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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-395

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY

OPERATING LICENSE

On August 6, 1982, the U. S. Nuclear Regulatory Commission (the Commission) issued Facility Operating License NPF-12 to South Carolina Electric & Gas Company and South Carolina Public Service Authority (licensees) authorizing operation of the Virgil C. Summer Nuclear Station, Unit No. 1 (the facility), at reactor core power levels not in excess of 2775 megawatts thermal in accordance with the provisions of the license, the Technical Specification and the Environmental Protection Plan with a condition limiting operation to five percent of full power (139 megawatts thermal).

The Commission has now issued Amendment No. 5 to Facility Operating License No. NPF-12 which authorizes operation of the Virgil C. Summer Nuclear Station, Unit No. 1, at reactor core power levels not in excess of 2775 megawatts thermal in accordance with the provisions of the amended license. The amendment includes a license condition restricting operation to 50 percent of full power until certain documentation concerning the steam generators is submitted for NRC staff review and approval. The amendment is effective as of the date of issuance.

The Virgil C. Summer Nuclear Station, Unit No. 1 is a pressurized water nuclear reactor located in Fairfield County, South Carolina, approximately 26 miles northwest of Columbia, South Carolina and approximately one mile east of the Broad River near Parr, South Carolina.

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The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's regulations in 10 CFR Chapter I, which are set forth in the amended license. Prior public notice of the overall action involving the proposed issuance of an operating license was published in the FEDERAL REGISTER on April 18, 1977 (42 FR 20203). The increase in power level authorized by this amendment is encompassed by that prior public notice.

The Commission has determined that the issuance of this license will not result in any environmental impacts other than those evaluated in the Final Environmental Statement since the activity authorized by the license is encompassed by the overall action evaluated in the Final Environmental Statement.

For further details with respect to this action, see (1) Amendment No. 5 to License No. NPF-12; (2) the Commission's Safety Evaluation Report, dated February 1981 (NUREG-0717), Supplement No. 1, dated April 1981, Supplement No. 2, dated May 1981, Supplement No. 3, dated January 1982, Supplement No. 4, dated August 1982, and Supplement No. 5, dated November 1982; (3) the Final Safety Analysis Report and amendments thereto; (4) the Final Environmental Statement, dated May 1981 (NUREG-0719); (5) the Environmental Report, dated February 1977, and supplements thereto; and (6) the Initial Decisions of the Atomic Safety and Licensing Board, dated July 20, 1982 and August 4, 1982.

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These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. 20555 and at the Fairfield County Library, Garden and Washington Streets, Winnsboro, South Carolina 29180. A copy of Amendment No. 5 to Facility Operating License No. NPF-12 may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing. Copies of the Safety Evaluation Report and its Supplements (NUREG-0717) and the Final Environmental Statement (NUREG-0719) may be purchased at current rates from the National Technical Information Service, Department of Commerce, 5285 Port Royal Road, Springfield, Virginia 22161, and through the NRC GPO sales program by writing to the U. S. Nuclear Regulatory Commission, Attention: Sales Manager, Washington, D. C. 20555. GPO deposit account holders may call 301-492-9530.

Dated at Bethesda, Maryland, this 12th day of November 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

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NUREG-0717
Supplement No. 5

Safety Evaluation Report
related to the operation of Virgil C. Summer Nuclear Station, Unit No. 1

Docket No. 50-395

South Carolina Electric and Gas Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

NOVEMBER 1982

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APPENDICES

1 INTRODUCTION AND GENERAL DESCRIPTION OF FACILITY

1.1 Introduction

Supplement No. 4 to the Nuclear Regulatory Commission Staff's Safety Evaluation Report in the matter of South Carolina Electric & Gas Company's application to operate the Virgil C. Summer Nuclear Station was issued in August 1982. At that time we identified issues for which we had taken positions and would require implementation and/or documentation after the issuance of the operating license. These were made conditions to the operating license which was issued on August 6, 1982. The purpose of this supplement to the Safety Evaluation Report is to provide our evaluation of the licensing conditions that have been resolved since the issuance of Supplement No. 4 to the Safety Evaluation Report and to update other areas where additional information has been received.

1.8 Licensing Conditions

The following is an update of those licensing conditions that have been resolved, modified, or added since the issuance of Supplement No. 4 to the Safety Evaluation Report.

1.8.25 Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment (Section 3.10)

This matter is now resolved as discussed in Section 3.10 of this supplement to the Safety Evaluation Report.

1.8.26 Environmental Qualification (Section 3.11)

Prior to startup after the first major shutdown or refueling outage after June 1983, the licensee shall correct the deficiencies for items 1, 2, 4, and 6 of Table 3-2 of Supplement No. 4 to the Safety Evaluation Report and provide updated component work sheets to the NRC staff.

1.8.27 Model D-3 Steam Generators (Section 5.4.2)

Prior to operation in excess of 2000 hours at power levels in excess of 5% of full power or operation at power levels in excess of 50% of full power, the licensee shall satisfy the NRC staff that appropriate surveillance measures and remedial action plans have been implemented with respect to the steam generator tube vibration problem.

1.8.31 Seismic Design Verification (Sections 3.7.4 and 17.5)

Prior to December 31, 1982 SCE&G shall provide a final report to the NRC staff delineating the final resolution of the actions taken to satisfy the recommendations of the independent design verification conducted by Stone & Webster Engineering Corporation.

1.8.34 Thermal Sleeves (Section 3.9.3)

Prior to startup after the first refueling outage, SCE&G shall provide justification for continued operation with the eight thermal sleeves removed from selected locations in the reactor coolant system.

1.8.35 Emergency Exercise (Section 13.3)

Prior to March 31, 1983, SCE&G shall conduct a limited emergency exercise similar to that conducted on May 5, 1982, but with full local government participation and partial State participation.

1.10 NRC Staff Contributors

This supplement to the Safety Evaluation Report is a product of the NRC staff. The following NRC staff members were the principal contributors to this report.

Name	Title
William F. Kane	Senior Project Manager
Charles G. Hammer	Mechanical Engineer
Shou-Nien Hou	Principal Mechanical Engineer
Arnold Jen-Hsu Lee	Senior Mechanical Engineer
David B. Matthews	Section Leader
Jai Raj Rajan	Senior Mechanical Engineer
John G. Spraul	Senior Quality Assurance Engineer
David Terao	Mechanical Engineer
Harold Walker	Materials Engineer

3 DESIGN CRITERIA FOR SYSTEMS, STRUCTURES, AND COMPONENTS

3.7 Seismic Design

3.7.4 Independent Design Verification Program

In Section 3.7.4 of Supplement No. 4 to the Safety Evaluation Report, the staff identified a license condition pertaining to the completion of the seismic design verification program and the submittal of a final report acceptable to the staff. The purpose of the program was to provide further assurance in the area of design verification. The independent design verification was performed by Stone & Webster Engineering Corporation (SWEC) on the piping system in the flow path of the turbine-driven portion of the emergency feedwater system to steam generator C.

The final report entitled, "Independent Design Verification Turbine Driven Portion Emergency Feedwater System," dated October 15, 1982 was transmitted to the staff in a letter from P. Dunlop (SWEC) to H. Denton dated October 16, 1982.

The program included three major areas of review: 1) a field walkdown for as-built verification, 2) an independent stress analysis and evaluation, and 3) a design control audit. This section addresses the first two tasks.

The purpose of the field walkdown was to determine whether the as-built condition of the piping subsystem was in accordance with the design layout as presented on the isometric drawings. The piping walkdown included identification of valve locations and orientation; support type location and orientation, and verification of other piping dimensions. Differences between the as-built condition and the design drawings were documented. SWEC evaluated these differences and concluded that they were minor and would have no significant effect on the piping stress analysis results. The overall conclusion of this task was that the field walkdown verified that the as-built condition of the piping system was in accordance with the design.

The stress analysis and evaluation task consisted of an independent stress analysis performed by SWEC of three piping subsystems and an evaluation of the results. The scope of the evaluation included a comparison of pipe stresses with code allowables and a comparison of pipe support, anchor, penetration and nozzle loads with the corresponding design loads. The load cases considered were dead load, design pressure, thermal, seismic and jet impingement loads.

All piping stresses from the independent analysis were found to be within the allowables for the three piping subsystem analyses performed.

Review of the support, anchor, penetration and nozzle loads resulted in several instances where the loads from SWEC's analysis were substantially larger than the design loads. These differences were evaluated by SWEC and were subsequently attributed to the following three potential generic discrepancies:

(1) Seismic effects from the diesel generator building were not included in the design analysis for one of the piping subsystems.

(2) Jet impingement loads, emanating from other piping subsystems, were misoriented or mislocated due to errors in the design specification for calculating jet loadings.

(3) Differences in the analysis results were found due to differences in modeling and small differences in the natural frequencies calculations between SWEC's computer program and the program used for the design.

As a result of potential discrepancy number 1, SWEC recommended that seismic response spectra and seismic anchor movements be reviewed for other piping systems in the facility. The licensee undertook a review and determined that four additional cases existed that required reanalysis due to utilization of incorrect seismic response spectra. One case affecting five piping subsystems required reanalysis due to use of improper seismic anchor movements. The piping and support analysis required for these cases indicated that no hardware modifications were necessary. The appropriate design drawings for these piping subsystems have been updated to reflect these analyses. This was considered by SWEC to be acceptable in resolving their finding.

Potential discrepancy number 2 resulted in a recommendation by SWEC that the design specification for calculating jet impingement loadings be updated to clearly reflect the design criteria. The applicant undertook a program of checking inputs to the specification and then checking the application of the report for safety related piping. This effort resulted in locating several discrepancies attributable to lack of clarity, excessive conservatisms, typographical errors and one calculation error in the document. Additionally, it was determined that jet impingement design had not been considered for Westinghouse analyzed reactor coolant loop branch piping and approximately twelve other piping cases. Corrective action was taken by the applicant to revise the design specification. Each of the affected analytical problems was reviewed for the corrected design input.

The loads for several supports increased but these supports were verified to be acceptable without hardware modifications.

The appropriate design drawings have been updated to incorporate the revised jet loading inputs. This was considered by SWEC to be acceptable in resolving their finding.

The third potential discrepancy involved the differences in the mathematical modeling techniques used for piping analysis by the licensee. The analytical differences attributed to this potential discrepancy included variations in pipe support stiffnesses, variations in lumped mass locations, geometrical differences, and differences in engineering judgement. SWEC concluded that these analytical differences were minor and would not have any significant generic ramifications.

Based on our review of the SWEC final report and on the information provided by SWEC at the meetings on October 13, 1982 and October 28, 1982, the staff concluded that SWEC has performed an adequate review of the analytical assumptions and technical procedures used in the analysis of the emergency feedwater system and that the assumptions and procedures used were consistent with project design specifications and commonly accepted standard industry practices. The staff therefore concludes that the independent design verification performed by SWEC provides additional assurance that the seismic requirements as stated in the applicant's design criteria have been met.

Prior to December 31, 1982 SCE&G shall provide a final report to the NRC staff delineating the final resolution of the actions taken to satisfy the recommendations of the independent design verification conducted by Stone & Webster Engineering Corporation.

3.9 Mechanical System and Components

3.9.3 ASME Code Class 1, 2, and 3 Components, Components Supports, and Core Support Structure

The NRC staff has reviewed the licensee's letters dated July 13, 1982 and September 29, 1982 to H. R. Denton regarding removal of eight thermal sleeves from selected nozzles in the reactor coolant system of the Virgil C. Summer Nuclear Station. These include (a) the 3-inch normal and alternate charging connections from the chemical and volume control system, (b) the 6-inch safety injection system high and low head connections, and (c) the 14-inch pressurizer surge line connection. Westinghouse has performed detailed stress evaluations with the thermal sleeves removed and the original welding surface ground smooth. Two-dimensional finite element techniques were used with models covering the nozzle field weld at the safe end and the nozzle crotch region. Effects of various operating transients and mechanical loads, including cumulative fatigue damage were evaluated. The analytical results meet the allowable limits set by Section III of the ASME Code.

The analyses discussed above for the Virgil C. Summer Nuclear Station are similar to the analyses performed for the McGuire, Trojan, and North Anna plants. Based on our review of these analyses, we have concluded that the analytical methods employed by Westinghouse are acceptable. The satisfactory results cited above for the Virgil C. Summer Nuclear Station with the eight thermal sleeves removed indicate that operation of the facility until the first scheduled refueling outage will not cause a safety concern.

Westinghouse stated in the July 14, 1982 meeting on McGuire Unit 1 that they plan to forward a generic resolution to the thermal sleeve problems for the affected plants to justify full-term operation with the thermal sleeves removed. We will require that this generic resolution be submitted by the licensee and approved by the NRC staff prior to startup after the first refueling outage for Virgil C. Summer Nuclear Station.

3.9.6 Inservice Testing of Pumps and Valves

In the Safety Evaluation Report, the staff stated that the licensee had committed to submit an inservice testing program for all ASME Code Class 1, 2, and 3 pumps and valves 30 days prior to loading fuel. The licensee has made submittals for inservice testing of pumps dated September 17, 1980, October 13, 1981, and August 12, 1982 and for inservice testing of valves dated September 17, 1980, December 17, 1981, January 25, 1982, and April 30, 1982.

The licensee has stated that the preservice and inservice testing programs for the above mentioned pumps and valves will meet the requirements of 10 CFR 50.55a(g), including the 1977 edition of the ASME Boiler and Pressure Vessel Code, Section XI through the Summer 1978 Addenda. The licensee requested relief from these code requirements pursuant to 10 CFR 50.55a(g)(1) for certain pump and valve tests.

At this time, we have not completed our detailed review of the licensee's submittals. However, we have evaluated their request for relief and based on our review, we find that it is impractical within the limitations of design, geometry, and accessibility for the licensee to meet certain of the ASME Code requirements. Imposition of those requirements would, in our view, result in hardships or unusual difficulties without a compensating increase in the level of quality or safety. Therefore, pursuant to 10 CFR 50.55a(g)(1), we believe that the relief that the licensee requested from the pump and valve testing requirements of the 1977 edition of Section XI of the ASME Code through the Summer 1978 Addenda should be granted until our detailed review is completed. If completion of our review results in additional testing requirements, we will require that the licensee comply with them.

3.10 Seismic Qualification of Seismic Category I Mechanical and Electrical Equipment

In Supplement No. 4 to the Safety Evaluation Report we stated that the licensee's seismic qualification program was complete with the exception of the limit switches for the pressurizer safety valves. The licensee had proposed to use these limit switches to comply with TMI Item II.D.3 requirements for pressurizer safety valve position indication.

In a letter dated August 26, 1982, the licensee provided seismic qualification review team forms and other testing information to support the qualification of the limit switches for the seismic environment. The staff has reviewed the submittal and concluded that the limit switches have been seismically qualified.

Our review of the licensee's seismic qualification program for mechanical and electrical equipment is now complete. We conclude that the licensee's program meets all applicable staff criteria as discussed in Supplement No. 4 to the Safety Evaluation Report, and is acceptable.

3.11 Environment Qualification of Mechanical and Electric Equipment

As specified in Section 3.11 of Supplement No. 4 to the Safety Evaluation Report, we required the licensee to establish environmental qualification of equipment by updating the component evaluation work sheets when the noted deficiencies were resolved. In a letter dated September 29, 1982, the licensee submitted updated component evaluation work sheets on the following two items: (1) isolation fuse blocks in heat tracing panels and (2) Triax connectors. The licensee also provided supporting documentation for Item (2). In a letter dated November 3, 1982 the licensee stated that the implementation of the surveillance and maintenance program was completed and that the final link in the program was implemented on November 1, 1982.

Based on our review of the information supplied on the updated component evaluation work sheets and the supporting documentation, we concur with the licensee on the qualification of the above items, and therefore find this qualification documentation acceptable.

The licensee has provided justification for full power operation of the facility with the noted deficiencies for the remaining four items of Table 3-2 of Supplement No. 4 to the Safety Evaluation Report. We have reviewed the justification provided and conclude that operation of the facility until the scheduled resolution of these deficiencies (first major shutdown or refueling outage after June 1983) is acceptable.

5 REACTOR COOLANT SYSTEM

5.4 Component and Subsystem Design

5.4.2 Steam Generators

In Section 5.2.4 of Supplement No. 4 to the Safety Evaluation Report, we reported that there was a generic problem with vibration-induced wear of tubes in the preheater section of Model D steam generators, the type used in this facility. This generic problem had been experienced on two foreign facilities. The only other operating domestic plant with Model D steam generators is McGuire Unit 1. McGuire Unit 1 has been in operation since late 1981 but has been limited to 50% power for most of this period except for short intervals at 75% and 100% power.

In Supplement No. 4 to the Safety Evaluation Report, we reported the operating experience at that time with Model D steam generators. On the basis of that information we were able to conclude that the facility could safely operate at power levels up to 5% of full power. In Supplement No. 4 to the Safety Evaluation Report we noted that an industry program was in place to permanently correct the cause of the damaging tube vibration. Operating License NPF-12 contained condition 2.C.14 which required staff approval of a detailed program for operation, prior to exceeding 5% of full power, pending permanent modifications to the facility.

Since that time the generic program to develop a permanent modification has proceeded. At our request, the three utilities that own plants with Model D-2 and D-3 steam generators have formed an independent design review group to review the Westinghouse program for correcting the tube vibration problems. The group has held two meetings with Westinghouse thus far and the staff has participated in each of the meetings. The Westinghouse program has progressed to the point where a design modification has been selected. This modification which includes a flow distribution component termed a manifold is located internal to the steam generator and is intended to reduce feedwater inlet turbulence to acceptable levels and achieve nearly uniform flow at the inlet. The manifold has undergone extensive testing including tests in a full-scale facility in a foreign country. Westinghouse has concluded that the tests demonstrate the adequacy of the manifold to reduce the vibration to acceptable levels.

Following the preparation of the generic design report by Westinghouse, the independent design review group will review the report and provide their evaluation. The staff will then complete its review of the matter and make a determination of whether, and under what conditions, the facilities with Model D-2 and D-3 steam generators will be permitted to operate at 100% power. Such a decision will likely occur early in 1983. Until that time, the power levels of the facilities will be restricted accordingly to assure that damaging tube vibration does not occur.

On October 4, 1982, the licensee requested that the facility be permitted to operate at power levels up to approximately 50% of full power until the permanent modifications are made to the steam generators. The licensee also provided justification for this request.

The requested program for operation consists of starting up and performing ascension tests, through the 50 percent power level tests in accordance with standard startup procedures. Prior to startup, a multi-frequency eddy current inspection of the three outboard rows (47, 48, and 49) of steam generator tubes will be performed to provide a baseline for comparison with future inspections of this region of the tube bundle. Following the 50 percent tests, the facility will continue to operate at approximately 50 percent power not exceeding 50 percent of full power feedwater flow to the main feedwater nozzle, for a period of up to 2000 hours of operation, including the time at or above 5 percent power during power ascension testing. Eddy current inspection of selected tubes shall be performed at the end of this period.

The licensee has provided operating data from two foreign facilities and McGuire Unit 1 to justify its program for interim operation. The licensee has stated that tube motion accelerometers have been installed inside tubes on other plants adjacent to the feedwater inlet where tube wear has been observed. The data from this instrumentation indicate to the licensee that its proposed operating limit on main feedwater flow is a prudent interim operating condition. The licensee has also provided data on tube wear for those facilities. One of these facilities operated from December 1981 through July 1982 representing 3500 hours of operation at 50% of full power main feedwater flow. Eddy current testing data were available before the period, after 1500 hours, and at the end of the period with no significant wear indicated. The licensee also cites data from McGuire Unit 1 whose operating history at power levels at or above 50% power substantially exceeds that proposed for Virgil C. Summer Nuclear Station.

We have reviewed the program proposed by the licensee for interim operation of the Virgil C. Summer Nuclear Station and conclude that it is acceptable. We base this conclusion on the extensive tube wear data available at McGuire Unit 1 and other operating facilities with Model D steam generators which indicate that significant tube wear would not occur during the interim operating program proposed by the licensee for Virgil C. Summer Nuclear Station. We will condition the operating license to require NRC staff approval of the program for operation of the facility beyond the scope of the program proposed by the licensee in its October 4, 1982 letter to the staff.

13 CONDUCT OF OPERATIONS

13.3 Emergency Planning

The NRC staff conclusion regarding onsite and offsite capabilities to respond to an emergency at Virgil C. Summer Nuclear Station was provided in Supplement No. 3 to the Safety Evaluation Report. At that time the applicant had installed, but had not yet completed testing of, an alert and notification system to be used to promptly inform the public within the plume exposure pathway Emergency Planning Zone. On January 30, 1982, the licensee conducted a full, system-wide test of the alert and notification system involving all four affected counties and the State emergency planning organization. The system, including both the sirens and the emergency broadcast systems, was again fully tested on May 2, 1982, as part of the annual exercise. The installation and testing of the system was reported by the licensee in a letter to the staff dated September 23, 1982, and was confirmed by the staff as reported in NRC Inspection Report Nos. 82-03, 82-33 and 82-44.

In letters dated September 23 and 29, 1982, the licensee provided justification for and requested an exemption, in accordance with the provisions of 10 CFR §50.12(a) and §50.47(c), from literal compliance with one requirement of Section IV.F.1.b of Appendix E to Part 50. That section provides that a full-scale exercise shall be conducted:

> "for each site at which a power reactor is located for which the first operating license for that site is issued after July 13, 1982, within one year before the issuance of the first operating license for full power, and prior to operation above 5% of rated power of the first reactor which will enable each State and local government within the plume exposure pathway EPZ and each State within the ingestion pathway EPZ to participate."

Justification for the request was provided by the licensee and the exemption was approved in a letter dated November 2, 1982 from B. J. Youngblood to O. W. Dixon, Jr. The license will be conditioned to require that SCE&G conduct a limited emergency exercise similar to that conducted on May 5, 1982 but with full local government participation and partial State participation.

Based on the above and the findings previously reported in Supplement No. 3 to the Safety Evaluation Report, the NRC staff has concluded that the onsite and offsite emergency preparedness at Virgil C. Summer Nuclear Station meets the requirements of 10 CFR 50.47(b) (with the exception of the exemption discussed above), Regulatory Guide 1.101, Revision 2, NUREG-0654/FEMA-REP-1, Revision 1 and is acceptable for operation at power levels in excess of 5 percent of rated power.

17 QUALITY ASSURANCE

17.5 Independent Design Verification Program

17.5.1 Background

The Stone & Webster Engineering Corporation (SWEC) has completed its program for the independent verification of the seismic design of the Virgil C. Summer Nuclear Station by performing an in-depth evaluation of one representative subsystem--namely, the flow path of the turbine-driven portion of the emergency feedwater system to steam generator C. This technique is intended to provide increased assurance that the overall design and construction of the unit have been properly conducted. The program was accomplished in accordance with documented procedures, and it included three major tasks: a field walkdown (as built verification), a stress analysis and evaluation, and a quality assurance audit. For the quality assurance audit, SWEC reviewed the design controls of the architect/engineer for the unit, Gilbert Associates, Incorporated (GAI), including the interface controls between GAI and Teledyne Engineering Services (TES), an organization contracted by the applicant to perform pipe stress analyses using inputs supplied by GAI. This evaluation addresses the quality assurance audit of GAI and TES by SWEC, the independent verification contractor.

17.5.2 Quality Assurance Audit Results

The SWEC quality assurance audit was divided into three parts which involved: (1) review of the GAI design control program, (2) verification that the program had been properly implemented, and (3) confirmation of consistent utilization of response spectra.

Part 1 involved the determination of whether adequate design control procedures were in place consistent with Appendix B to 10 CFR Part 50. SWEC's review of the GAI design control program focused on the procedures available for the control of vendor and GAI drawings, of specifications, of changes to documents, of interfaces between GAI and subcontractors, of computer programs, and of design verification.

Review of the design control procedures established the lack of approval of the procedure for maintenance and distribution of a specification index. The procedure, though unapproved, was found by SWEC to be adequate and in use. In addition, GAI has an approved procedure which covers both the generation and distribution of project lists. The review also revealed that there was no formal procedure governing the verification/certification/use of computer programs early in the project. The audit identified the use (in 1972) of one computer program for which there was no evidence of verification/certification. Followup work showed that this was an isolated case and that the program was acceptable for its use. Other than these two items, adequate procedures were verified to exist.

Part 2 involved the determination of whether the design control procedures in effect were properly implemented in the design documents for the seismic design work. The SWEC audit found some apparent documentation problems, resulting in some pipe stress analyses using inputs inconsistent with program requirements. Investigation by SWEC indicated that these inconsistencies would not affect the design adequacy. In addition, the applicant has committed to review each pipe stress analysis to eliminate any additional documentation problems. The documentation problems found to date have all been of a nature which do not affect the design, and any additional problems are expected to be of the same nature. If a problem should be found which does have safety-significance, it will be reported to the NRC. This is acceptable to the staff. The SWEC audit also found some confusion in the application of damping factors. The applicant is to clarify the damping factors used for piping analysis (at least as conservative as specified in the Final Safety Analysis Report) in a revision to a document entitled, "Piping Engineering Section-Nuclear Criteria for Piping Stress Analysis and Pipe Support Design." This is acceptable to the staff.

Other than noted above, part 2 of the SWEC audit showed that the procedures associated with the activities reviewed during the audit were adequately implemented.

Part 3 of the audit showed that the response spectra utilized in the pipe stress analyses audited were consistent with (and in some cases, more conservative than) the dynamic structural analysis output.

17.5.3 Conclusion

Based on the quality assurance audit portion of the SWEC program for the independent verification of the seismic design of the Virgil C. Summer Nuclear Station by performing an in-depth evaluation of the flow path of the turbine-driven portion of the emergency feedwater system to steam generator C, it is concluded that the quality assurance program established and implemented by the architect/engineer of the facility was generally effective in controlling the seismic design activities for the facility. While deficiencies were identified in the program controls and in their implementation, the overall design activities were adequately performed so that no adverse impact on safety was found. These results provide increased assurance that the overall design of the facility has been properly conducted and provide an acceptable basis for granting authority to operate the facility at power levels up to and including full power.

22 TMI REQUIREMENTS

In Supplement No. 4 to the Safety Evaluation Report, we identified TMI issue II.B.4 as complete pending verification by the staff that certain procedural matters were satisfactorily completed by the licensee prior to operation in excess of 5% of full power. These matters, as discussed in Supplement No. 1 to the Safety Evaluation Report, have been verified by Region II in Inspection Report No. 82-55.

Since the issuance of Supplement No. 4 to the Safety Evaluation Report we have received additional information to permit us to complete our review of TMI Item II.D.3.

II.D.3 Relief and Safety Valve Position Indication

Refer to Section 3.10 of this supplement to the Safety Evaluation Report for our evaluation of this matter.

23 CONCLUSION

We have determined that the amendment to the license supported by this supplement to the Safety Evaluation Report will not result in any environmental impacts other than those evaluated in the Final Environmental Statement since these actions are encompassed by the overall action evaluated in the Final Environmental Statement.

Prior public notice of the overall action involving issuance of an operating license amendment authorizing operation above 5 percent of full power, was published in the FEDERAL REGISTER on April 18, 1977 (42 FR 20203). The staff evaluation of the safety of the overall action is given the Safety Evaluation Report and its supplements (NUREG-0717).

Further, there is reasonable assurance that the health and safety of the public will not be endangered by operation in the manner authorized by the amendment, the activities authorized by the amendment will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. We, therefore, conclude that the proposed amendment is acceptable.

APPENDIX A

CHRONOLOGY OF NRC STAFF RADIOLOGICAL SAFETY REVIEW

July 2, 1982	Letter from applicant concerning offsite dose calculation manual.
July 2, 1982	Letter from applicant concerning process control program.
July 6, 1982	Letter from applicant concerning cable separation.
July 7, 1982	Letter from applicant concerning FSAR Chapter 14 tests.
July 8, 1982	Letter from applicant concerning reactor coolant system temperature instrumentation.
July 9, 1982	Letter from Stone & Webster concerning independent seismic design verification status report.
July 12, 1982	Letter from applicant transmitting Amendment No. 33 to the FSAR.
July 12, 1982	Letter to applicant concerning Technical Specifications.
July 12, 1982	Letter from applicant concerning record keeping.
July 13, 1982	Letter from applicant concerning thermal sleeves.
July 19, 1982	Letter from applicant concerning radiation monitoring instrumentation.
July 19, 1982	Letter from applicant concerning boron dilution.
July 20, 1982	Letter from applicant concerning core subcooling monitor (TMI Item II.F.2)
July 21, 1982	Representatives from Westinghouse, Argonne National Laboratory and NRC met in Bethesda, Md., concerning Model D and E steam generators. (Summary issued July 23, 1982.)
July 23, 1982	Letter from applicant concerning Technical Specifications.
July 23, 1982	Letter from applicant concerning emergency preparedness.
July 23, 1982	Letter from applicant concerning earthquake instrumentation.

July 23, 1982	Letter from applicant concerning Technical Specifications.
July 28, 1982	Letter from applicant concerning inadvertent boron dilution.
July 29, 1982	Letter from applicant concerning special low power physics test procedures.
July 29, 1982	Letter from applicant concerning Technical Specifications.
July 29, 1982	Letter from applicant concerning Technical Specifications.
July 30, 1982	Letter from applicant concerning safety and relief valve test report, NUREG-0737 Item II.D.1.
July 30, 1982	Letter from applicant concerning Technical Specifications.
August 2, 1982	Letter to applicant concerning monitoring program for service water pond structures.
August 3, 1982	Letter to applicant transmitting a review copy of the Technical Specifications.
August 4, 1982	Letter from applicant concerning reactor coolant system temperature instrumentation.
August 6, 1982	Letter to licensee transmitting Facility Operating License NPF-12 for 100% power, restricted to 5% power until further Commission approval.
August 12, 1982	Letter to licensee transmitting 2 copies of Supplement No. 4 to the Safety Evaluation Report.
August 12, 1982	Letter from licensee concerning inservice test program for pumps and valves.
August 13, 1982	Letter from licensee concerning physical security plan.
August 16, 1982	Letter to licensee transmitting 20 copies of Supplement No. 4 to the Safety Evaluation Report.
August 17, 1982	Letter from licensee concerning steam generator inspection ports, License Condition 2.C(13).
August 18, 1982	Letter from licensee requesting changes to Technical Specifications.
August 20, 1982	Letter to licensee transmitting Amendment No. 1 to Facility Operating License NPF-12 concerning fire-rated assemblies w/Technical Specifications change page.
August 23, 1982	Letter from licensee requesting an amendment to Operating License NPF-12 for relief from Technical Specification 3/4 3.7.10.

August 24, 1982	Letter from licensee requesting an amendment to Operating License NPF-12 for administrative changes to Technical Specifications.
August 24, 1982	Letter from licensee concerning physical security plan for the protection of nuclear material of low strategic significance.
August 25, 1982	Representatives from NRC, Duke Power Company, South Carolina Electric & Gas Company, and Tennessee Valley Authority met in Bethesda, Maryland to review with NRR management the proposed scope and content of the safety evaluation to be developed by the independent design review group on Model D steam generator modifications. (Summary issued August 27, 1982.)
August 26, 1982	Letter from licensee concerning seismic qualification, License Condition 23.
August 27, 1982	Letter to licensee transmitting Amendment No. 2 to operating license NPF-12 correcting certain inconsistencies in the Technical Specifications regarding containment radiation monitors and the containment purge and exhaust isolation.
September 1, 1982	Representatives from NRC and SCE&G met in Bethesda, Maryland to review the licensee's program for responding to License Condition 25 regarding confirmatory seismic analysis. (Summary issued September 2, 1982.)
September 3, 1982	Letter from licensee concerning Cadweld allegation.
September 15, 1982	Representatives from NRC, SCE&G and Dames & Moore met in Bethesda, Md., to review the licensee's program for responding to License Condition 25 regarding confirmatory seismic analyses. (Summary issued September 24, 1982.)

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Docket No. 50-395

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16. ABSTRACT (200 words or less)

The Safety Evaluation Report and its supplements pertain to the application for a license to operate the Virgil C. Summer Nuclear Station filed by the South Carolina Electric & Gas Company on December 10, 1976. The site is located in Fairfield County, South Carolina.

The Safety Evaluation Report related to operation was issued in February 1981. Supplement No. 1 containing updated information since issuance of the Safety Evaluation Report was issued in April 1981. Supplement No. 2 updating the emergency planning information was issued in May 1981. Supplement No. 3 and Supplement No. 4 updated previous information and closed out formerly outstanding issues and were issued in January 1982 and August 1982, respectively. This Supplement (5) discusses operation of the station above 5 percent power.

17. KEY WORDS AND DOCUMENT ANALYSIS

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17b. IDENTIFIERS/OPEN-ENDED TERMS

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DISTRIBUTION FOR AMENDMENT #5 TO FACILITY OPERATING LICENSE NPF-12

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

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