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

T. C. NICHOLS, JR.
VICE PRESIDENT AND GROUP EXECUTIVE

August 27, 1980

(Nuclear Operations)

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
Radiation Protection Questions -
Sheilding

Dear Mr. Denton:

South Carolina Electric and Gas Company, acting for itself and as agent for South Carolina Public Service Authority, herewith provides forty-five (45) copies of our responses to questions raised in an August 7, 1980 phone call with your Mr. John Minns regarding shielding of personnel after an accident. The responses are in the form of an FSAR change and will be included in the next FSAR amendment.

If you require additional information, please let us know.

Very truly yours,

T. C. Nichols, Jr.

T. C. Nichols, Jr.

RBC:TCN:rh

cc: G. H. Fischer
E. H. Crews, Jr.
V. C. Summer
W. A. Williams, Jr.
B. A. Bursey
D. A. Nauman
O. S. Bradham
M. A. Barnisin
R. B. Clary
H. T. Babb
O. W. Dixon
PRS/Woods
NPCF/Whitaker
File

8009030493

331.34 Provide a summary of the shielding design review required by our letter dated November 9, 1979, implementing the Lessons Learned item 2.1.6.b of NUREG-0578, and provide a description of the results of this review. Include in your description.

a. source terms used in the evaluation (NUREG-0578 specified that source terms in Regulatory Guide 1.3, 1.4 and 1.7 be used).

b. systems assumed to contain high levels of radioactivity in a post-accident situation including, but not limited to, containment residual heat removal, safety injection, CVCS, demineralizers, charging systems, reactor coolant filters, seal water filters, sample lines, liquid radwaste systems, and gaseous radwaste systems. If any of these systems or others that could contain high radioactivity were excluded, explain why such systems were excluded from review. You should verify that direct radiation from field run piping and scattered radiation (such as shine over shield walls) were included in the analysis.

c. specify areas where access is considered necessary for vital system operation after an accident. Your evaluation of areas to determine the necessary vital areas should include but not be limited to, consideration of the control room, Technical Support Center, Operational Support Center, recombiner hookup and control stations, hydrogen purge control stations, containment isolation reset control area, sampling and sample analysis areas, manual ECCS alignment area, motor control centers, instrument panels, emergency power supplies, security center and radwaste control panels. If any of these areas were not considered areas where access was necessary after an accident, explain why they are excluded.

d. Designation of the codes used for analysis, such as ORIGEN, ISOSHIELD, QUAD or others.

e. The projected doses to individuals for necessary occupancy times in vital areas.

f. A brief description of the proposed plant modifications resulting from the design review and confirmation that these modifications will be complete by full power operation.

RESPONSE

Refer to the revised Sections 12.1.1 and 12.1.3, and Appendix 12A. The report prepared for the shielding design review that was performed in response to NUREG-0578 has been incorporated into the V. C. Summer FSAR as Appendix 12A. The contents of Appendix 12A provide a comprehensive explanation of the shielding design review.

a. The origin of the source terms used in the NUREG-0578 shielding design review are explained in Section 12A.3-1 of Appendix 12A. The initial isotopic inventory of the shielding source terms are presented in Table 12A.3-2 of Appendix 12A.

b. The following systems either in their entirety or in part are assumed to contain high levels of radioactivity in a post-accident situation:

(1) Residual Heat Removal System

(2) Safety Injection System

(3) Reactor Building Spray System

(4) Reactor Coolant System

(5) Post-accident Hydrogen Removal System

This system is of concern only for the sections involved in the collection of the reactor building atmosphere sample.

(6) Nuclear Sampling System

This system is of concern for its role in the collection of the following samples: primary coolant hot legs, pressurizer liquid and steam spaces, residual heat removal loop.

(7) Chemical and Volume Control System

This system will be of concern only for those system sections involving high head injection and seal water injection. The high head injection section involves the charging/safety injection pumps and the associated piping required to charge the reactor coolant system. The pumps will initially receive their suction supply from the refueling water storage tank and then from the reactor building recirculation sump during the recirculation mode. The seal water injection section of the CVCS will be isolated prior to the start of the recirculation mode and will, therefore, not contain the high radioactive levels found in the reactor building recirculation sump. The CVCS sections involving the letdown and purification functions are not required for the post-accident situation, and will be isolated from the high levels of radioactivity contained in the coolant and sump liquids. The demineralizers and filters of the CVCS have been excluded from the review, since the amount of fuel damage present will render their usage undesirable.

(8) Radwaste Gas Handling System

Although not designed for reactor coolant degasification under post-accident conditions, the waste gas system was evaluated for the exposures emitted and received under the post-accident conditions.

The following systems are excluded from the review because they will be isolated upon the occurrence of an accident and will therefore not contain the high radiation levels resulting from the fuel damage:

(1) Radwaste Solids Handling System

(2) Radwaste Liquid Handling System

(3) Spent Fuel Cooling System

(4) Boron Recycle System

(5) Thermal Regeneration System

The direct radiation dose from all piping of significance to an area has been considered in the calculation of the doses presented in Appendix 12A. Contributions to the dose from scattered radiation were considered in the review during the calculation of both the dose and dose rates for the access areas in the plant.

c. The areas where access is considered necessary for the operation of vital systems after an accident are given in Sections 12A.4.1 through 12A.4.19.

For those areas specifically requested for consideration:

(1) Control Room = refer to Section 12A.4.15

(2) Technical Support Center = refer to Section 12A.4.14

(3) Operational Support Center = part of the Control Room, refer to Section 12A.4.15.

(4) Recombiner hookup and control station = refer to Sections 12A.4.11 and 12A.4.12

(5) Hydrogen purge control stations = the alternate purge system controls of the Post Accident Hydrogen Removal System are located in the Control Room, refer to Section 12.A.4.15.

(6) Containment isolation reset control area = in the Control Room, refer to Section 12.A.4.15

(7) Sampling and sample analysis areas = refer to Sections 12A.4.13, 12A.4.16, 12A.4.17, and 12A.4.18

(8) Manual ECCS alignment areas = refer to Sections 12A.4.3, 12A.4.7, 12A.4.9, and 12A.4.10

(9) Motor Control Centers = refer to Sections 12A.4.7 and 12A.4.10

(10) Instrument panels = refer to Sections 12A.4.3, 12A.4.5, 12A.4.6, 12A.4.8, 12A.4.11, 12A.4.12, and 12A.4.15

(11) Emergency power supplies = refer to Sections 12A.4.7 and 12A.4.10

(12) Security center = part of Service Building. Not a vital access area.

(13) Radwaste control panels = refer to Sections 12A.4.5 and 12A.4.8

d. Computer codes used in the analysis are INHEC, RWDS, and SDC.

(1) INHEC

For information on INHEC, refer to the following Topical Report:

"Computation of Radiological Consequences Using
INHEC Computer Program"
Gilbert/Commonwealth Companies
Topical Reports GAI-TR-101P-A (Proprietary)
and GAI-TR-101-A (Non-Proprietary)
March, 1976.

(2) RWDS

The RWDS computer code performs the calculation of the activity levels in both the fluid and components of a system. This code incorporates numerous options which allow for the analysis of various systems with a broad spectrum of characteristics. Using this computer code, it is possible to generate the isotopic inventory of the individual system components, as well as the system flow stream, after the individual isotopic components of the stream have been decayed independently of the others. Capable of handling the activities of several streams simultaneously, the RWDS code can also generate the total activity present when several streams are merged into a single stream. The activities of the inventory are summed by energy group, and by liquid and gaseous activity, so as to generate a input form from the list of isotopes and activities suitable for use in shielding analysis. The activities can be obtained in terms of either specific or total activities present in all the contributing streams.

(3) SDC

The SDC computer code is designed to calculate the gamma-ray shielding requirements for nuclear applications. The integration of the basic exponential attenuation point kernel over the various source geometries provides an uncollided gamma-ray flux. The biological dose rate is obtained by multiplying this uncollided flux by the product of a flux-weighted buildup factor and a dose conversion factor. The major options in the computer code permit calculation of either the required shield thickness when a dose rate is specified, or the dose rate when the shield thickness is given. The calculation of dose rates from unshielded sources as well as surface intensities for cylinders and spheres is also included.

e. Refer to Table 12A.4-2 for the radiation dose to individuals in the vital areas.

f. Plant modifications that may be required due to the review are currently under study and will be provided in a later submittal.

12A.3 SOURCE TERMS AND CALCULATIONAL METHODOLOGY

12A.3.1 SOURCE TERMS

12A.3.1.1 Basis for the Source Terms

The activity releases assumed in this review are based on the assumptions and regulatory positions contained in the Regulatory Guides 1.4 and 1.7. The activity assumed for liquid source term calculation is based on 100% of the noble gas inventory, 50% of the halogen core inventory and 1% of all other nuclides in the core inventory. The activity assumed for gaseous source term calculation is based on 100% of the noble gas core inventory and 25% of the halogen core inventory.

12.A.3.1.2 Liquid Source Terms

Two liquid sources are considered in the design review: (1) the undiluted fluid as found within the reactor coolant system, and (2) the diluted fluid as found within the reactor building recirculation sump. The first source term, undiluted reactor coolant, is required in the examination of those systems whose flow originates from either the Reactor Coolant System or an auxiliary system containing the undiluted primary fluid. The source terms are based on the dilution of the liquid activity inventory discussed in the first paragraph with the fluid volume of the Reactor Coolant System. This source is used for the examination of the reactor coolant fluid sampling section of the Nuclear Sampling System, and those portions of the Chemical and Volume Control System associated with the degasific tion of the reactor coolant fluid.

In the secoid liquid source term, consideration is given for the dilution of the liquid activity inventory discussed in the first paragraph with the fluid volume contained in the reactor building recirculation sump. The minimum fluid volume expected in the sump, and the individual contributors to that volume, are given in Table 12A.3-1. These source terms are utilized in the examination of those systems which receive their fluid supply from the reactor building recirculation sump. Those systems which are considered in the review are:

Residual Heat Removal System

Reactor Building Spray System

Safety Injection System

Nuclear Sampling System (RHR process fluid sample)

12A.3.1.3 Gas Source Terms

The gaseous source terms were determined for containment and the waste gas system using the activity releases described in Section 12A.3.1.1.

The containment airborne source term was based on the dilution of the gaseous activity inventory by the containment free volume atmosphere.

Although not designed for reactor coolant degasification under post-accident conditions, the waste gas system was evaluated for the exposures emitted and received under the post-accident conditions. The waste gas system is designed to remove the fission product gases from the reactor coolant contained in the Volume Control Tank (VCT). The amount of fission gases removed from the reactor coolant in the VCT and collected by the waste gas system can be related to the amount entering the VCT as follows:

(1) Stripping Efficiency (SE):

$$SE = \frac{C_R - C_L}{C_R - C_{L_{eq}}}$$

(2) Stripping Fraction (SF):

$$SF = \frac{C_R - C_L}{C_R}$$

where C_R = the gas concentration in the reactor coolant liquid entering the Volume Control Tank,

C_L = the gas concentration in the reactor coolant liquid leaving the Volume Control Tank,

$C_{L_{eq}}$ = the gas concentration in the reactor coolant liquid leaving the Volume Control Tank, assuming the ratio of the gas concentration in the liquid and gas phases in the Volume Control Tank follows Henry's Law.

The waste gas system source terms were determined for the degasification of the reactor coolant liquid by the calculation of the quantity of activity entering the Volume Control Tank via the normal letdown path. For the sake of conservatism, the stripping efficiency of this process is assumed to be 100%. Therefore, in the previous equation of the stripping fraction, $C_L = C_{L_{eq}}$, and the stripping fraction is:

$$SF = \frac{C_R - C_{L_{eq.}}}{CR}$$

Thus, the separation of the fission gases from the reactor coolant liquid in the VCT will follow Henry's Law. This results in the maximum theoretical gas concentration in the vapor phase of the VCT and, hence, the maximum quantity of gas enters the waste gas system.

12A.3.2 METHODOLOGY

12A.3.2.1 Calculation of Dose Rates

Dose rates for the areas of interest in this review were calculated by determining the potential contributing sources at a representative location and using the appropirate source term from Table 12A.3-2 adjusted for decay as required. The dose rate at the representative location was used as the general area dose rate for the area. The SDC computer code (Ref. 1) was used in performing the dose rate claculations. Energy groups required as input to the computer code were determined using the gamma ray energy and intensity data in Refs. 2 and 3 for the nuclides in Table 12A.3-2.

12A.3.2.2 Calculation of Doses to Personnel During Post Accident Access to Vital Areas

Personnel doses received in performing a specific task in a given vital area are calculated as the sum of the doses received during travel to and from the vital area and the dose received while performing the given operation in the vital area.

The doses received during travel are determined by calculating dose rates at selected locations (or at a single location if the dose rate along the travel route is relatively uniform) along the travel route using the methodology discussed in Section 12A.3.2.1 and multiplying the dose rates by the appropriate travel time for each selected location along the travel route.

Doses received while performing a given operation are determined by multiplying the dose rate for the given area by the time required to perform the operation. Dose rates for the given vital area are determined using the methodology discussed in Section 12A.3.2.1.

12A.3.2.3 Calculation of Integrated Doses to Safety Equipment

The integrated dose to a given item of safety equipment is determined by integrating the dose rate appropriate for the given item over the time period that it is required to be available to perform its safety function. Dose rates are calculated using the methodology discussed in Section 12A.3.2.1.

TABLE 12A.3-1

Containment Sump Minimum Liquid Inventory

Liquid Source	Liquid Volume (ft^3)
Refueling Water Storage Tank	46,791[a]
Safety Injection Accumulators	3,000[b]
Boron Injection Tank	120
Sodium Hydroxide Storage Tank	408[c]
Reactor Coolant System	9,146
Minimum Containment Sump Volume =	59,465 ft^3

NOTES: (a) Refueling Water Storage Tank at minimum operating water level at start of drawdown of the tank. The tank drawdown will be terminated at LO-LO level upon the automatic initiation of recirculation via the RHR system.

(b) The three (3) Safety Injection Accumulators have individual capacities of 1000 ft^3 each.

(c) The minimum usable volume of liquid in the Sodium Hydroxide Storage Tank.

TABLE 12A.3-2

V. C. Summer
Shielding Source Terms (T=0)

Isotope	Liquid[(1)] Source Activity (ci)	Gaseous[(2)] Source Activity (Ci)	Containment Sump Concentration (µCi/cc)	Reactor Coolant Concentration (µCi/cc)	Containment Airborne Concentration (µCi/cc)	Waste Gas Concentration (µCi/cc)
Br-84	7.5 + 6	3.75 + 6	4.46 + 3	2.90 + 4	7.20 + 1	6.67 + 3
Kr-87	3.70 + 7	3.70 + 7	2.20 + 4	1.43 + 5	7.10 + 2	1.30 + 6
Te-133	4.40 + 5	-	2.62 + 2	1.70 + 3	-	-
Cs-134	1.70 + 5	-	1.01 + 2	6.56 + 2	-	-
Cs-136	4.70 + 4	-	2.80 + 1	1.81 + 2	-	-
Cs-137	7.03 + 4	-	4.18 + 1	2.71 + 2	-	-
Ba-139	1.50 + 6	-	8.93 + 2	5.79 + 3	-	-
Br-83	3.20 + 6	1.60 + 6	1.90 + 3	1.24 + 4	3.07 + 1	2.85 + 3
Kr-83m	6.30 + 6	6.30 + 6	3.75 + 3	2.43 + 4	1.21 + 2	2.09 + 5
Kr-85m	1.90 + 7	1.90 + 7	1.13 + 4	7.34 + 4	3.65 + 2	5.06 + 5
Kr-85	7.42 + 5	7.42 + 5	4.42 + 2	2.86 + 3	1.42 + 1	1.15 + 4
Kr-88	5.50 + 7	5.50 + 7	3.27 + 4	2.12 + 5	1.06 + 3	1.63 + 6
Rb-88	5.41 + 5	-	3.22 + 2	2.09 + 3	-	-
Rb-89	6.80 + 5	-	4.05 + 2	2.63 + 3	-	-
Sr-89	7.40 + 5	-	4.40 + 2	2.86 + 3	-	-
Sr-90	5.00 + 4	-	2.98 + 1	1.93 + 2	-	-
Y-90	5.00 + 4	-	2.98 + 1	1.93 + 2	-	-
Sr-92	9.10 + 5	-	5.42 + 2	3.51 + 3	-	-
Y-92	1.00 + 6	-	5.95 + 2	3.86 + 3	-	-
Sr-93	1.10 + 6	-	6.55 + 2	4.25 + 3	-	-
Y-93	1.10 + 6	-	6.55 + 2	4.25 + 3	-	-
Mo-99	1.50 + 6	-	8.93 + 2	5.79 + 3	-	-
Tc-99m	1.30 + 6	-	7.74 + 2	5.02 + 3	-	-
Ru-103	1.10 + 6	-	6.55 + 2	4.25 + 3	-	-
Rh-103m	1.10 + 6	-	6.55 + 2	4.25 + 3	-	-
Ru-106	4.10 + 5	-	2.44 + 2	1.58 + 3	-	-
Rh-106	4.10 + 5	-	2.44 + 2	1.58 + 3	-	-
Te-132	1.10 + 6	-	6.55 + 2	4.25 + 3	-	-
I-132	5.90 + 7	2.95 + 7	3.51 + 4	2.28 + 5	5.66 + 2	5.24 + 4
Te-134	1.60 + 6	-	9.52 + 2	6.18 + 3	-	-
I-134	8.90 + 7	4.45 + 7	5.30 + 4	3.44 + 5	8.54 + 2	7.88 + 4

TABLE 12A.3-2 (Continued)

Isotope	Liquid [1] Source Activity (ci)	Gaseous [2] Source Activity (ci)	Containment Sump Concentration (μCi/cc)	Reactor Coolant Concentration (μCi/cc)	Containment Airborne Concentration (μCi/cc)	Waste Gas Concentration (μCi/cc)
Xe-138	1.50 + 8	1.50 + 8	8.93 + 4	5.79 + 5	2.88 + 3	6.64 + 6
Cs-138	1.50 + 6	-	8.93 + 2	5.79 + 3	-	-
Ba-140	1.40 + 6	-	8.33 + 2	5.41 + 3	-	-
La-140	1.50 + 6	-	8.93 + 2	5.79 + 3	-	-
Ce-143	1.20 + 6	-	7.14 + 2	4.63 + 3	-	-
Pr-143	1.20 + 6	-	7.14 + 2	4.63 + 3	-	-
Ce-144	9.20 + 5	-	5.48 + 2	3.55 + 3	-	-
Pr-144	1.20 + 6	-	7.14 + 2	4.63 + 3	-	-
Sr-91	9.20 + 5	-	5.48 + 2	3.55 + 3	-	-
Y-91m	-	-	-	-	-	-
Y-91	9.60 + 5	-	5.71 + 2	3.71 + 3	-	-
Zr-95	1.30 + 6	-	7.74 + 2	5.02 + 3	-	-
Nb-95m	-	-	-	-	-	-
Nb-95	1.30 + 6	-	7.74 + 2	5.02 + 3	-	-
Zr-97	1.30 + 6	-	7.74 + 2	5.02 + 3	-	-
Nb-97m	-	-	-	-	-	-
Nb-97	1.30 + 6	-	7.74 + 2	5.02 + 3	-	-
Ru-105	8.60 + 5	-	5.12 + 2	3.32 + 3	-	-
Rh-105m	8.60 + 5	-	5.12 + 2	3.32 + 3	-	-
Rh-105	5.50 + 5	-	3.27 + 2	2.12 + 3	-	-
Te-131	7.20 + 5	-	4.29 + 2	2.78 + 3	-	-
I-131	4.10 + 7	2.05 + 7	2.44 + 4	1.58 + 3	3.93 + 2	3.64 + 4
Xe-131m	6.51 + 5	6.51 + 5	3.88 + 2	2.51 + 5	1.25 + 1	1.15 + 4
I-133	7.80 + 7	3.90 + 7	4.64 + 4	3.01 + 5	7.49 + 2	6.91 + 4
Xe-133m	3.80 + 6	3.80 + 6	2.26 + 3	1.47 + 4	7.29 + 1	7.24 + 4
Xe-133	1.50 + 8	1.50 + 8	8.93 + 4	5.79 + 5	2.88 + 3	2.72 + 6
I-135	6.90 + 7	3.45 + 7	4.11 + 4	2.66 + 5	6.62 + 2	6.09 + 4
Xe-135m	4.20 + 7	4.20 + 7	2.50 + 4	1.62 + 5	8.06 + 2	1.77 + 6
Xe-135	2.90 + 7	2.90 + 7	1.73 + 4	1.12 + 5	5.57 + 2	7.18 + 5
Ba-141	1.30 + 6	-	7.74 + 2	5.02 + 3	-	-
La-141	1.30 + 6	-	7.74 + 2	5.02 + 3	-	-
Ce-141	1.40 + 6	-	8.33 + 2	5.41 + 3	-	-

TABLE 12A.3-2 (Continued)

NOTES:

(1) Based on 100% noble gas core inventory, 50% halogen core inventory, and 1% of all others core inventory.

(2) Based on 100% noble gas core inventory and 25% halogen core inventory.