SCE&G
A SCANA COMPANY

August 25, 2000
RC-00-0291

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U. S. Nuclear Regulatory Commission
Washington, DC 20555

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
TECHNICAL SPECIFICATION CHANGE REQUEST - TSP 99-0299
CONTAINMENT STRUCTURAL INTEGRITY, REVISION

Reference: Gary J. Taylor Letter to Document Control Desk, RC-99-0216, dated January 5, 2000

South Carolina Electric & Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, hereby submits a revised request for amendment to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS). This request is being submitted pursuant to 10 CFR 50.90.

This request proposes to change Technical Specification Section 3/4 6.1.6, including its bases and to add Section 6.8.4.h. The proposed changes support the new requirements of 10CFR50.55a which require licensees to update their Containment Vessel Structural Integrity Programs to incorporate the provisions of ASME Section XI, subsection IWL (1992 Edition with 1992 Addenda) and the five additional provisions found in 10CFR50.55a(b)(2)(viii).

The TS change request is contained in the following attachments:

Attachment I	Explanation of Changes Summary Marked-up Technical Specification Pages Revised Technical Specification Pages
Attachment II	Safety Evaluation
Attachment III	No Significant Hazards Evaluation Environmental Impact Considerations
Attachment IV	Description Of The Virgil C. Summer Nuclear Station Containment Inservice Inspection Program

A047

This Technical Specification Change Request is similar to requests submitted by Commonwealth Edison on June 17, 1997 for the Byron Nuclear Power Station, Units 1 and 2 (Facility Operating Licenses NPF-37 and NPF-66; TAC NOS. M99174 and M99175) and for the Braidwood Nuclear Power Station, Units 1 and 2 (Facility Operating Licenses NPF-72 and NPF-77; TAC NOS. M99176 and M99177), and by Entergy Operations, Inc. on April 9, 1999 and July 14, 1999 for Arkansas Nuclear One - Unit 1 (Facility Operating License DPR-51).

Details of this request have been discussed with the NRC Project Manager on several occasions in June and July of this year. Final wording was sent to the Project Manager via electronic communication on July 25, 2000.

This request is being revised to address comments made by the technical reviewer concerning the reporting of any significant degradation relative to containment operability. This revision provides the following statement to B 3/4.6.1.6 and to new section 6.8.4.h.:

> "In addition, any significant degradation which seriously challenges containment operability found during the inspection shall be reported to the NRC in accordance with Technical Specification 6.9.2 within 30 days. The report shall include the description of degradation, operability determination, root cause determination, and corrective actions taken."

This revision presents no new or different technical changes to the proposed amendment which has been reviewed and approved by the Plant Safety Review Committee and the Nuclear Safety Review Committee. Also, this revision has no impact on the Safety Evaluation or No Significant Hazards Determination previously performed for the original request.

There are no other outstanding changes to the subject Technical Specification sections.

As discussed with the NRC Project Manager, SCE&G requests approval of this proposed change by September 1, 2000. Approval of this Technical Specification change is needed to support the next tendon inspection scheduled for the Fall of 2000.

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

If you have any questions regarding this request or require additional information, please contact Mr. Jim Turkett at (803) 345-4047 or Mr. Donald Jones at (803) 345-4480.

Very truly yours,

KW Nettles for

Stephen A. Byrne

JT/SAB/dr
Attachments

c: N. O. Lorick
N. S. Carns
T. G. Eppink (without Attachments)
R. J. White
L. A. Reyes
K. R. Cotton
NRC Resident Inspector

P. Ledbetter
J. B. Knotts, Jr.
T. P. O'Kelley
RTS (TSP 99-0299)
File (813.20)
DMS (RC-00-0291)

STATE OF SOUTH CAROLINA :
: TO WIT :
COUNTY OF FAIRFIELD :

I hereby certify that on the 25 day of August 2000, before me, the subscriber, a Notary Public of the State of South Carolina personally appeared Kenneth W. Nettles, being duly sworn, and states that he has signature authority for the Vice President, Nuclear Operations of the South Carolina Electric & Gas Company, a corporation of the State of South Carolina, that he provides the foregoing response for the purposes therein set forth, that the statements made are true and correct to the best of his knowledge, information, and belief, and that he was authorized to provide the response on behalf of said Corporation.

WITNESS my Hand and Notarial Seal

Notary Public

My Commission Expires

JANUARY 17, 2010

Date

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revision is indicated by a marginal line.

Remove Page	Insert Page
3/4 6-8	3/4 6-8
3/4 6-9	---
3/4 6-10	---
B 3/4 6-2	B 3/4 6-2
B 3/4 6-2a	B 3/4 6-2a
6-12c	6-12c

SCE&G -- EXPLANATION OF CHANGES

Page	Affected Section	Bar #	Description of Change	Reason for Change
3/4 6-8	3.6.1.6	1	Reworded LCO to replace the specific requirements that describe the containment operability with a statement that the structural integrity of the containment is required to be OPERABLE.	To relocate the tendon operability criteria to a licensee controlled program which meets the criteria of 10CFR50.55a, including ASME Section XI, Subsection IWL.
	ACTION (previously numbered as 3.6.1.6.a and 3.6.1.6.b)	2	Deleted ACTIONS a. and b. and revised to permit 1 hour to restore containment operability or require a plant shutdown.	To relocate the tendon operability criteria to a licensee controlled program which meets the criteria of 10CFR50.55a, including ASME Section XI, Subsection IWL.
	4.6.1.6.1 4.6.1.6.2 4.6.1.6.3	3	[1] Removed the details of the tendon surveillance inspection. [2] Moved 4.6.1.6.2 and 4.6.1.6.3 from Page 3/4 6-10 to Page 3/4 6-8.	[1] To relocate the tendon operability criteria to a licensee controlled program which meets the criteria of 10CFR50.55a, including ASME Section XI, Subsection IWL. [2] Due to Repagination process.
3/4 6-9	4.6.1.6.1		Removed the details of the tendon surveillance inspection. Deleted page.	To relocate the tendon operability criteria to a licensee controlled program which meets the criteria of 10CFR50.55a, including ASME Section XI, Subsection IWL.

Page	Affected Section	Bar #	Description of Change	Reason for Change
3/4 6-10	4.6.1.6.1		Removed the details of the tendon surveillance inspection. Deleted Page.	To relocate the tendon operability criteria to a licensee controlled program which meets the criteria of 10CFR50.55a, including ASME Section XI, Subsection IWL.
	4.6.1.6.2		Deleted requirements of TS 4.6.1.6.2.	To relocate the tendon operability criteria to a licensee controlled program which meets the criteria of 10CFR50.55a, including ASME Section XI, Subsection IWL.
	4.6.1.6.3		Relocated to Page 3/4 6-8.	Due to repagination process.
B3/4 6-2	B3/4.6.1.6	1	Changed “Reactor Building” to “Containment”.	Changed subtitle to match TS section title.
		2	Revised BASES to be consistent with the requirements of 10CFR50.55a.	To relocate the tendon operability criteria to a licensee controlled program which meets the criteria of 10CFR50.55a, including ASME Section XI, Subsection IWL.
B3/4 6-2a	B3/4.6.1.6	1	...continuation from 3/4.6.1.6 on previous page.	To relocate the tendon operability criteria to a licensee controlled program which meets the criteria of 10CFR50.55a, including ASME Section XI, Subsection IWL.
6-12c	6.8.4.h	1	Added description of the Containment Inservice Inspection Program in new section 6.8.4.h.	To relocate the tendon operability criteria to a licensee controlled program which meets the criteria of 10CFR50.55a, including ASME Section XI, Subsection IWL.

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be ~~maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.~~ OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

INSERT 'A'

~~a. With the structural integrity of the containment not conforming to the requirements of Specification 4.6.1.6.1.b, perform an engineering evaluation of the containment to demonstrate the acceptability of containment tendons within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~b. With the structural integrity of the containment otherwise not conforming to the requirements of Specification 4.6.1.6, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days after completion of the inspection describing the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective actions taken.~~

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 The structural integrity of the containment ~~tendons~~ shall be demonstrated ~~at the end of one, three and five years following the initial containment structural integrity test and at five year intervals thereafter. The structural integrity of the tendons shall be demonstrated by:~~ IN ACCORDANCE WITH THE CONTAINMENT INSERVICE INSPECTION PROGRAM.

~~a. Determining that a representative sample* of at least 15 tendons (4 dome, 5 vertical, and 6 hoop) each has a lift off force of greater than or equal to 95% of its Base Value. If the lift off force of a selected tendon in a group lies between the 95% Base Value and 90% of the Base Value, one tendon on each side of this tendon shall be checked for its lift off force. If the lift off forces of the adjacent tendons are greater than or equal to 95% of their Base Values, the single deficiency shall be considered unique and acceptable. For tendon(s)~~

~~*For each inspection, the tendons shall be selected on a random but representative basis so that the sample group will change somewhat for each inspection; however, to develop a history of tendon performance and to correlate the observed data, one tendon from each group (dome, vertical, and hoop) may be kept unchanged after the initial selection.~~

Insert A (TS 3.6.1.6):

If the structural integrity of the containment is found to be inoperable, restore the containment to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS (Continued)

these requirements, a determination shall be made as to the cause of the occurrence and the tendon(s) shall be restored to the required level of integrity.

If the lift-off force of the selected tendon lies below 90% of its Base Value, the tendon shall be completely detensioned and a determination made as to the cause of the occurrence.

b. Determining that the average of the Normalized Lift Off Forces for each tendon group (vertical, dome and hoop) is greater than or equal to the minimum required average tendon force for the group. The minimum required average tendon force is 1160 kips for vertical tendons, 1063 kips for dome tendons, and 1000 kips for hoop tendons. The Normalized Lift Off Force for a tendon is obtained by adding the Normalizing Factor appearing in Table 4.6-2 to the lift off force. Failure to comply with this requirement may be evidence of abnormal degradation of the containment structure.

If the Normalized Lift-Off Force of any tendon is less than the applicable minimum required average tendon force, an investigation shall be conducted to determine the cause and extent of occurrence. This investigation shall include as a minimum the measurement of lift-off forces of tendons adjacent to the deficient tendon to determine if the average of the tendon lift-off forces in this region of the containment is equal to or greater than the minimum required average tendon force. Failure to comply with this requirement may be evidence of abnormal degradation of the containment structure.

c. Detensioning one tendon in each group (dome, vertical and hoop) from the representative sample. One wire shall be removed from each detensioned tendon and examined to determine:

1. That over the entire length of the tendon wire, the wire has not undergone corrosion, cracks or damage to the extent that an abnormal condition is indicated.
2. A minimum tensile strength value of 240,000 psi (guaranteed ultimate strength of the tendon material) for at least three wire samples (one from each end and one at mid-length) cut from each removed wire.

d. Determining for each tendon in the above representative tendon sample, that an analysis of a sample of the sheathing filler grease is within the following limits:

1.	Grease Voids	$\leq$ 5% of net duct volume
2.	Chlorides	$\leq$ 10 PPM
3.	Sulphides	$\leq$ 10 PPM
4.	Nitrates	$\leq$ 10 PPM
5.	Water Content	$\leq$ 10% by weight

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

~~If the inspections performed at 1, 3, and 5 years indicate no abnormal degradation of the tendon system, the number of sample tendons may be reduced to 3 dome, 3 vertical, and 3 hoop for subsequent inspections. Upon the completion of the five year inspection, the results of the first three inspections shall be evaluated to determine if an abnormal condition is evident for the tendon system. Based on the conclusions of this evaluation, the sample tendons with their Base Values and Normalizing Factors will be specified for all subsequent locations.~~

~~4.6.1.6.2 At the same inspection frequency as the tendons, the structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the adjacent concrete surfaces shall be determined by a visual inspection and verifying that no abnormal material or structural behavior is evident.~~

4.6.1.6.3 In accordance with the Containment Leakage Rate Testing Program, the structural integrity of the exposed accessible interior and exterior surfaces of the containment shall be determined by a visual inspection of these surfaces and verifying that no abnormal material or structural behavior is evident.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on reactor building internal pressure ensure that 1) the reactor building structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig and 2) the reactor building peak pressure does not exceed the design pressure of 57psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 53.5 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 53.5 psig which is less than design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on reactor building average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a steam line break accident.

3/4.6.1.6 ~~REACTOR~~ CONTAINMENT BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 53.5psig in the event of a steam line break accident. The measurement of containment tendon lift off force, the tensile tests of the tendon wires, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

~~The tendon lift off forces are evaluated to ensure that 1) the rate of tendon force loss is within predicted limits, and 2) a minimum required prestress level exists in the containment. In order to assess the rate of force loss, the lift off force for a tendon is compared with the force predicted for the tendon times a reduction factor of 0.95. This resulting force is referred to as the 95% Base Value. The predicted tendon force is equal to the original stressing force minus losses due to elastic shortening of the tendon, stress relaxation of the tendon wires, and creep and shrinkage of the concrete. The 5% reduction on the predicted force is intended to compensate for both uncertainties in the prediction techniques for the losses and for inaccuracies in the lift-off force measurements.~~

INSERT 'B'

Insert B (B 3/4.6.1.6)

The reactor building structural integrity limitations as described in the Containment Inservice Inspection Program (CISIP) ensure that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 53.5 psig in the event of a steam line break accident. The measurement of containment tendon lift off force, the tensile tests of the tendon wires, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability. Visual and other required examinations of tendons, anchorages, and surfaces are performed periodically in accordance with plant procedures. These procedures embody applicable requirements of the 1992 Addenda of ASME Code, Section XI, Subsection IWL as set forth in 10CFR50.55a. Any degradations exceeding the CISIP acceptance criteria will be reviewed under an engineering evaluation within 60 days of the completion of the inspection to determine what impact the degradation has on overall containment operability, if any.

In addition, any significant degradation which seriously challenges containment operability found during the inspection shall be reported to the NRC in accordance with Technical Specification 6.9.2 within 30 days. The report shall include the description of degradation, operability determination, root cause determination, and corrective actions taken.

The tendon lift-off forces are evaluated to ensure that 1) the rate of tendon force loss is within predicted limits, and 2) a minimum required tendon force level exists in the containment. In order to assess the rate of force loss, the average lift off force for a tendon is compared with 95% of the predicted force. The predicted force is calculated by subtracting the initial, time-dependent, and other losses where applicable from the original stressing force, consistent with the recommendations of Regulatory Guide 1.35.1, Revision 3 dated July 1990.

REACTOR BUILDING STRUCTURAL INTEGRITY (Continued)

In order for the tendon lift off force to be indicative of the level of prestress force in the containment, each measured force must be adjusted for the known differences which exist among the tendons due to original stressing force and elastic shortening loss. This adjustment is accomplished through the use of a Normalizing Factor ($NF_i(t)$). This factor is added to the lift off force, which results in the Normalized Lift Off Force. The Normalizing Factor is given by:

$$NF_i(t) = \{F_{ave}(o) - F_i(o)\} \{1 - \frac{SR(t)}{100}\} + \Delta F^T_{es} \{\frac{N - 2n + 1}{2N}\}$$

$\{F_{ave}(o) - F_i(o)\}$ is the group average lock-off force at original stressing, minus the original stressing force for the specific tendon.

SR(t) is stress relaxation (percent) which occurs at time t after original stressing.

ΔF^T_{es} is the total elastic shortening tendon force loss.

n is the stressing sequence comprising the specific tendon.

N is the total number of stressing sequences for the group of tendons which comprise the specific tendons.

i refers to the specific tendon.

t refers to the time after original stressing of the current inspection period.

The Base Values and Normalizing Factors of tendons selected for surveillances 4 through 10 are listed in Enclosure 8 of Attachment I to the Virgil C. Summer Nuclear Station Surveillance Test Procedure STP-160.001, "Containment Tendon Test," Revision 2. Based on experience from the first three tendon surveillances, STP-160.001 may have to be revised to add Base Values and Normalizing Factors for additional or alternate tendons not previously listed, but are required for a particular surveillance. The revision level of STP-160.001, as listed above, need not be updated in this Technical Specification where a change to Enclosure 8 of Attachment I only adds Base Values and Normalizing Factors for tendons not previously listed, but were used as additional or alternates for a particular surveillance. Base Values and Normalizing Factors listed in Enclosure 8 of Attachment I of STP-160.001 will not be revised prior to NRC approval.

The surveillance requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Proposed Revision 3 to Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments," April 1979, except that in place of the Lower Limit and 90% Lower Limit defined by these Regulatory Guides, the 95% Base Value and 90% Base Value, respectively, are used.

g. Containment Leakage Rate Testing Program (Continued)

2) Air lock testing acceptance criteria are:

a. Overall air lock leakage rate is $\leq 0.10\ L_a$ when tested at $\geq P_a$.

b. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 8.0 psig for at least 3 minutes.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

INSERT 'a'

Insert C (New section 6.8.4.h):

h. Containment Inservice Inspection Program

This program provides controls for monitoring containment vessel structural integrity including routine inspections and tests to identify degradation and corrective actions if degradation is found. The Containment Inservice Inspection Program, inspection frequencies and acceptance criteria shall be in accordance with 10CFR50.55a as modified by approved exemptions. Predicted lift-off forces shall be determined consistent with the recommendations of Regulatory Guide 1.35.1, Revision 3 dated July 1990.

Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the Containment Inservice Inspection Program shall undergo an engineering evaluation within 60 days of the completion of the inspection surveillance. The results of the engineering evaluation shall be reported to the NRC within an additional 30 days of the time the evaluation is completed. The report shall include the cause of the condition that does not meet the acceptance criteria, the acceptability of the concrete containment without repair of the item, whether or not repair or replacement is required and, if required, the extent, method, and completion of necessary repairs, and the extent, nature, and frequency of additional examinations.

In addition, any significant degradation which seriously challenges containment operability found during the inspection shall be reported to the NRC in accordance with Technical Specification 6.9.2 within 30 days. The report shall include the description of degradation, operability determination, root cause determination, and corrective actions taken.

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

If the structural integrity of the containment is found to be inoperable, restore the containment to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 The structural integrity of the containment shall be demonstrated in accordance with the Containment Inservice Inspection Program.

4.6.1.6.2 Deleted

4.6.1.6.3 In accordance with the Containment Leakage Rate Testing Program, the structural integrity of the exposed accessible interior and exterior surfaces of the containment shall be determined by a visual inspection of these surfaces and verifying that no abnormal material or structural behavior is evident.

Page 3/4 6-9 has been Deleted.
Page 3/4 6-10 has been Deleted.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on reactor building internal pressure ensure that 1) the reactor building structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig and 2) the reactor building peak pressure does not exceed the design pressure of 57 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 53.5 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 53.5 psig which is less than design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on reactor building average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a steam line break accident.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 53.5 psig in the event of a steam line break accident. The measurement of containment tendon lift off force, the tensile tests of the tendon wires, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

The reactor building structural integrity limitations as described in the Containment Inservice Inspection Program (CISIP) ensure that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 53.5 psig in the event of a steam line break accident. The measurement of containment tendon lift off force, the tensile tests of the tendon wires, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability. Visual and other required examinations of tendons, anchorages, and surfaces are performed periodically in accordance with plant procedures. These procedures embody applicable requirements of the 1992 Addenda of ASME Code, Section XI, Subsection IWL as set forth in 10CFR50.55a. Any degradations exceeding the CISIP acceptance criteria will be reviewed under an engineering evaluation within 60 days of the completion of the inspection to determine what impact the degradation has on overall containment operability, if any.

CONTAINMENT STRUCTURAL INTEGRITY (Continued)

In addition, any significant degradation which seriously challenges containment operability found during the inspection shall be reported to the NRC in accordance with Technical Specification 6.9.2 within 30 days. The report shall include the description of degradation, operability determination, root cause determination, and corrective actions taken.

The tendon lift-off forces are evaluated to ensure that 1) the rate of tendon force loss is within predicted limits, and 2) a minimum required tendon force level exists in the containment. In order to assess the rate of force loss, the average lift off force for a tendon is compared with 95% of the predicted force. The predicted force is calculated by subtracting the initial, time-dependent, and other losses where applicable from the original stressing force, consistent with the recommendations of Regulatory Guide 1.35.1, Revision 3 dated July 1990.

g. Containment Leakage Rate Testing Program (Continued)

2) Air lock testing acceptance criteria are:

a. Overall air lock leakage rate is $\leq 0.10\ L_a$ when tested at $\geq P_a$.

b. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 8.0 psig for at least 3 minutes.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

h. Containment Inservice Inspection Program

This program provides controls for monitoring containment vessel structural integrity including routine inspections and tests to identify degradation and corrective actions if degradation is found. The Containment Inservice Inspection Program, inspection frequencies and acceptance criteria shall be in accordance with 10CFR50.55a as modified by approved exemptions. Predicted lift-off forces shall be determined consistent with the recommendations of Regulatory Guide 1.35.1, Revision 3 dated July 1990.

Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the Containment Inservice Inspection Program shall undergo an engineering evaluation within 60 days of the completion of the inspection surveillance. The results of the engineering evaluation shall be reported to the NRC within an additional 30 days of the time the evaluation is completed. The report shall include the cause of the condition that does not meet the acceptance criteria, the acceptability of the concrete containment without repair of the item, whether or not repair or replacement is required and, if required, the extent, method, and completion of necessary repairs, and the extent, nature, and frequency of additional examinations.

In addition, any significant degradation which seriously challenges containment operability found during the inspection shall be reported to the NRC in accordance with Technical Specification 6.9.2 within 30 days. The report shall include the description of degradation, operability determination, root cause determination, and corrective actions taken.

SAFETY EVALUATION
REVISION OF REACTOR BUILDING POST TENSION TENDON SURVEILLANCE CRITERIA FOR THE VIRGIL C. SUMMER NUCLEAR STATION TECHNICAL SPECIFICATIONS

Description of Amendment Request

South Carolina Electric and Gas (SCE&G) proposes to revise the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) to incorporate the requirements for the surveillance of the Reactor Building Post Tensioned Tendon System as specified in Title 10 of the Code of Federal Regulations (10CFR) sections 50.55a(b)(2)(vi) and 50.55a(b)(2)(viii). SCE&G proposes to revise the following Technical Specifications:

> TS LCO 3.6.1.6 will be revised to replace the specific requirements that describe the containment operability with a statement that the structural integrity of the containment is required to be OPERABLE. The Action Statements for LCO 3.6.1.6 will be revised to permit 1 hour to restore containment operability or require a plant shutdown.
>
> TS SR 4.6.1.6.1 and 4.6.1.6.2 will be revised to delete the current detailed surveillance requirements and to refer to the Containment Inservice Inspection Program (CISIP). The CISIP is to be maintained as a separate Licensee controlled program. Refer to Attachment IV for a description of the CISIP.
>
> TS 6.8.4.h will be added to ensure the Containment Inservice Inspection Program is provided that demonstrates containment structural integrity.
>
> The BASES for TS 3/4.6.1.4 will be revised to refer to 10CFR50.55a(b)(2)(vi), 10CFR50.55a(b)(2)(viii), and Regulatory Guide 1.35.1, Revision 3 dated July 1990.

10CFR50.55a requires that the containment structural integrity be determined in accordance with the requirements of ASME Code Section XI, Subsection IWL, "Requirements for Class CC Concrete Components of Light Water Cooled Power Plants," 1992 Edition including 1992 Addenda and the five additional requirements and limitations that are stated in 50.55a(b)(2)(viii), "Examination of Concrete Containments."

In summary, the proposed changes to the Technical Specifications for surveillance of the Reactor Building post-tensioned tendon system include additional surveillance requirements to comply with current criteria required by 10CFR50.55a. None of the changes impact the tendon force provided by the tendon system. As enhancements to the surveillance program, the changes improve the capability to ensure the tendon system is performing as designed.

Safety Evaluation

10CFR50.55a has been amended[1] to incorporate, by reference, the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWL, 1992 Edition including 1992 Addenda. Additionally, 10CFR50.55a contains requirements and limitations that augment the ASME Code. The tendon surveillance requirements currently reflected in the Technical Specifications and FSAR comply with the criteria that was applicable when the plant was licensed including proposed Revision 3 to Regulatory Guide 1.35, April 1979. The surveillance requirements in 10CFR50.55a and the ASME Code Section XI include requirements beyond those in the current FSAR and Technical Specifications.

The post-tensioned tendon system is an integral structural force-resisting component of the Reactor Building. The post-tensioning system is comprised of three groups of tendons - horizontal or hoop, vertical, and dome. The tendon system provides a compressive force in the Reactor Building concrete that counteracts the internal pressure inside the building, postulated to occur during certain accidents such as LOCA and steam line breaks. The tendons were installed in ducts embedded within the Reactor Building concrete and were then tensioned to a prescribed force. The tendon surveillance program is designed to inspect the tendon system on a regular schedule to ensure that the system is performing as designed without occurrence of abnormal degradation.

Over time, the force in the individual tendons is expected to decrease due to losses caused by phenomena such as tendon wire relaxation and concrete creep and shrinkage. The amount of force initially provided by the tendon system design and installation accounted for these losses. The surveillance program measures the force in a small sample population of tendons from each of the three groups at predetermined time intervals. This ensures that the rate of tendon force loss is within predicted limits and that the minimum tendon force required to meet the design basis will be available through the time of the next scheduled surveillance.

The Nuclear Regulatory Commission has found the tendon surveillance requirements stated in and incorporated by reference in 10CFR50.55a are adequate to ensure that the tendon systems are performing as designed without occurrence of abnormal degradation.

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[1] Federal Register 63FR154, 41303 - 41311

NO SIGNIFICANT HAZARDS EVALUATION FOR THE REVISION OF REACTOR BUILDING POST TENSION TENDON SURVEILLANCE CRITERIA FOR THE VIRGIL C. SUMMER NUCLEAR STATION TECHNICAL SPECIFICATIONS

Description of Amendment Request

South Carolina Electric and Gas (SCE&G) proposes to revise the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) to incorporate the requirements for the surveillance of the Reactor Building Post Tensioned Tendon System as specified in Title 10 of the Code of Federal Regulations (10CFR) sections 50.55a(b)(2)(vi) and 50.55a(b)(2)(viii). SCE&G proposes to revise the following Technical Specifications:

> TS LCO 3.6.1.6 will be revised to replace the specific requirements that describe the containment operability with a statement that the structural integrity of the containment is required to be OPERABLE. The Action Statements for LCO 3.6.1.6 will be revised to permit 1 hour to restore containment operability or require a plant shutdown.
>
> TS SR 4.6.1.6.1 and 4.6.1.6.2 will be revised to delete the current detailed surveillance requirements and to refer to the Containment Inservice Inspection Program (CISIP). The CISIP is to be maintained as a separate Licensee controlled program. Refer to Attachment IV for a description of the CISIP.
>
> TS 6.8.4.h will be added to ensure the Containment Inservice Inspection Program is provided that demonstrates containment structural integrity.
>
> The BASES for TS 3/4.6.1.4 will be revised to refer to 10CFR50.55a(b)(2)(vi), 10CFR50.55a(b)(2)(viii), and Regulatory Guide 1.35.1, Revision 3 dated July 1990.

10CFR50.55a requires that the containment structural integrity be determined in accordance with the requirements of ASME Code Section XI, Subsection IWL, "Requirements for Class CC Concrete Components of Light Water Cooled Power Plants," 1992 Edition including 1992 Addenda and the five additional requirements and limitations that are stated in 50.55a(b)(2)(viii), "Examination of Concrete Containments."

In summary, the proposed changes to the Technical Specifications for surveillance of the Reactor Building post-tensioned tendon system include additional surveillance requirements to comply with current criteria required by 10CFR50.55a. None of the changes impact the tendon force provided by the tendon system. As enhancements to the surveillance program, the changes improve the capability to ensure the tendon system is performing as designed.

Basis for No Significant Hazards Consideration Determination

In accordance with 10CFR50.92, a proposed change to the Operating License involves no "significant hazards" if operation of the facility, in accordance with the proposed change, would not:

1) involve a significant increase in the probability or consequences of any accident previously evaluated;
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 to safety.

This request is evaluated against each of these criteria as follows:

1) This proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes revise the surveillance requirements for containment reinforced concrete and unbonded post-tensioning systems inservice examinations as required by 10CFR50.55a(b)(2)(vi) and 10CFR50.55a(b)(2)(viii). The revised requirements affect the inservice inspection program designed to detect structural degradation of the containment reinforced concrete and unbonded post-tensioning systems and do not affect the function of the containment reinforced concrete and unbonded post-tensioning system components. The reinforced concrete and unbonded post-tensioning systems are passive components whose failure modes could not act as accident initiators or precursors.

The proposed changes do not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. They do not involve the addition or removal of any equipment, or any design changes to the facility. Therefore, this proposed change does not represent a significant increase in the probability or consequences of an accident previously evaluated.

2) This proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new of different requirements or introduce a new accident initiator, accident precursor or malfunction mechanism. The proposed changes provide an NRC approved ASME Code inspection /testing methodology to assure age related degradation of the containment structure will not go undetected. The function of the containment reinforced concrete and unbonded post-tensioning system components are not altered by this change. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure. Therefore, this proposed change does not create the possibility of an accident of a different type than previously evaluated.

3) *This proposed change does not involve a significant reduction in a margin to safety.*

The Reactor Building internal design pressure is 57 psig and the maximum peak pressure from a postulated steam line break is 53.5 psig. The proposed change does not impact the margin of safety included in the design pressure compared to the peak calculated pressure because the proposed activity does not alter, in any way, the available force provided by the tendons. Additionally, the proposed activity does not affect the initial temperature conditions within the Reactor Building assumed in the accident analysis for a steam line break. Therefore, this proposed change does not involve a significant reduction in a margin to safety.

Environmental Impact Consideration

SCE&G has reviewed this request against the criteria of 10CFR51.22 for environmental considerations. Since this request involves (i) no significant hazard consideration, (ii) no significant change in the types or increase in the amounts of any effluents that may be released offsite, and (iii) no significant increase in individual or cumulative occupational radiation exposure, SCE&G has concluded that the proposed change meets the criteria given in 10CFR51.22 (c)(9) for a categorical exclusion from the requirement for an environmental impact statement.

DESCRIPTION
OF THE
VIRGIL C. SUMMER NUCLEAR STATION
CONTAINMENT INSERVICE INSPECTION PROGRAM

The Virgil C. Summer Nuclear Station (VCSNS) Containment Inservice Inspection Program (CISIP) prescribes the requirements and instructions for the examination and testing of ASME Class MC and Class CC components and component supports. The CISIP is being developed in accordance with the requirements of the 1992 Edition with the 1992 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsection IWE and IWL, as modified by NRC final rulemaking to 10CFR50.55a published in the Federal Register[2] on August 8, 1996.

The top tier program document for the CISIP will be, "Virgil C. Summer Nuclear Station, Containment Inservice Inspection Program Plan," which is currently being developed. This document will provide the basis for inclusion and identification of metal and concrete components which are required to be included in the CISIP and will summarize the requirements for the examination and testing of ASME Class MC and Class CC pressure retaining components. The new program requirements implemented by the procedures noted below constitute the new VCSNS CISIP.

Since commercial operation, inspections and tests of the containment systems have been performed in accordance with 10CFR50, Appendix J, Regulatory Guide 1.35, Proposed Rev. 3, April 1979, and additional criteria specified in the Bases of the Technical Specifications (TS), to ensure containment integrity and design function are maintained. Surveillance requirements for containment structural integrity are currently addressed in the VCSNS TS Section 3.6.1.6, "Containment Structural Integrity, Limiting Condition for Operation," and Surveillance Requirements 4.6.1.6.1 through 4.6.1.6.3. The detailed implementation of the referenced TS requirements is accomplished through the following specifications, design guidelines and surveillance test procedures:

G/C Specification SP-228, "Surveillance of Reactor Building Post Tensioning System," which provides program requirements and criteria for:

- tendon selection
- testing and inspection methodology
- acceptance criteria.

STP-160.001, "Containment Tendon Test," which provides detailed instructions for conducting the tendon testing and inspection.

STP-207.002, "Inspection of Containment," which provides criteria and detailed instructions for the visual inspection of/for:

- interior steel surfaces
- exterior concrete surfaces
- grease leakage.

Design Guide ST-04, "Reactor Building Tendon System Surveillance," which provides detailed practical guidance to engineering and other personnel involved in the implementation of the CISIP.

[2] Federal Register 63FR154, 41303 - 41311

These documents will be revised to reflect the requirements of 10CFR50.55a, including Section XI, Division 1, Subsection IWE and IWL of the ASME Boiler and Pressure Vessel Code, 1992 Edition with 1992 Addenda, as modified by 10CFR50.55a.

These changes include, but are not limited to the following:

- Requiring the inspection of all accessible grease caps
- Tracking the trend of tendon force loss on both an individual and a group tendon basis
- Evaluating consecutive period tendon force surveillance data for the same tendon in a group of tendons (common tendon),
- Changing the limit on tendon elongation to be within 10% of that recorded during the previous lift-off force measurement
- Including the following as reportable conditions:
  - the presence of free water in the sheathing filler grease
  - entrained water in the sheathing filler grease exceeding 10% by weight
  - grease voids exceeding 10% of the net sheathing duct volume
- Specifying the surveillance grace period to be not more than 1 year prior to nor 1 year later than the specified date for performance of the surveillance test
- Specifying the tendon force measurement accuracy to be at least 1.5% of the minimum ultimate tensile strength of the tendon
- Specifying the lock-off force in a re-tensioned tendon to not exceed 70% of the guaranteed ultimate tensile strength of the tendon wire
- Specifying visual examination of anchorage areas within 2 feet of the bearing plates
- Specifying pH testing of samples of free water found in the tendon sheathing filler grease (where sufficient free water is present for testing)
- Specifying the analysis of grease samples for reserve alkalinity
- Specifying the following additional Tendon Lift-off Force Acceptance Criteria:
  - measured force in not more than one tendon may be between 90 and 95% of the predicted force;
  - the measured force in all of the remaining tendons must not be less than 95% of the predicted force
- Requiring that the average of the lift-off forces for each group of tendons (i.e. dome, vertical or hoop) is equal to or greater than the minimum required tendon force for that group of tendons
- Requiring that the trend of the average lift-off forces for each group of tendons (i.e. dome, vertical or hoop) indicates the average force in each group of tendons will meet the minimum required average force until the next scheduled surveillance

These documents, once revised, will be the detailed implementation tools for the new CISIP. Additional technical aspects of the inspection and test are further implemented by vendor procedures, which are reviewed and approved by cognizant SCE&G personnel or authorized agent.