May 16, 2008

Mr. David A. Christian
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION, UNIT NO. 2 – ISSUANCE OF EXIGENT AMENDMENT RE: INTERIM ALTERNATE REPAIR CRITERIA FOR STEAM GENERATOR TUBE REPAIR (TAC NO. MD8504)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 258 to Renewed Facility Operating License No. DPR-37, for the Surry Power Station, Unit No. 2 (Surry 2). The amendment changes the Technical Specifications (TSs) in response to your application dated April 14, 2008, as supplemented by letter dated May 6, 2008.

This amendment allows a one-cycle revision to Surry 2 TSs. Specifically, TS 6.4.Q, “Steam Generator (SG) Program,” and TS 6.6.A.3, “Steam Generator Tube Inspection Report,” will be revised to incorporate an interim alternate repair criterion into the provisions for SG tube repair for use during the Surry 2, spring 2008 refueling outage and the subsequent operating cycle. The exigent circumstances and final no significant hazards consideration are addressed in Sections 5.0 and 6.0 of the enclosed Safety Evaluation.

The Notice of Issuance will be included in the Commission’s next biweekly *Federal Register* notice.

Sincerely,

/RA/

Siva P. Lingam, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-281

Enclosures:
1. Amendment No. 258 to DPR-37
2. Safety Evaluation

cc w/encls: See next page

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VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 258
Renewed License No. DPR-37

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

    A. The application for amendment by Virginia Electric and Power Company (the licensee) dated April 14, 2008, as supplemented by letter dated May 6, 2008, 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 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;

    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 3.B of Renewed Facility Operating License No. DPR-37 is hereby amended to read as follows:

   (B) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 258, are hereby incorporated in the renewed 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Melanie C. Wong, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes License No. DPR-37
and the Technical Specifications

Date of Issuance: May 16, 2008

ATTACHMENT

TO LICENSE AMENDMENT NO. 258

RENEWED FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Replace the following pages of the Licenses and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

| Remove Pages | Insert Pages |
| --- | --- |
| License | License |
| License No. DPR-37, page 3 | License No. DPR-37, page 3 |
| TSs | TSs |
| 6.4-12 | 6.4-12 |
| --- | 6.4-13 |
| 6.6-3 | 6.6-3 |
| --- | 6.6-3a |

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 258 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NO. 2

DOCKET NO. 50-281

## 1.0 INTRODUCTION

By letter dated April 14, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML0810800440), as supplemented by letter dated May 6, 2008 (ADAMS Accession No. ML081280322), Virginia Electric and Power Company (the licensee) submitted a request for changes to the Surry Power Station, Unit No. 2 (Surry 2), Technical Specifications (TSs). The requested changes would allow a one-cycle revision to Surry 2 TSs. Specifically, TS 6.4.Q, "Steam Generator (SG) Program," and TS 6.6.A.3, "Steam Generator Tube Inspection Report," will be revised to incorporate an interim alternate repair criterion into the provisions for SG tube repair for use during the Surry 2, spring 2008 refueling outage and the subsequent operating cycle. The supplement dated May 6, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination published in an individual notice in the *Federal Register* on April 25, 2008 (73 FR 22443), however, the licensee requested issuance of this amendment based on the exigent circumstances. As a result of exigent circumstances, a notice has been published in the *Daily Express* on May 12, and 13, 2008 referring to the individual notice in the *Federal Register* on April 25, 2008 (73 FR 22443), for public comments by May 15, 2008.

In its letter dated April 14, 2008, the licensee submitted Westinghouse Electric Company (WEC) topical reports, LTR-CDME-08-11 P-Attachment, "Interim Alternate Repair Criterion (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," dated January 31, 2008, and LTR-CDME-08-43 P-Attachment, "Response to NRC Request for Additional Information Relating to LTR-CDME-08-11 P-Attachment," dated March 18, 2008. Because the topical reports contained proprietary information, the licensee's submission included affidavits, requesting that the NRC withhold the proprietary information from the public. The NRC letter approving the withholding of the information from the public, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), paragraph 2.390(b)(5) and Section 103(b) of the Atomic Energy Act of 1954, as amended, was issued in a letter dated May 13, 2008 (ADAMS Accession No. ML081200924). This Safety Evaluation (SE) contains no proprietary information.

## 2.0 BACKGROUND

Surry 2 has three Westinghouse Model 51F SGs. There are 3342 thermally treated Alloy 600 tubes in each SG, each with an outside diameter of 0.875 inches and a nominal wall thickness of 0.050 inches. The tubes are hydraulically expanded for the full depth of the tubesheet at each end and are welded to the tubesheet at the bottom of each expansion. Until the fall of 2004, no instances of stress corrosion cracking (SCC) affecting the tubesheet region of thermally treated alloy 600 tubing had been reported, at Surry or other nuclear power plants in the United States. As a result, most plants, including Surry 2, had been using bobbin probes for inspecting the length of tubing within the tubesheet. Since bobbin probes are not capable of reliably detecting SCC in the tubesheet region, supplementary rotating coil probe inspections were used in a region extending from 3 inches above the top of the tubesheet (TTS) to 3 inches below the TTS. This zone includes the tube-expansion transition, which contains significant residual stress, and was considered a likely location for SCC to develop.

In the fall of 2004, crack-like indications were found in tubes in the tubesheet region of Catawba Nuclear Station, Unit 2 (Catawba), which has Westinghouse Model D5 SGs. Like Surry 2, the Catawba SGs employ thermally treated alloy 600 tubing that is hydraulically expanded against the tubesheet. At the time of cracking, Catawba had accumulated 14.7 effective full power years (EFPY) of service. While the service experience of the Surry 2 SGs is significantly more than the Catawba SGs, the hot-leg operating temperature at Surry 2 is significantly lower than at Catawba. The crack-like indications at Catawba were found in bulges (also called over-expansions) in the tubesheet region, in the tack expansion region, and near the tube-to-tubesheet weld. The tack expansion is an initial 0.7-inch-long expansion at each tube end and is formed prior to the hydraulic expansion over the full tubesheet depth. The purpose of the tack expansion is to facilitate performing the tube to tubesheet weld.

As a result of the Catawba findings, the licensee expanded the scope of rotating coil inspections to include overexpansions (OXPs) during the spring 2005 and fall 2006 Surry 2 refueling outages.

By letter dated April 14, 2008, the licensee submitted a license amendment request to change the TSs for Surry 2. The proposed changes would establish alternate repair criteria for portions of the SG tubes within the tubesheet, and would be applicable to Surry 2 during Refueling Outage 21 and the subsequent operating cycle.

## 3.0 REGULATORY EVALUATION

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of the TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant’s TSs. In 10 CFR 50.36(d)(5), administrative controls are stated to be, "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." This also includes the programs established by the licensee and listed in the administrative controls section of the TSs for the licensee to operate the facility in a safe manner. The Surry 2 requirements for performing SG tube inspections and repair are in TS 6.4.Q, while the Surry 2 requirements for reporting the SG tube inspections and repair are in TS 6.6.A.3.

In the improved standard technical specifications (STS) in NUREG-1431 for Westinghouse plants like Surry, TS 5.5.9 requires that an SG tube program be established and implemented to ensure that SG tube integrity is maintained. For Surry 2, SG tube integrity is maintained by meeting specified performance criteria (in TS 6.4.Q.2) for structural and leakage integrity, consistent with the plant design and licensing basis. TS 6.4.Q.1 requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected to confirm that the performance criteria are being met. TS 6.4.Q.4 also includes provisions regarding the scope, frequency, and methods of SG tube inspections. Of relevance to the subject amendment request, these provisions require that the number and portions of tubes inspected, and methods of inspection, shall be performed with the objective of detecting flaws of any type that may be present along the length of a tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The applicable tube repair criteria, specified in TS 6.4.Q.3, are that tubes found by an inservice inspection (ISI) to contain flaws with a depth equal to or exceeding 40 percent of the nominal tube-wall thickness shall be plugged.

The SG tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and, in addition, serve to isolate radiological fission products in the primary reactor coolant from the secondary coolant and the environment. For the purposes of this SE, SG tube integrity means that the tubes are capable of performing these safety functions in accordance with the plant design and licensing basis.

The General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 provide regulatory requirements in the GDC which state that the RCPB shall have “an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDCs 15 and 31), shall be of "the highest quality standards practical" (GDC 30), and shall be designed to permit "periodic inspection and testing . . . to assess . . . structural and leaktight integrity" (GDC 32). To this end, 10 CFR 50.55a specifies that RCPB components must meet Class 1 requirements in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Section 50.55a further requires, in part, that throughout the service life of a pressurized-water reactor (PWR) facility like Surry 2, ASME Code Class 1 components meet the requirements, except design and access provisions and pre-service examination requirements, in Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME Code, to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. Section XI requirements pertaining to ISI of SG tubing are augmented by additional requirements in the TSs.

As part of the plant-licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents (DBAs) such as a SG tube rupture and main steamline break (MSLB). These analyses consider primary-to-secondary leakage that may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of the 10 CFR Part 100 guidelines for offsite doses, GDC 19 criteria for control room operator doses, or some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis (e.g., a small fraction of these limits). No accident analysis for Surry 2 is being changed because of the proposed amendment and, thus, no radiological consequences of any accident analysis are being changed.

The licensee-proposed changes to TS 6.4.Q stay within the GDC requirements for the SG tubes and maintain the accident analysis and consequences that the NRC has reviewed and approved for the postulated DBAs for SG tubes. The proposed amendment is applicable to Refueling Outage 21 and the subsequent operating cycle of Surry 2. This license amendment is similar to those approved in 2008 for Wolf Creek, Vogtle, and Braidwood, but differs from previous one-cycle amendments related to SG tube inspections that were approved prior to 2008. First, the lowermost 4 inches of the tubesheet would not be excluded from the TS inspection requirements in TS 6.4.Q.4.c. The lowermost 4 inches would be subject to the same inspection requirements as the rest of the tubing. Second, any flaws found in the lowermost 4 inches of the tubesheet would not always be excluded from requirements to plug the tube. Under the proposed amendment, flaws found in the lowermost 4 inches of tubing would be subject to specific interim alternate repair criteria (IARC) in lieu of the aforementioned 40 percent depth-based criterion; the 40 percent criterion would continue to be applicable outside of the tubesheet region. Third, the proposed amendment would apply to both the hot- and cold-leg sides of the tubesheet. Fourth, the proposed amendment would include new reporting requirements to allow the NRC staff to monitor the implementation of the amendment. The proposed amendment would require the plugging of all tubes found with flaws in the upper 17 inches of the tubesheet region on both the hot- and cold-leg sides.

## 4.0 TECHNICAL EVALUATION

### 4.1 Proposed Changes to the TSs

TS 6.4.Q.3 currently states:

> "3. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged."

This criterion would be revised as follows, as noted in italic type:

> "3. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.
>
> *The following alternate tube repair criteria shall be applied as an alternative to the 40% depth-based criteria:*
>
> a. *For Unit 2 Refueling Outage 21 and the subsequent operating cycle, tubes with flaws having a circumferential component less than or equal to 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet do not require plugging. Tubes with flaws having a circumferential component greater than 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet shall be removed from service.*
>
> *Tubes with service-induced flaws located within the region from the top of the tubesheet to 17 inches below the top of the tubesheet shall be removed from*

*service. Tubes with service-induced axial cracks found in the portion of the tube below 17 inches from the top of the tubesheet do not require plugging.*

*When more than one flaw with circumferential components is found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet with the total of the circumferential components greater than 203 degrees and an axial separation distance of less than 1 inch, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.*

*When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet, and the total of these circumferential components exceeds 94 degrees, then the tube shall be removed from service. When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet and within 1 inch axial separation distance of a flaw above 1 inch from the bottom of the tubesheet, and the total of these circumferential components exceeds 94 degrees, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components."*

TS 6.6.A.3 currently states:

"3. A report shall be submitted within 180 days after $T_{avg}$ exceeds 200°F following completion of an inspection performed in accordance with the Specification 6.4.Q, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG."

TS 6.6.A.3 would be revised to add the following three additional reporting criteria:

*i. Following completion of a Unit 2 inspection performed in Refueling Outage 21 (and any inspections performed in the subsequent operating cycle), the number of indications and location, size, orientation, whether initiated on primary or secondary side for each service-induced flaw within the thickness of the tubesheet, and the total of the circumferential components and any circumferential overlap below 17 inches from the top of the tubesheet as determined in accordance with TS 6.4.Q.3.a,*

*j. Following completion of a Unit 2 inspection performed in Refueling Outage 21 (and any inspections performed in the subsequent operating cycle), the primary to secondary LEAKAGE rate observed in each steam generator (if it is not practical to assign leakage to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one steam generator) during the cycle preceding the inspection which is the subject of the report, and*

*k. Following completion of a Unit 2 inspection performed in Refueling Outage 21 (and any inspections performed in the subsequent operating cycle), the calculated accident leakage rate from the portion of the tube below 17 inches below the* top of *the tubesheet for the most limiting accident in the most limiting steam generator.*

4.2 Technical Evaluation

The tube-to-tubesheet joint consists of the tube, which is hydraulically expanded against the bore of the tubesheet; the tube-to-tubesheet weld located at the tube end; and the tubesheet. The joint was designed as a welded joint and not as a friction or expansion joint. The weld itself was designed as a pressure boundary element. It was designed to transmit the entire end-cap pressure load during normal and DBA conditions from the tube to the tubesheet with no credit taken for the friction developed between the hydraulically expanded tube and the tubesheet. In addition, the weld serves to make the joint leak tight.

The one-cycle amendments approved for other plants (such as Vogtle and Braidwood) prior to 2008, exempted the lower 4-inch portion of the tube within the 21-inch-deep tubesheet from inspection and exempted tubes with flaw indications in this region from being removed from service (i.e., plugged). These one-cycle amendments, in effect, redefined the pressure boundary at the tube-to-tubesheet joint as consisting of a friction or expansion joint with the tube hydraulically expanded against the tubesheet over the top 17 inches of the tubesheet. These amendments took no credit for the lower portion of the tube or the tube-to-tubesheet weld as contributing to the structural or leakage integrity of the joint.

The proposed amendment that is the subject of this safety evaluation (and similar amendments approved in 2008 for Wolf Creek, Vogtle, and Braidwood) differs fundamentally from the one-cycle amendments approved prior to 2008 and is a more conservative approach. The proposed amendment treats the tube-to-tubesheet joint as a welded joint in a manner consistent with the original design basis, with no credit taken for the friction developed between the hydraulically expanded tube and the tubesheet. The proposed amendment is intended to ensure that the aforementioned end-cap loads can be transmitted down the tube, through the tube-to-tubesheet weld, and into the tubesheet.

4.2.1 Proposed Change to TS 6.4.Q.3, “Provisions for SG tube repair criteria”

The 40 percent depth-based tube repair criterion in TS 6.4.Q.3 is intended to ensure, in conjunction with other elements of TS 6.4.Q, that tubes accepted for continued service (i.e., not plugged) satisfy the performance criteria for structural integrity in TS 6.4.Q.2.a and the performance criteria for accident induced leakage in TS 6.4.Q.2.b. The criterion includes an allowance for eddy current measurement error and incremental flaw growth prior to the next inspection of the tube. The proposed IARC in this amendment are alternatives to the 40 percent depth-based criterion.

4.2.1.1 Structural Integrity Considerations

The 40 percent depth-based criterion was developed to be conservative for flaws located anywhere in the SG, including free span regions. In the tubesheet, however, the tubes are constrained against radial expansion by the tubesheet and, therefore, are constrained against an axial (fish-mouth) rupture failure mode. The only potential structural failure mode within the tubesheet is a circumferential failure mode, leading to tube severance.

The proposed IARC would permit tubes with up to 100 percent through-wall flaws in the portion of the tube from 17 inches below the TTS to 1 inch above the bottom of the tubesheet to remain in service provided the circumferential component of these flaws does not exceed 203 degrees. The 203-degree criterion was determined on the basis of the remaining cross-sectional area of the tube needed to resist the limiting axial end-cap load on the tube and the pressure load on the flaw cross-section, using limit-load analysis, with safety factors consistent with those required by the performance criteria for structural integrity in the TS. Because the 203-degree criterion was determined on this basis, the NRC staff finds this approach acceptable.

For the portion of the tube from the bottom of the tubesheet to 1 inch above the bottom of the tubesheet, the proposed IARC would permit tubes with up to 100 percent through-wall flaws to remain in service provided the circumferential component of these flaws does not exceed 94 degrees. This criterion is based on the minimum tube-to-tubesheet weld cross-sectional area needed to resist the limiting axial end-cap load on the tube and the pressure load on the flaw cross-section, using limit load analysis, with safety factors consistent with those required by the performance criteria for structural integrity in the TS. A 203-degree crack in the tube wall immediately above the weld could potentially concentrate the entire end cap load to a 157-degree segment of the weld, whereas a minimum 266 degree segment (i.e., 360 minus 94 degrees) of weld is needed to resist the end-cap load with adequate safety margin. Thus, the 94-degree criterion for the tube in the lowermost 1-inch region is intended to ensure that the weld is not overstressed. Although the NRC staff did not do a detailed review of the specific limit-load methodology used to calculate the 94-degree criterion, the staff verified the conservatism of the 94-degree criterion by reviewing the ASME Code analysis and establishing that the ratio of calculated stresses to ASME Code allowable stresses in the weld conservatively lower bounds the ratio of the proposed required remaining weld arc length (i.e., 360 minus 94 degrees) to the total weld arc length of 360 degrees. The TS performance criteria for tube structural integrity are intended to ensure safety margins consistent with the ASME Code, Section III stress limits. Based on a comparison of the calculated maximum design stress to the ASME Code-allowable stress, the NRC staff concludes that the proposed 94-degree criterion ensures that the weld can react the end-cap loads with margins to failure consistent with the margins ensured by the ASME stress limits and is, therefore, acceptable.

The 203- and 94-degree criteria include an allowance for incremental flaw growth in the circumferential direction prior to the next inspection. The licensee states that no significant growth rate data exists for the specific case of circumferential cracking in the tubesheet expansion region. The licensee's growth rate estimate is based on a 95 percent upper bound value of available primary water stress corrosion crack (PWSCC) growth rate data for other tube locations. Given the lack of actual growth rate data for cracks that may potentially initiate in the lowermost 4 inches of the tube, the NRC staff attaches only a low level of confidence in the conservatism of the licensee's growth rate estimate. However, the NRC staff notes that the effect of any lack of conservatism in the licensee's estimate is mitigated somewhat by the fact that all of the SGs at Surry 2 will be inspected during Refueling Outage 22, should any crack indications be found during Refueling Outage 21, and appropriate actions would be taken based on the findings. In addition, the 203- and 94-degree criterion conservatively take no credit for the effects of friction between the tube and tubesheet in any portion of the tube-to-tubesheet joint, in reacting out a portion of the axial end cap load before it reaches the cracked cross-section. Thus, the NRC staff concludes that the 203- and 94-degree criteria are conservative, irrespective of growth rate uncertainties.

The 203- and 94-degree criteria do not include an explicit allowance for eddy current measurement error. The licensee will be utilizing an inspection technique that has been qualified for the detection of circumferential PWSCC in tube expansion transitions and in the tack expansion region just above the tube to tubesheet weld. The tack expansion is a 0.7-inch-long expansion of the tube in the tubesheet that is performed before the tube is hydraulically expanded for the entire depth of the tubesheet. A fundamental assumption behind the proposed 203- and 94-degree repair criteria is that all detected circumferential flaws in the lowermost 4 inches of the tube are fully 100% through wall, irrespective of the actual depth of the flaw. With this assumption, the license referenced an Electric Power Research Institute (EPRI) sponsored study that indicated the eddy current measurement of the crack arc length was conservative (i.e., larger than the actual crack size), and resulted in an estimate of the remaining cross sectional area that was always smaller than values obtained through direct measurement of cracks. Based on the EPRI study results and the assumption that all cracks are through-wall, any uncertainties related to measured arc length of the flaw are not expected to impair the conservatism of the 203- and 94-degree criteria.

The proposed IARC also includes criteria to account for interaction effects for multiple circumferential flaws that are in close proximity. The proposed criteria treat the multiple circumferential flaws located within 1 inch of one another as all occurring at the same axial location. The total arc length of the combined flaw is the sum of the individual flaw arc lengths with overlapping arc lengths counted only once. The licensee stated that the summation of cracks with both located more than 17 inches from the TTS and more than 1 inch from the bottom of the tube will be compared to the 203-degree criterion. The summation of cracks with one flaw located less than 1 inch from the bottom of the tubesheet and the other within 1 inch of the first (or both flaws within 1 inch of the bottom of the tubesheet) would be compared to the 94-degree criterion. Cracks located more than 1 inch apart from one another are assumed to act independently of each other. This 1-inch criterion was determined using a fracture mechanics approach to determine the axial distance from an individual crack tip at which the stress distribution reverts to a nominal stress distribution for an uncracked section. The 1-inch criterion is twice the calculated distance since twice this distance is the necessary separation between two cracks for the cracks to act independently of each other. The NRC staff reviewed the basis for the 1-inch criterion and the fracture mechanics approach to determining the criterion. Because the criterion is based on a valid fracture mechanics approach, the NRC staff finds it acceptable.

The proposed IARC would permit tubes with axial cracks in the lower most 4 inches of the tube to remain in service, irrespective of crack depth. The NRC staff finds this acceptable because axial cracks do not impair the ability of the tube or the weld to resist axial load and because the tube is fully constrained by the tubesheet against an axial failure mode.

Finally, the proposed IARC includes a requirement to plug all tubes in which flaws are detected in the upper 17-inch portion of the tube within the tubesheet. This adds to the conservatism of the 203- and 94-degree criteria since it mitigates any loss of tightness and, thus, any loss of friction between the tube and tubesheet due to flaws in the upper 17-inch region of the joint.

4.2.1.2 Accident Leakage Integrity Considerations

If a tube is assumed to contain a 100 percent through wall flaw some distance into the tubesheet, a potential leak path between the primary and secondary systems is introduced between the hydraulically expanded tubing and the tubesheet. Operational leakage integrity is assured by monitoring primary-to-secondary leakage relative to the applicable TS LCO limits in TS 3.1.C, “RCS Operational Leakage.” However, it must also be demonstrated that the proposed TS changes do not create the potential for leakage during DBAs to exceed the accident leakage performance criteria in TS 6.4.Q.2.b, including the leakage values assumed in the plant licensing basis accident analyses. The licensee states that this is ensured for Surry 2 by limiting primary-to-secondary leakage to 0.33 gpm in the faulted SG during an MSLB accident.

The leakage path between the tube and tubesheet has been modeled by the licensee’s contractor, Westinghouse, as a crevice consisting of a porous media. Using Darcy’s model for flow through a porous media, leak rate is proportional to differential pressure and inversely proportional to flow resistance. Flow resistance is a direct function of viscosity, loss coefficient, and crevice length. Westinghouse performed leak tests of tube-to-tubesheet joint mockups to establish loss coefficient as a function of contact pressure. Westinghouse states that the flow resistance varies as a log normal linear function of joint contact pressure, but due to the large scatter of the flow resistance test data, has been assumed to be constant with joint contact pressure at a value which conservatively lower bounds the data.

The above model relies, to some extent, on an assumed constant value of loss coefficient, based on a lower bound of the data. The staff was not able to conclude whether the assumed value of loss coefficient in the above approach was conservative; however, the NRC staff performed some evaluations regarding the potential for the normal operating leak rate to increase under steam-line break conditions. Making the conservative assumption that loss coefficient and viscosity remain constant (under both normal operating and steam-line break conditions), the ratio of steam-line break leak rate to normal operating leak rate is equal to the ratio of steam-line break differential pressure to normal operating differential pressure, multiplied by the ratio of effective crevice length under normal operating conditions ($L_{NOP}$) to effective crevice length under steam-line break conditions ($L_{SLB}$). Effective crevice length is the crevice length over which there is contact between the tube and tubesheet. By making the conservative assumption that loss coefficient and viscosity remain constant under normal operating and steam-line break conditions, the loss coefficient variable drops out of the equation (i.e. the value of loss coefficient assumed by the licensee no longer affects the result). Using various values of $L_{NOP}/L_{SLB}$, the NRC staff concluded that a factor of 2.5 reasonably bounded the potential increase in leakage from the lowermost 4 inches of tubing that would be realized in going from normal operating to steam-line break conditions.

The licensee stated in its April 14, 2008, license amendment request that it would apply the 2.5 factor in its condition monitoring (CM) and operational assessment (OA) upon implementation of the subject license amendment. Specifically, for the CM assessment, the licensee states that the component of leakage from the lowermost 4 inches for the most limiting SG during the prior cycle of operation will be multiplied by a factor of 2.5 and added to the total leakage from any other source and compared to allowable accident leakage limit. For the OA, the licensee stated that the difference in leakage from the allowable accident leakage limit and the accident leakage from other sources will be divided by 2.5 and compared to the observed (operational) leakage and that an administrative limit (for operational leakage) will be established not to exceed the calculated value. Since this properly addresses the factor of 2.5 that bounds the potential increase in leakage in the lowermost 4 inches of tubing, the NRC staff finds this acceptable.

In its letter dated April 14, 2008, the licensee submitted a regulatory commitment that stated the 2.5 factor would be used in the completion of its CM and OA upon implementation of the IARC in this amendment. This is an IARC because it applies only to Refueling Outage 21 and the subsequent operating cycle.

The NRC staff finds that reasonable controls for the licensee's implementation and subsequent evaluation of any changes to the regulatory commitment are provided by the licensee's administrative processes, including its commitment management program. The NRC staff has determined that the commitment does not warrant the creation of regulatory requirements, which would require prior NRC approval of subsequent changes. The NRC has agreed that NEI 99-04, Revision 0, provides reasonable guidance for the control of regulatory commitments made to the NRC staff (Regulatory Issue Summary 2000-17, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff," dated September 21, 2000). These commitments will be controlled in accordance with the licensee's commitment management program in accordance with NEI 99-04. Any change to the regulatory commitments is subject to licensee management approval and subject to the procedural controls established at the plant for commitment management in accordance with NEI 99-04, which include notification of the NRC. Also, the NRC staff may choose to verify the implementation and maintenance of these commitments in a future inspection or audit.

Based on this, the NRC staff concludes that the regulatory commitment addressed above for this amendment is acceptable.

#### 4.2.2 Proposed Change to TS 6.6.A.3, “Steam Generator Tube Inspection Report”

The NRC staff has reviewed the proposed new reporting requirements and found that they are sufficient to allow the NRC staff to monitor the implementation of the proposed amendment. Based on this conclusion, the NRC staff finds that the proposed new reporting requirements are acceptable.

#### 4.2.3 Considerations Relating to Tube-to-Tubesheet Welds

The STS and the Surry 2 TSs state specifically that the tube-to-tubesheet welds are not part of the tube. Therefore, the requirements of TS 6.4.Q do not apply to these welds. However, licensees typically visually inspect the tube ends (including the welds) for evidence of leakage while the SG primary manways are open to permit eddy current inspection of the tubes.

Eddy-current inspection of the SG tubes at Catawba Unit 2 revealed indications interpreted as cracks at or near the tube-to-tubesheet weld, suggesting the potential for such cracks in similar SGs, such as those at Surry 2. An industry peer review was recently conducted for the Catawba Unit 2, 2007 cold-leg tube-end indications to establish whether the reported indications are in the tube material or the welds. A consensus was reached that the indications most likely exist within the tube material. However, some of the indications extend close enough to the tube end that the possibility that the flaws extend into the weld could not be ruled out. An NRC staff member and an expert consultant from Argonne National Laboratory also reviewed these indications and concluded that the industry's position was reasonable. The peer review group and the NRC consultant also reviewed eddy-current signals from a tube-to-tubesheet mockup, which included a circumferential notch in one of the welds, and they concluded that this notch did not produce a detectable signal.

### 4.3 Summary

Based on the above evaluation, the NRC staff finds that the proposed license amendment, which is applicable only to Refueling Outage 21 and the subsequent operating cycle of Surry 2, ensures that SG tube structural and leakage integrity will be maintained during this period. Structural safety margins consistent with the design basis and leakage integrity within assumptions employed in the licensing basis accident analyses will also be maintained. Additionally, there will be no adverse impact on the ability of the tube-to-tubesheet welds to perform their safety-related function. Based on these findings, the NRC staff further concludes that the proposed amendment meets 10 CFR 50.36 and, thus, the proposed amendment is acceptable.

The current TSs and the proposed amendment do not address inspection requirements for the tube-to-tubesheet welds. There are no safety issues with respect to hypothetical cracks in the weld if it can be demonstrated that the axial end-cap loads in the tube are reacted by frictional forces developed between the tube and tubesheet before any portion of the end-cap load is transmitted to the weld.

Other plants, such as Wolf Creek, Vogtle, and Braidwood, requested amendments that would permanently limit the inspection scope of tubes within the tubesheet region, based on a Westinghouse analysis referred to as "H*/B*". Based on the industry peer review conducted for the Catawba Unit 2, 2007 cold-leg tube-end indications, the licensee has concluded that cracking exclusively in the weld is not a potential damage mechanism. Should it not be possible for the NRC staff to approve an acceptable H*/B* amendment within a reasonable time period, it is the NRC staff's position that the industry will need to develop inspection techniques (e.g., visual, eddy-current) capable of detecting weld cracks to ensure that the welds are capable of performing their safety related function. During the peer review, the NRC staff observed a demonstration of an available visual inspection technique for inspecting the welds, but raised questions on whether this technique was sufficiently reliable.

## 5.0 EXIGENT CIRCUMSTANCES

In its supplemental letter dated May 6, 2008, the licensee requested that this amendment be processed as an exigent amendment request pursuant to 10 CFR 50.91(a)(6)(vi) to expedite the approval of IARC for Surry 2 SG tube repairs.

Based on Surry's lower operating temperature and its effective degradation years (EDY) of operation, no flaws exceeding the repair criteria were expected and no flaws were found. As a result of this review, similar inspections on the Surry 2 steam generators were planned for the 2008 spring Unit 2 Refueling Outage. Similar to the Surry 1 results, minimal flaws were expected on the Surry 2 steam generator tubes in the tubesheet region.

The licensee initially submitted a license amendment request to the NRC to include the use of IARC for Surry 2 on April 14, 2008. This amendment request also addressed the NRC questions previously received by other licensees on the IARC. With no previous indication of degradation in the tubesheet tube ends, the lower EDY at both Surry units, and the lower Surry operating temperature, there was no basis for an exigent or emergency approval of the IARC license amendment request for Surry 2. Instead, the licensee requested an expedited review similar to other recent industry requests. The results of Surry 2 steam generator inspections being performed during the ongoing refueling outage identified indications that were not observed in the Unit 1 B steam generator in 2007. The results of two of the three Surry 2 steam generator inspections indicate that 54 tubes would require removal from service. Assuming the same results from the third steam generator inspection, approximately 75 tubes would be removed from service. As a result, the licensee submitted a supplemental letter dated May 6, 2008, requesting NRC to approve this amendment based on exigent circumstances. Per later conversation with the licensee, 117 tubes need to be plugged in lieu of 6 tubes if this IARC amendment is not approved.

The NRC staff agrees that based on the performed inspections the number of tubes to be plugged without the approval of this amendment was unanticipated. Without the issuance of this exigent amendment, there will be a significant increase in the number of Surry 2 steam generator tubes removed from service, the steam generator tube margin would be reduced unnecessarily, and additional exposure to station personnel would be incurred to plug these tubes. Therefore, exigent circumstances are present.

## 6.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that license amendments involve no significant hazards considerations if operation of the facility in accordance with the proposed amendments would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below.

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

Response: No

Of the various accidents previously evaluated, the proposed changes only affect the steam generator tube rupture (SGTR) event evaluation and the postulated steam line break (SLB), and locked rotor evaluations. Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this amendment request. Another faulted load consideration is a safe shutdown earthquake (SSE); however, the seismic analysis of Model F steam generators has shown that axial loading of the tubes is negligible during an SSE.

At normal operating pressures, leakage from PWSCC below 17 inches from the TTS is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region.

For the SGTR event, the required structural margins of the steam generator tubes is maintained by limiting the allowable ligament size for a circumferential crack to remain in service to 203 degrees below 17 inches from the TTS for the subsequent operating cycle. Tube rupture is precluded for cracks in the hydraulic expansion region due to the constraint provided by the tubesheet. The potential for tube pullout is mitigated by limiting the allowable crack size to 203 degrees subsequent operating cycle. These allowable crack sizes take into account eddy current uncertainty and crack growth rate. It has been shown that a circumferential crack with an azimuthal extent of 203 degrees for the 18-month SG tubing eddy current inspection interval meet the performance criteria of NEI 97-06, Rev. 2, “Steam Generator Program Guidelines” and Draft Regulatory Guide (RG) 1.121, “Bases for Plugging Degraded PWR Steam Generator Tubes.” Therefore, the margin against tube burst/pullout is maintained during normal and postulated accident conditions and the proposed change does not result in a significant increase in the probability or consequence of a SGTR.

The probability of a SLB is unaffected by the potential failure of a SG tube as the failure of a tube is not an initiator for a SLB event.  SLB leakage is limited by leakage flow restrictions resulting from the leakage path above potential cracks through the tube-to-tubesheet crevice. The leak rate during postulated accident conditions (including locked rotor) has been shown to remain within the accident analysis assumptions for all axial or circumferentially oriented cracks occurring 17 inches below the top of the tubesheet. Since normal operating leakage is limited to 150 gpd, the attendant accident condition leak rate, assuming all leakage to be from indications below 17 inches from the top of the tubesheet would be bounded by 470 gpd.  This value is within the accident analysis assumptions for the limiting design basis accident for Surry, which is the postulated SLB event.

Based on the above, the performance criteria of NEI-97-06, Rev. 2 and Draft Regulatory Guide (RG) 1.121 continue to be met and the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Does the proposed change create the possibility of a new or different accident from any accident previously evaluated?

Response: No

The proposed change does not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity is expected to be maintained for all plant conditions upon implementation of the interim alternate repair criteria. The proposed change does not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

Therefore, based on the above evaluation, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed change involve a significant reduction in a margin of safety?

Response: No

The proposed change maintains the required structural margins of the steam generator tubes for both normal and accident conditions. NEI 97-06, Rev. 2 and RG 1.121 are used as the basis in the development of the limited tubesheet inspection depth methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC staff for meeting GDC 14, 15, 31, and 32 by reducing the probability and consequences of an SGTR. RG 1.121 concludes that by determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service or repaired, the probability and consequences of a SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the ASME Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking in a tube or the tube-to-tubesheet weld, Reference 6 defines a length of remaining tube ligament that provides the necessary resistance to tube pullout due to the pressure induced forces (with applicable safety factors applied). Additionally, it is shown that application of the limited tubesheet inspection depth criteria will not result in unacceptable primary-to-secondary leakage during all plant conditions.

> Based on the above, it is concluded that the proposed changes do not result in any reduction of margin with respect to plant safety as defined in the Updated Final Safety Analysis Report or bases of the plant Technical Specifications.

The NRC staff has reviewed the licensee's analysis given above. Based on this review, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied, and therefore, the amendment request involves no significant hazards consideration. Therefore, the NRC staff has made a final determination that the proposed amendment does not involve a significant hazards consideration.

## 7.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments. The State official had no comments.

## 8.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements 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 amendments involve 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. The Commission has made a final finding that the amendments involved no significant hazards consideration. Accordingly, the amendments meet 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 amendments.

## 9.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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