September 21, 2005

Mr. D. E. Grissette
Vice President
Southern Nuclear Operating
Company, Inc.
P.O. Box 1295
Birmingham, AL 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS REGARDING THE STEAM GENERATOR TUBE SURVEILLANCE PROGRAM (TAC NOS. MC8078 AND MC8079)

Dear Mr. Grissette:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 138 to Facility Operating License NPF-68 and Amendment No. 117 to Facility Operating License NPF-81 for the Vogtle Electric Generating Plant (Vogtle), Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated August 12, 2005, as supplemented by letter dated August 24, 2005.

The amendment request involves a one time change to TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," regarding the required SG inspection scope for Vogtle, Unit 2, during Refueling Outage 11 and the subsequent operating cycle. The proposed changes modify the inspection requirements for portions of the SG tubes within the hot leg tubesheet region of the SGs. The license for Vogtle, Unit 1 is affected only due to the fact that Unit 1 and Unit 2 use common TSs.

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

Sincerely,

/RA/

Christopher Gratton, Sr. Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:
1. Amendment No. 138 to NPF-68
2. Amendment No. 117 to NPF-81
3. Safety Evaluation

cc w/encls: See next page

September 21, 2005

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Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

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3. Safety Evaluation

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Package Number: ML052630014 Amendment Number: ML052630011
Tech Spec Number: ML052650114 NRR-058

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SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS REGARDING THE STEAM GENERATOR TUBE SURVEILLANCE PROGRAM (TAC NOS. MC8078 AND MC8079)

Date: September 21, 2005

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SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 138
License No. NPF-68

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 1 (the facility) Facility Operating License No. NPF-68 filed by the Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated August 12, 2005, as supplemented by letter dated August 24, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-68 is hereby amended to read as follows:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 138, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Evangelos C. Marinos, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 21, 2005

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 117
License No. NPF-81

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 2 (the facility) Facility Operating License No. NPF-81 filed by the Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated August 12, 2005, as supplemented by letter dated August 24, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-81 is hereby amended to read as follows:

Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 117, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Evangelos C. Marinos, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 21, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 138

FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND

TO LICENSE AMENDMENT NO. 117

FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

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

Remove	Insert
5.5-10	5.5-10
5.5-11	5.5-11

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 138 TO FACILITY OPERATING LICENSE NPF-68

AND AMENDMENT NO. 117 TO FACILITY OPERATING LICENSE NPF-81

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By letter dated August 12, 2005 (Agencywide Documents Access and Management System (ADAMs) Accession No. ML052270248), as supplemented by letter dated August 24, 2005 (ADAMS Accession No. ML052370173), Southern Nuclear Operating Company, (the licensee), requested changes to the Technical Specifications (TSs) for Vogtle Electric Generating Plant (Vogtle), Units 1 and 2. The supplemental letter dated August 24, 2005, provided clarifying information that did not change the scope of the August 12, 2005, application and the initial proposed no significant hazards consideration determination.

The amendment requests involves a one time change to TS 5.5.9, “Steam Generator (SG) Tube Surveillance Program,” regarding the required SG inspection scope for Vogtle, Unit 2, during Refueling Outage 11 and the subsequent operating cycle. The proposed changes modify the inspection requirements for portions of the SG tubes within the hot leg tubesheet region of the SGs. The license for Vogtle Unit 1 is affected only due to the fact that Unit 1 and Unit 2 use common TSs. Specifically, the proposed changes would modify:

TS 5.5.9.d.1.f, Plugging Limit

Two new paragraphs have been added to state:

> For Unit 2 during Refueling Outage 11 and the subsequent operating cycle, this definition does not apply to degradation identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet. Degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging.
>
> For Unit 2 during Refueling Outage 11 and the subsequent operating cycle, degradation identified in the portion of the tube from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be plugged upon detection.

TS 5.5.9.d.1.h, Tube Inspection

A new paragraph has been added to state:

> For Unit 2 during Refueling Outage 11 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded.

1.1 Background

Vogtle, Units 1 and 2 have four Westinghouse Model F SGs. There are 5626 thermally treated Alloy 600 tubes in each of the steam generators. The tubes have an outside diameter of 11/16 inch, a wall thickness of 0.040-inch, and are supported by seven stainless steel tube support plates and a flow distribution baffle. The tube support plate holes are quatrefoil shaped. The U-bend region of the tubes in rows 1 through 10 was stress relieved after bending. A total of 55 tubes and 42 tubes are plugged in Vogtle, Units 1 and 2, respectively. Vogtle, Unit 1 is currently in Cycle 13 operation. Vogtle, Unit 2 is in Cycle 11 operation.

The licensee has been using bobbin probes for inspecting the length of tubing within the tubesheet; however, the bobbin probe is not capable of reliably detecting stress corrosion cracks (SCCs) in the tubesheet region should such cracks be present. For this reason, the licensee has been supplementing the bobbin probe inspections with rotating coil probes 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 zone located at the TTS. The expansion transition contains significant residual stress and was considered a likely location for a SCC should it develop. Until the fall of 2004, there had not been any reported instances of SCCs affecting the tubesheet region of thermally treated Alloy 600 tubing, either at Vogtle or elsewhere in the U.S.

In the fall of 2004, crack-like indications were found in tubes in the tubesheet region of Catawba Nuclear Station (Catawba), Unit 2, which have Westinghouse model D5 SGs. Like the Vogtle SGs, the Catawba, Unit 2 SGs have thermally treated Alloy 600 tubing that is hydraulically expanded against the tubesheet. The crack-like indications at Catawba, Unit 2 were found in bulges (or over-expansions) in the tubesheet region, in the tack roll region, and in the tube-to-tubesheet weld. (The tack expansion is an initial 0.7-inch long expansion at each tube end and formed prior to the hydraulic expansion over the full tubesheet depth. Its purpose was to facilitate performing the tube-to-tubesheet weld.) Crack-like indications were found in a bulge in one tube and in the tack expansion in nine tubes. Approximately 6 of the 196 tube-to-tubesheet weld indications extended into the parent tube.

As a result of the Catawba, Unit 2 findings, the Vogtle licensee plans to expand the scope of previous rotating coil inspections to address the potential for cracks within the thickness of the tubesheet down to 17 inches below the TTS. However, the licensee believes that any flaws located below 17 inches below the TTS (i.e., in the bottom 4 inches of the tubesheet region, including the tack expansion region and the tubing in the vicinity of the welds) have no potential to impair tube integrity and, thus, do not pose a safety concern. To avoid the unnecessary plugging of tubes as would be required by the TS should inspection reveal cracks in this region, the licensee is proposing on a one time basis to revise the TS such that tubes found to contain flaws in the lower 4 inches of the tubesheet region need not be plugged and that the lower

4-inch region be excluded from current inspection requirements. In addition, the licensee proposed new requirements requiring that all tubes found with degradation in the upper region of the tubesheet be plugged.

2.0 REGULATORY EVALUATION

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 coolant from the secondary coolant and the environment. For the purposes of this Safety Evaluation (SE), tube integrity means that the tubes are capable of performing these functions in accordance with the plant design and licensing basis.

Title 10 of the *Code of Federal Regulations* (10 CFR) establishes the fundamental regulatory requirements with respect to the integrity of the SG tubing. Specifically, the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 state that the RCPB shall have “an extremely low probability of abnormal leakage ... and gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDC 15 and 31), shall be of "the highest quality standards possible" (GDC 30), and shall be designed to permit "periodic inspection and testing ... to assess ... structural and leak tight integrity" (GDC 32). To this end, 10 CFR 50.55a specifies that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code). Section 50.55a further requires, in part, that throughout the service life of a pressurized-water reactor (PWR) facility, 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 [ISI] 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 SG tube surveillance requirements in the TS.

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 an SG tube rupture and main steamline break (MSLB). These analyses consider the primary-to-secondary leakage through the tubing that may occur during these events and must show that the offsite radiological consequences do not exceed the GDC 19 criteria for control room operator doses or the applicable limits of the 10 CFR Part 100 guidelines for offsite doses or some fraction, thereof, as appropriate to the accident, or the Nuclear Regulatory Commission (NRC)-approved licensing basis (e.g., a small fraction of these limits).

Under the plant TS SG surveillance program requirements, the licensee is required to monitor the condition of the SG tubing and to plug tubes as necessary. Specifically, the licensee is required to perform periodic inspections of and to remove from service by plugging all tubes found to contain flaws with sizes exceeding the acceptance limit, termed "plugging limit." The tube plugging limits were developed with the intent of ensuring that degraded tubes: (1) maintain factors of safety against gross rupture consistent with the plant design basis (i.e., consistent with the stress limits of the ASME Code, Section III); and (2) maintain leakage integrity consistent with the plant licensing basis while, at the same time, allowing for potential flaw size measurement error and flaw growth between SG inspections. The required frequency and scope of tubing examinations and the tube plugging limits are specified in TS 5.5.9, "Steam Generator (SG) Tube Surveillance Requirements."

The subject TS amendment request concerns the portions of the tubing that are subject to the TS SG tube surveillance requirements, including any necessary plugging or repairs, and the inspection methods to be employed. TS 5.5.9 defines a tube inspection as an inspection of the SG tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. This includes the full length of tubing within the thickness of the tubesheet on the hot leg side.

The proposed license amendment would limit the required inspections and plugging in the 21-inch thick tubesheet region to the upper 17 inches of the tubesheet region and is conceptually similar to permanent amendments approved by the NRC staff for a number of plants. Examples include the F* criteria approved for Westinghouse SGs where the tubes were hard roll expanded inside the tubesheet and the W* criteria approved for plants where the tubes were explosively expanded against the tubesheet. In the case of the F* criteria, the required inspection zone was limited to approximately the upper 1.5-inch zone below the TTS. The W* criteria required an inspection zone extending approximately 8 inches below the TTS. The larger required inspection zone for W* relative to F* results from the explosively expanded joints not exhibiting as much residual interference fit as do hard rolled joints. The proposed license amendment for Vogtle is similar to recently approved amendments at other plants where the tubes are hydraulically expanded against the tubesheet.

3.0 TECHNICAL EVALUATION

The tube-to-tubesheet joint consists of the tube, which is hydraulically expanded against the tubesheet and the tube-to-tubesheet weld located at the tube end. The joint was designed as a welded joint in accordance with the ASME Code, Section III, not as a friction or expansion joint. The weld itself was designed as a pressure boundary element in accordance the ASME Code, Section III. 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 licensee, in effect, is proposing on a one-time basis to exempt the lower 4 inches of the 21-inch deep tubesheet region from a tube inspection (see proposed change to TS 5.5.9.d.1.h, “Tube Inspection”) and to exempt tubes with flaw indications in the lower 4-inch zone from the need to plug (see proposed revision to TS 5.5.9d.1.f, “Plugging Limit”). The latter part of this proposal (i.e., to exempt tubes from plugging) is needed as a practical matter since, although rotating coil probe inspections will not be performed in this region, the bobbin probe will necessarily be recording any signals produced in this zone. This proposal, in effect, redefines the pressure boundary at the tube-to-tubesheet joint as consisting of a friction or expansion joint with the tube assumed to be hydraulically expanded against tubesheet over the top 17 inches of the tubesheet region. Under this proposal, no credit is taken for the lower 4 inches of the tube or the tube-to-tubesheet weld in contributing to the structural or leakage integrity of the joint. The lower 4 inches of the tube and weld are assumed not to exist.

The regulatory standard by which the NRC staff has evaluated the subject license amendment is that the amended TSs should continue to ensure that tube integrity will be maintained. This includes maintaining structural safety margins consistent with the plant design basis as embodied in the stress limit criteria of the ASME Code, Section III as is discussed in Section 4.1

below. In addition, this includes limiting the potential for accident induced primary to secondary leakage to values not exceeding those assumed in the licensing basis accident analyses. Maintaining tube integrity in this manner ensures that the amended TS are in compliance with all applicable regulations. The NRC staff's evaluation of joint structural integrity and leakage integrity is discussed in Sections 3.1 and 3.2 of this SE, respectively.

The licensee is also proposing on a one-time basis to plug all tubes found with degradation in the upper 17-inch region of the tubesheet (see proposed revision to TS 5.5.9d.1.f, "Plugging Limit"). "Degradation" is defined in the TSs to mean service-induced cracking, wastage, wear, or general corrosion occurring on either the inside or outside of a tube. This definition is not limited to flaws that exceed the current TS 40 percent plugging limit. The NRC staff finds this proposed requirement acceptable since it is more conservative than the current TS 40 percent plugging limit and will provide added assurance that the length of tubing along the entire proposed 17-inch inspection zone will be effective in resisting tube pullout under tube end cap pressure loads and in resisting primary-to-secondary leakage between the tube and tubesheet.

3.1 Joint Structural Integrity

Westinghouse has conducted analysis and testing to establish the engagement (embedment) length of hydraulically expanded tubing inside the tubesheet that is necessary to resist pullout under normal operating and design basis accident conditions. Pullout is the structural failure mode of interest since the tubes are radially constrained against axial fishmouth rupture by the presence of the tubesheet. The axial force that could produce pullout derives from the pressure end cap loads due to the primary to secondary pressure differentials associated with normal operating and design basis accident conditions. The licensee's contractor, Westinghouse, determined the required engagement distance on the basis of maintaining a factor of three against pullout under normal operating conditions and a factor of 1.4 against pullout under accident conditions. Pullout was conservatively treated as tube slippage relative to the tubesheet of 0.25 inches. The NRC staff concurs that these are the appropriate safety factors to apply to demonstrate structural integrity. As documented in detail in a SE accompanying the NRC staff's approval of new performance based SG TS for Farley Units 1 and 2 (Reference: Letter, Sean Peters, NRC, to L. M. Stinson, Vice President, Southern Nuclear Operating Company, "Joseph M. Farley Nuclear Plant, Units 1 and 2, re: Issuance of Amendments to Facilitate Implementation of Industry Initiative NEI 97-06, Steam Generator Program Guidelines," dated September 10, 2004, (ADAMS Accession No. ML042570427), the NRC staff has concluded that these safety factor criteria are consistent with the design basis; namely the stress limit criteria in the ASME Code, Section III.

The resistance to pullout is the axial friction force developed between the expanded tube and the tubesheet over the engagement distance. The friction force is a function of the radial contact pressure between the expanded tube and the tubesheet. The radial contact pressure derives from several contributors including (1) the contact pressure associated directly with the hydraulic expansion process itself, (2) additional contact pressure due to differential radial thermal expansion between the tube and tubesheet under hot operating conditions, (3) additional contact pressure caused by the primary pressure inside the tube, and (4) additional or reduced contact pressure associated with tubesheet bore dilation (distortion) caused by tubesheet bow (deflection) as a result of the primary-to-secondary pressure load acting on the tubesheet. Westinghouse employed a combination of pullout tests and analyses, including finite element analyses, to evaluate these contributors. Based on these analyses and tests, Westinghouse concludes that the required engagement distances to ensure the safety factor criteria against pullout are achieved vary from about 3.59 to 8.51 inches depending on the radial location of the tube within the tube bundle, with the largest engagement distances needed toward the center of the bundle.

The NRC staff has not reviewed the Westinghouse analyses in detail and, thus, has not reached a conclusion with respect to whether 3.59 to 8.51 inches of engagement (termed H* criterion by Westinghouse) is adequate to ensure that the necessary safety margins against pullout are maintained. The licensee, therefore, is proposing on a one time basis to inspect the tubes in the tubesheet region such as to ensure a minimum of 17 inches of effective engagement, well in excess of the 3.59 to 8.51 inches that the Westinghouse analyses indicate are needed. Based on the following considerations, the NRC staff concludes the proposed 17-inch engagement length is clearly acceptable to ensure the structural integrity of the tubesheet joint.

- The NRC staff estimates based on pullout tests that the radial contact pressure produced by the hydraulic expansion and differential radial thermal expansion is such as to require an engagement distance of 8.6 inches to ensure the appropriate safety margins against pullout based on no-slip. This estimate is a mean minus one standard deviation estimate based on six pullout tests. This estimate ignores the effect on needed engagement distance from internal primary pressure in the tube and tubesheet bore dilations associated with tubesheet bow. The NRC staff notes that from a tube pullout standpoint, the use of a “no slip” criterion is conservative. Allowing slippage of about 0.2 to 0.3 inches decreases the necessary engagement distance to 5.1-inch, again ignoring the effect on needed engagement distance from internal primary pressure in the tube and tubesheet bore dilations associated with tubesheet bow.

- The internal primary pressure inside the tube under normal operating and accident conditions also acts to tighten the joint relative to unpressurized conditions, thus reducing the necessary engagement distance.

- Tubesheet bore dilations caused by tubesheet bow under primary to secondary pressure can increase or decrease contact pressure depending on the tube location within the bundle and on location along the length of the tube in the tubesheet region. Basically, the tubesheet acts as a flat, circular plate under an upward acting net pressure load. The tubesheet is supported axially around its periphery with a partial restraint against tubesheet rotation provided by the SG shell and channel head. The SG divider plate provides a spring support against upward displacement along a diametral mid-line. Over most of the tubesheet away from the periphery, the bending moment resulting from the applied primary to secondary pressure load can be expected to put the tubesheet into tension at the top and compression at the bottom. Thus, the resulting distortion of the tubesheet bore (tubesheet bore dilation) tends to be such as to loosen the tube to tubesheet joint at the top of the tubesheet and to tighten the joint at the bottom of the tubesheet. The amount of dilation and resulting change in joint contact pressure would be expected to vary in a linear fashion from top to bottom of the tubesheet. Given the neutral axis to be at approximately the axial mid-point of the tubesheet thickness (i.e., 10.5 inches below the top of the tubesheet), tubesheet bore dilation effects would be expected to further tighten the joint from 10 inches below the TTS to 17 inches below the TTS which would be the lower limit of the proposed tubesheet region inspection zone. Combined with the effects of the joint tightening associated with the primary pressure inside the tube, contact pressure over at least a 6.5-inch distance should be considerably higher than the contact pressure simulated in the above mentioned pull out tests. A similar logic applied to the periphery of the tubesheet leads the staff to conclude that at the top 10.5 inches of the tubesheet region, contact pressure should be considerably higher than the contact pressure simulated in the above mentioned pull out tests. Thus, the NRC staff concludes that the proposed 17-inch engagement distance (or inspection zone) is acceptable to ensure the structural integrity of the tubesheet joint.

3.2 Joint Leakage Integrity

If no credit is to be taken for the presence of the tube-to-tubesheet weld, a potential leak path between the primary to secondary is introduced between the hydraulically expanded tubing and the tubesheet. In addition, not inspecting the tubing in the lower 4 inches of the tubesheet region may lead to an increased potential for 100 percent throughwall flaws in this zone and the potential for leakage of primary coolant through the crack and up between the hydraulically expanded tubes and tubesheet to the secondary system. Operational leakage integrity is assured by monitoring primary to secondary leakage relative to the applicable TS limiting condition for operation (LCO) limits. However, it must also be demonstrated that the proposed TS changes do not create the potential for leakage during design basis accidents that may exceed values assumed in the licensing basis accident analyses. The licensee states that this is ensured by limiting primary to secondary leakage to 0.35 gallons per minute (gpm) in the faulted SG during a MSLB.

To support its H* criterion (discussed above), Westinghouse has developed a detailed leakage prediction model that considers the resistance to leakage from cracks located within the thickness of the tubesheet. The NRC staff has not reviewed or accepted this model. For the proposed one time 17-inch inspection zone, Westinghouse cited a number of qualitative arguments supporting a conclusion that a minimum 17-inch engagement length ensures that leakage during MSLB will not exceed two times the observed leakage during normal operation. Westinghouse refers to this as the “bellwether approach.” Thus, for a SG leaking at 150 gallons per day or 0.10 gpm under normal operating conditions (consistent with the licensee’s regulatory commitment with this value stated in a letter dated August 24, 2005, (ADAMs Accession No. ML052370173) , Westinghouse estimates that leakage would not be expected to exceed 0.20 gpm, which is less than the 0.35 gpm assumed in the licensing basis accident analyses for MSLB.

The factor of 2 upper bound is based on the Darcy equation for flow through a porous media where leakage rate would be proportional to differential pressure. Westinghouse considered normal operating pressure differentials between 1200 and 1400 pound per square inch (psi) and accident differential pressures on the order of 2560 to 2650 psi, essentially a factor of 2 difference. The factor of 2 as an upper bound is based on a premise that the flow resistance between the tube and tubesheet remains unchanged. Westinghouse states that the flow resistance varies as a log normal linear function of joint contact pressure. The NRC staff finds that the factor of 2 upper bound is reasonable, given the stated premise. The NRC staff notes that the assumed linear relationship between leak rate and differential pressure is conservative relative to alternative models such as Bernoulli or orifice models, which assume the leak rate to be proportional to the square root of differential pressure.

The NRC staff reviewed the qualitative arguments developed by Westinghouse regarding the conservatism of the aforementioned premise, namely the conservatism of assuming that flow resistance between the expanded tubing and the tubesheet does not decrease under the most limiting accident relative to normal operating conditions. Most of the Westinghouse observations are based on insights derived from the finite element analyses performed to assess joint contact pressures and from test data relating leak flow resistance to joint contact pressure, neither of which has been reviewed by the NRC staff in detail. Among the Westinghouse observations is that for all tubes there is at least an 11-inch zone in the upper 17 inches of the tubesheet where there is an increase in joint contact pressure, and, thus, leak flow resistance, due to higher primary pressure inside the tube and changes in tubesheet bore dilation along the length of the tubes. In Section 3.1 above, the NRC staff observed that there is at least a 6.5-inch zone over which changes in tubesheet bore dilations when going from unpressurized to pressured conditions should result in an increase in joint contact pressure. The contact pressure due to changes in tubesheet bore dilation should increase further over the 6.5-inch zone under the increased pressure loading on the tubesheet during accident conditions. Considering the higher pressure loading in the tube when going from normal operating to accident conditions, the NRC staff estimates on a qualitative basis that the length over which contact pressure would be expected to increase should be at least 10 inches, which is reasonably consistent with the more detailed Westinghouse analysis.

Although joint contact pressures and leak flow resistance decrease over other portions of the tube length, Westinghouse expects a net increase in total leak flow resistance on the basis of its insights from leakage test data that leak flow resistance is more sensitive to changes in joint contact pressure as contact pressure increases due to the linear log normal nature of the relationship. The NRC staff's depth of review did not permit it to credit this aspect of the Westinghouse assessment. However, it is clear from the above discussion that there should be no significant reduction in leakage flow resistance when going from normal operating to accident conditions.

Finally, the NRC staff has considered that undetected cracks in the lower 4 inches are unlikely to produce leakage rates during normal operation that would approach the TS LCO operational leakage limits during normal operation, thus providing additional confidence that such cracks will not result in leakage in excess of the values assumed in the accident analyses. Any axial cracks will be tightly clamped by the tubesheet against opening of the crack faces. In addition, minimal end cap pressure load should remain in the tube below 17 inches and, thus, any circumferential cracks would be expected to remain tight. Thus, irrespective of the flow resistance in the upper 17 inches of the tubesheet between the tube and tubesheet, the tightness of the cracks themselves should limit leakage to very small values.

Based on the above, the NRC staff concludes that there is reasonable assurance that the proposed one time exclusion of the lower 4 inches of the tubes in the tubesheet region from the tube inspection and plugging and repair requirements will not impair the leakage integrity of the tube-to-tubesheet joint.

3.5 Summary

The NRC staff finds that the proposed one time license amendment ensures that the structural and leakage integrity of the tube-to-tubesheet joint will be maintained with structural safety margins consistent with the design basis, with leakage integrity within assumptions employed in the licensing basis accident analyses and, thus, in accordance with the applicable regulations without undue risk to public health and safety. Based on this, the NRC staff concludes that the proposed changes to TS 5.5.9 are acceptable.

4.0 NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission's regulations in 10 CFR 50.92, "Issuance of amendment," paragraph (c), state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment 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.

The following analysis was provided by the licensee in its letter dated August 12, 2005:

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

   No. The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed changes that alter the SG inspection criteria do not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed changes will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

   Of the applicable accidents previously evaluated, the limiting transients with consideration to the proposed changes to the SG tube inspection criteria, are the SG tube rupture (SGTR) event and the steam line break (SLB) accident.

   During the SGTR event, the required structural integrity margins of the SG tubes will be maintained by the presence of the SG tubesheet. SG tubes are hydraulically expanded in the tubesheet area. Tube rupture in tubes with cracks in the tubesheet is precluded by the constraint provided by the tubesheet. This constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet and from the differential pressure between the primary and secondary side. Based on this design, the structural margins against burst, discussed in Regulatory Guide (RG) 1.12 1, "Bases for Plugging Degraded PWR [Pressurized-Water Reactor] SG Tubes," are maintained for both normal and postulated accident conditions.

The proposed changes do not affect other systems, structures, components or operational features. Therefore, the proposed changes result in no significant increase in the probability of the occurrence of a SGTR accident.

At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) below the proposed limited inspection depth 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. The consequences of an SGTR event are affected by the primary-to-secondary leakage flow during the event.

Primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed change since the tubesheet enhances the tube integrity in the region of the hydraulic expansion by precluding tube deformation beyond its initial hydraulically expanded outside diameter.

The probability of a SLB is unaffected by the potential failure of a SG tube as this failure is not an initiator for a SLB.

The consequences of a SLB are also not significantly affected by the proposed changes. During a SLB accident, the reduction in pressure above the tubesheet on the shell side of the SG creates an axially uniformly distributed load on the tubesheet due to the reactor coolant system pressure on the underside of the tubesheet. The resulting bending action constrains the tubes in the tubesheet thereby restricting primary-to-secondary leakage below the midplane.

Primary-to-secondary leakage from tube degradation in the tubesheet area during the limiting accident (i.e., SLB) is limited by flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of potential crack face opening as compared to free span indications. The primary-to-secondary leak rate during postulated SLB accident conditions would be expected to be less than that during normal operation for indications near the bottom of the tubesheet (i.e., including indications in the tube end welds). This conclusion is based on the observation that while the driving pressure causing leakage increases by approximately a factor of two, the flow resistance associated with an increase in the tube-to-tubesheet contact pressure, during a SLB, increases by up to approximately a factor of three. While such a leakage decrease is logically expected, the postulated accident leak rate could be conservatively bounded by twice the normal operating leak rate if the increase in contact pressure is ignored. Since normal operating leakage is administratively limited (by NEI [Nuclear Energy Instriture] 97-06) to less than 0.10 gpm (150 gpd) in the Vogtle Unit 2 steam generators, the attendant accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be bounded by 0.20 gpm, which is less than the accident analysis assumption of 0.35 gpm included in Section 15.1.5 of the Vogtle Unit 2 UFSAR. Hence it is reasonable to omit any consideration of inspection of the tube, tube end weld, bulges/over expansions or other anomalies below 17 inches from the top of the hot leg tubesheet. Therefore, the consequences of a SLB accident remain unaffected.

Based on the above discussion, the proposed changes do not involve an increase in the consequences of an accident previously evaluated.

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

No. The proposed changes do not involve the use or installation of new equipment and the currently installed equipment will not be operated in a new or different manner. No new or different system interactions are created and no new processes are introduced. The proposed changes will not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases.

Based on this evaluation, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

No. The proposed changes maintain the required structural margins of the SG tubes for both normal and accident conditions. Nuclear Energy Institute (NEI) 97- 06, "Steam Generator Program Guidelines," Revision 1 and Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," are used as the bases in the development of the limited hot leg tubesheet inspection depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. RG 1.12 1 describes a method acceptable to the NRC for meeting General Design Criteria (GDC) 14, "Reactor coolant pressure boundary," GDC 15, "Reactor coolant system design," GDC 31, "Fracture prevention of reactor coolant pressure boundary," and GDC 32, "Inspection of reactor coolant pressure boundary," by reducing the probability and consequences of a SGTR. RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation 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 American Society of Mechanical Engineers (ASME) Code.

Application of the limited hot leg tubesheet inspection depth criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the proposed limited hot leg tubesheet inspection depth criteria.

Therefore, [the licensee concludes that] the proposed changes do not involve a significant hazards consideration under the criteria set forth in 10 CFR 50.92(c).

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff determines that the proposed amendment involves no significant hazard consideration.

5.0 STATE CONSULTATION

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

6.0 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 made a no significant hazards finding with respect to this amendment.

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.

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

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