VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

January 18, 2002

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

Serial No.	01-720
NL&OS/ETS	R2
Docket Nos.	50-338
	50-339
License Nos.	NPF-4
	NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS CHANGES
ELIMINATION OF SEISMIC EFFECTS FROM CONTROL ROD DROP TIMES
REQUEST FOR ADDITIONAL INFORMATION

In a June 22, 2000 letter (Serial No. 00-307) and a July 26, 2001 letter (Serial No. 01-359), Virginia Electric and Power Company (Dominion) requested amendments to the Facility Operating Licenses NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively. The proposed changes would add a risk-informed license condition. The license condition will eliminate the consideration of the effects of a concurrent seismic event on the rod control cluster assembly (RCCA) drop time for the non-LOCA accident analyses. In a November 16, 2001 letter, the NRC requested additional information regarding the analysis used to develop the seismic allowance currently applied to the rod control cluster assembly. The attachment to this letter provides the requested information.

If you have any further questions or require additional information, please contact us.

Very truly yours,

Leslie N. Hartz
Vice President - Nuclear Engineering

Attachment

Commitments made in this letter: None

A075

cc: U.S. Nuclear Regulatory Commission
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SN: 01-720
Docket Nos.: 50-338/339
Subject: Proposed TS Change RAI – Elim. Of Seismic Effects from CRD Times

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

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

Acknowledged before me this 18th day of January, 2002.

My Commission Expires: March 31, 2004.

Maggie McClure
Notary Public

(SEAL)

Response to Request for Additional Information
Elimination of Seismic Effects from Control Rod Drop Times

NRC Question 1

Assuming (1) control rods are subject to the postulated seismic-related control rod drop time delay for applicable seismic events, and (2) required reactor trip signals are successful, would reactor coolant system (RCS) pressure open the pressurizer power-operated relief valve(s) and/or the safety relief valves? If so, would the valves be required to function in a steam or water environment? Provide a discussion on the associated risk of these considerations in terms of Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" guidance.

Response:

The June 22, 2000 (Serial No. 00-307) submittal contains a section that discusses defense-in-depth. In this section, the Loss of Load Accident is discussed as the limiting scenario for overpressure transient analyses. The present analysis of record (AOR) makes several conservative assumptions. Using these assumptions, pressure relief is required. However, the reactor actually trips on reactor trip following turbine trip, which occurs sooner than the high pressurizer pressure trip as assumed in the AOR. When the reactor trips on the first signal actually received, pressure relief is not required even if a delay in the rod drop time equivalent to that assumed to result from a seismic event should occur. In actuality, the reactor trip on turbine trip occurs very quickly and limits the system pressure response.

The AOR is but one of many possible scenarios that result from transient operation. As discussed above for a loss of load due to a turbine trip, the PORVs are not expected to open because the turbine trip initiates a reactor trip. For slower transients (such as rod withdrawal or a turbine runback) the transient dynamics may be such that the PORVs open before the reactor trip occurs. When the PORVs open before the reactor trips there is no impact with respect to the rod drop time issue by definition. Similarly, if the trip occurs more than a second or two before the PORV setpoint is reached, the PORVS will not open. Therefore, there is no impact from the rod drop time.

For the rod drop time to have an impact on whether or not the PORV opens, the transient dynamics must be such that the PORVs would open at or just after reactor trip. In these cases, the seismic effect could contribute to an increased likelihood of PORV demand. However, the frequency of these "smart" scenarios is small by definition because they require the near simultaneous occurrence of a PORV demand at the time of trip with a concurrent seismic event.

In the conservative AOR for the loss of load scenario only steam is relieved into the pressurizer relief tank. This result is obtained even though the PORVs are not assumed to be operable and the safety valves operate at the high end of the pressure setpoint range. More generally, it is expected that only the PORVs would be required to operate and that the small impact on pressure would not change the dynamics of the system such that water relief occurs instead of steam relief. The pressurizer steam volume is large compared to the size of the PORVs so steam relief would be expected to last much longer than a pressure rise due only to the seismic delay on control rod insertion.

From a Regulatory Guide 1.174 perspective, the risk associated with this event would be evaluated as a transient event in which the RCS integrity is potentially lost as a result of the stuck open PORV. A review of the transient event tree from the internal events model shows that the conservative assumptions regarding pressure relief are not included in this event tree. The RCS integrity function has been removed from the tree because the original IPE analysis showed that the sequence frequency for the transient with a loss of RCS integrity was about four orders of magnitude smaller than the initiating event frequency for a small LOCA. The increase in risk for those pressure increases resulting from a delay in rod drop time can be approximated as the product of the transient initiating event frequency, the RCS integrity unavailability, and the conditional probability of core damage from a small LOCA. For the proposed change the CDF increase is 1.01E-8/yr. The transient initiating event frequency is 1.95E0/yr. The RCS integrity unavailability is 1.22E-5. The small LOCA conditional core damage probability is the ratio of the small LOCA contribution to core damage frequency to the initiating event frequency (8.95E-6/yr/2.1E-2/yr). Thus, the increase in risk for a pressure increase due to a delay in rod drop from a seismic event that only impacts the internal events PRA is conservatively shown to be less than 10E-6/yr.

NRC Question 2

Please provide for review your seismic risk analysis of the failure of a turbine trip-reactor trip scenario mentioned in your July 26, 2001 submittal. Discuss the risk significance of this consideration using the guidance in R.G. 1.174.

Response:

The enclosure to the July 26, 2001 RAI response (Serial No. 00-359) contains a discussion of the seismic risk due to failure of RCS integrity following a seismically induced loss of offsite power. This discussion can be summarized as follows:

1. A seismically induced loss of offsite power (LOOP) was found to contribute to core damage frequency in the Surry Seismic PRA developed in response to GL 88-20, Supplement 4 (IPEEE).
2. The impact of overpressure events is explicitly considered in the analysis because the final node of the seismic event tree, CCDP, contains cut sets from the event tree for a LOOP from the internal events model. These cut sets represent the random failures that can occur including failures leading to a loss of RCS integrity.
3. Thus, sequence number S10 from the seismic event tree explicitly considers the "seismic risk of the failure of the turbine trip-reactor trip scenario." The sequence frequency is 3.3E-6/yr and the CCDP from random events is 0.12.

4. The top cut sets from the CCDP evaluation were provided and PORV failure to close was not among the top contributors.
5. Finally, the North Anna event tree that is equivalent to the event tree used to quantify the CCDP for the Surry Seismic PRA was provided to show that the contribution to core damage frequency from a stuck open PORV following a LOOP is negligible.

The precise increase in core damage frequency for the proposed change is difficult to quantify for two reasons. The requisite models are not available and the change is very small. The above summary of the previous submittal illustrates both points. We have used inputs from a combination of the North Anna internal events model and the Surry seismic model as the best available tools. These models indicate that the contribution to core damage from a stuck open PORV following a seismic event with a loss of offsite power is not among the top cut sets. In fact, it is not even a developed end state because the likelihood is so small. (See sequence T1P30 or T11RC from the event tree of the July 26 submittal also attached herein.) The most likely outcome of the seismic event is that the switchyard would fail following the event, but the diesels would start and power would be available to close the PORV. If one or both PORVs would fail-to-close, power would be available to close the PORV block valves.

The RCS integrity function, 1RC-11, in the enclosed event tree is the only failure for Sequence T1P30. The top fifty cut sets for this function are also enclosed. The top four cut sets from this list are those discussed in the meeting with the NRC staff in March 2001. As can be seen from the list even if one of the basic events was increased by a factor of 10 the result would still be several orders of magnitude below the dominant cut sets presented in the July 26, 2001 submittal (Attachment 4, page 9 of 10). That is, the dominant cut sets are on the order of 1E-2 while the cut sets for failure of the PORV would be on the order of 1E-5. The seismic sequence frequency is 3.3E-6/year. This number includes the seismic convolution of the hazard and the fragility along with the random failures. The dominant random failures are going to continue to be in the 1E-2 range so a change in the less dominant random cut sets from 1E-5 to 1E-4 is not going to increase the overall sequence frequency.

The guidance in Regulatory Guide 1.174 regarding risk significance recommends that an application for a change to the licensing basis should include an evaluation of the change in core damage frequency and large early release frequency as a result of the proposed change. The proposed change has been evaluated using bounding calculations. Additionally, the Surry seismic PRA model was used to infer the seismic impact assuming that the most likely result of the seismic event would be a loss of the switchyard. The estimated increase in CDF using either method is less than 10E-6/yr. This small increase in CDF in combination with a baseline CDF in the 10E-5 range means that the proposed change to the license basis would be evaluated against the acceptance criteria for Region III. In this region, small risk increases are permitted without requiring a detailed analysis of the change in total core damage frequency. Based on these conclusions the small increase in risk associated with the elimination of the seismic penalty from the control rod drop time meets the acceptance criteria in RG 1.174.

NRC Question 3

The submittal dated June 22, 2000 indicated that if accident analyses are re-performed, the rod drop time of 2.7 seconds would need to be increased. Such a change would involve concurrent reactor protection system changes (e.g., reductions in high pressurizer pressure and/or low RCS flow reactor trip setpoints). The submittal indicated this would have the potential to reduce normal operating margin and increase the potential for reactor trip events and associated plant equipment transients. Please discuss the potential magnitude of the decreases in the setpoints.

Response:

Our evaluation of the effects of using a full core of advanced fuel products (with their associated higher pressure drop and reduced thimble tube ID) concludes that measured rod drop times could potentially increase by as much as 0.5 seconds at North Anna. This increase results from the fact that advanced fuel assemblies have a higher hydraulic resistance and core pressure drop than current generation fuel. This higher pressure drop forces more flow up the RCCA guide tubes, creating more resistance to control rod insertion and, therefore, slightly delayed drop times. The seismic effect amplifies the magnitude of the estimated drop time increase.

Current measured drop times provide slightly in excess of 0.5 second of margin to the 2.7 seconds Technical Specification limit (see attached Figure). However, much of this margin is currently allocated for the seismic allowance.

If the seismic penalty is not eliminated, Dominion estimates that the safety analysis rod drop time will have to be increased from 2.7 seconds to 3.2 seconds to ensure adequate margins for BOC startup tests.

Sensitivity studies have been performed with our safety analysis models to estimate the protection setpoint adjustments that would be required to offset a 0.5 second additional delay in rod drop time. The results are summarized below.

Case 1:

Transient - Complete loss of RCS flow
Acceptance Criterion - Hot channel DNBR
Source of protection- Low RCS Flow
Current safety analysis protection setpoint - 87.0% of full flow
Required adjusted setpoint to offset 0.5 second rod drop time delay:
88.9% of full flow (+1.9%).

Case 2:

Transient - Loss of external electrical load
Acceptance Criterion - Peak RCS pressure
Source of protection- High pressurizer pressure reactor trip
Current safety analysis protection setpoint - 2381 psig
(2360 psig Technical Specification Setpoint + Instrument Uncertainty + Margin)
Required adjusted setpoint to offset 0.5 second rod drop time delay: 2355 psig (-26 psig)
(2334 psig Technical Specification Setpoint + Instrument Uncertainty + Margin)

The second case is of particular concern, since it essentially consumes (by greater than 50%) the currently allocated operating margin between the nominal pressurizer PORV setpoint (2335 psig) and the high pressure reactor trip setpoint. Therefore, the loss of margin has the potential to significantly reduce the effectiveness of the pressure control system to prevent reactor trips.

Given the demonstrably low probability of a significant seismic event, Dominion continues to believe that elimination of the seismic allowance from the control rod drop time requirement will result in an enhancement to overall reactor safety for North Anna following introduction of full cores of advanced design fuel.

Figure 2.2

NORTH ANNA UNIT 1 - CYCLE 13 STARTUP PHYSICS TESTS
ROD DROP TIME - HOT FULL FLOW CONDITIONS

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2						1.73		1.62		1.75					
3							1.68		1.70						
4				1.75		1.68				1.79		1.75			
5					1.78						1.78				
6		1.64		1.80		1.72		1.74		1.75		1.79		2.07	
7			1.73				1.67		1.70				1.76		
8		1.81				1.72				1.70				1.92	
9			1.75				1.68		1.71				1.57		
10		1.80		1.78		1.70		1.74		1.76		1.77		1.80	
11					1.85						1.76				
12				1.69		1.81				1.71		1.73			
13							1.72		1.72						
14						1.97		1.75		1.76					
15															

X.XX --> ROD DROP TIME TO DASHPOT ENTRY (SEC)

Minimum Cut Set Solution for fault tree RC100 , Serial no.= 7

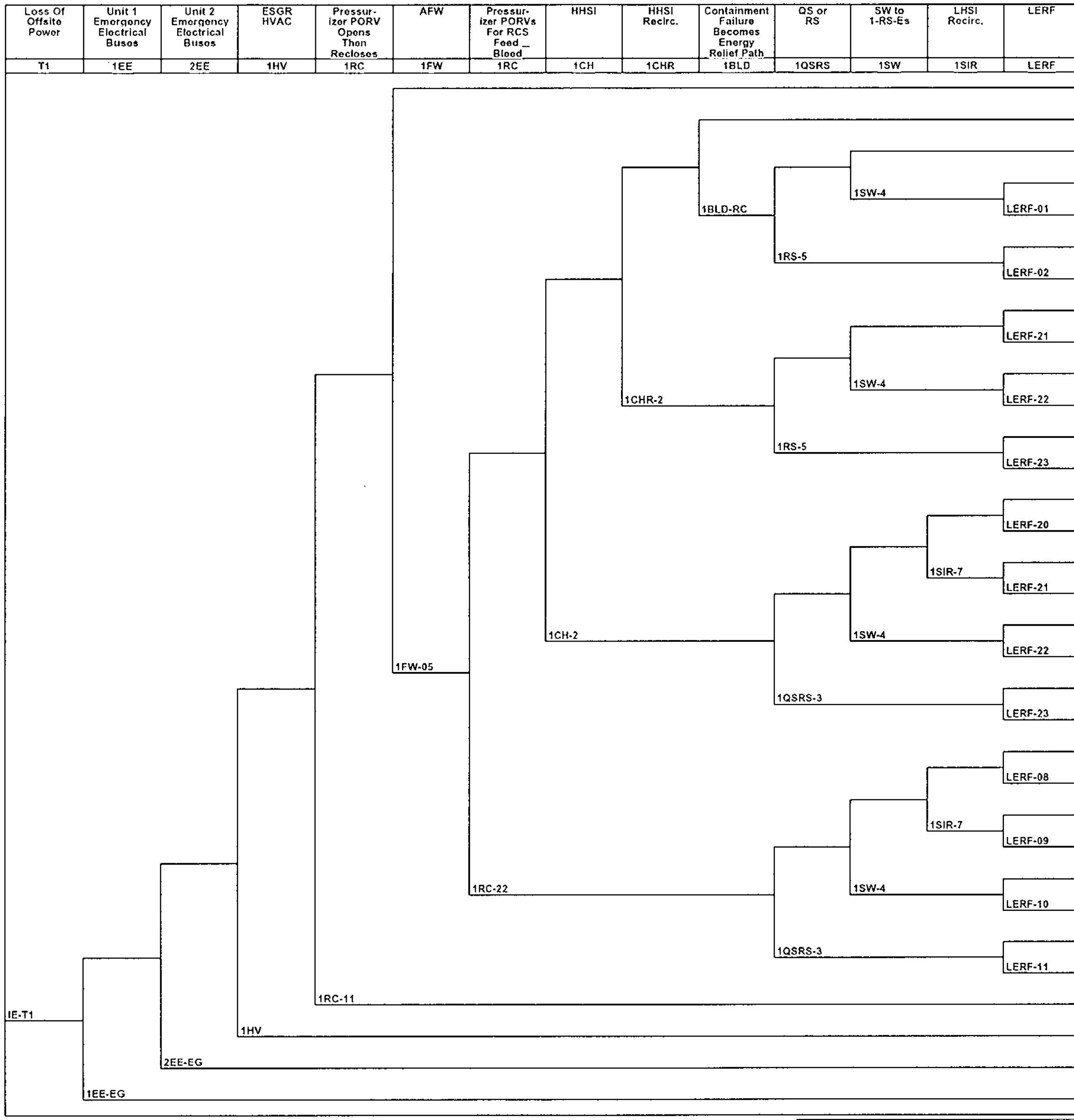
Performed : 21:19 13 Feb 1998
Cut Set Equation produced is : 1RC-11.EQN

RCS PORVs Fail To Reclose -TR NAPS Unit 1 At-Power PSA, N7B

Top event: GRC1112
Top event unavailability (r.ev. appr)= 1.224E-005
Cutoff value used = 1.00E-009
Number of Boolean Indicated Cut Sets = 5.107597E+00
Number of MCS in equation file = 65
MINIMAL CUT SETS SORTED BY UNAVAILABILITY

1.	3.126E-006	1RCPORV-T3	1RCRV--FO-1455C	HEP-1E0-22
2.	3.126E-006	1RCPORV-T3	1RCRV--FO-1456	HEP-1E0-22
3.	1.812E-006	1RCMOV-FO-1536	1RCPORV-T3	1RCRV--FO-1455C
4.	1.812E-006	1RCMOV-FO-1535	1RCPORV-T3	1RCRV--FO-1456
5.	2.489E-007	1EEEDG-TM-EEEG1J	1RCPORV-T3	1RCRV--FO-1456
		HEP-0OP6:3		
6.	2.489E-007	1EEEDG-TM-EEEG1H	1RCPORV-T3	1RCRV--FO-1455C
		HEP-0OP6:3		
7.	1.873E-007	1EEEDG-FS-1J	1RCPORV-T3	1RCRV--FO-1456
		HEP-0OP6:3		
8.	1.873E-007	1EEEDG-FS-1H	1RCPORV-T3	1RCRV--FO-1455C
		HEP-0OP6:3		
9.	1.742E-007	1EEEDG-FR-1H	1RCPORV-T3	1RCRV--FO-1455C
		HEP-0OP6:3		
10.	1.742E-007	1EEEDG-FR-1J	1RCPORV-T3	1RCRV--FO-1456
		HEP-0OP6:3		
11.	1.475E-007	1EGEDG-CC-ALL	1RCPORV-T3	1RCRV--FO-1455C
12.	1.475E-007	1EGEDG-CC-ALL	1RCPORV-T3	1RCRV--FO-1456
13.	7.613E-008	1EEEDG-TM-EEEG1H	1RCPORV-T3	1RCRV--FO-1455C
		AACEDG-FS-DG0M		
14.	7.613E-008	1EEEDG-TM-EEEG1J	1RCPORV-T3	1RCRV--FO-1456
		AACEDG-FS-DG0M		
15.	5.729E-008	1EEEDG-FS-1J	1RCPORV-T3	1RCRV--FO-1456
		AACEDG-FS-DG0M		
16.	5.729E-008	1EEEDG-FS-1H	1RCPORV-T3	1RCRV--FO-1455C
		AACEDG-FS-DG0M		
17.	5.329E-008	1EEEDG-FR-1H	1RCPORV-T3	1RCRV--FO-1455C
		AACEDG-FS-DG0M		
18.	5.329E-008	1EEEDG-FR-1J	1RCPORV-T3	1RCRV--FO-1456
		AACEDG-FS-DG0M		
19.	3.885E-008	1EEEDG-TM-EEEG1J	1RCPORV-T3	1RCRV--FO-1456
		AACEDG-TM-DG0M		
20.	3.885E-008	1EEEDG-TM-EEEG1H	1RCPORV-T3	1RCRV--FO-1455C
		AACEDG-TM-DG0M		
21.	3.506E-008	1EEEDG-TM-EEEG1H	1RCPORV-T3	1RCRV--FO-1455C
		AACEDG-FR-DG0M		
22.	3.506E-008	1EEEDG-TM-EEEG1J	1RCPORV-T3	1RCRV--FO-1456
		AACEDG-FR-DG0M		
23.	2.924E-008	1EEEDG-FS-1J	1RCPORV-T3	1RCRV--FO-1456
		AACEDG-TM-DG0M		

24.	2.924E-008	1EEEDG-FS-1H AACEDG-TM-DG0M	1RCPORV-T3	1RCRV--FO-1455C
25.	2.720E-008	1EEEDG-FR-1J AACEDG-TM-DG0M	1RCPORV-T3	1RCRV--FO-1456
26.	2.720E-008	1EEEDG-FR-1H AACEDG-TM-DG0M	1RCPORV-T3	1RCRV--FO-1455C
27.	2.639E-008	1EEEDG-FS-1J AACEDG-FR-DG0M	1RCPORV-T3	1RCRV--FO-1456
28.	2.639E-008	1EEEDG-FS-1H AACEDG-FR-DG0M	1RCPORV-T3	1RCRV--FO-1455C
29.	2.454E-008	1EEEDG-FR-1H AACEDG-FR-DG0M	1RCPORV-T3	1RCRV--FO-1455C
30.	2.454E-008	1EEEDG-FR-1J AACEDG-FR-DG0M	1RCPORV-T3	1RCRV--FO-1456
31.	5.781E-009	1EEEDG-TM-EEEG1H 1RCRV--FO-1455C	1EPBKR-FC-15F1	1RCPORV-T3
32.	5.781E-009	1EEEDG-TM-EEEG1J 1RCRV--FO-1456	1EPBKR-FC-15D1	1RCPORV-T3
33.	5.586E-009	1EEBKR-SO-15J8	1RCPORV-T3	1RCRV--FO-1456
34.	5.586E-009	1EEBKR-SO-15H8	1RCPORV-T3	1RCRV--FO-1455C
35.	5.586E-009	1EEBKR-SO-14H1-7	1RCPORV-T3	1RCRV--FO-1455C
36.	5.586E-009	1EEBKR-SO-14H1-1	1RCPORV-T3	1RCRV--FO-1455C
37.	5.586E-009	1EEBKR-SO-14J1	1RCPORV-T3	1RCRV--FO-1456
38.	5.586E-009	1EEBKR-SO-14J5	1RCPORV-T3	1RCRV--FO-1456
39.	4.351E-009	1EEEDG-FS-1H 1RCRV--FO-1455C	1EPBKR-FC-15F1	1RCPORV-T3
40.	4.351E-009	1EEEDG-FS-1J 1RCRV--FO-1456	1EPBKR-FC-15D1	1RCPORV-T3
41.	4.046E-009	1EEEDG-FR-1H 1RCRV--FO-1455C	1EPBKR-FC-15F1	1RCPORV-T3
42.	4.046E-009	1EEEDG-FR-1J 1RCRV--FO-1456	1EPBKR-FC-15D1	1RCPORV-T3
43.	3.590E-009	1EEBKR-FO-15J2 HEP-00P6:3	1RCPORV-T3	1RCRV--FO-1456
44.	3.590E-009	1EEBKR-FO-15H2 HEP-00P6:3	1RCPORV-T3	1RCRV--FO-1455C
45.	3.485E-009	1EGEDG-CC-1H-1J HEP-00P6:3	1RCPORV-T3	1RCRV--FO-1455C
46.	3.485E-009	1EGEDG-CC-1H-1J HEP-00P6:3	1RCPORV-T3	1RCRV--FO-1456
47.	3.159E-009	1EETFM-LP-1H1	1RCPORV-T3	1RCRV--FO-1455C
48.	3.159E-009	1EETFM-LP-1J	1RCPORV-T3	1RCRV--FO-1456
49.	2.012E-009	1EEBUS-LU-1H	1RCPORV-T3	1RCRV--FO-1455C
50.	2.012E-009	1EEBUS-LU-1J1	1RCPORV-T3	1RCRV--FO-1456



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
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P02	T11FW	OK	
P03	T11FW1BLD	OK	
P04	T11FW1BLD1SW	OK	
P05	T11FW1BLD1SWLERF	1	5.89E-10
P06	T11FW1BLD1QSRS	OK	
P07	T11FW1BLD1QSRSLERF	2	1.42E-10
P08	T11FW1CHR	OK	
P09	T11FW1CHRLERF	21	1.27E-07
P10	T11FW1CHR1SW	OK	
P11	T11FW1CHR1SWLERF	22	0.00E+00
P12	T11FW1CHR1QSRS	OK	
P13	T11FW1CHR1QSRSLERF	23	0.00E+00
P14	T11FW1CH	OK	
P15	T11FW1CHLERF	20	1.67E-06
P16	T11FW1CH1SIR	OK	
P17	T11FW1CH1SIRLERF	21	5.82E-09
P18	T11FW1CH1SW	OK	
P19	T11FW1CH1SWLERF	22	2.05E-09
P20	T11FW1CH1QSRS	OK	
P21	T11FW1CH1QSRSLERF	23	9.95E-08
P22	T11FW1RC	OK	
P23	T11FW1RCLERF	8	8.94E-07
P24	T11FW1RC1SIR	OK	
P25	T11FW1RC1SIRLERF	9	4.86E-09
P26	T11FW1RC1SW	OK	
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P28	T11FW1RC1QSRS	OK	
P29	T11FW1RC1QSRSLERF	11	1.37E-07
P30	T11RC	TR	1.40E-06 TTR
P31	T11HV	TR	7.08E-04 TTR
P32	T12EE	TR	3.48E-05 2TR
P33	T11EE	TR	1.36E-04 TTR

P:\APPS\NUPRA\MODEL\N7B\ETREES\T1.EVT 01/07/2002 15:19:47 WinNUPRA 2.0
Quantification Date: 2-27-98 4:46:28pm TOTAL CMF = 2.94E-006
Licensed to: VEPCO

NORTH ANNA PROBABILISTIC SAFETY ASSESSMENT
97 UPDATE MODEL (N7B)

T1: LOSS OF OFF-SITE POWER (LOOP)
(96FEB = 1EP)