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SOUTHERN COMPANY
Energy to Serve Your World℠

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NL-06-1706

U. S. Nuclear Regulatory Commission
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Washington, D. C. 20555-0001

Vogtle Electric Generating Plant
Revision to Request for License Amendment Related to Technical Specification 5.5.9
"Steam Generator (SG) Tube Surveillance Program"

Ladies and Gentlemen:

On July 28, 2006, the NRC advised Southern Nuclear Operating Company (SNC) that because the application for amendment to the Vogtle Electric Generating Plant (VEGP) Technical Specifications as requested by letter dated July 20, 2006, differed from a previously reviewed VEGP precedent for addressing steam generator tube ends, an extensive NRC review of the proposed amendment would be required. Therefore, the VEGP application for amendment to the Technical Specifications is being revised via this letter to modify the inspection and plugging requirements for steam generator hot leg side tube ends only. The original application proposed modifying the inspection and plugging requirements for both the cold leg and hot leg sides of the steam generator tube ends.

The basic approach for the requested amendment remains the same as that described in the July 20, 2006 letter, in which the proposed one-time change would revise TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," to incorporate changes in the SG inspection scope for VEGP Unit 1 during Refueling Outage 13 and the subsequent operating cycle, and VEGP Unit 2 during Refueling Outage 12 and the subsequent operating cycle. The proposed changes modify the inspection and plugging requirements for portions of SG tubes within the hot leg side of the tubesheet region of the SGs only.

The attached revised amendment request is subdivided as shown below.

Enclosure 1 provides a revised basis for the proposed change, an evaluation determining that the proposed change involves no significant hazards consideration as defined in 10 CFR 50.92, and the evaluation that determines this change satisfies the criteria of 10 CFR 51.22 for categorical exclusion from the requirements for an environmental assessment. The 10 CFR 50.92 evaluation provided with our July 20, 2006 submittal has not changed and remains valid with respect to this revision to the application for amendment to the Technical Specifications. In addition, the evaluation performed pursuant to 10 CFR 51.22 has not changed.

Enclosure 2 includes the revised set of marked-up TS pages with the proposed changes indicated for VEGP, incorporated into the proposed TS requirements for the steam generator program based on TSTF-449.

Enclosure 3 includes the associated revised set of typed TS pages with the proposed changes incorporated for VEGP.

The previously provided Westinghouse technical reports and associated application for withholding, affidavit, proprietary information notice, and copyright notice for information proprietary to Westinghouse Electric Company, LLC, continue to apply to this revised amendment request.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-06-2176 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC; P. O. Box 355; Pittsburgh, Pennsylvania 15230-0355.

Southern Nuclear Operating Company requests approval of the proposed license amendments September 1, 2006, in order to support the VEGP-1 and VEGP-2 refueling outages that are currently scheduled to begin September 17, 2006, and March 4, 2007, respectively. The proposed changes will reduce the potential for unnecessary plugging of SG tubes which could further adversely impact the ability of VEGP-1 and VEGP-2 to achieve their licensed power level. In addition, personnel responsible for the tube plugging activities will not be subject to additional radiation dose by having to unnecessarily plug SG tubes. The proposed changes would be implemented within 30 days of issuance of the amendment.

(Affirmation and signature are on the following page.)

Mr. D. E. Grissette states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

This letter contains no NRC commitments. If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

Don E. Grissette

Sworn to and subscribed before me this 4th *day of* August*, 2006.*

Glenda C. Sparks
Notary Public

GLENDA C. SPARKS NOTARY PUBLIC GEORGIA STATE AT LARGE

My commission expires: 11/10/06

DEG/DRG/daj

Enclosures:
1. Basis for the Proposed Change - Revised
2. Markup of Proposed Technical Specifications Page Changes for VEGP - Revised
3. Typed Pages for Technical Specification Changes for VEGP - Revised

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. T. E. Tynan, General Manager – Plant Vogtle
RType: CVC7000

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. C. Gratton, NRR Project Manager – Vogtle
Mr. G. J. McCoy, Senior Resident Inspector – Vogtle

State of Georgia
Mr. L. C. Barrett, Commissioner – Department of Natural Resources

Enclosure 1

Vogtle Electric Generating Plant
Request for Technical Specifications Amendment (Revised)
Steam Generator Tube Surveillance Program
Basis for the Proposed Change - Revised

Enclosure 1
Vogtle Electric Generating Plant
Request for Technical Specifications Amendment
Steam Generator Tube Surveillance Program

Basis for the Proposed Change - Revised

1.0 Description

The proposed one-time change would revise Technical Specification (TS) steam generator (SG) program requirements to incorporate changes to SG tubing inspection and plugging requirements for Vogtle Electric Generating Plant (VEGP) Unit 1, during Refueling Outage 13 and the subsequent operating cycle and VEGP Unit 2, during Refueling Outage 12 and the subsequent operating cycle. The proposed changes modify the inspection and plugging requirements for portions of SG tubes within the tubesheet region of the SGs by excluding approximately 4 inches at the tube ends on the hot leg side of the SG from inspection and/or plugging requirements. This single-cycle change for both units is based on structural analysis and leak rate evaluation results and constitutes a redefinition of the primary-to-secondary pressure boundary. This change is supported by Westinghouse Electric Company LLC as described in LTR-CDME-06-58-P, "Steam Generator Tube Alternate Repair Criteria for the Portion of the Tube Within the Tubesheet at the Vogtle 1 & 2 Electric Generating Plant for One Cycle Application", dated July 11, 2006.

The NRC approved a similar one-time change for VEGP, Unit 2, to apply during Refueling Outage 11 and subsequent operating cycle, by letter from NRC to D. E. Grissette (Southern Nuclear Operating Company), "Vogtle Electric Generating Plant, Units 1 and 2, Issuance of Amendments Regarding the Steam Generator Tube Surveillance Program (TAC NOS. MC8078 AND MC8079)" dated September 21, 2005.

Southern Nuclear Operating Company (SNC) submitted a request to amend the VEGP TS by letter NL-06-0124, Don E. Grissette to U. S. Nuclear Regulatory Commission, dated March 29, 2006, with supplemental changes being proposed by letter NL-06-0990, Don E. Grissette to U. S. Nuclear Regulatory Commission, dated June 5, 2006. The TS amendment request proposed incorporating TSTF-449, Steam Generator Tube Integrity, into the VEGP TS; therefore, the proposed changes addressed within this letter are written into the proposed specification changes and associated bases changes of SNC letters NL-06-0124 and NL-06-0990.

On July 28, 2006, the NRC advised Southern Nuclear Operating Company (SNC) that because the application for amendment to the Vogtle Electric Generating Plant (VEGP) Technical Specifications as requested by letter NL-06-0708 dated July 20, 2006, Don E. Grissette to U. S. Nuclear Regulatory Commission (Reference 7), differed from a previously reviewed VEGP precedent for addressing steam generator tube ends, an extensive NRC review of the proposed amendment would be required. Therefore, the VEGP application for amendment to the Technical Specifications is being revised via this letter to modify the inspection and plugging requirements for steam generator hot leg side tube ends only. The original application proposed modifying the inspection and plugging requirements for both the cold leg and hot leg sides of the steam generator tube ends.

2.0 Proposed Change

Proposed changes to TS 5.5.9, Steam Generator (SG) Program, are to be incorporated into the technical specification pages as proposed in NL-06-0124 and NL-06-0990, and are summarized below.

TS 5.5.9.c, "Provisions for SG tube repair criteria"

Alternate repair criterion #2 has been added as described below:

"2. *For Unit 1 during Refueling Outage 13 and the subsequent operating cycle and for Unit 2 during Refueling Outage 12 and the subsequent operating cycle, degradation identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging.*

For Unit 1 during Refueling Outage 13 and the subsequent operating cycle and for Unit 2 during Refueling Outage 12 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 hot leg tubesheet shall be plugged upon detection."

TS 5.5.9.d, "Provisions for SG tube inspections"

The following sentence will be added:

"For Unit 1 during Refueling Outage 13 and the subsequent operating cycle and for Unit 2 during Refueling Outage 12 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded."

3.0 Background

VEGP Units 1 and 2 are four loop Westinghouse-designed plants with Model F steam generators (SG) having nominal 11/16 inch (OD) thermally treated A600TT tubes, full-depth hydraulically expanded tubesheet joints, and broached hole quatrefoil tube support plates constructed of stainless steel. The tubesheet is approximately 21 inches thick and there are 5626 tubes in each SG. A total of 55 and 42 tubes are plugged in VEGP Units 1 and 2, respectively. VEGP Unit 1 is currently in Cycle 13 operation. VEGP Unit 2 is in Cycle 12 operation.

Indications of SG tube cracking were reported by Catawba Nuclear Station Unit 2 based on the results from the nondestructive, eddy current examination of the SG tubes during the fall 2004 outage as described in NRC Information Notice 2005-09, Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds. The SGs at the Catawba 2 plant are type Westinghouse Model D5 with 3/4 inch nominal outside diameter (OD) Alloy 600 tubing (A600TT). The tube indications at Catawba 2 were reported approximately 7 inches below the top of the tubesheet (TTS) on the hot leg (HL) side in one tube, and just above the tube-to-tubesheet welds in a region of the tube known as the tack expansion (TE) in several other tubes. Finally, indications were also reported in the tube-end welds (TEWs), also known as tube-to-tubesheet welds, joining the tube to the tubesheet with a small number of those indications extending into the tubes.

Because of the indications detected in tubing within the tubesheet at Catawba 2, additional rotating probe inspections were performed in overexpanded locations in tubing within the hot leg tubesheet at VEGP Unit 1 in eddy current inspections performed in the spring 2005 refueling outage. VEGP Unit 1 reported circumferential indications in two SG tubes in SG 4 (one tube had two indications) in overexpanded locations within the hot leg tubesheet. In fall 2005, eddy current inspection was performed in 2 of 4 SGs of the VEGP Unit 2 SG tubing. No degradation was detected in the tubesheet region in the sample inspections performed during this Unit 2 inspection. Because of the Catawba fall 2004 inspection results and the VEGP Unit 1 spring 2005 results, there are 3 general issues with regard to the VEGP Units 1 and 2 SG tubes:

1. indications in internal bulges and overexpansions within the hot leg tubesheet;
2. indications at the elevation of the tack expansion transition; and
3. indications in the hot leg tube-to-tubesheet welds and propagation of these indications into the adjacent tube material.

The SG inspection scope is governed by TS 5.5.9, Nuclear Energy Institute 97-06, "Steam Generator Program Guidelines", Electric Power Research Institute (EPRI) "Pressurized Water Reactor Steam Generator Examination Guidelines" (SG Examination Guidelines), SG degradation assessments which SNC prepares to support each SG tubing inspection, and SNC procedures. Criterion IX, "Control of Special Processes," of 10 CFR Part 50, Appendix B, requires in part that nondestructive testing is to be accomplished by qualified personnel using qualified procedures in accordance with the applicable criteria. The inspection techniques and equipment were capable of reliably detecting the known and potential specific degradation mechanisms applicable to VEGP. The inspection techniques, essential variables, and equipment were qualified to Appendix H of the SG Examination Guidelines, "Performance Demonstration for Eddy Current Examination."

The SG degradation assessment (DA) is prepared by SNC prior to each SG inspection. The DA is performed to identify degradation mechanisms that may be present, and includes a review of operating experience. A validation is performed to verify that the eddy current techniques utilized are capable of detecting those flaw types that are identified in the degradation assessment. Based on operating experience from both VEGP and other plants, sample inspections of bulged and overexpanded locations within the tubesheet will be specified elements of SG eddy current inspection. This sample is based on the guidance contained in the SG Examination Guidelines and TS 5.5.9. The inspection plan is expanded according to industry guidelines if necessary due to confirmed degradation (i.e. tube crack indications).

Constraint provided by the hot leg tubesheet precludes tube burst for cracks within the tubesheet. The criteria for tube burst described in Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," Revision 2 dated May 2005 (Reference 3), and NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," dated August 1976 (Reference 4), are satisfied due to the constraint provided by the tubesheet. Through application of the limited hot leg tubesheet inspection scope described herein, the existing operating leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur during a postulated SLB event.

Implementation of this proposed methodology involves limited inspection of the tubes within the tubesheet to depths of 17 inches from the top of the tubesheet on the hot leg side using specialized rotating eddy current probes. The limited tubesheet inspection length of tubing must be demonstrated to be non-degraded below the top of the tubesheet interface on the hot leg side. If cracks are found within the top of tubesheet to 17 inches below the hot leg side top of tubesheet, the tube must be removed from service.

4.0 <u>Technical Analysis</u>

The proposed TS change is intended to preclude unnecessarily plugging tubes in the VEGP Units 1 and 2 SGs. An analysis was performed as technical justification to identify the portion of the tube within the hot leg side tubesheet necessary to maintain structural and leakage integrity for both normal operating and accident conditions. The revised TS requirements will limit inspections to identifying and plugging degradation in this portion of the tubes. The technical justification for the inspection and repair methodology is provided in Westinghouse Electric Company LTR-CDME-06-58-P, "Steam Generator Tube Alternate Repair Criteria for the Portion of the Tube Within the Tubesheet at the Vogtle 1 & 2 Electric Generating Plant for One Cycle Application", July 11, 2006 (Reference 1). The evaluation is based on the use of finite element model structural analyses and a bounding leak rate evaluation based on the change in contact pressure between the tube and the tubesheet between normal operating and postulated accident conditions. The evaluation considered the requirements of the ASME Code, Regulatory Guides, NRC Generic Letters, NRC Information Notices, the Code of Federal Regulations, NEI 97-06, and additional industry requirements.

The following bullets are two of the conclusions of the evaluation:

- The structural integrity of the primary-to-secondary pressure boundary is unaffected by tube degradation of any magnitude below a tube location-specific depth ranging from 2.3 to 7.0 inches depending on the tube leg and bundle zone being considered.
- The accident condition leak rate integrity can be bounded by twice the normal operational leak rate as a result of unlimited degradation below 17 inches from the top of the approximately 21-inch thick tubesheet.

Based on these conclusions a redefinition of the pressure boundary can be effected while still assuring that the structural and leak rate performance criteria would be met during both normal operation and limiting postulated accident conditions. Implementation of the redefinition of the pressure boundary results in the elimination of the need for the inspection of the tubes below a depth on the order of 17 inches from the top of hot leg tubesheet, which includes eliminating the need to inspect the region of the hot leg side SG tubes referred to as the tack expansion including tubing immediately adjacent to the tube-to-tubesheet weld, and the tack expansion transition near the bottom of the tubesheet. The tube-to-tubesheet weld is excluded from the definition of the SG tubing.

The determination of the required engagement depth was based on results from finite element model structural analyses and a steam line break to normal operation comparative leak rate evaluation.

The limited tubesheet inspection criteria were developed for the hot leg side tubesheet region of the VEGP Model F SGs considering the most stringent loads associated with plant operation, including transients and postulated accident conditions. The limited tubesheet inspection criteria were selected to prevent tube burst and axial separation due to axial pullout forces acting on the tube and to ensure that the steam line break (SLB) leakage limits are not exceeded. Reference 1 provides technical justification for allowing tubes with indications that are below 17 inches from the top of the hot leg tubesheet (i.e., within approximately four inches of the tube end on the hot leg side) to remain in service.

The portion of the tube in the tubesheet with the highest safety significance is the length of tube that is engaged in the tubesheet from the secondary face that is required to maintain structural and leakage integrity over the full range of steam generator operating conditions, including the most limiting accident conditions. The evaluation determined that degradation in tubing below the portion of the tube with the highest safety significance does not require repair and serves as the basis for the tubesheet inspection program. The determination of the portion of the tube within the tubesheet with the highest safety significance is based upon evaluation and testing programs that quantified the tube-to-tubesheet radial contact pressure for bounding plant conditions as described in LTR-CDME-06-58-P.

The tube-to-tubesheet radial contact pressure provides resistance to tube pull-out and resistance to leakage during plant operation and transients. Temperature effects and upward bending of the tubesheet due to primary and secondary differential pressure during normal and transient conditions result in the tube-to-tubesheet contact pressure increasing below the neutral plane of the tube sheet. Due to these effects, the tubesheet bore tends to dilate near the top of the tubesheet and constricts the tube near the bottom of the tubesheet.

The hydraulically expanded tube-to-tubesheet joints in Model F SGs are not leak-tight without the tube end weld. Considerations were also made with regard to the potential for primary-to-secondary leakage during postulated faulted conditions. However, the leak rate during postulated accident conditions would be expected to be less than that during normal operation for indications near the bottom of the tubesheet based on the evaluation (Reference 1) which shows that while the driving pressure increases by about a factor of almost two, the flow resistance increases because the tube-to-tubesheet contact pressure also increases. Depending on the depth within the tubesheet, the relative increase in resistance could easily be larger than that of the pressure potential. Therefore, the leak rate under normal operating conditions could exceed its allowed value before the accident condition leak rate would be expected to exceed its allowed value. This approach is termed an application of the "bellwether principle." While such a decrease in the leak rate is expected, the postulated accident leak rate could conservatively be taken to be bounded by twice the normal operating leak rate if the increase in contact pressure is ignored.

Since normal operating leakage is limited by the TS changes proposed in SNC letter NL-06-0124 and by NEI 97-06 to less than 0.10 gpm (150 gpd) throughout one SG in the VEGP Units 1 and 2 SGs, the attendant accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be bounded by 0.20 gpm in the faulted SG which is less than the accident analysis assumption of 0.35 gpm to the affected SG included in Section 15.1.5 of the VEGP Updated Final Safety Analysis Report (FSAR). Hence it is reasonable to omit any consideration of inspection of the tube, tube end weld, bulges/overexpansions or other anomalies below 17 inches from the top of the hot leg tubesheet.

Testing and analyses have shown that tube-to-tubesheet engagement lengths of approximately 2.3 to 7.0 inches were sufficient to maintain structural integrity (i.e., resist tube pull-out resulting from loading considering differential pressures of three times the normal operating pressure difference and considering differential pressures of 1.4 times the limiting accident pressure difference). The variation of the required engagement length is a function of the radial tube location within the tube bundle. Additional conservatism is being added to the minimum structural distances of 2.3 to 7.0 inches by performing sampling inspections to depths of 17 inches below the top of the tubesheet, which traverses below the neutral plane. The increase in contact pressure at this depth significantly increases the tube structural strength and resistance to leakage.

Therefore, the proposed inspection sampling length of 17 inches from the top of the hot leg side tubesheet provides a high level of confidence that the structural and leakage criteria are maintained during normal operating and accident conditions and degradation found in the portion of the tube below 17 inches from the top of the hot leg does not require plugging.

In accordance with the EPRI PWR Steam Generator Examination Guidelines and NEI 97-06, "Steam Generator Program Guidelines" (Reference 3), SNC will implement the following inspection requirements in order to use the limited tubesheet inspection methodology:

1. Perform a 40% minimum inspection of the hot leg side tubes of the two scheduled steam generators using rotating pancake probe (RPC) technology from three inches above the top of the hot leg tubesheet to three inches below the top of the tubesheet. Expand to 100% of the affected SG and 20% of the unaffected SGs in this region only if cracking is found that is not associated with a bulge or overexpansion as described below.

2. Perform an inspection of the hot leg side tubes using RPC technology to a depth of 17 inches below the top of the tube sheet in order to inspect (1) for Unit 1, 100% of bulges and overexpansions in SG 4, and at least 20% of bulges and overexpansions in SGs 1, 2, and 3; and (2) for Unit 2, a 40% sample of bulges and overexpansions in the two scheduled SGs.

    a. Bulge refers to a tube diameter deviation within the tubesheet of 18 volts or greater as measured by bobbin coil probe based on review of the previous cycle bobbin data; and

    b. Overexpansion refers to a tube diameter deviation within the tubesheet of 1.5 mils or greater as measured by bobbin coil probe based on review of the previous cycle bobbin data.

3. If cracking is found in the sample population of bulges or overexpansions, the inspection scope will be increased to 100% of the bulges and overexpansions population for the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet in the affected SG and a 20% sample of each of the unscheduled SGs.

4. If cracking is reported at one or more tube locations not designated as either a top of the tubesheet expansion transition, a bulge, or an overexpansion, an engineering evaluation will be performed. This evaluation will determine the cause for the signal, e.g., some other tube anomaly, in order to identify a critical area for the expansion of the inspection. This expanded inspection will be limited to the identified critical area within 17 inches from the top of the hot leg tubesheet.

SNC will implement the following plugging criteria and acceptance criteria:

- Degradation below 17 inches from the top of hot leg tubesheet is acceptable.
- Degradation within 17 inches from the top of hot leg tubesheet must be plugged.

In summary:

- Reference 1 notes that the structural integrity requirements of NEI 97-06, and RG 1.121, are met by sound tube engagement lengths ranging from approximately 2.3 to 7.0 inches from the top of the tubesheet. The region of the tube below those elevations, including the tube-to-tubesheet weld, is not needed for structural integrity during normal operation or accident conditions. SNC will, however, perform sampling inspections to a depth of 17 inches from the top of the hot leg tubesheet.
- The leak rate during postulated faulted events would be bounded by twice the leak rate during normal operation.
- NEI 97-06 defines the tube as extending from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, but specifically excludes the tube-to-tubesheet weld from the definition of the tube.
- The welds were originally designed and analyzed as the primary pressure boundary in accordance with the requirements of Section III of the 1971 edition of the American Society of Mechanical Engineers (ASME) Code, Summer 1972 Addenda for the VEGP Units 1 and 2 SGs. This proposed license amendment request, in effect, redefines the primary pressure boundary from the hot leg tube end weld to 17 inches below the top of the hot leg tube sheet.
- Section XI of the ASME Code deals with the in-service inspection of nuclear power plant components. The ASME Code (i.e., Editions 1971 through 2004) specifically recognizes that the SG tubes are under the purview of the NRC through the implementation of the requirements of the TS as part of the plant operating license.

5.0 Regulatory Analysis

5.1 No Significant Hazards Consideration

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed 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.

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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) 1.121, "Bases for Plugging Degraded PWR 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.

The hydraulically expanded tube-to-tubesheet joints in Model F SGs are not leak-tight without the tube end weld. Considerations were also made with regard to the potential for primary-to-secondary leakage during postulated faulted conditions. However, the leak rate during postulated accident conditions would be expected to be less than that during normal operation for indications near the bottom of the tubesheet based on the evaluation (Reference 1) which shows that while the driving pressure increases by about a factor of almost two, the flow resistance increases because the tube-to-tubesheet contact pressure also increases. Depending on the depth within the tubesheet, the relative increase in resistance could easily be larger than that of the pressure potential. Therefore, the leak rate under normal operating conditions could exceed its allowed value before the accident condition leak rate would be expected to exceed its allowed value. This approach is termed an application of the "bellwether principle." While such a decrease in the leak rate is expected, the postulated accident leak rate could conservatively be taken to be bounded by twice the normal operating leak rate if the increase in contact pressure is ignored.

Since normal operating leakage is limited by the TS changes proposed in SNC letter NL-06-0124 and by NEI 97-06 to less than 0.10 gpm (150 gpd) throughout one SG in the VEGP Units 1 and 2 SGs, the attendant accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be bounded by 0.20 gpm in the faulted SG which is less than the accident analysis assumption of 0.35 gpm to the affected SG included in Section 15.1.5 of the VEGP Updated Final Safety Analysis Report (FSAR). Hence it is reasonable to omit any consideration of inspection of the tube, tube end weld, bulges/overexpansions or other anomalies below 17 inches from the top of the hot leg tubesheet.

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," 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 tubesheet inspection depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. RG 1.121 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 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 tubesheet inspection depth criteria.

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

5.2 Applicable Regulatory Requirements Criteria

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include technical specifications (TS) as part of the license. The Commission's regulatory requirements related to the content of the TS are contained in Title 10, Code of Federal Regulations (10 CFR), Section 50.36, "Technical specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings and control settings, (2) limiting conditions for operation (LCO), (3) surveillance requirements, (4) design features, and (5) administrative controls. The SG tube inspection requirements are included in the TS in accordance with 10 CFR 50.36(c)(5), "Administrative Controls." As stated in 10 CFR 50.59, "Changes, tests, and experiments," paragraph (c)(1)(i), a licensee is required to submit a license amendment pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," if a change to the TS is required. Furthermore, the requirements of 10 CFR 50.59 necessitate that the NRC approve the TS changes before the TS changes are implemented. SNC's submittal meets the requirements of 10 CFR 50.59(c)(1)(i) and 10 CFR 50.90. RG 1.121 margins against burst are maintained for both normal and postulated accident conditions due to the constraint provided by the tubesheet. NRC Information Notice 2005-09, Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," dated April 7, 2005, provides additional regulatory insight regarding SG tube degradation.

5.3 Environmental Assessment

This amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) as follows:

(i) The amendment involves no significant hazards consideration.

As described above, the proposed change involves no significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed change does not involve the installation of any new equipment, or the modification of any equipment that may affect the types or amounts of effluents that may be released offsite. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupation radiation exposure.

The proposed change does not involve plant physical changes, or introduce any new mode of plant operation. Therefore, there is no significant increase in individual or cumulative occupational radiation exposure.

Based on the above, SNC concludes that the proposed change meets the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.22 relative to requiring a specific environmental assessment by the Commission.

6.0 References

1. Westinghouse Electric Company LTR-CDME-06-58-P, "Steam Generator Tube Alternate Repair Criteria for the Portion of the Tube Within the Tubesheet at the Vogtle 1 & 2 Electric Generating Plant for One Cycle Application", dated July 11, 2006.

2. NRC Information Notice 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," dated April 7, 2005.

3. NEI 97-06, "Steam Generator Program Guidelines," Revision 2, dated May 2005.

4. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," dated August 1976.

5. Southern Nuclear Operating Company Letter NL-06-0124, "Vogtle Electric Generating Plant Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity."

6. Southern Nuclear Operating Company Letter NL-06-0990, "Southern Nuclear Operating Company Response to NRC Questions Regarding the Vogtle Electric Generating Plant Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity."

7. Southern Nuclear Operating Company Letter NL-06-0708, "Request for License Amendment Related to Technical Specification 5.5.9, Steam Generator (SG) Tube Surveillance Program," dated July 20, 2006.

Enclosure 2

Vogtle Electric Generating Plant
Request for Technical Specifications Amendment
Steam Generator Tube Surveillance Program

Markup of Proposed Technical Specification – Revised

5.5.9 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. 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 may be applied as an alternative to the 40% depth based criteria:

1. For Unit 2 during Refueling Outage 11 and the subsequent operating cycle, 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.

2. For Unit 1 during Refueling Outage 13 and the subsequent operating cycle, and for Unit 2 during Refueling Outage 12 and the subsequent operating cycle, degradation identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging.

   For Unit 1 during Refueling Outage 13 and the subsequent operating cycle and for Unit 2 during Refueling Outage 12 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 hot leg tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the 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. 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. For Unit 1 during Refueling Outage 13 and the subsequent operating cycle and for Unit 2 during Refueling Outage 12 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

Enclosure 3

Vogtle Electric Generating Plant
Request for Technical Specifications Amendment
Steam Generator Tube Surveillance Program

Typed Pages for Technical Specification – Revised

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. 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 may be applied as an alternative to the 40% depth based criteria:

1. For Unit 2 during Refueling Outage 11 and the subsequent operating cycle, 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.

2. For Unit 1 during Refueling Outage 13 and the subsequent operating cycle, and for Unit 2 during Refueling Outage 12 and the subsequent operating cycle, degradation identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging.

   For Unit 1 during Refueling Outage 13 and the subsequent operating cycle, and for Unit 2 during Refueling Outage 12 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 hot leg tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

present along the length of the 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. 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. For Unit 1 during Refueling Outage 13 and the subsequent operating cycle, and for Unit 2 during Refueling Outage 12 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic nondestructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

(continued)

5.5 Programs and Manuals (continued)

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1980:

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass ≤ 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1980 at the system flow rate specified below ± 10%.

ESF Ventilation System	Flow Rate
Control Room Emergency Filtration System (CREFS)	19,000 CFM
Piping Penetration Area Filtration and Exhaust (PPAFES)	15,500 CFM

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1980 at the system flow rate specified below ± 10%.

ESF Ventilation System	Flow Rate
CREFS	19,000 CFM
PPAFES	15,500 CFM

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than or equal to the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and greater than or equal to the relative humidity specified below.

ESF Ventilation System	Penetration	RH
CREFS	.2%	70%
PPAFES	10%	95%

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the charcoal adsorbers, and CREFS cooling coils is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flow rate specified below ± 10%.

ESF Ventilation System	Delta P	Flow Rate
CREFS	7.1 in. water gauge	19,000 CFM
PPAFES	6 in. water gauge	15,500 CFM

e. Demonstrate that the heaters for the CREFS dissipate ≥ 95 kW when corrected to 460 V when tested in accordance with ASME N510-1989.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

(continued)

5.5 Programs and Manuals (continued)

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Waste Processing System, the quantity of radioactivity contained in each Gas Decay Tank, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be limited to 10 curies per outdoor tank in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures."

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Gaseous Waste Processing System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas decay tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is limited to ≤ 10 curies per tank, excluding tritium and dissolved or entrained noble gases. This surveillance program provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting concentrations would be less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable

(continued)

5.5 Programs and Manuals

5.5.13 Diesel Fuel Oil Testing Program (continued)

ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits, or an API gravity or specific gravity within limits when compared to the supplier's certificate;

   2. a flash point within limits for ASTM 2D fuel oil, and, if gravity was not determined by comparison with supplier's certification, a kinematic viscosity within limits for ASTM 2D fuel oil; and

   3. a clear and bright appearance with proper color.

b. Other properties for ASTM 2D fuel oil are within limits within 30 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance frequencies.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. a change in the TS incorporated in the license; or

   2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

(continued)

5.5 Programs and Manuals

5.5.14 Technical Specifications (TS) Bases Control Program (continued)

d. Proposed changes that meet the criteria of (b) above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

(continued)

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 MS and FW Piping Inspection Program

This program shall provide for the inspection of the four Main Steam and Feedwater lines from the containment penetration flued head outboard welds, up to the first five-way restraint. The extent of the inservice examinations completed during each inspection interval (ASME Code Section XI) shall provide 100% volumetric examination of circumferential and longitudinal welds to the extent practical. This augmented inservice inspection is consistent with the requirements of NRC Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," November 1975 and Section 6.6 of the FSAR.

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995, as modified by the following exceptions:

1. Leakage rate testing for containment purge valves with resilient seals is performed once per 18 months in accordance with LCO 3.6.3, SR 3.6.3.6 and SR 3.0.2.

2. Containment personnel air lock door seals will be tested prior to reestablishing containment integrity when the air lock has been used for containment entry. When containment integrity is required and the air lock has been used for containment entry, door seals will be tested at least once per 30 days during the period that containment entry(ies) is (are) being made.

3. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified

(continued)

5.5 Programs and Manuals

5.5.17 Containment Leakage Rate Testing Program (continued)

by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC. At the discretion of the licensee, the containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage.

4. A one time exception to NEI 94-01, Rev. 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

Section 9.2.3:	The next Type A test, after the March 2002 test for Unit 1 and the March 1995 test for Unit 2, shall be performed within 15 years.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 37 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.2% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criteria are $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$,

   2) For each door, the leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

5.5 Programs and Manuals (continued)

5.5.18 Configuration Risk Management Program

The Configuration Risk Management Program (CRMP) provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The program applies to technical specification structures, systems, or components for which a risk-informed allowed outage time has been granted. The program shall include the following elements:

a. Provisions for the control and implementation of a Level 1 at power internal events PRA-informed methodology. The assessment shall be capable of evaluating the applicable plant configuration.

b. Provisions for performing an assessment prior to entering the LCO Condition for preplanned activities.

c. Provisions for performing an assessment after entering the LCO Condition for unplanned entry into the LCO Condition.

d. Provisions for assessing the need for additional actions after the discovery of additional equipment out of service conditions while in the LCO Condition.

e. Provisions for considering other applicable risk significant contributors such as Level 2 issues and external events, qualitatively or quantitatively.

5.5.19 Battery Monitoring and Maintenance Program

This program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," of the following:

a. Actions to restore battery cells with float voltage < 2.13 V, and

b Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates.