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United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS TO EXTEND THE INSPECTION INTERVAL FOR REACTOR COOLANT PUMP FLYWHEELS USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS
SALEM GENERATING STATION - UNIT 1 AND UNIT 2
DOCKET NO. 50-272 AND 50-311
FACILITY OPERATING LICENSE NO. DPR-70 AND DPR-75

In accordance with the provisions of 10 CFR 50.90, PSEG Nuclear, LLC (PSEG) hereby transmits a request for amendment of the Technical Specifications (TS) for Salem Generating Station Unit 1 and Unit 2.

The proposed amendment will extend the reactor coolant pump (RCP) motor flywheel examination frequency from the currently approved 10-year inspection interval, to an interval not to exceed 20 years. The changes are consistent with Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-421, "Revision to RCP Flywheel Inspection Program (WCAP-15666)." The availability of this TS improvement was announced in the *Federal Register* on October 22, 2003 as part of the consolidated line item improvement process (CLIIP).

Attachment 1 provides a description of the proposed change, the requested confirmation of applicability and plant-specific verifications. Attachment 2 provides the existing TS pages marked up to show the proposed changes.

PSEG does not have specific schedule needs for this proposed change and requests approval of the proposed license amendment in accordance with the normal NRC review schedule for this type of request. PSEG requests implementation within 60 days of receipt of the approved amendment.

Should you have any questions regarding this request, please contact Mr. Michael Mosier at (856) 339-5434.

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

Sincerely,

Michael H. Brothers
Vice President – Operations

Attachments (2)

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SALEM GENERATING STATION – UNIT 1 AND UNIT 2
DOCKET NO. 50-272 AND 50-311
CHANGE TO TECHNICAL SPECIFICATIONS
EXTENSION OF THE INSPECTION INTERVAL FOR REACTOR COOLANT PUMP FLYWHEELS

DESCRIPTION AND ASSESSMENT

1.0 INTRODUCTION

The proposed amendment will extend the reactor coolant pump (RCP) motor flywheel examination frequency from the currently approved 10-year inspection interval, to an interval not to exceed 20 years. The changes are consistent with Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-421, "Revision to RCP Flywheel Inspection Program (WCAP-15666)."[1] The availability of this TS improvement was announced in the *Federal Register* on October 22, 2003 as part of the consolidated line item improvement process (CLIIP).

2.0 DESCRIPTION

Consistent with the NRC-approved TSTF-421, the proposed TS change includes the following revision to the Reactor Coolant Pump Flywheel Inspection TS Surveillance Requirement (SR) 4.4.10.1.1 (Unit 1) and SR 4.4.11.1 (Unit 2):

The examination interval for the RCP flywheels is changed from approximately 10-year intervals coinciding with the Inservice Inspection schedule as required by ASME Section XI to 20-year intervals.

3.0 BACKGROUND

The background for this application is adequately addressed by the NRC Notice of Availability published on October 22, 2003 (68 FR 60422), NRC Notice for

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1 Salem Unit 1 and Unit 2 have not adopted Standard Technical Specifications (STS); therefore, the requirements of STS 5.5.7, "Reactor Coolant Pump Flywheel Inspection Program" referenced in TSTF 421, are actually addressed by Salem Unit 1 SR 4.4.10.1.1 and Salem Unit 2 SR 4.4.11.1. NRC Regulatory Issue Summary 2000-06, "CLIIP for Adopting STS Changes for Power Reactors", permits adoption of CLIIP changes for Licensees that have not converted to STS, but have determined that the TSTF is applicable to their facility.

Comment published on June 24, 2003 (68 FR 37590), TSTF-421, WCAP-15666, "Extension of Reactor Coolant Pump Motor Flywheel Examination," and the related NRC safety evaluation (SE) dated May 5, 2003.

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

The applicable regulatory requirements and guidance associated with this application are adequately addressed by the NRC Notice of Availability published on October 22, 2003 (68 FR 60422), NRC Notice for Comment published on June 24, 2003 (68 FR 37590), TSTF-421, WCAP-15666, and the related NRC SE.

5.0 TECHNICAL ANALYSIS

PSEG has reviewed the model SE published on June 24, 2003 (68 FR 37590), and verified its applicability as part of the CLIIP. This verification included a review of the NRC staff's model SE, as well as the information provided to support TSTF-421 (including WCAP-15666 and the related SE dated May 5, 2003). The change in risk for extending the inservice inspection interval (to 20 years) is acceptable when compared to Regulatory Guide 1.174 acceptance guidelines.

PSEG has concluded that the justifications presented in the TSTF proposal and the model SE prepared by the NRC staff are applicable to Salem Unit 1 and Unit 2 and justify this amendment for the incorporation of the changes to the Salem Unit 1 and Unit 2 TS.

6.0 REGULATORY ANALYSIS

A description of this proposed change and its relationship to applicable regulatory requirements and guidance was provided in the NRC notices related to the CLIIP, TSTF-421, topical report WCAP-15666, and the associated SE.

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

PSEG has reviewed the proposed no significant hazards consideration determination published on June 24, 2003 (68 FR 37590) as part of the CLIIP. PSEG has concluded that the proposed determination presented in the notice is applicable to Salem Unit 1 and Unit 2 and it is presented below to satisfy the requirements of 10 CFR 50.91(a).

In accordance with the criteria set forth in 10 CFR 50.92, PSEG has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change to the RCP flywheel examination frequency does not change the response of the plant to any accidents. The RCP will remain highly reliable and the proposed changes will not result in a significant increase in the risk of plant operation. Given the extremely low failure probabilities for the RCP motor flywheel during normal/accident conditions and the extremely low probability of a LOCA/LOOP, and even assuming a conditional core damage probability (CCDP) of 1.0 (complete failure of safety systems), the Core Damage Frequency (CDF) and change in risk would still not exceed the NRCs acceptance guidelines contained in RG-1.174 (less than 1.0E- 6 per year). Moreover, considering the uncertainties involved in this evaluation, the risk associated with the postulated failure of an RCP motor flywheel is significantly low. Even if all four RCP motor flywheels are considered in the bounding plant configuration case, the risk is still acceptably low. Since the evaluation results for CDF and the conservative assumption that failure of the RCP motor flywheel is assumed to result directly in core damage and also a large early release (CDF = LERF), calculations were not performed for the large early release frequency (LERF). The CDF and LERF results are below the NRC's LERF acceptance guidelines.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility, or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of and accident previously evaluated. Further, the proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with the safety analysis assumptions and resultant consequences. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed change in flywheel inspection frequency does not involve any change in the design or operation of the RCP. The change to examination frequency does not change any existing accident scenarios, nor create any new or different accident scenarios.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed), or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter any assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Response: No

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by this change. The proposed changes will not result in plant operation in a configuration outside of the design basis. The calculated impact on risk is insignificant and meets the acceptance criteria contained in Regulatory Guide 1.174. There are no significant mechanisms for inservice degradation of the RCP flywheel.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

8.0 ENVIRONMENTAL EVALUATION

PSEG has reviewed the environmental evaluation included in the model SE published on June 24, 2003 (68 FR 37590) as part of the CLIIP. PSEG has concluded that the staff's findings presented in that evaluation are applicable to Salem Unit 1 and Unit 2 and the evaluation is hereby incorporated by reference for this application.

9.0 PRECEDENT

This application is being made in accordance with the CLIIP. PSEG is not proposing any substantive variations or deviations from the TS changes described in TSTF-421 or the NRC staff's model SE published on June 24, 2003 (68 FR 37590). Note that Salem Unit 1 and Unit 2 have not converted to STS; therefore there are some administrative differences in TS numbering (See footnote 1).

10.0 REFERENCES

1. Federal Register Notice: Notice of Availability of Model Application Concerning Technical Specification Improvement Regarding Extension of Reactor Coolant Pump Motor Flywheel Examination for Westinghouse Plants Using the Consolidated Line Item Improvement Process, published October 22, 2003, (68 FR 60422).
2. Federal Register Notice: Notice of Opportunity to Comment on Model Safety Evaluation on Technical Specification Improvement Regarding Extension of Reactor Coolant Pump Motor Flywheel Examination for Westinghouse Plants Using the Consolidated Line Item Improvement Process, published June 24, 2003 (68 FR 37590).
3. Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-421, "Revision to RCP Flywheel Inspection Program (WCAP-15666)," Revision 0, November 2001.
4. WCAP-15666, "Extension of Reactor Coolant Pump Motor Flywheel Examination," July 2001. [including WCAP-15666-A, Revision 1 dated October 2003]
5. NRC letter dated May 5, 2003, from H. Berkow to R. Bryan (WOG) transmitting Safety Evaluation of WCAP-15666.

ATTACHMENT 2 LCR S04-03
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TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Salem Unit 1 and Unit 2, Facility Operating License DPR-70 and DPR-75, are affected by this change request:

Technical Specification	Page
4.4.10.1.1 (Unit 1)	3/4 4-33
4.4.11.1 (Unit 2)	3/4 4-33

SALEM UNIT 1 AND UNIT 2 MARKED-UP TECHNICAL SPECIFICATION PAGES

Insert 1 *(Addition to existing SR 4.4.10.1.1 (Unit 1) and SR 4.4.11.1 (Unit 2))*

The inspection frequency will ensure that each reactor coolant pump flywheel is inspected at 20-year intervals.

In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

Insert 1

4.4.10.1.2 Augmented Inservice Inspection Program for Steam Generator Channel Heads - The steam generator channel heads shall be ultrasonically inspected during each of the first three refueling outages using the same ultrasonic inspection procedures and equipment used to generate the baseline data. These inservice ultrasonic inspections shall verify that the cracks observed in the stainless steel cladding prior to operation have not propagated into the base material. The stainless steel clad surfaces of the steam generator channel heads shall also be visually inspected during the above outages. This may be accomplished by direct visual examination or by remote means such as television camera. If the visual examination, either direct or remote, reveals detectable cladding indications, a record shall be made by means of a video tape recording or photographs for comparison purposes.

REACTOR COOLANT SYSTEM

3.4.11 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.11.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.11.1.

APPLICABILITY: ALL MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1 In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

Insert 1 →

4.4.11.2 Augmented Inservice Inspection Program for Steam Generator Channel Heads - The No. 21 Steam Generator channel head shall be ultrasonically inspected in a selected area during each of the first three refueling outages using the same ultrasonic inspection procedures and equipment used to generate the baseline data. These inservice ultrasonic inspections shall verify that the cracks observed in the stainless steel cladding prior to operation have not propagated into the base material.