September 28, 2006

Mr. William Levis
Senior Vice President & Chief Nuclear Officer
PSEG Nuclear LLC - N09
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NO. 2, ISSUANCE OF AMENDMENT RE: CHANGES TO TECHNICAL SPECIFICATIONS FOR STEAM GENERATOR TUBE INSPECTIONS (TAC NO. MC8429)

Dear Mr. Levis:

The Commission has issued the enclosed Amendment No. 256 to Facility Operating License No. DPR-75 for the Salem Nuclear Generating Station, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application dated September 21, 2005, as supplemented by letters dated June 28, 2006, and August 4, 2006.

The amendment revises the extent of steam generator tube inspections in the hot-leg side of the tubesheet.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Stewart N. Bailey, Senior Project Manager
Plant Licensing Branch 1-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-311

Enclosures:
1. Amendment No. 256 to License No. DPR-75
2. Safety Evaluation

cc w/encls: See next page

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PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 256
License No. DPR-75

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by PSEG Nuclear LLC, acting on behalf of itself and Exelon Generation Company, LLC, dated September 21, 2005, as supplemented by letters dated June 28, 2006, and August 4, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-75 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 256, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Brooke D. Poole, Acting Chief
Plant Licensing Branch 1-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and
Technical Specifications

Date of Issuance: September 28, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 256

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Replace the following page of Facility Operating License No. DPR-75 with the attached revised page as indicated. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
4	4

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3/4 4-10	3/4 4-10
3/4 4-12	3/4 4-12
3/4 4-13	3/4 4-13
3/4 4-13a	3/4 4-13a
B 3/4 4-3a	B 3/4 4-3a
- -	B 3/4 4-3b
- -	B 3/4 4-3c

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 256 TO FACILITY OPERATING

LICENSE NO. DPR-75

PSEG NUCLEAR LLC

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

DOCKET NO. 50-311

1.0 INTRODUCTION

By letter dated September 21, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML052720520), PSEG Nuclear LLC (PSEG or the licensee) requested a license amendment for the Salem Nuclear Generating Station, Unit No. 2 (Salem). The amendment would revise the Technical Specifications (TSs) to change the scope of the steam generator (SG) tube inspections required in the SG tubesheet region by applying a methodology called W* (W-star).

The W* methodology was developed for plants with SG tubes that were expanded into the tubesheet region using the Westinghouse explosive tube expansion (WEXTEX) process. The proposed amendment would revise TS Surveillance Requirement (SR) 4.4.6.4.a.8 to exclude from inspection the bottom portion of the tubes within the tubesheet region. The SR would only require inspection of a length of tube called the W* distance, which is in the upper portion of the hot-leg tubesheet region. Currently, the TSs require, in part, an inspection of the entire portion of the SG tube within the hot-leg tubesheet region.

The proposed change will also (1) revise SR 4.4.6.4.a.6 on SG tube repair criteria, (2) add SR 4.4.6.2.d to require 100 percent inspection of the tube in the tubesheet for the hot-leg W* distance, (3) add SRs 4.4.6.5.b.4 and 4.4.6.5.d to include new reporting and Nuclear Regulatory Commission (NRC or Commission) notification requirements associated with implementing the W* methodology, (4) add definitions on SRs 4.4.6.4.a.10, 4.4.5.4.a.11, and 4.4.5.4.a.12 related to the W* methodology, and (5) revise TS Bases 3/4.4.6 to add information about the W* methodology.

The original application was supplemented by letters dated June 28, 2006 (ADAMS Accession No. ML061870376) and August 4, 2006 (ADAMS Accession No. ML062280230). These supplements provided additional information that did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 7, 2006 (71 FR 2594).

2.0 REGULATORY EVALUATION

SG tubes are an integral part of the reactor coolant pressure boundary and serve to isolate radiological fission products in the primary coolant from the secondary coolant and the environment. Because of the importance of SG tube integrity, the NRC requires the performance of periodic inservice inspections of SG tubes. These inspections detect, in part, degradation in the tubes resulting from interaction with the SG operating environment. Inservice inspections may also provide a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken. Plant TSs provide acceptance criteria for SG tube inspections. Tubes with degradation that exceeds the tube repair limits specified in a plant's TSs are removed from service by plugging or are repaired by sleeving (if this repair technique has been approved by the NRC for use at the plant).

In reviewing requests of this nature, the NRC staff verifies that a methodology exists that maintains the structural and leakage integrity of the tubes consistent with the plant design and licensing bases. Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the SG tubes. Leakage integrity refers to limiting primary-to-secondary leakage during normal operation, plant transients, and postulated accidents. The NRC staff verifies that the applicable requirements in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, "General Design Criteria" (GDC), including GDC-14 and GDC-32, are satisfied. The staff's evaluation is based, in part, on ensuring that the structural margins inherent in Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR [pressurized-water reactor] Steam Generator Tubes," are maintained. The staff also verifies that a conservative methodology is used to determine the amount of primary-to-secondary leakage that may occur during design-basis accidents, such that the radiological dose consequences will meet the applicable criteria. For Salem, the radiological dose criteria are specified in 10 CFR 50.67, "Accident Source Term."

The NRC has approved amendments that are similar to the one proposed by PSEG. For example, the NRC approved similar W* criteria for one cycle of operation at Beaver Valley Power Station, Unit No. 1 (ADAMS Accession No. ML042730591), and it approved similar W* criteria for permanent use at Sequoyah Nuclear Plant, Unit No. 2 (ML051160012), and Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 (ML052970219).

3.0 TECHNICAL EVALUATION

3.1 Background

Salem is a 4-loop, Westinghouse-designed plant with Model 51 SGs. Each SG contains approximately 3400 tubes. The SG tubes are mill-annealed Alloy 600 with an outside diameter of 0.875 inches and a wall thickness of 0.050 inches. Each tube is roll-expanded for approximately 2.75 inches into the bottom of the tubesheet, then secured into the remaining portion of the tubesheet by an explosive expansion process referred to as WEXTEX. The tubesheet is approximately 21 inches thick and each tube is expanded for essentially the full thickness of the tubesheet. The WEXTEX process forms an interference fit between the tube and tubesheet. The transition from the expanded portion of the tube to the unexpanded portion of the tube is referred to as the WEXTEX transition or the expansion transition. Each tube is also welded to the primary side of the tubesheet near the tube end. This weld provides a leak-tight boundary and also provides resistance to tube pullout. Each SG contains seven tube support plates to provide lateral support to the tubes. The tube supports are carbon steel plates with drilled holes through which the tubes are inserted.

The existing TSs for Salem do not take into account the reinforcing effect of the tubesheet on the external surface of the expanded tube. The tubesheet constrains the tube and complements tube integrity in the tubesheet region by essentially precluding tube deformation beyond the expanded outside diameter of the tube. The resistance to both tube rupture and tube collapse is significantly enhanced by the tubesheet reinforcement. In addition, the proximity of the tubesheet to the expanded tube significantly reduces the leakage from any through-wall defect. Based on these considerations, power reactor licensees have proposed, and the NRC has approved, alternate repair criteria for SG tube defects located in the lower portion of the tubesheet, when these defects are a specific distance below the expansion transition or the top of tube sheet (TTS), whichever is lower.

The W* methodology defines a distance, referred to as the W* distance, such that any type of tube degradation below this distance is considered acceptable (i.e., even if inspections below this region identified degradation, the regulatory requirements pertaining to tube structural and leakage integrity would be met provided there were no flaws within the W* distance). The W* distance is determined by calculating the amount of undegraded tubing, termed the W* length, needed to address tube pullout and leakage concerns. This W* length is measured from the bottom of the WEXTEX transition (BWT). In addition to the W* length, non-destructive examination (NDE) uncertainties must be accounted for when determining the W* distance. These uncertainties include, but are not limited to, the uncertainties in determining the location of the BWT and the total inspection distance below this point (i.e., W* length). These uncertainties are addressed in the W* methodology. The W* distance is defined as the larger of the following two distances as measured from the TTS: (a) 8 inches below the TTS, or (b) 7 inches below the BWT, plus the uncertainty associated with determining the distance below the BWT (approximately 0.12 inches).

The generic W* analysis presented in WCAP-14797, Revision 2, uses bounding, non-plant-specific values for secondary system pressure and primary temperature to determine the W* length for two regions of the tube bundle. This analysis considers the forces acting to pull the tube out of the tubesheet (i.e., from the internal pressure in the tube) and the forces acting to keep the tube in place. These latter forces are a result of friction forces arising from (1) the residual preload from the WEXTEX expansion process, (2) the differential thermal expansion between the tube and the tubesheet, and (3) the internal pressure in the tube within the tubesheet. The generic W* distances were determined from lower bound tube-pull forces for WEXTEX expansions (based on a smooth tubesheet hole) in order to maximize the W* distance and bound the variability in WEXTEX expansions. In addition, the effects of tubesheet bow due to pressure and thermal differentials across the tubesheet were considered. Tubesheet bow causes dilation of the tubesheet holes from the secondary face to approximately the midpoint of the tubesheet, which reduces the ability of the tubes to resist pullout. The amount of tubesheet bow varies across the tube bundle with tubes in the periphery (referred to as Zone A tubes) experiencing less bow than tubes in the interior of the SG tube bundle (referred to as Zone B tubes). The analysis indicates that the W* length is 5.2 inches for Zone A and 7.0 inches for Zone B. WCAP-14797 also considered the uncertainties associated with NDE.

The NRC staff notes that while the most limiting region of the tube bundle is Zone B, if tubes in this region began to pull out of the tubesheet, they would be constrained by contact with neighboring tubes. As a result, the likelihood that a tube would pull out of the tubesheet is small. This effect was not considered in the development of the W* distance and adds conservatism to the evaluation.

As an additional conservatism in WCAP-14797, Revision 2, 800 pounds per square inch secondary-side pressure was assumed in the crevice when calculating SG tube contact pressures, but no secondary-side crevice pressure was assumed when calculating contact pressures for the WEXTEX leak rate test specimens.

3.2 Licensee's Proposal

The licensee's basis for only inspecting from the TTS to the W* distance is documented in the September 21, 2005, license amendment request, in WCAP-14797, Revision 2, and in the licensee's letters dated June 28 and August 4, 2006. The operating conditions assumed in the generic WCAP-14797 analysis bound the operating conditions at Salem such that the W* distance calculated using plant-specific conditions would be less than the generic W* distance identified in WCAP-14797. The generic W* distance is conservative for Salem for the following reasons:

- The generic analysis assumes a hot-leg temperature of 590 degrees Fahrenheit, whereas the limiting hot-leg operating temperature at Salem is approximately 602 degrees Fahrenheit. Therefore, the generic analysis provides less thermal tightening of the WEXTEX joint than would be present in the Salem SGs.

- The secondary-side pressure at Salem is lower than the secondary-side pressure assumed in the generic analysis, resulting in a higher differential pressure. Since a higher differential pressure corresponds to more pressure tightening of the WEXTEX joint, the actual plant conditions would produce greater pressure tightening than assumed in the generic analysis.

The licensee's proposal is also more conservative than the methods discussed in WCAP-14797 for the following reasons:

- The licensee will plug all tubes with service-induced degradation in the W* distance. Consequently, axial cracks will not remain in service within the W* distance (i.e., the flexible W* length discussed in WCAP-14797 will not be applied).

- The licensee will use a bounding leakage methodology, based on tube-to-tubesheet contact pressures, that is more conservative than the DENTFLO Code leakage model presented in WCAP-14797.

- The licensee will conservatively apply the greater W* length, calculated for Zone B tubes, to all tubes in the SG.

Since the operating conditions (e.g., pressure and temperature) can change at the plant, the licensee stated that additional controls will be established to ensure that the primary temperature and SG secondary-side pressure remain within the bounds of (or remain conservative with respect to) the W* tube integrity evaluation. If these parameters do not remain within the specified range, an evaluation will be required to assess the impact on the W* tube integrity evaluation.

3.3 Tube Structural Integrity

The licensee's proposal to use W* has the potential to allow SG tubes with defects to remain in service (in particular, in the uninspected portion of the tube); therefore, the licensee must demonstrate that the tubes returned to service using the W* methodology will maintain adequate structural integrity. Tube rupture and pullout are the two potential modes of structural failure considered for tubes returned to service under the W* methodology.

In order for a tube to rupture due to tube flaws located within the tube sheet, a flaw would need to propagate above the TTS. If the flaws remain entirely within the tubesheet, the reinforcement provided by the tubesheet will prevent tube rupture. The licensee's proposal requires an examination of the W* distance and plugging of any service-related degradation therein. Therefore, any known flaws remaining in service following the examinations will be located a minimum of 8 inches below the TTS. Industry operating experience shows that the growth rate of flaws within the tubesheet over one cycle of operation are well below the rate necessary to propagate a flaw of the 8 inches necessary to reach the TTS. Therefore, it is unlikely that any of these flaws will grow in an axial direction and extend outside the tubesheet. Thus, tube burst is precluded for flaws left in service using the W* methodology due to the reinforcement provided by the surrounding tubesheet.

In the event that flaws are located within the W* distance and are not detected during the inspection, or if new flaws initiate in the W* distance during the operating cycle following the inspection, there is a potential that these flaws could grow in the axial direction and extend outside the tubesheet. Therefore, the NRC staff considered the conditions that would be necessary to structurally fail a tube with this type of flaw. SG tube rupture is primarily a function of flaw geometry (i.e., length), the differential pressure across the tube wall, and the flaw location. In order for a tube to burst due to axial, through-wall flaws, the flaws must exceed a certain length (typically on the order of one-half inch or longer) and have no external restraint (i.e., the flaw must occur in the free span). Partially through-wall flaws would require additional length in order to become susceptible to spontaneous rupture, based on empirical models for tube burst. Thus, for tubes with partially through-wall flaws (i.e., tubes with undetected flaws slightly below the TTS), the flaws would have to extend a significant distance above the tubesheet to degrade the margins of structural integrity for the tube. In addition, restriction of a flaw on one end by the tubesheet would further elevate the burst pressure of a flawed tube. Flaw growth rates necessary to reach a critical flaw size are unlikely to occur. Therefore, flaws in the W* distance under either of the two scenarios described above should maintain adequate margins against tube burst.

The other postulated mode of structural failure for tubes remaining in service according to the W* methodology is pullout of the tube from the tubesheet due to axial loading on the tube. Differential pressures from the primary side to the secondary side of the SG impart axial loads into each tube. These loads are reacted at the tube-to-tubesheet interface. Axial tube loading during normal operating conditions can be significant. The peak postulated loading, however, occurs during events involving a depressurization of the SG secondary side (e.g., main steamline break (MSLB)). The presence of circumferentially-oriented degradation within a tube under axial loading decreases the load-bearing capability of the affected tube. If a tube becomes sufficiently degraded, these loads could lead to an axial separation of the tube.

Resistance to tube pullout is provided by the interference fit created during the WEXTEX process. In addition, increasing the temperature of the system and the internal pressure of the tube creates a tighter interference fit between the tube and the tubesheet to further resist tube pullout. The analysis supporting the licensee's proposed amendment addressed the limiting conditions necessary to maintain adequate structural integrity of the tube-to-tubesheet joint. Specifically, the tube must not experience excessive displacement relative to the tubesheet under bounding loading conditions, with appropriate factors of safety considered.

The W* methodology is based, in part, on an assessment that used analytical calculations and laboratory experiments to justify the acceptability of any type of tube degradation below the W* distance. This assessment included pullout tests of prototypical SG tube-to-tubesheet joints to evaluate the length of sound tubing necessary to maintain the appropriate structural margins. The test specimens were subjected to internal pressurization and axial loadings at various temperatures in order to demonstrate acceptable structural capabilities under a range of conditions. Despite using configurations with lower structural capabilities than expected of actual in-service SG tubes, the test program demonstrated that tubes remaining in service according to the W* methodology resisted pullout from the tubesheet with margins meeting or exceeding those inherent in RG 1.121. In addition, the licensee's tubes are most likely experiencing denting at the tube support plates, which would further restrain the tubes against pullout and would likely prevent the axial pressure load necessary to cause tube pullout. This effect was not considered in the development of the W* distance and adds conservatism to the evaluation.

In summary, the W* repair criteria was established, in part, to limit the potential for the growth of cracks into the freespan region above the TTS and to maintain adequate strength to resist tube pullout. The confinement of the surrounding tubesheet will prevent tube structural failure by tube burst for all flaws left in service using the proposed alternate repair criteria. Repair of all service-induced degradation within the W* distance will ensure that tube pullout from the tubesheet under the limiting conditions is precluded. On these bases, the NRC staff has concluded that tubes returned to service using the W* repair criteria will maintain adequate structural integrity.

3.4 Tube Leakage Integrity

In assessing leakage integrity of SGs under postulated accident conditions, the leakage from all sources (i.e., all types of flaws at all locations and all non-leak-tight repairs) must be assessed. The combined leakage from all sources must be less than a plant-specific limit that is determined based on radiological dose consequences. Since the W* methodology does not require inspections below the W* distance, there is a potential for flaws that could cause leaks to exist below the W* distance. As a result, the licensee has developed a methodology, as part of the W* methodology, for determining the amount of accident-induced primary-to-secondary leakage from flaws located at any depth in the tubesheet. The licensee's leakage methodology includes two models: (1) a constrained crack model for leakage from indications located between the TTS and 12 inches below the TTS, and (2) a crevice model for leakage from indications more than 12 inches below the TTS. The licensee's proposed methodology, and the NRC staff's review of this methodology, are discussed below.

3.4.1 Determination of the Number of Flaw Indications

The leakage methodology requires a determination of the total number of indications within the tube increments identified above (e.g., within the W* distance, from 8 to 12 inches below TTS, and more than 12 inches below TTS), and the leak rate from these indications. The total number of indications within the W* distance is determined by inspection. The licensee has proposed to plug all indications in the W* distance.

To estimate the number of indications between 8 and 12 inches below the TTS, the licensee will calculate a linear fit to the cumulative inspection data (of indications detected below the TTS) from all SGs and extrapolate the data, using a 95-percent probability prediction bound, to the tube increment from 8 to12 inches below the TTS. Although these indications would be expected to be distributed over all four SGs, the licensee will conservatively assume all projected indications in this region are in one (or each) SG. In addition, since the tube inspection actually extends deeper into the tubesheet than the W* distance (due to the characteristics of the inspection systems), some indications between 8 and 12 inches from the TTS may be detected during inspection.

To estimate the number of indications below 12 inches, the licensee will conservatively assume that all tubes left in service contain a 360-degree, through-wall circumferential flaw (i.e., a tube sever) 12 inches below the TTS.

3.4.2 Calculation of Leakage from Flaws

The licensee will use a constrained crack leakage methodology for indications from the TTS to 12 inches below the TTS. The licensee will use a 95th-percentile prediction bound on the leak rate as a function of contact pressure for the full set of constrained crack leak data. Contact pressure at the location (or postulated location) of the flaw is determined from finite element analysis, based on the tube radial position and depth of the indication within the tubesheet.

Accident-induced leakage from flaws located within the W* distance is not anticipated since all flaws in this region are repaired upon detection. This limits the size of these flaws. Nonetheless, there is a potential that significant indications could arise during the operating cycle. To this end, the licensee will perform an assessment if severe new indications of cracking are identified. This assessment will include potential leakage, which will be calculated using the constrained crack leakage model discussed above.

For the projected set of indications between 8 and 12 inches below the TTS, the licensee will assign a constrained crack leak rate of 0.0033 gallons per minute (gpm), which is based on the 95th-percentile prediction bound for a 360-degree through-wall crack at 8 inches below the TTS, the location in this span where contact pressure is the lowest. A similar leak rate model was approved previously for Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2.

To address the potential for indications more than 12 inches below the TTS, the licensee will assign a leakage value of 9×10^{-5} gpm to each tube left in service, based on a crevice leakage model of a 360-degree through-wall crack at 12 inches below TTS (the limiting location in this tube segment). The licensee calculated this leakage value using a 90th-percentile upper-bound leak rate from nominal 3-inch crevice depth leak rate specimens. Given past plant-specific and industry operating experience, the NRC staff considers the assumption that all tubes contain circumferential, through-wall flaws 12 inches below the TTS, along with application of a 90th-percentile upper-bound leak rate to these flaws, to be conservative. This approach has been approved previously by the NRC for use at Beaver Valley Power Station, Unit No. 1, Sequoyah Nuclear Plant, Unit No. 2, and Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2.

Based on its review of the licensee's proposal, the NRC staff considers the leak rates from the licensee's leakage methodology acceptable. A number of conservative assumptions were included in this methodology:

1. All indications below the W* distance within the tubesheet are assumed to be through-wall. Historical inspection data indicates only a fraction of these indications are through-wall. In addition, all indications below the W* distance are assumed to be leaking, although industry operating experience has demonstrated negligible leakage under normal operating conditions, even when cracks are located in a tube-to-tubesheet expansion transition zone.

2. The constrained crack leak rates (applied to indications from the TTS to 12 inches below the TTS) are based on tests simulating the resistance to crack leakage provided by the tubesheet constraining the crack opening. In reality, leakage from tubes with through-wall flaws located within a tubesheet is also restricted by the crevice (or, more precisely, the interference fit) between the tube and tubesheet. The licensee's leakage methodology takes no credit for the leakage restriction resulting from the tube-to-tubesheet crevice.

3. The crack indications in SG tubes are typically stress-corrosion cracks, which tend to be tighter and provide greater resistance to leakage than the fatigue cracks used in the constrained crack test samples. Therefore, the leakage testing provided conservative results.

4. The licensee conservatively assumes all tubes in service contain a 360-degree tube sever located 12 inches below the TTS and, therefore, each inservice tube contributes to the leakage total.

5. The leak rates from flaws between 8 and 12 inches below the TTS were determined based on the worst-case tube in the SG (i.e., the greatest hole dilation resulting in a lower contact pressure). All other radial positions within the SG would be expected to have lower leak rates due to higher contact pressures.

6. The number of indications detected by inspection of all four SGs is used to project the number of indications in each SG. Although these indications would be expected to be spread over all four SGs, the licensee is conservatively assuming that all postulated indications within this range are located in each of the four SGs.

The licensee's application of the W* criteria and accompanying leakage methodology will be used to determine the amount of leakage from flaws below the W* distance. This leakage will be combined with the leakage from all other sources to ensure that it is less than the plant-specific allowable limits. In addition, the licensee will be required to assess whether the results of the inspection were consistent with expectations with respect to the number of flaws and their severity. In the event that the results are not consistent, the licensee will be required to describe proposed corrective actions in accordance with the TSs. On this basis, the NRC staff has concluded that the licensee has an acceptable methodology for ensuring leakage integrity can be maintained.

3.5 Reporting Requirements

As part of the Annual Operating Report required by SR 4.4.6.5, the licensee will report the following with respect to implementation of the W* inspection methodology: the number of indications, the orientation of each indication, the severity (depth) of each indication, the tube surface on which the indication initiated (internal or external), the cumulative number of indications detected in the tubesheet as a function of elevation, the condition monitoring and operational assessment MSLB leak rate (including calculated MSLB leak rate from all other sources), and an assessment of whether the inspection results were consistent with expectations regarding the number of flaws and their severity (and if not consistent, a description of the proposed corrective action). In addition, the NRC staff will be notified prior to returning the SGs to service if the estimated MSLB leak rate described above exceeds the design and licensing basis.

3.6 Summary

The NRC staff's approval of the licensee's proposal is based on the licensee's demonstration that the applicable structural integrity and leakage integrity requirements will be met. The licensee has demonstrated that SG tubes left in service will maintain adequate structural integrity to meet the applicable regulatory requirements, and will maintain adequate leakage integrity to maintain the radiological dose consequences within the appropriate limits following a design-basis event. Finally, the licensee will evaluate and report unexpected, significant degradation of the SG tubes identified through inspections.

The NRC staff concludes the licensee's proposed methodology for assessing structural and leakage integrity for flaws in the tubesheet region is acceptable. Therefore, the NRC staff concludes that the licensee's proposal to limit the tube inspection scope in the hot-leg tubesheet is an acceptable approach.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment. By letter dated March 7, 2006 (ADAMS Accession No. ML061160122), the State official stated that it had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (71 FR 2594). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (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.

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