UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 4, 2010

Mr. Thomas D. Gatlin
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88
Jenkinsville, SC 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1, ISSUANCE OF AMENDMENT REGARDING ALTERNATIVE SOURCE TERM IMPLEMENTATION (TAC NO. ME0663)

Dear Mr. Gatlin:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 183 to Renewed Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1, in response to your application dated February 17, 2009, as supplemented on June 15, December 1, and December 23, 2009, January 14, and July 16, 2010. The amendment implements an alternative source term application methodology for analyzing the radiological consequences for six design-basis accidents. The current radiological source term licensing basis remains unchanged for equipment qualification, NUREG-0737 evaluations other than for control room habitability, and Final Safety Analysis Report accidents not included in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-basis Accidents at Nuclear Power Reactors."

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's next Biweekly *Federal Register* notice.

Sincerely,

Robert Martin

Robert E. Martin, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures:
1. Amendment No. 183 to NPF-12
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 183
Renewed License No. NPF-12

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by South Carolina Electric & Gas Company (the licensee), dated February 17, 2009, as supplemented by letters dated June 15, December 1, and December 23, 2009, January 14, and July 16, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Appendix A Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-12 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 183, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of its date of issuance and shall be implemented within ninety (90) days.

FOR THE NUCLEAR REGULATORY COMMISSION

Gloria Kulesa, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachments:
Changes to the Technical
Specifications

Date of Issuance: October 4, 2010

October 4, 2010

Mr. Thomas D. Gatlin
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88
Jenkinsville, SC 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1, ISSUANCE OF AMENDMENT REGARDING ALTERNATIVE SOURCE TERM IMPLEMENTATION (TAC NO. ME0663)

Dear Mr. Gatlin:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 183 to Renewed Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, UnitNo. 1, in response to your letters dated February 17, 2009, as supplemented on June 15, December 1, and December 23, 2009, January 14, and July 16, 2010. The amendment implements an alternative source term application methodology for analyzing the radiological consequences for six design-basis accidents. The current radiological source term licensing basis remains unchanged for equipment qualification, NUREG-0737 evaluations other than for control room habitability, and Final Safety Analysis Report accidents not included in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-basis Accidents at Nuclear Power Reactors."

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's next Biweekly *Federal Register* notice.

Sincerely,

/RA/

Robert E. Martin, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures:
1. Amendment No. 183 to NPF-12
2. Safety Evaluation

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NAME	RMartin	SRohrer	TTate*	RDenning*
DATE	09/21/10	09/16/10	07/08/10	10/01/09
OFFICE	NRR/DCI/CSGB/BC	OGC	ITSB/BC	NRR/LPL2-1/BC
NAME	RTaylor*	MDreher	REliott	GKulesa
DATE	02/04/10	09/15/10	0917/10	10/04/10

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ATTACHMENT TO LICENSE AMENDMENT NO. 183

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
License	License
3	3
TS	TS
VIII	VIII
X	X
XIV	XIV
XV	XV
3/4 3-42	3/4 3-42
3/4 3-44	3/4 3-44
3/4 3-45	3/4 3-45
3/4 7-40	3/4 7-40
3/4 9-4	3/4 9-4
3/4 9-9	3/4 9-9
6-12f	6-12f
6-12g	6-12g

(3) SCE&G, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as amended through Amendment No. 33;

(4) SCE&G, pursuant to the Act and 10 CFR Parts 30, 40 and 70 to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) SCE&G, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus of components; and

(6) SCE&G, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as my be produced by the operation of the facility.

C. This renewed license shall be deemed to contain, and is subject to, the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

SCE&G is authorized to operate the facility at reactor core power levels not in excess of 2900 megawatts thermal in accordance with the conditions specified herein and in Attachment 1 to this renewed license. The preoccupation tests, startup tests and other items identified in Attachment 1 to this renewed license shall be completed as specified. Attachment 1 is hereby incorporated into this renewed license.

(2) Technical Specifications and Environmental Protection Plan

The technical Specifications contained in Appendix A, as revised through Amendment No 183 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area (RM-G8)	1	*	≤ 15 mR/hr	10^{-1} - 10^{4} mR/hr	25
b. Deleted					
2. PROCESS MONITORS					
a. Deleted					
b. Containment					
i. Deleted					
ii. Particulate and Gaseous Activity (RM-A2) - RCS Leakage Detection	1	1, 2, 3 & 4	N/A	10 - 10^{6} cpm	26
c. Control Room Isolation (RM-A1)	1	ALL MODES	≤ 2 x background	10 - 10^{6} cpm	29

* With fuel in the storage pool or building

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 27 - Deleted

ACTION 28 - Deleted

ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the emergency mode of operation.

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area (RM-G8)	S	R	M	*
b. Deleted				
2. PROCESS MONITORS				
a. Deleted				
b. Containment				
i. Deleted				
ii. Particulate and Gaseous Activity - RCS Leakage Detection (RM-A2)	S	R	M	1, 2, 3 & 4
c. Control Room Isolation (RM-A1)	S	R	M	ALL MODES
d. Noble Gas Effluent Monitors (High Range)				
i. Main Plant Vent (RM-A13)	S	R	M	1, 2, 3 & 4
ii. Main Steam Lines (RM-G19A, B, C)	S	R	M	1, 2, 3 & 4
iii. Reactor Building Purge Supply & Exhaust System (RM-A14)	S	R	M	1, 2, 3 & 4

* With fuel in the storage pool or building

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location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

a) Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

b) Inspect 100% of the tubes at sequential periods of 144, 108, 72 and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

c) If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

5. Provisions for monitoring operational primary-to-secondary leakage.

I. Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989.

1. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	Flowrate
Control Room Emergency Filtration System	21,270 SCFM
Reactor Building Cooling Units	60,270 ACFM

2. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	Flowrate
Control Room Emergency Filtration System	21,270 SCFM

3. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity specified below.

ESF Ventilation System	Penetration	RH	Face Velocity (fps)
Control Room	<2.5%	70%	0.667

4. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	Delta P	Flowrate
Control Room	<6 in. W.G.	21,270 SCFM
Reactor Building Cooling Units	<3 in. W.G.	60,270 ACFM

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the VFTP test frequencies.

m. Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Filtration System (CREFS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 183

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-395

1.0 INTRODUCTION

By application dated February 17, 2009 (Reference 1), as supplemented on June 15, December 1, and December 23, 2009, January 14, and July 16, 2010, (References 2 through 6), South Carolina Electric & Gas Company (SCE&G, the licensee), submitted a license amendment request (LAR) for Virgil C. Summer Nuclear Station, Unit 1 (VCSNS). The LAR provides the technical specification (TS) changes and design-basis accident (DBA) radiological consequence analyses to support implementation of alternative source term (AST) methodology, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.67 (10 CFR 50.67), "Accident source term," using the guidance described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design-basis Accidents at Nuclear Power Reactors," (RG 1.183). Thus the existing accident radiological source term from TID-14844 (Reference 7) has been replaced for the six following VCSNS DBAs: loss of coolant accident (LOCA), main steamline break (MSLB), fuel-handling accident (FHA), SG tube rupture (SGTR), reactor coolant pump locked rotor accident (LRA), and the control rod ejection accident (CREA). The licensee states that the current radiological source term licensing basis, as in TID-14844, will remain the licensing basis for equipment qualification, NUREG-0737 evaluations other than for control room habitability, and Final Safety Analysis Report (FSAR) accidents not included in RG 1.183.

The supplements dated June 15, December 1, and December 23, 2009, January 14, and July 16, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the Federal Register on March 24, 2009 (74 FR 12395).

2.0 REGULATORY EVALUATION

The Nuclear Regulatory Commission (NRC, the Commission) staff evaluated the radiological consequences of the postulated DBAs against the dose criteria specified in 10 CFR 50.67. The

applicable criteria are 5 rem Total Effective Dose Equivalent (TEDE) in the control room (CR), 25 rem TEDE at the exclusion area boundary (EAB), and 25 rem TEDE at the outer boundary of the low population zone (LPZ).

The regulatory requirements upon which the NRC staff based its acceptance are Standard Review Plan (SRP) 15.0.1, General Design Criteria (GDC) 19, and the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 and Table 6 of RG 1.183. The licensee has not proposed any deviation or departure from the guidance provided in RG 1.183. The NRC staff's evaluation is based upon the following regulatory codes, guides, and standards, in addition to relevant information in the VCSNS FSAR and TSs:

- 10 CFR Part 50.67, "Accident source term."
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants": Criterion 19, "Control room."
- NUREG-0800 Standard Review Plan (SRP) Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases."
- NUREG-0800 SRP Section 6.4, "Control Room Habitability Systems."
- NUREG-0800 SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Term."
- NUREG/CR-6604, "RADTRAD: Simplified Model for RADionuclide Transport and Removal And Dose Estimation."
- NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes."
- NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design-basis Accidental Releases of Radioactive Materials from Nuclear Power Stations."
- NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data."
- RG 1.23, "Onsite Meteorological Programs."
- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants."
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design-basis Accidents at Nuclear Power Reactors."
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants."

- RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors."
- NRC Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms."

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the regulatory and technical analyses performed by the licensee in support of its proposed license amendment, as they relate to the radiological consequences of DBA analyses. Information regarding these analyses was provided as Attachment 10 to the February 17, 2009, submittal. The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impacts of the proposed license amendment. The NRC staff also performed independent calculations to confirm the conservatism of the licensee's analyses. However, the findings of this safety evaluation (SE) are based on the descriptions and results of the licensee's analyses and other supporting information docketed by the licensee.

3.1 Atmospheric Dispersion Estimates

In support of the LAR, dated February 17, 2009, the licensee generated new control room (CR) atmospheric dispersion factors ($^{\chi}/_{Q}$ values) for use in evaluating the radiological consequences of the following six DBAs: LOCA, FHA, CREA, MSLB, SGTR, and the LRA. The licensee initially used the $^{\chi}/_{Q}$ values listed in the VCSNS FSAR for postulated releases to the exclusion area boundary (EAB) and the low population zone (LPZ). The licensee subsequently provided updated EAB and LPZ $^{\chi}/_{Q}$ values by letter dated December 23, 2009. These CR, EAB and LPZ $^{\chi}/_{Q}$ values represent a change from those in the current VCSNS FSAR.

3.1.1 Meteorological Data

To calculate the CR $^{\chi}/_{Q}$ values, the licensee used onsite hourly meteorological data for calendar years 2002 through 2006 as an input to the ARCON96 atmospheric dispersion computer code (NUREG/CR-6331). The licensee used a joint wind speed, wind direction, and atmospheric stability frequency distribution (JFD) for the 3-year period from July 2003 through June 2006 as an input to the PAVAN computer code (NUREG/CR-2858) to calculate the updated EAB and LPZ $^{\chi}/_{Q}$ values. The July 2003 through June 2006 data file is a subset of the calendar year 2002 through 2006 data.

The NRC staff performed a quality review of the ARCON96 hourly meteorological database using the methodology described in NUREG-0917. Further review was performed using computer spreadsheets. Wind speed and wind direction data were measured at heights of 10 meters and 61 meters above the ground. Temperature difference data, which were used to determine atmospheric stability class, were measured between the 61-meter and 10-meter levels. The NRC staff noted some apparent anomalies in temporal trends and between measurement heights in the 2002 through 2006 data file set and, by letter dated October 26, 2009, NRC staff requested additional information regarding the data. In its December 23, 2009, response, the licensee provided a supplemental July 2003 through June 2006, 3-year meteorological data. Since justification of the representativeness for this 3-year period being indicative of long-term site trends was not available in time to perform the AST analyses, the full 5-year data set was

developed and used in the February 17, 2009, AST LAR. The licensee provided additional information and comparisons of meteorological data and calculated $^{\chi}/_{Q}$ values to justify use of the 3-year data period.

The NRC staff performed a review of the verified July 2003 through June 2006 data set. Combined data recovery for the 3-year period exceeded 90 percent which meets the recommendation of RG 1.23, Revision 0, "Onsite Meteorological Programs." Stable and neutral atmospheric conditions were generally reported to occur at night and unstable and neutral conditions during the day, as expected. The daily duration of stable and unstable conditions was consistent with expected meteorological conditions. Winds were predominantly from southwesterly and northeasterly directions at both the 10- and 61-meter levels. With regard to wind speed, the 10-meter wind speed was reported to be somewhat higher in 2005 than the balance of the 3-year period, with an apparent anomaly that the 10-meter wind speed exceeded the 61-meter wind speed more than 16 percent of the year. This was more than twice as frequently as during the other years. To assess the possible impact of this difference, NRC staff performed selected comparisons of $^{\chi}/_{Q}$ calculations using both the full July 2003 through June 2006 3-year data set and also a data set from which the 2005 data had been deleted.

In addition, the NRC staff generated a joint frequency distribution (JFD) using the verified July 2003 through June 2006 data set. A comparison of this JFD with the JFD used by the licensee as an input to calculate the EAB and LPZ $^{\chi}/_{Q}$ values showed reasonably good agreement. The NRC also generated a JFD which deleted calendar year 2005 from the verified data for use in making additional comparison calculations.

In summary, the NRC staff reviewed available information relative to the onsite meteorological measurements program and the 2002 through 2006 five-year meteorological data files originally provided by the licensee. On the basis of this review, the NRC staff has concluded that the verified 2003 through June 2006 three-year data files subsequently provided by the licensee give an adequate representation of the site conditions to facilitate calculation of the specific $^{\chi}/_{Q}$ values used in the dose assessments performed in support of this LAR. However, because of the apparent uncertainties discussed above, the NRC staff notes that these data may not be considered acceptable for use in other dose assessments or other meteorological applications without further NRC review and approval.

3.1.2 Control Room Atmospheric Dispersion Factors

To assess the CR post-accident atmospheric dispersion conditions, the licensee generated $^{\chi}/_{Q}$ values using the ARCON96 computer code and guidance provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." In the February 17, 2009 LAR, the licensee postulated releases for six DBAs from the following locations to Intake A and Intake B.

- Reactor Building nearest point
- Main Plant Vent
- Purge Exhaust
- Intermediate Building Blowout Panel
- Main Steam (MS) Steam Safety Value (SSV) A (A-relief only)

- MS SSV A (Relief's B, C, D & E)
- MS SSV B
- MS SSV C
- MS Power-Operated Relief Valve (PORV) A
- MS PORV B
- MS PORV C

All releases were modeled as ground level point sources using the straight line horizontal distance as the distance of separation between each postulated source and receptor. In RG 1.194 it states that ARCON96 is an acceptable methodology for assessing CR $^{\chi}/_{Q}$ values for use in DBA radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and concluded that there is no unusual siting, building arrangement, release characterization, source-receptor configuration, meteorological regime, or terrain condition that precludes use of this model in support of the current LAR for VCSNS.

Based on the resultant calculations, in Table 4.1.2 of the LAR, the licensee identified the limiting source and receptor pairs as being: (a) the reactor building's nearest point to Intake A, (b) the Main Plant Vent to Intake B and (c) the main steam safety relief valves (Reliefs B, C, D, E) to Intake B. The licensee confirmed that the $^{\chi}/_{Q}$ values used in the dose assessments represent the limiting cases should a loss of offsite power or other single failure occur. The $^{\chi}/_{Q}$ values associated with each of these release locations were also used to analyze unfiltered inleakage to the CR for the dose assessment.

The NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and requested additional information regarding a few of the listed inputs when compared with a drawing provided in the February 17, 2009, LAR. In its December 23, 2009, response, the licensee provided comparison calculations of selected cases, supplemental figures, and further analysis to support that use of the inputs in the current LAR, including that use of the 2002 through 2006 meteorological data rather than the verified July 2003 through June 2006 data, did not result in significant differences in the resultant CR $^{\chi}/_{Q}$ values and estimated doses for the assessment associated with this specific LAR.

In summary, the NRC staff qualitatively reviewed the inputs to the ARCON96 computer runs for the CR $^{\chi}/_{Q}$ value assessment and found them adequately consistent with site configuration drawings and NRC staff practice. NRC staff performed comparison calculations for a small number of random cases, including calculations based on an NRC-generated file that deleted the 2005 meteorological data. On the basis of this review, the NRC staff has concluded that the limiting CR $^{\chi}/_{Q}$ values identified in the February 17, 2009, LAR are acceptable for use in the dose assessment associated with this LAR. However, because of the uncertainties discussed in Section 3.1.1 regarding the meteorological data set used to generate these $^{\chi}/_{Q}$ values, the NRC staff notes that these data may not provide adequate representation of the site for other applications. Therefore, the $^{\chi}/_{Q}$ values should not be considered acceptable for use in other dose assessments or other meteorological applications without further NRC review and approval.

3.1.3 EAB and LPZ Atmospheric Dispersion Factors

Section 4.1 of Attachment 2 to the February 17, 2009, LAR stated that the EAB and LPZ $^{\chi}/_{Q}$ values listed in Table 4.1-1 and used in the AST dose analyses were the current licensing basis values described Section 2.3.4 of the VCSNS FSAR. Section 4.1 further stated that use of the $^{\chi}/_{Q}$ values previously approved by the NRC staff during the initial facility licensing is acceptable for use in the AST analyses as discussed in RG 1.183, Section 5.3. However, the NRC staff noted that the values listed in Table 4.1-1 were not the same as those in NUREG-0717, "Safety Evaluation Report related to the operation of Virgil C. Summer Nuclear Station, Unit No.1," dated February 1981 (NUDOCS Number 8103030656). Further, it was not clear to NRC staff when the EAB and LPZ $^{\chi}/_{Q}$ values had become the current licensing basis values. Therefore, NRC staff requested that the licensee cite a reference for NRC approval of the $^{\chi}/_{Q}$ values used in support of the current LAR.

In the December 23, 2009, response, the licensee provided new EAB and LPZ $^{\chi}/_{Q}$ values based on the guidance in RG 1.145. The licensee calculated these $^{\chi}/_{Q}$ values using the PAVAN atmospheric dispersion computer code and the verified July 2003 through June 2006 three-year data set discussed in Section 3.1.1. The data were divided into a relatively large number of wind speed categories at the lower wind speeds to generate the JFD input file. In RIS 2006-04 it states that JFDs used as input to PAVAN should have a large number of wind speed categories at the lower wind speeds in order to produce the best results. Other inputs included EAB and LPZ distances of 1609 and 4828 meters, respectively, a building minimum cross-sectional area of 1740 square meters, and a containment height of 44.8 meters. The release was considered to be ground level.

The licensee also provided a comparison of the new EAB and LPZ $^{\chi}/_{Q}$ values with the $^{\chi}/_{Q}$ values provided in the February 17, 2009, LAR and those in NUREG-0717. The December 23, 2009, response noted that the LAR $^{\chi}/_{Q}$ values were limiting for the EAB and the NUREG-0717 values were limiting for the LPZ, although any combination of the $^{\chi}/_{Q}$ values would lead to acceptable dose results. The licensee proposed use of the new July 2003 through June 2006 data set, $^{\chi}/_{Q}$ values in the dose assessment related to the current LAR. This approach would achieve consistency with current, accepted practices and also establish new FSAR EAB and LPZ $^{\chi}/_{Q}$ values supported by in-house, plant-specific calculations.

The NRC staff qualitatively reviewed the inputs to the PAVAN computer run and found them generally consistent with site configuration drawings and NRC staff practice. The NRC staff also evaluated the resulting atmospheric dispersion estimates by running the PAVAN computer model using the verified July 2003 through 2006 meteorological data set and by deleting the 2005 meteorological data. In both cases, the results were similar to or less limiting than the results obtained by the licensee.

On the basis of this review, the NRC staff finds the new $^{\chi}/_{Q}$ values listed in Table 3.1.2 of this SE acceptable for use in the EAB and LPZ dose assessments in the current LAR. However, because of the uncertainties discussed in Section 3.1.1 regarding the meteorological data set used to generate these $^{\chi}/_{Q}$ values, the NRC staff notes that these data may not provide adequate representation of the site for other applications. Therefore, the $^{\chi}/_{Q}$ values should not be considered acceptable for use in other dose assessments or other meteorological applications without further NRC review and approval.

3.2 Radiological Consequences of Design-basis Accidents

To support the proposed implementation of an AST, the licensee analyzed the radiological dose consequences and provided all major inputs and assumptions for the following DBAs:

- Loss-of-Coolant Accident (LOCA)
- Fuel-Handling Accident (FHA)
- Main Steam Line Break (MSLB) Accident
- Steam Generator Tube Rupture (SGTR) Accident
- Locked-Rotor Accident (LRA)
- Control Rod Ejection Accident (CREA)

As a minimum effort to revise the VCSNS licensing basis to incorporate a full implementation of the AST, RG 1.183, Position 1.2.1, specifies that the DBA LOCA must be re-analyzed using the appropriate guidance therein. In accordance with this RG 1.183 guidance, the licensee re-analyzed the six DBAs listed above, which includes the design-basis LOCA at VCSNS.

The licensee's submittal reports the results of the radiological consequence analyses for the above DBAs to show compliance with 10 CFR 50.67 dose acceptance criteria, or fractions thereof, as defined in SRP 15.0.1, for doses offsite and in the control room. For full implementation of the AST DBA analysis methodology, the dose acceptance criteria specified in 10 CFR 50.67 provides an alternative to the previous whole body and thyroid dose guidelines stated in 10 CFR 100.11 and GDC 19. The subject LAR is considered a request for full implementation of the AST.

RG 1.183, Regulatory Position 3.1, "Fission Product Inventory", states the following:

> "The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the [Emergency Core Cooling System] ECCS evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 or ORIGEN-ARP."

For accident analyses postulating fuel damage, the licensee conforms to RG 1.183, Regulatory Position 3.1, by multiplying the default isotopic specific activities calculated and presented in Table 1.4.3.2-2 of NUREG/CR-6604, for a typical pressurized-water reactor (PWR), by the VCSNS-specific maximum full power level (2,958 MWth) and release fractions. In addition, in a letter dated July 16, 2010 (Reference 6), the licensee stated that, for iodine and noble gas isotopes, the isotopic activities used were calculated using the ORIGEN2 code and presented in the proprietary "Westinghouse Radiation Analysis Manual for Virgil C. Summer," which was used to determine the current licensing basis source term. The licensee compared these iodine and noble gas activities to those that were determined from the NUREG-6604 default PWR specific activities, and then used the larger of the two values in their analysis. After inspection of the isotopic activities that were determined by the licensee, the NRC staff finds that the values used

are consistent with and more conservative than those expected following VCSNS operation at a maximum full-power level and using the licensed values for enrichment and burnup. Therefore, they are acceptable for AST accident analyses.

As stated in RG 1.183, the release fractions associated with the light-water reactor (LWR) core inventory released into containment for the design-basis LOCA and non-LOCA events have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 megawatt days per metric ton of uranium (MWd/MTU) provided that the maximum linear heat generation rate does not exceed 6.3 kilowatt per foot (kw/ft) peak rod average power for burnups exceeding 54,000 MWd/MTU.

To perform independent confirmatory dose calculations for the DBAs, the NRC staff used the NRC-sponsored radiological consequence computer code, "RADTRAD: Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.03, as described in NUREG/CR-6604. The RADTRAD code, developed by the Sandia National Laboratories for the NRC, estimates transport and removal of radionuclides and the resulting radiological consequences at selected receptors.

The following sections discuss the NRC staff review of the DBA dose assessment performed by the licensee to support the LAR submittal of February 17, 2009, including all supporting supplements.

3.2.1 Loss-of-Coolant Accident (LOCA)

The current VCSNS design-basis LOCA analysis is based on the traditional accident source term described in Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The current licensing basis radiological consequence analysis for the postulated LOCA is provided in the VCSNS Final Safety Analysis Report (FSAR) Chapter 15.4.1, "Major Reactor Coolant System Pipe Ruptures (Loss-Of-Coolant Accident)." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated LOCA. This reanalysis was performed to demonstrate that the engineered safety features (ESFs) designed to mitigate the radiological consequences at VCSNS will remain adequate after implementation of the AST.

In addition to the LAR, the licensee submitted the AST-based reanalysis of the LOCA including the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their reanalysis of the postulated design-basis LOCA. Specifically, the NRC staff's guidance for the assumptions is detailed in Appendix A of RG 1.183.

3.2.1.1 Activity Source

For the LOCA analysis the licensee assumed that the core isotopic inventory that is available for release into the containment atmosphere is based on maximum full-power operation of the core at 2,958 MWth, or 1.02 times the current licensed thermal power level of 2,900 MWth, in order to account for the ECCS evaluation uncertainty. Additionally, the current licensed values for fuel enrichment and burnup are assumed when determining the core isotopic inventory.

The core inventory release fractions and release timing for the gap and early in-vessel release phases of the DBA-LOCA were taken from RG 1.1.83, Tables 2 and 4, respectively. Also consistent with RG 1.183 guidance, the licensee assumes that the radioactive iodine speciation released from failed fuel is 95.00 percent aerosol (particulate), 4.85 percent elemental, and 0.15 percent organic. Whereas, the radioactive iodine speciation released from steaming and flashed liquid is 97.00 percent elemental and 3.00 percent organic.

3.2.1.2 Transport Methodology and Assumptions

The licensee calculated the onsite and offsite dose consequences of the design-basis LOCA by modeling the transport of activity released from the core to the environment, while accounting for appropriate activity dilution, holdup, and removal mechanisms.

The NRC staff has reviewed the licensee's assessment of the following potential post-LOCA activity release pathways:

- Containment Leakage
- ESF Equipment Leakage
- Refueling Water Storage Tank (RWST) Release of ECCS Leakage
- Hydrogen Purge Line Leakage

Also, the NRC staff has reviewed the licensee's assessment of the following potential post-LOCA shine dose pathways:

- Containment Shine
- External Cloud Shine

The following sections detail the NRC staff's review of the licensee's analysis of these post-accident contributors to both control room and offsite dose.

Containment Leakage

The current VCSNS design-basis containment leak rate, L_a, is equal to 0.2 percent volume per day (percent per day), at containment peak pressure, as expressed in VCSNS FSAR Table 6.2-1. The licensee assumed the design-basis leak rate L_a for the first 24 hours of the accident, then a subsequent reduction to half of this value for the remainder of the accident duration, as is consistent with the guidance of RG 1.183.

The airborne activity in the VCSNS containment following the postulated LOCA is mitigated by natural deposition of fission products in aerosol form, RB cooling unit (RBCU) particulate filtration, and removal by the containment spray system. The following subsections discuss the evaluation of the credit that is taken for activity mitigation by these processes.

When more than one model is used simultaneously for the various mechanisms for iodine removal for the same iodine species in a dose analysis, this usage should consider the effect of one model on the others. Because each model used by the licensee to simulate the removal of activity does not necessarily account for removal through the other models, the use of the

referenced natural deposition, containment filtration, and spray removal models in the same region of containment, during the same time period, is recognized as potentially non-conservative.

Although natural deposition, containment filtration, and sprays are all acting on the overall in-containment aerosol and elemental iodine source term, the total effect from each of these removal mechanisms is not the same as would be found by simply adding the removal coefficients for each model for a given time period. Therefore, to the NRC staff, this treatment of containment removal modeling is generally found to be unacceptable. NRC staff confirmatory calculations showed that the dose consequences of the postulated LOCA with only sprays credited bounded the dose consequences assuming both sprays and natural deposition are credited. However, with regard to natural deposition, it is understood that in the presence of the filtration and sprays, the additional effect of crediting natural deposition is minimal. The licensee for VCSNS did not credit natural deposition when sprays were assumed to be operation. It can also be concluded, from the NRC staff's confirmatory calculations based on the licensee's submitted dose analyses, that adequate conservatism was used in modeling containment filtration and spray systems such that the non-conservative effect of the two systems on one another is sufficiently accounted for. In addition, as discussed in the following sections, there is adequate justification that both the filtering units and spray systems will indeed be available to mitigate post-LOCA activity leakage. Therefore, for this amendment request, the NRC staff does not deem it necessary for the licensee to recalculate the iodine removal, and subsequent resulting LOCA dose consequences, using a more conservative modeling of iodine removal mechanisms. The NRC staff's confirmatory analysis indicates that, while the inclusion of the effects of natural deposition in conjunction with ESF iodine removal mechanisms is, in theory, non-conservative, the overall iodine removal, as determined by the licensee in this case, is conservative and is therefore acceptable.

Reactor Building (RB) Cooling System

As noted, for releases into containment, the licensee credits the VCSNS RB cooling system for mixing and recirculation of air and entrained activity between the sprayed and unsprayed regions of containment. The RBCUs and associated discharge ductwork of this system supply cooling air flow to the unsprayed regions below the operating floor. This air flow establishes a convective flow path whereby the air-steam mixture from the unsprayed regions and from inside the secondary shield walls are directed upward into the suction of the RBCUs. For this reason, in their submitted LOCA design analysis, the licensee stated that mixing and recirculation of the building atmosphere for the volumes between the lowest elevation of the RB and the elevation of the RBCUs (approximately 67 feet above the operating floor) is provided. The licensee also states in their LOCA analysis that the RBCU fans operate at high speed during normal periods and at slow speeds during post-LOCA periods, and that the units are serviced by cooling water from the industrial cooling system during normal periods and by the service water system during post-LOCA or loss of offsite power conditions. For normal operation, 3 out of 4 fans operate. For LOCA, one fan in each train operates. The automatic change in operating mode from normal to emergency involves changing from motors operating at high speed to motors operating at slow speed, changing the unit cooling water from that supplied by the industrial cooling water system to that from the service water system, and automatically closing the RB high efficiency particulate air (HEPA) filter bypass damper. The active RBCU discharge valves will automatically align so that 2 of the 4 valves, one per train, will open when their associated RBCU fans are selected to run; the

other two valves will close when their RBCU fans are not selected to run. When operating in the normal mode, the RBCUs are tripped upon the receipt of a safety injection or loss of offsite power signal and are then automatically started at slow speed in accordance with the ESF actuation system and the ESF loading sequence of the related emergency diesel generator.

The total time for switchover from normal to emergency operation is completed in approximately 33 seconds as determined by the licensee. As a design-basis, the licensee assumed the RBCUs to be fully operational at 52 seconds post-LOCA and run for the duration of the accident. The reactor building cooling unit's recirculation flow rate is 60,270 ± 10 percent actual cubic feet per minute (acfm), per TS 4.6.3.b.1 Therefore, the licensee determined the minimum allowable system flow to be 54,243 acfm; however, the licensee rounded this number down to 54,200 cfm for conservatism in modeling the recirculation. The licensee also conservatively assumed the RBCU recirculation HEPA to remove only 90 percent of the particulate form of iodine particulate.

Because the licensee incorporated limiting uncertainty values in their analysis, and made conservative assumptions about the function of their RBCU, the licensee's treatment of the RBCU System and its mitigation of post-LOCA activity releases is conservative and consistent with current VCSNS design-basis assumptions. Therefore, this treatment is acceptable to the NRC staff.

Natural Deposition

The licensee's submitted LOCA design analysis assumed removal of airborne activity in aerosol form by natural deposition in containment following the postulated LOCA using Powers' simplified natural deposition model in the dose consequences computer code described in RADTRAD and its supplements. Powers' simplified natural deposition model is described in NUREG/CR-6189, "A Simplified Model of Aerosol Removal By Natural Processes in Reactor Containments." Natural deposition was only credited when the sprays were assumed to be not in operation, and the licensee conservatively used the 10th percentile confidence interval (90 percent probability) removal values implemented in the RADTRAD code. The Powers natural deposition model was derived by correlation of results of Monte Carlo uncertainty analyses of detailed models of aerosol behavior in the containment under accident conditions. The Powers model has been reviewed and found generally acceptable by the NRC staff, and the NRC staff finds that the use of this model, as implemented in the NRC computer code, RADTRAD, to also be acceptable, as discussed in RG 1.183.

Reactor Building Sprays

The licensee modeled the iodine removal function of the VCSNS RB spray system following the postulated LOCA. As described by the licensee in VCSNS FSAR Section 6.2.2.2.1.1, three spray header and nozzle assemblies are provided for each of the two spray subsystems. Each spray subassembly has 165 spray nozzles. The spray nozzles are one-piece, stainless steel units that deliver water at a nominal rate of 15.2 gallons per minute (gpm). A nominal spray flow of 2,500 gpm is provided by each spray subsystem. The licensee assumed the containment building atmosphere to be a single, well-mixed volume. This assumption is acceptable to the NRC staff because the licensee also determined that sprays cover at least 90 percent of the building volume thereby providing adequate mixing of the unsprayed compartments.

In their LOCA design analysis, the licensee stated that, during the injection phase following a LOCA, the spray pumps receive fluid from the RWST and the sodium hydroxide storage tank. Subsequently, these solutions are carried through the pump discharge lines to the spray headers and are then sprayed into the RB atmosphere through the spray nozzles. The spray is collected in the bottom of the RB with water from ECCS and the reactor coolant system (RCS) in the RB recirculation sumps. Upon receipt of the RWST "lo-lo" level signal in conjunction with the safety injection signal, the RB recirculation sump isolation valves automatically open. During the recirculation phase, the spray pumps may take suction from the sumps in the RB.

As described by the licensee in their submittal, operation of the spray system is initiated automatically following a LOCA by signals from the engineered safety features (ESF) actuation system when the RB pressure increases to the actuation setpoint. Following a spray actuation signal, the RB spray pumps are loaded onto the onsite diesel generators in the case of a loss of offsite power. The licensee determined the total time for RB spray water to start being delivered through the spray nozzles to the RB atmosphere is 47 seconds following LOCA initiation. This includes a 10-second start time for emergency diesel generators (EDGs), 5 seconds for the spray pump to reach full capacity, and 32 seconds to fill the RB spray line up to the nozzles. For conservatism in their analysis, the licensee assumed no filling of lines during the 5-second pump acceleration time, the sprays start 52 seconds after the LOCA occurs, and that the sprays will operate for only the first 4 hours of the accident.

In their LOCA design analysis, the licensee assumed removal of airborne activity by RB sprays following the postulated LOCA using the simplified aerosol removal model, as incorporated in the dose consequences computer code described in RADTRAD and its supplements. This spray removal model is described in NUREG/CR-5966, "A Simplified Model of Aerosol Removal By Containment Spray," and NUREG-0800, Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 4. The system-specific parameters discussed above were input into the model to calculate the activity removal effectiveness of the VCSNS RB sprays.

By selecting limiting parameters and assuming minimal system effectiveness, the licensee's model of activity release mitigation by the spray system is treated conservatively and in accordance with the applicable guidance, and is therefore acceptable to the NRC staff.

ESF Equipment Leakage

The guidance in RG 1.183, Appendix A, Section 5.2, states that leakage from the ESF system should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation system above which the TSs, or licensee commitments to item III.D.1.1 of NUREG-0737, would require declaring such systems inoperable. As a design-basis at VCSNS, it is assumed that leakage of coolant from ESF systems is possible upon initiation of those systems. The VCSNS TSs do not provide a specific limitation on the quantity of operational leakage that is allowed from the residual heat removal (RHR) and RB spray ESF systems. However, VCSNS FSAR Tables 6.2-52b and 6.3-4 provide estimates of 870 and 2060 cubic centimeters per hour (cc/hr) for the RB Spray and RHR systems, respectively. Currently, as expressed in the VCSNS FSAR Section 15.4.1.4.2, post-LOCA recirculation leakage of 12,000 cc/hr is assumed as the design-basis ESF leakage value. This conservative value is more than four times the combined estimated maximum leakage of 2930 cc/hr, as discussed above, and is therefore acceptable to the NRC staff.

The licensee's DBA analysis assumed leakage to start at 1460 seconds, which the licensee determined to be the earliest time the recirculation mode is entered, which is consistent with AST activity release assumptions. The leakage was assumed to continue for the duration of the accident. The licensee conservatively took no credit for holdup or filtration of this leakage in the auxiliary building by assuming activity released by the recirculation loop leakage to be immediately available for release to the atmosphere. Using this model, the licensee performed a sensitivity analysis to determine the maximum allowable ESF recirculation leakage that would still allow them to meet the applicable CR operator dose consequence limits.

The licensee assumed that, with the exception of iodine, all radioactive material in the recirculating coolant is retained in the liquid phase, and also in accordance with RG 1.183, the chemical form of the released iodine activity is assumed to be 97.00 percent elemental and 3.00 percent organic. The licensee uses a constant enthalpy assumption and an associated equation to calculate the flashing fraction of leaked coolant. The licensee determined this value to be less than 10 percent; however, in accordance with RG 1.183 guidance, a conservative flashing fraction of 10 percent is used.

RWST Release of ECCS Leakage

The licensee determined that leakage through the RWST and sodium hydroxide (NaOH) tank was negligible based on plant procedures, which require closure of the 20-inch RWST outlet valve and closure of the 3-inch NaOH outlet valve following the transition to recirculation. The result of this action is 3-valve isolation and a minimum of 2-valve isolation with an assumed single failure. Based upon this configuration and procedural action, the licensee's conclusion is acceptable to the NRC staff, as only negligible leakage of low activity fluid is to be expected.

Hydrogen Purge Line Leakage

VCSNS Licensing Amendment No. 170 deleted the TS requirement associated with the hydrogen recombiners and hydrogen monitors as a result of changes to 10 CFR 50.44. Consequently, post-LOCA containment purge for hydrogen control is not required at VCSNS and was not considered in the licensee's analysis. Therefore, because the VCSNS containment is not routinely purged, the NRC staff agrees that purge line leakage at VCSNS is not an activity release path that needs to be considered as a design-basis.

Containment Shine

For the calculation of internal containment cloud shine dose to the control room, the licensee applied the source term that was calculated using the model of accident releases into containment, as described in Section 3.2.1.2 of this SE. The containment volume and associated airborne activity cloud was modeled as a cylindrical source of air with a height of 3688.1 centimeters (cm) and a radius of 1920.24 cm. This cylindrical volume was modeled with a 4-foot concrete shield and at a transition distance of approximately 4400 cm from the control room wall. The VCSNS control room dose receiver was conservatively modeled at a distance of 2 feet away from the 2-foot thick concrete control room wall for the 720-hour accident duration, and ignoring any additional shielding provided by the floor and ceiling thickness.

The licensee used the MicroShield code to calculate the time dependent dose rates in the control room that result from containment activity cloud shine. Using a selection of 16 time steps, the

licensee calculated individual dose rates, and then integrated the results to determine the total shine dose contribution from the containment cloud over the 30-day duration of the accident. The licensee assumed a very conservative combination of dose receiver location and duration of that receiver in that position. Also, the licensee carried forward their conservative assumptions for activity transport, consistent with the regulatory guidance, to the generation of the containment cloud. Therefore, the licensee's conservative model is acceptable to the NRC staff for calculation of the internal containment cloud shine dose contribution to the 30-day control room dose.

External Cloud Shine

For the calculation of external cloud shine dose to the control room, the licensee applied the source term that was calculated using the model of accident releases into containment and out to the environment. As modeled by the licensee in their submitted LOCA design analysis, airborne activity that leaks from the RB results in a radioactive cloud external to the VCSNS control building. During the postulated accident, this cloud will in turn result in additional dose to the control room operator inside the control building. For their design-basis model, the licensee calculated the unprotected dose (the dose assuming no shielding) to control room personnel using the RADTRAD computer program to simulate the containment and ESF release models. Then, the licensee applied a reduction factor (RF) to account for the shielding effectiveness of the control building concrete. The licensee used the following equation to calculate their RF:

$$RF=B(ux) \cdot e^{(-ux)}$$

where:

B(ux) = buildup factor
u = attenuation factor
X = concrete thickness of Control Structure

Assuming the 2-foot thick control building concrete wall thickness and an average gamma energy of 0.733 mega electron volts (MeV), the licensee appropriately applied a buildup factor of 30. The average gamma energy of 0.733 MeV is based upon that provided in "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criteria GDC 19," K. G. Murphy and K. M. Campe, USAEC, 13th AEC Air Cleaning Conference, August 1974. Since this simplified and approved methodology applied by the licensee implements a conservative and appropriate reference, the NRC staff finds this acceptable.

Summary for LOCA

The licensee concluded that the radiological consequences at the EAB, LPZ, and control room are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in SRP 15.0.1 and RG 1.183. These accident-specific dose acceptance criteria for the LOCA are a TEDE of 25 rem at the EAB for the worst 2 hours, 25 rem at the outer boundary of the LPZ and 5 rem for access to and occupancy of the control room for the duration of the accident. The NRC staff finds that the licensee used sufficiently conservative analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the VCSNS FSAR as design bases. The NRC staff also performed independent calculations of the dose consequences of the postulated LOCA releases, using the licensee's assumptions for input to the RADTRAD computer code. The NRC staff's

calculations confirmed the licensee's dose results. The NRC staff found the major parameters and assumptions used by the licensee, as presented in Table 3.2.1 of this SE, to be acceptable. The results of the licensee's design-basis radiological consequence calculation are provided in Table 3.2 of this SE.

The EAB, LPZ, and CR doses estimated by the licensee for the LOCA were found to meet the applicable accident dose criteria. Therefore the NRC staff finds this acceptable.

3.2.2 Fuel-Handling Accident (FHA)

The current VCSNS design-basis FHA analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The VCSNS licensing basis analysis is presented in FSAR Chapter 15.4.5, "Fuel Handling Accidents." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated FHA. This reanalysis was performed to demonstrate that the engineered safety features (ESFs) designed to mitigate the radiological consequences at VCSNS will remain adequate after implementation of the AST.

The licensee submitted the AST-based reanalysis of the FHA as part of the LAR. Included in this reanalysis are the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their reanalysis of the postulated design-basis FHA. Specifically, the NRC staff's guidance is detailed in Appendix B of that document.

The VCSNS design-basis consists of two accident scenarios: (1) a refueling accident occurring inside containment, and (2) a refueling accident occurring outside containment in the VCSNS Fuel Handling Building (FHB). The postulated FHA inside containment is the drop of a spent fuel assembly onto the core during refueling which results in damage to the fuel assemblies. The FHA outside containment is postulated as the drop of a spent fuel assembly into the spent fuel pool in the FHB, which results in damage to the fuel assemblies and the release of the volatile gaseous radioactivity.

Activity Source

For the FHA analysis, the licensee assumed that the core isotopic inventory is based on maximum full-power operation of the core at 2,958 MWth, or 1.02 times the current licensed thermal power level of 2,900 MWth, in order to account for the ECCS evaluation uncertainty. The licensee assumed the design-basis radial peaking factor of 1.70. Additionally, the licensee accounted for current licensed values for fuel enrichment and burnup when determining the core isotopic inventory. The fraction of core isotopic activity assumed to be available for release from the gap of failed fuel (i.e., fuel experiencing cladding failure as a result of the drop) is provided in Table 3 of RG 1.183. To account for gap fraction uncertainty, in fuel that does not meet the criteria specified in footnote 11 of RG 1.183, the licensee multiplied these gap fractions by a factor of two. This is a conservative approach that is found to be acceptable to the NRC staff.

The licensee assumed that, as a design-basis, spent fuel will have decayed for at least 72 hours after shutdown before being handled. For both the FHA inside and outside of containment, the licensee assumed that 314 pins will be damaged (264 pins in the dropped assembly and 50 pins

in the impacted assembly), as the result of the postulated FHA, and thus release all of their available gap activity over a 2-hour period. This is consistent with the current VCSNS licensing basis, as described in VCSNS FSAR Section 15.4.5, and the guidance expressed in RG 1.183.

Transport Methodology and Assumptions

Per VCSNS FSAR Section 15.4.5, the minimum water depth over the reactor core when handling fuel and over the spent fuel in the FHB is 23 feet. Therefore, consistent with RG 1.183, Appendix B.2, an overall effective iodine decontamination factor (DF) of 200 was assumed, which resulted in an iodine chemical speciation released from the water of 57 percent elemental and 43 percent organic. All particulate radionuclides were assumed to be retained in the water. The 23-foot water depth coverage easily bounds the available water coverage for a drop over the reactor core. Therefore, the licensee's assumption is conservative.

For the FHA inside containment, the licensee assumed activity released to the environment to be released at a ground level from a limiting point on the RB (see Section 3.1). For the FHA outside of containment, the conditions and parameters assumed by the licensee are identical to those utilized in the FHA inside containment, except that the activity released to the environment is released through the fuel-handling building exhaust system. In both cases, the activity release is modeled to occur within a time period not to exceed 2 hours, in accordance with the guidance of RG 1.183. The licensee accomplished this by purging a nominal free air volume, arbitrarily set to 1.0E+04 cubic feet, at a very high flowrate (1.0E+10 cfm). Using this model, the licensee assumed an immediate release of all leakage directly to the environment without credit for holdup, plateout, or dilution.

For the FHA releases of activity from inside the FHB through the FHB exhaust system, the licensee took no credit for treatment by the available HEPA and charcoal filters. Therefore, both the FHA inside and outside containment are modeled with identical activity releases but differing $^{\chi}/_{Q}$ values; and due to the bounding $^{\chi}/_{Q}$ value for the release to the control room, the FHA inside containment is the bounding DBA.

Summary for Fuel-Handling Accident (FHA)

The licensee concluded that the radiological consequences at the EAB, LPZ, and control room are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in SRP 15.0.1 and RG 1.183. These accident-specific dose acceptance criteria for the FHA are a TEDE of 6.3 rem at the EAB for the worst 2 hours, 6.3 rem at the outer boundary of the LPZ and 5 rem in the control room for the duration of the accident. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the VCSNS FSAR as design bases. The NRC staff also performed independent calculations to verify the conservatism of certain parameters used by the licensee. The NRC staff found the major parameters and assumptions used by the licensee, as presented in Table 3.2.2 of this SE, to be acceptable. The results of the licensee's design-basis radiological consequence calculation are provided in Table 3.2 of this SE. The EAB, LPZ, and CR doses estimated by the licensee for the FHA accident were found to meet the applicable accident dose criteria and are therefore acceptable.

3.2.3 Main Steam Line Break (MSLB) Accident

The current VCSNS design-basis MSLB analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The VCSNS licensing basis analysis is presented in FSAR Chapter 15.4.2.1, "Major Rupture of a Main Steam Line." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated MSLB. This reanalysis was performed to demonstrate that the engineered safety features (ESFs) designed to mitigate the radiological consequences at VCSNS will remain adequate after implementation of the AST.

The licensee submitted the AST-based reanalysis of the MSLB accident as part of the LAR. Included in this reanalysis are the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their reanalysis of the postulated design-basis MSLB accident. Specifically, the NRC staff's guidance is detailed in Appendix E of that document.

The design-basis MSLB accident is typically defined as the pre-trip guillotine-type rupture of a main steamline. The licensee modeled unimpeded blowdown of steam from the broken steam line. As a VCSNS design-basis, the steam release arising from a rupture of a main steam line results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure; however, the core is ultimately shut down by the boric acid injection delivered by the VCSNS safety injection system. Activity is introduced into the nuclear steam supply system (secondary side) through SG tube leakage, also referred to as primary-to-secondary leakage.

Activity Source

With regard to the postulated activity available for release following a design-basis MSLB, the licensee considered the following two cases as a design-basis in order to determine the maximum offsite and control room dose:

- Pre-accident Iodine Spike (PIS) Case: For this case the licensee assumed that a reactor transient has occurred prior to the postulated MSLB and has raised the primary RCS iodine concentration to 60 times the TS 3.4.8 limit of 1.0 μCi/gm dose equivalent (DE) I-131, as is consistent with the guidance of RG 1.183, when no fuel failure is assumed.

- Concurrent Iodine Spike (CIS) Case: For this case the licensee assumed that the RCS transient associated with the MSLB causes an iodine spike in the primary RCS. It is assumed that the iodine release rate from the fuel rods to the primary RCS increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value, as is consistent with the guidance of RG 1.183, when no fuel failure is assumed. The licensee conservatively assumed a spike duration of 8 hours; that is, the iodine activity increases during the 8 hours of the transient as a result of release from the

defective fuel at the spiked iodine equilibrium appearance rate. The 8-hour assumption is consistent with the guidance of RG 1.183.

In addition, for the two cases considered, the licensee also assumed that the TS maximum secondary coolant iodine concentration is available for release, as inferred by RG 1.183 guidance.

As shown in VCSNS TS 3.7.1.4, and as input to the MSLB analysis, the maximum secondary coolant iodine concentration at VCSNS is 0.1 μCi/gm DE I-131.

Consistent with RG 1.183 guidance, the licensee assumed that the radioactive iodine speciation released from failed fuel is 95.00 percent aerosol (particulate), 4.85 percent elemental, and 0.15 percent organic. Whereas, the radioactive iodine speciation released from the SGs is 97.00 percent elemental and 3.00 percent organic.

Transport Methodology and Assumptions

The current licensing basis at VCSNS restricts primary-to-secondary leakage rate to a TS limited rate of 1 gallon per minute (gpm) apportioned between all SGs in a manner that maximizes the calculated dose consequence. The licensee assumed that, prior to the event, this leakage is distributed throughout the three SGs. The licensee conservatively analyzed the MSLB using 24 hours for the cooldown time, when following plant procedures would require 14 hours for shutdown cooling to be achieved. The licensee assumed that the activity associated with the 1 gpm leak is released to the environment via the faulted SG at a rate of 0.35 gpm for the 24 hour duration of the event with no credit taken for any reduction or mitigation, i.e., a partition factor of 1.0. These assumptions are consistent with the applicable guidance of RG 1.183 and conservative in that the actual maximum value allowed by VCSNS TS 3.4.6.2.c for any one SG is 150 gallons per day (~0.1 gpm).

The licensee assumed the remaining 0.65 gpm to be released to the environment via the two intact SGs for the 24-hour duration of the event crediting a partition factor of 100 in accordance with the guidance of RG 1.183. In addition, the licensee conservatively assumed that the secondary coolant activity initially contained in the faulted SG is released to the environment with a partition factor of 1.0 (no partitioning), whereas partitioning could theoretically be calculated. By selecting limiting parameters and by assuming maximized releases, the licensee conservatively modeled MSLB accident activity transport. Therefore, the NRC staff finds this release methodology to be acceptable.

Summary for MSLB

The licensee stated that the radiological consequences at the EAB, LPZ, and control room are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in SRP 15.0.1 and RG 1.183. These accident-specific dose acceptance criteria for the MSLB assuming a PIS are a TEDE of 25 rem at the EAB for the worst 2 hours, 25 rem at the outer boundary of the LPZ, and 5 rem for the control room. For the MSLB assuming a CIS, the accident-specific dose acceptance criteria are a TEDE of 2.5 rem at the EAB for the worst 2 hours, 2.5 rem at the outer boundary of the LPZ, and 5 rem for the control room. The NRC staff

finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the VCSNS FSAR as design bases.

The NRC staff also performed an independent calculation of the dose consequences of the MSLB accident using the licensee's assumptions for input to the RADTRAD computer code. The NRC staff's calculation confirmed the licensee's dose results. The NRC staff found the major parameters and assumptions used by the licensee, as presented in Table 3.2.3, to be acceptable. The results of the licensee's design-basis radiological consequence calculation are provided in Table 3.2. The EAB, LPZ, and CR doses estimated by the licensee for the MSLB accident were found to meet the applicable accident dose criteria and are therefore acceptable.

3.2.4 Steam Generator Tube Rupture (SGTR) Accident

The current VCSNS design-basis SGTR accident analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The VCSNS licensing basis analysis is presented in FSAR Chapter 15.4.3, "Steam Generator Tube Rupture." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated SGTR. This reanalysis was performed to demonstrate that the engineered safety features (ESFs) designed to mitigate the radiological consequences at VCSNS will remain adequate after implementation of the AST.

The licensee submitted the AST-based reanalysis of the SGTR accident as part of the LAR. Included in this reanalysis are the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their reanalysis of the postulated design-basis SGTR accident. Specifically, the NRC staff's guidance is detailed in Appendix F of that document.

Typically, the limiting SGTR accident is analyzed as a complete double-ended tube break that is postulated to occur due to a complete failure of a tube-to-sheet weld or the rapid propagation of a circumferential crack. The SGTR allows primary coolant to leak into the secondary system via the SG. The primary coolant transfer causes the pressurizer level to decrease, provided that the tube leak rate exceeds the charging pump capacities and causes the level in the affected SG to increase. In the event of a coincident loss of offsite power, or failure of the condenser steam dump system, discharge of activity to the atmosphere takes place via the SG safety and/or power-operated relief valves. At VCSNS, the operator is expected to determine that a SG tube rupture has occurred, and to identify and isolate the faulted SG on a restricted time scale in order to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the faulty unit. The recovery procedure can be carried out on a time scale which ensures that break flow to the secondary system is terminated before water level in the affected SG rises into the main steam pipe.

For a double-ended SG tube rupture, the leak rate typically exceeds the charging pump capacities and, consequently, the pressurizer level will decrease. The decrease in the pressurizer level and the inability of the heaters to maintain pressurizer pressure causes the RCS pressure to decrease.

The drop in the pressure will cause a reactor trip and ensure that the Departure from Nucleate Boiling (DNB) fuel design limit is not exceeded. The licensee assumed no fuel failure resulting from a design-basis SGTR accident at VCSNS.

Activity Source

The licensee determined that there is no fuel failure associated with the design-basis SGTR accident. As a result, the licensee considered the following two cases of postulated activity release following a design-basis SGTR, in order to determine the maximum offsite and control room dose:

- Pre-accident Iodine Spike Case: For this case the licensee assumed that a reactor transient has occurred prior to the postulated MSLB and has raised the primary RCS iodine concentration to 60 times the TS 3.4.8 limit of 1.0 μCi/gm DE I-131, as is consistent with the guidance of RG 1.183, when no fuel failure is assumed.

- Concurrent Iodine Spike Case: For this case the licensee assumed that the RCS transient associated with the MSLB causes an iodine spike in the primary RCS. It is assumed that the iodine release rate from the fuel rods to the primary RCS increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value, as is consistent with the guidance of RG 1.183, when no fuel failure is assumed. Also consistent with RG 1.183 guidance, the spike duration persists for a period of 8 hours.

In addition, for the two cases considered, the licensee assumed that the TS maximum secondary coolant iodine concentration is available for release, as inferred by RG 1.183 guidance. As shown in VCSNS TS 3.7.1.4, and as input to the MSLB analysis, the maximum secondary coolant iodine concentration at VCSNS is 0.1 μCi/gm DE I-131.

Consistent with RG 1.183 guidance, the licensee assumes that the radioactive iodine speciation released from the SGs is 97.00 percent elemental and 3.00 percent organic.

Transport Methodology and Assumptions

As stated in its submittal, the licensee assumed that a portion of the primary-to-secondary leakage through the SGTR flashes to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. The leakage that immediately flashes to vapor was assumed to rise through the bulk water of the SG and enter the steam space and then was assumed to be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the SG bulk water was credited. Partitioning of this release to the condenser would typically be credible, but was not credited by the licensee. All leakage that does not immediately flash was assumed to mix with the bulk water. The radioactivity within the bulk water was assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 was assumed for the leakage from the bulk water. This value is consistent with the regulatory guidance of RG 1.183.

The current licensing basis at VCSNS restricts primary-to-secondary leakage rate to a TS limited rate of 1 gpm apportioned between all SGs in such a manner that maximizes the calculated dose consequence. The licensee assumed that, prior to the event, this leakage is distributed throughout the three SGs. Recognizing that an extended plant cooldown may be required under natural circulation conditions, the licensee conservatively analyzed the SGTR using 24 hours for the cooldown time. The activity associated with the 1 gpm leak was conservatively assumed to be released to the environment via the two intact SGs for the 24-hour duration of the event. A partition coefficient for iodine of 100 was assumed for this leakage. These assumptions are consistent with the applicable guidance of RG 1.183 and conservative in that the actual maximum value allowed by VCSNS TS 3.4.6.2.c for any one SG is 150 gallons per day (~0.1 gpm). For the initial release of secondary coolant activity to the environment, the licensee also assumed a partitioning factor of 100.

Where applicable, the licensee's model is consistent with the guidance of RG 1.183. In addition, by selecting limiting parameters and by assuming maximized releases, the licensee conservatively modeled SGTR accident activity transport. Therefore, the NRC staff finds this release methodology to be acceptable.

Summary for SGTR

The licensee stated that the radiological consequences at the EAB, LPZ, and control room are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in SRP 15.0.1 and RG 1.183. These accident-specific dose acceptance criteria for the SGTR assuming a PIS are a TEDE of 25 rem at the EAB for the worst 2 hours, 25 rem at the outer boundary of the LPZ, and 5 rem for the control room. For the SGTR assuming a CIS, the accident-specific dose acceptance criteria are a TEDE of 2.5 rem at the EAB for the worst two hours, 2.5 rem at the outer boundary of the LPZ, and 5 rem for the control room. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the VCSNS FSAR as design bases. The NRC staff also performed an independent calculation of the dose consequences of the SGTR accident using the licensee's assumptions for input to the RADTRAD computer code. The NRC staff's calculation confirmed the licensee's dose results. The NRC staff found the major parameters and assumptions used by the licensee, as presented in Table 3.2.4, to be acceptable. The results of the licensee's design-basis radiological consequence calculation are provided in Table 3.2. The EAB, LPZ, and CR doses estimated by the licensee for the SGTR accident were found to meet the applicable accident dose criteria and are therefore acceptable.

3.2.5 Locked Rotor Accident (LRA)

The current VCSNS design-basis LRA analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The VCSNS licensing basis analysis is presented in FSAR Chapter 15.4.4, "Single Reactor Coolant Pump Locked Rotor." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated LRA. This reanalysis was performed to demonstrate that the ESFs designed to mitigate the radiological consequences at VCSNS will remain adequate after implementation of the AST. The licensee submitted the AST-based reanalysis of the LRA as part of the LAR. Included in this reanalysis are the assumptions, parameters, and newly calculated offsite and control room doses

associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their reanalysis of the postulated design-basis LRA. Specifically, the NRC staff's guidance is detailed in Appendix G of that document.

The licensee has defined the design-basis LRA as the instantaneous seizure of a single reactor coolant pump (RCP) shaft. Typically, such a seizure is postulated to occur due to a mechanical failure or a loss of component cooling to the pump shaft seals. The reactor coolant flow through the core would be asymmetrically reduced as the result of a shaft seizure on one pump. The licensee postulated that, at the beginning of the LRA, i.e., at the time the shaft in one of the RCPs is assumed to seize, the plant is assumed to be in operation under the most adverse steady-state operating conditions. The licensee determined these conditions to be maximum steady-state power level, maximum steady-state pressure, and maximum steady-state coolant average temperature. Finally, flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low flow signal.

Activity Source

To determine the maximum offsite and control room dose resulting from the postulated design-basis LRA, the licensee assumed 15.0 percent fuel cladding failure in their reanalysis. This failed fuel fraction is consistent with the current VCSNS design-basis, as shown VCSNS FSAR Chapter 15.4.4.3. The licensee's current licensing basis states that, based on the transient analyses, less than 15.0 percent of the fuel rods will exceed the safety analysis limit DNB ratio; therefore, the use of 15.0 percent in the design-basis LRA dose consequence analysis is conservative. For the postulated LRA, the initial thermal power is assumed to be 2,958 MWth, which is a factor of 1.02 times the current licensed thermal power of 2,900 MWth, in order to account for the ECCS evaluation uncertainty. The licensee assumed the design-basis radial peaking factor of 1.70. Additionally, the licensee accounted for current licensed values for fuel enrichment and burnup when determining the core isotopic inventory. Table 4.7-2 of Attachment 2 to the licensee's LAR submittal shows the core inventory used by the licensee for the LRA analysis. To account for gap fraction uncertainty in fuel that does not meet the criteria specified in footnote 11 of RG 1.183, the licensee multiplied these gap fractions by a factor of two. This is a conservative approach that is found to be acceptable to the NRC staff.

In addition to the activity released into the coolant from fuel that was damaged during the accident, the licensee assumed that RCS equilibrium specific activity based on operation with 1 percent fuel defects would also be available for release. This specific activity results from the release into the RCS of fission and corrosion products in VCSNS-characteristic quantities.

Consistent with RG 1.183 guidance, the licensee assumes that the radioactive iodine speciation released from failed fuel is 95.00 percent aerosol (particulate), 4.85 percent elemental, and 0.15 percent organic. Whereas, the radioactive iodine speciation released from the SGs is 97.00 percent elemental and 3.00 percent organic.

Transport Methodology and Assumptions

As described in the licensee's submittal, it was assumed that, following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the RCS causing the coolant to expand. At the same time, heat transfer to the shell side of the SGs is reduced, first because the reduced

flow results in a decreased tube side film coefficient, and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the SGs causes a pressure increase throughout the RCS. This actuates the automatic spray system, opens the power-operated relief valves (PORVs), and opens the pressurizer safety valves, in that sequence.

The licensee states that the three PORVs are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, the licensee does not credit the pressure reducing effect of the PORVs or the sprays. Considering this effect would reduce the magnitude of the release and subsequently the consequences of the accident.

The insertion of control rods, due to a low RCS flow trip, will terminate the power increase; however, a limited number of fuel pins are calculated to experience DNB for a short period of time and are therefore assumed to fail. The initial primary RCS activity and gap activity released into the primary RCS from the failed fuel, then leaks into and combines with the activity in the secondary system. Releases to the environment occur from the SGs, via the atmospheric relief valves (or safety valves), and from the condenser. The licensee does not, however, credit the condensers for holdup or dilution of any airborne activity release. This is consistent with the assumption of a coincident loss of offsite power.

As shown in Section 4.7.3 of Attachment 2 of the licensee's submittal, the licensee assumed constant RCS leakage following the postulated LRA until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C, as specified by RG 1.183. The licensee assumed a 24-hour cooldown period, based on the time at which the RHR system starts operation to cool down the plant. This is a change from the current VCSNS licensing basis assumption of 8 hours as shown in VCSNS FSAR Section 15.4.4.4, and upon approval of this LAR, the 24-hour cooldown period will become the new design-basis of VCSNS. This longer cooldown period is conservative, as it will result in a calculated protracted release of activity and higher subsequent dose consequences. The steam released from 0 to 24 hours is based on the amount of steam required to remove the residual heat from the primary and secondary systems, the decay heat generated in the core, and the reactor coolant pump heat.

The licensee assumed that the LRA occurs, releasing the iodine, noble gas, and alkali metal activities into the primary system to mix immediately and homogeneously with the primary RCS. For VCSNS, the current licensing basis primary-to-secondary leakage rate is 1 gpm. The licensee assumed all noble gas isotopes from the gas gap of the failed and defective fuel to be released from the primary RCS through the SGs without credit for reduction or mitigation. For the alkali metals and iodine, the licensee assumed that the primary-to-secondary leakage to the SGs mixed instantaneously and homogeneously with the secondary water without flashing. The radioactivity in the secondary water was then assumed to vaporize at a rate that is the function of the steaming rate and the partition coefficient. The partition factor for iodine of 100 may be assumed in accordance with RG 1.183. The licensee also conservatively applied this same partition factor to the alkali metals that were released. Because, as a design-basis, the licensee did not postulate any SG failure, tube uncovery and certain flashing phenomena do not apply.

The licensee assumed no credit for cleanup mechanisms (spray, filtration, plateout) in the primary or secondary systems for any releases.

Summary for LRA

The licensee stated that the radiological consequences at the EAB, LPZ, and control room are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in SRP 15.0.1 and RG 1.183. The accident-specific dose acceptance criteria for the LRA at VCSNS are a TEDE of 2.5 rem at the EAB for the worst 2 hours, 2.5 rem at the outer boundary of the LPZ and 5 rem for access to and occupancy of the control room. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the VCSNS FSAR as design bases. The NRC staff also performed an independent calculation of the dose consequences of the LRA using the licensee's assumptions for input to the RADTRAD computer code. The NRC staff's calculation confirmed the licensee's dose results. The NRC staff found the major parameters and assumptions used by the licensee, as presented in Table 3.2.5, to be acceptable. The results of the licensee's design-basis radiological consequence calculation are provided in Table 3.2. The EAB, LPZ, and CR doses estimated by the licensee for the LRA were found to meet the applicable accident dose criteria and are therefore acceptable.

3.2.6 Control Rod Ejection Accident (CREA)

The current VCSNS design-basis CREA analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The VCSNS licensing basis analysis is presented in FSAR Chapter 15.4.6, "Rupture of a Control Rod Drive Mechanism (Rod Cluster Control Assembly Ejection)." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated CREA. This reanalysis was performed to demonstrate that the ESFs designed to mitigate the radiological consequences at VCSNS will remain adequate after implementation of the AST.

The licensee submitted the AST-based reanalysis of the CREA as part of the LAR. Included in this reanalysis are the assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their reanalysis of the postulated design-basis CREA. Specifically, the NRC staff's guidance is detailed in Appendix H of that document.

As stated in Section 4.8.1 of Attachment 2 to the licensee' submittal, the licensee has defined the design-basis CREA as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster control assembly (RCCA) and drive shaft. The licensee stated that the consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. The insertion of RCCAs will terminate the event.

The licensee has identified two separate credible scenarios for activity release to the environment for the postulated rod ejection accident. The two scenarios are as follows: (1) a breach of the reactor pressure vessel (RPV) and containment leakage for 30 days, or (2) primary-to-secondary leakage and secondary steaming via the relief valves until cold shutdown, i.e., a secondary RCS release.

Activity Source

To determine the maximum offsite and control room dose resulting from the postulated design-basis CREA, the licensee assumed 10.0 percent fuel cladding failure, 0.25 percent fuel melt, and RCS equilibrium specific activity of fission and corrosion products based on operation with 1 percent fuel defects. The failed fuel fraction is consistent with the current VCSNS design-basis, as shown in VCSNS FSAR Chapter 15.4.6.4.3. The licensee's current licensing basis states that it is assumed that fission products are released from the gaps of all rods entering DNB, and in all cases considered, less than 10 percent of the rods entered DNB based on a detailed three-dimensional analysis. Although limited fuel melting at a hot spot was predicted for the full power cases, in practice melting is not expected since the analysis conservatively assumed that the hot spots before and after ejection were coincident. Therefore, the NRC staff finds that the assumption of 10.0 percent fuel cladding failure and 0.25 percent fuel melt for the VCSNS design-basis CREA dose consequence analysis is conservative. For the postulated CREA, the initial thermal power is assumed to be 2,958 MWth, which is a factor of 1.02 times the current licensed thermal power of 2,900 MWth, in order to account for the ECCS evaluation uncertainty. The licensee assumed the design-basis radial peaking factor of 1.70. Additionally, the licensee accounted for current licensed values for fuel enrichment and burnup when determining the core isotopic inventory, as presented in Table 4.8-2 of Attachment 2 to the licensee's LAR. In addition to the aforementioned sources of activity, the licensee also accounted for the availability and release of the maximum secondary coolant iodine concentration of 0.1 μCi/gm DE I-131. This activity will typically not contribute significantly to the doses resulting from sources of activity coming directly from damaged fuel, but consideration of this activity is conservative and appropriate.

The licensee assumed that 100 percent of the noble gas, 25 percent of the iodine, and 30 percent of the alkali metals contained in the fuel, which is estimated to experience melting, is released. For the 10 percent of the fuel that experiences cladding failure, 20 percent of the noble gas, 20 percent of the iodine, and 24 percent of the alkali metals are assumed to be released, which represents a conservative factor of 2 added to the release fraction guidance of RG 1.183 to account for gap fraction uncertainty in fuel that does not meet the criteria specified in footnote 11 of RG 1.183. These assumptions are conservative, consistent with the applicable regulatory guidance, and therefore acceptable to the NRC staff.

Consistent with RG 1.183 guidance, the licensee assumes that the radioactive iodine speciation released from failed fuel is 95.00 percent aerosol (particulate), 4.85 percent elemental, and 0.15 percent organic. Whereas, the radioactive iodine speciation released from the SGs is 97.00 percent elemental and 3.00 percent organic.

Transport Methodology and Assumptions

Consistent with Appendix H of RG 1.183, the licensee considered two release cases. In the first, 100 percent of the activity released from the fuel was assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100 percent of the activity released from the fuel was assumed to be completely dissolved in the primary coolant and available for release to the secondary system.

Containment Leakage Release

For the containment leakage release, the ejected control rod is assumed to breach the RPV, effectively causing the equivalent of a small break loss of coolant accident. The containment leakage release pathway considers a release of activity from the primary RCS directly into containment, where the licensee assumes it to mix instantaneously and homogeneously in that free volume. The licensee assumed that the activity leaks from the containment atmosphere to the environment at the design-basis containment leak rate for the first 24 hours of the accident. The current VCSNS design-basis containment leak rate, L_a, is equal to 0.2 percent per day, at containment peak pressure, as expressed in VCSNS FSAR Table 6.2-1. The licensee then assumed that, based on the containment pressure decreasing over time, the leak rate is reduced to 50 percent of the TS value (0.1 percent per day) at 24 hours and for the duration of the accident. This is consistent with the guidance of RG 1.183.

The licensee takes credit for natural deposition removal of aerosols utilizing the RADTRAD 10 percent Power's model discussed in Section 3.2.1 of this evaluation. Conservatively, the licensee takes no credit for the removal of released activity by the containment sprays or the internal containment recirculation HEPA filters.

Secondary RCS Release

For the secondary RCS release pathway, the licensee assumed no breach of the RPV following the rod ejection. In this case, the licensee assumed that RCS integrity was maintained and all activity from damaged fuel that has been mixed with the RCS leaks to the secondary RCS through the SG tubes at a TS-limited rate of 1.0 gpm of total leakage. Activity is then available for release to the environment through steaming of the SG PORVs. In addition to the activity released from the primary-to-secondary coolant, iodine activity in the secondary coolant at the TS limit was also assumed to be released.

Summary for CREA

The licensee stated that the radiological consequences at the EAB, LPZ, and control room are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in SRP 15.0.1 and RG 1.183. The accident-specific dose acceptance criteria for the CREA at VCSNS are a TEDE of 6.3 rem at the EAB for the worst two hours, 6.3 rem at the outer boundary of the LPZ and 5 rem for access to and occupancy of the control room. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the VCSNS FSAR as design bases. The NRC staff also performed an independent calculation of the dose consequences of the CREA using the licensee's assumptions for input to the RADTRAD computer code. The NRC staff's calculation confirmed the licensee's dose results. The NRC staff found the major parameters and assumptions used by the licensee, as presented in Table 3.2.6, to be acceptable. The results of the licensee's design-basis radiological consequence calculation are provided in Table 3.2. The EAB, LPZ, and CR doses estimated by the licensee for the CREA were found to meet the applicable accident dose criteria and are therefore acceptable.

3.2.7 Control Room Habitability and Modeling

The current VCSNS DBA analyses, as shown in FSAR Chapter 15, do not calculate control room (CR) dose. Therefore, the control room dose model provided in the revised DBA accident analyses that support this AST-based LAR, represents a change in the VCSNS licensing basis.

The licensee's revised analyses credits post-accident isolation with filtered recirculation and pressurization. The CR ventilation system is modeled as providing 1,000 actual cfm (acfm) ± 25 percent of makeup air in normal mode and 980 acfm ± 25 percent of makeup air in emergency mode. As stated by the licensee, this system provides 95 percent filtration of aerosol, elemental, and organic forms of iodine. The licensee assumed that the emergency mode of the CR ventilation system is initiated and available from time equal 0 hours for the LOCA, time equal 0.5 hours for the FHA, and time equal 2 hours for the MSLB, SGTR, LRA, & CREA SG release case (the CREA containment release case assumed 0.5 hours). As the bounding time is 0 hours, the new VCSNS design-basis will require the licensee to ensure automatic isolation of the CR from the onset of any given DBA, and maintain the filtration system's availability for the duration of the accident.

Also associated with the emergency mode of the VCSNS CR is a recirculation system that provides a flow rate of 21,250 cfm ± 10 percent (-10 percent was used and is conservative for recirculation flow) and 95 percent filtration of aerosol, elemental, and organic forms of iodine. The total unfiltered inleakage into the VCSNS CR assumed by the licensee was 200 acfm, and then an additional 10 acfm was added for CR ingress and egress, bringing the total to 210 acfm. This value was also assumed for the accident duration.

The licensee has determined that, due to the sensitivity of control room intake flow rate to pressure and temperature, current industry practice indicates that a correction factor should be applied to all flow into their CR. Therefore, each of the aforementioned intake and unfiltered inleakage flows were increased by the licensee-calculated factor of 1.0328; emergency mode makeup flow becomes 1,265 cfm and unfiltered inleakage becomes 243 cfm when the 25percent uncertainty is also conservatively considered. Inclusion of the calculated correction factor in the licensee's analysis yields conservative results and is therefore acceptable to the NRC staff.

The licensee's response to Generic Letter 2003-01, dated November 18, 2005 (Reference 9), indicates that the maximum measured unfiltered inleakage into the VCSNS CR was 41 scfm. Therefore, the modeled unfiltered inleakage value can be viewed as conservative and providing margin for future measurements of control room inleakage.

3.2.8 Summary – Section 3.2

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of the postulated DBA analyses with the proposed TS changes. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0. The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0 and finds, with reasonable assurance, that the licensee's estimates of the Control Room, EAB, and LPZ doses will comply with these criteria. The NRC staff further finds reasonable assurance that VCSNS, as modified by this approved license

amendment, will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and input parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological consequences of the DBAs.

3.3 Containment and Ventilation – Technical Specifications

3.3.1 Introduction

The requested amendment to the VCSNS licensing basis and TS supports full implementation of AST methodology with the exceptions that the current TID-14844 accident source term will remain the licensing basis for equipment qualification, NUREG-0737 evaluations other than CR habitability envelope (CRHE) doses, and FSAR accidents not included in RG 1.183.

3.3.2 Regulatory Evaluation

The NRC staff reviewed the licensee's application to ensure that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) activities proposed will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or the health and safety of the public. The NRC staff's review evaluated the licensee's assessment of the impact of the proposed changes to the VCSNS TS on design-basis analyses. The NRC staff evaluated the licensee's application base on the requirements of 10 CFR Part 50, Appendix A, General Design Criterion 19, "Control Room," as supplemented by Sections 6.4 and 6.4.1 of NUREG-0800, "Standard Review Plan", Generic Letter 2003-01, "Control Room Habitability" and Technical Specification Task Force (TSTF) traveler TSTF-51, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," Revision 2.

3.3.3 Technical Evaluation

The licensee is proposing to eliminate the following TSs and the associated limiting condition for operation and surveillance requirements:

3/4.7.11 "Spent Fuel Pool Ventilation System."
3/4.9.4 "Reactor Building Penetration."
3/4.9.8 "Reactor Building Purge Supply and Exhaust Isolation System."

With respect to TS 3/4.7.11, the licensee stated that a dose analysis performed in accordance with the guidance of RG 1.183 assumed no credit for filtration and no credit for the removal of radionuclides for a fuel-handling building release. The analysis found that the CR and offsite doses are within the regulatory limits of 10 CFR 50.67. Consequently the licensee concluded that removal of the spent fuel pool ventilation sub-system is justified. The licensee further stated that "This change is consistent with TSTF-51 which allows deletion of the OPERABILITY requirements during CORE ALTERATIONS for ESF mitigation features previously credited for the FHA."

With respect to TS 3/4.9.4, the licensee stated that a dose analysis performed in accordance with the guidance of RG 1.183 assumed no credit for filtration and no credit for radionuclide removal for a RB release, found that the resulting control room and offsite doses are also within the regulatory limits of 10 CFR 50.67. The licensee further stated that since the only accident postulated to occur that results in a significant radioactive release is the FHA the proposed TS change omitting operability requirements for RB penetration is justified.

For TS 3/4.9.8, the dose analysis, performed by the licensee in accordance with the guidance of RG 1.183 assumed no filtration, holdup, or radionuclide removal for a RB release. Again, the resulting control room and offsite doses are within the regulatory limits of 10 CFR 50.67. Consequently, the licensee concluded that the proposed TS change omitting operability requirements for the RB purge supply and exhaust isolation system is justified.

Section 50.36(c)(2)(ii)(C) states that "[A TS limiting condition for operation of a nuclear reactor must be established for] a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." For TSs 3/4.7.11, 3/4.9.4 and 3/4.9.8, the NRC staff evaluation determined that, because of the adoption of the AST, and as a result of conclusions that resulted from the dose analysis performed in accordance with the guidance of RG 1.183, these TSs and associated limiting conditions for operation and surveillance requirements are no longer required in order for VCSNS to be in compliance with the requirements of 10 CFR 50.36(c)(2)(ii)(C) and therefore removing them as TSs is acceptable.

With respect to Administrative Controls: Section 6.8.4.l "Ventilation Filter Testing Program (VFTP)," As a result of the removal of TSs 3/4.7.11, 3/4.9.4 and 3/4.9.8 the licensee proposed that corresponding changes be made in the VFTP. The VFTP supports TSs 3/4.7.11, 3/4.9.4 and 3/4.9.8. Therefore proposed changes to Section 6.8.4.l are as follows:

> 6.8.4. l.1 ESF ventilation systems, delete the requirement for an inplace test for the high efficiency particulate air (HEPA) filter in the spent fuel pool ventilation system.
>
> 6.8.4. l.2 ESF ventilation system, delete the requirement for an inplace test for charcoal adsorber in the spent fuel pool ventilation system.
>
> 6.8.4. l.3 ESF ventilation system, delete the requirement for a laboratory test of a sample of charcoal adsorber from the spent fuel pool ventilation system.
>
> 6.8.4. l.4 ESF ventilation system, delete the requirement to demonstrate a pressure drop across the combined HEPA filter, the prefilter, and the charcoal adsorber in the spent fuel pool ventilation system.

The NRC staff finds the requested changes acceptable because the associated TS has been eliminated, and consequently these tests and a demonstration of pressure drop, as applicable, are not required in order to maintain safety and demonstrate compliance with applicable regulations.

The licensee proposed deletion of the following radiation monitoring instrumentation, associated notes and action statements in TS table 3.3-6 and monitoring instrumentation surveillance requirements and note from TS table 4.3-3.

Table 3.3-6: Radiation Monitoring Instrumentation.

1. Area Monitors

    b. RB manipulator crane area (RM-G17A or RM-G17B)

2. Process Monitor

    a. spent fuel pool exhaust ventilation system (RM-A6)

        i. gaseous activity
        ii. particulate activity

    b. containment

        i. gaseous activity purge & exhaust isolation (RM-A4)

Action Statements:

ACTION 27- With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.11.

ACTION 28 – With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.8.

Table 4.3-3: Radiation Monitoring Instrumentation Surveillance Requirements.

1. Area Monitors

    b. RB manipulator crane area (RM-G17A or RM-G17B)

2. Process Monitors

    a. spent fuel pool exhaust area ventilation system (RM-A6)

        i. gaseous activity
        ii. particulate activity

    b. containment

        i. gaseous activity and purge exhaust isolation (RM-A4)

For Table 3.3-6: "Radiation Monitoring Instrumentation."

The licensee stated that the new dose analysis performed in accordance with RG 1.183 assumes no filtration or radionuclide removal and that the containment is open for the duration of the event involving an FHA and that under these assumptions, the resulting control room and offsite doses are within the regulatory limits of 10 CFR 50.67. The NRC staff finds the licensee's assumptions to be acceptable and finds that the proposed TS change omitting the use of RM-G17A or RM-G17B and the associated action 28 to initiate automatic isolation of the containment purge and exhaust is acceptable.

Since the NRC staff finds the licensee's assumptions in the new dose analysis performed in accordance with RG 1.183 of no filtration or radionuclide removal for the fuel handling building release to be acceptable and since the resulting control room and offsite doses are within the regulatory limits of 10 CFR 50.67, the NRC staff consequently finds the removal of operability requirements for the spent fuel pool exhaust ventilation system radiation monitor (RM-A6), i. gaseous activity, and ii. particulate activity and associated ACTION 27 to be acceptable.

As previously indicated, the new analysis is performed in accordance with the guidance of RG 1.183 to meet the regulatory requirements of 10 CFR 50.67. During the FHA, the containment is assumed to be open for the duration of the event. The licensee stated that since the only accident postulated to occur during mode 6 [Refueling] that results in a significant radioactive release is the FHA, the proposed TS change omitting the use of RM-A4 to initiate automatic isolation of the containment purge and exhaust is justified. The NRC staff finds this acceptable because the new dose analysis performed in accordance with the guidance of RG 1.183 meets the regulatory requirements of 10 CFR 50.67 while taking no credit for RM-A4.

Because the NRC staff finds these proposed changes to Table 3.3-6 acceptable, the NRC staff concludes that the proposed removal of the two notes (** With irradiated fuel in the storage pool, and *** Alarm/trip setpoint will be per the Operational Dose Calculation Manual when purge exhaust operations are in progress) is administrative and is therefore acceptable.

With respect to TS Table 4.3-3: "Radiation Monitoring Instrumentation Surveillance Requirements," Area monitor instrument 1b, "Reactor Building manipulator crane area (RM-G17A or RM-G17B)," the NRC staff finds that, as a result of the proposed changes in table 3.3-6 where it was determined that automatic isolation of the containment purge and exhaust is no longer required for mitigation of the FHA, the associated surveillance requirement for RM-G17A or RM-G17B is no longer necessary the removal of item 1.b is acceptable.

For TS Table 4.3-3, Item 2.a. "Spent fuel pool exhaust area ventilation system (RM-A6), i. gaseous activity, and ii. particulate activity," the NRC staff finds that, because of the proposed changes in table 3.3-6 (where the operational requirements of RM-A6 is no longer required since filtration or radionuclide removal for a fuel handling building release is no longer required for mitigation of the fuel handling accident) the associated surveillance requirement for RM-A6 is no longer necessary and the removal of item 2.a is acceptable.

For TS Table 4.3-3, Item 2.b.i, "Gaseous Activity Purge and Exhaust Isolation (RM-A4)," the staff finds that consistent with the change in TS Table 3.3-6 to delete the operational requirements for RM-A4, based on the conclusion that automatic isolation of the containment purge and exhaust is

no longer required for mitigation of the FHA, the associated surveillance requirement for RM-A4 is no longer required and therefore the removal of item 2.b.i is acceptable.

Since the NRC staff finds the proposed changes in Table 4.3-3 acceptable, the NRC staff also finds the proposed removal of the note (** With irradiated fuel in the storage pool) is administrative and is acceptable.

The NRC staff also reviewed and found acceptable the licensee's proposed changes to the corresponding TS Table of Content and index entries. The licensee made associated changes to the TS Bases.

3.4 Maintenance of pH in Containment Sump Water

3.4.1 Introduction

The NRC staff reviewed the licensee's analysis for maintaining sump pool pH ≥ 7 for 30 days following a LOCA. This review is based on the licensee's submittals dated February 17, 2009, June 15, 2009, and December 1, 2009.

3.4.2 Regulatory Evaluation

Implementation of the alternative source term (AST) by the licensee required re-analyzing several DBAs using new source terms pursuant to 10 CFR, Part 50, Section 50.67, "Accident source term". As a result of improved understanding of the mechanisms of the release of radioactivity, the current accident source term may be replaced by a less restrictive AST. However, this replacement is subject to performing a successful re-evaluation of the major DBAs. The guidance for implementation of an AST is provided in RG 1.183.

3.4.3 Technical Evaluation

According to NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," iodine released from the damaged core to the containment after a LOCA is composed of 95 percent cesium iodide which is a highly ionized salt, soluble in water. Iodine in this form does not present any radiological problems since it remains dissolved in the sump water and does not enter the containment atmosphere.

However, in the radiation field existing in the containment, some of this iodine could be transformed from the ionic to the elemental form, which is scarcely soluble in water and can be, therefore, released to the containment atmosphere. Conversion of iodine to the elemental form depends on several parameters, of which pH is one of the more important. Maintaining pH basic in the sump water, i.e., greater than 7.0, will ensure that this conversion will be minimized. The licensee used the method described in NUREG/CR-5950, "Iodine Evolution and pH Control," for calculating generation of this elemental iodine. The licensee's calculations indicate that at the higher sump water pH, less of the iodides is converted into elemental form and at a pH of 7 or higher, elemental iodine generated from this source becomes insignificant relative to the elemental iodine released directly to the containment from the damaged core. The pH of the sump water in the VCSNS is controlled by the sodium hydroxide buffer which is formed by the addition of sodium hydroxide to the containment spray from the sodium hydroxide storage tank

and the boric acid dissolved in the sump water after a LOCA. After a LOCA, several acids are either generated or are added to the containment. Relative amounts of these acids and that of sodium hydroxide determine the value of pH reached by the containment sump water.
After a LOCA, boric acid from the reactor coolant system, accumulators, refueling water storage tank, and sodium hydroxide storage tank is discharged into the sump. For the minimum pH design case, the licensee determined the maximum mass for boric acid to be 58,245 pound-mass or 0.2708 moles per liter. Also, the pH will continuously decrease as a result of hydrochloric and nitric acid formation in the containment. Hydrochloric acid (HCl) is formed from decomposition of chlorinated polymer cable insulation by radiation; the licensee conservatively overestimated the cable insulation weight. The licensee used a generation rate of 9.97 $x10^{-4}$ moles of HCl per pound of insulation per Megarad (Mrad), and states that this is more conservative than the value from NUREG-5950 since it produces 2.2 times more HCl per pound of exposed material. Nitric acid is formed in the containment by irradiation of water and air. The amount of nitric acid produced is proportional to the time-integrated dose rate for gamma and beta radiation. Both acids are strong acids and will contribute to lowering sump pH. In order to neutralize the boric, hydrochloric and nitric acids, the licensee chose to buffer the sump pool water by using a sodium hydroxide buffer. Such buffering action is intended to maintain basic pH in the sump pool despite the presence of the acids. The licensee has calculated that by adding 3829 pound-mass or 2.7516 $x10^{-2}$ moles per liter of sodium hydroxide solution, at a reference temperature of 77 °F, from the sodium hydroxide storage tank to the sump pool during the injection phase, it will maintain the pH in the sump water as basic.

The NRC staff has independently verified the licensee's calculations and finds that by using sodium hydroxide as a buffer in the quantity specified, the pH of the sump will remain above 7 for 30 days following a LOCA.

3.4.4 Summary for Section 3.4

The NRC staff reviewed the licensee's assumptions, methodology, and conclusions regarding the pH of sump water and the corresponding fraction of the dissolved iodine in the sump water that is converted into the elemental form. The calculations were made for the 30-day period following a LOCA. The NRC staff performed independent calculations to verify the licensee's calculation. From the results of these calculations, the NRC staff concludes that although the value of pH varied with time it does not decrease below 7.5. Maintaining pH above 7 resulted in a negligible fraction of the dissolved iodine converting into elemental form and a low release of radioactive iodine to the environment. Based on these considerations the NRC staff finds the licensee's proposal acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South Carolina State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment includes changes to a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that these parts of the amendment involve no significant increase in the amounts, and

no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, these parts of the amendment meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). The amendment also includes parts that relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, these parts of the amendment meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment. The Commission previously issued a proposed finding in the Federal Register March 24, 2009, (74 FR 12395) that the amendment involves no significant hazards consideration, and there has been no public comment on such finding.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from SCE&G to NRC, "License Amendment And Related Technical Specification Changes to Implement Full-Scope Alternative Source Term in Accordance with 10 CFR 50.67," February 17, 2009, Agencywide Documents Access and Management System (ADAMS) Accession No. ML090720887.

2. Letter from SCE&G to NRC, Supplement to LAR, June 15, 2009, ADAMS Accession No. ML091680061.

3. Letter from SCE&G to NRC, Supplement to LAR, December 1, 2009, ADAMS Accession No. ML093410537.

4. Letter from SCE&G to NRC, Response to RAI, December 23, 2009, ADAMS Accession No. ML100150990.

5. Letter from SCE&G to NRC, Supplemental response to RAI, January 14, 2010, ADAMS Accession No. ML100210969.

6. Letter from SCE&G to NRC, providing supplemental information, July 16, 2010, ADAMS Accession No. ML102020199.

7. J. J. DiNunno et al., "Calculation of Distance Factors for Power and Test Reactor Sites," USAEC TID-14844, U.S. Atomic Energy Commission (now USNRC), 1962.

8. NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation."

9. Letter from SCE&G to NRC, Response to NRC Generic Letter 2003-01, "Control Room Habitability," November 18, 2005, ADAMS Accession No. ML053260454.

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Table 3.1.2

VCSNS Control Room Atmospheric Dispersion Factors

Design-basis Accident	Source / Receptor	$^{\chi}/_{Q}$ Values				
		0-2 Hours	2-8 Hours	8-24 Hours	24-96 Hours	96-720 Hours
		sec/m^3	sec/m^3	sec/m^3	sec/m^3	sec/m^3
LOCA, FHA, CREA	Reactor Building Nearest Point to Intake A	1.39E-03	1.17E-03	5.70E-04	4.17E-04	3.00E-04
FHA	Main Plant Vent to Intake B	7.43E-04	5.41E-04	2.75E-04	2.16E-04	1.49E-04
MSLB, SGTR, LRA, CREA	MS SSV A (Reliefs B, C,D,E) to Intake B	1.51E-03	1.17E-03	5.75E-04	4.18E-04	3.10E-04

Table 3.1.3

VCSNS Offsite Atmospheric Dispersion Factors

Offsite Dose Location	$^{\chi}/_{Q}$ Values				
	0-2 Hours	0-8 Hours	8-24 Hours	24-96 Hours	96-720 Hours
	sec/m^3	sec/m^3	sec/m^3	sec/m^3	sec/m^3
EAB	1.24E-4	----	----	----	----
LPZ	----	2.42E-5	1.68E-5	7.55E-6	2.40E-6

Table 3.2

Licensee Calculated Radiological Consequences of Design-basis Accidents

Design-basis Accident	Control Room		[a]EAB		LPZ	
	[b]Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)	[c]Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)	[d]Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)
LOCA[e]						
DBA Base	1.01E+00	5.0	4.87E+00	25	5.42E-01	25
Max ESF	5.00E+00	5.0	9.88E+00	25	5.99E+00	25
Max CR	5.00E+00	5.0	4.87E+00	25	5.42E-01	25
FHA	7.60E-01	5.0	4.29E+00	6.3	1.06E+00	6.3
MSLB						
PIS	1.15E+00	5.0	1.96E+00	25	1.30E-01	25
CIS	3.70E-01	5.0	7.80E-01	2.5	1.60E-01	2.5
SGTR						
PIS	1.18E+00	5.0	2.08E+00	25	1.20E-01	25
CIS	3.70E-01	5.0	7.10E-01	2.5	4.70E-02	2.5
LRA	2.43E+00	5.0	2.20E+00	2.5	4.50E-01	2.5
CREA						
Cont.	1.71E+00	5.0	4.31E+00	6.3	9.50E-01	6.3
SG	2.38E+00	5.0	2.52E+00	6.3	4.80E-01	6.3

a The licensee calculated the EAB dose for the worst 2-hour period of the accident duration.
b The licensee's control room dose results have been rounded to three significant digit precision.
c The licensee's EAB dose results have been rounded to three significant digit precision.
d The licensee's LPZ dose results have been rounded to three significant digit precision.
e In analyzing their design-basis LOCA, the licensee presented cases of maximized allowable control room (CR) unfiltered inleakage and maximized ESF leakage. In these cases, the limiting dose location, the control room, has been intentionally calculated to the applicable dose limit. The resulting doses of all cases are presented; the latter two for information.

Table 3.2.1

Key Parameters Used in Radiological Consequence Analysis of Loss of Coolant Accident

Parameter	Value
Reactor Core Power, MWth	2958
Containment Volume, ft^3 Total	1,840,000
Spray Delay Time, seconds	52
Spray Iodine Removal Coefficient, hr^{-1} Elemental Organic Aerosol/Particulate 52 – 2176 seconds 2176 seconds – 4 hours	 20 Not credited. 5.68 0.568
Sprayed and Unsprayed Region Mixing Rate, acfm	54,200
Primary Containment Leakage Rate, volume percent per day 0 to 24 hours 24 hours to 30 days	 0.2 0.1
Aerosol Iodine Natural Deposition Model	Powers 10th Percentile
Reactor Building Cooling System Flow Rates, acfm	54,200
Reactor Building Cooling System Activity Removal, percent Aerosol/Particulate 52 seconds to 30 days Elemental Organic	 90 0 0
ECCS Leakage Start, seconds	1460
ECCS Leakage, cm^3/hr	12,000
ECCS Leakage Flashing, percent	10
ECCS Iodine Release Species, percent Elemental Organic	 97 3
Atmospheric Dispersion Factors	Tables 3.1.2 and 3.1.3

Table 3.2.2

Key Parameters Used in Radiological Consequence Analysis of Fuel Handling Accident

Parameter	Value
Reactor Core Power, MWth	2958
Peaking Factor	1.7
Number of Failed Fuel Rods	314
Fuel Decay Time, hr	72
Fraction of Core Inventory in Fuel Gap Kr-85 I-131 Other Noble Gases Other Iodines Alkali Metals	 0.20 0.16 0.10 0.10 0.24
Minimum Water Depth Above Damaged Fuel, ft	23
Iodine Decontamination Factor	200
Iodine Speciation Upon Release from Fuel Gap, percent Elemental Organic	 99.85 0.15
Iodine Speciation Upon Release from Pool, percent Elemental Organic	 57 43
Fuel Activity Release Duration, hr	2
Atmospheric Dispersion Factors	Tables 3.1.2 and 3.1.3

Table 3.2.3

Key Parameters Used in Radiological Consequence Analysis of Main Steam Line Break Accident

Parameter	Value
Reactor Core Power, MWth	2958
Peaking Factor	1.7
Failed Fuel (defects), percent	1.0
Primary RCS Equilibrium Activity, µCi/gm DE I-131 Preaccident Iodine Spike Factor Concurrent Iodine Release Rate Spike Factor	1.0 60 500
Secondary RCS Equilibrium Activity, µCi/gm DE I-131	0.1
Fraction of Core Inventory in Fuel Gap Kr-85 I-131 Other Noble Gases Other Iodines Alkali Metals	 0.20 0.16 0.10 0.10 0.24
Iodine Speciation from Failed Fuel, percent Elemental Organic Aerosol/Particulate	 4.85 0.15 95.0
Iodine Speciation from Steam Generator, percent Elemental Organic	 97 3
Total Release Duration, hr	24
Steam Release from SG, lbm Faulted SG 0 – 30 minutes Intact SGs 0 – 2 hours 2 hours – 8 hours 8 hours – 24 hours	 406,000 343,700 733,900 1,200,000
RCS Primary-to-Secondary Leak Rate, gpm	1.0
Atmospheric Dispersion Factors	Tables 3.1.2 and 3.1.3

Table 3.2.4

Key Parameters Used in Radiological Consequence Analysis of Steam Generator Tube Rupture Accident

Parameter	Value
Reactor Core Power, MWth	2958
Peaking Factor	1.7
Primary RCS Equilibrium Activity, µCi/gm DE I-131 Preaccident Iodine Spike Factor Concurrent Iodine Release Rate Spike Factor	1.0 60 335
Secondary RCS Equilibrium Activity, µCi/gm DE I-131	0.1
Iodine Speciation, percent Elemental Organic	 97 3
Total Release Duration, hr	24
RCS Primary-to-Secondary Leak Rate, gpm	1.0
Coolant Release from Tube to Faulted SG, lbm 0 – 385 seconds 385 seconds – 30 minutes	 19,400 73,500
Steam Release from Intact SG, lbm 0 – 2 hours 2 hours – 8 hours 8 hours – 24 hours	 381,400 924,900 1,200,000
Steam Release from Faulted SG, lbm 0 – 30 minutes	 56,800
Atmospheric Dispersion Factors	Tables 3.1.2 and 3.1.3

Table 3.2.5

Key Parameters Used in Radiological Consequence Analysis of Locked Rotor Accident

Parameter	Value
Reactor Core Power, MWth	2958
Peaking Factor	1.7
Failed Fuel, percent Cladding Failure Fuel Defects	 15.0 1.0
Fraction of Core Inventory in Fuel Gap Kr-85 I-131 Other Noble Gases Other Iodines Alkali Metals	 0.20 0.16 0.10 0.10 0.24
Iodine Speciation from Failed Fuel, percent Elemental Organic Aerosol/Particulate	 4.85 0.15 95.0
Iodine Speciation Released from SG to Environment, percent Elemental Organic	 97 3
Time Until Shutdown Cooling is Established, hr	24
RCS Primary-to-Secondary Leak Rate, gpm	1.0
SG Blowdown Flow Rate, lbm/hr	12,756
Coolant Mass Release (SG Steaming), lbm 0 – 2 hours 2 hours – 8 hours 8 hours – 24 hours	 447,900 868,300 1,200,000
Total RCS Coolant Mass, lbm	400,000
Atmospheric Dispersion Factors	Tables 3.1.2 and 3.1.3

Table 3.2.6

Key Parameters Used in Radiological Consequence Analysis of Control Rod Ejection Accident

Parameter	Value
Reactor Core Power, MWth	2958
Peaking Factor	1.7
Failed Fuel, percent Cladding Failure Fuel Melt Fuel Defects	 10 0.25 1.0
Fraction of Core Inventory in Fuel Gap Noble Gas Iodine Alkali Metals	 0.20 0.20 0.24
Iodine Speciation from Failed Fuel, percent Elemental Organic Aerosol/Particulate	 4.85 0.15 95.0
Secondary RCS Equilibrium Activity, µCi/gm DE I-131	0.1
Containment Volume, ft^3 Total	 1,840,000
Aerosol Iodine Natural Deposition Model	Powers 10th Percentile
Primary Containment Leakage Rate, volume percent per day 0 to 24 hours 24 hours to 30 days	 0.2 0.1
Coolant Mass Release (SG Steaming), lbm 0 – 2 hours 2 hours – 8 hours 8 hours – 24 hours	 447,900 868,300 1,200,000
RCS Primary-to-Secondary Leak Rate, gpm	1.0
Atmospheric Dispersion Factors	Tables 3.1.2 and 3.1.3

Table 3.2.7

Key Parameters Used in Modeling the Control Room for Design-basis Radiological Consequence Analyses

Parameter	Value
Control Room Volume, ft^3	226,040
Recirculation Flow Rate, cfm	19,125
Emergency Filtration System Initiation Delay, hours	0
Emergency Filtered Intake Flow Rate, cfm	1265
Recirculation Filter Efficiency, percent Elemental Organic Aerosol/Particulate	 95 95 95
Emergency Intake Filter Efficiency, percent Elemental Organic Aerosol/Particulate	 95 95 95
Unfiltered Inleakage, cfm	243
Occupancy Factors 0 – 24 hours 24 – 96 hours 96 – 720 hours	 1.0 0.6 0.4
Breathing Rate, m^3/sec	3.5E-04
Atmospheric Dispersion Factors	Table 3.1.2