December 23, 2009
RC-09-0159

SCE&G®
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Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Sir / Madam:

Subject: VIRGIL C. SUMMER NUCLEAR STATION (VCSNS) UNIT 1
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
FOR LICENSE AMENDMENT REQUEST - LAR-04-02911 (ALTERNATIVE SOURCE TERM)

Reference:
1. NRC Letter (ADAMS Accession No. ML092960067) to Mr. Jeffrey B. Archie dated October 26, 2009, "Virgil C. Summer Nuclear Station, Unit 1 - Request for Additional Information (TAC NO. ME0663)"
2. SCE&G Letter (ADAMS Accession No. ML090720887) (LAR-04-02911) from Mr. Jeffrey B. Archie to USNRC Document Control Desk dated February 17, 2009, "License Amendment and Related Technical Specification Changes to Implement Full-Scope Alternative Source Term in Accordance with 10 CFR 50.67".

South Carolina Electric & Gas Company (SCE&G) hereby submits a response to the Request for Additional Information (RAI) items identified in the referenced NRC letter (Ref. 1). These RAIs resulted from NRC review of LAR-04-02911 (Ref. 2) requested by SCE&G.

Attachment I to this letter provides a response to all of the RAIs. In addition, a CD is enclosed in support of the RAI responses. The CD includes meteorological monitoring data validation referenced in RAI response #1, and referenced drawings in the RAI responses #2 and #3.

As described in the last paragraph in the response to Question 6, new EAB and LPZ doses will be provided by January 15, 2010 demonstrating acceptable doses for each of the postulated accidents considered in the VCSNS LAR.

Should you have questions, please call Bruce Thompson at (803) 931-5042.

A001
NRR

I certify under penalty of perjury that the foregoing is true and correct.

12/23/2009

Executed on

For Jeffrey B. Archie

Jeffrey B. Archie
Vice President, Nuclear Operations

JHW/JBA/cm
Attachments I & 1 CD

c: (without Attachments unless noted)
K. B. Marsh
S. A. Byrne
N. S. Carns
J. H. Hamilton
R. J. White
W. M. Cherry
T. P. O'Kelley
P. Ledbetter
L. A. Reyes (with Attachments)
R. E. Martin (with Attachments)
NRC Resident Inspector
K. M. Sutton
NSRC
RTS (CR-04-02911)
File (813.20)
PRSF (RC-09-0159)

VCS AST RAIs

Meteorological

1. **Please explain how the VCSNS hourly meteorological data from 2002 through 2006, as provided in support of the February 17, 2009, alternative source term (AST) license amendment request (LAR), in general, was processed from the raw measurements and discuss the VCSNS site meteorological characteristics. The U.S. Nuclear Regulatory Commission (NRC) staff has noted some apparent anomalies in temporal trends and between measurement heights in the five year data file. The following are several examples of staff observations and estimates, but should not be regarded as all inclusive.**

RESPONSE:

VCSNS is located near the center of South Carolina (26 miles northwest of Columbia, SC), approximately 140 miles northwest of the Atlantic Ocean and 100 miles southeast of the Appalachian Mountains. Plant elevation is approximately 435' above mean sea level. The terrain consists of rolling hills; elevations range from 200' near Columbia to over 600' near Little Mountain. The Broad River lies approximately 1 mile to the west, and Parr Dam is located approximately 2-1/2 miles southwest.

The climate is temperate, having long hot summers and cool winters. The Appalachian Mountain chain frequently retards the approach of cold fronts during the winter months. General climatology is discussed further in section 2.3.1 of the FSAR.

General VCSNS site meteorological climate characteristics are discussed in FSAR section 2.3.2. Onsite meteorological data is provided by a meteorological tower located at the same elevation as the Reactor Building, yet is far enough away from plant structures to reduce their influence on observed conditions. Further information on the meteorological tower instrumentation and data collection, as well as local site characteristics and their influences on the meteorological data, may be found in Section 2.3.3 of the FSAR.

The meteorological (met) monitoring program is described in FSAR section 2.3.3.2. With regard to the five year data file, raw data from the VCSNS met tower was retrieved and processed for complete calendar years 2002 through 2006 (a set of five complete years) in Revision 2 to calculation DC00040-080. As described in DC00040-080, 60-minute averaged raw data (per Section 6 of RG 1.23) was converted to ARCON96-specific inputs as required per NUREG/CR-6331. This included date/time/wind speed format and unit conversions, and calculation of atmospheric stability classifications (Pasquill Stability Classes according to FSAR Table 2.3-120). Invalid measurements were screened out, with the highest yearly percentage of invalid measurements less than or equal to 5.14% - which is well within the required 90% data recovery rate per Section 5 of RG 1.23. Within the VCSNS LAR, the five year data set was utilized to calculate control room χ/Q's.

There were a number of observations made relative to the five year data file, including the following:

- **In 2002, winds at both the 10- and 61-meter levels appear to be reported from the north about 14 percent of the year, whereas from 2003 through 2006 northerly winds were reported to occur only about 3 to 6 percent of the time.**

- **In 2003, wind direction was reported as either 360° or 0° with a wind speed of 0.3 meters per second for a consecutive 19 hour period on Julian days 112 and 113. In 2002, beginning in hour 23 of Julian day 280, the reported wind direction at both heights was identical or within two degrees of each other for 77 consecutive hours.**

- **In 2005, wind speed was reported to decrease with height about 15 percent of the year. Further, in 2005, the 10-meter wind speed was reported to remain unchanged from the previous hour about 12 percent of the time.**

In recognition of these and other observations, an additional three-year meteorological data set that had been previously verified and validated by an outside consultant (ABS Consulting) was obtained. As explained below, its use confirms the continued applicability of the control room $^{\chi}/_{Q}$'s used in the VCSNS LAR.

An additional set of ARCON96 inputs were developed and validated by an outside consultant (ABS Consulting) for the proposed VCSNS Units 2 and 3 Combined Operating License (COL) application. The validation package was previously submitted via letter from SCE&G to the NRC , *V. C. Summer Nuclear Station Units 2 and 3 Project No. 743 Submittal of Supporting Documents for the March 31, 2008 Combined License Application for Virgil C. Summer Nuclear Station Units 2 and 3,* June 11, 2008 (CD #5 from Attachment 1 of that letter). Although submitted in support of Units 2 and 3, the data was based on the Unit 1 meteorological tower and is thus applicable to Unit 1. An additional copy of that validation package is attached.

Prior to a meteorological tower being constructed near the proposed Units 2 and 3, the COL used data collected over a three year time period (July 2003 through June 2006) at the VCSNS Unit 1 meteorological tower. Justification of the representativeness for this three year time period being indicative of long-term stable site trends was not available in time to perform the AST analyses, therefore the full five year data set used in the February 17, 2009 AST LAR was developed and used.

As described in Section 2.3.3.4.1.2 of the VCSNS Units 2 and 3 COL application part 2 FSAR (letter from SCE&G to NRC, *V. C. Summer Nuclear Station Units 2 and 3 Project No. 743 Submittal of a Combined License Application for V. C. Summer Nuclear Station Units 2 and 3*, March 27, 2008), the three year data collected (July 2003 through June 2006) was generally representative of the long-term climatological conditions at the site. Relatively close agreement for average wind speed and direction, wind direction persistence, prevailing wind direction, frequency of calm, and atmospheric stability is seen with the three year data and offsite long-term meteorological data from the Columbia, SC National Weather Service (NWS).

To provide evidence to show how well the three year data compares with long-term conditions at the site, the three year onsite wind data (July 2003 through June 2006) was compared with two periods of long-term wind data (1951-1960 and 1956-1975) at Columbia, SC NWS. The results of the

comparison are shown in the table below (taken from the VCSNS Units 2 and 3 COL FSAR Table 2.3-218).

Table 2.3-218
Comparison of Onsite Data with Long-Term Climatological Data

Wind Frequency Distribution (%)

Wind Direction	VCSNS Onsite Data 3 years (7/2003–6/2006)	Columbia NWS 10 years (1951–1960)	Columbia NWS 20 years (1956–1975)
N	3.8	4.9	6.8
NNE	5.2	6.5	6.5
NE	**9.0**	**8.1**	**7.9**
ENE	6.6	5.3	7.0
E	4.1	3.7	6.3
ESE	2.2	3.1	4.4
SE	2.9	3.1	3.3
SSE	5.6	3.0	2.6
S	7.1	4.5	6.3
SSW	9.0	7.4	6.4
SW	**11.6**	**10.1**	**10.7**
WSW	10.5	7.4	9.8
W	9.2	5.4	8.4
WNW	4.1	4.7	5.5
NW	3.4	4.3	4.2
NNW	2.8	4.1	4.0

As clearly shown in the table, the wind frequency distributions between the Columbia, SC NWS and the VCSNS site are in agreement, with the same bimode prevailing wind (southwest and northeast), and most of the winds are from four of the west southerly wind sectors (south-southwest, southwest, west-southwest, and west).

Since there are no differential temperature measurements made at the NWS, a comparison of stability classes was made based on one year of onsite data (1975) and the three years of recent onsite data. The results shown in the table below indicate a reasonable agreement with the highest frequencies occurring at classes D and E. The major difference (22.4% vs. 15.3%) was in stability classes F and G, which is inconsequential to the development of the 5 percentile $^{\chi}/_{Q}$.

Annual Percentage by Stability Class (%)						
	A	B	C	D	E	F&G
1975	3.6	2.0	5.4	35.0	31.8	22.4
(2003–2006)	8.8	6.5	8.6	34.3	22.2	15.3

Therefore, the three year onsite data is generally representative of the long-term climatological conditions at the site and provides a reasonable basis to prove continued applicability of the control room χ/Q's.

To assess any potential differences, the relative concentrations from the Reactor Building nearest point to control room intake 'A' and the Main Steam Safety Relief (MSSR) 'A' to control room intake 'B' were calculated and compared. The five year data relative concentrations were taken from DC00040-079, Rev. 1 and are compared in the table below to the ARCON96 runs with the three year data:

Time Period	Relative Concentration RB Nearest Point to Intake 'A' AST Submittal	Relative Concentration RB Nearest Point to Intake 'A' 3 yr data	Ratio of 3 yr Data to AST Submittal
0 – 2 hr	1.39E-03	1.38E-03	0.99
2 -8 hr	1.17E-03	1.14E-03	0.97
8 – 24 hr	5.70E-04	5.56E-04	0.98
1 – 4 d	4.17E-04	4.27E-04	1.02
4 – 30 d	3.00E-04	3.10E-04	1.03

Time Period	Rel. Conc. MSSR 'A' to Intake 'B' AST Submittal	Rel. Conc. MSSR 'A' to Intake 'B' 3 yr data	Ratio of 3 yr Data to AST Submittal
0 – 2 hr	1.51E-03	1.51E-03	1.00
2 -8 hr	1.17E-03	1.14E-03	0.97
8 – 24 hr	5.75E-04	5.63E-04	0.98
1 – 4 d	4.18E-04	4.25E-04	1.02
4 – 30 d	3.10E-04	3.15E-04	1.02

Although minor differences may exist, both data sets produce essentially the same control room χ/Q's, thus providing assurance that the current control room χ/Q's, and therefore doses, as calculated in the February 17, 2009 AST LAR are acceptable.

2. What is the scale of Figure 2 of Calculation No. DC00040-079 in Attachment 10 of the VCSNS LAR dated February 17, 2009? Where on the figure are the assumed points of release from the Refueling Water Storage Tank (RWST) Overflow, the Reactor Building wall, the Pressure Relief Area, and the three sets of safety and relief valves?

RESPONSE:

Figure 2 of Calculation No. DC00040-079 is not a scaled drawing. Attached marked up drawings of E-001-062 (FSAR Figure 1.2-26) provide the requested locations for each intake location. The scale on the attached drawings is:

1 inch ≈26.5 feet

3. With respect to the summary of the ARCON96 inputs listed in Table 5.1-3 of Calculation No. DC00040-079:

- **Why was zero meters used as the release height input for the Reactor Building wall rather than either the control room intake height or the midpoint height of the Reactor Building?**

RESPONSE:

Examining the section drawing E-002-002 (FSAR Figure 1.2-10) in concert with attached marked up drawing E-001-062 (FSAR Figure 1.2-26) shows that there is no outside direct path between the reactor building outside wall and the control building structure below elevation 511 feet. Also above elevation 511 feet there are no penetrations through the reactor building wall. Likely leakage paths through the reactor building west side facing the control building are generally below elevation 436 feet. Penetrations through the reactor building at elevations 436 feet and 463 feet are into penetration areas sealed by pressure tight doors. Therefore, it was considered that use of elevation 436 feet as the reactor building release location was appropriate to determine the $^{\chi}/_{Q}$ values for this release path.

Nevertheless, ARCON96 was rerun for the limiting case of the RB nearest point to intake 'A' setting the release height equal to the receptor height. The values are provided below.

Time	Intake 'A' DC00040-079 Values	Intake 'A' Release height equal to receptor height	% Difference
0-2 h	1.39E-03	1.48E-03	6.5%
2-8 h	1.17E-03	1.16E-03	-0.9%
8-24 h	5.70E-04	5.72E-04	0.4%
1-4 d	4.17E-04	4.26E-04	2.2%
4-30 d	3.00E-04	3.02E-04	0.7%

The revised $^{\chi}/_{Q}$ values were input to RADTRAD and resulted in a negligible change to the calculated control room 30 day dose for the LOCA, i.e., the calculated dose changed from 1.01 rem TEDE to 1.03 rem TEDE.

- **What is the basis for the release height of 8.2 meters for the Pressure Relief Area?**

RESPONSE:

As shown in drawing E-002-004 (FSAR Figure 1.2-14), the pressure relief area is a horizontal panel located on the Intermediate Building roof. Per page 9 of Calculation No. DC00040-079 in Attachment 10 of the VCSNS LAR dated February 17, 2009, the release height is the height (elevation of the IB roof - 463 ft) minus grade elevation (436 ft) = 27 ft or 8.2 meters.

- **Confirm that the directions from the intake to release are as follows: a) MS POR 'A' to Intake 'A' is 64°, b) MS SR's 'A' to Intake 'A' is 57°, c) RWST Overflow to Intake 'A' is 22°, and d) RWST Overflow to Intake 'B' is 18°.**

RESPONSE:

The directions to MS POR 'A' were slightly inaccurate in Calculation No. DC00040-079 while the directions and distances for the RWST were reversed. The direction of 57° for MS SR's 'A' to Intake 'A' was acceptable. The original and updated values are shown as follows:

Release Point	Direction from Intake to Release			
	DC00040-079 Values		Updated Values	
	Intake 'A'	Intake 'B'	Intake 'A'	Intake 'B'
MS POR 'A'	64°	59°	49°	52°
RWST Overflow	22°	18°	18°	22°
	Straight-line Horizontal Distance (m)			
RWST Overflow	51.9	55.7	55.7	51.9

ARCON96 was rerun for the MS POR 'A' cases. The values are provided below.

Time	DC00040-079 Values		Updated Values		% Difference	% Difference
	Intake 'A'	Intake 'B'	Intake 'A'	Intake 'B'	Intake 'A'	Intake 'B'
0-2 h	1.34E-03	1.37E-03	1.33E-03	1.37E-03	-0.7%	0.0%
2-8 h	1.01E-03	1.03E-03	9.66E-04	1.00E-03	-4.4%	-2.9%
8-24 h	4.97E-04	5.07E-04	4.80E-04	4.95E-04	-3.4%	-2.4%
1-4 d	3.64E-04	3.77E-04	3.68E-04	3.80E-04	1.1%	0.8%
4-30 d	2.69E-04	2.72E-04	2.66E-04	2.72E-04	-1.1%	0.0%

NOTE: The χ/Q values provided in DC00040-079 are conservative for MS POR 'A', and the χ/Q values for the RWST overflow do not change other than the labeling of the intakes.

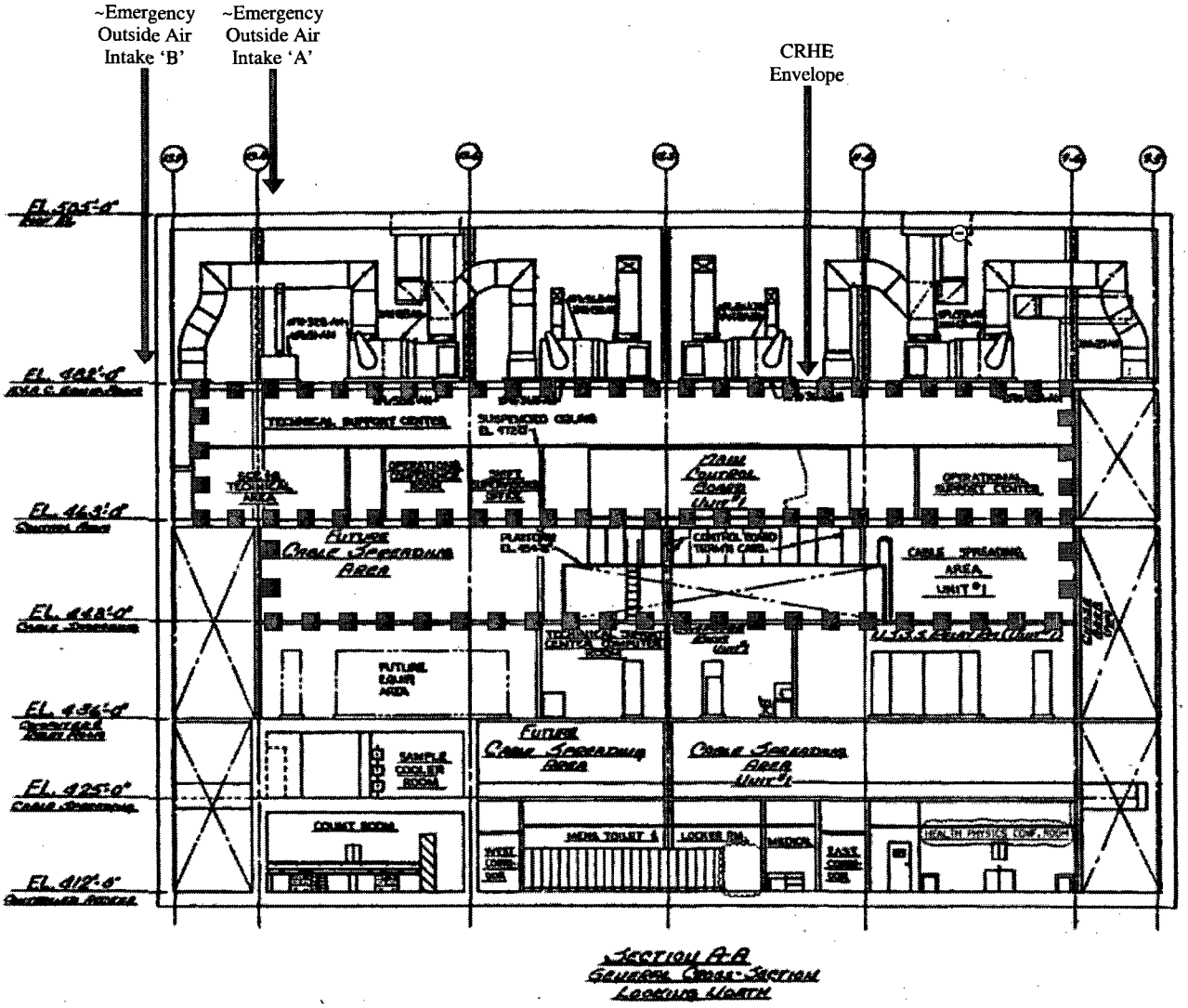
None of the above release points were used in any subsequent design basis radiological analyses.

4. Given that Intakes A and B are near the corner of the control building, furthest from the postulated release locations, please provide the location of the control room envelop on Figure 2. What atmospheric dispersion factors (χ/Q values) were used as inputs to the dose assessments for unfiltered inleakage into the control room and why are use of those values appropriate?

RESPONSE:

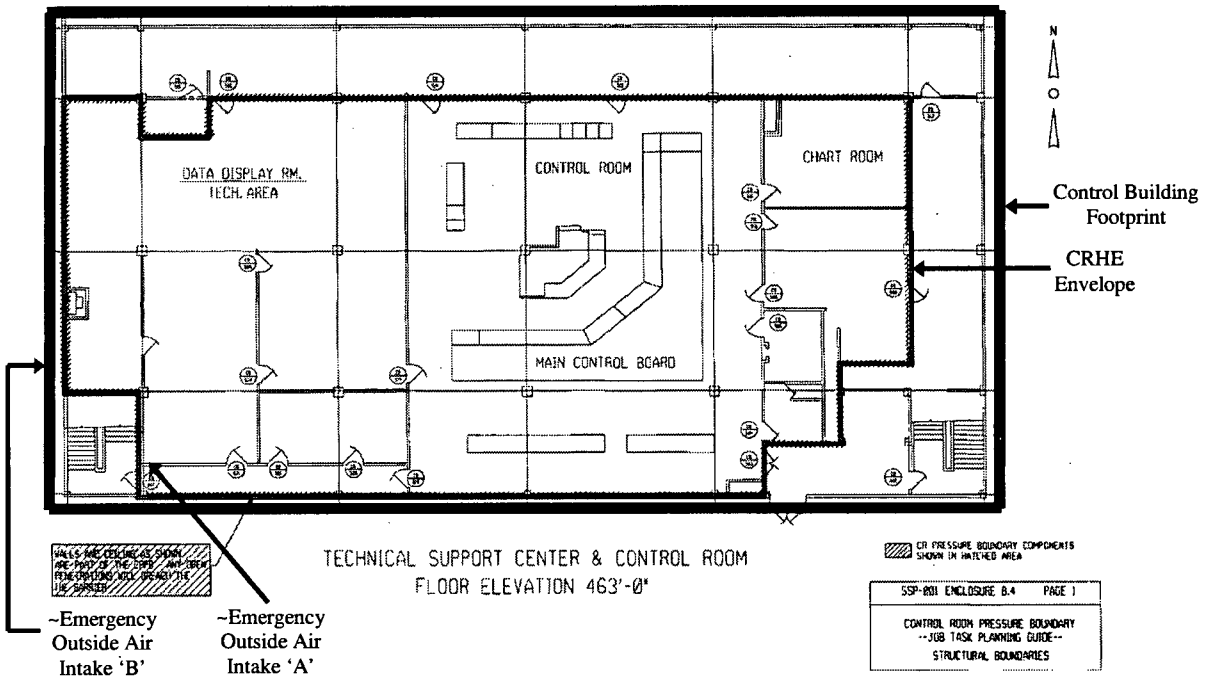
The Control Room Habitability Envelope (CRHE) occupies the majority of the 448' and 463' control building elevations as shown in the next two figures, as well as certain sections of the HVAC ductwork. These figures also show the approximate location of the outside air intakes, which are the emergency outside air intakes as the normal outside air intake structures are on the control building roof.

Control Building Elevation View (from drawing E-005-003, FSAR Figure 1.2-17):



SECTION A-B
GENERAL CROSS-SECTION
LOOKING NORTH

Control Building Plan View 463' Elevation (from enclosure 8.4 of SSP-001):



The atmospheric dispersion factors used as input to dose assessments from control room unfiltered inleakage were taken from calculation DC00040-079, Revision 1. This calculation determined the relative concentrations at the control room emergency outside air intakes 'A' and 'B' (shown in the two figures above) from several release points.

Control room emergency outside air intakes 'A' and 'B' were selected as appropriate locations for CRHE unfiltered air inleakage. Per Technical Specification Administrative Controls 6.8.4.m.4, the Control Room Envelope (CRE) Habitability Program includes measurement of the CRE pressure (relative to external areas adjacent to the CRE) when operating with one train in emergency mode. Station Administrative Procedure (SAP) 603 ensures this control room differential pressure is greater than 0.125 inches water gauge. CRHE pressurization eliminates unfiltered inleakage through the CRHE boundary from adjacent areas. However, this pressurization is achieved for each train operating in emergency mode by bringing in outside air (from control room emergency outside air intakes 'A' and 'B') at a nominal 1000 scfm, passing it through HEPA filters and a charcoal adsorber, and combining the filtered outside air to the nominal 20,000 scfm being recirculated from the CRHE (drawing D-912-140, FSAR Figure 9.4-1).

Therefore, because the CRHE is pressurized, the most credible location for any steady-state inleakage would be on the suction side of the fans of each train where a low pressure exists. Besides the control room emergency outside air intakes (where dispersion factors were calculated),

Besides the control room emergency outside air intakes (where dispersion factors were calculated), the normal outside air intake structures as well as the ductwork on the suction side of the fans could also be a source of inleakage.

The two normal outside air intake structures (one each for the 'A' and 'B' trains) are both located on the control building roof at the 505' elevation, similar to the 'A' emergency outside air intake. The 'A' emergency outside air intake $^{\chi}/_{Q}$'s would be slightly non-conservative for these normal outside air intakes (since the normal air intakes are slightly closer to potential release paths than the emergency intakes), but are still considered representative.

The ductwork on the suction side of the fans is contained in the control building at the 482' elevation (drawing E-005-003, FSAR Figure 1.2-17). The control room HVAC equipment completely occupies this elevation and is bounded by four concrete walls with few electrical penetrations, the control building roof, and doors leading to the control building stairwells. However, dispersion factors were not calculated, due to the torturous path as well as the very low measured inleakage as described below.

CRHE tracer gas inleakage tests were recently performed in March 2005 by NUCON International, Inc. On the nights 22-26 March, 2005, nine test configurations were evaluated for total inleakage into the CRHE by tracer gas techniques (ASTM E741-00 and NUCON 12-366 Rev. 1). The measured inleakage for each test configuration is summarized below:

Test Configuration	Inleakage (CFM)	Inleakage (CFM)
1 — 'A' Train Normal	15	±23
2 — 'A' Train Emergency	15	±29
3 — 'A' & 'B' Trains Emergency	0	±34
4 — 'A' & 'B' Emergency Damper 245B Open	0	±38
5 — 'A' & 'B' Emergency Damper 22B Modulated	0	±36
6 — 'B' Train Normal	0	±30
7 — 'B' Train Emergency	41	±16
8 — 'A' & 'B' Emergency Damper 245A Open	12	±23
9 — 'A' & 'B' Emergency Damper 22A Modulated	0	±33

Except for test configuration 7, the inleakage values for the tests are statistically zero since the values lie within the 95% uncertainty limits. For test configuration 7, which has a small inleakage of 41 cfm, the value falls slightly outside the 95% uncertainty limits.

Per NRC Letter Virgil C. Summer Nuclear Station – NRC Receipt of Response to Generic Letter 2003-01 "Control Room Habitability" (TAC NO. MB9860), from Robert E. Martin to Jeffery Archie, 10/24/2006, an Unanticipated Unfiltered Inleakage (UUI) value of 41 CFM was reported to the NRC. In addition, to provide margin, a conservative unfiltered inleakage of 243 cfm was assumed in all AST accident analyses to calculate the radiological consequences provided in the February 17, 2009 VCSNS LAR.

For the reasons described above, it was judged that the calculated dispersion coefficients for the control room outside emergency air intakes 'A' and 'B' were appropriate for CRHE unfiltered inleakage.

5. Regarding the design basis accidents analyzed in support of this LAR, please confirm that the generated χ/Q values model the limiting doses and that all potential release scenarios were considered, including those due to loss of offsite power or other single failures.

RESPONSE:

The χ/Q values used in the dose assessments represent the limiting cases should a loss of offsite power or other single failure occur. In the DBA LOCA case, the reactor building leakage and ECCS leakage are assumed to be released directly to the environment at the reactor building nearest point to Intake 'A'.

The activity release from the main steam line break, steam generator tube rupture, and locked rotor are assumed to be released through MS SR's 'A' to Intake 'B'.

The activity release from the rod ejection event is the RB nearest point to Intake 'A' for the containment release pathway, and MS SR 'A' to Intake 'B' for the steam generator release pathway.

For the fuel handling accident inside containment, the activity released to the atmosphere is assumed to occur as ground level releases from the reactor building nearest point to Intake 'A'. For the fuel handling accident outside containment, the activity is released to the atmosphere through the plant vent via the safety related fuel handling building emergency ventilation system to Intake 'B'.

6. Section 4.1 of Attachment 2 to the February 17, 2009, LAR notes that use of the χ/Q values previously approved by the NRC staff during the initial facility licensing is acceptable for use in the alternative source term analyses as discussed in Regulatory Guide 1.183, Section 5.3. The exclusion area boundary (EAB) and low population zone (LPZ) χ/Q values are listed in Table 4.1-1 of the LAR. However, NRC staff notes that these values are not the same as those in NUREG-0717, Supplement 4, "Safety Evaluation Report related to the operation of Virgil C. Summer Nuclear Station, Unit No. 1," dated August 1982. Therefore, please cite a reference for NRC approval of the current EAB and LPZ licensing basis χ/Q values used in support of the current LAR.

RESPONSE:

The exclusion area boundary (EAB) and low population zone (LPZ) χ/Q values listed in Table 4.1-1 of the LAR are the values provided in the original VCSNS FSAR Table 15A-3. From section 2.3.4 of NUREG-0717, the values in NUREG-0717 Table 15-3 were the NRC's independent confirmatory values from the raw metrology data provided. The two sets of values are compared below.

Time Period	VCS LAR	NUREG-0717	Ratio
	EAB	EAB	
0 – 2 hours	4.08E-04	3.3E-04	0.81
	LPZ	LPZ	
0 – 8 hours	2.37E-05	4.1E-05	1.73
8 - 24 hours	2.44E-06	2.6E-05	10.66
1 - 4 days	1.11E-06	1.0E-05	9.01
4 - 30 days	6.28E-07	2.6E-06	4.14

As noted above, the LAR value for the EAB is conservative relative to the NUREG-0717 value, while the LPZ values in NUREG-0717 are conservative. A comparison of the difference in the results for the LOCA analysis using the two sets of values is provided as follows:

Time Period	VCS LAR Rem TEDE	NUREG-0717 REM TEDE	Ratio
	EAB	EAB	
0 – 2 hours	4.87	3.93	0.81
Time Period	LPZ	LPZ	
0 – 30 days	0.54	1.29	2.39

As shown, the EAB dose would decrease while the LPZ dose would increase using the NUREG-0717 χ/Q's. All values are well below the 25 Rem TEDE regulatory limit of 10CFR100.

Comparing the calculated dose to the dose criteria in Attachment 1 of the VCSNS February 17, 2009 LAR, the locked rotor accident has the least margin in EAB and LPZ dose. Therefore, this analysis was re-performed as well to determine the differences in the two sets of χ/Q's:

	VCS LAR Rem TEDE	NUREG-0717 REM TEDE	Ratio
EAB	2.20	1.78	0.81
LPZ	0.45	1.08	2.40

All values are below the 2.5 Rem TEDE regulatory limit of RG 1.183.

In addition to the above, the NRC-sponsored PAVAN computer code, as described in NUREG/CR-2858 was used to estimate ground-level χ/Q's for potential accidental releases of radioactive material to the atmosphere using the latest validated three year meteorological data set.

The PAVAN model input data is presented below:

- Meteorological data: 3-year (July 1, 2003 to June 30, 2006 as discussed in response to RAI #1) composite onsite joint frequency distributions of wind speed, wind direction, and atmospheric stability
- Type of release: ground-level (a default height of 10 meters was used)
- Wind sensor height: 10 meters
- Vertical temperature difference: (61 meters – 10 meters)
- Number of wind speed categories: 12 (including calm)
- EAB Distance - 1609 meters, for all downwind sectors
- LPZ Distance – 4828 meters for all downwind sectors
- Building cross-sectional area – 1740 m^2 (DC00040-079)
- Containment height – 44.8 m (DC00040-079).

The PAVAN analysis values are compared to the above in the following table.

Time Period	VCS LAR	NUREG-0717	PAVAN	Ratio NUREG-0717/PAVAN
	EAB	**EAB**	**EAB**	**EAB**
0 – 2 hours	4.08E-04	3.3E-04	1.24E-04	2.66
	LPZ	**LPZ**	**LPZ**	**LPZ**
0 – 8 hours	2.37E-05	4.1E-05	2.42E-05	1.69
8 - 24 hours	2.44E-06	2.6E-05	1.68E-05	1.55
1 - 4 days	1.11E-06	1.0E-05	7.55E-06	1.32
4 - 30 days	6.28E-07	2.6E-06	2.40E-06	1.08

The comparison above indicates that for the EAB, the current VCS LAR value is the most conservative, while for the LPZ the NUREG-0717 values are conservative. Although any combination of the above χ/Q's will lead to acceptable dose results, it is VCSNS's position that the most appropriate values to apply are the new PAVAN results. This approach will achieve consistency with current, accepted practices (RG-1.183, Section 5.3) and establish new FSAR χ/Q's that are supported by in-house, plant specific calculations. To support this revised approach, new EAB and LPZ doses will be provided by January 15, 2010 demonstrating acceptable doses for each of the postulated accidents considered in the VCSNS LAR.