VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

August 18, 2004

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

Serial No. 04-494
NL&OS/ETS R0
Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY (Dominion)
NORTH ANNA POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATION CHANGES
IMPLEMENTATION OF ALTERNATE SOURCE TERM
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

In a letter dated September 12, 2003 (Serial No. 03-464) 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 August 3, 2004 telephone conference call, the NRC Staff requested additional information regarding the dose analysis methods/assumptions, charcoal filter testing efficiencies, and the proposed Technical Specifications changes. Information regarding the dose analysis, including some corrected analysis input data (i.e., pages 50 and 68 of the original dose assessment) is included in the attachment to this letter.

During the phone call the NRC provided their position on the assumptions to be used for evaluating the radiological consequences of a fuel handling accident consistent with RG 1.183. This position specifically addresses the use of effective decontamination factors (DFs) for fuel handling accidents. Based on the NRC position on the use of effective DFs instead of the DFs for elemental and organic species, additional dose assessment will be required to support the requested Technical Specification changes. Therefore, the remainder of the requested information will be provided in a subsequent letter.

Due to the number of program and procedure changes necessary to implement these changes, we continue to request ninety days from the issuance date of the amendments to implement the Technical Specifications changes. If you have any further questions or require additional information, please contact Mr. Thomas Shaub at (804) 273-2763.

Very truly yours,

Eugene S. Grecheck
Vice President – Nuclear Support Services

A001

Attachment
1. Request for Additional Information and Revised Analysis Input Data

Commitments made in this letter: None

cc: U.S. Nuclear Regulatory Commission
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SN: 04-494
Docket Nos.: 50-338/339
Subject: AST RAI - Dose Analysis

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Eugene S. Grecheck who is Vice President – Nuclear Support Services of Virginia Electric and Power Company. He has affirmed before me that he 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 18th day of August, 2004.

My Commission Expires: 3/31/08

Maggie McClure
Notary Public

(SEAL)

Attachment

Proposed Technical Specification Changes
Implementation of Alternate Source Term

Response to Request for Additional Information
and
Revised Analysis Input Data pages 58 and 68
From September 12, 2003 letter (Serial No. 03-464)

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

NRC Question:

Provide the basis for reducing the ground level X/Q value calculated with ARCON96 by a factor of 5 at the end of the release phase for the steam generator tube rupture.

Dominion Response:

The North Anna safety analysis of a steam generator tube rupture (SGTR) accident models large quantities of radioactive steam flowing out of the steam generator power operated relief valves (PORVs). The radiological impact on the control room of this large steam release is determined with atmospheric dispersion factors.

From Regulatory Guide 1.194 Section 6, "Plume Rise", the excerpted paragraph below discusses the reduction of atmospheric dispersion factors (X/Q's) for releases from steam relief valves and atmospheric dump valves. The North Anna steam generator PORVs are examples of atmospheric dump valves.

"In lieu of mechanistically addressing the amount of buoyant plume rise associated with energetic releases from steam relief valves or atmospheric dump valves, the ground level X/Q value calculated with ARCON96 (on the basis of the physical height of the release point) may be reduced by a factor of 5. This reduction may be taken only if 1) the release point is uncapped and vertically oriented and 2) the time-dependent vertical velocity exceeds the 95th-percentile wind speed (at the release point height) by a factor of 5."

As was discussed in our letter dated November 20, 2003 (Serial No. 03-464A), the North Anna steam generator PORVs are uncapped and vertically oriented. The 95^{th}-percentile wind speed at the PORV release height is 5.72 meters per second. Five times this value for wind speed is 28.6 meters per second. Therefore, if the vertical velocity of the steam exiting the steam generator PORVs after a SGTR accident is greater than 28.6 meters per second, the atmospheric dispersion factors used to model the PORVs can be reduced by a factor of five. In the letter dated November 20, 2003 (Serial No. 03-464A) Dominion provided the vertical steam velocities for the steam generator PORVs. After isolation of the affected steam generator, the vertical steam velocity of the unaffected steam generator PORVs was determined based on a conservatively low average steam release velocity instead of a minimum velocity. However, to apply this factor of five reduction for the 8 hour release duration, the vertical velocity of the exiting steam must be greater than 28.6 meters per second at all times, even at the end of the accident when steam velocities are at their lowest.

During the first 30 minutes of a SGTR accident, initial plant cooldown, reactor coolant system (RCS) depressurization and secondary side isolation are expected to occur and are modeled in the safety analysis. Subsequent to the first 30 minutes of the SGTR accident, the emergency procedures direct the operators to proceed with plant cooldown. The cooldown instructions limit the cooldown rate to <100 °F/hour. Therefore, residual heat removal (RHR) initiation conditions (350 °F) would be expected to be reached in 2-3 hours.

To conservatively model both integrated steam releases and low steam PORV exit velocity for X/Q considerations, a slow cooldown is modeled with 350 °F being reached in 8 hours. At 8 hours, the rate of steam release required to remove decay heat plus reactor coolant pump heat (no LOOP case) is approximately 32 lbm/sec. For the LOOP case (no reactor coolant pump heat) the steam release requirement is reduced to about 21.5 lbm/sec.

For RCS conditions at 350 °F, the secondary steam pressure will be about 135 psia. At this pressure, it is estimated that a single steam generator PORV (atmospheric dump valve) can pass somewhere between 13-17 lbm/sec, depending on the model used. This means that both steam generator PORVs will be needed at the end of the cooldown. Assuming an equal distribution, at 8 hours each PORV will be passing around 10-11 lbm/sec for the LOOP case.

For a 10" discharge pipe and assuming isoenthalpic expansion of 10 lbm/sec from saturated steam at 135 psi to superheated steam at atmospheric conditions, the steam exit velocity is estimated at about 560 ft/sec or 170 meters/sec. 170 meters per second is well above the required threshold of 28.6 meters per second to qualify for the factor of five reduction in the PORV atmospheric dispersion factors.

For a locked rotor accident (LRA), the heat in the reactor core which has to be removed through cooldown with the steam generator PORVs is the same as the heat released during a SGTR accident. The primary difference is that there are 3 steam generator PORVs available for cooldown during a LRA instead of the 2 flowing steam during a SGTR accident. This means that the PORV steam flow during a LRA cooldown will be 2/3 of the steam flow during a SGTR accident cooldown. Because the vertical velocity of the SGTR PORV steam flow is so high, even 2/3 of this will be far higher than the threshold required to justify a reduction of the atmospheric dispersion factors by a factor of 5. Therefore, the PORV atmospheric dispersion factors used to model the control room dose during a LRA can also be reduced by a factor of 5.

Conclusion

At the end of the steam generator tube rupture accident, the vertical velocity of the steam exiting the unaffected steam generator PORVs exceeds the threshold required to qualify for the factor of five reduction in the PORV atmospheric dispersion factors. Additionally, at the end of a locked rotor accident, the vertical velocity of the steam exhausting to atmosphere via the three steam generator PORVs also exceeds the 28.6 meters/sec requirement to qualify for the factor of five reduction in PORV atmospheric dispersion factors.

Revised Input Data Pages 58 and 68
From Letter Dated September 12, 2003 (Serial No. 03-464)

Table 3.3-4: Analysis Assumptions and Key Parameter Values Employed in the SGTR Analysis

Primary and Secondary Side Parameters

Primary system volume (cubic feet)	9786
SG Steam volume (cubic feet/SG)	3838
SG Liquid volume (cubic feet/SG)	2054
SG Liquid mass (gm/SG)	4.43E+07
Control room volume (cubic feet)	2.30E+05
Primary System Mass (lb or gm)	4.37845E+05 lb or 1.986E+08 gm.
Secondary System Mass (lb or gm per generator)	7200 lb/SG or 3.266E+06 gm/SG
Steam Mass Dilution	2.81E+05

Full Power Properties	Steam Generator	RCS Coolant Liquid
Temperature (degrees F)	525.24	580.8
Pressure (psia)	850	2250
Density (gm/cc)	0.76096	0.71669

SGTR Flow Rates (all flow rates are in cubic feet per minute)

LOOP - Affected SG

Time	RCS to SG Liquid	RCS to SG Steam	SG Liquid to Steam	SG Steam to Environment
0 - 103 sec	9.096E+01	1.164E+01	1.494E+03	0.000E+00
103 - 232 sec	7.659E+01	3.470E+00	1.972E+02	4.994E+03
232 -1800 sec	7.878E+01	4.860E+00	1.308E+02	3.313E+03

LOOP - Unaffected SG

Time	RCS to SG Liquid	SG Liquid to Environment
0 - 103 sec	0.1337	0
103 - 232 sec	0.1337	356.09
232 -1800 sec	0.1337	100.81
1800 - 2 hrs	0.1337	53.80
2 hrs - 8 hrs	0.1337	38.71

NO-LOOP - Affected SG

Time	RCS to SG Liquid	RCS to SG Steam	SG Liquid to Steam	SG Steam to Environment
0 - 107 sec	8.021E+01	1.838E+01	1.494E+03	0.000E+00
107 - 196 sec	8.505E+01	3.190E+00	2.726E+02	6.904E+03
196 - 1800 sec	8.228E+01	2.460E+00	1.521E+02	3.850E+03

NO-LOOP - Unaffected SG

Time	RCS to SG Liquid	SG Liquid to Environment
0 - 107 sec	0.1337	0
107- 196 sec	0.1337	484.85
196 -1800 sec	0.1337	184.30
1800 - 2 hrs	0.1337	65.49
2 hrs - 8 hrs	0.1337	50.35

Indicates revisions from original submittal

Table 3.5-2: Analysis Assumptions & Key Parameter Values Employed In the Locked Rotor Analysis

NSSS Parameters		
Core Power		2958 MWt
Number of Fuel Assemblies		157
Primary System (RCS) Volume		9786 ft^3
Steam Generator Liquid Volume		6162 ft^3
Steam Generator Steam Volume		11,514 ft^3
Radial Peaking Factor		1.65
Fuel Failure During Event		13%
RG-1.183 Non-LOCA Gap Fractions		See Table 3.5-1
Main Control Room (MCR) Parameters		
Free Volume		2.30E5 ft^3
Normal Intake and Exhaust Flow Rate		3500 CFM
No Isolation of the Control Room		
Onsite Atmospheric Dispersion Factors		
Main Control Room Normal Intake		
0 – 2 hours		2.08E-3 sec/m^3
2 – 8 hours		1.64E-3 sec/m^3
8 – 24 hours		6.46E-4 sec/m^3
24 – 96 hours		4.50E-4 sec/m^3
96 – 720 hours		3.36E-4 sec/m^3
Leak Rates		
Description		**Flow rate (cfm)**
Primary to Secondary Leakage		0.1337
SG liquid to steam	0 hr	441.6
	0.25 hr	249.9
	0.5 hr	65.5
	1 hr	---
	2 hr	50.3
	8 hr	0
SG steam to Environment	0 hr	11183
	0.25 hr	6328
	0.5 hr	1658
	1 hr	---
	2 hr	1275
	8 hr	0

| Indicates revisions from original submittal