July 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: CORRECTION TO AMENDMENT NOS. 240 AND 221, FOR NORTH ANNA POWER STATION (TAC NOS. MC0776 AND MC0777)

Dear Mr. Christian:

On June 15, 2005, the U.S. Nuclear Regulatory Commission (NRC) issued Amendment Nos. 240 and 221 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Units 1 and 2. These amendments were 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.

The amendments revised 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. Virginia Electric and Power Company informed the NRC staff of inaccuracies and editorial corrections to the safety evaluation (SE) supporting the amendment. The NRC staff has resolved this by revising the appropriate wording in the SE. The corrected SE pages are included as an enclosure to this letter. Revisions are identified by lines in the margin.

The NRC regrets any inconvenience this may have caused. If you have any questions, please contact me at (301) 415-1157.

Sincerely,

/RA/
John Honcharik, 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: Pages 2, 5 and 18 of SE

cc w/encls: See next page

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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."

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.
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 recirculation 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.

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 not anticipated that the equipment hatch can be successfully completed within the 30-minute recommendation of RG 1.183, the licensee has committed to closing the hatch, which requires actions from inside of containment, within 45 minutes, 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.

North Anna Power Station, Units 1 & 2

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