NMC
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Prairie Island Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

JUL 0 3 2007

L-PI-07-048
10 CFR 50.90

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

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

License Amendment Request (LAR) Incorporating Technical Specification Task Force (TSTF) Industry Travelers TSTF-479, TSTF-485, and TSTF-497

Pursuant to 10 CFR 50.90, the Nuclear Management Company, LLC (NMC) hereby requests an amendment to the Technical Specifications (TS) for the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2 to revise TS 1.4, "Frequency", to incorporate changes consistent with industry traveler TSTF-485, "Correct Example 1.4-1", and revise TS 5.5.7, "Inservice Testing Program", to incorporate changes consistent with TSTF-479, "Changes to Reflect Revision of 10 CFR [Code of Federal Regulations] 50.55a", and TSTF-497, "Limit Inservice Testing Program SR [Surveillance Requirement] 3.0.2 Application to Frequencies of 2 Years or Less". NMC has evaluated the proposed changes in accordance with 10 CFR 50.92 and concluded that they involve no significant hazards consideration.

The enclosure to this letter contains the licensee's evaluation of the proposed changes.

NMC requests approval of this LAR within one calendar year of the submittal date. Upon Nuclear Regulatory Commission (NRC) approval, NMC requests 90 days to implement the associated changes. In accordance with 10 CFR 50.91, NMC is notifying the State of Minnesota of this LAR by transmitting a copy of this letter and enclosure to the designated State Official.

If there are any questions or if additional information is needed, please contact Mr. Dale Vincent, P.E., at 1-651-388-1121.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on JUL 03 2007

Michael D. Wadley

Michael D. Wadley
Site Vice President, Prairie Island Nuclear Generating Plant Units 1 and 2
Nuclear Management Company, LLC

Enclosure: Evaluation of Proposed Changes

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
State of Minnesota

ENCLOSURE

Evaluation of the Proposed Changes

License Amendment Request (LAR) Incorporating Technical Specification Task Force (TSTF) Industry Travelers TSTF-479, TSTF-485, and TSTF-497

ATTACHMENTS:

1. Technical Specification Pages (Markup)
2. Bases Pages (Markup) (For information only)
3. Technical Specification Pages (Retyped)

1. SUMMARY DESCRIPTION

This LAR is a request to amend Operating Licenses DPR-42 and DPR-60 for Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2.

The Nuclear Management Company, LLC (NMC) requests Nuclear Regulatory Commission (NRC) review and approval of proposed revisions to Technical Specification (TS) 1.4, "Frequency", and TS 5.5.7, "Inservice Testing Program". The proposed revisions will incorporate changes consistent with industry traveler TSTF-479, "Changes to Reflect Revision of 10 CFR [Code of Federal Regulations] 50.55a", Revision 0; TSTF-485, "Correct Example 1.4-1", Revision 0; and TSTF-497, "Limit Inservice Testing Program SR [Surveillance Requirement] 3.0.2 Application to Frequencies of 2 Years or Less", Revision 0. The Technical Specifications, with the revisions proposed in this LAR, meet applicable regulatory guidance.

2. DETAILED DESCRIPTION

2.1 Proposed Changes

A brief description of the associated proposed TS changes is provided below along with a discussion of the justification for each change. The specific wording changes to the TS are provided in Attachments 1 and 3 to this enclosure.

TS 1.4, "Frequency": This LAR proposes to modify the second paragraph of Example 1.4-1 to be consistent with the requirements of SR 3.0.4. This change is acceptable because it incorporates the changes in NRC approved TSTF-485, "Correct Example 1.4-1".

TS 5.5.7, "Inservice Testing Program": This LAR proposes revisions to TS 5.5.7.a which replace references to Section XI of the American Society of Mechanical Engineers (ASME) Code with references to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code), TS 5.5.7.b which restrict extension of Frequencies to those Frequencies specified as 2 years or less and TS 5.5.7.d which reference ASME OM Code. These changes are acceptable because they incorporate changes in NRC approved TSTF-479, "Changes to Reflect Revision of 10 CFR 50.55a", and TSTF-497, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less".

This LAR also proposes changes to TS 5.5.7.b which take exception to the limitation in the PINGP SR 3.0.2 which does not apply the 1.25 times extension to Frequencies of 24 months. This change is acceptable because: PINGP SR 3.0.2 contains a unique restriction; it implements the intent of the reference to SR 3.0.2 in TS 5.5.8, paragraph (b.) in NUREG-1431, "Standard Technical

Specifications, Westinghouse Plants", Revision 3.1 (NUREG-1431); and SR 3.0.2 states that exceptions are stated in the individual Specifications. Further discussion of the appropriateness of this change is provided in Section 4.0 below.

Although Bases changes are not a part of this LAR, Attachment 2 to this enclosure includes marked up Bases pages for information. The following Bases have been marked up with changes consistent with the intent of TSTF-479; due to the specific PINGP wording these Bases markups may differ from TSTF-479: B 3.4.10; B 3.4.11; B 3.4.15; B 3.5.2; B 3.6.5; B 3.7.1; B 3.7.2; B 3.7.3; B 3.7.5; and B 3.7.8. Due to the specific PINGP wording the PINGP Bases equivalent to following NUREG-1431, "Standard Technical Specifications, Westinghouse Plants" (NUREG-1431), Bases revised by TSTF-479 were not affected and are not included: B 3.4.12; B 3.6.12; and B 3.8.1.

In summary these changes are acceptable because they are consistent with current regulatory guidance.

2.2 Background

Preparation of this LAR followed the guidance of NEI 06-02, "License Amendment Request Guidelines", (Reference 1), Appendix D. Paragraph D3, 2 states:

> Maximize the use of cross-references to previously published information presented in the Traveler and NRC approval documentation to minimize the repetition of information. Repetition can be confusing because the NRC reviewer must compare the information restated in the LAR with the information in the Traveler and NRC documentation to ensure there are no differences.

Thus, this LAR makes extensive use of cross-references to the NRC-approved Travelers which are being adopted.

Revisions to TS 1.4

This LAR proposes changes consistent with TSTF-485. See TSTF-485, Revision 0, Section 3.0 for Background discussion. License Amendments 167 and 157 issued October 20, 2004 for PINGP Units 1 and 2 respectively, incorporated TSTF-359, "Increase Flexibility in MODE Restraints," Revision 9, into the PINGP TS.

Revisions to TS 5.5.7

This LAR requests changes to TS 5.5.7 consistent with TSTF-479 and TSTF-497. See TSTF-479, Revision 0, and TSTF-497, Revision 0, Sections 3.0 for Background discussions.

On July 26, 2002, the NRC issued License Amendments 158/149 which approved the NMC request to convert the PINGP TS to the format and content guidance of NUREG-1431 (conversion to improved TS). The conversion to improved TS included requirements in SR 3.0.2 unique to PINGP which restrict Frequency extensions for SRs with a stated Frequency of 24 months. This restriction was not intended to apply to Inservice Testing (IST) Frequency extensions. This LAR proposes changes to TS 5.5.7.b to clarify the use of extensions with IST.

With the TS changes proposed in this LAR the plant will continue to operate safely and the health and welfare of the public is protected.

3. TECHNICAL EVALUATION

PINGP is a two unit plant located on the right bank of the Mississippi River approximately 6 miles northwest of the city of Red Wing, Minnesota. The facility is owned by the Northern States Power Company (NSP) and operated by NMC. Each unit at PINGP employs a two-loop pressurized water reactor designed and supplied by Westinghouse Electric Corporation. The initial PINGP application for a Construction Permit and Operating License was submitted to the Atomic Energy Commission (AEC) in April 1967. The Final Safety Analysis Report (FSAR) was submitted for application of an Operating License in January 1971. Unit 1 began commercial operation in December 1973 and Unit 2 began commercial operation in December 1974.

The PINGP was designed and constructed to comply with NSP's understanding of the intent of the AEC General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967. PINGP was not licensed to NUREG-0800, "Standard Review Plan (SRP)."

<u>TS 1.4, "Frequency" proposed changes</u>

See TSTF-485, Revision 0, Section 4.0 for Technical Analysis discussion applicable to the proposed TS 1.4 changes. NMC has reviewed the TSTF-485 Technical Analysis and concluded that the analysis is applicable to PINGP, Units 1 and 2, and justifies this amendment for the incorporation of the changes into PINGP TS.

<u>TS 5.5.7, "Inservice Testing Program" proposed changes</u>

See TSTF-479, Revision 0, and TSTF-497, Revision 0, for Technical Analysis discussion applicable to the proposed TS 5.5.7 changes. NMC has reviewed the TSTF-479 and TSTF-497 Technical Analysis sections and concluded that the analyses are applicable to PINGP, Units 1 and 2, and justify this amendment for the incorporation of the changes into PINGP TS.

This LAR proposes changes to TS 5.5.7.b that are consistent with, but not identical to, the changes in TSTF-479 and TSTF-497. TS 5.5.7.b references SR 3.0.2 which for

PINGP has unique wording that does not allow extensions for SRs specified with a 24 month (2 year) Frequency. Therefore, to eliminate confusion, the first sentence of TS 5.5.7.b specifies that SR 3.0.2 is applicable to IST Frequencies of less than 2 years. This is a minor deviation from the TS change provided in TSTF-479 and TSTF-497 and does not change the intent or application of the TS.

In addition to changes to TS 5.5.7 that are consistent with TSTF-479 and TSTF-497, TS 5.5.7.b is revised to include an exception to the provision in SR 3.0.2 which does not allow Frequency extension for SRs with a specified Frequency of 24 months. Prior to conversion to ITS, PINGP TS 4.0.A.2 specified that, "The intervals between tests scheduled for refueling shutdowns shall not exceed two years." This limitation applied to SR Frequencies specified in the old TS format as "each refueling shutdown", "once each refueling interval", "each reactor refueling shutdown", "each refueling outage", or "each refueling shutdown on staggered test basis" as discussed in the Improved Technical Specifications (ITS) conversion NRC Safety Evaluation (SE), Reference 2. Since the pre-ITS TS allowed up to two years (24 months) for these SR Frequencies, NMC chose to set the Frequency for these SRs at 24 months in the ITS. Furthermore, NMC chose not to perform evaluations for these SRs based on the guidance of Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle" and, in lieu of such evaluations, proposed in SR 3.0.2 to not allow the interval extension (1.25 times the interval specified). For some Surveillances specified in the pre-ITS TS as "once per 18 months" or similar wording, NMC did perform evaluations based on the guidance of GL 91-04 to support specifying the Frequency as 24 months in ITS.

The reference in TS 5.5.7.b to SR 3.0.2 could be interpreted to prohibit the interval extension for IST Program Frequencies specified as 24 months. This would be an unintended consequence of the unique requirements in PINGP TS 3.0.2. This LAR proposes to remedy this situation by including a provision in TS 5.5.7.b which states that, as an exception to SR 3.0.2, IST Program testing with a Frequency specified as 2 years (24 months) is met if the test is performed within 30 months.

The interval extension limitation on 24 month SRs in SR 3.0.2 was specifically included to apply to the SRs explicitly listed in the TS with the Frequency specified as 24 months. The NRC ITS SE, Reference 2, stated:

> The current fuel cycle at PINGP cannot exceed 24 months and the CTS [current TS] does not include a provision for any extension of this surveillance interval. The CTS 4.0.A.2 states that 'The intervals between tests scheduled for refueling shutdowns shall not exceed two years.' The proposed ITS SR 3.0.2 retains this CTS requirement by specifying in part: '. . . The specified Frequency is met for each SR with a specified Frequency of 24 months if the Surveillance is performed within 24 months, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met . . .'.
>
> The licensee has categorized each ITS SR with a 24-month frequency into three

groups, depending on their origin and CTS frequency requirements.

Three Tables are included in the NRC SE which list all of the 24-month SRs included in the PINGP TS to which the SR 3.0.2 interval extension limitation was intended to apply. The pre-ITS TS did not specify any IST Program testing intervals except for reactor coolant pump flywheel testing which was specified for approximately 3-year and 10-year intervals. Thus the interval extension limitation of SR 3.0.2 was not intended to apply to IST Program testing.

Application of the 25% interval extension to IST Program testing specified as "24-months" is supported by the NRC approved TSTF-479 technical justification presented above. The Background Section of TSTF-497 discusses the February 23, 2006, meeting between the NRC Staff and the TSTF where the NRC presented their position that TSTF-479 did not adequately justify applying SR 3.0.2 to Frequencies specified in the IST Program as greater than 2 years. NRC support for use of the 25% interval extension with 24-month testing was reaffirmed in the Background discussion in TSTF-497, which states, "The NRC stated that they would accept applying SR 3.0.2 to IST Frequencies not listed in the Inservice Testing Program table provided that those Frequencies are specified in the Inservice Testing Program as 2 years or less." The "SR 3.0.2" to which the NRC referred is Standard TS which includes the 25% interval extension without the limitation included in the PINGP TS. Thus, the interval extension limitation in the PINGP SR 3.0.2 should not be applied to IST Program Frequencies specified as "2 years" or "24 months".

The NRC Staff also reaffirmed that the 25% extension applies to IST Frequencies specified as "2-years" in its Safety Evaluation, page 3, for the Cooper Nuclear Station adoption of TSTF-479 (Reference 3) which stated, "Application of SR 3.0.2 to frequencies of 2 years or less, however, is consistent with the staff position contained in NUREG-1482, 'Guidelines for Inservice Testing at Nuclear Power Plants.'"

An exception in 5.5.7.b is an appropriate remedy within the format of the TS because SR 3.0.2 specifically states, "Exceptions to this Specification are stated in the individual Specifications."

Conclusions

This LAR proposes TS changes which will make Example 1.4-1 consistent with SR 3.0.4, update references to ASME Code which applies to the Inservice Testing Program and clarify that a 25% interval extension applies to IST Program testing with Frequencies specified as 24 months. These changes are consistent with NRC approved industry travelers TSTF-479, TSTF-485 and TSTF-497, the basis for NRC approval of the PINGP improved TS, and NRC Staff intent. Operation and maintenance of the Prairie Island Nuclear Generating Plant with the proposed TS revisions will continue to protect the health and safety of the public.

4. REGULATORY SAFETY ANALYSIS

4.1 Applicable Regulatory Requirements/Criteria

The Applicable Regulatory Requirements and Criteria are addressed in Section 5.2 of Technical Specification Task Force (TSTF) industry traveler TSTF-479, Revision 0; Section 5.2 of TSTF-485, Revision 0; and Section 5.0 (Regulatory Analysis) of TSTF-497, Revision 0. The Nuclear Management Company has reviewed these sections of TSTF-479, TSTF-485 and TSTF-497 and concluded that the information is applicable to the Prairie Island Nuclear Generating Plant, Units 1 and 2, and justifies this amendment for the incorporation of the changes into the Prairie Island Nuclear Generating Plant Technical Specifications.

4.2 Precedent

The NRC approved TSTF-479 (Reference 4) by letter dated December 6, 2005 (Reference 5) and accepted TSTF-497 in October, 2006 (Reference 6). TSTF-479 states, "The first plant to submit an amendment request based on this Traveler will be considered the 'lead plant' submittal and a generic Safety Evaluation will be written for the Traveler." The requested changes to Technical Specification 5.5.7, which are consistent with TSTF-479 and TSTF-497, are similar to those granted to the Wolf Creek Generating Station in NRC Safety Evaluation dated November 15, 2006, (Reference 7) which appears to be the first approved license amendment request based on TSTF-479 for a Westinghouse plant. The Nuclear Management Company has reviewed the NRC Safety Evaluation in Reference 7 and concluded it is applicable to the Prairie Island Nuclear Generating Plants, Units 1 and 2, to the extent that it has generic application, applies to Prairie Island Nuclear Generating Plant Technical Specification 5.5.7, and justifies this amendment for the incorporation of the changes into the Prairie Island Nuclear Generating Plant Technical Specifications.

4.3 Significant Hazards Consideration

This license amendment request proposes administrative changes to revise Technical Specification 1.4, "Frequency", for consistency with Surveillance Requirement 3.0.4 and Limiting Condition for Operation 3.0.4, and Technical Specification 5.5.7, "Inservice Testing Program", for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves. The proposed changes improve the Technical Specification guidance for usage and incorporate revisions to the American Society of Mechanical Engineers Code that may result in a net improvement in the measures for testing pumps and valves.

The No Significant Hazards Considerations are addressed in Section 5.1 of TSTF-479, Revision 0; Section 5.1 of TSTF-485, Revision 0; and Section 5.0 (Regulatory Analysis) of TSTF-497, Revision 0. The Nuclear Management Company has reviewed these sections of TSTF-479, TSTF-485 and TSTF-497 and concluded that the information is applicable to the Prairie Island Nuclear Generating Plant, Units 1 and 2, and is hereby incorporated by reference to satisfy the requirements of 10CFR 50.91(a).

4.4 Conclusions

In conclusion, based on the considerations discussed in the TSTFs referenced 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. ENVIRONMENTAL CONSIDERATION

A 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 significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. REFERENCES

1. NEI 06-02, "License Amendment Request Guidelines", December 2006, Accession No. ML070360327.

2. Prairie Island Nuclear Generating Plant, Units 1 and 2 – Issuance of Amendments RE: Conversion to Improved Technical Specifications (TAC Nos. MB0695 and MB0696), Accession No. ML022070654.

3. Cooper Nuclear Station – Issuance of Amendment RE: Technical Specification (TS) Changes Associated with Inservice Testing Program, Section 5.5.6, Under TS Programs and Manuals (TAC No. MD0335), dated September 6, 2006, Accession No. ML061440049.

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

5. Status of TSTF 343, 479, 482, 485, dated December 6, 2005, Accession No. ML053460302.

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

7. Wolf Creek Generating Station – Issuance of Amendment RE: Revision to Technical Specification 5.5.8 on the Inservice Testing Program (TAC No. MC9726), dated November 15, 2006, Accession No. ML062980233.

ENCLOSURE, ATTACHMENT 1

Technical Specification Pages (Markup)

1.4-3
5.0-12
5.0-13

3 pages follow

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. T~~t~~he Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the MODE or other specified condition or the LCO is considered not met (in accordance with SR 3.0.1) and LCO~~.~~ ~~Failure to do so would result in a violation of SR~~ 3.0.4 becomes applicable.

All markup per TSTF-485

5.5 Programs and Manuals (continued)

5.5.7 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~~specified in Section XI of~~ the ASME ~~Boiler and Pressure Vessel~~ Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

Markup per TSTF-479

ASME OM ~~Boiler and Pressure Vessel~~ Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Semiquarterly	At least once per 46 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

Mark-up consistent with TSTF-479 and TSTF-497 (revised due to plant specific SR 3.0.2 wording)

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as less than 2 years in the Inservice Testing (IST) Program for performing inservice testing activities. As an exception to SR 3.0.2, IST Program testing with the Frequency specified as 2 years (24 months) is met if the test is performed within 30 months (1.25 times the interval specified in the Frequency);

Plant specific markup

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

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

Markup per TSTF-479

5.5 Programs and Manuals (continued)

5.5.8 Steam Generator (SG) Program

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

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

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

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and, except for flaws addressed through application of the alternate repair criteria discussed in

ENCLOSURE, ATTACHMENT 2

Bases Pages (Markup)

(For Information Only)

B 3.4.10-5
B 3.4.11-8
B 3.4.15-7
B 3.5.2-11
B 3.6.5-11
B 3.6.5-13
B 3.7.1-4
B 3.7.1-5
B 3.7.2-7
B 3.7.3-7
B 3.7.5-11
B 3.7.5-14
B 3.7.8-17

13 pages follow

BASES

ACTIONS

B.1 and B.2 (continued)

and without challenging plant systems. With any RCS cold leg temperatures at or below the OPPS enable temperature specified in the PTLR, overpressure protection is provided by the LTOP function. 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 both pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

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

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

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III, with the 1968 Winter Addendum.
2. USAR, Section 14.
3. WCAP-7769, Rev. 1, June 1972.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

ACTIONS (continued)

G.1 and G.2

If the Required Actions of Condition F are not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12 and LCO 3.4.13.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed. The basis for the Frequency of 92 days is the ASME Code for Operation and Maintenance of Nuclear Power Plants~~, Section XI~~.

This SR is modified by two Notes. Note 1 modifies this SR by stating that it is not required to be performed with the block valve closed in accordance with the Required Action of Condition B or E. Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable.

Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1 (continued)

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. A differential pressure of at least 150 psid is sufficient to ensure the valves are seated.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

REFERENCES

1. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
2. NUREG-0677, May 1980.
3. Letter from Robert A. Clark, NRC, to L. O. Mayer, NSP, Subject: "Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves," dated April 20, 1981.
4. American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants~~, Boiler and Pressure Vessel Code, Section XI~~.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by ~~Section XI of~~ the ASME Code. This type of testing may be accomplished by measuring the pump developed head at a single 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 within the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program~~, which encompasses Section XI~~ of the ASME Code. ~~Section XI of~~ ~~T~~The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This test is met when control board indications and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing, the appropriate pump breakers have opened and closed, and all automatic valves have been placed in the proper position required to establish a safety injection flow path to the reactor coolant system.

This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.2

Operating each containment fan coil unit on low motor speed for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly.

Motor current is measured and compared to the nominal current expected for the test condition. 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 coil units and controls, the two train redundancy available, and the low probability of significant degradation of the containment cooling train occurring between Surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.5.3

Verifying that cooling water flow rate to each containment fan coil unit is ≥ 900 gpm provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 4).

Terminal temperatures of each fan coil unit are also observed. This test includes verifying operation of all essential features including low motor speed, cooling water valves and normal ventilation system dampers. The 24 month Frequency is based on; the need to perform these Surveillances under the conditions that apply during a plant outage; the known reliability of the Cooling Water System; the two train redundancy available; and, the low probability of a significant degradation of flow occurring between Surveillances.

SR 3.6.5.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded. Flow and differential pressure are normal tests of centrifugal pump performance required by ~~Section XI of~~ the ASME Code (Ref. 6). Since the

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.7

This SR requires verification that each containment cooling train actuates upon receipt of an actual or simulated safety injection signal. The 24 month Frequency is based on engineering judgment. See SR 3.6.5.5 and SR 3.6.5.6, above, for further discussion of the basis for the 24 month Frequency.

SR 3.6.5.8

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

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criteria 37, 38, 41, 42, 49, 52, and 58 through 61 issued for comment July 10, 1967, as referenced in USAR Section 1.2.
2. USAR Section 6.4.
3. USAR, Section 14.5.
4. USAR, Section 6.3.
5. USAR, Section 5.2.
6. American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants.

BASES

ACTIONS

B.1 and B.2 (continued)

12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code~~, Section XI~~ (Ref. 4), requires that safety and relief valve tests be performed in accordance with ~~ANSI/ASME OM-1-1987 (~~Reference~~.~~ 5~~)~~. According to Reference 5, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-1 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to within a nominal ± 1% of their setpoint during the Surveillance. The lift settings, according to Table 3.7.1-1, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

BASES

SURVEILLANCE REQUIREMENTS

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. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. USAR, Section 11.4.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. USAR, Section 14.4.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants~~, Boiler and Pressure Vessel Code, Section XI~~.
5. ASME OM Code, Appendix I, Inservice Testing of Pressure Relief Devices in Light-Water Reactor Power Plants~~ANSI/ASME OM-1-1987~~.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.2

This SR verifies each MSIV can close on an actual or simulated main steam isolation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage.

The Frequency of MSIV testing is every 24 months. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 11.7.

2. USAR, Section 14.5.

3. License Amendment 133/125, issued November 18, 1997, "Voltage-based Steam Generator Tube Repair Criteria."

4. 10 CFR 100.11.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.3.2

This SR verifies that each MFRV and MFRV bypass valve can close on an actual or simulated FWI signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage.

The Frequency for this SR is every 24 months. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 11.9.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Differential pressure is a normal test of centrifugal pump performance required by ~~Section XI of~~ the ASME Code (Ref. 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code~~, Section XI~~ (Ref. 2) satisfies this requirement. The Inservice Testing Program specifies the Frequency for testing each pump. This test is considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. This deferral is based on the inservice testing requirements not met; all other requirements for OPERABILITY must be satisfied.

BASES (continued)

REFERENCES

1. USAR, Section 11.9.

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

3. USAR, Section 14.4.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.6 (continued)

Operating experience has shown that these components usually pass the Surveillance when performed. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 10.4.
2. USAR, Section 6.
3. ASME ~~Boiler and Pressure Vessel~~ Code for Operation and Maintenance of Nuclear Power Plants~~, Section XI~~.

ENCLOSURE, ATTACHMENT 3

Technical Specification Pages (Retyped)

1.4-3
5.0-11
5.0-12

3 pages follow

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the MODE or other specified condition or the LCO is considered not met (in accordance with SR 3.0.1) and LCO 3.0.4 becomes applicable.

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

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

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

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the USAR, Section 4.1.4, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT or PT) of exposed surfaces of the removed flywheels may be conducted at approximately 10 year intervals coinciding with the Inservice Inspection schedule as required by ASME Section XI.

5.5.7 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:

5.5 Programs and Manuals

5.5.7 Inservice Testing Program (continued)

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
Semiquarterly	At least once per 46 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and other normal and accelerated Frequencies specified as less than 2 years in the Inservice Testing (IST) Program for performing inservice testing activities. As an exception to SR 3.0.2, IST Program testing with the Frequency specified as 2 years (24 months) is met if the test is performed within 30 months (1.25 times the interval specified in the Frequency);

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.