proposed values for receiver pressures.

F.1

With a Required Action and associated Completion Time not met, or one or more DGs with diesel fuel oil, lube oil, or starting air subsystem inoperable for reasons other than addressed by Conditions A through E, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

(continued)

BASES

REFERENCES

1. FSAR, Section 9.5.4.2.
2. Regulatory Guide 1.137.
3. ANSI N195-1976, Appendix B.
4. FSAR, Chapter 6.
5. FSAR, Chapter 15.
6. ASTM Standards: D4057-81; D975-07b; D976-91; D4737-90; D1796-83; D2276-89, Method A.
7. ASTM Standards, D975, Table 1.
8. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

~~9~~8. "Emergency Diesel Generator and Diesel Fuel Oil Systems Instrumentation Uncertainty Calculation", 13-JC-DG-203, Parts 23 and 51

BASES

LCO

Only one SDC loop is required for decay heat removal in MODE 6, with water level ≥ 23 ft above the top of the reactor vessel flange. Only one SDC loop is required because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one SDC loop must be in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE SDC train is composed of an OPERABLE SDC pump (LPSI or CS) capable of providing flow to the SDC heat exchanger for heat removal. SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow ~~(current Section XI)~~, if required.

The LCO is modified by a Note that allows the required operating SDC loop to be removed from service for up to 1 hour in each 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, surveillance testing of ECCS pumps, and RCS to SDC isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

One SDC loop must be in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Water Level - Fuel Assemblies."

(continued)

BASES

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both SDC loops must be OPERABLE. Additionally, one loop of the SDC System must be in operation in order to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE SDC train is composed of an OPERABLE SDC pump (LPSI or CS) capable of providing flow to the SDC heat exchanger for heat removal. SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow ~~(current Section XI)~~, if required.

Both SDC pumps may be aligned to the Refueling Water Tank (RWT) to support filling the refueling cavity or for performance of required testing.

The LCO is modified by a Note that allows a required operating SDC loop to be removed from service for up to 1 hour in each 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, surveillance testing of ECCS pumps, and RCS to SDC isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

This LCO is modified by a Note that allows one SDC loop to be inoperable for a period of 2 hours provided the other loop is OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing plant configuration. This consideration should include that the core time to boil is not short, there is no draining operation to further reduce RCS water level and that the capacity exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the non-operating loop during a time when these tests are safe and possible.

(continued)