VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

December 10, 2001

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Serial No.: 01-658
LR/DWL R0
Docket Nos.: 50-280/281
50-338/339
License Nos.: DPR-32/37
NPF-4/7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
SURRY AND NORTH ANNA POWER STATIONS UNITS 1 AND 2
REQUEST FOR ADDITIONAL INFORMATION
LICENSE RENEWAL APPLICATIONS

In an October 17, 2001 letter, the NRC requested additional information regarding the license renewal applications (LRAs) for Surry and North Anna Power Stations. The attachment to this letter contains the responses to the Requests for Additional Information (RAIs) associated with the severe accident mitigation alternatives (SAMA) analysis provided in the LRAs.

Should you have any questions regarding this submittal, please contact Mr. J. E. Wroniewicz at (804) 273-2186.

Very truly yours,

David A. Christian
Senior Vice President – Nuclear Operations and Chief Nuclear Officer

Attachment

Commitments made in this letter: None

A001

cc:

U. S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW
Suite 23T85
Atlanta, GA 30303-8931

Mr. M. J. Morgan
NRC Senior Resident Inspector
North Anna Power Station

Mr. R. A. Musser
NRC Senior Resident Inspector
Surry Power Station

Mr. J. E. Reasor, Jr.
Old Dominion Electric Cooperative
Innsbrook Corporate Center
4201 Dominion Blvd.
Suite 300
Glen Allen, VA 23060

Ms. Ellie Irons, EIR Program Manager
Virginia Dept. of Environmental Quality
629 East Main St., 6th Fl
Richmond, VA 23219

Mr. David Paylor, Program Coordinator
Virginia Dept. of Environmental Quality
P.O. Box 10009
Richmond, VA 23240-0009

Mr. Joe Hassell, Environmental Manager
Virginia Dept. of Environmental Quality
Water Division
P.O. Box 10009
Richmond, VA 23240-0009

Mr. Frank Daniel, Regional Director
Virginia Dept. of Environmental Quality
Tidewater Regional Office
5636 Southern Blvd.
Virginia Beach, VA 23462

Mr. Gregory Clayton, Regional Director
Virginia Dept. of Environmental Quality
Northern Virginia Regional Office
13901 Crown Ct.
Woodbridge, VA 22193

Mr. Frank Fulgham, Program Manager
Virginia Dept. of Agriculture & Consumer Services
Office of Plant & Pest Services
1100 Bank St.
Richmond, VA 23219

Mr. David Brickley, Agency Director
Virginia Dept. of Conservation & Recreation
203 Governor St.
Richmond, VA 23219

Mr. William Woodfin, Director
Virginia Dept. of Game & Inland Fisheries
4010 West Broad St.
Richmond, VA 23230

Mr. Robert Hicks, Director
Virginia Dept. of Health
Office of Environmental Health Services
1500 East Main St., Room 115
Richmond, VA 23219

Ms. Kathleen S. Kilpatrick, Director
Virginia Dept. of Historic Resources
State Historic Preservation Office
2801 Kensington Ave.
Richmond, VA 23221

Dr. Ethel Eaton, Archeologist Senior
Virginia Dept. of Historic Resources
State Historic Preservation Office
2801 Kensington Ave.
Richmond, VA 23221

Mr. Robert W. Grabb, Assistant Commissioner
Virginia Marine Resources Commission
2600 Washington Ave.
Newport News, VA 23607

Dr. John Olney, Associate Professor
Virginia Institute of Marine Science
School of Marine Science
Gloucester Point, VA 23062

Mr. John Simkins
Virginia Dept. of Transportation
Environmental Division
1401 East Broad St.
Richmond, VA 23219

Mr. Robert Burnley
Virginia Economic Development Partnership
901 East Byrd St.
Richmond, VA 23219

Mr. William F. Stephens, Director
Virginia State Corporation Commission
Division of Energy Regulation
1300 East Main St., 4th Fl., Tyler Bldg.
Richmond, VA 23219

Mr. Michael Cline, State Coordinator
Commonwealth of Virginia
Department of Emergency Management
10501 Trade Rd.
Richmond, VA 23236-3713

Mr. Terry Lewis, County Administrator
P.O. Box 65
Surry, VA 23883

Mr. Lee Lintecum
Louisa County Administrator
P.O. Box 160
Louisa, VA 23093

Mr. Douglas C. Walker
Acting Spotsylvania County Administrator
P.O. Box 99
Spotsylvania, VA 22553

Ms. Brenda G. Bailey, County Administrator
P.O. Box 11
Orange, VA 22960

Chairman Reeva Tilley
Virginia Council on Indians
P.O. Box 1475
Richmond, VA 23218

Mr. Don Lillywhite, Director
Economics Information Services
Virginia Employment Commission
State Data Center
703 East Main St., Room 213
Richmond, VA 23219

Mr. Alan Zoellner
Government Information Department
Swem Library
College of William and Mary
Landrum Dr.
P.O. Box 8794
Williamsburg, VA 23187-8794

Mr. Walter Newsome
Government Information Resources
Alderman Library
University of Virginia
160 McCormick Rd.
P.O. Box 400154
Charlottesville, VA 22904-4154

SN: 01-658
Docket Nos.: 50-280/281
50-338/339

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing responses to requests for additional information supporting the applications for renewal of the operating licenses for Surry, Units 1 and 2, and North Anna, Units 1 and 2, were acknowledged before me, in and for the County and Commonwealth aforesaid, today by David A. Christian who is Senior Vice President - Nuclear Operations and Chief Nuclear Officer of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing documents in behalf of that Company, and that the statements in the documents are true to the best of his knowledge and belief.

Acknowledged before me this 10th day of December, 2001.

My Commission Expires: 3-31-04.

Maggie McClure
Notary Public

(SEAL)

Attachment

License Renewal – Response to SAMA RAIs
Serial No. 01-658

Response to Request for Additional Information

Dated October 17, 2001

Surry and North Anna Power Stations, Units 1 and 2

License Renewal Application Environmental Reports

Appendix G, Severe Accident Mitigation Alternatives Analysis

Virginia Electric and Power Company
(Dominion)

Questions Related to Surry Power Station (SPS)

SPS RAI No. 1:

The license renewal SAMA analysis is based on an internal events core damage frequency (CDF) of 3.8 x 10^{-5} per reactor-year, which is about 50 percent of the value reported in the Individual Plant Evaluation (IPE, i.e., 7.4 x 10^{-5} per reactor-year). Also, you indicate on page G-18 of the submittal that the level 2 Probabilistic Risk Assessment (PRA) for SPS was updated for the purpose of the Severe Accident Mitigation Alternatives (SAMA) evaluation. In this regard please provide the following:

a. a discussion of the reasons for the reduction in CDF, including a description of the major changes in PRA models/assumptions and plant hardware/procedures, and their respective impacts on CDF,

b. a breakdown of the internal events CDF (3.8 x 10^{-5} per reactor-year) by initiating event, specifically, transients, Loss-of-coolant accident (LOCA), Steam generator tube rupture (SGTR), and Interfacing system LOCA (ISLOCA),

c. a description of the changes to the level 2 analysis and their impact on results,

d. the updated conditional probabilities associated with each release category for each plant damage state (PDS) (i.e., the Containment Matrix), the list of updated plant damage state definitions if different from IPE, and the updated PDS frequencies and release class frequencies, and

e. a description of the internal and external peer reviews of the PRA used in the SAMA analysis.

Dominion Response:

Response to 1a: Three major model revisions occurred between the time the IPE was submitted and the preparation of the license renewal SAMA analyses. The first was an update prior to conducting the IPEEE fire analysis. The second was an update in 1997. The third update occurred in 1998. Each of the updates incorporated significant plant modifications, corrected model errors or enhanced the model with state-of-the-art improvements as appropriate. Some model updates were also made to support implementation of the maintenance rule. A summary of the changes made is listed below:

- Revised the AFW fault tree to incorporate full flow test capability per design change.

- Incorporated design change for additional ESGR chillers located in MER-5.
- Modified HEP to account for reduced time to hot leg recirculation in LLOCA.
- The swing diesel model was modified to allow the possibility of being on either unit
- Provided additional functions for mitigating the SGTR.
- Modified the modeling of the Loss of ESGR cooling initiating event.
- Modified the modeling of the Loss of 4160 V emergency bus.
- Added a fault tree model for the Station Blackout Diesel.
- Added a fault tree model for AMSAC
- Added a fault tree model of the Bearing Cooling, Service Air, Instrument Air, Auxiliary Building Ventilation and Fire Protection Systems that provide support functions for balance of plant components or backup mitigating functions.
- Revised the CC fault trees to include providing cooling to the reactor coolant pumps and the instrument air compressors
- Added fault tree models for the station service busses and switchyard busses.

The individual contribution of each of the changes to the PRA is not available, but collectively they account for the reduction of CDF to 3.8E-5/yr. However, a breakdown of the change in core damage frequency for each initiator is available and is shown in the table below.

Model Comparison of CDF For Each Accident Group							
Accident Group	**Accident Type**	**S7B With TM-0**	**S7B With TM-1**	**S7B With 3-Yr**	**S97-04C With TM-0**	**S97-04C With TM-1**	**91 Aug (IPE)**
LOCA Transients	Small LOCA, S2	9.68E-6	1.39E-5	1.09E-5	1.29E-5	1.36E-5	1.14E-5
	Medium LOCA, S1	5.12E-6	5.86E-6	5.97E-6	5.09E-6	5.84E-6	5.30E-6
	Large LOCA, A	4.58E-6	4.94E-6	4.98E-6	4.60E-6	4.94E-6	4.57E-6
	SGTR, T7	1.65E-6	2.62E-6	2.26E-6	9.30E-6	9.96E-6	1.04E-5
	Inter System LOCA, Vx	1.62E-6	1.62E-6	1.62E-6	1.62E-6	1.62E-6	1.62E-6
	RCP Seal LOCA, T4	7.74E-7	7.74E-7	7.74E-7	7.77E-7	7.77E-7	7.74E-7
	Reactor Vessel Rupture, Rx	2.66E-7	2.66E-7	2.66E-7	2.67E-7	2.67E-7	2.66E-7
	Subtotal	2.37E-5	3.00E-5	2.68E-5	3.46E-5	3.70E-5	3.43E-5
Electrical Transients	Loss of 4160V AC Emerg. Bus 1H, T9A	1.72E-6	3.57E-6	3.48E-6	N/A	N/A	N/A
	Loss of 4160V AC Emerg. Bus 1J, T9B	1.09E-6	5.50E-6	1.93E-6	N/A	N/A	N/A
	Loss of DC Bus 1A , T5A	7.23E-7	2.81E-6	9.41E-7	6.86E-7	1.23E-6	6.84E-7
	Loss of DC Bus 1B, T5B	7.21E-7	2.80E-6	9.39E-7	6.85E-7	1.24E-6	6.84E-7
	Loss of Offsite Power, T1	7.22E-7	1.41E-6	2.18E-6	4.23E-6	8.51E-6	7.11E-6
	Station Blackout, T1A	5.22E-8	5.23E-7	3.30E-7	2.56E-6	7.13E-6	8.09E-6
	Sub-total	5.03E-6	1.66E-5	9.80E-6	8.85E-6	1.81E-5	1.66E-5
General Transients	Loss of SW, T6	8.41E-7	1.14E-6	8.53E-7	1.30E-6	1.51E-6	1.29E-6
	Loss of FW, T2/T2A/T3	2.41E-7	6.68E-7	4.07E-7	2.39E-6	2.47E-6	3.23E-6
	ATWS, TH/TL	4.00E-9	6.97E-9	4.50E-9	3.29E-7	3.40E-7	3.17E-7
	Loss of ESGR HVAC, T8A/T8B	8.78E-9	8.78E-9	8.78E-9	1.63E-5	1.63E-5	1.81E-5
	Steam Line Break, Ts1/Ts2	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	3.23E-11
	Sub-total	1.10E-6	1.82E-6	1.27E-6	2.03E-5	2.06E-5	2.29E-5
Grand Total		2.99E-5	4.84E-5	3.79E-5	6.31E-5	7.58E-5	7.38E-5

Response to 1b: The CDF for Surry grouped by sequence and by category is in the Table below. It should be noted that in this table the Initiating Event groupings have been provided in slightly more detail than the table above. For example the T1A trees with conditional failures are shown individually in the table below while they are combined into a single T1A contribution in the response to question 1a.

Init Event	Description	CDF	% of Total CDF
S2	Small LOCA	1.10E-05	29.00%
S1	Medium LOCA	5.97E-06	15.80%
A	Large LOCA	4.98E-06	13.20%
T9A	Loss of 4KV bus 1H	3.48E-06	9.20%
T7	SGTR	2.26E-06	6.00%
T1	Loss of Offsite Power	2.12E-06	5.60%
T9B	Loss of 4KV bus 1J	1.93E-06	5.10%
VX	ISLOCA	1.62E-06	4.30%
T5A	Loss of 125V DC bus 1A	9.24E-07	2.40%
T5B	Loss of 125V DC bus 1B	9.21E-07	2.40%
T6	Loss of Circ Water	8.53E-07	2.30%
T4	RCP Seal LOCA	7.74E-07	2.00%
RX	RV Rupture	2.66E-07	0.70%
T1AQ	SBO with stuck open PORV	2.11E-07	0.60%
T3	Transient with MFW available	1.68E-07	0.40%
T2A	Loss MFW, FW recoverable	1.26E-07	0.30%
T2	Loss of MFW, FW not recoverable	1.13E-07	0.30%
T1ALt	SBO with AFW failure	6.90E-08	0.20%
T1Q	LOOP with stuck open PORV	6.72E-08	0.20%
T1A	SBO	2.23E-08	0.10%
T8B	Loss of ESGR chillers	6.80E-09	0.00%
TH	ATWS from high power	3.98E-09	0.00%
T1ASI	SBO with seal LOCA	2.18E-09	0.00%
T8A	Loss of ESGR HVAC	1.98E-09	0.00%
TL	ATWS from low power	5.19E-10	0.00%
Ts2	MSLB outside containment	0.00E+01	0.00%
Ts1	MSLB inside containment	0.00E+01	0.00%
TOTAL		**3.79E-05**	**100%**
LOCAs		2.20E-05	58.0%
Transients		9.30E-06	24.5%
LOOP/SBO		2.49E-06	6.6%
SGTR		2.26E-06	6.0%
ISLOCA		1.62E-06	4.3%
RV Rupture		2.66E-07	0.7%
ATWS		4.50E-09	0.0%
Total		**3.79E-5**	**100%**

Response to 1c: For Surry and North Anna the level 2 model consists of several logic trees. These trees are used to evaluate accident progression for each plant damage state as well as for appropriate grouping of the accident sequences into source term categories. Accident progression is evaluated using Containment Event Trees. These trees are quantified using probabilities that are assigned using rules. The source term grouping diagram is a tree that combines accident sequences based on similarity of source term. The source terms used in the IPE were obtained from MAAP3B and were not updated for this license renewal work.

A small number of changes were made to the level 2 model prior to performing the SAMA analysis. First, the Surry and North Anna level 2 models were made consistent. So, both models determine Source Term Categories using the same grouping criteria. This change reflects the status of the level 2 modeling at the time the IPE was completed. The North Anna work structure was an improvement to the work done for Surry. Since the level 2 models were converted to LERF models shortly after the IPE/IPEEE process was completed there was no incentive to change the level 2 models until a need was identified. Hence, prior to beginning the SAMA analysis a unified Source Term Category grouping was implemented which essentially was to use the approach presented in the North Anna IPE.

The general containment event tree (CET) was also slightly modified to reflect recent experimental results in severe accident analysis research. In particular the probabilities of the following events were modified:

- The conditional probability of early containment failure from Direct Containment Heating or Hydrogen combustion is set to zero.
- Changed the probabilities of vessel depressurization before vessel rupture slightly.
- Set the Alpha failure mode (steam explosion) to zero.

The net impact of these changes is small because the magnitude of the changes is small. The Alpha mode failure probability is set to zero but it was already no larger than .008. Similarly, the branch fractions for mode of induced primary system failure changed such that the branch fraction of hot leg failures increased from 0.72 to 0.88 and the branch fraction for No RCS failure decreased from .26 to 0.1. The direction of the change is a shift of the plant damage state frequency from source term categories that would be considered LERF to those that would not be considered LERF either because there was no containment failure or if the containment did fail it would be late.

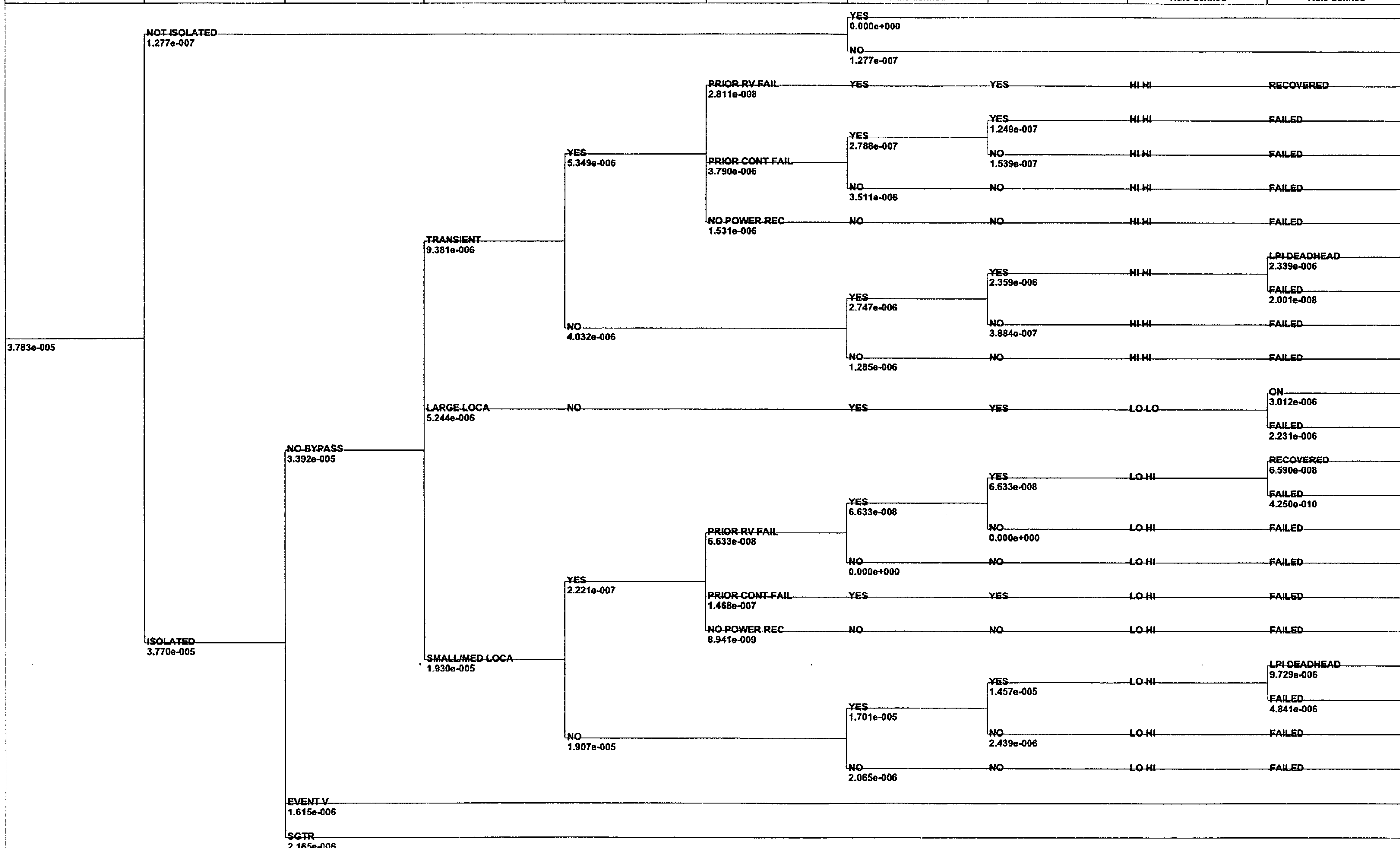
Response to 1d: The PDS definitions are provided in the figure on the following page, which displays how the level 1 sequences are separated into the 25 PDSs.

The PDS frequencies are provided in the following table:

Baseline PDS Frequencies

PDS:	Base Freq
1	0.00E+00
2	1.28E-07
3	2.81E-08
4	1.25E-07
5	1.54E-07
6	3.51E-06
7	1.53E-06
8	2.34E-06
9	2.00E-08
10	3.88E-07
11	1.29E-06
12	3.01E-06
13	2.23E-06
14	6.59E-08
15	4.25E-10
16	0.00E+00
17	0.00E+00
18	1.47E-07
19	8.94E-09
20	9.73E-06
21	4.84E-06
22	2.44E-06
23	2.07E-06
24	1.62E-06
25	2.17E-06
Total	**3.78E-05**

CRITERIA>	CONISOLAT	CONBYPASS	TRANLOCA	SBO	POWRECOV	RECSPRAYS	CNHEATREM	RCSPRESS	INVESSINJ
ENTRY FROM LEVEL 1 PDS TREE	CONTAINMENT ISOLATION STATUS	CONTAINMENT BYPASS	TRANSIENT OR LOCA TYPE	STATION BLACKOUT	POWER RECOVERY PRIOR RV FAILURE PRIOR CONT FAIL NO POWER RECOVERY	CONTAINMENT RECIRCULATION SPRAYS	CONTAINMENT HEAT REMOVAL	RCS PRESSURE DURING CORE DAMAGE/AT VESSEL FAILURE	STATUS OF INVESSEL INJECT ON.LPI DEADHEAD RECOVERED,FAILED
No Rule defined	Rule defined	Rule defined	Rule defined	Rule defined	Rule defined	Rule defined	Rule defined	Rule defined	Rule defined



PDS #	Frequency from sps
1	0
2	1.277e-007
3	2.811e-008
4	1.249e-007
5	1.539e-007
6	3.511e-006
7	1.531e-006
8	2.339e-006
9	2.001e-008
10	3.884e-007
11	1.285e-006
12	3.012e-006
13	2.231e-006
14	6.590e-008
15	4.250e-010
16	0
17	0
18	1.468e-007
19	8.941e-009
20	9.729e-006
21	4.841e-006
22	2.439e-006
23	2.065e-006
24	1.615e-006
25	2.165e-006

VIRGINIA ELECTRIC POWER COMPANY
SURRY POWER STATION PRA

PLANT DAMAGE STATE GROUPING LOGIC
REV. 0

C:\Sama\sps.PDD
Last Saved: Thursday, November 08, 2001
Last Updated: Thursday, January 01, 1970
WinNUCAP 1.0 Licensed to: VEPCO

The conditional probabilities associated with each release category for each plant damage state are provided in the following table:

Containment Event Tree Endstate Conditional Probabilities For Each PDS

	PDS												
STC	1	2	3	4	5	6	7	8	9	10	11	12	13
1	0.00E+00	0.00E+00	2.78E-01	7.74E-01	0.00E+00	0.00E+00	0.00E+00	1.82E-01	9.28E-01	0.00E+00	0.00E+00	4.70E-02	9.40E-01
2	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
4	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
5	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
6	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
7	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
8	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
9	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.76E-01	0.00E+00	0.00E+00	0.00E+00
10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.68E-02	0.00E+00	0.00E+00	0.00E+00
11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.83E-02	0.00E+00	0.00E+00	0.00E+00
12	0.00E+00	0.00E+00	0.00E+00	2.65E-02	5.57E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.20E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
14	0.00E+00	0.00E+00	1.85E-04	0.00E+00	5.79E-02	6.05E-01	6.05E-01	6.16E-04	6.16E-04	6.16E-04	6.05E-01	0.00E+00	0.00E+00
15	0.00E+00	0.00E+00	1.15E-04	0.00E+00	4.71E-02	3.77E-01	3.77E-01	3.84E-04	3.84E-04	3.84E-04	3.77E-01	0.00E+00	0.00E+00
16	0.00E+00	0.00E+00	1.59E-02	1.81E-01	0.00E+00	0.00E+00	0.00E+00	5.52E-03	5.32E-02	0.00E+00	0.00E+00	3.00E-03	6.00E-02
17	7.50E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
18	0.00E+00	1.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
20	0.00E+00	0.00E+00	6.87E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.94E-01	0.00E+00	0.00E+00	0.00E+00	9.50E-01	0.00E+00
21	2.50E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
24	0.00E+00	0.00E+00	1.80E-02	1.80E-02	1.80E-02	1.80E-02	1.80E-02	1.80E-02	1.80E-02	1.80E-02	1.80E-02	0.00E+00	0.00E+00
SUM	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00

Containment Event Tree Endstate Conditional Probabilities For Each PDS (continued)

	PDS											
STC	14	15	16	17	18	19	20	21	22	23	24	25
1	2.96E-01	9.88E-01	0.00E+00	0.00E+00	9.00E-01	0.00E+00	9.88E-01	9.88E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
4	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
5	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
6	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
7	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
8	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
9	0.00E+00	0.00E+00	8.84E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.84E-01	0.00E+00	0.00E+00	0.00E+00
10	0.00E+00	0.00E+00	5.74E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.74E-02	0.00E+00	0.00E+00	0.00E+00
11	0.00E+00	0.00E+00	4.88E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.88E-02	0.00E+00	0.00E+00	0.00E+00
12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
14	1.85E-03	6.16E-03	6.16E-03	6.16E-01	0.00E+00	6.16E-01	6.16E-03	6.16E-03	6.16E-03	6.16E-01	0.00E+00	0.00E+00
15	1.15E-03	3.84E-03	3.84E-03	3.84E-01	0.00E+00	3.84E-01	3.84E-03	3.84E-03	3.84E-03	3.84E-01	0.00E+00	0.00E+00
16	6.83E-04	2.28E-03	0.00E+00	0.00E+00	1.00E-01	0.00E+00	2.28E-03	2.28E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
20	7.00E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.50E-01	0.00E+00
23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.50E-01	0.00E+00
24	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.00E+00
SUM	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00

The release category frequencies are provided in the following table:

Source Term Category	Frequency
1	1.73E-05
2	0.00E+00
3	0.00E+00
4	0.00E+00
5	0.00E+00
6	0.00E+00
7	0.00E+00
8	0.00E+00
9	2.50E-06
10	1.62E-07
11	1.38E-07
12	8.91E-08
13	4.92E-08
14	5.22E-06
15	3.26E-06
16	2.28E-07
17	0.00E+00
18	1.28E-07
19	0.00E+00
20	4.78E-06
21	0.00E+00
22	1.37E-06
23	2.42E-07
24	2.33E-06

The Source Term Category frequencies are calculated by a matrix multiplication of the PDS frequency table by the CET Endstate conditional probability table. For each PDS, the conditional probability to go to each STC is calculated, and then the frequency of each STC is summed over all the PDSs, and shown in the STC frequency table.

An example of the matrix multiplication is provided by examining the calculation of STCs 22 and 23, which are ISLOCAs with and without fission product scrubbing, respectively. For each of the 25 PDSs, the frequency from the PDS table is multiplied by its fraction shown in the CET Endstate table. For all of the PDSs except for 24, the conditional CET fraction to STCs 22 and 23 is zero. For PDS 24, the conditional CET fraction to STC 22 is 8.50E-01, and for STC 23 it is 1.50E-01. Therefore, the STC 22 frequency is 1.62E-6 * 8.5E-1 = 1.37E-6, and the STC 23 frequency is 1.62E-6 * 1.5E-1 = 2.42E-7. The calculation of the other STC frequencies is more complicated because the fractions in the CET Endstate table is

usually not zero for more than just one PDS and the contribution from all the PDSs must be summed.

Response to 1e: Peer reviews were required by the implementation guidance given for GL 88-20. A discussion of the review team and key results is presented in Chapter 5 of the final IPE report for Surry submitted to the NRC staff on August 30, 1991 (letter No. 91-134A). In summary, the peer review consisted of a review of the level 1 and level 2 model using a team of individuals consisting of outside contractors and station personnel. The outside contractors had expertise in both level 1 analysis and level 2 analysis. The updated PRA model used in the SAMA analysis was reviewed as the pilot in the Westinghouse Owners Group (WOG) peer certification effort. The WOG peer review used an independent team of six reviewers who were onsite for a week to review the model. A rigorous review process has been defined for use by all Westinghouse plants.

The MACCS model was developed and independently reviewed by Dominion personnel and an outside contractor. The SAMA benefit calculations were performed and independently reviewed by an outside contractor.

SPS RAI No. 2:

As indicated on page 4-45, the top 100 cutsets of the updated level 1 PRA were examined for the SAMA analysis:

a. Please indicate the total fractional contribution to CDF and risk that result from these 100 top cutsets,

b. Please confirm whether an importance analysis was performed, and if so, did such analysis identify any additional SAMAs, and

c. If an importance analysis was not performed, then please justify its exclusion basis, considering that importance ranking of initiators has the potential to identify possible actions that would not appear in a listing of top cutsets.

Dominion Response:

Response to 2a: The combined CDF from the top 100 cutsets is 2.33E-5/yr, or 61.5% of the total CDF. The contribution to the total plant risk, including external events (i.e., the total "benefit" of removing all of the core damage and containment release sequences), is $3.2 Million. The total risk from the top 100 core damage cutsets is $1.8 Million. This equates to 57% of the total plant risk.

Response to 2b: An importance analysis was not performed to identify potential SAMAs.

Response to 2c: Review of the top 100 cutsets was chosen over an importance analysis for the identification of potential SAMAs because they contain the dominant contributors to risk. The top 100 cutsets contain 61.5% of the total CDF, so any cutsets not in the top 100 would not be expected to have a significant impact on the benefit calculation. The CDF of cutset #101 is 4.8E-8/yr, or 0.1% of the total CDF, and the top 100 cutsets contain the entire ISLOCA frequency and much of the SGTR contribution to offsite consequences. Since none of the SAMAs identified from the top 100 cutsets were found to be cost beneficial, it is not likely that SAMAs from the cutsets below the top 100 would be either.

It should also be noted that Surry PRA staff were surveyed for possible SAMA candidates based on their experience, so the importance of initiators was indirectly included in this way. In addition a review of the initiating events in the importance list shows that there are five initiating events with a RRW above 1.05. These initiators include the three LOCAs, the SGTR and the LOOP. Several SAMAs were considered for each of these initiators.

SPS RAI No. 3:

To account for the risk impact of internal fires and external events, you multiplied the internal events IPE CDF by 2, except for the contributions due to bypass events (i.e., ISLOCA and SGTR), on the grounds that external events would not impact the frequency of bypass scenarios. Risk analyses at other commercial nuclear power plants indicate that external events could be large contributors to core damage and the overall risk to the public. Even though the submittal indicates that the contributions of external events are not significant, the quantitative influence of SAMAs applicable to internal fires/floods and external events cannot be accurately evaluated by just doubling the estimated internal events CDF for selected sequences. In view of the fact that the characteristics of the internal and external events scenarios are, in general, considerably different and include different levels of uncertainties and conservatism, please provide the following:

a. further discussion of the uncertainties associated with the calculated CDF for internal events (including internal flooding) for SPS, (e.g., the mean and median CDF estimates and the 5th and 95th percentile values of the uncertainty distribution) and the rationale for not explicitly considering the upper end of the uncertainty distribution in the SAMA evaluation process.

b. justification that by doubling the internal events CDF, one can reliably bound the risk of core damage due to all initiators at SPS, including the impact of uncertainties in PRA results. This justification should be based on plant-specific considerations and sound PRA arguments, and

c. the technical rationale for including only a very limited number of SAMA candidates directed towards mitigation of external events.

Dominion Response:

Response to 3a: The point estimate yearly CDF was calculated to be 3.7E-5/yr. Using a random sampling tool to assess the data uncertainty distributions, the CDF distribution by confidence level was tabulated as follows:

Confidence	Unavailability
99.5	7.06E-004
99.0	3.96E-004
97.5	1.81E-004
95.0	1.16E-004
90.0	7.18E-005
80.0	4.63E-005
75.0	4.05E-005
70.0	3.63E-005
60.0	3.04E-005
50.0	2.60E-005
40.0	2.23E-005
30.0	1.93E-005
25.0	1.78E-005
20.0	1.62E-005
10.0	1.30E-005
5.0	1.11E-005
2.5	9.66E-006
1.0	8.14E-006
0.5	7.38E-006

The 5% confidence is 1.1E-5/yr, and the 95% confidence is 1.2E-4/yr. Consistent with traditional PRA calculations, the SAMA analysis was performed as a best-estimate probabilistic evaluation. For this reason, the 95% end of the uncertainty distribution was not used in the SAMA evaluation process. For the same reason the 5% end was not used. The SAMA items were screened out because in each case, their cost was greater than twice the benefit that was conservatively calculated.

A review of the table in Appendix G which provide quantitative results shows two relevant facts. First, the vast majority of SAMAs that were evaluated resulted in a cost benefit that was significantly lower than the costs to implement. For example many of the system enhancement SAMAs show a cost reduction on the order of \$100K. Any safety-related system modification would cost three or four times that amount as a minimum. Second, the majority of the SAMAs were evaluated in a bounding manner; that is, assuming that the SAMA would entirely eliminate the contribution to core damage frequency. In other words, the results tables show

conclusively that there is substantial margin in the results. The margin accounts for uncertainty.

Response to 3b: The treatment of uncertainties is described above in the response to part a of this question. The contribution from external events was treated by doubling the internal events contribution. This sufficiently bounds the risk from external events for the following reasons:

- The Surry IPEEE found that containment response to core damage external events was similar to that from the internal events in the IPE. The Surry IPEEE found no external events vulnerabilities in terms of containment bypass or isolation failure, so the offsite consequences can be bounded by using an internal events profile.
- The CDF from external events is 1.3E-5/year. This compares to a base CDF of 3.7E-5/year from the internal events model used to calculate SAMA benefit. Therefore, the doubling approach is considered conservative since an argument could me made that the internal events benefit numbers would only need to be increased by as little as 35% to account for the external events contribution.

This approach of doubling risk to account for external events is consistent with the approach of the Turkey Point SAMA submittal.

Response to 3c: The Surry as-built, as-operated plant is already designed to prevent damage from design basis external events. Buildings housing safety-related equipment are designed to withstand seismic acceleration as well as wind speeds and associated missiles from tornadoes. Thus, the primary impact of external events based on the design of the plant is a loss of offsite power. The Surry switchyard is designed with substantial redundancy. Any section of the switchyard can be switched out without jeopardizing power to any other section.

Should a loss of offsite power occur there are three emergency diesel generators available to power 3-of-the-4 emergency busses and an alternate AC diesel that can power bus 2H, 1J or both. The AAC diesel is not safety-related but has been procured using design principles to minimize the common cause failure probability of the AAC diesel and the emergency diesels. As a result the Surry electrical power distribution system already has diversity and redundancy. Improvements upon this design would require substantial redesign and would not be cost beneficial given the relatively small contribution to core damage frequency.

SPS RAI No. 4:

On page G-9 you state that the base case does not include sheltering as part of the emergency planning assumptions in the MELCOR Accident Consequence Code System (MACCS) calculations. On the other hand, on page G-28 you state that "(a)nother sensitivity run was made for the time to take shelter (MACCS parameter DLTSHL) which used 7200 seconds, whereas the base case used 5400 seconds." Please explain this discrepancy. **During the telephone call between the staff and the licensee on August 30, 2001, the licensee clarified what is presented in the Environmental Report (i.e., that the duration used for sheltering was 0 and the time to take shelter was 7200). The staff concluded that there was no discrepancy in the Environmental Report and that no response to this question is necessary.**

Dominion Response: No further response required.

SPS RAI No. 5:

In Table G.2-2 for SAMAs 35 and 36 (containment venting to remove decay heat), the calculated CDF is reduced by about 4.9 percent. Please explain the process by which the proposed vent is envisioned to remove the core-generated decay heat, thereby preventing core damage.

Dominion Response:

Response to 5: The theory is that as steam is released into containment (e.g., via a LOCA or Feed & Bleed), the containment will heat up. Installing a containment vent would allow a release of the pressure and heat that the steam would add to containment. Therefore, if injection to the core was maintained, core damage could be averted. This concept is analogous to the containment vents used in many BWRs.

The evaluation of this SAMA's benefit was made by setting all event tree nodes related to failure of containment heat removal to zero. This analysis was very conservative, because such a containment vent might not be successful in preventing core damage in all scenarios.

SPS RAI No. 6:

Please provide the basis for the low release fractions (i.e., lower than a near complete release to environment) for noble gases for source term categories (STCs) 2, 5, and 21. In addition, please explain the zero release fractions for Tellurium (STCs 2 and 21), Strontium (STCs 2 and 13), Rubidium (STC 13), Lanthanum (STCs 2, 11, 13, 15, and 21), and Cesium (STCs 11, 13, 15, 18, 21, and 22).

Dominion Response:

Response to 6: The low release fractions to the environment for noble gases are primarily due to the post accident subatmospheric conditions inside the containment. Following a LOCA a high containment pressure signal would actuate the containment spray systems which would quickly depressurize the containment below atmospheric pressure. The subatmospheric containment pressure would retain some fission products inside containment which would result in low release fractions. The zero release fractions for Tellurium (STCs 2 and 21), Strontium (STCs 2 and 13), Rubidium (STC 13), Lanthanum (STCs 2, 11, 13, 15, and 21), and Cesium (STCs 11, 13, 15, 18, 21 and 22) were the result of setting those MAAP/MACCS adjusted values from the Surry IPE that were less than 10^{-7} to zero. On a relative basis, release fractions of less than 10^{-7} magnitude would have a negligible impact on the population dose and therefore would have a negligible impact on the resulting SAMA analysis.

SPS RAI No. 7:

Please respond to the following questions related to calculated cost, benefit, and screening criteria for specific SAMAs and SAMA candidates:

a) SAMA 21 (procedural enhancements for loss of component cooling water or service water) is listed as having negligible benefit (for the option without a completely new, independent pump) because it is able to delay, but not prevent, system failure. It is not clear why delaying system failure (and presumably delaying core damage) has zero risk benefit. Please provide specific results from the risk analysis to show that this benefit is negligible. **During the telephone call between the staff and the licensee on August 30, 2001, the licensee pointed out the explanation related to SAMA 21 on page G-56 of the Environmental Report. After further reviewing the basis for the licensee's conclusion in the Environmental Report, the staff concluded that the explanation was sufficient and that no response to this question is necessary.**

b. SAMAs 43 and 44 (reactor cavity flooding) may be expected to impact environmental release due to impacts on both ex-vessel coolability and decontamination of ex-vessel release. Please provide specific results from the risk analysis that support assigning no risk benefit to these actions.

c. SAMA 70 (emphasizing steps in recovery of offsite power after a station blackout [SBO]) is estimated to have a bounding benefit of $33K. Since this SAMA could be implemented as a procedural change, it is not clear that the cost would exceed twice the benefit. Please provide and justify the cost estimate for implementation of this SAMA. Furthermore, please discuss whether any existing procedures for coping with SBOs already address SAMA 70.

d. SAMA candidate 95 (enhancing inspection activities to prevent ISLOCAs) is screened out by stating that it is not feasible to institute a 100 percent effective and complete inspection program, and that even if it were possible, such a program would extend plant outage durations and thus make the cost excessive. Given the fact that the potential benefit for eliminating ISLOCAs is substantial (about $253K), please explain why less complete or less costly measures were not considered for potential cost-benefit (e.g., rotating partial inspections during outages, or installing radiation or level alarms at strategic locations).

e. SAMA candidate 141 (enlarging the refueling water storage tank [RWST]) was apparently identified from examination of the top cutsets in the SPS PRA. However, it is screened out by stating that this change has already been implemented. Please state whether the PRA that was used takes credit for the change, and, if not, provide justification for why even further enlargement was not considered past the preliminary screening phase (since it would then still have non-zero benefit).

f. Given that the submittal indicates that SAMA candidate 151 (creation of a boron injection system [BIS]) has already been implemented, please explain why SAMA candidate 148 (creation of an alternate or backup BIS) can be screened out as "not applicable".

g. SAMA candidate 102 (installing limit valves to prevent containment isolation failure) is the only SAMA candidate dealing directly with failure to isolate the containment. Please explain why other methods for coping with containment isolation failure (e.g., procedural changes aimed at increasing isolation recovery probability before core damage) are not worth consideration. Also, please provide the frequency of core damage events at SPS accompanied by failure to isolate containment, and a bounding value for the realizable risk benefit due to such events.

h. No SAMA candidate was evaluated that would involve the refill of an affected steam generator following an un-isolated SGTR event. Refill of a damaged steam generator is typically a part of severe accident management guidelines and could be accomplished using alternative water sources (e.g., fire water). Please provide justifications for not considering this seemingly beneficial and low-cost SAMA. **During the telephone call between the staff and the licensee on August 30, 2001, the licensee pointed out that this issue is covered by SAMA number 91 in the Environmental Report. The staff concluded that there was no discrepancy in the Environmental Report and that no response to this question is necessary.**

i. What is the basis for the estimated cost of SAMA 47 (Core Melt Source Reduction System [COMSORS]) that would cause it to exceed twice the bounding benefit of $1.6 million?

Dominion Response:

Response to 7a: No further response required.

Response to 7b: There was an error discovered in the treatment of analysis case DEB, which was used as the benefit calculation of SAMAs 43 and 44. The DEB analysis was performed setting the Containment Event Tree (CET) event EXVCOOL to always "Cooled," and never "Not Cooled." This resulted in no change in risk to three significant digits. However, this method is not completely correct because a previous question in the CET checks the status of the Recirc Sprays. If they are failed, then the EXVCOOL question is not asked, and a "Not Cooled" state is assumed. Since SAMAs 43 and 44 provide a new or redundant means to flood the cavity, even the situations in which Recirc Sprays have failed should yield a "Cooled" state.

However, the analysis case SCB, which was used to evaluate SAMAs 42 and 54, is considered a bounding estimate of the benefit for SAMAs 43 and 44. Analysis case SCB found the maximum benefit from filtering all fission products in the containment to be only $45k. SAMA 43 involves creating a new system, which would cost at least an order of magnitude more than $45k to create and install. SAMA 44 suggests the use of existing equipment such as fire pumps to flood the cavity. This option would be less costly than SAMA 43, but would still require substantial analysis, training and installation and new piping costs, which would exceed twice the maximum benefit that could be obtained.

Response to 7c: The operators have a set of procedures to use when dealing with a loss of AC Power. The procedures include an emergency procedure and supporting abnormal and operating procedures. The operators currently retrain on accident scenarios at regular intervals. The LOOP scenarios are currently covered at approximately two-year intervals. Scheduling operator training is an involved process that is based on the need to cover a large amount of material. SAMA 70

was identified to look at the improvement in operator response to the existing procedures as a result of additional emphasis on the use of the existing process. This approach would involve every operator spending more time in training either on the simulator or in the classroom. The cost for SAMA 70 is the aggregate cost of additional training for every operator throughout the year. As the curriculum is heavily loaded the only way to implement this SAMA would be more training time as it is judged that it would not be practical to eliminate or trade off any of the current material.

Response to 7d: Less complete measures were not considered as a potential SAMA because of the commitment to an aging management program required as part of the license renewal process. The piping and other passive components in Safety Injection System that would be most likely to lead to an ISLOCA are included in the scope of this aging management program. The aging management program represents a commitment to maintain inspection of these components at the appropriate level.

Response to 7e: The SPS has not actually installed a larger RWST. Rather, SAMA 141 was screened by taking credit for the RWST makeup capability, which serves the same function as increasing the size of the RWST. Page G-51 of the Surry Environmental Report lists this SAMA as resulting from discussions with the PRA group. The essence of these discussions is that the model includes recovery of the RWST. The importance of this recovery action is small (RRW=1.02). Hence, the benefit would have been small compared to the cost of installing a new, larger tank even if as a result, the recovery action were always successful.

Response to 7f: SAMA 151 was screened as not being applicable because the SAMA that was proposed was specifically a redundant Standby Liquid Control (SLC) system, which is a BWR system. However, a redundant Boron Injection System could be proposed for SPS/NAPS. Such a system would have negligible benefit since, as presented in the evaluation of SAMAs 145/146, the benefit of eliminating all ATWSs at SPS is only $1k. Therefore, this SAMA is eliminated because of its low benefit.

Response to 7g: Containment isolation failure is less significant at Surry than at other plants because its containment is a sub-atmospheric design. Large, pre-existing leaks, for example, need not be considered because they would be noticed during the course of normal operation. Plant damage states 1 and 2 are core damage with failure of containment isolation, and their frequency sums to 1.28E-7/yr. If these PDSs are completely eliminated, the bounding benefit becomes $8,600 after being doubled for external events ($4,300 from internal events). Because of the low frequency associated with core damage and containment isolation failure, no other methods for coping with containment isolation failure (including procedural changes) were considered. The low bounding benefit justifies that assumption as being valid, since even procedural changes would have costs greater than twice the bounding benefit.

Response to 7h: No further response required.

Response to 7i: The COMSORS concept involves placing a huge mass of glass-like material under the reactor vessel. The theory was that a molten core exiting the vessel would combine with the material to form a vitrified compound that would contain the fission products within. Besides the cost of the material (and the analysis costs), the reactor cavity would have to be expanded to be able to house it. Expanding the size of the reactor cavity alone would cost millions of dollars, which was the reason the COMSORS was screened out. In reality, this SAMA probably could have been screened as only being applicable to a new plant, since any substantial modifications to the containment cavity really are not feasible to an existing plant.

SPS RAI No. 8:

Please clarify the benefit and estimated cost values reported in Table 4-6, and how they were obtained. Specifically:

a. It is the staff's understanding that the "Benefit" values reflect a doubling to account for external events, and that the "Estimated Cost" values also reflect a doubling (the "2 x" in the equation, "2 x benefit") to account for cost uncertainty (page 4-48). Is this correct?

b. On page 4-70, for SAMA 47, the benefit is given as $1.6 million. We assumed that this was determined by multiplying 100 percent times the maximum benefit that can be obtained by mitigating the consequences of a core damage accident, namely $1.6 million. If this is the case, there appears to be an inconsistency in the value for SAMA 42 (page 4-69), a similar SAMA in the sense that the reduction in CDF is zero, for which the bounding benefit is estimated to be $45K. The corresponding Reduction in Person-Rem Offsite is 4.9 percent. According to our calculation (4.9% times $1.6 million), the bounding benefit should be $78K, not $45K. Please explain this apparent discrepancy.

Dominion Response:

Response to 8a: The staff's understanding is correct.

Response to 8b: The benefit of a SAMA that affects offsite releases is not directly proportional to the offsite person-REM incurred, so the percentage reduction cannot simply be multiplied by the maximum benefit of removing all releases. Other factors that impact the offsite benefits include averted crop damage, property damage, evacuation costs, etc. The benefit for SAMA 42 was conservatively estimated by eliminating the release for all non-bypass events (i.e., all events except SGTR,

ISLOCA and containment isolation failure, which would not benefit from in-containment scrubbing). The offsite consequences of these release events (and how they affect offsite crop damage, etc.) varies from the other release types, with bypass events being the most significant. Therefore, the calculated benefit came to $45K compared to the $78K that would be calculated by a direct multiplication of the percentage difference in offsite person-REM.

Note also that the benefit calculation is conservative in that it credited removal of all fission products from the containment prior to release, while in reality, even a fission product scrubbing mechanism would not be able to prevent noble gas releases.

Questions Related to North Anna Power Station (NAPS)

NAPS RAI No. 1:

The license renewal SAMA analysis is based on an internal events CDF of 3.5 x 10^{-5} per reactor-year, which is about 50 percent of the value reported in the IPE (i.e., 6.8 x 10^{-5} per reactor-year). Also, on page G-18 of the submittal, it is stated that the level 2 PRA for NAPS was updated for the purpose of the SAMA evaluation. In this regard please provide the following:

a. a discussion of the reasons for the reduction in CDF, including a description of the major changes in PRA models/assumptions and plant hardware/procedures, and their respective impacts on CDF,

b. a breakdown of the internal event CDF (3.5 x 10^{-5} per reactor-year) by initiating event, specifically, transients, LOCA, SGTR, and ISLOCA,

c. a description of the changes to the level 2 analysis and their impact on results,

d. the updated conditional probabilities associated with each release category for each PDS (i.e., the Containment Matrix), the list of updated plant damage state definitions if different from IPE, and the updated PDS frequencies and release class frequencies, and

e. a description of the internal and external peer reviews of the PRA used in the SAMA analysis.

Dominion Response:

Response to 1a: Three major model revisions occurred between the time the IPE was submitted and the preparation of the license renewal SAMA analyses. The first was an update prior to conducting the IPEEE fire analysis. The second was an update in 1997. The third update occurred in 1998. A fourth update was completed in 2000 in order to implement updates identified by the maintenance rule expert panel. Each of the updates incorporated significant plant modifications, corrected model errors or enhanced the model with state-of-the-art improvements as appropriate. Some model updates were also made to support implementation of the maintenance rule. A summary of the changes made is listed below:

- Revised the Emergency Diesel Generator (EDG) fault tree to include explicit representation of the fuel oil transfer pumps.
- Incorporated the alternate AAC diesel fully into the model.
- Added a CW system fault tree to include the dependency of the steam dumps on this system.

- Revised fault trees to allow manual start of the C Charging Pump fault trees when the automatically started pump is unavailable.
- Added charging pump failure modes for loss of ventilation
- Added Service Water dependency to the Instrument Air Heat Exchangers failure modes and the Charging Pumps failure modes
- Added a model of the cross-tie between the Unit 1 and Unit 2 charging pumps.
- Added the dependency of the Reactor Coolant Pumps on Component Cooling.
- Revised the Service Water system fault tree to reflect all four Service Water pumps.
- Added Test and Maintenance Faults for numerous components to support determination of performance criteria for the maintenance rule.

The individual contribution of each of the changes to the PRA is not available, but collectively they account for the reduction of CDF to 3.5E-5/yr. However, a breakdown of the change in core damage frequency for each initiator is available and is shown in the table below.

Model Comparison of CDF For Each Accident Group							
Accident Group	**Accident Type**	**96 Feb With TM-0**	**96 Feb With TM-1**	**N7B With TM-0**	**N7B With TM-1**	**N7B With 3-YR**	**92 Dec (IPE)**
LOCA Transients	Small LOCA (S2)	8.32E-6	1.16E-5	7.00E-6	8.95E-6	7.41E-6	1.01E-5
	SGTR (T7)	6.76E-6	8.90E-6	4.14E-6	4.92E-6	4.16E-6	7.02E-6
	Medium LOCA (S1)	6.33E-6	7.21E-6	4.66E-6	5.43E-6	4.81E-6	6.64E-6
	Large LOCA (A)	3.89E-6	4.27E-6	4.02E-6	4.37E-6	4.10E-6	4.09E-6
	Inter System LOCA (Vx)	1.60E-6	1.60E-6	1.60E-6	1.60E-6	1.60E-6	1.60E-6
	Reactor Vessel Rupture (Rx)	2.66E-7	2.66E-7	2.66E-7	2.66E-7	2.66E-7	2.68E-7
	RCP Seal LOCA (T4)	1.81E-8	3.32E-8	1.81E-8	3.17E-8	2.09E-8	1.07E-8
	Subtotal	2.72E-5	3.39E-5	2.17E-5	2.56E-5	2.24E-5	2.97E-5
Electrical Transients	Station Blackout (T1A)	4.77E-6	4.82E-6	4.66E-6	6.05E-6	5.19E-6	7.98E-6
	Loss of Off-Site Power (T1, T1HV, T1Q)	8.60E-7	3.24E-6	2.70E-6	4.71E-6	3.29E-6	1.19E-5
	Loss of 1-EE-SW-1H (T9A)	1.96E-7	3.12E-7	2.55E-7	3.74E-7	2.78E-7	3.68E-6
	Loss of 1-EE-SW-1J (T9B)	1.91E-7	2.18E-7	2.14E-7	2.77E-7	2.29E-7	6.49E-7
	Loss of 1-EP-CB-12A (T5A)	9.31E-9	2.18E-8	2.08E-8	4.13E-8	2.53E-8	1.11E-7
	Loss of 1-EP-CB-12C (T5B)	7.92E-9	2.04E-8	2.08E-8	3.92E-8	2.45E-8	1.09E-7
	Sub-total	6.03E-6	8.63E-6	7.87E-6	1.15E-5	9.04E-6	2.44E-5
General Transients	Loss of Feedwater (T23 or the IPE T2, T3, T2HV, T2AHV, T3HV)	8.74E-7	1.76E-6	1.95E-6	3.29E-6	2.32E-6	6.88E-6
	Loss of SW (T6)	5.39E-7	5.26E-7	5.41E-7	5.41E-7	5.41E-7	4.52E-9
	ATWS (TH, THMFW, (TL)	4.24E-7	4.24E-7	4.34E-7	4.40E-7	4.40E-7	4.20E-7
	Loss of ESGR HVAC, (T8)	2.87E-7	2.94E-7	2.86E-7	3.22E-7	2.89E-7	6.56E-6
	Sub-total	2.12E-6	3.00E-6	3.21E-6	4.59E-6	3.59E-6	1.39E-5
Grand Total		3.56E-5	3.88E-5	3.28E-5	4.17E-5	3.50E-5	6.79E-5

Response to 1b: The CDF for North Anna grouped by sequence and by category is in the Table below. It should be noted that in this table the Initiating Event groupings have been provided in slightly more detail than the table above. For example the T1 trees with conditional failures are shown individually in the table below while they are combined into a single T1 contribution in the response to question 1a.

Init Event	**Description**	**CDF**	**% of Total CDF**
S2	Small LOCA	7.41E-06	21.20%
T1A	SBO	5.19E-06	14.80%
S1	Medium LOCA	4.82E-06	13.80%
T7	SGTR	4.16E-06	11.90%
A	Large LOCA	4.10E-06	11.70%
T1	Loss of Offsite Power	2.38E-06	6.80%
T23	Transient/loss of MFW	2.33E-06	6.60%
VX	ISLOCA	1.60E-06	4.60%
T1HV	Loss of offsite power and loss of ESGR HVAC	9.02E-07	2.60%
T6	Loss of Service Water	5.41E-07	1.50%
TH	ATWS from high power	4.40E-07	1.30%
T8	Loss of ESGR HVAC	2.89E-07	0.80%
T9A	Loss of 4KV AC bus 1H	2.78E-07	0.80%
RX	RV Rupture	2.66E-07	0.80%
T9B	Loss of 4KV AC bus 1J	2.29E-07	0.70%
T5A	Loss of 125V DC bus 1A	2.53E-08	0.10%
T5B	Loss of 125V DC bus 1B	2.45E-08	0.10%
T4	RCP Seal LOCA	2.09E-08	0.10%
T1Q	Loss of offsite power with stuck open PORV	5.96E-09	0.00%
TL	ATWS from low power	4.35E-10	0.00%
Total		**3.50E-05**	**100%**
LOCAs		1.63E-05	46.64%
LOOP/SBO		8.48E-06	24.21%
SGTR		4.16E-06	11.88%
Transients		3.74E-06	10.68%
ISLOCA		1.60E-06	4.57%
ATWS		4.40E-07	1.26%
RV Rupture		2.66E-07	0.76%
Total		**3.50E-05**	**100%**

Response to 1c: For Surry and North Anna the level 2 model consists of several logic trees. These trees are used to evaluate accident progression for each plant damage state as well as for appropriate grouping of the accident sequences into source term categories. Accident progression is evaluated using Containment Event Trees. These trees are quantified using probabilities that are assigned using rules. The source term grouping diagram is a tree that combines

accident sequences based on similarity of source term. The source terms used in the IPE were obtained from MAAP3B and were not updated for this license renewal work.

A small number of changes were made to the level 2 model prior to performing the SAMA analysis. First, the Surry and North Anna level 2 models were made consistent. So, both models determine Source Term Categories using the same grouping criteria. This change reflects the status of the level 2 modeling at the time the IPE was completed. The North Anna work structure was an improvement to the work done for Surry. Since the level 2 models were converted to LERF models shortly after the IPE/IPEEE process was completed there was no incentive to change the level 2 models until a need was identified. Hence, prior to beginning the SAMA analysis a unified Source Term Category grouping was implemented which essentially was to use the approach presented in the North Anna IPE.

The general containment event tree (CET) was also slightly modified to reflect recent experimental results in severe accident analysis research. In particular the probabilities of the following events were modified:

- The conditional probability of early containment failure due to Direct Containment Heating or Hydrogen burns is set to zero.
- Changed the probabilities of vessel depressurization before vessel rupture slightly.
- Set the Alpha failure mode (steam explosion) to zero.

The net impact of these changes is small because the magnitude of the changes is small. The Alpha mode failure probability is set to zero but it was already no larger than .008. Similarly, the branch fractions for mode of induced primary system failure changed such that the branch fraction of hot leg failures increased from 0.72 to 0.88 and the branch fraction for No RCS failure decreased from .26 to 0.1. The direction of the change is a shift of the plant damage state frequency from source term categories that would be considered LERF to those that would not be considered LERF either because there was no containment failure or if the containment did fail it would be late.

Response to 1d: The PDS definitions have not changed since the IPE.

The PDS frequencies are provided in the following table:

Baseline PDS Frequencies

PDS:	Base Freq
1	8.17E-08
2	4.10E-08
3	7.12E-07
4	1.43E-06
5	1.33E-09
6	2.77E-09
7	1.43E-06
8	3.50E-06
9	7.77E-09
10	2.70E-09
11	2.75E-07
12	3.10E-06
13	1.25E-06
14	2.11E-07
15	2.87E-07
16	7.74E-07
17	1.98E-07
18	1.74E-06
19	1.23E-06
20	7.59E-06
21	5.04E-06
22	9.87E-09
23	3.62E-07
24	1.60E-06
25	4.16E-06
Total	**3.50E-05**

The conditional probabilities associated with each release category for each plant damage state are provided in the following table:

Containment Event Tree Endstate Conditional Probabilities For Each PDS

	PDS												
STC	1	2	3	4	5	6	7	8	9	10	11	12	13
1	0.00E+00	0.00E+00	2.78E-01	7.74E-01	0.00E+00	0.00E+00	0.00E+00	1.82E-01	9.28E-01	0.00E+00	0.00E+00	4.70E-02	9.40E-01
2	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
4	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
5	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
6	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
7	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
8	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
9	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.76E-01	0.00E+00	0.00E+00	0.00E+00
10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.68E-02	0.00E+00	0.00E+00	0.00E+00
11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.83E-02	0.00E+00	0.00E+00	0.00E+00
12	0.00E+00	0.00E+00	0.00E+00	2.65E-02	5.57E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.20E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
14	0.00E+00	0.00E+00	1.85E-04	0.00E+00	5.79E-02	6.05E-01	6.05E-01	6.16E-04	6.16E-04	6.16E-04	6.05E-01	0.00E+00	0.00E+00
15	0.00E+00	0.00E+00	1.15E-04	0.00E+00	4.71E-02	3.77E-01	3.77E-01	3.84E-04	3.84E-04	3.84E-04	3.77E-01	0.00E+00	0.00E+00
16	0.00E+00	0.00E+00	1.59E-02	1.81E-01	0.00E+00	0.00E+00	0.00E+00	5.52E-03	5.32E-02	0.00E+00	0.00E+00	3.00E-03	6.00E-02
17	7.50E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
18	0.00E+00	1.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
20	0.00E+00	0.00E+00	6.87E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.94E-01	0.00E+00	0.00E+00	0.00E+00	9.50E-01	0.00E+00
21	2.50E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
24	0.00E+00	0.00E+00	1.80E-02	1.80E-02	1.80E-02	1.80E-02	1.80E-02	1.80E-02	1.80E-02	1.80E-02	1.80E-02	0.00E+00	0.00E+00
SUM	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00

Containment Event Tree Endstate Conditional Probabilities For Each PDS (continued)

	PDS											
STC	14	15	16	17	18	19	20	21	22	23	24	25
1	2.96E-01	9.88E-01	0.00E+00	0.00E+00	9.00E-01	0.00E+00	9.88E-01	9.88E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
4	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
5	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
6	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
7	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
8	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
9	0.00E+00	0.00E+00	8.84E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.84E-01	0.00E+00	0.00E+00	0.00E+00
10	0.00E+00	0.00E+00	5.74E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.74E-02	0.00E+00	0.00E+00	0.00E+00
11	0.00E+00	0.00E+00	4.88E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.88E-02	0.00E+00	0.00E+00	0.00E+00
12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
14	1.85E-03	6.16E-03	6.16E-03	6.16E-01	0.00E+00	6.16E-01	6.16E-03	6.16E-03	6.16E-03	6.16E-01	0.00E+00	0.00E+00
15	1.15E-03	3.84E-03	3.84E-03	3.84E-01	0.00E+00	3.84E-01	3.84E-03	3.84E-03	3.84E-03	3.84E-01	0.00E+00	0.00E+00
16	6.83E-04	2.28E-03	0.00E+00	0.00E+00	1.00E-01	0.00E+00	2.28E-03	2.28E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
20	7.00E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
22	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.50E-01	0.00E+00
23	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.50E-01	0.00E+00
24	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.00E+00
SUM	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00

The release category frequencies are provided in the following table:

Source Term Category	Frequency
1	1.76E-05
2	0.00E+00
3	0.00E+00
4	0.00E+00
5	0.00E+00
6	0.00E+00
7	0.00E+00
8	0.00E+00
9	6.95E-07
10	4.51E-08
11	3.84E-08
12	3.85E-08
13	4.24E-10
14	2.22E-06
15	1.38E-06
16	5.77E-07
17	6.12E-08
18	4.10E-08
19	0.00E+00
20	6.35E-06
21	2.04E-08
22	1.36E-06
23	2.40E-07
24	4.29E-06

The Source Term Category frequencies are calculated by a matrix multiplication of the PDS frequency table by the CET Endstate conditional probability table. For each PDS, the conditional probability to go to each STC is calculated, and then the frequency of each STC is summed over all the PDSs, and shown in the STC frequency table.

An example of the matrix multiplication is provided by examining the calculation of STCs 22 and 23, which are ISLOCAs with and without fission product scrubbing, respectively. For each of the 25 PDSs, the frequency from the PDS table is multiplied by its fraction shown in the CET Endstate table. For all of the PDSs except for 24, the conditional CET fraction to STCs 22 and 23 is zero. For PDS 24, the conditional CET fraction to STC 22 is 8.50E-01, and for STC 23 it is 1.50E-01. Therefore, the STC 22 frequency is 1.60E-6 * 8.5E-1 = 1.36E-6, and the STC 23 frequency is 1.60E-6 * 1.5E-1 = 2.40E-7. The calculation of the other STC frequencies is more complicated because the fractions in the CET Endstate table is usually not zero for more than just one PDS and the contribution from all the PDSs must be summed.

Response to 1e: Peer reviews were required by the implementation guidance given for GL 88-20. A discussion of the review team and key results is presented in Chapter 5 of the final IPE report for North Anna submitted to the NRC staff on December 14, 1992 (letter No. 92-774). In summary, the peer review consisted of a review of the level 1 and level 2 model using a team of individuals consisting of outside contractors and station personnel. The outside contractors had expertise in both level 1 analysis and level 2 analysis. The updated PRA model used in the SAMA analysis is based on the peer reviewed IPE model.

NAPS RAI No. 2:

As indicated on page 4-45, the top 100 cutsets of the updated level 1 PRA were examined for the SAMA analysis:

a. Please indicate the total fractional contribution to CDF and risk that result from these 100 top cutsets,

b. Please confirm whether an importance analysis was performed, and if so, did such analysis identify any additional SAMAs, and

c. If an importance analysis was not performed, then please justify its exclusion basis, considering that importance ranking of initiators has the potential to identify possible actions that would not appear in a listing of top cutsets. Note that, in your discussion of Level 1 PRA results in the North Anna IPE, you provide both a list of top cutsets (Table 3.4.1-3) and F-V, RRW, and RAW importance ranking (Table 3.4.1-6), as well as several pages of discussion (on pages 3-116 through 3-119) on the most significant events for risk reduction based on this importance ranking.

Dominion Response:

Response to 2a: The combined CDF from the top 100 cutsets is 2.46E-5/yr, or 70.2% of the total CDF. The contribution to the total plant risk, including external events (i.e., the total "benefit" of removing all of the core damage and containment release sequences), is $3.6 Million. The total risk from the top 100 core damage cutsets is $2.6 Million. This equates to 72% of the total plant risk.

Response to 2b: An importance analysis was not performed to identify potential SAMAs.

Response to 2c: Review of the top 100 cutsets was chosen over an importance analysis for the identification of potential SAMAs because they contain the dominant contributors to risk. The top 100 cutsets contain 70.2% of the total CDF, so any cutsets not in the top 100 would not be expected to have a

significant impact on the benefit calculation. The CDF of cutset #101 is 4.9E-8/yr, or 0.1% of the total CDF, and the top 100 cutsets contain the entire ISLOCA frequency and much of the SGTR contribution to offsite consequences. Since none of the SAMAs identified from the top 100 cutsets were found to be cost beneficial, it is not likely that SAMAs from the cutsets below the top 100 would be either.

It should also be noted that North Anna PRA staff were surveyed for possible SAMA candidates based on their experience, so the importance of initiators was indirectly included in this way. In addition a review of the initiating events in the importance list shows that there are five initiating events with a RRW above 1.05. These initiators include the three LOCAs, the SGTR and the LOOP. Several SAMAs were considered for each of these initiators.

NAPS RAI No. 3:

To account for the risk impact of internal fires and external events, you multiplied the internal events IPE CDF by 2, except for the contributions due to bypass events (i.e., ISLOCA and SGTR), on the grounds that external events would not impact the frequency of bypass scenarios. Risk analyses at other commercial nuclear power plants indicate that external events could be large contributors to core damage and the overall risk to the public. Even though the submittal indicates that the contributions of external events are not significant, the quantitative influence of SAMAs applicable to internal fires/floods and external events cannot be accurately evaluated by just doubling the estimated internal events CDF for selected sequences. In view of the fact that the characteristics of the internal and external events scenarios are, in general, considerably different and include different levels of uncertainties and conservatism, please provide the following:

a. further discussion of the uncertainties associated with the calculated CDF for internal events (including internal flooding) for NAPS, (e.g., the mean and median CDF estimates and the 5^{th} and 95^{th} percentile values of the uncertainty distribution) and the rationale for not explicitly considering the upper end of the uncertainty distribution in the SAMA evaluation process.

b. justification that by doubling the internal events CDF, one can reliably bound the risk of core damage due to all initiators at NAPS, including the impact of uncertainties in PRA results. This justification should be based on plant-specific considerations and sound PRA arguments, and

c. the technical rationale for including only a very limited number of SAMA candidates directed towards mitigation of external events.

Dominion Response:

Response to 3a: The point estimate yearly CDF was calculated to be 3.5E-5/yr. Using a random sampling tool to assess the data uncertainty distributions, the CDF distribution by confidence level was tabulated as follows:

Confidence	Unavailability
99.5	4.31E-003
99.0	1.88E-003
97.5	4.87E-004
95.0	1.84E-004
90.0	9.02E-005
80.0	5.44E-005
75.0	4.65E-005
70.0	4.08E-005
60.0	3.35E-005
50.0	2.85E-005
40.0	2.45E-005
30.0	2.11E-005
25.0	1.96E-005
20.0	1.80E-005
10.0	1.44E-005
5.0	1.23E-005
2.5	1.09E-005
1.0	9.37E-006
0.5	8.46E-006

The 5% confidence is 1.2E-5/yr, and the 95% confidence is 1.8E-4/yr. Consistent with traditional PRA calculations, the SAMA analysis was performed as a best-estimate probabilistic evaluation. For this reason, the 95% end of the uncertainty distribution was not used in the SAMA evaluation process. For the same reason the 5% end was not used. The SAMA items were screened out because in each case, their cost was greater than twice the benefit that was conservatively calculated.

A review of the table in Appendix G which provide quantitative results shows two relevant facts. First, the vast majority of SAMAs that were evaluated resulted in a cost benefit that was significantly lower than the costs to implement. For example many of the system enhancement SAMAs show a cost reduction on the order of $100K. Any safety-related system modification would cost three or four times that amount as a minimum. Second, the majority of the SAMAs were evaluated in a bounding manner; that is, assuming that the SAMA would entirely eliminate the contribution to core damage frequency. In other words, the results tables show conclusively that there is substantial margin in the results. The margin accounts for uncertainty.

Response to 3b: The treatment of uncertainties is described above in the response to part a of this question. The contribution from external events was treated by doubling the internal events contribution. This sufficiently bounds the risk from external events for the following reasons:

- The North Anna IPEEE found that containment response to core damage external events was similar to that from the internal events in the IPE. The North Anna IPEEE found no external events vulnerabilities in terms of containment bypass or isolation failure, so the offsite consequences can be bounded by using an internal events profile.
- The CDF from external events is 3.9E-6/year. This compares to a base CDF of 3.5E-5/year from the internal events model used to calculate SAMA benefit. Therefore, the doubling approach is considered conservative since an argument could me made that the internal events benefit numbers would only need to be increased by as little as 11% to account for the external events contribution.

This approach of doubling risk to account for external events is consistent with that used in the Turkey Point SAMA submittal.

Response to 3c: The North Anna as-built, as-operated plant is already designed to prevent damage from design basis external events. Buildings housing safety-related equipment are designed to withstand seismic acceleration as well as wind speeds and associated missiles from tornadoes. Thus, the primary impact of external events based on the design of the plant is a loss of offsite power.

However, the NAPS switchyard is designed with substantial redundancy. Any section of the switchyard can be switched out without jeopardizing power to any other section. The 500-kV switchyard was initially designed as a ring bus but was changed to a "breaker-and-a-half" design with the addition of the third transmission line. The transmission interconnections are as follows:

1. A 500-kV line to the east to a 500-kV switching station near Ladysmith, Virginia, provides a connection.
2. A 500-kV line to the north to a substation near Morrisville, Virginia provides a second connection to the North Anna Switchyard 500-kV system.
3. A 500-kV line to the south to a substation near Midlothian, Virginia provides a third connection to the North Anna Switchyard 500-kV system.
4. A 230-kV line to the west to a substation near Gordonsville, Virginia provides a connection to the North Anna Switchyard 230-kV system.

Each of the four transmission lines leaves North Anna on a different right-of-way. The entire output of the two units can be carried on any one of three 500-kV lines. The 230-kV line can only carry approximately one-third of the output of one

nuclear unit due to the size of the 500/230-kV transformer. Thus, the 500-kV system is extensive and interconnects with neighboring utility grids to the north, south, and west. The additions to the system associated with North Anna Units 1 and 2 and the interconnection to the 230-kV system further strengthen this system and increase its reliability.

Should a loss of offsite power occur there are four emergency diesel generators available to power each of the emergency busses and an alternate AC diesel that can power bus any single emergency bus. The AAC diesel is not safety-related but has been procured using design principles to minimize the common cause failure probability of the AAC diesel and the emergency diesels. As a result the North Anna electrical power distribution system already has diversity and redundancy. Improvements upon this design would require substantial redesign and would not be cost beneficial given the relatively small contribution to core damage frequency.

NAPS RAI No. 4:

On page G-9 you state that the base case does not include sheltering as part of the emergency planning assumptions in MACCS calculations. On the other hand, on page G-28 you state that "(a)nother sensitivity run was made for the time to take shelter (MACCS parameter DLTSHL) which used 7200 seconds, whereas the base case used 5400 seconds." Please explain this discrepancy. **During the telephone call between the staff and the licensee on August 30, 2001, the licensee clarified what is presented in the Environmental Report (i.e., that the duration used for sheltering was 0 and the time to take shelter was 7200). The staff concluded that there was no discrepancy in the Environmental Report and that no response to this question is necessary.**

Dominion Response: No further response required.

NAPS RAI No. 5:

In Table G.2-2 for SAMAs 35 and 36 (containment venting to remove decay heat), the calculated CDF is reduced by about 0.7 percent. Please explain the process by which the proposed vent is envisioned to remove the core generated decay heat, thereby preventing core damage.

Dominion Response:

Response to 5: The theory is that as steam is released into containment (e.g., via a LOCA or Feed & Bleed), the containment will heat up. Installing a containment vent would allow a release of the pressure and heat that the steam would add to containment. Therefore, if injection to the core was maintained,

core damage could be averted. This concept is analogous to the containment vents used in many BWRs.

The evaluation of this SAMA's benefit was made by setting all event tree nodes related to failure of containment heat removal to zero. This analysis was very conservative, because such a containment vent might not be successful in preventing core damage in all scenarios.

NAPS RAI No. 6:

Please provide the basis for the low release fractions (i.e., lower than a near complete release to environment) for noble gases for source term categories (STCs) 2, 5, and 21. In addition, please explain the zero release fractions for Tellurium (STCs 2 and 21), Strontium (STCs 2 and 13), Rubidium (STC 13), Lanthanum (STCs 2, 11, 13, 15, and 21), and Cesium (STCs 11, 13, 15, 18, 21, and 22).

Dominion Response:

Response to 6: The low release fractions to the environment for noble gases are primarily due to the post accident subatmospheric conditions inside the containment. Following a LOCA a high containment pressure signal would actuate the containment spray systems that would quickly depressurize the containment below atmospheric pressure. The subatmospheric containment pressure would retain some fission products inside containment that would result in low release fractions. Zero release fractions for Tellurium (STCs 2 and 21), Strontium (STCs 2 and 13), Rubidium (STC 13), Lanthanum (STCs 2, 11, 13, 15, and 21), and Cesium (STCs 11, 13, 15, 18, 21 and 22) were the result of setting those MAAP/MACCS adjusted values from the Surry IPE that were less than 10^{-7} to zero. On a relative basis, release fractions of less than 10^{-7} magnitude would have a negligible impact on the population dose and therefore would have a negligible impact on the resulting SAMA analysis. For North Anna it was determined that the Surry MAAP runs were applicable based on the similarity of the plant designs. Thus, the same source term results are presented.

NAPS RAI No. 7:

Please respond to the following questions related to calculated cost, benefit, and screening criteria for specific SAMAs and SAMA candidates:

a. SAMA 21 (procedural enhancements for loss of component cooling water or service water) is listed as having negligible benefit ($0) because it is able to delay, but not prevent, system failure. It is not clear why delaying system failure (and presumably delaying core damage) has zero risk benefit. Please provide specific results from the risk analysis to show that this benefit is negligible. **During the telephone call between the staff and the licensee on August 30, 2001, the licensee pointed out the explanation related to SAMA 21 on page G-56 of the Surry Power Station Environmental Report. The same basis would apply to North Anna Power Station. After further reviewing the basis for the licensee's conclusion in the Environmental Report, the staff concluded that the explanation was sufficient and that no response to this question is necessary.**

b. SAMAs 43 and 44 (reactor cavity flooding) may be expected to impact environmental release due to impacts on both ex-vessel coolability and decontamination of ex-vessel release. Please provide specific results from the risk analysis that support assigning no risk benefit to these actions.

c. In SAMA 60, the estimated benefit for providing longer battery capability is stated to be $876K, while in SAMAs 61 and 64, the maximum benefit for extending battery power is given as $29K. In addition, from screened SAMA candidate 66, increasing battery reliability apparently has negligible risk benefit, so the above stated benefits are presumably due only to extended battery operating time. Please explain the reason for this apparent discrepancy in benefit for extending battery operating time.

d. SAMA 70 (emphasizing steps in recovery of offsite power after a SBO) is estimated to have a bounding benefit of $72K. Since this SAMA could be implemented as a procedural change, it is not clear that the cost would exceed twice the benefit. Please provide and justify the cost estimate for implementation of this SAMA. Furthermore, please discuss whether any existing procedures for coping with SBOs already address SAMA 70.

e. SAMA candidate 93 (enhancing inspection activities to prevent ISLOCAs) is screened out by stating that it is not feasible to institute a 100 percent effective and complete inspection program, and that even if it were possible, such a program would extend plant outage durations and thus make the cost excessive. Given the fact that the potential benefit for eliminating ISLOCAs is substantial (about $220K), please explain why

less complete or less costly measures were not considered for potential cost-benefit (e.g., rotating partial inspections during outages, or installing radiation or level alarms at strategic locations).

f. SAMA candidate 139 (enlarging the RWST) was apparently identified from examination of the top cutsets in the NAPS PRA. However, it is screened out by stating that this change has already been implemented. Please state whether the PRA that was used takes credit for the change, and, if not, provide justification for why even further enlargement was not considered past the preliminary screening phase (since it would then still have non-zero benefit).

g. Given that the submittal indicates that SAMA candidate 149 (creation of a BIS) has already been implemented, please explain why SAMA candidate 146 (creation of an alternate or backup BIS) can be screened out as "not applicable".

h. SAMA candidate 100 (installing limit valves to prevent containment isolation failure) is the only SAMA candidate dealing directly with failure to isolate the containment. Please explain why other methods for coping with containment isolation failure (e.g., procedural changes aimed at increasing isolation recovery probability before core damage) are not worth consideration. Also, please provide the frequency of core damage events at NAPS accompanied by failure to isolate containment, and a bounding value for the realizable risk benefit due to such events.

i. No SAMA candidate was evaluated that would involve the refill of an affected steam generator following an un-isolated SGTR event. Refill of a damaged steam generator is typically a part of severe accident management guidelines and could be accomplished using alternative water sources (e.g., fire water). Please provide justifications for not considering this seemingly beneficial and low-cost SAMA. **During the telephone call between the staff and the licensee on August 30, 2001, the licensee pointed out that this issue is covered by SAMA number 89 in the Environmental Report. The staff concluded that there was no discrepancy in the Environmental Report and that no response to this question is necessary.**

j. What is the basis for estimated cost of SAMA 47 (Core Melt Source Reduction System [COMSORS]) that would cause it to exceed twice the bounding benefit of $2.2 million?

Dominion Response:

Response to 7a: No further response required.

Response to 7b: There was an error discovered in the treatment of analysis case DEB, which was used as the benefit calculation of SAMAs 43 and 44. The DEB analysis was performed setting the Containment Event Tree (CET) event EXVCOOL to always "Cooled," and never "Not Cooled." This resulted in no change in risk to three significant digits. However, this method is not completely correct because a previous question in the CET checks the status of the Recirc Sprays. If they are failed, then the EXVCOOL question is not asked, and a "Not Cooled" state is assumed. Since SAMAs 43 and 44 provide a new or redundant means to flood the cavity, even the situations in which Recirc Sprays have failed should yield a "Cooled" state.

However, the analysis case SCB, which was used to evaluate SAMAs 42 and 54, is considered a bounding estimate of the benefit for SAMAs 43 and 44. Analysis case SCB found the maximum benefit from filtering all fission products in the containment to be only $14k. SAMA 43 involves creating a new system, which would likely cost at least one or two orders of magnitude more than $14k to create and install. SAMA 44 suggests the use of existing equipment such as fire pumps to flood the cavity. This option would be less costly than SAMA 43, but would still require substantial analysis, training and installation and new piping costs, which would exceed twice the maximum benefit that could be obtained.

Response to 7c: For each of these SAMAs, the benefit should have been listed as $29K because each one deals with mitigating the consequences of an SBO. The benefits were calculated assuming more mitigation for SAMA 60 than the other SAMAs. For SAMA 60, the benefit calculation set all battery depletion to zero, while SAMAs 61 and 64 only set the depletion to zero for Station Blackout (SBO) events. Both methods would have come up with the same result with the exception that the evaluation of SAMA 60 did not disregard a conservative assumption related to SGTR events. When an AC bus failed after a SGTR in the NAPS PRA, no credit was taken for the batteries because it was conservatively assumed that the batteries would deplete prior to depressurization of the RCS through secondary side steam dump. However, such depressurization occurs 1-2 hours before battery depletion, and as such, the assumed failure was conservative.

Because SAMA 60 would be so costly, the original bounding estimate of $876k was not refined further, but for SAMAs 61 and 64, the more detailed evaluation was performed. In evaluating the SAMA benefit only for SBO sequences and disregarding the SGTR effect, the actual benefit of $29K was obtained.

Response to 7d: The operators have a set of procedures to use when dealing with a loss of AC Power. The procedures include an emergency procedure and supporting abnormal and operating procedures. The operators currently retrain on accident scenarios at regular intervals. The LOOP scenarios are currently covered at approximately two-year intervals. Scheduling operator training is an involved process that is based on the need to cover a large amount of material.

SAMA 70 was identified to look at the improvement in operator response to the existing procedures as a result of additional emphasis on the use of the existing process. This approach would involve every operator spending more time in training either on the simulator or in the classroom. The cost for SAMA 70 is the aggregate cost of additional training for every operator throughout the year. As the curriculum is heavily loaded the only way to implement this SAMA would be more training time as it is judged that it would not be practical to eliminate or trade off any of the current material.

Response to 7e: Less complete measures were not considered as a potential SAMA because of the commitment to an aging management program required as part of the license renewal process. The piping and other passive components in Safety Injection System that would be most likely to lead to an ISLOCA are included in the scope of this aging management program. The aging management program represents a commitment to maintain inspection of these components at the appropriate level.

Response to 7f: The NAPS has not actually installed a larger RWST. Rather, SAMA 139 was screened by taking credit for the RWST makeup capability, which serves the same function as increasing the size of the RWST. Therefore, further enlargement of the RWST is not applicable. The RWST makeup is credited in the PRA. Page G-49 of the NAPS Environmental Report lists this SAMA as resulting from discussions with the PRA group. The importance of this recovery action is small based on the fact that the components used to crosstie the RWST are truncated from the final solution. Hence, the benefit of having additional water available is negligible and would have been small compared to the cost of installing a new, larger tank even if as a result, the recovery action were always successful.

Response to 7g: SAMA 149 was screened as not being applicable because the SAMA that was proposed was specifically a redundant Standby Liquid Control (SLC) system, which is a BWR system. However, a redundant Boron Injection System could be proposed for SPS/NAPS. Such a system would have negligible benefit since, as presented in the evaluation of SAMAs 143/144, the benefit of eliminating all ATWSs at SPS is only $20k. Therefore, this SAMA is eliminated because of its low benefit.

Response to 7h: Containment isolation failure is less significant at North Anna than at other plants because its containment is a sub-atmospheric design. Large, pre-existing leaks, for example, need not be considered because they would be noticed during the course of normal operation. Plant damage states 1 and 2 are core damage with failure of containment isolation, and their frequency sums to 1.23E-7/yr. If these PDSs are completely eliminated, the bounding benefit becomes $5,800 after being doubled for external events ($2,900 from internal events). Because of the low frequency associated with core damage and containment isolation failure, no other methods for coping with containment

isolation failure (including procedural changes) were considered. The low bounding benefit justifies that assumption as being valid, since even procedural changes would have costs greater than twice the bounding benefit.

Response to 7i: No further response required.

Response to 7j: The COMSORS concept involves placing a huge mass of glass-like material under the reactor vessel. The theory was that a molten core exiting the vessel would combine with the material to form a vitrified compound that would contain the fission products within. Besides the cost of the material (and the analysis costs), the reactor cavity would have to be expanded to be able to house it. Expanding the size of the reactor cavity alone would cost millions of dollars, which was the reason the COMSORS was screened out. In reality, this SAMA probably could have been screened as only being applicable to a new plant, since any substantial modifications to the containment cavity really are not feasible to an existing plant.

NAPS RAI No. 8:

Please clarify the benefit and estimated cost values reported in Table 4-6, and how they were obtained. Specifically:

a. It is the staff's understanding that the "Benefit" values reflect a doubling to account for external events, and that the "Estimated Cost" values also reflect a doubling (the "2 x" in the equation, "2 x benefit") to account for cost uncertainty (page 4-48). Is this correct?

b. On page 4-71 for SAMA 38 the benefit is given as $2K. We assumed that this was determined by multiplying 0.1 percent times the maximum benefit that can occur for mitigating the consequences of a core damage accident, namely $2.2 million, and rounding to the nearest thousand. If this is the case, there appears to be an inconsistency in the value for SAMA 42, a similar SAMA in the sense that the reduction in CDF is zero, where the bounding benefit is estimated to be $14K. The corresponding Reduction in Person-Rem Offsite is 1.1 percent. According to our calculation (1.1% times $2.2 million), the bounding benefit should be $24K, not $14K. Please explain this apparent discrepancy.

Dominion Response:

Response to 8a: The staff's understanding is correct.

Response to 8b: The benefit of a SAMA that affects offsite releases is not directly proportional to the offsite person-REM incurred, so the percentage reduction cannot simply be multiplied by the maximum benefit of removing all

releases. Other factors that impact the offsite benefits include averted crop damage, property damage, evacuation costs, etc.

The benefit for SAMA 38 was calculated by adjusting the Level 2 code's Decomposition Event Tree (DET) for containment failure due to hydrogen burns such that the burns would never fail containment. It turns out that failure due to hydrogen burns is a small contributor to risk, and the benefit is only $2k.

Eliminating the containment failure due to hydrogen burns does not alter containment failure due to other causes. This differs from SAMA 42, which was conservatively estimated by eliminating the release for ALL non-bypass events (i.e., all events except SGTR, ISLOCA and containment isolation failure, which would not benefit from containment scrubbing), of which releases due to hydrogen burns is only a subset. The calculated benefit came to $14K for this case. Note also that the benefit calculation is conservative in that it credited removal of all fission products from the containment prior to release, while in reality, even a fission product scrubbing mechanism would not be able to prevent noble gas releases.

The offsite consequences of containment failure events (and how they affect offsite crop damage, etc.) varies from the other release types such as containment bypass. If all release modes affected offsite risk the same way, then the direct multiplication could be performed, which would yield $24k as stated in the RAI.

NAPS RAI No. 9:

The gross electrical rating of NAPS is greater than the generic plant rating of 910 MWe. As a result, on page 4-44, you indicated that a scaling factor of 1.08 would be applied to the applicable formula. It is not apparent that this was done. Please confirm whether the indicated scaling factor was used.

Dominion Response:

Response to 9: The omission of the 1.08 scaling factor from the report was a typographical error. The factor was properly used in the spreadsheet that compiled the benefit calculations.