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SCE&G®
A SCANA COMPANY

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RC-03-0007

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTN: K. R. Cotton

Ladies and Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION (VCSNS)
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
LICENSE AMENDMENT REQUEST - LAR 02-2767
STEAM GENERATORS - ONE-TIME EXCLUSION OF INSPECTION FREQUENCY

Pursuant to 10 CFR 50.90, South Carolina Electric & Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, hereby requests an amendment to Technical Specification (TS) 4.4.5.3a, "Steam Generator Surveillance Requirements." The proposed one-time change revises the steam generator inservice inspection frequency requirements in TS 4.4.5.3.a for V. C. Summer Nuclear Station (VCSNS) immediately after refueling outage RF-12. The change would allow a 58-month maximum inspection interval after two inspections resulting in C-1 classification, rather than a 40 month maximum inspection interval. This change is proposed to eliminate premature/unnecessary steam generator inspections, due to a shortened operating cycle, which will result in significant dose, schedule, and cost savings.

The VCSNS steam generators (SGs) were replaced during RF-8, Fall 1994. The replacement steam generators (RSGs) are the Westinghouse Delta 75 design, which incorporates significant improvements, including thermally treated Alloy 690 tubing. Inspections performed pursuant to this surveillance requirement in RF-9 and RF-10 both resulted in C-1 classification. The latest 100% bobbin inspection of all three VCSNS SGs, performed in RF-12, found no indications of stress corrosion cracking or any other active damage mechanism.

During the VCSNS RF-12, October, 2000, following the fourth cycle of operation after replacement, 100% of the steam generator tubes were inspected full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) with eddy current bobbin coil. Additionally, a sample of 322 tubes (10%) in SG B were inspected at the hot leg top of tubesheet area (+3/-3 inches), 14 low row u-bends in SG C with a Rotating coil probe, and approximately 65 areas of special interest with the Rotating coil probe. No defective or degraded tubes were indicated.

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Because of the identification of a crack in Loop "A" Reactor Coolant System (RCS) hot leg pipe, RF-12 was extended approximately four months for repairs. The following operating period for cycle 13 was reduced to 13 months, with RF-13 commencing in April, 2002. Cycle 14 operating period is anticipated to be 16 months, from June 4, 2002 to October 18, 2003, with RF-14 commencing on October 18, 2003. At that time, 36 calendar months will have elapsed since the previous inspection, but only 29 Effective Full Power Months (EFPM). An inspection performed at that time would be premature, based on the excellent condition of the RSGs.

These inspection results, along with the shortened cycle 13, improved RSG design, and industry experience with thermally treated Alloy 690 tubing, provide the basis for proposing this one-time extension of the maximum inspection interval.

SCE&G is aware of ongoing industry and NRC work regarding a generic license change (GLCP) to this section of Technical specifications. This request complies with proposed Technical Specifications changes to date, and with lengthened inspection intervals proposed in EPRI Steam Generator Inspection guidelines, Rev. 6. We believe that our site-specific thermally treated Alloy 690 experience, and overall industry experience with thermally treated Alloy 690 tubes provides an acceptable approval basis for our one-time request.

Attachment 1 to this letter provides the No Significant Hazards Determination and Attachment 2 provides the TS page marked up with the proposed change. Attachment 3 provides the retyped TS page. There are no changes proposed to the Bases for TS 3/4.4.5.

The VCSNS Plant Safety Review Committee and the Nuclear Safety Review Committee have reviewed and approved the proposed change. SCE&G has notified the State of South Carolina in accordance with 10CFR50.91(b).

SCE&G requests approval of the proposed change prior to March. 31, 2003 to support postponing SG tube inspections currently planned for RF-14, which is scheduled to begin in October 2003.

If you have any questions or require additional information regarding this proposed amendment to TS 4.4.5.3.a, please contact Mr. Melvin N. Browne at (803) 345-4141.

I certify under penalty of perjury that the foregoing is true and correct.

1/14/03
Executed on

Stephen A. Byrne

JT/SAB/dr

Enclosures:
Evaluation of the proposed change
Attachment(s): 3
1. Proposed Technical Specification Change - Mark-up
2. Proposed Technical Specification Change - Retyped
3. List of Regulatory Commitments

c: N. O. Lorick
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RTS (LAR 02-2767)
File (813.20)
DMS (RC-03-0007)

**Subject: LICENSE AMENDMENT REQUEST - LAR 02-2767
STEAM GENERATORS – ONE-TIME EXCLUSION OF INSPECTION FREQUENCY**

1.0 DESCRIPTION

South Carolina Electric & Gas Company (SCE&G) proposes an amendment to revise the V. C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) Surveillance Requirements (SR) to revise SR 4.4.5.3.a.

NEI 97-06 and the associated EPRI guidelines are the documents that provide consistency in managing steam generator (SG) programs. These guidelines have been developed through NEI and EPRI to provide methods to ensure accurate assessment of SG tube integrity, and to extend the reliable, cost effective, operating life of the steam generators. To this end, VCSNS has committed to the above as part of an industry wide effort of SG maintenance.

Inspections of the SG tubing ensure that the structural integrity of this portion of the reactor coolant system is maintained. In-service inspections of SG tubes are essential in order to maintain surveillance of tube condition in the event there is mechanical damage or progressive damage due to design or manufacturing errors or in-service conditions such as wear or corrosion. In-service inspection of the tubes also provides the means to characterize the condition such that corrective action can be taken in the case of identified tube degradation.

VCSNS current Technical Specifications are more rigorous for in-service inspection frequency requirements of the steam generators than the NEI guidance. Based on the superior performance of the steam generators to date and the following discussions, VCSNS requests a one-time amendment to Technical Specification 4.4.5.3.a to allow a 58-month maximum inspection interval after two inspections resulting in a C-1 classification, rather than a 40-month inspection interval. This change is proposed to eliminate premature/unnecessary steam generator inspections due to a shortened operating cycle, which will result in significant dose, schedule, and cost savings.

2.0 PROPOSED CHANGE

Specifically the proposed changes would revise the following:

Currently, TS 4.4.5.3.a states, in part:

> If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

The proposed change is to add a note following the paragraph cited above:

> Note: A one-time inspection interval of once per 58 months is allowed for the inspection performed immediately following refueling outage RF-12.

3.0 BACKGROUND

VCSNS replaced three Westinghouse model D3 Steam Generators with replacement model Delta 75 Steam Generators in the fall of 1994. One pre-service examination and 4 inservice examinations, which are described in this submittal, have been performed since replacement and have confirmed that no degradation of tubing has occurred. Degradation assessments have further concluded that no active degradation mechanisms are present.

VCSNS has performed inspections on the replacement Steam Generators that exceed the referenced Technical Specification surveillance requirements. The station's Steam Generator inspection program also complies with industry requirements outlined in NEI 97-06 and supporting EPRI guidelines.

4.0 TECHNICAL ANALYSIS

4.1 Steam Generator Inspection History

The VCSNS steam generators were replaced during Refuel 8 (fall 1994). The three replacement steam generators are the Westinghouse Delta 75 design. Each Steam Generator contains 6307 thermally treated alloy 690 tubes with an outside diameter of 0.6875 inches and a nominal wall thickness of 0.040 inches.

Inspections were performed pursuant to the technical specifications surveillance requirements in the four consecutive outages following replacement, Refuel 9, 10, 11, and 12, and found no evidence of any degradation. Each resulted in Technical Specifications category C-1 classification. The latest inspection of all three steam generators, performed in Refuel 12 (fall 2000), found no indications of stress corrosion cracking or any other damage mechanism. The scope of this latest inspection is described below.

One hundred percent of all SG tubes were full length eddy current inspected using bobbin coil in Refuel 12. In detail, SG A was examined full length with a bobbin probe; SG B was examined full length bobbin with 322 tubes examined with a rotating coil probe hot leg top of tubesheet; SG C was examined full length with a bobbin probe and a U-bend study was completed on 14 SG C Row 1 tubes comparing a mid frequency to a high frequency rotating coil probe with calibration and subsequent validation of the VCSNS site technique. These inspections were performed in accordance with guidance specified by Appendix H of EPRI SG Examination Guidelines.

Results for Refuel 12 eddy current after 5.4 effective full power years were that no indications of stress corrosion cracking were present. This data supports the laboratory test results of Alloy 690 TT tubing which indicates superior performance over previous mill annealed Inconel 600 tubing. A full description of the Refuel 12 inspection results was reported to the NRC pursuant to Technical Specifications Section 4.4.5.5.

Because of the identification of a crack in A (Alpha) reactor coolant system hot leg piping, Refuel 12 was extended approximately 4 months. This resulted in Cycle 13 being reduced to 13 operating months, with Refuel 13 commencing on April 20, 2002. No steam generator inspections were performed in Refuel 13. Cycle 14 is expected to be 16 operating months commencing on June 4, 2002 and completing on October 18, 2003. At that time, 36 calendar months will have elapsed since the previous inspection, but only 29 effective full power months (EFPM) would be realized. A VCSNS steam generator inspection performed at that time would be premature based on the excellent condition of the steam generators. Cycle 15 is conservatively estimated to be 17 operating months following a 1-month refuel outage. Extending the inspection interval would allow VCSNS to operate an additional 17 EFPM to the spring 2005 refueling, for a total of 46 EFPM, or 54 calendar months. This extension of the Technical Specifications limit is supported as described below.

4.2 Steam Generator Design Improvements

The Delta 75 SG design employs the key characteristics of the proven Westinghouse Model F design. The most reliable features carried over from Model F include the use of previously mentioned Alloy 690 thermally treated tube material, Type 405 stainless steel tube support plate material, trifoil-shaped broached flat contact tube support holes, hydraulically expanded tube sheet joints, minimum gap U-bend construction, and foreign material exclusion in the design of the feedwater distribution headers. In addition, stress relief was required for all tubes in the 17 innermost rows.

The corrosion resistance of thermally treated Alloy 690 has not only been proven by years of extensive laboratory testing, but has been proven in actual plant operation at plants of the same steam generator replacement vintage as VCSNS such as D.C. Cook 2 and Indian Point 3. To date, none of these units have reported any corrosion degradation. Also, since test conditions have been more hostile than actual plant service conditions, expectations for Alloy 690 tubing in the field should result in improved performance. VCSNS will continue to track the evolution of Alloy 690 tubing for degradation concerns through association with EPRI tools such as the SG Degradation Database and related communication. Industry experience with Alloy 690 will be factored into VCSNS SG inspection planning and integrity assessments.

Current industry data shows that some plants with replacement steam generators have experienced mechanical degradation (wear) at tube support plates, anti-vibration bars, and wear related to loose parts. Wear is the volumetric removal of material caused by the mechanical action of one material in contact with another and is associated with secondary side flow that induces vibration and subsequent contact between the objects.

The potential for loose parts entering the steam generators is minimized by the design of the feedwater ring spray nozzle assemblies, each consisting of a series of 0.25 inch diameter outlet holes, which function to trap potential foreign objects that may be introduced from feedwater systems. The 0.25 inch dimension is smaller than the tube pitch, thus, if parts are small enough to pass through these holes, they will also pass between the tubes and be transported into the low velocity regions of the bundle. There they will have the least potential to cause service induced wear. One possible loose part was reported in Refuel 11, in SG "A" at the top of the tube sheet. The presence of the item was evaluated, and the item was left in place until the next scheduled shutdown. During Refuel 12 the loose part was identified (a piece of wire ≈ 0.5 inches long) and removed with no indications present on any of the surrounding tubes after the part was removed. No evidence of additional loose parts was identified in the tube bundle through FOSAR inspections and eddy current, and no indication of loose part damage to SG tubing was observed. As far as minimizing loose parts during maintenance, VCSNS is committed to a strict FME program when any SG closure is opened to the environment.

Eddy current examinations thus far have indicated that wear is not present in the VCSNS steam generators. Periodic future inspections will continue to pay close attention to this degradation mode, since it is the only potential degradation mode currently identified for VCSNS.

The operational assessment performed following Refuel 12 provides additional support for the VCSNS extension request. In this assessment, it was assumed that the largest through wall indication (9%) found in Refuel 12, was 0% at baseline inspection. (Actual analysis confirmed that this indication was, in fact, present at baseline and had not grown). The calculated growth rate was 1.74% TW/EFPY. At this growth rate, the largest indication left in service would not reach structurally significant size (57.27% TW) until 27.7 EFPY, or approximately 18.5 cycles from Refuel 12, assuming 18 month cycles.

VCSNS implements a Steam Generator program that complies with the current revision to NEI 97-06 and supporting EPRI Steam Generator Guidelines. The current revision to the EPRI Steam Generator examination guideline, Revision 6, issued in October 2002, allows Alloy 690 replacement steam generators to be inspected at an initial interval (following the first in-service examination) of 144 EFPM, with a 50% inspection in all Steam Generators required at or before 72 EFPM. This request for extension of the surveillance interval (from 29 to 46 EFPM) falls well within this guideline. Also, this extension is preceded by 3 partial and 1 full inspection in 4 consecutive cycles. For information, the previous Revision 5 to the EPRI Steam Generator guideline allowed a 60 EFPM rolling time frame to achieve 100% tube inspection of each type of repair, regardless of material type. Revision 5, however, provided an additional limitation that no steam generator should operate more than two cycles between inspections.

There is also an ongoing industry effort through NEI to revise this portion of Technical Specifications via a generic license change (GLCP). As stated earlier, VCSNS is requesting an extension of the more restrictive Technical Specification frequency of steam generator inspection criteria. While this is a relaxation of certain current requirements, the commitment to other performance criteria such as more restrictive primary to secondary leakage and rigorous condition monitoring and operational assessments that are currently not required by Technical Specifications provide added support for this request. Although the entire GLCP effort is not complete, the proposed change also falls within the criteria addressed in the GLCP.

In summary, the following bases provide support for assuring that VCSNS steam generator performance criteria will be maintained through the extended operating period.

- Improved quality of materials.
- Improved design features.
- Results of examinations performed to date.
- Reduced operating time due to extended outages.
- Operational Assessment analysis.
- VCSNS Steam Generator Management Program controls.
- Changes in industry examination guidelines for Alloy 690 replacement steam generators.

4.3 Steam Generator Performance

Background

The steam generator tubes in pressurized water reactors have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary and as such, are relied upon to maintain the primary system pressure and inventory. As part of the RCS boundary, the tubes are unique in that they are also relied upon as a heat transfer medium between the primary and secondary systems such that heat may be removed from the primary system. Additionally, the steam generator tubes also isolate the radioactive fission products in the primary coolant from the secondary system.

Steam generator tube integrity is necessary in order to satisfy the above safety functions. Tube degradation related to corrosion, denting, and wear can impair tube integrity if they are not managed effectively. Maintaining this integrity ensures that the tubes are capable of performing their safety function consistent with the VCSNS licensing basis.

Frequency of Verification of Tube Integrity

The existing Technical Specification base inspection intervals classifies the condition of the steam generator into one of three categories based on the overall results of the previous inspection. The surveillance frequency described by the current VCSNS steam generator inspection program is performance based and is dependent on the location and severity of active degradation mechanisms and their anticipated growth rate.

Tube Sample Selection

The existing Technical Specification bases tube selection on steam generator conditions and industry and plant experience. This change to a performance-based program as defined in the VCSNS Steam Generator Management Program requires that a degradation assessment be performed before inspections and references EPRI Guidelines for details on tube selection. This approach is dependent on actual steam generator conditions and operational experience. This method is more rigorous than the existing Technical Specification.

Inspection Techniques

The VCSNS Steam Generator Management Program requires the performance of a degradation assessment prior to steam generator inspections to identify active and potential degradation modes and mechanisms and NDE techniques that are effective in detecting their existence. Tube inspection techniques are chosen to reliably detect flaws that may progress during an operating cycle beyond performance criteria limits. Identification of damage not predicted in the Degradation Assessment requires revision to the Degradation Assessment and initiation of corrective actions to address the identified condition.

Performance Criteria

The performance criteria used for VCSNS steam generators are based on structural integrity, accident induced leakage, and operational leakage. The structural and accident induced leakage criteria were developed deterministically and are consistent with the VCSNS licensing basis. The operational leakage criterion was based on providing added assurance against tube rupture at normal operating and accident conditions.

The Structural Integrity performance criterion is:

> Steam generator tubing shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation and a safety factor of 1.4 under all design basis accidents, including any additional loading combinations required by existing design and licensing basis.
>
> The structural integrity performance criterion is based on ensuring that there is reasonable assurance that a steam generator tube will not burst during normal or postulated accident conditions.

The Accident-Induced Leakage performance criterion is:

> The primary to secondary accident induced leakage rate for the limiting design basis accident, other than a steam generator tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rate for an individual steam generator. Leakage is not to exceed 1 gpm per steam generator, except for specific types of degradation at specific locations where the NRC has approved greater accident-induced leakage as part of the VCSNS licensing basis. Exception to the 1-gpm limit can be applied if approved by the NRC in conjunction with approved Alternate Repair Criteria.
>
> Primary to secondary leakage is a factor in the dose releases outside containment resulting from a limiting design basis accident. The potential primary to secondary leak rate during postulated design basis accidents must not exceed the offsite radiological dose consequences required by 10 CFR Part 100 guidelines or the radiological consequences to control room personnel required by GDC-19.

The Operational Leakage performance criterion is:

> The RCS operational primary to secondary leakage through any one steam generator shall be limited to 150 gallons per day.
>
> Plant shutdown should commence if primary to secondary leakage exceeds 150 gallons per day at room temperature conditions from any one steam generator.

The above Performance Criterion are controlled through the VCSNS Steam Generator Management Program. Any future change in any of the performance criteria will require prior NRC approval.

Repair Criteria

VCSNS steam generator tubes will be repaired as follows:

- Tubes that do not meet the performance criteria will be plugged.
- Tubes that are pressure tested or the location of tubes removed for metallurgical examination will be plugged.
- Tubes identified with a damage form or mechanism for which no depth sizing capability exists shall be plugged. In certain cases (e.g., loose parts wear), technical justification may be developed to justify leaving some small indications in place.
- Tubes with defects (indications of degradation which exceed the repair limits) as defined in the Degradation Assessment shall be plugged.
- New repair methods shall be reviewed and approved by the NRC.

Exceeding Performance or Repair Criteria

VCSNS will monitor steam generator performance as described in the Steam Generator Management Program. This includes meeting the performance criteria during all MODES of plant operation as well as during steam generator inspections and subsequent assessments.

Failure to meet a Performance Criterion or Failure to implement a plugging or repair will require operational assessments and NRC reporting as described in the Steam Generator Management Program. VCSNS will follow NEI 97-06 Guidelines for reporting requirements, which is more rigorous than the existing Technical Specification reporting requirement.

4.4 Conclusion

This Technical Specification change does not affect the design of the steam generators, their method of operation, or primary coolant chemistry control. The change does not adversely impact any previously evaluated design basis accident.

A steam generator tube rupture (SGTR) event is a VCSNS analyzed event. In this analysis, a bounding primary to secondary leakage rate equal to the operational leakage rate limit in the Technical Specifications plus the leakage rate associated with a double-ended rupture of a single tube is assumed.

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of the 10 CFR Part 100 limits following a SGTR accident in conjunction with an assumed primary to secondary leakage rate of 1 gallon per minute. For other design basis accidents such as main steam line break or under LOCA conditions, the tubes are assumed to retain their structural integrity (i.e., assumed not to rupture). The present Technical Specification limits of steam generator leakage are assumed in these analyses.

For accidents that do not involve fuel damage, the reactor coolant specific activity levels are assumed at the Technical Specification values. For accidents that do involve fuel damage, the primary coolant specific activity values are a function of the accident conditions. None of the accident assumptions are affected by this change.

Based on the above evaluation, the proposed change does not affect the consequences of a SGTR or any other design basis accident.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

South Carolina Electric & Gas Company (SCE&G) has evaluated the proposed changes to the VCSNS TS described above against the significant Hazards Criteria of 10CFR50.92 and has determined that the change does not involve any significant hazard. The proposed change is part of a steam generator program that defines a performance-based approach to maintaining steam generator tube integrity. The Steam Generator Management Program includes performance criteria that define the basis for steam generator operability and provide reasonable assurance that the steam generator will remain capable of fulfilling its safety function of maintaining reactor coolant pressure boundary integrity. The existence of a Steam Generator Management Program is more rigorous than existing Technical Specifications. The following is provided in support of this conclusion.

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

   Response: No.

   The proposed one-time extension of the Technical Specification inspection interval does not involve changing any structure, system or component or affect plant operations. It is not an initiator of any accident and does not change any FSAR safety analyses. As such, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

   Probability of an Accident

   The VCSNS Steam Generator Management Program includes provisions that are more rigorous than existing Technical Specification requirements. The topics addressed by the program include:

   - Steam generator performance criteria, including a reduced operational leakage limit.
   - Steam generator repair criteria and repair methods.
   - Steam generator inspections that include Degradation Assessments, Condition Monitoring Assessments, and Operational Assessments.
   - NDE technique requirements.

   The results of the above program requirements demonstrated that all performance requirements were met during Refuel 12.

Consequences of an Accident

The consequences of design basis accidents are, in part, functions of the specific activity in the primary coolant and the primary to secondary leakage rates resulting from an accident. Therefore, limits are included in the Technical Specifications for operational leakage and for specific activity in the reactor coolant to ensure the plant is operated in its analyzed condition.

The VCSNS program requires a 150-gallon per day per steam generator limit for leakage prior to an accident. This limit is a reduction in the current Technical Specification value. The post accident leak rate remains at the same value assumed by the accident analysis (1 gallon per minute). Since the new operational leakage limit is more conservative than the existing value, it will not increase the likelihood or consequences of an accident.

In consideration of the above, past 100% eddy current results after 5.4 EFPY of operation, and the current leak free condition of the steam generators, extending the tube inspection frequency does not involve a significant increase in the consequences of a previously evaluated accident.

Summary

The proposed change does not affect the design of the steam generators, their method of operation, or primary coolant chemistry controls. The change does not adversely impact any other previously evaluated design basis accident.

Therefore, the change does not affect the consequences of a SGTR or any other accident.

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

Response: No.

The proposed one-time extension of the Technical Specification inspection interval does not involve changing any structure, system or component or affect plant operations. It is not an initiator of any accident and does not change any FSAR safety analyses.

Primary to secondary leakage that may be experienced during plant conditions is expected to remain within current accident analysis assumptions.

The proposed change does not affect the design of the steam generators, their method of operation, or primary coolant chemistry controls. In addition, the change does not impact any other plant system or component.

Therefore, the change does not create the possibility of a new or different type of accident or malfunction from any accident previously evaluated.

3. Does this change involve a significant reduction in margin of safety?

Response: No.

The steam generator tubes are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system pressure and inventory. As part of the RCS boundary, the tubes are unique in that they are also relied upon as a heat transfer medium between the primary and secondary systems such that heat may be removed from the primary system. Additionally, the steam generator tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of the steam generator is maintained by ensuring the integrity of its tubes.

Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. Extending the tube inspection frequency will not alter the design function of the steam generators. Previous inspections conducted during Refuel 12 demonstrate that there is no active tube damage mechanism. The improved design of the Model Delta 75 generator also provides reasonable assurance that leakage is not likely to occur over the next operating period.

For the above reasons, the margin of safety is unchanged and overall plant safety will be maintained by the proposed Technical Specification revision.

Pursuant to 10 CFR 50.91, the preceding analyses provide a determination that the proposed Technical Specification change poses no significant hazard as delineated by 10 CFR 50.92.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," 1, 2,4,14, 30,31, and 32, define requirements for the reactor coolant pressure boundary (RCPB) with respect to structural and leakage integrity. Steam generator tubing and tube repairs constitute a major fraction of the RCPB surface area. Steam generator tubing and associated repair techniques and components, such as tube plugs, must be capable of maintaining reactor coolant inventory and pressure.

General Design Criteria 19 of 10 CFR 50, Appendix A, defines requirements for the control room and for the radiation protection of the operators working within it. Accidents involving the leakage or burst of steam generator tubing comprise a challenge to the habitability of the control room. Steam generator tubing and associated repair techniques and components, such as tube plugs, must be capable of maintaining reactor coolant inventory and pressure in order to prevent excessive leakage and the resulting radiation doses to the control room operator.

10 CFR 50, Appendix B, establishes quality assurance requirements for the design, construction, and operation of safety related components. The pertinent requirements of this appendix apply to all activities affecting the safety related function of steam generators that include inspection, testing, operating, and maintaining.

10 CFR 50.65, the Maintenance Rule, classifies steam generators as risk significant components because they are relied upon to remain functional during and after design basis events. Steam generators are monitored under a (2) of the Maintenance Rule against industry established performance criteria. If performance criteria are not met, a cause determination shall be done and the results evaluated to determine if goals should be established per a (1) of the Maintenance Rule.

10 CFR 100, Reactor Site Criteria, establishes the reactor-siting criteria with respect to the risk of public exposure to the release of radioactive fission products. Accidents involving the leakage or burst of steam generator tubing may comprise a challenge to containment and therefore involve increased risk of radioactive release. Steam generator tubing and associated repair techniques and components, such as tube plugs, must be capable of maintaining reactor coolant inventory and pressure in order to prevent excessive leakage.

VCSNS Technical Specifications include a requirement to shut the plant down when primary to secondary leakage exceeds an established threshold.

VCSNS complies with the above regulatory requirements.

5.2.1 Regulations

The regulatory basis for TS 4.4.5, "Steam Generators," is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

5.2.2 Design Bases (FSAR)

FSAR Chapter 5 describes the steam generator tubes and tube sheet as forming the pressure boundary across which heat is transferred and which is designed to prevent or control the transfer of activity within the core to the secondary system.

FSAR Appendix 3A discusses the SCE&G position on Regulatory Guide 1.83 and the exceptions applied to the recommendations of the Regulatory Guide.

5.2.3 Approved Methodologies

Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes", Revision 1, July 1976, discusses programs that are acceptable for determining steam generator tube integrity. This guide was utilized in the formation of the steam generator inspection program at VCSNS.
The VCSNS steam generator inspection program has been updated in accordance with the guidelines of NEI 97-06, an industry initiative related to steam generator tube integrity.

5.2.4 Analysis

The Acceptance Criteria for the steam generator tube inspections are described in Regulatory Guide 1.83, Revision 1, July 1976.

These criteria were established to reduce the probability and consequences of steam generator tube failures through periodic inservice inspection for early detection of defects and deterioration.

5.2.5 Conclusion

The technical justification performed by SGE&G demonstrates that the proposed amendment has no impact on the design basis function of the steam generator and its associated tubes. The change represents a preference of extending an inspection based on a shorter operating interval and previous inspection results. Therefore, the proposed License amendment is in compliance with previously established regulatory requirements.

The one-time extension to the inspection frequency requirements allows synchronization with respect to the schedule prescribed by NEI 97-06 guidelines. Therefore, the proposed License amendment is in compliance with GDC 32.

6.0 ENVIRONMENTAL CONSIDERATION

SCE&G has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20 (Reference 7), or would change an inspection or surveillance requirement. SCE&G has evaluated the proposed change and has determined that the change does not involve, (i) a significant hazards consideration, (ii) a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. As discussed above, the proposed changes do not involve a significant hazards consideration and the analysis demonstrates that the consequences of a steam generator tube rupture are well within design boundaries. Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51 (Reference 6), specifically 10 CFR 51.22(c)(9). Therefore, pursuant 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

7.0 REFERENCES

1. FSAR Section(s) 5 and Appendix 3A
2. 10 CFR 50, Appendix A, GDC
3. 10 CFR 50.46
4. 10 CFR 50, Appendix B
5. 10 CFR 100
6. 10 CFR 51
7. 10 CFR 20

ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

Attachment to License Amendment No. XXX
To Facility Operating License No. NPF-12
Docket No. 50-395

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 Pages	Insert Pages
3/4 4-13	3/4 4-13

SCE&G -- EXPLANATION OF CHANGES

Page	Affected Section	Bar #	Description of Change	Reason for Change
3/4 4-13	4.4.5.3.a	1	Add * at end to identify footnote.	To note exemption statement to be applied.
	4.4.5.3.a	2	Add footnote	To identify a one-time extension of inservice inspection of steam generator tubing.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection after the steam generator replacement shall be performed after at least 6 Effective Full Power Months from the time of the replacement but within 24 calendar months of initial criticality after the steam generator replacement. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.*

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A main steam line or feedwater line break.

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* A one time inspection interval of once per 58 months is allowed for the inspection performed immediately following refueling outage RF-12.

ATTACHMENT II

PROPOSED TECHNICAL SPECIFICATION CHANGES (RETYPED)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection after the steam generator replacement shall be performed after at least 6 Effective Full Power Months from the time of the replacement but within 24 calendar months of initial criticality after the steam generator replacement. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.*

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or feedwater line break.

* A one-time inspection interval of once per 58 months is allowed for the inspection performed immediately following refueling outage RF-12.

ATTACHMENT III

LIST OF REGULATORY COMMITMENTS

There are no regulatory commitments created due to this License Amendment Request. The proposed change allows a frequency extension to an existing inspection commitment.