SCE&G
A SCANA Company

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John L. Skolds
Vice President
Nuclear Operations

June 4, 1992

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

Attention: Mr. G. F. Wunder

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
PROPOSED SCHEDULE FOR SUBMITTAL OF INFORMATION
SUPPORTING STEAM GENERATOR REPLACEMENT (REM 6000)

On January 9, 1992, South Carolina Electric & Gas Company (SCE&G) met with NRC representatives to discuss the Virgil C. Summer Nuclear Station (VCSNS) Steam Generator Replacement Project. In this meeting, SCE&G presented licensing package options for NRC review of material supporting required Technical Specification changes. It was agreed that early submittal of discrete packages of analyses would assist NRC in supporting the SCE&G Fall 1994 replacement schedule. Attached is a proposed, detailed schedule for submittal of information to support the replacement effort.

SCE&G plans to make three submittals.

1. August 31, 1992 - Delta 75 Design Requirements and Performance Data (requested by Mr. Hermann).

2. April 30, 1993 - Steamline break and loss of coolant accident mass and energy releases (inside and outside containment), the associated containment integrity and equipment qualification effects analyses, steam generator tube rupture evaluation, and the majority of the radiological analyses. This corresponds to sections 3.3.5, 3.4, 3.5 and 3.8 of the attachment.

3. October 29, 1993 - The remainder of the analyses/information listed in the attachment will be provided.

The dates listed above are target dates. Major changes to these dates or the submittal package contents will be communicated to the NRC in a timely manner. SCE&G would appreciate any suggestions/comments on the proposed submittals. If you have any questions on this subject, please contact Ms. April Rice at 803-345-4232.

Very truly yours,

John L. Skolds

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ARR:lcd
Attachment

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File (810.39)

Proposed Submittal Schedule to Support
Steam Generator Replacement Technical Specification Changes

Title	Submittal		
	1	2	3
REPLACEMENT STEAM GENERATOR SAFETY EVALUATION List of Tables List of Figures List of Acronyms Executive Summary			X
1 Introduction 1.1 Content of Current Licensing Basis 1.2 Purpose of Identified Technical Specifications 1.3 Description of the Proposed Changes 1.4 Discussion of Changes Request			X
2 Basis for Evaluations/Analyses Performed 2.1 Design Power Capability Parameters 2.1.1 Discussion of Parameters 2.1.2 References 2.2 NSSS Design Transients 2.3 Control/Protection System Setpoints			X
3 Safety Evaluations/Analyses			
3.1 LOCA (Large Break and Small Break) 3.1.1 Large Break LOCA 3.1.2 Small Break LOCA 3.1.3 References			X
3.2 LOCA Hydraulic Forcing Functions Impact			X
3.3 Non-LOCA Safety Evaluations 3.3.1 Introduction 3.3.2 Reactor Protection System and Engineered Safety Features Setpoints Assumed in Evaluations 3.3.3 Methodology 3.3.3.1 Initial Conditions 3.3.3.2 RCCA Insertion Characteristics 3.3.3.3 Reactivity Coefficients			X

Target Submittal Dates:

1: August 31, 1992

2: April 30, 1993

3: October 29, 1993 [Final Submittal]

Title	Submittal		
	1	2	3
3.3.3.4 Residual Decay Heat			
3.3.3.5 Computer Codes Utilized			
3.3.4 Non-LOCA Safety Evaluations			
3.3.4.1 Uncontrolled RCCS Bank Withdrawal from a Subcritical Condition			
3.3.4.2 Uncontrolled RCCS Bank Withdrawal at Power			
3.3.4.3 Rod Cluster Control Assembly Misalignment			
3.3.4.4 Uncontrolled Boron Dilution			
3.3.4.5 Startup of an Inactive Loop			
3.3.4.6 Loss of Reactor Coolant Flow			
3.3.4.7 Loss of External Electrical Load			
3.3.4.8 Loss of Normal Feedwater Flow			
3.3.4.9 Loss of Offsite Power to the Station Auxiliaries			
3.3.4.10 Excessive Heat Removal Due to Feedwater System Malfunctions			
3.3.4.11 Excessive Load Increase Incident			
3.3.4.12 Accidental Depressurization of the Reactor Coolant System			
3.3.4.13 Accidental Depressurization of the Main Steam System			
3.3.4.14 Inadvertent Operation of the ECCS			
3.3.4.15 Major Rupture of Main Steam Line			
3.3.4.16 Major Rupture of a Main Feedwater Line			
3.3.4.17 Single RCP Locked Rotor			
3.3.4.18 Rupture of Control Rod Drive Mechanism Housing			
3.3.4.19 Anticipated Transients Without Trip (ATWS)			

Target Submittal Dates:

1: August 31, 1992

2: April 30, 1993

3: October 2[illegible], 1993 [Final Submittal]

Title	Submittal 1	Submittal 2	Submittal 3
3.3.5 Steamline Break Mass/Energy Releases		X	
3.3.5.1 Inside Containment			
3.3.5.2 Outside Containment			
3.3.6 Conclusions of Non-LOCA Safety Evaluations			
3.3.7 References			
3.4 Containment Analyses		X	
3.4.1 Short-Term Containment Analysis			
3.4.2 Long-Term Containment Analysis			
3.4.2.1 Main Steamline Break Containment Integrity			
3.4.2.2 LOCA Containment Integrity			
3.4.3 References			
3.5 Steam Generator Tube Rupture		X	
3.6 Post-LOCA Hot Leg Recirculation Time			X
3.6.1 Introduction			
3.6.2 Event Description			
3.6.3 Methodology			
3.6.4 Results			
3.6.5 References			
3.7 Reactor Cavity Pressure Analysis			X
3.8 Radiological Analysis		X	X
3.8.1 Introduction			
3.8.2 Source Terms			
3.8.3 Loss of Offsite Power			
3.8.4 Waste Gas Decay Tank Rupture			
3.8.5 Break in a CVCS Instrument Line			
3.8.6 Large Break LOCA			
3.8.7 Main Steam Line Break			
3.8.8 Steam Generator Tube Rupture			
3.8.9 Locked Rotor			
3.8.10 Fuel Handling Accident			
3.8.11 RCCA Ejection			
3.8.12 References			

Target Submittal Dates:

1: August 31, 1992

2: April 30, 1993

3: October 29, 1993 [Final Submittal]

Title	Submittal		
	1	2	3
3.9 Primary Components Evaluations			X
3.9.1 Reactor Vessel			
3.9.1.1 Reactor Vessel Structural Evaluation			
3.9.1.2 Reactor Vessel Integrity			
3.9.2 Reactor Internals			
3.9.3 Steam Generators			
3.9.3.1 Thermal-Hydraulic Performance Evaluation			
3.9.3.2 Structural			
3.9.4 Pressurizer			
3.9.5 Reactor Coolant Pumps			
3.9.6 Reactor Coolant Piping and Suports			
3.9.7 Control Rod Drive Mechanism			
3.9.8 Application of Leak-Before-Break Methodology			
3.9.9 Conclusions			
3.9.10 References			
3.10 Fluid and Auxiliary Systems Evaluations			X
3.10.1 Introduction			
3.10.2 Discussion of Evaluations Performed			
3.10.2.1 Fluid Systems Evaluation			
3.10.2.2 Auxiliary Equipment Evaluation			
3.10.2.3 NSSS/Balance of Plant Interface			
3.10.3 Conclusions			
3.11 Fuel Structural Evaluation			X
3.11.1 Fuel Assembly Structural Evaluation			
3.11.2 Fuel Rod Structural Evaluation			
3.12 Technical Specification/Reactor Trip & ESF Impact			X

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Title	Submittal		
	1	2	3
LICENSING REVIEW CRITERIA Appendix 1 10CFR 50.59 (Evaluation of an Unreviewed Safety Question/Response to Seven Questions) Appendix 2 10 CFR 50.92 (Significant Hazards Determination for Issuance of Amendment) Appendix 3 10CFR 50.36 (Technical Specification Modifications)			X
Appendix 4 Delta 75 Design Requirements and Performance Data	X		

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