Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10CFR50.90

December 17, 2008
3F1208-06

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

Subject: Crystal River Unit 3 – License Amendment Request #300, Revision 0
Addition of TSTF 479, Revision 0 and TSTF 497, Revision 0 to the Inservice Testing (IST) Program

Dear Sir:

Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc, hereby submits License Amendment Request (LAR) #300, Revision 0. The proposed LAR will revise the Crystal River Unit 3 (CR-3) Improved Technical Specifications (ITS) Administrative Controls, Section 5.6, to revise the Inservice Testing (IST) Program to incorporate the Technical Specification Task Force (TSTF) Traveler 479, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a," and TSTF Traveler 497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less." The current CR-3 ITS Section 5.6.2.9, Inservice Testing Program, requires conformance to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI. The ASME Code requirements have changed, and per 10 CFR 50.55a(f)(4), CR-3 must meet the inservice test requirements set forth in the ASME Operation and Maintenance (OM) Code.

CR-3 is transitioning to the Fourth 10-year Interval of the IST Program and is moving to the 2001 Edition through the 2003 Addenda of the ASME Boiler and Pressure Vessel Code. This edition of the Code was still referenced in Paragraph 50.55a(b)(3) on May 12, 2007, twelve months prior to the start of the Crystal River Unit 3 Fourth 10-year interval. The incorporation of these travelers will ensure that the CR-3 ITS are in agreement with the edition of the Code being used with the IST program.

FPC requests approval of this LAR by April 12, 2009, with a 30 day implementation period. This will permit implementation of the amendment coincident with implementing the Fourth 10-year Interval Inservice Testing Program, scheduled to start May 11, 2009. FPC is not proposing any variations or deviations from the ITS changes described in TSTF 479, Revision 0, or TSTF-497, Revision 0.

Progress Energy Florida, Inc.
Crystal River Nuclear Plant
15760 W. Powerline Street
Crystal River, FL 34428

A047
NRR

No new regulatory commitments are made in this letter.

The CR-3 Plant Nuclear Safety Committee has reviewed this request and recommended it for approval.

If you have any questions regarding this submittal, please contact Mr. Dan Westcott, Supervisor, Licensing and Regulatory Programs at (352) 563-4796.

Sincerely,

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

DEY/par

Attachments:
A. Description of the Proposed Change, Background, Justification for the Request, Determination of No Significant Hazards Consideration, and the Environmental Assessment
B. Proposed Technical Specification Page Changes Strikeout and Shadowed Text Format
C. Proposed Technical Specification Page Changes Revision Bar Format
D. Proposed Revised Bases Pages (For Information Only) Strikeout and Shadowed Text Format

xc: NRR Project Manager
Regional Administrator, Region II
Senior Resident Inspector
State Contact

STATE OF FLORIDA

COUNTY OF CITRUS

Dale E. Young states that he is the Vice President, Crystal River Nuclear Plant for Florida Power Corporation, doing business as Progress Energy Florida, Inc.; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 17 day of December, 2008, by Dale E. Young.

Carolyn E Portmann

Signature of Notary Public
State of Florida

CAROLYN E. PORTMANN
Notary Public - State of Florida
My Commission Expires Mar 1, 2010
Commission # DD 524380
Bonded By National Notary Assn.

(Print, type, or stamp Commissioned Name of Notary Public)

Personally Known ✓ -OR- Produced Identification _____

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #300, REVISION 0

ATTACHMENT A

DESCRIPTION OF THE PROPOSED CHANGE, BACKGROUND, JUSTIFICATION FOR THE REQUEST, DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION, AND THE ENVIRONMENTAL ASSESSMENT

DESCRIPTION OF THE PROPOSED CHANGE BACKGROUND, JUSTIFICATION FOR THE REQUEST, DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION, AND THE ENVIRONMENTAL ASSESSMENT

1.0 DESCRIPTION OF PROPOSED CHANGE

The proposed change would revise the Crystal River Unit 3 (CR-3) Improved Technical Specifications (ITS) to incorporate the changes identified in Technical Specification Task Force (TSTF) Traveler 479, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a," and TSTF Traveler 497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less." The proposed change will revise ITS Section 5.6.2.9, Inservice Testing Program, for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2, and Class 3. The proposed change will also limit applying Surveillance Requirement (SR) 3.0.2 to surveillances with a frequency of two years or less. The proposed wording for Section 5.6.2.9 is as follows:

> 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. Provisions that inservice testing of ASME Code Class 1, 2, and 3 pumps, valves, and snubbers shall be performed in accordance with the ASME **Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code)** and applicable Addenda as required by 10 CFR 50.55a;
>
> b. Testing frequencies specified in the ASME **OM** Code and applicable Addenda;
>
> c. The provisions of SR 3.0.2 are applicable to the above required Frequencies and **to other normal and accelerated Frequencies specified as two years or less in the Inservice Testing Program** for performing inservice testing activities;
>
> d. The provisions of SR 3.0.3 are applicable to inservice testing activities; and
>
> e. Nothing in the ASME **OM** Code shall be construed to supersede the requirements of any TS.

2.0 BACKGROUND

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 testing of nuclear plant components. Until 1990, the ASME Code requirements addressing the inservice testing of pumps and valves were contained in Section XI, Subsection 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 for pumps and valves. Since the establishment of the 1990 Edition of the OM Code, the rules for the inservice testing for pumps and valves are no longer being updated in Section XI. As identified in NRC SECY-99-017, "Proposed Amendment to 10 CFR 50.55a," dated January 13, 1999, the NRC has generally considered the evolution of the ASME Code to result in a net improvement in

the measures for inspecting piping and components and testing pumps and valves. CR-3 had previously adopted the 1989 Edition of the Code for testing pumps and valves during the Third ten-year interval.

The Technical Specification Task Force recognized the fact that many plants were about to transition to the next ten-year inservice inspection and testing interval, and developed the travelers to standardize the wording for the inevitable ITS changes. The NRC approved TSTF 479, Revision 0 in December 2005 and TSTF 497, Revision 0 in July 2006. The NRC approved both TSTF travelers as administrative changes to the ITS NUREGs and incorporated them into Revision 3.1 of the ITS NUREGs.

3.0 JUSTIFICATION FOR THE REQUEST

The purpose of the Inservice Testing (IST) Programs 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, "Codes and Standards," defines the requirements for applying industry codes to each licensed nuclear powered facility. Licensees are required by 10 CFR 50.55a(f)(4)(i) to initially prepare programs to perform inservice testing of certain ASME Section III, Code Class 1, 2, and 3 pumps and valves during the initial 120-month interval. The regulations require that programs be developed utilizing the latest edition and addenda incorporated into paragraph (b) of 10 CFR 50.55a on the date 12 months prior to the date of issuance of the operating license subject to the limitations and modifications identified in paragraph (b).

NRC regulations also require that the Inservice Testing Programs be revised during successive 120-month intervals to comply with the latest edition and addenda of the Code incorporated by reference in paragraph (b), 12 months prior to the start of the interval.

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

The Technical Specification Inservice Testing Program is revised to indicate that the provisions of SR 3.0.2 are applicable to other IST Frequencies that are not specified in the Program. The Inservice Test Program may have frequencies for testing that are based on risk and do not conform to the standard testing frequencies specified in the Technical Specifications. For example, an Inservice Testing Program may use ASME Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light-Water Reactor plants," in lieu of stroke time testing. The Frequency of the Surveillance may be determined through a mix of risk informed and performance based means in accordance

with the Inservice Testing Program. This is consistent with the guidance in NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," which indicates that the 25% extension of the interval specified in the Frequency would apply to increased frequencies the same way that it applies to regular frequencies. If a test interval is specified in 10 CFR 50.55a, the TS SR 3.0.2 Bases indicate that the requirements of the regulation take precedence over the Technical Specifications.

However, at the February 23, 2006 meeting between the NRC and the Technical Specification Task Force, members of the Component Branch of the NRC stated that TSTF 479 did not provide an adequate justification for applying SR 3.0.2 to Frequencies specific in the IST Program as greater than two years, and the NRC would not approve plant specific amendments based on TSTF-479 incorporating this change without further justification. After consideration, the TSTF declined to develop a technical justification for applying SR 3.0.2 to IST Frequencies specified as greater than two years at this time due to inadequate cost benefit.

The Third 10-year inservice testing interval for CR-3 will conclude on May 10, 2009 since the Fourth 10-year inservice testing interval begins on May 11, 2009. The Code of record for the Third 10-year inservice testing interval has been the 1989 Edition of the ASME Code, Section XI, with no addenda. The Fourth 10-year inservice testing interval for CR-3 will use the 2001 Edition through the 2003 Addenda of the ASME Code as the Code of record. Currently, the CR-3 ITS does not address the OM Code, and as such, will not be completely and administratively prepared for the Fourth 10-year inservice testing interval. The proposed changes are necessary to achieve consistency with the inservice testing requirements of 10 CFR 50.55a, beginning with the Fourth 10-year inservice testing interval.

Therefore, FPC is providing License Amendment Request #300, Revision 0, that incorporates the approved wording found in TSTF 479 and TSTF 497.

4.0 NO SIGNIFICANT HAZARDS CONSIDERATION

Florida Power Corporation (FPC) has evaluated the proposed License Amendment Request (LAR) against the criteria of 10 CFR 50.92(c) to determine if any significant hazards consideration is involved. FPC has concluded that this proposed LAR does not involve a significant hazards consideration. The following is a discussion of how each of the 10 CFR 50.92(c) criteria is satisfied.

(1) Does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change revises the CR-3 ITS, Section 5.6.2.9, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, Class 2, and Class 3. 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. The proposed change does not

involve the addition or removal of any equipment, or any design changes to the facility. Therefore, this proposed change does not involve an increase in probability or consequences of an accident previously evaluated.

(2) *Does not create the possibility of a new or different kind of accident from any accident previously evaluated.*

The proposed change revises the CR-3 ITS, Section 5.6.2.9, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, Class 2, and Class 3. 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 involve a change in the methods governing normal plant operation. The proposed change will not introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in types or increases in the amounts of any effluents that may be released offsite and there is no increase in individual or cumulative occupational exposure. Therefore, the proposed change does not create the possibility of an accident of a different kind than previously evaluated.

(3) *Does not involve a significant reduction in a margin of safety*

The proposed change revises the CR-3 ITS, Section 5.6.2.9, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, Class 2, and Class 3. 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 the methods governing normal plant operation. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves. The safety function of the affected pumps and valves will be maintained. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

5.0 ENVIRONMENTAL IMPACT EVALUATION

10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if the amendment changes a requirement with respect to use of a facility component within the restricted area provided that (i) the amendment involves no significant hazards consideration, (ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and (iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Florida Power Corporation (FPC) has reviewed this License Amendment Request (LAR) and has determined that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22, no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the proposed license amendment. The basis for this determination is that for this amendment:

(i) The proposed license amendment does not involve a significant hazards consideration, as described in the significant hazards evaluation.

(ii) As discussed in the Justification for the Request and the No Significant Hazards Consideration, this change does not result in a significant change or significant increase in the release associated with any Design Basis Accident. The CR-3 ITS Inservice Testing Program will remain in compliance with the requirements of 10 CFR 50.55a. Likewise, there will be no significant change in the types or a significant increase in the amounts of any effluents released offsite during normal operation.

(iii) The proposed amendment does not have any impact on the performance of required surveillance testing that ensures pumps and valves remain capable of performing as designed. As specific Code required tests have evolved, this proposed change ensures that the CR-3 Inservice Testing Program evolves with the Code. Therefore, the proposed LAR does not result in a significant increase to the individual or cumulative occupational radiation exposure.

6.0 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

NRC regulation, 10 CFR 50.55a, defines the requirements for applying industry codes to each licensed nuclear powered facility. Licensees are required by 10 CFR 50.55a(f)(4)(i) to initially prepare programs to perform inservice testing of certain ASME Section III, Code Class 1, 2, and 3 pumps and valves during the initial 120-month interval. The regulations require that programs be developed utilizing the latest edition and addenda incorporated into paragraph (b) of 10 CFR 50.55a on the date 12 months prior to the date of issuance of the operating license subject to the limitations and modification identified in paragraph (b).

NRC regulations also require that the Inservice Testing Programs be revised during successive 120-month intervals to comply with the latest edition and addenda of the Code incorporated by reference in paragraph (b), 12 months prior to the start of the interval.

This Technical Specification change will not reduce the leak-tightness of the containment nor the integrity of the Reactor Coolant System. Therefore, 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) Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

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ATTACHMENT B

PROPOSED TECHNICAL SPECIFICATION PAGE CHANGES

STRIKEOUT AND SHADOWED TEXT FORMAT

5.6 Procedures, Programs and Manuals (continued)

5.6.2.9 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. Provisions that inservice testing of ASME Code Class 1, 2, and 3 pumps, valves, and snubbers shall be performed in accordance with ~~Section XI of~~ the ASME ~~Boiler and Pressure Vessel~~ Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR 50.55a;

b. Testing frequencies specified in ~~Section XI of~~ the ASME OM ~~Boiler and Pressure Vessel~~ Code and applicable Addenda;

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

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

e. Nothing in the ASME ~~Boiler and Pressure Vessel~~ OM Code shall be construed to supersede the requirements of any TS.

(continued)

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

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LICENSE AMENDMENT REQUEST #300, REVISION 0

ATTACHMENT C

PROPOSED TECHNICAL SPECIFICATION PAGE CHANGES

REVISION BAR FORMAT

5.6 Procedures, Programs and Manuals (continued)

5.6.2.9 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. Provisions that inservice testing of ASME Code Class 1, 2, and 3 pumps, valves, and snubbers shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR 50.55a;

b. Testing frequencies specified in the ASME OM Code and applicable Addenda;

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

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

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

(continued)

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

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LICENSE AMENDMENT REQUEST #300, REVISION 0

ATTACHMENT D

PROPOSED REVISED BASES PAGES
(FOR INFORMATION ONLY)

STRIKEOUT AND SHADOWED TEXT FORMAT

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 (continued)

To meet the Code requirements, CR-3 typically removes the valves and ships them to the vendor to be bench tested. Alternately, the valves may be tested in-place. If tested in-place, pressurizer safety valves are to be tested one at a time and in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 2% for OPERABILITY; however, valves removed for testing or maintenance are required to be reset to ± 1% as part of the Surveillance to allow for drift.

The Note allows entry into MODE 3 with the lift settings outside the SR limits. This permits testing and examination of the safety valves at pressures and temperatures near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is based on an 18 hour outage time for each valve. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe. As mentioned earlier, this allowance is not utilized at the present time since current practice is to remove the valves and send them to the vendor for testing and lift setting adjustment.

REFERENCES

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

2. B&W Report 86-1200382-00, November 1990.

3. B&W Topical Report BAW-10043, "Overpressure Protection for B&W's Pressurized Water Reactors."

4. ASME~~, Boiler and Pressure Vessel~~ Code for the Operation and Maintenance of Nuclear Power Plants (ASME OM Code) Code~~, Section XI~~.

BASES

ACTIONS

A.1 and A.2

With the PORV inoperable, the PORV must be restored or the flow path isolated within 1 hour. The block valve should be closed and power must be removed from the block valve to eliminate the potential for inadvertent PORV opening and depressurization.

B.1.1, B.1.2, B.2.1, and B.2.2

If the block valve is inoperable, it must be restored to OPERABLE status or the flowpath isolated within 1 hour. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status, the Required Action is to close the block valve and remove power or close the PORV and remove power to its associated solenoid valve. Either of the two Required Actions will render the PORV isolated. The 1 hour Completion Times are consistent with an allowance of some time for correcting minor problems, restoring the valve to operation, and establishing correct valve positions and restricting the time without adequate protection against RCS depressurization.

C.1 and C.2

If the Required Action and associated Completion Time is not met, the plant must be placed in MODE in which the requirement does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The Completion Times are reasonable, based on operating experience, to reach the specified MODES and conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

Block valve cycling verifies that it can be closed if needed. The Frequency of 92 days is based on ASME OM Code, ~~Section XI~~ (Ref. 3) requirements. Block valve cycling, as stated in the Note, is not required to be performed when it

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1 (continued)

is closed for isolation since cycling could increase the hazard of an existing degraded flow path.

SR 3.4.10.2

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish plant conditions most representative of those under which the valve is expected to operate and is consistent with the recommendations of NRC Generic Letter 90-06.

PORV cycling demonstrates its function. The Frequency of 24 months is based on a typical refueling cycle and industry accepted practice.

REFERENCES

1. NUREG-0737, November 1980.

2. NRC IE Bulletin 79-05B, April 1979.

3. ASME~~, Boiler and Pressure Vessel~~ Code for the Operation and Maintenance of Nuclear Power Plants (ASME OM Code) Code~~, Section XI~~.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (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 were not individually leakage tested, one valve could have failed completely and not detected provided the other valve in series met the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

The ASME~~,~~ OM Code ~~Section XI~~ (Ref. 3) permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential). Reference 3 allows this reduced pressure testing for those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening, e.g., check valves. In such cases, the observed rate should be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

The Frequency of testing is a combination of ASME Code and PIV Order requirements.

The Inservice Testing Program implements the ~~American Society of Mechanical Engineers (~~ASME~~)~~ OM Code~~, Section XI~~ (Ref. 5), cold shutdown performance requirement. This requirement is based on the need to perform this Surveillance under conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the plant at power.

The Frequency of prior to entering MODE 2 whenever the plant has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months was contained in the April 20, 1981 PIV Order (Ref. 6). It was intended to provide confidence the valves re-seated following any period of extended operation with flow through the valves. The 7 day value is based on NUREG 1366 recommendations (Ref. 4).

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable RCS conditions to allow for performance of this Surveillance. The Note

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

that allows this provision is complimentary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months.

SR 3.4.13.2 and SR 3.4.13.3

Verifying ACIS is OPERABLE ensures that RCS pressure will not pressurize the DHR system beyond its design pressure of 330 psig on the suction side and 450 psig on the discharge side of the pump. The setpoint is adjusted to account for elevation differences between the pressure instrument and the drop line and is set so RCS hot leg pressure must be < 284 psig to open the valves. This setpoint ensures the DHR design pressure will not be exceeded and the DHR relief valves will not lift. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

REFERENCES

1. NUREG-75/014, Appendix V, October 1975.

2. NUREG-0677, NRC, May 1980.

3. ASME~~, Boiler and Pressure Vessel Code, Section XI, Article IWV-3423(e).~~ Code for the Operation and Maintenance of Nuclear Power Plants (ASME OM Code)

4. NUREG-1366, December 1992.

~~5. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWV-3422~~ Deleted

6. NRC Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves dated 4/20/81. Includes Technical Evaluation Report, "Primary Coolant System Pressure Isolation Valves," prepared by the Franklin Research Center.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1 (continued)

The 31 day frequency is appropriate because the valves are operated under administrative control, and an inoperable valve position would only affect a single train. This frequency has been shown to be acceptable through operating experience.

SR 3.5.2.2

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the American Society of Mechanical Engineers (ASME) Code (Ref. 4). This type of testing may be accomplished by measuring the pump's developed head at only one point of the pump's characteristic curve and this point may be anywhere on the 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 plant accident analysis. SRs are specified in the Inservice Testing Program, which encompasses ~~Section XI of~~ the ASME OM Code. ~~Section XI of~~ Ithe ASME OM Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.3 and SR 3.5.2.4

These SRs demonstrate that each automatic ECCS valve that is not locked, sealed, or otherwise secured in position, actuates to its required position on an actual or simulated ESAS signal and that each ECCS pump starts on receipt of an actual or simulated ESAS signal. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the ESAS testing, and equipment performance is monitored as part of the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.7

Periodic inspections of the reactor building emergency sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and to preserve access to the location. This Frequency has been found to be sufficient to detect abnormal degradation and has been confirmed by operating experience.

REFERENCES

1. 10 CFR 50.46.

2. FSAR, Section 6.1.

3. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

4. American Society of Mechanical Engineers, ~~Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWP-3000.~~ Code for the Operation and Maintenance of Nuclear Power Plants (ASME OM Code).

5. BAW-2295-A, Revision 1, Justification for Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems.

6. FSAR, Section 4.3.10.1.

7. Letter from NRC to FPC, 3N1098-15, dated October 29, 1998, "Issuance of Exemption from the Requirements of 10 CFR 50, Appendix K, Section I.D.1 - Crystal River Unit 3 (TAC No. M99892)".

BASES

ACTIONS

B.1 and B.2 (continued)

status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock and purge valve with resilient seal leakage limits for SR 3.6.2.1 and 3.6.3.6 does not constitute a failure of this Surveillance unless the contribution from these penetrations causes overall Type A, B, and C leakage to exceed limits. SR Frequencies are as required by the Containment Leakage Rate Testing Program. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Inspection Program. Testing and Frequency are in accordance with Subsections IWE and IWL of the ~~1992~~ 2001 Edition through the 2003 Addenda of the ASME Code and 10 CFR 50.55a.

Abnormal degradation shall be determined by engineering evaluation. In the event abnormal degradation is detected, a Special Report shall be submitted in accordance with ITS 5.7.2.b. The impact of large-scale tendon degradation should also be evaluated with respect to Containment OPERABILITY. In this context, containment structural integrity is analogous to containment OPERABILITY.

(continued)

BASES

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. FSAR, Sections 14.2.2
3. FSAR, 5.2.1.1
4. ~~1992~~ 2001 Edition, through the 2003 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWE and IWL
5. 10 CFR 50.67
6. NEI 94-01, Revision ~~0~~ 1, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J"
7. ANSI/ANS-56.8 1994, "American National Standard for Containment System Leakage Testing Requirement"

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.2

Operating each required containment cooling train fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration 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 a significant degradation of the containment cooling trains occurring between surveillances and has been shown to be acceptable through operating experience.

It is preferable to run the fans in slow speed for this SR since this provides additional confidence the post-accident containment cooling train circuitry is OPERABLE.

SR 3.6.6.3

Verifying that each RB 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 RB Spray System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.4

Verifying an emergency design cooling water flow rate of ≥ 1780 gpm to each required containment cooling system heat exchangers (fan cooling coils) ensures the design flow rate assumed in the safety analysis is being achieved. The SR verifies that, with the SW System in the post-accident ES alignment, adequate flow is provided to the heat exchangers to remove the design basis reactor building heat load. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. While the heat exchangers can be aligned to the SW System during normal operations, other critical normal-running SW loads make it impractical to verify accident flow rate to the heat exchangers with the plant on-line. On an ES actuation, these normal-running loads are isolated and the SW flow

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.8 (continued)

For activities, such as a valve repair/replacement, a visual inspection would be the preferred post-maintenance test since small debris in a localized area is the most likely concern. A smoke or air test would be appropriate following an event where a large amount of debris entered the system or water was actually discharged through the spray nozzles. For an inadvertent actuation of the Reactor Building Spray system, an air or smoke test should be performed at the next outage of sufficient duration.

REFERENCES

1. FSAR, Section 1.4.

2. FSAR, Section 14.2.2.5.9.

3. FSAR, Section 6.3.

4. RO-2787 Requirement Outline, Reactor Building Fan Assemblies, Addendum B, February 19, 1971.

5. BAW-2295-A, Revision 1, Justification for Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems.

6. ASME Code for the Operation and Maintenance of Nuclear Power Plants (ASME OM Code). ~~,Boiler and Pressure Vessel Code, Section XI.~~

BASES

ACTIONS

A.1 and A.2 (continued)

The Completion Time of 12 hours for Required Action A.2 is based on operating experience in resetting all four channels of the overpower trip protective function and on the low probability of the occurrence of a transient that could result in OTSG overpressure during this period.

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status in the associated Completion Time or there are no OPERABLE MSSV with a nominal lift setpoint of 1050 (± 3%) on an OTSG, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR demonstrates MSSV OPERABILITY by verifying each valves lift setpoint in accordance with the Inservice Testing Program. The ASME OM Code, ~~Section XI~~ (Ref. 8) requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1981 (Ref. 9). According to Reference 9, the following tests are required for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

The ANSI/ASME Standard requires the testing of all valves every 5 years, with a minimum of 20% of the valves tested every 24 months. Reference 8 provides the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-1 allows a ± 3% setpoint tolerance for OPERABILITY; however, valves removed for maintenance or testing are required to be reset to ± 1% following re-installation in order to allow for drift. Administrative limits on the as-left low-end MSSV setpoint (i.e. 1050 psig, -1%) have been established to reduce the probability of an inadvertent opening of the valve during normal plant operation.

This SR is modified by a Note that provides an SR 3.0.4 type exception and 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 is corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, 1971.
2. B&W Document 51-1174336-00, March 2, 1989.
3. FSAR, Section 10.3.5.
4. FSAR, Section 14.1.2.8.
5. FSAR, Section 14.2.2.9.
6. FSAR, Section 14.2.2.2.
7. SER on Amendment 77, dated April 25, 1985.
8. ASME~~, Boiler and Pressure Vessel Code, Section XI, Article IWV-3500~~ Code for the Operation and Maintenance of Nuclear Power Plants (ASME OM Code).
9. ANSI/ASME OM-1-1981.
10. B&W Document 86-1219188-00.

BASES

ACTIONS (continued)

B.1 and B.2

If the Required Action and associated Completion Time are not met, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is in accordance with the limit specified in the Inservice Testing Program. This MSIV closure time is established based upon design specifications for the valves and is consistent with the accident and containment analyses. This Surveillance is normally performed upon returning the plant to operation following a refueling outage, because the MSIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the plant generating power. As the MSIVs are not to be tested at power, they are exempt from the ASME OM Code~~, Section XI~~ (Ref. 7) quarterly valve stroke requirements.

The Frequency for this SR is in accordance with the Inservice Testing Program and is based on the typical refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at this Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish plant conditions most representative of those under which the acceptance criterion was generated.

(continued)

BASES

REFERENCES

1. Enhanced Design Basis Document for the Main Steam System.

2. FSAR, Section 10.2.1.4.

3. 10 CFR 50, Appendix A, GDC 57.

4. FSAR, Section 14.2.2.1.7.

5. FSAR, Section 14.2.2.2.

6. 10 CFR 50.67.

7. ASME~~, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400~~ Code for the Operation and Maintenance of Nuclear Power plants (ASME OM Code).

BASES

ACTIONS

D.1 (continued)

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valve in the flowpath, the MFW ump trip feature provided on low OTSG pressure, and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

E.1 and E.2

If the MFIVs cannot be restored to OPERABLE status, or closed, or isolated within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that MFIV closure time is within the acceptance criteria in the Inservice Testing Program. In order to be consistent with the safety analysis as documented in FTI Summary Analysis Report 86-1266223-00, "CR-3 MSLB with MFP Trip Failure," the required stroke time of the MFIVs, except for the low load block valves FWV-31 and FWV-32, is 34 seconds which includes an EFIC signal process delay and valve closure from the time of OTSG low pressure of 585 psig. The actual EFIC actuation on OTSG low pressure is greater than or equal to 600 psig. The lower analysis pressure is conservative. The low load block valves FWV-31 and FWV-32 are required to stroke close in 67 seconds which includes an EFIC signal process delay and valve closure. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The MFIVs should 1 not be tested at power since even a part stroke exercise increases the risk of a valve closure, and the risk of a plant transient with the plant generating power. As these valves are not tested at power, they are exempt from the ASME OM Code~~, Section XI~~ (Ref. ~~2~~3) quarterly valve stroke requirements.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish the test conditions most representative of those under which the acceptance criterion was generated.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 (continued)

The Frequency for this SR is in accordance with the Inservice Testing Program. Operating experience has shown that these components usually pass the Surveillance when performed at this Frequency.

REFERENCES

1. FSAR, Section 10.2.1.2.

2. FSAR, Section 7.2.4.2.

3. ASME~~, Boiler and Pressure Vessel Code, Section XI~~ Code for the Operation and Maintenance of Nuclear Power Plants (ASME OM Code).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.2 (continued)

introduce cold EFW into the OTSGs while they are operating, this test is performed on recirculation at a reduced flow rate.

Periodically comparing the reference differential pressure developed at this reduced flow detects trends that might be indicative of degrading pump performance. Performance of inservice testing discussed in the ASME OM Code, ~~Section XI (Ref. 5),~~ at 3 month intervals, is satisfied by this SR. The 45 day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 5.

This SR is modified by a Note indicating that the SR may be deferred until suitable test conditions are established. This SR 3.0.4 type exception may be necessary during any plant start-up because there is insufficient steam pressure in the secondary side of the OTSGs to perform this SR on the turbine-driven pump.

SR 3.7.5.3

This SR verifies that EFW can be delivered to the appropriate OTSG in the event of any accident or transient that generates an EFIC signal by demonstrating that each automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, actuates to its correct position on an actual or simulated actuation signal.

Valves secured in the correct position need not demonstrate the capability to achieve this configuration. ADVs also need not demonstrate the capability to satisfy this SR since their operation is not credited as part of any DBA.

The SR also verifies the EFW control and block valves actuate to the isolation position on a simulated or actual vector valve control signal.

This SR is a test of the integrated system response to an actuation signal and as such, it is not necessary to verify the EFW System actuates on each EFIC signal. Any of the initiation signals described in the Background Section of these Bases is adequate, given that the various EFIC

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.5 (continued)

of EFW flow paths must be demonstrated before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment, in view of other administrative controls to ensure that the flow paths are OPERABLE. To further ensure EFW 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 EFW tank to the OTSGs is properly aligned. This requirement is based upon the recommendation of NUREG 0737. The Frequency was modified slightly during ITS development (prior to entering MODE 2) to provide an SR 3.0.4 type exception. As written, the SR allows the plant to achieve and maintain MODE 3 conditions in order to perform the verification.

SR 3.7.5.6

Verifying battery terminal voltage ensures the ability of the battery to perform the intended function. The voltage requirements are based on the nominal design voltage of the battery. The 7 day frequency is consistent with IEEE-450.

REFERENCES

1. Enhanced Design Basis Document for the Emergency Feedwater and Emergency Feedwater Initiation and Control System.

2. BAW-10043, "Overpressure Protection for B&W Reactors," dated May 1972.

3. FSAR, Section 10.5.

4. 10 CFR 50, Appendix A.

5. ASME~~, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Subsection IWP.~~ Code for the Operation and Maintenance of Nuclear Power Plants (ASME OM Code).

6. Deleted.

7. FPC calculation 187-0008, Rev. 6.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, plant safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following periodic governor replacement, corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. However, the Note acknowledges that credit may be taken for unplanned events that satisfy this SR.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. FSAR, Chapter 8.
3. Regulatory Guide 1.9, Rev. 3, July 1993.
4. FSAR, Chapter 6.
5. FSAR, Chapter 14.
6. Regulatory Guide 1.93, Rev. 0, December 1974.
7. Generic Letter 84-15.
8. 10 CFR 50, Appendix A, GDC 18.
9. Regulatory Guide 1.108, Rev. 1, August 1977.
10. Regulatory Guide 1.137, Rev. 1, October 1979.
11. ANSI C84.1-1982.
12. ASME~~, Boiler and Pressure Vessel Code, Section XI~~ Code for the Operation and Maintenance of Nuclear Power Plants (ASME OM Code).
13. Deleted.