

AUG 02 2005



LR-N05-0386  
LCR S04-03

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

**SUPPLEMENT TO REQUEST FOR CHANGE TO TECHNICAL SPECIFICATION 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 NOS. 50-272 AND 50-311  
FACILITY OPERATING LICENSE NOS. DPR-70 AND DPR-75**

Reference: LR-N04-0345, "Request for Change to Technical Specification to Extend the Inspection Interval for Reactor Coolant Pump Flywheels, Using the Consolidated Line Item Improvement Process," dated September 27, 2004.

By letter dated September 27, 2004, PSEG Nuclear LLC (PSEG) submitted a proposed Technical Specifications (TS) amendment for the Salem Generating Station, Units 1 and 2. The proposed amendment would extend the inspection interval for Reactor Coolant Pumps flywheels from the currently approved 10 year inspection interval, to an interval not to exceed 20 years. While developing the implementation plan for the proposed amendment, a potential conflict was identified. The existing TS surveillances (Unit 1 SR 4.4.10.1.1 and Unit 2 SR 4.4.11.1) reference Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975, which contains the 10 year inspection interval. The proposed change added a sentence after the Regulatory Guide reference specifying an inspection interval of 20 years, which was meant to supercede the referenced 10 year interval. However, one could interpret that two different inspection intervals would then be specified within the same surveillance requirement. PSEG is revising the previous proposed change to clarify the intent and adopt the TSTF-421 language. TSTF-421 specifically replaces the Regulatory Position C.4.b(1) and C.4.b(2) inspection intervals with the 20 year inspection interval.

Attachment 1 provides the revised marked-up pages. The changes adopt the TSTF-421 language and are thus consistent with the CLIIP. The previous no significant hazards consideration remains valid and it has been determined that no additional information is necessary to support this change.

A047

Should you have any questions regarding this request, please contact Mr. Wayne P. Grau at (856) 339-1172.

Sincerely



Thomas P. Joyce  
Site Vice President  
Salem Generating Station

Attachment 1

C Mr. S. Collins, Administrator - Region I  
U. S. Nuclear Regulatory Commission  
475 Allendale Road  
King of Prussia, PA 19406

U. S. Nuclear Regulatory Commission  
ATTN: Mr. S. Bailey, Licensing Project Manager - Salem  
Mail Stop 08B1  
Washington, DC 20555

USNRC Resident Inspector Office – (Salem X24)

Mr. K. Tosch, Manager IV  
Bureau of Nuclear Engineering  
P. O. Box 415  
Trenton, NJ 08625

**PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UPS)**

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

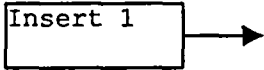
In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

|| REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

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.