VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

May 7, 2004

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 03-464D
NL&OS/ETS R0
Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM
REQUEST FOR ADDITIONAL INFORMATION
DOSE ASSESSMENT AND VENTILATION SYSTEMS

In a letter dated September 12, 2003 (Serial No. 03-464), Virginia Electric and Power Company (Dominion) requested amendments, in the form of changes to the Technical Specifications to Facility Operating Licenses Numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively. The proposed changes were requested based on the radiological dose analysis margins obtained by using an alternate source term consistent with 10 CFR 50.67. In an April 8, 2004 facsimile, the NRC staff requested additional information regarding dose assessment and operation of heating ventilation and air conditioning systems. The information requested is provided in the attachment to this letter.

If you have any further questions or require additional information, please contact Mr. Thomas Shaub at (804) 273-2763.

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachments

Commitments made in this letter: None

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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President - Nuclear Engineering, of Virginia Electric and Power Company. She has affirmed before me that she is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of her knowledge and belief.

Acknowledged before me this 7th day of May, 2004.

My Commission Expires: May 31, 2008.

Maggie McClure
Notary Public

(SEAL)

Attachment 1

Proposed Technical Specification Changes
Implementation of Alternate Source Term
Request for Additional Information
Dose Assessment and Ventilation System

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company
(Dominion)

NORTH ANNA ALTERNATIVE SOURCE TERM

In a letter dated September 12, 2003 (Serial No. 03-464), Virginia Electric and Power Company (Dominion) requested amendments, in the form of changes to the Technical Specifications to Facility Operating Licenses Numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively. The proposed changes were requested based on the radiological dose analysis margins obtained by using an alternate source term consistent with 10 CFR 50.67. In an April 8, 2004 facsimile, the NRC staff requested additional information regarding dose assessment and operation of heating ventilation and air conditioning systems. The information requested is provided below.

Dose Assessment

NRC Question 1

Although you use FGR-11 internal dose conversion factors in the dose calculations of design basis accidents (DBAs), you used Regulatory Guide (RG) 1.109 thyroid dose conversion factors in the calculation of iodine appearance rate for iodine spiking and in the revised definition of dose equivalent I-131 in the TS. Why did you not use the same dose conversion factors for both cases? Why is this formulation acceptable?

Dominion Response:

Technical Specification 3.4.16 limits North Anna reactor coolant system (RCS) specific activity for normal operations to 1 μCi/gm dose equivalent I-131. The proposed change to the definition of dose equivalent I-131 (TS 1.1, Definitions) allows dose equivalent iodine to be calculated using either TID-14844 or RG 1.109 dose conversion factors. This proposed change is consistent with the current Technical Specification definition for dose equivalent I-131 at Surry Power Station (previously approved by the NRC as part of the Surry power uprate in a letter dated August 3, 1995, TAC Nos. M90364 and M90365).

RG 1.183 requires that the preaccident and concurrent iodine spikes used in the design basis analysis be based on the maximum value permitted by Technical Specifications. The use of FGR-11 dose conversion factors to calculate dose is consistent with the Total Effective Dose Equivalent methodology described in RG 1.183.

The use of either the RG 1.109 or TID-14844 dose conversion factors to perform the Technical Specification surveillance for dose equivalent I-131 will restrict plant operations to a lower total allowable iodine inventory in the RCS than would be attainable using FGR-11 dose conversion factors. As is stated in the question, the 1 μCi/gm dose equivalent I-131 inventory calculated using RG 1.109 dose conversion factors was used to establish the design basis analysis source term for both the preaccident and concurrent iodine spikes. The use of the RG 1.109 dose conversion factors to determine the design basis analysis source term bounds the use of TID-14844 dose conversion factors. Therefore, use of the RG 1.109 dose conversion factors in the design basis analysis is consistent with the proposed change in the Technical Specification definition of dose equivalent I-131 and the requirement to use the maximum value permitted by Technical Specifications.

It is acceptable for the preaccident and concurrent iodine spike source terms to be based on RG 1.109 dose conversion factors and the doses to be calculated using FGR-11 dose conversion factors because the source term bounds allowable plant operating parameters as defined in the Technical Specifications.

NRC Question 2

How were the break flow rates calculated for the steam generator tube rupture? How were the steaming rates calculated?

Dominion Response:

Break Flow Rates

Break flow rates for the first 30 minutes after tube rupture were calculated using Dominion's RETRAN plant transient model. The break flow rates are calculated assuming the Extended Henry (subcooled) and Moody (saturated) critical flow correlations with a contraction coefficient of 1.0 to maximize the calculated break flow. Use of a smaller contraction coefficient could be justified based on available data. The break model also assumes a double-ended break of a single tube located near the top of the tube bundle and models frictional losses on both sides of the ruptured tube.

The fraction of SGTR break flow that flashes to steam is quite sensitive to the assumed break enthalpy. Therefore, this effect must be considered when determining the conditions to model. Two cases were considered for calculating the bounding break flow and subsequent steam flow rates: the first assumed forced primary system flow (offsite power available) and the second assumed primary system natural circulation (offsite power not available). The fraction of tube rupture break flow that flashes to steam (the "flashing fraction") is substantially different between the two cases. The effect of the difference in RCS flow rates between the two cases affects the break enthalpy and was accounted for in the calculation of the flashing fractions in the RETRAN plant transient model.

Steaming rates

Steaming rates for the first 30 minutes were calculated using Dominion's RETRAN plant transient model. Once the system transient predicted opening of the atmospheric steam dumps (the SG power operated relief valves [PORV's]), the PORV on the ruptured generator was assumed to fail open. The nominal opening setpoint was reduced for instrument errors to predict an early opening. Condenser steam dump valves were assumed to be unavailable, as this maximizes steam release to the atmosphere.

The RETRAN PORV model varies the steam flow as a function of calculated steam pressure. The PORV is modeled as a RETRAN break junction with break area and discharge coefficient chosen to yield the design valve flow rate at the set pressure (425,244 lbm/hr at a pressure of 1035 psig) using the isoenthalpic expansion flow model.

After 30 minutes, the stuck open PORV is assumed to be isolated. North Anna emergency procedures are structured to provide a prompt diagnosis of a failed open PORV, and contingency actions are provided for local isolation with a manual valve in the main steam valve house. Dominion evaluations have shown that these actions can be accomplished within 30 minutes.

Steam releases from the intact steam generators between 30 minutes and 8 hours are calculated in a separate energy balance (not in RETRAN). The calculation involves determining the amount of steam required to cool the reactor coolant system at a constant cooldown rate from its condition at 30 minutes (as calculated by RETRAN) to 350 °F (Residual Heat Removal System entry condition) at 8 hours. The energy balance accounts for:

- Decay heat
- Sensible heat in the reactor coolant
- Sensible heat in the secondary coolant (contents of the steam generators)
- Reactor coolant pump heat (case with offsite power available)
- Stored energy in the core
- Stored energy in the reactor coolant system metal

The relatively slow cooldown rate assumed (under 30 °F/hr) maximizes the steam release by integrating the decay heat over a long interval. In an actual event, cooldown rates would be expected to exceed this value, since current emergency procedures for post-SGTR cooldown specify a cooldown rate ≤100 °F/hr.

NRC Question 3

How were the steaming rates calculated for the main steamline break?

Dominion Response:

The steaming rates for the main steamline break are documented in Westinghouse report WCAP-11431, which is the basis for the existing analysis in North Anna UFSAR Section 6.2.1. WCAP-11431 evaluated the containment response to a main steamline rupture for a variety of break situations, including variations of possible plant system configurations, using the LOFTRAN computer code. This approach considered the following items:

- Nuclear kinetics characteristics and power generation in the reactor core,

- Stored energy in both the primary reactor system and the secondary steam plant,
- Main and auxiliary feedwater system operation,
- Safety systems operation (e.g., reactor trip, steamline isolation, ECCS),
- Break characteristics, and
- Blowdown characteristics.

WCAP-11431 contains 20 cases using 4 different power levels and five different break sizes. Due to flow limiters installed in the steam generator outlet nozzle, the maximum steamline break size at any location is limited to 1.4 ft^2. Case 2 in WCAP-11431 was the basis for the Dominion main steamline break dose analysis. This case assumed 102% power and a 1.4 ft^2 break size. The mass release rates for Case 2 in WCAP-11431 were integrated over three time periods: 0 to 10 seconds, 10 to 180 seconds and 180 to 1800 seconds to determine average flow rates for the three time periods. The three time periods represented the periods of relatively high, medium and low flow. The average flow rates were then used for the dose analysis, with the following modification. The average flow rate for the 10 to 180 seconds time period and the 180 to 1800 seconds time period were both scaled up to ensure that no significant radioactivity remained in the affected steam generator at the end of the 30 minute release period. This adjustment was made as a means to comply with the guidance in Regulatory Guide 1.183 (Appendix E, Item 5.5.1) which states that during periods of steam generator dryout all primary to secondary leakage in the affected steam generator should be released to the environment without mitigation.

NRC Question 4

How were the steaming rates calculated for the locked rotor accident?

Dominion Response:

Since the locked rotor accident (LRA) involves no break flow through a faulted or ruptured steam generator, all 3 steam generators are used for plant cooldown. Dominion calculations indicate that after approximately 1 minute all 3 steam generators have essentially equal steam flows. No new steaming rates were calculated for the locked rotor accident (LRA) dose analysis. The steaming rates for the steam generator tube rupture (SGTR) accident were also used for the locked rotor accident. This was based on a review of steam releases calculated for both events, which concluded that the SGTR releases bound either event.

NRC Question 5

For the fuel handling accident (FHA), the pool decontamination factor (DF) was modeled as a 99.8-percent efficient filter for elemental iodine. You state that this corresponds to an elemental iodine DF of 500. Does this also correspond to an overall effective iodine DF of 200?

Dominion Response:

RG 1.183 indicates that an elemental iodine decontamination factor (DF) of 500 is used with the organic DF of 1 to derive the effective DF and that the difference in DFs for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. The effective iodine DF that corresponds to these parameters is 286 and is derived as follows:

Total Iodine/$DF_{effective}$ = Organic Iodine/$DF_{organic}$ + Elemental Iodine/$DF_{elemental}$

Assuming that the total iodine is 1 and substituting in the elemental and organic DFs and the form released from the fuel we get the following:

$1/DF_{effective} = 0.0015/1 + 0.9985/500 = 0.003497$,
or
$DF_{effective} = 286$,

and the iodine above the water is composed of 57% elemental (1-0.0015/0.003497) and 43% organic (0.0015/0.003497) species.

NRC Question 6

To support revisions to TS 3.9.4, "Containment Penetrations," you assume no containment closure exists at the time of the FHA. You state in the submittal that closure of the containment after radiological release from dropped fuel may not occur based on the level of radioactivity in containment and the impact on personnel who would be required to close openings from inside the containment. The NRC staff has previously required licensees to provide for quick closure of the containment after an FHA with radioactivity release to contain the release and provide defense-in-depth protection of the public. Understanding that the principles of ALARA may be fulfilled by not requiring closure of containment, how does this compensate for the loss of the ability to contain a radioactivity release?

Dominion Response:

The radiological analysis of a fuel handling accident in containment did not credit containment closure. All available radioactivity is assumed to escape to the environment over a two-hour period. The doses from a fuel handling accident are less than those specified in 10 CFR 50.67 and Regulatory Guide 1.183 for the EAB, LPZ, and control room without closure of containment. In the design basis analysis of the fuel handling accident in containment, approximately 90% of the EAB and LPZ dose is realized in the first 30 minutes. During a fuel handling accident (assuming design basis conditions), workers attempting to close the equipment hatch within the first 30 minutes of the event might be exposed to levels of radioactivity that could exceed the emergency worker dose limit. However, realizing that a design basis fuel handling event bounds expected radiological conditions and acceptable conditions probably will exist in the containment, Dominion agrees with the concept of defense-in-depth and has committed in Dominion letter 03-464C, dated April 20, 2004 to establishing containment closure within 45 minutes following the decision to isolate containment.

NRC Question 7

The revisions to the requirements for the emergency core cooling system (ECCS) pump room exhaust air cleanup system (PREACS) operability in TS 5.5.2 are based on controlling the ECCS PREACS filtered leakage and ECCS PREACS unfiltered leakage based on the most recent evaluation of the control room unfiltered inleakage and maximizing the control room calculated dose. How does this assure that ECCS leakage is what is assumed in the DBA dose calculations?

Dominion Response:

Technical Specification (TS) 5.5.2 is a programmatic TS. As such, a new Technical Requirements Manual (TRM) Appendix A section is being developed as part of the TS Implementation Plan that corresponds to the new Specification 5.5.2.c. This new TRM Appendix A section requires that the ECCS leakage curve developed as part of the Alternative Source Term (AST) LOCA analysis [See Figure 3.1-1 from page 42 of Attachment 1 of the AST submittal (Serial No. 03-464) dated September 12, 2003] be incorporated as the acceptance criteria in station procedures. It should be noted that Figure 3.1-1 contains three lines that represent three different control room inleakage rates. For station procedures, a curve will be selected that corresponds to an unfiltered inleakage that is greater than the most recently measured control room inleakage. This limiting curve will be incorporated into station procedures and maintained as discussed in the response to Heating Ventilation & Air Conditioning question No. 3.

When TS 3.7.12, "ECCS PREACS" is not applicable, as defined by TS 5.5.2, all ECCS leakage will be considered unfiltered in station procedures. During periods when TS 3.7.12 is applicable, ECCS leakage in the safeguards building and charging pump cubicles in the auxiliary building is directly filtered by ECCS PREACS and will be considered filtered in station procedures. Finally, there are portions of the ECCS piping located in the quench spray basement and auxiliary building that are not filtered by ECCS PREACS. ECCS leakage in these areas is always considered unfiltered leakage in station procedures. Please refer to the discussion on pages 31 through 32 of Attachment 1 to the Virginia Electric and Power Company letter dated September 12, 2003 (Serial No. 03-464) for a more complete description of ECCS leakage and the ECCS PREACS response to a LOCA.

The procedures are also being revised to include guidance for entering into Action B of Tech Spec 3.7.10 and 3.7.13 since exceeding the Allowable ECCS leakage in Figure 3.1-1 could lead to exceeding the 5 rem TEDE control room dose limit, which would indicate an inoperable MCR/ESGR boundary. These procedure changes are consistent with the proposed revision to Bases 3.7.10 and 3.7.13 ACTIONS B.1 that indicate excessive control room inleakage or ECCS leakage are examples that could result in

Main Control Room / Emergency Switchgear Room boundary being declared inoperable.

NRC Question 8

In the loss-of-coolant accident, you assume control room isolation for the first hour, with up to an assumed 500 cfm unfiltered inleakage. Have you performed testing of your control room envelope to confirm this value? If not, please explain how you determined this value is bounding for your control room envelope.

Dominion Response:

Tracer gas testing was accomplished in September of 2003 to support the North Anna Units 1 and 2 AST license amendment request and response to the NRC's Generic Letter (GL) 2003-01. The results of the tracer gas testing to determine the control room unfiltered inleakage were documented in Dominion's response to question 1a of GL 2003-01 (letter Serial No. 03-373A, dated March 30, 2004). The control room unfiltered inleakage was measured to be 150 $\pm$ 3 cfm.

Heating Ventilation & Air Conditioning

NRC Question 1

With respect to the requested change to TS 3.7.10, this change may not meet the single-failure criterion. Discuss how the requested change is consistent with the single failure criterion and the function of providing circulating air to support equipment operability and human habitability.

Dominion Response:

The current dose consequence accident analyses assume that the control room is isolated and pressurized by bottle air in the event of a radiological accident and one train of the emergency ventilation system (EVS) provides filtered recirculation air flow in the Main Control Room / Emergency Switchgear Room (MCR/ESGR). After depletion of the bottled air (at least 60 minutes), a second train of EVS is used for pressurization. As a result, three trains of EVS are required to meet the single failure criterion. It should be noted that the cooling and recirculation of MCR/ESGR air for equipment operability and human habitability is performed by a separate, safety grade, redundant air conditioning system that is internal to the MCR/ESGR and is governed by TS 3.7.11.

Based on the Alternative Source Term dose consequence analyses, which did not assume filtered recirculation air flow during Modes 1 through 4, changes are proposed to LCO 3.7.10 since only one MCR/ESGR EVS train is required to provide pressurization. Specifically, the Loss of Coolant Accident analysis credits EVS pressurization after the first hour and isolation of the control room but does not credit recirculation of the air within the control room. The Main Steam Line Break, Steam

Generator Tube Rupture, and Locked Rotor Accident do not credit EVS recirculation, pressurization, or isolation to meet the dose limits. Therefore, in order to accommodate single failure 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 Fuel Handling Accident (FHA) are governed by TS 3.7.14. The FHA only credits one train of EVS for recirculation and only requires two trains to meet the single failure criterion.

NRC Question 2

If the requested change to TS 3.7.10 is not acceptable, then what is the justification for eliminating Surveillance Requirement (SR) 3.7.10.3?

Dominion Response:

The justification for eliminating SR 3.7.10.3 is based on the acceptability of the changes to TS 3.7.10 and the assumptions in the AST dose analyses for the control room. SR 3.7.10.3, which verifies automatic actuation of each MCR/ESGR EVS train, is no longer required because, as discussed in the response to Heating Ventilation & Air Conditioning question No. 1, fan operation is not credited in the first hour of the LOCA. One hour provides adequate time for operator action to manually start a train of EVS.

NRC Question 3

With respect to TS 3.7.12, sufficient justification has not been provided to support this change. This change will be based on a curve that will be developed from an evaluation that has not been conducted. With this request, there is sufficient uncertainty for the NRC staff to be concerned with the lack of reasonable assurance. What is the technical justification to support your request?

Dominion Response:

The results of the tracer gas testing to determine the control room unfiltered inleakage were documented in Dominion's response to question 1a of Generic Letter (GL) 2003-01 (letter Serial No. 03-373A) dated March 30, 2004. The control room unfiltered inleakage was measured to be 150 $\pm$ 3 cfm by tracer gas testing.

In the analyses done for the AST submittal, the control room dose for the LOCA was calculated at three different levels of unfiltered control room inleakage – 125 cfm, 250 cfm and 500 cfm. Please refer to Figure 3.1-1 from page 42 of Attachment 1 of the AST submittal (letter Serial No. 03-464) dated September 12, 2003. To establish the ECCS leakage limits associated with 125 cfm unfiltered leakage into the control room (Figure 3.1-1), the dose contribution from 125 cfm of unfiltered inleakage from both the containment and RWST releases and the shine from the control room filters were summed. This sum was subtracted from the control room dose limit of 5 REM TEDE to generate the remaining margin available to accommodate the ECCS leakage contribution toward the total control room dose. Next, the ECCS contribution to the control room dose was calculated based on the value of unfiltered ECCS leakage that generated a control room dose which consumed the remaining margin. This was repeated using the value of filtered ECCS leakage that generated a control room dose, which consumed the remaining margin. Finally, the ECCS contribution to the control room dose was calculated with the value of unfiltered ECCS leakage which consumed one half of the remaining margin and the value of filtered ECCS leakage which consumed one half of the remaining margin. These values of unfiltered and filtered ECCS leakage were plotted on a graph with filtered ECCS leakage on the abscissa and unfiltered ECCS leakage on the ordinate. The line through these data points showed the maximum allowable amounts of unfiltered and filtered ECCS leakage for a control room with a measured unfiltered inleakage of 125 cfm or less. Similar calculations were made for unfiltered control room inleakages of 250 cfm and 500 cfm and the results were plotted on the graph with the curve for 125 cfm.

The intent was to generate enough curves on the graph to make sure that the result of the control room unfiltered inleakage evaluation was bounded by one or more of the curves. Each curve is intended to represent a set of conditions, including ECCS leakage, MCR inleakage, RWST leakage, etc., that meets the control room dose limit (5 Rem TEDE). Now that the unfiltered control room inleakage had been determined to be 150 ± 3 cfm, the curve for 250 cfm of unfiltered control room inleakage will be used to limit the ECCS leakage. With an unfiltered control room inleakage of 150 ± 3 cfm, use of the 250 cfm curve to limit the ECCS leakage will be conservative. This is because the margin will be smaller (i.e., lower allowed filtered and unfiltered ECCS leakage) than the margin if the curve were based on 150 ± 3 cfm of unfiltered control room inleakage. If a future unfiltered control room inleakage evaluation results in a value sufficiently different from 150 ± 3 cfm, then a curve other than the 250 cfm curve may be used. For example, if a future evaluation shows only 90 cfm of unfiltered control room inleakage, then the 125 cfm curve for control room inleakage may be used as bounding in the analysis to determine the ECCS leakage.

Additionally, future changes to plant facilities or operational conditions may result in changes to one or more of the inputs used to generate the curves of permissible ECCS leakage. It is anticipated that new curves will be added or existing curves revised to address such changes in plant facilities or operating conditions, which affect the permissible ECCS leakage. These changes will employ the methodology described above, which was used in development of the curves in the AST submittal, and will be implemented under the provisions of 10 CFR 50.59.

NRC Question 4

With respect to TS 5.5.10, in accordance with the TS, North Anna should be in compliance with RG 1.52, Rev 2. As such, the requested change is not consistent with the RG. See RG 1.52, Table 2. What is the technical justification for the requested change?

Dominion Response:

The Pump Room Exhaust Air Cleanup System (PREACS) filters and Main Control Room/Emergency Switchgear Room (MCR/ESGR) emergency ventilation filters were originally designed and tested to comply with RG 1.52 Rev. 2 as described in the letter from Virginia Electric and Power Company to the NRC dated November 29, 1999, Serial No 99-339B. This letter was written in response to NRC Generic Letter (GL) 99-02, "Laboratory Testing Of Nuclear – Grade Activated Charcoal", June 3, 1999, in which it was recommended that ASTM D3803-89 be used for charcoal testing.

In the GL 99-02 response a Technical Specification change was requested to revise the methyl penetration acceptance criteria for the MCR/ESGR emergency ventilation charcoal filters and the PREACS charcoal filters when tested in accordance with ASTM D3803-89. These Technical Specification changes (amendments 224 and 205) were approved by the NRC in a letter dated November 20, 2000 (TAC NOS. MA7869 AND MA7870), and were as follows:

	GL 99-02 change to methyl penetration acceptance criteria	
	From less than or equal to	To less than or equal to
MCR/ESGR	1.0%	2.5%
PREACS	1.0%	5.0%

The basis for these changes for the control room ventilation charcoal was the current design basis accident analysis that considers the MCR/ESGR filter efficiency assumptions of 95% for elemental iodine and 95% for methyl iodide. Filter efficiency is defined as 100% - (2 x penetration). The relationship between penetration and filter efficiency is defined in ASTM D3803-89 and the factor of 2 is the safety factor discussed in GL 99-02. Therefore, applying a safety factor of 2 to the methyl iodide test penetration of less than or equal to 2.5% is consistent with the MCR/ESGR 95% methyl iodide filter efficiency. Similarly, the design basis analysis assumption for the PREACS 90% filter efficiency is consistent with a tested penetration of less than or equal to 5%.

The revised design basis analysis with Alternative Source Term (AST) assumed 95% efficiency for elemental iodine and 70% efficiency for methyl iodide for both the MCR/ESGR filters and the PREACS filters. The requested changes to TS 5.5.10 were based on these assumed filter efficiencies, a safety factor of 2, and the discussion on pages 47 through 49 of Attachment 1 to the Virginia Electric and Power Company letter dated September 12, 2003 (Serial No. 03-464). The requested changes to TS 5.5.10 based on the AST are summarized below.

	AST change to methyl penetration acceptance criteria	
	From less than or equal to	To less than or equal to
MCR/ESGR	2.5%	10%
PREACS	5.0%	15%

Additionally, the capability of the charcoal to remove elemental iodine is much greater than the ability to remove methyl iodide and testing with methyl iodide envelopes the removal efficiency requirement of the elemental iodine assumed in the analysis. As noted by the NCS Corporation letter of July 10, 2000 to Virginia Power in Attachment 2, "As a general rule you may expect penetration through nuclear grade activated carbon to increase 20 to 100 times when switching from elemental iodine to methyl iodide testing."

NRC Question 5

With respect to SR 3.7.13.4, sufficient technical justification has not been provided for eliminating make-up flow. Please provide your rationale and adequate technical justification.

Dominion Response:

The dose consequence accident analysis performed for the LOCA assumes that the Main Control Room / Emergency Switchgear Room (MCR/ESGR) is isolated with less than or equal to 500 cfm of unfiltered inleakage. For the first hour after the accident, pressurization by the Bottled Air System provides fresh air and minimizes unfiltered inleakage. The validation of the pressurization is performed by measurement of MCR/ESGR pressure using installed pressure gages in the control room and establishing differential pressures greater than or equal to 0.05 inches water gage relative to all adjacent areas, for at least 60 minutes. During the first hour of the accident bottled air makeup flow is the only source of pressurization for the MCR/ESGR, and this makeup flow is not being eliminated. However, the dose consequence accident analysis does not consider any specific amount of bottle air to be released into the control room to maintain MCR/ESGR at a positive pressure relative to adjacent areas; therefore, the requirement to measure makeup flow rate from the bottled air system is redundant to the pressurization surveillance and was requested to be eliminated from SR 3.7.13.4.

Attachment 2

NCS corporation letter of July 10, 2000 to Virginia Power

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company
(Dominion)

NCS CORPORATION
1385 West Goodale Boulevard Columbus. Ohio 43212
614-340-3700 • FAX 614-340-3707 • www.ncsc.com

July 10, 2000

Mr. Gene Henry
Virginia Power Company
Surry Power Station
5530 Hog Island Road
Surry, Virginia 23883

Dear Mr. Henry

Regarding: Methyl iodide versus elemental iodine penetration through carbon.

Nuclear grade activated carbon, when tested in accordance with ASTM D3803-1989 (methyl iodide at 30°C, 24.4 m/min, and 70% relative humidity) to a penetration of 15%, is more conservative than testing the same carbon in accordance with ASTM D3803-1979 (elemental iodine at 30°C, 24.4 m/min, and 95% relative humidity) to a penetration of 5%.

As a general rule, you may expect the radioiodine penetration through nuclear grade activated carbon to increase from 20 to 100 times when switching from elemental iodine to methyl iodide testing.

If you have questions regarding this matter, please contact me at 614-340-3700.

Sincerely,

John R. Pearson
Vice President
NCS Corporation