June 15, 2005

Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, Virginia 23060-6711

SUBJECT: NORTH ANNA POWER STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS ON IMPLEMENTATION OF ALTERNATE SOURCE TERM (TAC NOS. MC0776 AND MC0777)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. and to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Units 1 and 2, in response to your application dated September 12, 2003, as supplemented by letters dated November 20, 2003, March 30, April 20, May 7, May 27, August 18, and November 3, 2004, and February 17, 2005.

These amendments revise the Technical Specifications to incorporate a full-scope application of an alternate source term methodology in accordance with Title 10 of the *Code of Federal Regulations*, Section 50.67.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA

Stephen Monarque, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No. 240 to NPF-4
2. Amendment No. 221 to NPF-7
3. Safety Evaluation

cc w/encls: See next page

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VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 240
Renewed License No. NPF-4

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

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated September 12, 2003, as supplemented by letters dated November 20, 2003, March 30, April 20, May 7, May 27, August 18, and November 3, 2004, and February 17, 2005, 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 Technical Specifications as

indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-4 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 240, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Evangelos C. Marinos, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 15, 2005

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 221
Renewed License No. NPF-7

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

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated September 12, 2003, as supplemented by letters dated November 20, 2003, March 30, April 20, May 7, May 27, August 18, and November 3, 2004, and February 17, 2005, 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 Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-7 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 221, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Evangelos C. Marinos, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 15, 2005

ATTACHMENT TO

LICENSE AMENDMENT NO. 240 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-4

LICENSE AMENDMENT NO. 221 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NOS. 50-338 AND 50-339

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
TS 1.1-2	TS 1.1-2
TS 3.7.10-1	TS 3.7.10-1
TS 3.7.10-2	TS 3.7.10-2
TS 3.7.13-1	TS 3.7.13-1
TS 3.7.13-3	TS 3.7.13-3
TS 3.9.4-1	TS 3.9.4-1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 240 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-4

AND

AMENDMENT NO. 221 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By letter dated September 12, 2003, as supplemented by letters dated November 20, 2003, March 30, April 20, May 7, May 27, August 18, and November 3, 2004, and February 17, 2005, Virginia Electric and Power Company (the licensee) submitted a proposed license amendment to incorporate full-scope application of an alternate source term (AST) methodology in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.67, "Accident source term." The supplements dated November 20, 2003, March 30, April 20, May 7, May 27, August 18, and November 3, 2004, and February 17, 2005, provided clarifying information only and did not change the Nuclear Regulatory Commission staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 9, 2003 (68 FR 68670) or expand the scope of the original application.

The licensee requested these license amendments in order to incorporate the AST in accordance with 10 CFR 50.67, revise the definition of dose equivalent Iodine (I)-131, eliminate the surveillance requirement (SR) to test the bottled air flow rate, relax containment closure requirements after a sufficient decay time to allow for movement of fuel, and revise the Technical Specifications (TS) requirements for the main control room and emergency switchgear room emergency ventilation systems. Furthermore, the licensee proposed to revise the design-basis accident (DBA) dose analyses to allow for a positive containment pressure for up to 4 hours after the DBA instead of the current limit of 1 hour.

2.0 REGULATORY EVALUATION

In the past, power reactor licensees have typically used U.S. Atomic Energy Commission Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962, as the basis for DBA analysis source terms. The power reactor siting regulation, 10 CFR Part 100, Section 11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," which contains offsite dose limits in terms of whole body and thyroid dose, makes reference to TID-14844.

In December 1999, the NRC issued 10 CFR 50.67, "Accident Source Term," which provides a mechanism for licensed power reactors to replace the traditional accident source term (TID-14844) used in their DBA analyses with an AST. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors." 10 CFR 50.67 requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of affected DBAs. The licensee's application dated September 12, 2003, as supplemented, addresses these requirements in proposing to use the AST, described in RG 1.183, as the North Anna DBA source term. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100 for a loss-of-coolant accident (LOCA), steam generator tube rupture (SGTR), main steamline break (MSLB) accident, fuel-handling accident (FHA), and locked-rotor accident (LRA).

The accident source term is intended to be representative of a major accident involving significant core damage and is typically postulated to occur in conjunction with a large LOCA. As a result of significant core damage, fission products are available for release into the containment environment. An AST is an accident source term that is different from the accident source term used in the original design and licensing of the facility and has been approved for use under 10 CFR 50.67. Although an acceptable AST is not set forth in the regulations, RG 1.183 identifies an AST that is acceptable to the NRC staff for use at operating reactors.

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The NRC staff used the following regulations, design criteria, and guides to evaluate the licensee's request.

10 CFR 50.67, “Accident source term,” requires a licensee to apply for a license amendment when revising its current accident source term in DBA radiological consequence analyses.

RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” as supplemented in Regulatory Position 4.4 and 10 CFR Part 50 Appendix A, General Design Criterion 4.

RG 1.109, Revision 1, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I,” as it relates to the calculation of Dose Equivalent I-131.

10 CFR Part 50, Appendix A, General Design Criterion 19 (GDC-19), “Control Room,” as supplemented by NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” (SRP) Section 6.4, “Control Room Habitability.”

NUREG-0800, SRP Section 15.0.1, “Radiological Consequence Analysis Using Alternative Source Terms.”

NUREG-0800, SRP Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” Rev. 2, 1988.

RG 1.23, “Onsite Meteorological Programs.”

NUREG/CR-5966, “A Simplified Model of Aerosol Removal by Containment Sprays,” USNRC, Office of Nuclear Regulatory Research, Prepared by Sandia National Laboratories, June 1993.

NUREG-0917, “Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data.”

RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” June 2003.

NUREG/CR-6331, Revision 1, “Atmospheric Relative Concentrations in Building Wakes.”

NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” February 1995.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the regulatory and technical analyses related to the radiological consequences of the DBA that were done by the licensee in support of this proposed license amendment. Information regarding these analyses was provided in Section 3 of the submittal dated September 12, 2003, and in supplementary letters dated November 20, 2003, and March 30, April 20, May 7, May 27, August 18 and November 3, 2004, and February 17, 2005. The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess these impacts and did independent calculations to confirm the conservatism of the licensee’s analyses. The findings of this SE input are based on the descriptions of the analyses and other supporting information submitted by the licensee. The NRC staff also considered relevant information in the North Anna Updated Final Safety Analysis Report (UFSAR) and the TS.

The licensee requested the following changes to the Bases and TS.

1. Allow positive containment pressure for up to 4 hours after the DBA versus the current allowance of 1 hour in the Bases for TS 3.6.4, “Containment Pressure,” TS 3.6.6, “Quench Spray System,” and TS 3.6.7, “Recirculation Spray System.”

2. Define recently irradiated fuel as fuel that has occupied part of a critical reactor core within the previous 100 hours in the Bases of TS 3.9.4, “Containment Penetrations,” and TS 3.7.15, “Fuel Building Ventilation System.”

3. Revise TS Limiting Condition for Operation (LCO) 3.7.10, “Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS) - Modes 1, 2, 3, and 4" to require two MCR/ESGR EVS trains to be operable rather than two trains from the affected unit and one train from the other unit.

4. Delete SR 3.7.10.3.

5. Delete the requirement to measure the bottled air flow rate in SR 3.7.13.4.

6. Change the TS definition of Dose Equivalent I-131 to include an allowance to use dose conversion factors from RG 1.109 in calculating Dose Equivalent I-131.

7. Revise TS 3.9.4, “Containment Penetrations” to require the air lock door to be closed during refueling operations.

3.1 Accident Dose Calculations

For its analysis of DBAs, the licensee used the guidance in RG 1.183. The licensee also used dose conversion factors from Federal Guidance Report (FGR) Nos. 11 and 12, which the NRC staff found acceptable, as documented in RG 1.183. The licensee calculated the TEDE to support the full implementation of AST at North Anna, Units 1 and 2. The NRC staff finds the licensee’s analysis meets the criteria of 10 CFR 50.67(b)(2).

In accordance with RG 1.183, a licensee is not required to reanalyze all DBAs for the purpose of the application, just those affected by the proposed changes. However, upon approval of this amendment, the AST and the TEDE criteria will become the licensing basis for all subsequent (except equipment qualification) radiological consequence analyses intended to show compliance with 10 CFR Part 50. The licensee did an evaluation of previously analyzed DBAs to decide which, if any, were affected by the proposed amendment. The licensee reanalyzed the radiological consequences of the following affected DBA events:

- LOCA
- FHA
- SGTR
- MSLB
- LRA

3.1.1 LOCA

The design-basis LOCA assumes that the accident is initiated by a major rupture of the reactor coolant system piping, and that the emergency core cooling systems (ECCS) do not provide sufficient cooling so that significant core melting occurs. The licensee analyzed the radiological consequences of a design-basis LOCA using the guidance in RG 1.183 for AST analyses. RG 1.183 provides a description of an alternative radiological source term in terms of the magnitude and mix of the fission-product radionuclides released from the fuel, their physical and chemical form, and the timing of their release. The licensee’s analyses projected the dose at the exclusion area boundary (EAB) for the worst 2 hours, and both the low-population zone (LPZ) boundary and in the control room for the duration of the accident, which is taken to be 30 days. In addition, the licensee’s analyses projected the dose for these three sites using atmospheric dispersion factors that were reviewed by the NRC staff in Section 3.2, “Atmospheric Relative Concentration Estimates” of this SE. The licensee also determined the projected dose in the control room due to control room filter loading.

3.1.1.1 Control Room Modeling and Filter Loading Analysis

To estimate the dose in the control room from airborne radioactivity, the licensee’s analysis assumed that the control room is automatically isolated and pressurized at the onset of the LOCA by the bottled air system, but it does not take credit for the injection of clean air by the bottled air system. After 1 hour, the air bottles are depleted and one train of the MCR/ESGR EVS is initiated to provide filtered outside air for pressurization of the control room. The licensee's analysis does not take credit for filtered recirculation in the control room. Regarding the dose to the control room operators caused by the airborne release from containment leakage, ECCS component leakage, and refueling water storage tank (RWST) leakage, the licensee assumed 250 cubic feet per minute (cfm) unfiltered inleakage for the duration of the LOCA.

The licensee also calculated the dose to the control room operators from the gamma radiation shine caused by the buildup of radioactive material on the MCR/ESGR EVS charcoal and HEPA filters. In its filter shine dose analyses, the licensee assumed that the MCR/ESGR EVS is recirculating air to model the buildup of the radioactive materials on the recirculation filters. By contrast, the licensee's analysis of the dose in the control room due to the LOCA release to the outside atmosphere did not assume filtered recirculation, in order to maximize the airborne radioactive material in the control room air, and therefore the calculated control room dose. The licensee conservatively assumed that both the intake and pressurization filters are on the same level of the control room envelope where the control room operators would be performing their duties for the duration of the LOCA. The licensee assumed a control room unfiltered inleakage of 500 cfm for the filter loading shine dose analysis, which would give a higher filter loading than the assumption used for the dose from the airborne release. All other source term and release pathway assumptions are the same as used in the airborne release dose analyses for the containment leakage and ECCS leakage pathways, with the exception of the assumed maximum ECCS leakage. The licensee's filter shine dose analyses assumed either 63,000 cubic centimeters per hour (cc/hr) filtered or 6,140 cc/hr unfiltered ECCS leakage. However, the licensee states that ECCS leakage will be limited to 17,200 cc/hr filtered or 1,700 cc/hr unfiltered based on the control room dose analysis for the airborne release.

To determine the isotopic loading on the intake filter, two additional dose computer code calculations were made for both the containment leakage and ECCS leakage pathways. In the first calculation control room recirculation, unfiltered inleakage, outflow from the control room, and filtration of the intake flow were not modeled. The second calculation was the same as the first, with the exception that the intake flow was assumed to be filtered. The difference in the control room isotopic inventory between the two runs is the amount deposited on the intake filter media. A similar pair of runs was performed to determine the amount deposited on the recirculation filter. The dose due to a release from leakage through the RWST was not specifically modeled; rather it was estimated by determining the ratio of I-131 inventory released from the RWST to that from the ECCS leakage and multiplying the ECCS filter loading dose by that ratio. After the isotopic inventories for the intake and recirculation filters were determined, the licensee used the ORIGENS computer code to calculate the gamma spectrum for each filter. Then, a conservative dose receptor point was selected to maximize the shine dose to a control room operator and the shine dose was calculated using the QADS code. Both QADS and ORIGENS are a part of the SCALE computer code system, which the NRC staff finds appropriate for use in shielding and dose calculations.

The licensee determined the total control room dose by adding together the doses calculated for the releases of radionuclides from the containment leakage, the ECCS leakage, the RWST leakage, and the control room filter loading shine. The total dose was maximized to determine the maximum allowable ECCS leakage as discussed in Section 3.1.1.4. The NRC staff finds that the licensee's calculated LOCA control room dose meets the GDC 19 and 10 CFR 50.67 dose criterion of 5 rem TEDE.

3.1.1.2 Iodine Removal Coefficient

The removal of particulate iodine by the containment sprays can be calculated according to SRP 6.5.2 or NUREG/CR-5966. According to NUREG-1465, the simplified model developed for NUREG/CR-5966 is considered more realistic than the SRP 6.5.2 methodology. The NUREG/CR-5966 model calculates the spray removal coefficient (λ) as a function of the containment spray water flux and spray fall height. Since a containment spray system may include separate quench and recirculation systems with different spray elevations and pump flow rates, the user determines the inputs to the model based on the spray systems that would be operating at a given time during the accident.

The North Anna AST application credits the containment spray systems with removal of the aerosol fission products during a LOCA according to the 10 percentile (most conservative) equation in NUREG-CR/5966. The spray systems at North Anna, Units 1 and 2 consist of three subsystems: quench, inside recirculation, and outside recirculation. The licensee made the conservative assumption that only one of the two spray trains in each of three subsystems would be operating. The removal coefficients were calculated as a function of time, with time intervals set by the start and stop times of spray header flows, the end of the early in-vessel release phase, and the limitation of 10 time steps in the RADTRAD code used to calculate radiological dose.

The NRC staff evaluated the assumptions made by the licensee, the way the methodology was applied, and the calculated λ values. The North Anna UFSAR was reviewed to identify the volume of the containment system and the characteristics of the spray subsystem components (elevation, pump flow rates, and droplet diameter). In its letter dated March 30, 2004, the licensee provided additional information and clarification about key parameters, in particular the selection of time intervals and the calculation of water flux.

The NRC staff then performed hand calculations of the λ values for particulate iodine removal using the equations from NUREG/CR-5966 (p. 153-154) and SRP 6.5.2 (p. 11-12), and compared them to the values in the licensee's application. The uncertainty (±) in the coefficients of the NUREG/CR-5966 equation was applied to determine the minimum and maximum values for a given set of conditions. The λ value was calculated at three time intervals corresponding to operation of the quench spray only, the recirculation spray only, and the two sprays concurrently. In the case of both sprays operating, λ values were calculated separately for each system and then summed. For the removal coefficient for elemental iodine, the licensee used the value of 10 hr^{-1} , which was already approved in the UFSAR, and as such this was not reviewed by the NRC staff.

The table below summarizes the particulate iodine removal coefficients submitted by the licensee and calculated by the NRC staff. As noted above, the range of λ values in the NRC staff calculations is a result of applying the uncertainty in the NUREG/CR-5966 model. The coefficients (λ values) calculated by the NRC staff using both methodologies agree with those in the licensee's application. In all cases the range of values that the NRC staff calculated using the NUREG/CR-5966 model bound the values in the application. The λ value calculated from the SRP for the end of the accident period (>8.6 hours, recirculation spray only) is higher than those calculated from the NUREG/CR-5966 model. This is because the equation in the SRP corresponds to any decontamination factor (DF) greater than 50, which is reached about

4 hours earlier in the analysis period. The coefficients that the NRC staff calculated using the more realistic methodology support the validity of the coefficients that the licensee submitted.

Spray Removal Phase (selected time interval)	**Aerosol Removal Coefficient (λ), hr^{-1}**		
	Licensee NUREG/CR-5966	NRC Staff NUREG/CR-5966	NRC Staff SRP 6.5.2
Quench Pump Only (time interval 0.025 - 0.080 hr)	3.73	3.65 - 4.42	3.87
Recirculation plus Quench (time interval 0.133 - 1.56 hr)	16.7	14.9 - 20.7	16
Recirculation only (time >8.6 hr)	1.42	1.25 - 1.80	1.6 (DF>50)

The NRC staff concludes that the spray removal coefficients for particulate iodine calculated by the licensee are acceptable because they were calculated using a realistic methodology, conservative inputs were used in applying the methodology, and the values are consistent with the NRC staff's review.

<u>Release Paths</u>

Once dispersed in the containment, the release to the environment is assumed to occur through the following three pathways: leakage from the containment, leakage from ECCS components, and leakage through the RWST.

3.1.1.3 LOCA Containment Leakage Pathway

The licensee's analysis of the LOCA, which was in accordance with the guidance in RG 1.183, assumes that the fission products from the core melting are released from the core into the containment atmosphere. A portion of the iodine in the containment atmosphere can be removed by the containment spray during operation of the containment spray systems. Containment leakage is assumed to occur until the containment is depressurized to subatmospheric pressure, which can be achieved within 4 hours after the start of the event. The licensee's analysis of the radiological consequences of LOCA containment leakage estimates the doses in the control room, at the EAB, and at the LPZ.

North Anna Units 1 and 2 have a subatmospheric containment design with the following acceptance criteria for the design-basis LOCA containment integrity analyses: (1) the calculated containment peak pressure must be less than 45 psig, (2) the containment must be depressurized to less than atmospheric pressure within 1 hour, and (3) the calculated peak pressure after 1 hour must be less than 0.0 psig. The licensee proposed to change TS Bases Sections 3.6.4, "Containment Pressure," 3.6.6, "Quench Spray System," and 3.6.7, "Recirculation Spray System," to revise the second and third criteria to state that, "The radiological consequences analysis demonstrates acceptable results provided the containment pressure decreases to 0.5 psig in 1 hour and does not exceed 0.5 psig for the interval from 1 to 4 hours following the DBA. Beyond 4 hours the containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment."

The NRC staff's acceptance of this change is based on the radiological calculations performed to show that this change will conform to the revised acceptance criteria. The licensee indicated that the LOCA analysis has assumed continued leakage during the 1-4 hour interval after the DBA, but at the diminished rate corresponding to a containment pressure of 0.5 psig. Beyond 4 hours, the pressure is assumed to be less than 0.0 psig, terminating leakage from the containment.

The licensee stated that the containment is modeled with a volumetric leak rate of 0.1 percent per day during the first hour (peak TS leak rate, L_a at maximum containment pressure) and 0.021 percent per day for the next 3 hours. The leak rate of 0.021 percent per day corresponded to the revised maximum allowable containment pressure of 0.5 psig for hours 1 through 4. The NRC staff confirmed that the calculated leak rate of 0.021 percent per day in the Surry AST was also applicable to North Anna. During the review of the Surry AST submittal, the NRC staff determined that the assumed leak rate of 0.021 percent per day was acceptable to use since it was conservative.

The licensee has not proposed any changes to the primary containment structure, heat removal systems, containment integrity (peak pressure and temperature) accident analyses, containment leakage rate testing, or the TS associated with any of these. The change in containment leak rate assumptions is only for offsite and control room dose analyses.

Note that the design-basis LOCA analysis for determining peak containment pressure and temperature demonstrates that a containment pressure of 0 psig is reached within 1 hour. Therefore, the assumption made for dose analysis is conservative in comparison to the design-basis containment LOCA analysis with respect to containment leakage.

3.1.1.4 LOCA ECCS Leakage Pathway

The ECCS fluid consists of the contaminated water in the sump of the containment, which is assumed to include 40 percent of the core inventory of the iodine isotopes. During a LOCA event, the ECCS fluid is pumped from the containment sump to the recirculation spray headers in order to cool and clean the containment atmosphere. Because some of the recirculation pumps are located in a building outside the containment, there is a potential for ECCS fluid leakage to occur in the outside atmosphere. In the licensee's model, recirculation of the containment sump fluid is assumed to start at 288.5 seconds after the LOCA. Ten percent of the iodine isotopes in the leaked ECCS fluid is assumed to become airborne and available for release to the outside environment. To model the amount of fluid leaking from the ECCS components that are located outside the containment, the licensee has backcalculated the maximum allowable ECCS leakage based on the limiting control room dose. The licensee calculated the offsite and control room doses for the containment leakage and RWST leakage pathways, and determined that the calculated control room dose had lesser margin to the dose criterion than the offsite doses. The ECCS modeled leakage was then increased until the control room dose from just the ECCS leakage pathway equaled the difference between the control room habitability dose criterion of 5 rem TEDE and the total of the dose from the containment leakage and RWST leakage release pathways. The allowable ECCS leakage will be limited based on the licensee's analysis to 1,700 cc/hr (modeled in the dose analysis as twice the limit or 3,400 cc/hr) if the leakage is unfiltered by the auxiliary building charcoal filters or 17,200 cc/hr (modeled as 34,400 cc/hr) if the leakage is filtered.

3.1.1.5 LOCA ECCS Back Leakage to RWST

Following a design-basis LOCA, the ECCS suction water source is switched from the RWST at the end of the injection phase to the containment sump for recirculation. The RWST is isolated during the recirculation phase and is located in the plant yard. This switch from ECCS injection to recirculation is assumed to occur at 30 minutes after the LOCA. The licensee has modeled the leakage of contaminated ECCS fluid through the isolation valves into the RWST, where subsequent leakage of evolved airborne iodine may be released to the outside environment through the vent located at the top of the RWST. The ECCS fluid leakage into the RWST includes 40 percent of the core inventory of iodine isotopes, and 10 percent of the iodine in the leaked fluid is assumed to evolve into the RWST air space. The licensee has assumed that a total 2,400 cc/hr of ECCS fluid is leaked into the RWST, starting at 30 minutes and continuing for the rest of the duration of the accident (30 days). Four cfm of outside air intake and release from the RWST is assumed for the same time period.

3.1.1.6 Evaluation of LOCA Analysis

The NRC staff reviewed the information provided in the licensee's submittals and the North Anna UFSAR and also performed independent calculations that confirmed the licensee's dose results. The licensee's analysis used assumptions and inputs that follow the guidance in RG 1.183. The NRC staff finds the licensee's calculated dose results are within the dose limits specified in 10 CFR 50.67 and GDC 19, and these calculated dose results are also within the dose acceptance criteria in SRP 15.0.1.

3.1.2 FHA

Regarding the analysis of the dose consequences of an FHA, the licensee assumed that cladding damage occurs to all of the fuel rods in one assembly. This event could occur either in the containment or the fuel building, with different release pathways. In support of the proposed changes to the TS requirements on containment closure and fuel building ventilation systems, the licensee assumed that the fuel has undergone 100 hours of radioactive decay and that the gap activity from the damaged fuel is released instantaneously into the spent fuel pool for the accident in the fuel building, or into the reactor cavity for the accident in the containment. The overlying water in the spent fuel pool or the reactor cavity is at least 23 feet above the top of the fuel. As such, the licensee was able to apply an iodine effective DF of 200 from RG 1.183 in order to model retention of iodine in the water. Appendix B of RG 1.183 identifies acceptable radiological analysis assumptions for an FHA. The licensee did not take credit for dilution or mixing of the release of radionuclides into the fuel building or containment. The licensee assumed the release from the spent fuel pool or reactor cavity is transported to the outside environment without filtration within 2 hours. The licensee determined the possible release locations for an FHA in the containment and an FHA in the fuel building. This allowed the licensee to model the FHA using the offsite and bounding control room atmospheric dispersion factor for the release of radionuclides from the personnel air lock through Vent Stack A. The atmospheric dispersion factors are discussed below in Section 3.2.

The licensee modeled isolation of the control room at the start of the FHA for the duration of the accident, with filtered pressurization initiated at 1 hour after the event. The licensee assumed 400 cfm of control room unfiltered inleakage.

The NRC staff reviewed the information provided in the licensee's submittals and the UFSAR. The NRC staff found that the licensee's analysis used assumptions and inputs that follow the guidance in RG 1.183 and are, therefore, acceptable. Additionally, the NRC staff did independent calculations that confirmed the licensee's conclusions. The NRC staff determined that the licensee's calculated dose results are within the dose limits specified in 10 CFR 50.67 and GDC 19. Furthermore, the NRC staff determined that the calculated offsite dose results are also within the dose acceptance criteria in SRP 15.0.1 for the FHA, which for the offsite dose is well within 10 CFR 50.67 limits (i.e., 6.3 rem TEDE).

3.1.3 SGTR

The postulated SGTR assumes that the reactor coolant flows through a break in the steam generator tubes into the secondary system and becomes available for release to the outside environment as the reactor is cooled down by steaming through the power-operated relief valves (PORVs) and steam generator safety valves. The affected steam generator is assumed to discharge steam to the environment for 30 minutes until the steam generator is isolated. The unaffected steam generators discharge steam to the environment for 8 hours until the reactor coolant system has cooled down enough to allow for the operation of the residual heat removal (RHR) system for the remainder of the cooldown.

Consistent with RG 1.183 guidance, the licensee considered two reactor coolant activity cases for the SGTR: one with a pre-existing iodine spike and another with an accident-initiated iodine spike. Appendix F of RG 1.183 identifies acceptable radiological analysis assumptions for an SGTR. The licensee also considered these conditions both with and without loss of offsite power (LOOP). For the pre-existing iodine spike case, the licensee assumed that the reactor coolant activity was at the TS maximum allowable iodine concentration level of 60 μCi/gm dose equivalent I-131 when the SGTR occurs. Regarding the accident-initated iodine spike case, the licensee assumed the reactor coolant activity was initially at the TS limit of 1 μCi/gm dose equivalent I-131; however, the fuel accident caused the radioiodine to be released from the fuel to the reactor coolant system at a rate of 335 times the normal radioiodine appearance rate for 8 hours. The initial reactor coolant system and secondary system liquid radionuclide inventories and concentrations had previously been determined for 1-percent failed fuel. These values are documented in Tables 11.1-6 and 11.1-7 of the North Anna UFSAR. For the analysis, these values were normalized to the TS limit for normal operation of 1 μCi/gm dose equivalent I-131. The licensee's modeling of the transport of the radionuclides from the reactor coolant system to the secondary system, then into the environment, is consistent with Appendix F of RG 1.183. In addition, the licensee's analysis of the iodine release from the coolant was found to be consistent with RG 1.183 guidance.

The licensee calculated the dose consequences of an SGTR using the offsite and bounding control room atmospheric dispersion factor for the release of radionuclides from the PORVs to the normal control room air intake. The atmospheric dispersion factors are discussed below in Section 3.2. The licensee's SGTR control room dose analysis did not assume control room isolation or EVS operation for the duration of the event. Although the licensee assumed the normal control room ventilation system unfiltered intake rate into the control room, the licensee's analysis also assumed an additional unfiltered inleakage of 500 cfm.

The NRC staff reviewed the information provided in the licensee's submittals and the UFSAR and also performed independent calculations that confirmed the licensee's dose results. The licensee's analysis used assumptions and inputs that followed the guidance in RG 1.183. As such, the NRC staff found this to be acceptable. The NRC staff performed independent calculations that confirmed the licensee's dose results are within the limits of 10 CFR 50.67 and GDC 19. In addition, the NRC staff determined that the offsite dose results are within the dose acceptance criteria specified in SRP 15.0.1 for the SGTR, which for the pre-existing iodine spike case is within 10 CFR 50.67 limits (25 rem TEDE) and for the accident-induced iodine spike case is a small fraction of 10 CFR 50.67 limits (i.e., 2.5 rem TEDE).

3.1.4 MSLB

The accident considered is a break in one of the main steamlines leading from the steam generator (affected generator) to the turbine. This MSLB is assumed to occur concurrently with a LOOP. As a result, the condenser is lost and cooldown of the reactor coolant system is through the release of steam from the steam generators. The affected steam generator is assumed to release steam for 30 minutes until it dries out, at which time it is isolated. In addition, the unaffected steam generators continue to release steam for 8 hours through the PORVs until the reactor coolant system has sufficiently cooled down to allow for the operation of the RHR system.

Consistent with RG 1.183 guidance, the licensee considered two reactor coolant activity cases for the MSLB: one with a pre-existing iodine spike and another with an accident-initiated iodine spike. In the pre-existing iodine spike case, the licensee assumed that the reactor coolant activity was at the TS maximum allowable iodine concentration level of 60 µCi/gm dose equivalent I-131 when the MSLB occurs. For the accident-initiated iodine spike case, the licensee assumed the reactor coolant activity was initially at the TS limit of 1 µCi/gm dose equivalent I-131. The accident causes the iodine to be released from the fuel at a rate 500 times the release rate (corresponding to 1 µCi/gm dose equivalent I-131) for a period of 8 hours. The licensee's modeling of the transport of the radionuclides from the reactor coolant system to the secondary system and into the environment is consistent with Appendix E of RG 1.183. Appendix E of RG 1.183 identifies acceptable radiological analysis assumptions for an MSLB. The licensee's analysis of the iodine release from the coolant is also consistent with RG 1.183 guidance.

The licensee calculated the dose consequences of an MSLB using the offsite and bounding control room atmospheric dispersion factor for the release of radionuclides from the PORVs to the normal control room air intake. The atmospheric dispersion factors are discussed below in Section 3.2. The licensee's MSLB control room dose analysis did not assume control room isolation or EVS operation for the duration of the event. Although the licensee assumed the normal control room ventilation system unfiltered intake rate into the control room, the licensee's analysis also assumed an additional unfiltered inleakage of 500 cfm.

The NRC staff reviewed the information provided in the licensee's submittals and the UFSAR and also performed independent calculations that confirmed the licensee's dose results. The licensee's analysis used assumptions and inputs that followed the guidance in RG 1.183. The NRC staff finds this to be acceptable. In addition, the NRC staff finds that the licensee's calculated dose results are within the dose limits specified in 10 CFR 50.67 and GDC 19.

The licensee's offsite dose results were determined to be within the dose acceptance criteria in SRP 15.0.1, which for the pre-existing iodine spike case is within 10 CFR 50.67 limits (25 rem TEDE) and for the accident-induced iodine spike case is a small fraction of 10 CFR 50.67 limits (i.e., 2.5 rem TEDE).

3.1.5 LRA

The LRA begins with either the instantaneous seizure of the rotor or a break in the shaft of a reactor coolant pump, resulting in a sudden decrease in reactor coolant flow through the core. This results in degradation of core heat transfer, which could result in fuel damage. The licensee has also assumed a coincident turbine trip and LOOP resulting in radionuclide release through the steam generator PORVs and safety valves. The release of radionuclides is assumed to continue for 8 hours until the RHR system is activated for reactor cooldown.

Although the current core cooling and departure from nucleate boiling (DNBR) analysis predicts no fuel failure would occur as a result of the LRA, the licensee assumes 13 percent of the core experiences fuel failure to bound potential future core design changes. The licensee also used a bounding fuel radial peaking factor that is a larger value than currently necessary to accommodate transition to a new fuel vendor and future core design changes.

The licensee's modeling of the transport of the radionuclides from the reactor coolant system to the secondary system, then into the environment, is consistent with Appendix G of RG 1.183. Appendix G of RG 1.183 identifies acceptable radiological analysis assumptions for an LRA. In addition, the licensee's Iodine release from the coolant is also consistent with RG 1.183 guidance.

The licensee calculated the dose consequences of an LRA using the offsite and bounding control room atmospheric dispersion factor for the release of radionuclides from the PORVs to the normal control room air intake. The atmospheric dispersion factors are discussed below in Section 3.2. The licensee's LRA control room dose analysis did not assume control room isolation or EVS operation for the duration of the event. Although the licensee assumed the normal control room ventilation system unfiltered intake rate into the control room, the licensee's analysis also assumed an additional unfiltered inleakage of 500 cfm.

The NRC staff reviewed the information provided in the licensee's submittals and the UFSAR and also performed independent calculations that confirmed the licensee's dose results. The licensee's analysis used assumptions and inputs that followed the guidance in RG 1.183. The NRC staff finds to be acceptable. The licensee's calculated dose results are within the dose limits specified in 10 CFR 50.67 and GDC 19. The NRC staff also determined that the licensee's offsite dose results for the LRA are also within the dose acceptance criteria specified in SRP 15.0.1, which is a small fraction of 10 CFR 50.67 limits (i.e., 2.5 rem TEDE).

3.2 Atmospheric Relative Concentration Estimates

3.2.1 Meteorological Data

The licensee used 5 years of hourly onsite meteorological data collected during calendar years 1997 through 2001 to generate new atmospheric dispersion factors (χ/Q values) for use in this license amendment request (LAR). Wind speed and direction were measured at 10 and 48.4 meters above the ground and the atmospheric stability categorization was based on temperature difference measurements between these two levels. The licensee stated that the data were collected in accordance with the guidance found in RG 1.23, "Onsite Meteorological Programs." These data were provided for NRC staff review in the form of hourly meteorological data files (for input into the ARCON96 atmospheric dispersion computer code). The data were used to generate control room air intake χ/Q values for the design-basis LOCA, FHA, SGTR, MSLB, and LRA events evaluated in this LAR. The resulting atmospheric dispersion factors represent a change from those used in the current UFSAR. The licensee used previously approved licensing basis χ/Q values for the EAB and LPZ χ/Q dose assessments.

The NRC staff did a quality review of the ARCON96 hourly meteorological database using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further NRC staff review was performed using computer spreadsheets. Examination of the data revealed that the data recovery during the 5-year period was consistently in the upper 90th percentiles, other than the upper level wind direction measurements in 2001, which had a recovery rate in the lower 90th percentile. These recovery rates meet the recommendation of RG 1.23.

With respect to atmospheric stability measurements, some stability conditions were reported with a higher variability of occurrence than at a typical site. The licensee noted that some of the apparent aberrant conditions were similar to concurrent observations at Surry, a neighboring licensee site, and was consistent with local weather or climate variations. In addition, the licensee identified two possible differences between North Anna and other nuclear power plant sites. They stated that the delta-T measurement interval used to estimate atmospheric stability at North Anna is smaller than that used at some other nuclear power plant sites. The smaller measurement interval may exaggerate the reported occurrence of stable and unstable stability readings under some conditions. Further, the North Anna site is adjacent to Lake Anna and the meteorological measurement tower is near the shoreline. The licensee noted that the water temperature appears to have a small effect on the stability measurements primarily in late spring and early fall when winds blowing over the lake can warm or cool the air near the ground. As a result of these considerations, NRC staff examined other available North Anna data and performed limited comparison calculations of the control room χ/Q values using onsite data for the 3-year period from 1996 through 1998. NRC staff judged that the 1996 through 1998 data should provide an adequate comparison with the 1997 through 2001 period. The 1996 data were provided as Enclosure 6 to a letter dated April 13, 2004, in support of another licensing action.

With respect to wind speed and direction, frequency distributions for each measurement channel were reasonably similar from year to year and when comparing measurements between the two heights and factoring in terrain effects.

In summary, the NRC staff has reviewed the available information relative to the ARCON96 meteorological data input files provided by the licensee. On the basis of this review, the NRC staff concludes that the 1997 through 2001 data provides an acceptable basis for estimating control room χ/Q values for the DBA assessments addressed in this LAR. However, these data should not be considered acceptable for use in other dose assessments without further NRC staff review.

3.2.2 Control Room Atmospheric Dispersion Factors

The licensee made numerous control room χ/Q calculations using guidance from Draft RG DG-1111 and, subsequently, RG 1.194, both entitled, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” and provided χ/Q values for the six postulated source and receptor pairs that were judged to result in the limiting dose cases based upon factors such as plant layout, design, and operation. The χ/Q values were calculated using 1997 through 2001 onsite meteorological data and the ARCON96 atmospheric computer code (NUREG/CR-6331, Revision 1, “Atmospheric Relative Concentrations in Building Wakes”). The resulting control room χ/Q values are presented in Table 1.

NRC staff performed a limited qualitative review of the inputs and assumptions and found them generally consistent with site configuration drawings and NRC staff practice. Specific areas of note are as follows:

- All releases were considered to be ground level. Postulated releases from the containment buildings were modeled as diffuse sources. In cases where more than one release scenario could be postulated, the most limiting pathway was assumed and the largest applicable χ/Q values, considering source category, operational line-ups, and accident timing, were selected for use in the dose assessment. When the limiting χ/Q values were not associated with a single source/receptor pair for the duration of the postulated accident, the most limiting χ/Q value for each time step was selected from among the possible source/receptor pairs.

- Because plant vent stack B is less than 10 meters from the control room air intake at location C-6 and RG 1.194 states that ARCON96 should not be used for source/receptor distances less than about 10 meters, the licensee stated that C-6 will be procedurally precluded from use as an emergency air intake. As specified by emergency procedures, the fan will only be operated in recirculation mode to recirculate air within the control room envelop but not draw air from outdoors. NRC staff notes that if the emergency procedure is changed to permit outside air to enter at location C-6, the licensee will need to calculate appropriate χ/Q values and assess the potential impact on doses.

- When modeling the dose for the SGTR and LRA, the licensee assumed plume rise and reduced the ground level χ/Q values calculated using ARCON96 by a factor of five, consistent with guidance in RG 1.194 for postulated releases from steam relief and atmospheric dump valves that meet certain criteria. Among the criteria, RG 1.194 states that the time-dependent vertical velocity must exceed the 95^{th} percentile wind speed at the release point height by a factor of at least 5. The licensee estimated a 95^{th} percentile wind speed of 5.7 meters per second (m/s) at the release height of 20.7 meters. NRC staff made confirmatory estimates from the 1997 through 2001 meteorological data and concluded that the licensee estimates appear reasonable. Thus, to ensure a ratio of at least a factor of five, the minimum vertical effluent exit speed at any time from any PORV would need to be at least 28.5 m/s. The licensee estimated the minimum speed from each of PORVs to be about 170 m/s. NRC staff approximations confirm that the ratio of the estimated minimum effluent exit speed to the 95^{th} percentile wind speeds is greater than a factor of five.

- As noted in Section 3.2.1 above, NRC staff made limited comparative χ/Q estimates using onsite meteorological data from 1996 through 1998. Resulting estimates were not significantly different from the licensee estimates for the cases considered.

In summary, the NRC staff has reviewed the licensee's assessments of control room post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. On the basis of this review, the NRC staff concludes that the χ/Q values in Table 1 are acceptable for use in DBA control room dose assessments addressed in this LAR. However, these data should not be considered acceptable for use in other dose assessments without further NRC staff review.

3.2.3 EAB/LPZ Atmospheric Dispersion Factors

With respect to the EAB and LPZ χ/Q values, the licensee has used its licensing basis values that have been previously approved at the time of licensing for North Anna, Units 1 and 2. As such, the NRC staff did not review these χ/Q values as a part of this LAR. The values are presented in Table 1.

3.3 <u>TS Definition of Dose Equivalent I-131</u>

The licensee has proposed changes to the TS definition of Dose Equivalent I-131 so that RG 1.109 DCFs can be used in the calculation of the coolant activity for TS surveillance. The purpose of calculating the dose equivalent I-131 is to help determine the overall iodine concentration in the reactor coolant for purposes of limiting the activity concentration. The use of RG 1.109 DCFs has been accepted in the past for other plants and is in the current standard Westinghouse TS. For consistency's sake, the NRC staff prefers the licensee use the same DCFs in the TS definition that are used in the licensee's DBA offsite and control room dose analyses (in this case, FGR-11). However, the licensee has demonstrated that estimated DBA releases from reactor coolant that contain a mix and concentration of isotopes based on RG 1.109 DCFs are in fact acceptable with respect to the dose criteria specified in both 10 CFR 50.67 and SRP 15.0.1. As such, the NRC staff finds the proposed change in the TS definition of Dose Equivalent I-131 to be acceptable.

3.4 <u>Ventilation System TS Changes</u>

3.4.1 Changes to TS 3.7.10

The licensee is proposing to revise TS 3.7.10 LCO to allow for two MCR/ESGR EVS trains to be operable during Modes 1, 2, 3, and 4. TS 3.7.10 LCO currently requires operability of two trains from the affected unit and one train from the other unit.

In addition, the licensee proposed the following changes to TS 3.7.10 Actions A and B in order to accommodate the proposed LCO change. Action A will be changed from "One required LCO 3.7.10.a or LCO 3.7.10.b MCR/ESGR EVS train inoperable," to "One required MCR/ESGR EVS train inoperable." Action B will be changed from "Two or more required LCO 3.7.10.a or LCO 3.7.10.b MCR/ESGR EVS trains inoperable due to inoperable MCR/ESGR boundary," to "Two required MCR/ESGR EVS trains inoperable due to inoperable MCR/ESGR boundary." The completion times for Actions A and B are 7 days and 24 hours, respectively, and will not change as a result of granting the proposed change.

Finally, the licensee is proposing to eliminate SR 3.7.10.3, which requires verification every 18 months that each LCO 3.7.10.a MCR/ESGR EVS train actuates on an actual or simulated actuation signal.

By letter dated February 17, 2005, the licensee stated that during a radiological accident while in Modes 1 through 4, the current dose consequence accident analyses assumes that the control room is isolated and pressurized by bottled air, and one train of the EVS provides filtered recirculation air flow in the MCR/ESGR. After depletion of the bottled air, which does not occur for at least 60 minutes, a second train of EVS is used for pressurization. As a result, three trains of EVS are currently required to meet the single failure criterion. It should be noted that cooling of the MCR/ESGR for equipment operability and human habitability is performed by a separate, safety-grade, redundant air conditioning system that recirculates the MCR/ESGR air and is governed by TS 3.7.11.

Based on the AST dose consequence analyses, which does not credit filtered recirculation air flow during Modes 1 through 4, the licensee proposed changes to LCO 3.7.10 since only one 100-percent capacity MCR/ESGR EVS train is required to provide pressurization. Specifically, the LOCA analysis credits EVS pressurization after the first hour and isolation of the control room, but the analyses does not credit recirculation of the air within the control room at any time. The MSLB, SGTR, and LRA events do not credit EVS recirculation, pressurization, or isolation in order to meet the dose limits. Therefore, in order to accommodate single failure criterion and maintain the MCR/ESGR habitable from a dose perspective for the AST case, only two EVS trains are required to be operable to meet the single failure criterion. The EVS requirements for the FHA are governed by TS 3.7.14. The FHA only credits one train of EVS for pressurization and only requires two trains to meet the single failure criterion.

The NRC staff reviewed the licensee’s information, dated February 17, 2005, and Section 9.4 of the UFSAR, and found in the AST accident analyses that no credit is taken for filtered recirculation air flow in Modes 1, 2, 3, and 4. Therefore, it is no longer necessary to maintain two MCR/ESGR EVS trains operable on the affected unit and one MCR/ESGR EVS train operable from the unaffected unit. Note that it is the NRC staff’s understanding that with this proposed change there will still be two trains of MCR/ESGR EVS per reactor unit, and each of these trains is capable of supplying 100 percent of the required capacity in the event of a DBA. Therefore, a single failure of an active component will not result in loss of the system’s functional performance. As such, the NRC staff finds the licensee’s proposed changes to TS 3.7.10 LCO to be acceptable.

Regarding the proposed change to SR 3.7.10.3, the licensee stated in its letter dated November 3, 2004, that the accident analyses take credit for operation of the air bottle system to pressurize the control room to $0.05 inches of water, consistent with the existing SR 3.7.13.4. As such, the NRC staff finds the requested change to SR 3.7.10.3 to be acceptable.

3.4.2 Changes to TS 3.7.13

The licensee is requesting that an editorial correction be made to Required Action C.1 of TS 3.7.13. The correction is to change “train” to “trains.” This is an editorial correction that does not change the intent of Required Action C.1; therefore the NRC staff finds it acceptable. In addition, the licensee is proposing to delete the part of SR 3.7.13.4 that requires monitoring the bottled air flow rate during the 18-month surveillance frequency. The SR would be changed from “verify two required MCR/ESGR bottled air system trains can maintain a positive pressure of $0.05 inches water gauge, relative to the adjacent areas at a makeup flow rate of $340 cfm for at least 60 minutes,” to “verify two required MCR/ESGR bottled air system trains can maintain a positive pressure of $0.05 inches water gauge, relative to the adjacent areas for at least 60 minutes.”

The licensee has determined that the flow rate portion of SR 3.7.13.4 is considered redundant and should be deleted since no credit is taken in the analyses for the cleansing effect of the bottled air. The requirement to pressurize the control room to $0.05 inches of water is retained because isolation and initiation of the bottled air system is credited in the LOCA and FHA analyses. Therefore, the requirement to measure 340 cfm of bottle airflow is deleted from SR 3.7.13.4. The NRC staff’s finds the requested change to SR 3.7.13.4 acceptable.

3.4.3 Associated Changes to TS Bases

In addition to the above TS changes, the licensee is changing the TS Bases to correspond to these TS changes, including the definition of recently irradiated fuel. The licensee is proposing that recently irradiated fuel be defined as fuel that has occupied part of a critical reactor core within the previous 100 hours in the Bases of TS 3.9.4 and 3.7.15.

The licensee stated that by defining recently irradiated fuel as fuel that has been part of a critical reactor core within the previous 100 hours, TS Bases 3.9.4 and 3.7.15 will not be applicable for movement of fuel that is conducted more than 100 hours after shutdown. Thus, these TS Bases will not be applicable for core offloads at North Anna that begin more than 100 hours after core shutdown. This change impacts the radiological consequences of the design-basis FHA. No other DBAs are impacted by these changes.

Although the TS Bases is a licensee-controlled document, the NRC staff’s review and assessment finds that this position is reflected in the accident analyses and therefore this proposed change is acceptable. This change is accounted for in the accident analysis and is consistent with TSTF-51. TSTF-51 was requested by the licensee and granted by the NRC staff in license amendment numbers 231 and 212 issued on April 5, 2002.

3.5 <u>Containment Closure Requirements - Proposed Change to TS 3.9.4</u>

The licensee is proposing a revision to TS 3.9.4 concerning the closure of containment penetrations during the handling of recently irradiated reactor fuel. With this revision, the licensee is defining “recently irradiated” as fuel that “has occupied part of a critical reactor core within the previous 100 hours.”

Although this revision limits the requirement to close containment during fuel handling, the licensee will implement procedures, consistent with the recommendation of RG 1.183, to ensure the capability to close the equipment hatch (the limiting case opening) beyond the 100-hour decay specification. A breach log to track containment openings will be maintained. Pre-designated individuals, including radiological protection personnel that have been trained and briefed prior to fuel movement, will be available to perform closure duties. Cables, hoses, etc. that penetrate the equipment hatch will be provided with quick disconnects. In addition, equipment needed to accomplish closure actions will be pre-staged.

In the event of an actual FHA, it is likely that the containment purge ventilation will isolate, resulting in the postulated release of airborne radioactive material through the open equipment hatch. However, there is considerable uncertainty in determining the motive force, transport pathway, and timing of this release. Although it is anticipated that the equipment hatch closure can be successfully completed within the 30-minute recommendation of RG 1.183, the licensee has committed only to closing the hatch, which requires actions from inside of containment if radiological conditions permit. Qualified Health Physics Technicians will monitor radiological conditions prior to and during closure activities. The need for additional protective measures (i.e., use respiratory protection devices or KI prophylaxis) or whether radiological conditions preclude completion of the hatch closure will be determined consistent with the licensee's emergency plan.

The licensee has demonstrated that containment closure is not necessary to meet the acceptance criteria in RG 1.183 for a postulated FHA. The containment closure actions committed to are to provide an additional defense-in-depth to the plant design. They are not vital actions required to mitigate the consequences of, or the recovery from, the postulated accident. Therefore, the proposed revision to TS 3.9.4 concerning containment closure during fuel handling is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments 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. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (68 FR 68672). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of a full-scope implementation of an AST and above-listed changes to TS on the definition of dose equivalent I-131, containment closure requirements, containment pressurization requirements, and control room ventilation system requirements at North Anna, Units 1 and 2. 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 above. The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses will continue to comply with these criteria. Therefore, the proposed changes to the TS and the implementation of an AST are acceptable with regard to the radiological consequences of postulated DBAs.

This licensing action is considered to be a full implementation of the AST. With this approval, the previous accident source term in the North Anna, Units 1 and 2 design basis is superseded by the AST proposed by the licensee. The previous offsite and control room accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67 or fractions thereof as defined in RG 1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in the North Anna, Units 1 and 2 design basis.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: June 15, 2005

TABLE 1

NORTH ANNA RELATIVE CONCENTRATION (χ/Q) VALUES (sec/m^3)

Exclusion Area Boundary (EAB) and Low Population Zone (LPZ)

EAB	0-2 hr	3.10×10^{-4}
LPZ	0-8 hr	1.10×10^{-5}
	8-24 hr	7.30×10^{-6}
	1-4 days	3.00×10^{-6}
	4-30 days	8.20×10^{-7}

Control Room

Source/Receptor	Accidents	0-2 hr	2-8 hr	8-24 hr	1-4 days	4-30 days
Vent stacks/ECR*	LOCA	3.75×10^{-3}	2.65×10^{-3}	1.03×10^{-3}	7.77×10^{-4}	5.70×10^{-4}
RWST** vent/ECR	LOCA	2.18×10^{-3}	1.42×10^{-3}	4.89×10^{-4}	3.84×10^{-4}	2.72×10^{-4}
Containment/ECR	LOCA	1.23×10^{-3}	9.02×10^{-4}	3.57×10^{-4}	2.55×10^{-4}	1.91×10^{-4}
Personnel airlock/ECR	FHA	3.75×10^{-3}	2.60×10^{-3}	1.03×10^{-3}	7.03×10^{-4}	5.52×10^{-4}
PORV/NCR*	SGTR, LRA	2.08×10^{-3}	1.64×10^{-3}	6.46×10^{-4}	4.50×10^{-4}	3.36×10^{-4}
PORV/NCR	MSLB	1.04×10^{-2}	8.20×10^{-3}	3.23×10^{-3}	2.25×10^{-3}	1.68×10^{-3}

* ECR is the emergency control room air intake and NCR is the normal control room air intake.
** RWST is the refueling water storage tank.

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