UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 6, 2000

Mr. Stephen A. Byrne
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88
Jenkinsville, South Carolina 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT (TAC NO. MA7366)

Dear Mr. Byrne:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 148 to Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1. The amendment changes the Technical Specifications (TS) in response to your application dated January 5, 2000, as supplemented August 25, 2000.

This amendment changes TS 3/4 6.1.6, including its Bases, and adds TS 6.8.4.h. The changes support the new requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a, 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 10 CFR 50.55a(b)(2)(viii).

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's Bi-weekly Federal Register notice. This completes the staff's efforts on TAC No. MA7366.

Sincerely,

Karen Cotton

Karen Cotton, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures:
1. Amendment No. 148 to NPF-12
2. Safety Evaluation

cc w/enclosures: See next page

September 6, 2000

Mr. Stephen A. Byrne
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88
Jenkinsville, South Carolina 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT (TAC NO. MA7366)

Dear Mr. Byrne:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 148 to Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1. The amendment changes the Technical Specifications (TS) in response to your application dated January 5, 2000, as supplemented August 25, 2000.

This amendment changes TS 3/4 6.1.6, including its Bases, and adds TS 6.8.4.h. The changes support the new requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a, 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 10 CFR 50.55a(b)(2)(viii).

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's Bi-weekly Federal Register notice. This completes the staff's efforts on TAC No. MA7366.

Sincerely,

/RA/

Karen Cotton, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures:
1. Amendment No. 148 to NPF-12
2. Safety Evaluation

cc w/enclosures: See next page

DOCUMENT NAME: G:\PDII-1\SUMMER\SUMMDma7366.WPD

OFFICE	PDII-1/PM	PDII-2/LA	OGC	PDII-1/SC
NAME	KCotton:mw	EDunnington		REmch
DATE	9/5/00	8/30/00	9/5/00	9/6/00

OFFICIAL RECORD COPY

AMENDMENT NO. 148 TO FACILITY OPERATING LICENSE NO. NPF-12 - SUMMER, UNIT NO. 1

DISTRIBUTION
PUBLIC
PD II-2 Rdg.
RidsNrrDlpmLpdii
RidsNrrDlpmLpdii1
RidsNrrPMKCotton
EDunnington (paper copy)
RidsOgcRp
GHill (2)
RidsNrrDripRtsb (WBeckner)
RidsAcrsAcnwMailCenter
RidsRgn2MailCenter

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 148
License No. NPF-12

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

   A. The application for amendment by South Carolina Electric & Gas Company (the licensee), dated January 5, 2000, as supplemented August 25, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-12 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 148 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: September 6, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 148

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 6-8	3/4 6-8
3/4 6-9	3/4 6-9
3/4 6-10	---
B3/4 6-2	B3/4 6-2
B3/4 6-2a	B3/4 6-2a
6-12c	6-12c

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.

SURVEILLANCE REQUIREMENTS (Continued)

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

Next Page is 3/4 6-11

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 148 TO FACILITY OPERATING LICENSE NO. NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-395

1.0 INTRODUCTION

On January 7, 1994, the Nuclear Regulatory Commission (NRC) published a proposed amendment to the regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsections IWE and IWL of Section XI, Division I of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). The final rule, Subpart 50.55a(g)(6)(ii)(B) of Title 10 of the *Code of Federal Regulations* (10 CFR), became effective on September 9, 1996, and requires licensees to implement Subsections IWE and IWL, with specified modifications and limitations, by September 6, 2001.

By letter dated January 5, 2000, as supplemented August 25, 2000, South Carolina Electric & Gas Company (SCE&G, the licensee) submitted an amendment to the technical specifications (TS) for Virgil C. Summer Nuclear Station (V. C. Summer). The licensee proposes changes to revise the TS to conform to the new regulatory requirements. The major portion of the changes involve replacing the requirements for tendon surveillance with inspection procedures in accordance with the requirements of Subsections IWE and IWL of the Code and 10 CFR 50.55a(g)(6)(ii)(B). The licensee also proposed a change to the reporting requirements. In response to the staff's concerns regarding the reporting requirements, the licensee revised the reporting requirements as described in the August 25, 2000, supplement. The August 25, 2000, supplement revised the proposed wording of Bases Section 3/4.6.1.6 and TS 6.8.4.h to clarify the reporting requirements; the clarification did not impact the initial no significant hazards consideration determination.

2.0 BACKGROUND

The licensee proposed the following changes to the current TS:

Change 1 Reworded the Limiting Condition for Operation (LCO) in Section 3.6.1.6 to replace the specific requirements for the structural integrity of containment by the statement, "The structural integrity of the containment shall be OPERABLE."

Change 2 Deleted the original ACTIONS a. and b. in Section 3.6.1.6 (Sections 3.6.1.6.a and 3.6.1.6.b) and replaced them by the words, "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."

Change 3 Deleted the tendon surveillance inspection procedures and requirements (Section 4.6.1.6.1 on Pages 3/4 6-8 through 3/4 6-10), and replaced them by the words, "The structural integrity of the containment shall be demonstrated in accordance with the Containment Inservice Inspection Program."

Change 4 Deleted Section 4.6.1.6.2 and moved Section 4.6.1.6.3 from Page 3/4 6-10 to Page 3/4 6-8.

Change 5 Replaced the word "Reactor" in the title of Bases Section 3/4.6.1.6 by the word "Containment."

Change 6 Replaced the second paragraph of Bases Section 3/4.6.1.6 (Page B3/4 6-2) and the description of "Reactor Building Structural Integrity" (Page B 3/4 6-2a) by two new paragraphs in which the Bases for containment building structural integrity are to be demonstrated through the implementation of the Containment Inservice Inspection Program (CISIP). The licensee claimed that the procedures of this program meet the requirements of the 1992 Addenda of ASME Code, Section XI, Subsection IWL as set forth in10 CFR 50.55a. Also, the licensee committed that 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, the licensee added a new paragraph which states that any significant degradation which seriously challenges containment operability found during the inspection shall be reported to the NRC in accordance with TS 6.9.2 within 30 days. The report shall include the description of degradation, operability determination, root cause determination, and corrective actions taken.

Change 7 Added a description of the CISIP in new Section 6.8.4.h which stated that the frequencies and acceptance criteria shall be in accordance with 10 CFR 50.55a. Also, the licensee committed that 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, the licensee added a new paragraph which states that any significant degradation which seriously challenges containment operability found during the inspection shall be reported to the NRC in accordance with TS 6.9.2 within 30 days. The report shall include the description of degradation, operability determination, root cause determination, and corrective actions taken.

3.0 EVALUATION

In its submittal, the licensee, in addition to editorial and some minor changes, removed the details of its tendon surveillance inspection from the current TS, and proposed to demonstrate the containment structural integrity through the implementation of its CISIP. According to the licensee, the CISIP has incorporated the requirements for the surveillance of the reactor building post-tensioned tendon system as specified in 10 CFR 50.55a(b)(2)(vi) and 10 CFR 50.55a(b)(2)(viii). The licensee also added a new section (Section 6.8.4.h) to specify the reporting requirements. In addition, the licensee stated, in the revised Section B 3/4.6.1.6, that through the implementation of the CISIP (the measurement of tendon lift 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), the containment structural integrity can be ensured to withstand a maximum pressure of 53.5 psig in the event of a steamline break accident.

The staff reviewed the proposed changes to the TS related to containment vessel structural integrity and found that the elements contained in the revised TS are acceptable on the basis of the following requirements: (1) the containment structural integrity will be maintained via the provision of proposed TS 4.6.1.6; (2) the procedures of the CISIP are in accordance with the 1992 Addenda of ASME Code, Section XI, Subsection IWL as set forth in 10 CFR 50.55a; and (3) any degradation which exceeds the acceptance criteria detected through the CISIP will be reported to the NRC and will include a statement of the corrective actions taken and an engineering evaluation report.

For the reporting requirements (TS B 3/4.6.1.6 and TS 6.8.4.h), the licensee committed, in the January 5, 2000, submittal, that any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the CISIP 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 response to the staff's concern regarding the 90-day reporting commitment, the licensee submitted an additional submittal dated August 25, 2000, to add a new paragraph to Bases Section 3/4.6.1.6 and TS 6.8.4.H of the revised TS, and stated that in addition, any significant degradation which seriously challenges the containment operability found during the inspection shall be reported to the NRC, including the description of degradation, operability condition, root cause determination, and corrective action taken within 30 days. To report the event to the NRC within 30 days, and follow up by an engineering evaluation report within 90 days, will not adversely affect plant safety, and meets the reporting requirements in 10 CFR 50.73. Therefore, the staff finds these changes acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (65 FR 9010). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Approval of the proposed TS changes does not relieve the licensee of its responsibility to report, pursuant to 10 CFR 50.73(a)(2)(ii), any event or condition that results in the condition of the nuclear power plant being seriously degraded. These conditions include serious degradation of the containment concrete structure, such as de-lamination of the dome concrete, corrosion of prestressing elements or anchorage components extending more than two tendons, tendon force trend not meeting the requirement of 10 CFR 50.55a(b)(2)(ix)(B), and widespread corrosion of the steel liner plate.

Principal Contributor: T. Cheng

Date: September 6, 2000

cc:

Mr. R. J. White
Nuclear Coordinator
S.C. Public Service Authority
c/o Virgil C. Summer Nuclear Station
Post Office Box 88, Mail Code 802
Jenkinsville, South Carolina 29065

Resident Inspector/Summer NPS
c/o U.S. Nuclear Regulatory Commission
Route 1, Box 64
Jenkinsville, South Carolina 29065

Chairman, Fairfield County Council
Drawer 60
Winnsboro, South Carolina 29180

Mr. Henry Porter
Division of Radioactive Waste Management
Bureau of Land & Waste Management
Department of Health & Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Mr. Bruce C. Williams, General Manager
Nuclear Plant Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station, Mail Code 303
Post Office Box 88
Jenkinsville, South Carolina 29065

Mr. Melvin N. Browne, Manager
Nuclear Licensing & Operating Experience
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station, Mail Code 830
Post Office Box 88
Jenkinsville, South Carolina 29065

J. B. Knotts, Jr., Esquire
Winston & Strawn Law Firm
1400 L Street, N.W.
Washington, D.C. 20005-3502