SOUTH CAROLINA ELECTRIC & GAS COMPANY
POST OFFICE 764
COLUMBIA, SOUTH CAROLINA 29218

O. W. DIXON, JR.
VICE PRESIDENT
NUCLEAR OPERATIONS

July 30, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Operating License No. NPF-12
Generic Letter 82-33
EOP Generation Package

Dear Mr. Denton:

In response to the requirements of Generic Letter 82-33, "Supplement 1 to NUREG-0737-Requirements for Emergency Response Capability," Section 7, the Emergency Operating Procedure (EOP) Generation Package for the Virgil C. Summer Nuclear Station is herewith submitted as enclosures I through III to this letter. South Carolina Electric and Gas Company's (SCE&G) upgraded EOPs are based on Westinghouse Owners Group Emergency Response Guidelines (ERGs), revision 1, and were developed with the assistance of Essex Corporation to ensure human factor concerns were adequately addressed.

Enclosure I compares the Virgil C. Summer Nuclear Station with the ERG generic plant described in the Executive Volume of the ERGs. Enclosure II outlines the specific areas where SCE&G's EOPs differ from the Westinghouse Owners Group ERGs, revision 1. Enclosure III is the Virgil C. Summer Nuclear Station Administrative Procedure (SAP-207) which details the development of EOPs.

SAP-207 addresses requirements found in Generic Letter 82-33, Item 7.2.b. Section 6, "Technical Guidelines," details elements of the plant specific technical guidelines and how these guidelines are used for procedure development. The Emergency Operating Procedures Writers Guide, Attachment I to this procedure, is based on guidelines published by the Institute of Nuclear Power Operations (INPO). Section 10.0, "EOP Validation," describes the procedure validation program which was also designed using INPO guidelines. The training program is addressed in Section 11.0, "EOP Training."

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SCE&G's plant specific procedure validation program is scheduled to begin in August 1984. Formal operator training of the upgraded procedures should begin in October 1984.

As stated in our March 22, 1983 letter to the Staff, justification for manual reactor coolant pump (RCP) trip was to be submitted along with this EOP generation package. This justification is found in the Westinghouse Report, "Justification of Manual RCP Trip for Small Break LOCA Events," transmitted to the NRC by the Westinghouse Owners Group in a letter dated March 12, 1984.

If you should have any questions, please advise.

Very truly yours,

O. W. Dixon, Jr.

AMM/OWD/gj
Enclosures

cc: V. C. Summer
T. C. Nichols, Jr./O. W. Dixon, Jr.
E. H. Crews, Jr.
E. C. Roberts
W. A. Williams, Jr.
D. A. Nauman
J. P. O'Reilly
Group Managers
O. S. Bradham
C. A. Price
C. L. Ligon (NSRC)
K. E. Nodland
R. A. Stough
G. Percival
C. W. Hehl
J. B. Knotts, Jr.
NPCF
File

ENCLOSURE I
CGSS-01-1004-NO

COMPARISON OF THE VIRGIL C. SUMMER NUCLEAR STATION TO THE HIGH PRESSURE REFERENCE PLANT

The V. C. Summer Nuclear Station is a Westinghouse Model 312 three loop nuclear power plant rated at 2775 MWT with a General Electric 950 Turbine/Generator rated at 950 MWE.

Differences between the V. C. Summer Nuclear Station and the High Pressure Reference Plant are as follows: (Refer to ERG Executive Volume).

2.2	Safety Injection Signal	-	In addition to the reference plant SI signals there is a steamline Differential Pressure SI at 97 psi. This does not have a block/reset function.
2.2	Safety Injection Block Signal	-	Low Pressure Ste block/reset is c P-12, Low-Low R(
2.2	Turbine Driven EF Pump Start	-	Loss of Power f buses versus lo RCP Buses.
2.2	Main Steam Isolation	-	High Steam Flow coincident with Low-Low RCS Tavg versus High Steam Pressure rate below P-11.
2.2	Main Steam Isolation Block	-	Dependant on Low-Low Tavg (P-12) versus Low Pressure (P-11). Steamline ΔP SI is not blockable.
2.7	Reactor Coolant System	-	3 loop versus 4 loop. 3 Pzr PORV versus 2 PZR PORV. Cold Overpressure Protection System is automatically placed in service. Proposal currently submitted to eliminate COPS.

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2.8 Hot Leg Recirculation Mode	-	Only recirculates from the Rx. Bldg. sump to the RCS hotlegs.
2.8 Charging/SI Subsystem	-	A seperate line is provided for cold leg recirculation in addition to thru the BIT.
2.8 High-Head SI Subsystem	-	Not applicable to V. C. Summer.
2.8 SI-Accumulator Subsystem	-	3 Charging/SI pumps and no positive displacement pump. Seal return and charging pump mini-flow returns to the volume control tank, not charging pump suction.
2.11 Component Cooling Water System	-	Does not provide water to the containment fan coolers.
2.12 Service Water System	-	Provides cooling water to the containment fan coolers via Service Water Booster Pumps. Provides automatic emergency makeup water to the Emergency Feedwater System.
2.13 Containment Spray System	-	Includes a Sodium Hydroxide System.
2.14 Containment Atmosphere Control System	-	Includes both charcoal and HEPA filters.

2.16	Main Feedwater & Condensate System	- The main feedwater system is also isolated on a low flow, coincident with low feedwater temperature or high Intermediate Building sump levels. All main feedwater pumps are turbine driven. Between the condensate pumps and feedwater pumps are four (4) Feedwater Booster Pumps and a deaerator. Shutoff head of the Feedwater Booster Pumps is approximately 350 psig.
2.17	Emergency (Auxiliary) Feedwater System	- The Emergency Feedwater System is completely separated from the Main Feedwater system and injects to steam generators via a separate nozzle. Any Emergency Feedwater Pump can supply all three Steam Generators. The alternate water supply is the Service Water System.
2.18	Steam Generator Blowdown	- Includes automatic diversion to a holdup system in the event of high blowdown radiation.

2.24 Electrical Power System	-	The electrical power supply consists of two independent off-site power supplies feeding two independent on-site emergency power supplies. During a blackout condition non-essential loads are locked-out on the emergency A. C. buses.
2.25 Pneumatic Power System	-	Two of the three Pressurizer PORV's are supplied by a high pressure nitrogen system with accumulators instead of control air.

Instrument and Control Requirements (Table 3) are consistent with the reference plant.

Comparison of the Virgil C. Summer Nuclear Station Emergency Operating Procedures to the Westinghouse Owners Group Emergency Response Guidelines.

The Virgil C. Summer Nuclear Station Emergency Operating Procedures (EOP's) are consistent with the Westinghouse Owners Group Emergency Response Guidelines (ERG's), Rev. 1, with the following exceptions:

1. ERG Procedures ES-0.3, "Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS)", is not included in the EOP set. The difference between this procedure and ES-0.2, "Natural Circulation Cooldown", includes monitoring reactor vessel level indication and provides direction to the Operators for dealing with reactor vessel void formation. This is included in EOP-1.3, "Natural Circulation". Since the Virgil C. Summer Nuclear Station is a T COLD plant as documented in the ERG Background document for ES-0.2, the lack of a plant specific EOP based on ES-0.3 is not considered a significant safety concern.

2. ERG Procedure FR-H.2, "Response To Steam Generator Over-Pressure", is not included in the EOP set. To reach S/G overpressure higher than the highest steamline safety valve setpoint requires the failure of five (5) code safety valves and a steamline power relief valve. The ERG procedure response basically requires the Operator to dump steam to relieve the high pressure; however, if steam dump capabilities were available the conditions would not exist to begin with. Neither the initiating conditions nor the recommended Operator responses are considered credible. The ERG FR-H.2 Background document

acknowledges that most plants have five (5) steamline safety valves and does not postulate how the entry conditions could conceivably be reached.

A complete listing of plant specific EOP's versus the applicable ERG procedure is included as Attachment I to this enclosure. Based on the results of the validation program, some consolidation of the procedures may become necessary. Therefore the final EOP numbering as shown on this listing may change.

EOP TITLES AND NUMBERING

NUMBER	TITLE	ERG REV. 1
EOP-1.0	Reactor Trip/ Safety Injection Actuation	E-0
EOP-1.1	Reactor Trip Recovery	ES-0.1
EOP-1.2	Safety Injection Recovery	ES-1.1
EOP-1.3	Natural Circulation	ES-0.2
EOP-1.4	Rediagnosis	ES-0.0
EOP-2.0	Loss Of Reactor Or Secondary Coolant	E-1
EOP-2.1	Post-LOCA Cooldown And Depressurization	ES-1.2
EOP-2.2	Transfer To Cold Leg Recirculation	ES-1.3
EOP-2.3	Loss Of Emergency Coolant Recirculation	ECA-1.1
EOP-2.4	Total Loss Of The Residual Heat Removal System	-
EOP-2.5	LOCA Outside Containment	ECA-1.2
EOP-3.0	Faulted Steam Generator Isolation	E-2
EOP-3.1	Uncontrolled Depressurization Of Generators	ECA-2.1
EOP-4.0	Steam Generator Tube Rupture	E-3
EOP-4.1	Post-SGTR Cooldown	ES-3.1, 3.2, 3.3
EOP-4.2	SGTR With Loss Of Reactor Coolant: Subcooled Recovery	ECA-3.1
EOP-4.3	SGTR With Loss Of Reactor Coolant: Saturated Recovery	ECA-3.2

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Attachment I
Enclosure II
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EOP TITLES AND NUMBERING

NAME	TITLES	ERG REV. 1
EOP-4.4	SGTR. Without Pressurizer Pressure Control	ECA-3.3
EOP-6.0	Loss Of All AC Power	ECA-0.0
EOP-6.1	Loss Of All AC Power Recovery Without SI Required	ECA-0.1
EOP-6.2	Loss Of All AC Power Recovery With SI Required	ECA-0.2
EOP-7.0	Refueling Emergency	-
EOP-8.0	Control Room Evacuation	-
EOP-9.0	High Radiation Outside Containment	-
EOP-10.0	Malfunction Of Control System	-
EOP-11.0	Emergency Boration	-
EOP-12.0	Monitoring Of Critical Safety Functions	F-01 thru 6.0
EOP-13.0	Response To Abnormal Nuclear Power Generation	FR-S.1
EOP-13.1	Response To Loss Of Core Shutdown	FR-S.2
EOP-14.0	Response To Inadequate Core Cooling	FR-C.1
EOP-14.1	Response To Degraded Core Cooling	FR-C.2
EOP-14.2	Response To Saturated Core Cooling Conditions	FR-C.3
EOP-15.0	Response To Loss Of Secondary Heat Sink	FR-H.1

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Attachment I
Enclosure II
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EOP TITLES AND NUMBERING

NUMBER	TITLE	ERG REV. 1
EOP-15.1	Response To Steam Generator High Level	FR-H.3
EOP-15.2	Response To Loss Of Normal Steam Release Capabilities	FR-H.4
EOP-15.3	Response To Steam Generator Low Level	FR-H.5
EOP-16.0	Response To Imminent Pressurized Thermal Shock	FR-P.1
EOP-16.1	Response To Anticipated Thermal Shock	FR-P.2
EOP-17.0	Response To High Reactor Building Pressure	FR-Z.1
EOP-17.1	Response To Reactor Building Flooding	FR-Z.2
EOP-17.2	Response To High Reactor Building Radiation Level	FR-Z.3
EOP-18.0	Response To High Pressurizer Level	FR-I.1
EOP-18.1	Response To Low Pressurizer Level	FR-I.2
EOP-18.2	Response To Void In Reactor Vessel	FR-I.3