Dominion
North Anna Power Station
JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

INITIAL CONDITIONS

Reactor power is 100%

Reactor Coolant System boron concentration is 1250 ppm.

In-service boric acid storage tank concentration is 14,660 ppm.

Core age is 7038 MWD/MTU

Movement of control rods is not desired.

Tavg is higher than Tref.

PCS is not available.

INITIATING CUE

You are requested to calculate the amount of boric acid required to lower Tavg by 2°F.

Record the answer below.

Gallons of boric acid _______________

Dominion
North Anna Power Station
JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

TASK

Calculate the amount of boric acid required to reduce Tavg by 2°F.

TASK STANDARDS

The correct amount of boron acid was calculated.

K/A REFERENCE:

G2.1.37 (4.3/4.6)

ALTERNATE PATH:

N/A

TASK COMPLETION TIMES

Validation Time = 15 minutes   Start Time = _______
Actual Time = _______ minutes   Stop Time = _______

PERFORMANCE EVALUATION

Rating   [ ] SATISFACTORY   [ ] UNSATISFACTORY

Candidate (Print) ________________________________

Evaluator (Print) ________________________________

Evaluator's Signature / Date ________________________________

EVALUATOR'S COMMENTS

____

____

____

Dominion
North Anna Power Station

JOB PERFORMANCE MEASURE
(Evaluation)

OPERATOR PROGRAM

READ THE APPLICABLE INSTRUCTIONS TO THE CANDIDATE

Instructions for Simulator JPMs

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

Instructions for In-Plant JPMs

I will explain the initial conditions, and state the task to be performed. All steps, including any required communications, shall be simulated for this JPM. Under no circumstances are you to operate any plant equipment. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS

Reactor power is 100%

Reactor Coolant System boron concentration is 1250 ppm.

In-service boric acid storage tank concentration is 14,660 ppm.

Core age is 7038 MWD/MTU

Movement of control rods is not desired.

Tavg is higher than Tref.

PCS is not available.

INITIATING CUE

You are requested to calculate the amount of boric acid required to lower Tavg by 2°F.

Record the answer below.

Gallons of boric acid _______________

EVALUATION METHOD

Perform if conducted in the simulator or in a laboratory (use Performance Cue(s))

Simulate if conducted in the station or on a dead simulator (use Simulation Cue(s))

TOOLS AND EQUIPMENT

Calculator

Station Curve Book (or copies of 1-SC-2.1, 1-SC-2.2, 1-SC-2.4, and 1-SC-3.4)

Station Data Book (or a copy of the ITC curve)

If requested, provide a copy of 1-OP-8.3.4

PERFORMANCE STEPS

START TIME ___________

1	Determine isothermal temperature coefficient (ITC).	Procedure Step N/A

Critical Step	SAT [ ] UNSAT [ ]

Standards	From the ITC curve in the Reactor Data Book, determines ITC at 100% power and 1250 ppm boron is -19.0 pcm/°F ($\pm$ 0.5 pcm).

Notes/Comments

If requested at any time during this task, provide a copy of 1-OP-8.3.4.

2	Determine desired reactivity insertion.	Procedure Step N/A

Critical Step	SAT [ ] UNSAT [ ]

Standards	Multiplies the amount RCS temperature is high by ITC. (2°F) x (-19.0 $\pm$ 0.5 pcm/°F) = -38 $\pm$ 1 pcm

Notes/Comments

3	Determine boron coefficient.	Procedure Step N/A

Critical Step	SAT [ ] UNSAT [ ]

Standards	From 1-SC-3.4, determines boron coefficient at 7038 MWD/MTU is -6.75 pcm/ppm ($\pm$ 0.03 pcm/ppm).

Notes/Comments

4	Determine desired change in boron concentration.	Procedure Step N/A

Critical Step	SAT [ ] UNSAT [ ]

Standards	Divides the desired reactivity insertion by the boron coefficient. (-38 $\pm$ 1 pcm) ÷ (-6.75 $\pm$ 0.03 pcm/ppm) = 5.6 $\pm$ 0.2 ppm

Notes/Comments

5	Determine gallons of boric acid required to raise RCS boron concentration 5.6 $\pm$ 0.2 ppm.	Procedure Step N/A

Critical Step	SAT [ ] UNSAT [ ]

Standards	Using 1-SC-2.2, determines the gallons of boric acid required to be added. $\text{Gal. acid} = \frac{(9262\ ft3) \times (44.779\ lbm/ft3)}{8.22\ lbm/gal} \times \ln\left(\frac{14{,}660\ ppm - 1250\ ppm}{14{,}660\ ppm - 1255.6 \pm 0.2ppm}\right)$ $= 50{,}455 \times \ln\left(\frac{13{,}410}{13404.4 \pm 0.2}\right)$ = 21 gallons of acid ($\pm$ 1gallon)

Notes/Comments

>>>>> END OF EVALUATION <<<<<

STOP TIME ___________

BLENDED FLOW

The general equations for Blended Flow boron concentration calculations are as follows:

- For a given acid flow, Boric Acid Storage Tanks (BAST) boron concentration, and Primary Grade water (PG) flow, the boron concentration in the blended flow will be as follows:

$$\mathit{Boron\ Conc\ of\ Blended\ Flow} = \frac{(\mathit{Gallons\ of\ Acid}) \times (\mathit{BAST\ Boron\ Conc})}{(\mathit{Gallons\ of\ Acid}) + (\mathit{Gallons\ of\ PG})}$$

- For a given acid flow, BAST boron concentration, and desired boron concentration in the Blended Flow, the PG flow will be as follows:

$$\mathit{Gallons\ of\ PG} = \frac{\mathit{Gallons\ of\ Acid} \times [(\mathit{BAST\ Boron\ Conc}) - (\mathit{Desired\ Blended\ Flow\ Boron\ Conc})]}{\mathit{Desired\ Blended\ Flow\ Boron\ Conc}}$$

- For a given PG flow, BAST boron concentration, and desired boron concentration in the Blended Flow, the acid flow will be as follows:

$$\mathit{Gallons\ of\ Acid} = \frac{(\mathit{Gallons\ of\ PG}) \times (\mathit{Desired\ Blended\ Flow\ Boron\ Conc})}{(\mathit{BAST\ Boron\ Conc}) - (\mathit{Desired\ Blended\ Flow\ Boron\ Conc})}$$

The following tables contain the gallons of PG needed per gallon of acid for a given BAST boron concentration and desired Blended Flow boron concentration. These values may be used to determine the total PG and acid required for a given makeup. For example, if a 2600 gallon, 2350 ppm makeup is desired and the BAST is at 14,000 ppm, the table shows that 4.96 gallons of PG will be required for each gallon of acid in the makeup. The multiplier for the PG and acid flows will be {2600 ÷ (1 + 4.96)} = 436.2. Therefore, (436.2 x 1) = 436 gallons of acid and (436.2 x 4.96) = 2164 gallons of PG will be needed for this 2600 gallon makeup.

APPROVED BY: [signature] DATE: 8/16/01

BLENDED FLOW

Gallons of PG Required per Gallon of Acid for Desired Blended Flow Boron Concentration

Desired Boron Concentration of Blended Flow (ppm)	PG Flow Required Per Gallon of 12,950 ppm Acid Flow	PG Flow Required Per Gallon of 13,500 ppm Acid Flow	PG Flow Required Per Gallon of 14,000 ppm Acid Flow	PG Flow Required Per Gallon of 14,500 ppm Acid Flow	PG Flow Required Per Gallon of 15,000 ppm Acid Flow	PG Flow Required Per Gallon of 15,500 ppm Acid Flow	PG Flow Required Per Gallon of 15,750 ppm Acid Flow
10	1294.00	1349.00	1399.00	1449.00	1499.00	1549.00	1574.00
50	258.00	269.00	279.00	289.00	299.00	309.00	314.00
100	128.50	134.00	139.00	144.00	149.00	154.00	156.50
150	85.33	89.00	92.33	95.67	99.00	102.33	104.00
200	63.75	66.50	69.00	71.50	74.00	76.50	77.75
250	50.80	53.00	55.00	57.00	59.00	61.00	62.00
300	42.17	44.00	45.67	47.33	49.00	50.67	51.50
350	36.00	37.57	39.00	40.43	41.86	43.29	44.00
400	31.38	32.75	34.00	35.25	36.50	37.75	38.38
450	27.78	29.00	30.11	31.22	32.33	33.44	34.00
500	24.90	26.00	27.00	28.00	29.00	30.00	30.50
550	22.55	23.55	24.45	25.36	26.27	27.18	27.64
600	20.58	21.50	22.33	23.17	24.00	24.83	25.25
650	18.92	19.77	20.54	21.31	22.08	22.85	23.23
700	17.50	18.29	19.00	19.71	20.43	21.14	21.50
750	16.27	17.00	17.67	18.33	19.00	19.67	20.00
800	15.19	15.88	16.50	17.13	17.75	18.38	18.69
850	14.24	14.88	15.47	16.06	16.65	17.24	17.53
900	13.39	14.00	14.56	15.11	15.67	16.22	16.50
950	12.63	13.21	13.74	14.26	14.79	15.32	15.58
1000	11.95	12.50	13.00	13.50	14.00	14.50	14.75
1050	11.33	11.86	12.33	12.81	13.29	13.76	14.00
1100	10.77	11.27	11.73	12.18	12.64	13.09	13.32
1150	10.26	10.74	11.17	11.61	12.04	12.48	12.70
1200	9.79	10.25	10.67	11.08	11.50	11.92	12.13

BLENDED FLOW

Gallons of PG Required per Gallon of Acid for Desired Blended Flow Boron Concentration

Desired Boron Concentration of Blended Flow (ppm)	PG Flow Required Per Gallon of 12,950 ppm Acid Flow	PG Flow Required Per Gallon of 13,500 ppm Acid Flow	PG Flow Required Per Gallon of 14,000 ppm Acid Flow	PG Flow Required Per Gallon of 14,500 ppm Acid Flow	PG Flow Required Per Gallon of 15,000 ppm Acid Flow	PG Flow Required Per Gallon of 15,500 ppm Acid Flow	PG Flow Required Per Gallon of 15,750 ppm Acid Flow
1200	9.79	10.25	10.67	11.08	11.50	11.92	12.13
1250	9.36	9.80	10.20	10.60	11.00	11.40	11.60
1300	8.96	9.38	9.77	10.15	10.54	10.92	11.12
1350	8.59	9.00	9.37	9.74	10.11	10.48	10.67
1400	8.25	8.64	9.00	9.36	9.71	10.07	10.25
1450	7.93	8.31	8.66	9.00	9.34	9.69	9.86
1500	7.63	8.00	8.33	8.67	9.00	9.33	9.50
1550	7.35	7.71	8.03	8.35	8.68	9.00	9.16
1600	7.09	7.44	7.75	8.06	8.38	8.69	8.84
1650	6.85	7.18	7.48	7.79	8.09	8.39	8.55
1700	6.62	6.94	7.24	7.53	7.82	8.12	8.26
1750	6.40	6.71	7.00	7.29	7.57	7.86	8.00
1800	6.19	6.50	6.78	7.06	7.33	7.61	7.75
1850	6.00	6.30	6.57	6.84	7.11	7.38	7.51
1900	5.82	6.11	6.37	6.63	6.89	7.16	7.29
1950	5.64	5.92	6.18	6.44	6.69	6.95	7.08
2000	5.48	5.75	6.00	6.25	6.50	6.75	6.88
2050	5.32	5.59	5.83	6.07	6.32	6.56	6.68
2100	5.17	5.43	5.67	5.90	6.14	6.38	6.50
2150	5.02	5.28	5.51	5.74	5.98	6.21	6.33
2200	4.89	5.14	5.36	5.59	5.82	6.05	6.16
2250	4.76	5.00	5.22	5.44	5.67	5.89	6.00
2300	4.63	4.87	5.09	5.30	5.52	5.74	5.85
2350	4.51	4.74	4.96	5.17	5.38	5.60	5.70
2400	4.40	4.63	4.83	5.04	5.25	5.46	5.56
2500	4.18	4.40	4.60	4.80	5.00	5.20	5.30
2600	3.98	4.19	4.38	4.58	4.77	4.96	5.06
2700	3.80	4.00	4.19	4.37	4.56	4.74	4.83
2800	3.63	3.82	4.00	4.18	4.36	4.54	4.63
2900	3.47	3.66	3.83	4.00	4.17	4.34	4.43
3000	3.32	3.50	3.67	3.83	4.00	4.17	4.25

BORON ADDITION

The general equation for the gallons of Boric Acid from the Boric Acid Storage Tanks (BAST) to add to the Reactor Coolant System (RCS) to increase the RCS boron concentration is as follows:

$$Gallons\ of\ Acid = \frac{(Volume\ of\ RCS)\times(Density\ of\ RCS\ Water)}{Density\ of\ Charging\ Flow}\times ln\left(\frac{BAST\ ppm - Initial\ RCS\ ppm}{BAST\ ppm - Desired\ RCS\ ppm}\right)$$

Below are typical values to use in the equation:

Density of Charging Flow: 8.22 lbm/gallon

BAST ppm: 14,350 ppm
(This is an average of the 12,950 ppm minimum value and 15,750 ppm maximum value listed in Tech Specs 3.1.2.7 and 3.1.2.8)

Volume of RCS:
9759 ft^3 with the Pressurizer Solid
8757 ft^3 with the Pressurizer level at 28.4%
9262 ft^3 with the Pressurizer level at 64.5%

Density of RCS Water:

RCS Temp	RCS Pressure	RCS Water Density
100°F	14.7 psia	61.999 lbm/ft^3
200°F	350 psia	60.176 lbm/ft^3
300°F	400 psia	57.380 lbm/ft^3
400°F	900 psia	53.877 lbm/ft^3
500°F	2000 psia	49.643 lbm/ft^3
547°F	2250 psia	47.056 lbm/ft^3
580.8°F	2250 psia	44.779 lbm/ft^3

NOTE: RCS Water above 100°F is typically a subcooled liquid and **NOT** a saturated liquid. Density values should be determined for the given RCS temperature and pressure.

APPROVED BY: [signature]

DATE: 8/16/01

BORON ADDITION

Gallons of Boric Acid to Add per Desired PPM Change in RCS Boron Concentration For Given RCS Conditions
Assumed BAST Acid Concentration = 14,350 ppm
Values Are Most Accurate For 100 ppm Borations But May Be 3% Low For 1000 ppm Borations

Initial RCS Boron Concentration (ppm)	Gallons of Acid to Add per PPM RCS = 100°F PZR Solid	Gallons of Acid to Add per PPM RCS = 200°F PZR Solid	Gallons of Acid to Add per PPM RCS = 200°F 28.4% PZR Level	Gallons of Acid to Add per PPM RCS = 300°F 28.4% PZR Level	Gallons of Acid to Add per PPM RCS = 400°F 28.4% PZR Level	Gallons of Acid to Add per PPM RCS = 500°F 28.4% PZR Level	Gallons of Acid to Add per PPM RCS = 547°F 28.4% PZR Level	Gallons of Acid to Add per PPM RCS = 580.8°F 64.5% PZR Level
10	5.156	5.004	4.491	4.282	4.021	3.704	3.511	3.534
100	5.188	5.036	4.519	4.309	4.046	3.728	3.534	3.556
200	5.226	5.072	4.551	4.340	4.075	3.755	3.559	3.582
300	5.263	5.108	4.583	4.370	4.104	3.781	3.584	3.607
400	5.301	5.145	4.617	4.402	4.133	3.808	3.610	3.634
500	5.339	5.182	4.650	4.434	4.163	3.836	3.636	3.660
600	5.378	5.220	4.684	4.466	4.194	3.864	3.663	3.687
700	5.417	5.259	4.719	4.499	4.224	3.893	3.690	3.714
800	5.458	5.297	4.754	4.533	4.256	3.921	3.717	3.741
900	5.498	5.337	4.789	4.567	4.288	3.951	3.745	3.769
1000	5.540	5.377	4.825	4.601	4.320	3.980	3.773	3.798
1100	5.582	5.418	4.861	4.636	4.353	4.011	3.802	3.826
1200	5.624	5.459	4.899	4.671	4.386	4.041	3.831	3.856
1300	5.669	5.501	4.937	4.707	4.420	4.072	3.860	3.885
1400	5.712	5.544	4.975	4.744	4.454	4.104	3.890	3.916
1500	5.757	5.587	5.014	4.781	4.489	4.136	3.921	3.946
1600	5.803	5.632	5.053	4.818	4.524	4.169	3.951	3.977
1700	5.848	5.676	5.093	4.857	4.560	4.202	3.983	4.009
1800	5.895	5.722	5.134	4.896	4.597	4.236	4.015	4.041
1900	5.942	5.768	5.176	4.935	4.634	4.270	4.047	4.073

BORON ADDITION

Gallons of Boric Acid to Add per Desired PPM Change in RCS Boron Concentration For Given RCS Conditions
Assumed BAST Acid Concentration = 14,350 ppm
Values Are Most Accurate For 100 ppm Borations But May Be 3% Low For 1000 ppm Borations

Initial RCS Boron Concentration (ppm)	Gallons of Acid to Add per PPM RCS = 100°F PZR Solid	Gallons of Acid to Add per PPM RCS = 200°F PZR Solid	Gallons of Acid to Add per PPM RCS = 200°F 28.4% PZR Level	Gallons of Acid to Add per PPM RCS = 300°F 28.4% PZR Level	Gallons of Acid to Add per PPM RCS = 400°F 28.4% PZR Level	Gallons of Acid to Add per PPM RCS = 500°F 28.4% PZR Level	Gallons of Acid to Add per PPM RCS = 547°F 28.4% PZR Level	Gallons of Acid to Add per PPM RCS = 580.8°F 64.5% PZR Level
2000	5.991	5.815	5.218	4.975	4.671	4.304	4.080	4.107
2100	6.041	5.862	5.261	5.016	4.710	4.340	4.114	4.141
2200	6.090	5.911	5.304	5.058	4.749	4.376	4.148	4.175
2300	6.141	5.961	5.349	5.100	4.788	4.412	4.182	4.209
2400	6.192	6.011	5.394	5.143	4.829	4.449	4.218	4.245
2500	6.245	6.061	5.439	5.187	4.870	4.487	4.253	4.281
2600	6.299	6.114	5.486	5.231	4.912	4.526	4.290	4.318
2700	6.353	6.166	5.533	5.276	4.954	4.565	4.327	4.355
2800	6.408	6.220	5.581	5.322	4.997	4.604	4.364	4.393
2900	6.465	6.275	5.630	5.369	5.041	4.645	4.403	4.431
3000	6.522	6.330	5.680	5.416	5.086	4.686	4.442	4.470

BORON DILUTION

The general equation for the gallons of Primary Grade Water (PG) to add to the Reactor Coolant System (RCS) to dilute the RCS boron concentration is as follows:

$$\text{Gallons of PG} = \frac{(\text{Volume of RCS}) \times (\text{Density of RCS Water})}{\text{Density of Charging Flow}} \times \ln\left(\frac{\text{Initial RCS Boron Concentration}}{\text{Desired RCS Boron Concentration}}\right)$$

Below are typical values to use in the equation:

Density of Charging Flow: 8.22 lbm/gallon

Volume of RCS:
9759 ft^3 with the Pressurizer Solid
8757 ft^3 with the Pressurizer level at 28.4%
9262 ft^3 with the Pressurizer level at 64.5%

Density of RCS Water:

RCS Temp	RCS Pressure	RCS Water Density
100°F	14.7 psia	61.999 lbm/ft^3
200°F	350 psia	60.176 lbm/ft^3
300°F	400 psia	57.380 lbm/ft^3
400°F	900 psia	53.877 lbm/ft^3
500°F	2000 psia	49.643 lbm/ft^3
547°F	2250 psia	47.056 lbm/ft^3
580.8°F	2250 psia	44.779 lbm/ft^3

NOTE: RCS Water above 100°F is typically a subcooled liquid and <u>NOT</u> a saturated liquid. Density values should be determined for the given RCS temperature and pressure.

APPROVED BY: TA Kendig

DATE: 10/8/01

BORON DILUTION

Dilution Table for RCS Temperature of 100°F, RCS Pressure of 14.7 psia
Pressurizer is Assumed to be Solid

Initial RCS Boron Concentration (ppm)	Gallons PG to Dilute 10 ppm	Gallons PG to Dilute 20ppm	Gallons PG to Dilute 50 ppm	Gallons PG to Dilute 100 ppm	Gallons PG to Dilute 200 ppm	Gallons PG to Dilute 500 ppm	Gallons PG to Dilute 1000 ppm
100	7755	16425	51020				
200	3776	7755	21175	51020			
300	2495	5078	13420	29845	80865		
400	1864	3776	9829	21175	51020		
500	1487	3005	7755	16425	37600		
600	1237	2495	6405	13420	29845	131886	
700	1059	2134	5455	11347	24767	92212	
800	926	1864	4750	9829	21175	72196	
900	822	1654	4207	8670	18498	59690	
1000	740	1487	3776	7755	16425	51020	
1100	672	1351	3424	7015	14771	44616	176501
1200	616	1237	3133	6405	13420	39674	131886
1300	568	1141	2887	5892	12296	35737	107932
1400	528	1059	2677	5455	11347	32522	92212
1500	492	988	2495	5078	10533	29845	80865
1600	461	926	2337	4750	9829	27580	72196
1700	434	871	2197	4462	9213	25638	65312
1800	410	822	2074	4207	8670	23953	59690
1900	388	779	1963	3980	8187	22478	55000
2000	369	740	1864	3776	7755	21175	51020
2100	351	704	1774	3591	7367	20016	47596
2200	335	672	1692	3424	7015	18978	44616
2300	321	643	1618	3272	6696	18043	41996
2400	307	616	1550	3133	6405	17196	39674
2500	295	591	1487	3005	6137	16425	37600
2600	284	568	1429	2887	5892	15721	35737
2700	273	547	1376	2778	5665	15074	34052
2800	263	528	1326	2677	5455	14479	32522
2900	254	509	1280	2583	5260	13930	31125
3000	246	492	1237	2495	5078	13420	29845

BORON DILUTION

Dilution Table for RCS Temperature of 200°F, RCS Pressure of 350 psia
Pressurizer is Assumed to be Solid

Initial RCS Boron Concentration (ppm)	Gallons PG to Dilute 10 ppm	Gallons PG to Dilute 20ppm	Gallons PG to Dilute 50 ppm	Gallons PG to Dilute 100 ppm	Gallons PG to Dilute 200 ppm	Gallons PG to Dilute 500 ppm	Gallons PG to Dilute 1000 ppm
100	7527	15942	49520				
200	3665	7527	20553	49520			
300	2422	4929	13026	28967	78488		
400	1809	3665	9540	20553	49520		
500	1443	2916	7527	15942	36495		
600	1201	2422	6216	13026	28967	128008	
700	1028	2071	5294	11013	24038	89501	
800	899	1809	4611	9540	20553	70073	
900	798	1606	4084	8415	17955	57935	
1000	718	1443	3665	7527	15942	49520	
1100	652	1311	3324	6809	14336	43304	171312
1200	598	1201	3041	6216	13026	38507	128008
1300	552	1108	2802	5718	11935	34686	104759
1400	512	1028	2598	5294	11013	31566	89501
1500	478	959	2422	4929	10223	28967	78488
1600	448	899	2268	4611	9540	26769	70073
1700	421	845	2133	4331	8942	24884	63391
1800	398	798	2013	4084	8415	23249	57935
1900	377	756	1905	3863	7946	21817	53383
2000	358	718	1809	3665	7527	20553	49520
2100	341	684	1722	3486	7150	19428	46197
2200	325	652	1642	3324	6809	18420	43304
2300	311	624	1570	3176	6499	17512	40761
2400	298	598	1504	3041	6216	16690	38507
2500	286	574	1443	2916	5957	15942	36495
2600	275	552	1387	2802	5718	15258	34686
2700	265	531	1335	2696	5498	14631	33051
2800	256	512	1287	2598	5294	14053	31566
2900	247	494	1243	2507	5105	13520	30210
3000	239	478	1201	2422	4929	13026	28967

BORON DILUTION

Dilution Table for RCS Temperature of 200°F, RCS Pressure of 350 psia
Pressurizer is Assumed to be at 28.4%

Initial RCS Boron Concentration (ppm)	Gallons PG to Dilute 10 ppm	Gallons PG to Dilute 20ppm	Gallons PG to Dilute 50 ppm	Gallons PG to Dilute 100 ppm	Gallons PG to Dilute 200 ppm	Gallons PG to Dilute 500 ppm	Gallons PG to Dilute 1000 ppm
100	6754	14305	44436				
200	3288	6754	18442	44436			
300	2173	4423	11688	25993	70429		
400	1623	3288	8560	18442	44436		
500	1295	2617	6754	14305	32748		
600	1077	2173	5578	11688	25993	114865	
700	922	1858	4751	9882	21570	80311	
800	806	1623	4137	8560	18442	62878	
900	716	1441	3664	7551	16111	51986	
1000	644	1295	3288	6754	14305	44436	
1100	585	1176	2982	6110	12864	38858	153722
1200	536	1077	2728	5578	11688	34554	114865
1300	495	994	2514	5131	10709	31125	94003
1400	460	922	2331	4751	9882	28325	80311
1500	429	861	2173	4423	9174	25993	70429
1600	402	806	2035	4137	8560	24021	62878
1700	378	759	1914	3886	8024	22329	56883
1800	357	716	1806	3664	7551	20862	51986
1900	338	678	1710	3466	7130	19577	47902
2000	321	644	1623	3288	6754	18442	44436
2100	306	613	1545	3128	6416	17433	41453
2200	292	585	1474	2982	6110	16529	38858
2300	279	560	1409	2850	5832	15714	36576
2400	268	536	1350	2728	5578	14976	34554
2500	257	515	1295	2617	5345	14305	32748
2600	247	495	1245	2514	5131	13692	31125
2700	238	477	1198	2419	4934	13129	29658
2800	229	460	1155	2331	4751	12611	28325
2900	221	444	1115	2250	4581	12132	27108
3000	214	429	1077	2173	4423	11688	25993

BORON DILUTION

Dilution Table for RCS Temperature of 300°F, RCS Pressure of 400 psia
Pressurizer is Assumed to be at 28.4%

Initial RCS Boron Concentration (ppm)	Gallons PG to Dilute 10 ppm	Gallons PG to Dilute 20ppm	Gallons PG to Dilute 50 ppm	Gallons PG to Dilute 100 ppm	Gallons PG to Dilute 200 ppm	Gallons PG to Dilute 500 ppm	Gallons PG to Dilute 1000 ppm
100	6441	13640	42371				
200	3135	6441	17586	42371			
300	2072	4217	11145	24785	67157		
400	1548	3135	8163	17586	42371		
500	1235	2495	6441	13640	31226		
600	1027	2072	5319	11145	24785	109528	
700	880	1772	4530	9423	20568	76580	
800	769	1548	3945	8163	17586	59957	
900	683	1374	3494	7200	15362	49571	
1000	614	1235	3135	6441	13640	42371	
1100	558	1122	2844	5826	12267	37052	146580
1200	512	1027	2602	5319	11145	32948	109528
1300	472	948	2398	4893	10212	29678	89635
1400	438	880	2223	4530	9423	27009	76580
1500	409	821	2072	4217	8748	24786	67157
1600	383	769	1941	3945	8163	22904	59957
1700	361	723	1825	3706	7651	21291	54240
1800	341	683	1722	3494	7200	19893	49571
1900	323	647	1630	3305	6799	18668	45676
2000	306	614	1548	3135	6441	17586	42371
2100	292	585	1473	2982	6118	16623	39527
2200	278	558	1405	2844	5826	15761	37052
2300	266	534	1344	2717	5561	14984	34877
2400	255	512	1287	2602	5319	14281	32948
2500	245	491	1235	2495	5097	13640	31226
2600	236	472	1187	2398	4893	13055	29678
2700	227	454	1143	2307	4705	12519	28280
2800	219	438	1101	2223	4530	12025	27009
2900	211	423	1063	2145	4368	11568	25849
3000	204	409	1027	2072	4217	11145	24785

BORON DILUTION

Dilution Table for RCS Temperature of 400°F, RCS Pressure of 900 psia
Pressurizer is Assumed to be at 28.4%

Initial RCS Boron Concentration (ppm)	Gallons PG to Dilute 10 ppm	Gallons PG to Dilute 20ppm	Gallons PG to Dilute 50 ppm	Gallons PG to Dilute 100 ppm	Gallons PG to Dilute 200 ppm	Gallons PG to Dilute 500 ppm	Gallons PG to Dilute 1000 ppm
100	6047	12808	39784				
200	2944	6047	16512	39784			
300	1946	3960	10465	23272	63057		
400	1453	2944	7664	16512	39784		
500	1160	2343	6047	12808	29320		
600	965	1946	4994	10465	23272	102841	
700	826	1664	4254	8848	19312	71904	
800	722	1453	3704	7664	16512	56296	
900	641	1290	3281	6760	14425	46545	
1000	577	1160	2944	6047	12808	39784	
1100	524	1053	2670	5470	11518	34790	137631
1200	480	965	2443	4994	10465	30937	102841
1300	443	890	2251	4594	9588	27867	84163
1400	411	826	2087	4254	8848	25360	71904
1500	384	770	1946	3960	8214	23272	63057
1600	360	722	1822	3704	7664	21506	56296
1700	339	679	1713	3480	7184	19992	50928
1800	320	641	1617	3281	6760	18678	46545
1900	303	607	1531	3103	6384	17528	42888
2000	288	577	1453	2944	6047	16512	39784
2100	274	549	1383	2800	5744	15608	37114
2200	261	524	1320	2670	5470	14799	34790
2300	250	501	1262	2551	5221	14069	32747
2400	240	480	1208	2443	4994	13409	30937
2500	230	461	1160	2343	4786	12808	29320
2600	221	443	1115	2251	4594	12258	27867
2700	213	427	1073	2166	4417	11755	26553
2800	205	411	1034	2087	4254	11291	25360
2900	198	397	998	2014	4102	10862	24271
3000	192	384	965	1946	3960	10465	23272

BORON DILUTION

Dilution Table for RCS Temperature of 500°F, RCS Pressure of 2000 psia
Pressurizer is Assumed to be 28.4%

Initial RCS Boron Concentration (ppm)	Gallons PG to Dilute 10 ppm	Gallons PG to Dilute 20ppm	Gallons PG to Dilute 50 ppm	Gallons PG to Dilute 100 ppm	Gallons PG to Dilute 200 ppm	Gallons PG to Dilute 500 ppm	Gallons PG to Dilute 1000 ppm
100	5572	11801	36658				
200	2713	5572	15214	36658			
300	1793	3649	9642	21443	58101		
400	1339	2713	7062	15214	36658		
500	1068	2159	5572	11801	27016		
600	889	1793	4602	9642	21443	94759	
700	761	1533	3919	8152	17795	66254	
800	665	1339	3413	7062	15214	51872	
900	591	1189	3023	6229	13291	42887	
1000	532	1068	2713	5572	11801	36658	
1100	483	970	2460	5041	10613	32056	126815
1200	443	889	2251	4602	9642	28505	94759
1300	408	820	2074	4233	8835	25677	77549
1400	379	761	1923	3919	8152	23367	66254
1500	354	710	1793	3649	7568	21443	58101
1600	332	665	1679	3413	7062	19816	51872
1700	312	626	1579	3206	6619	18421	46926
1800	295	591	1490	3023	6229	17210	42887
1900	279	560	1410	2859	5882	16150	39517
2000	265	532	1339	2713	5572	15214	36658
2100	252	506	1274	2580	5293	14382	34198
2200	241	483	1216	2460	5041	13636	32056
2300	230	462	1162	2351	4811	12964	30174
2400	221	443	1113	2251	4602	12355	28505
2500	212	425	1068	2159	4410	11801	27016
2600	204	408	1027	2074	4233	11295	25677
2700	196	393	989	1996	4070	10831	24466
2800	189	379	953	1923	3919	10403	23367
2900	183	366	920	1856	3779	10008	22363
3000	177	354	889	1793	3649	9642	21443

BORON DILUTION

Dilution Table for RCS Temperature of 547°F, RCS Pressure of 2250 psia
Pressurizer is Assumed to be at 28.4%

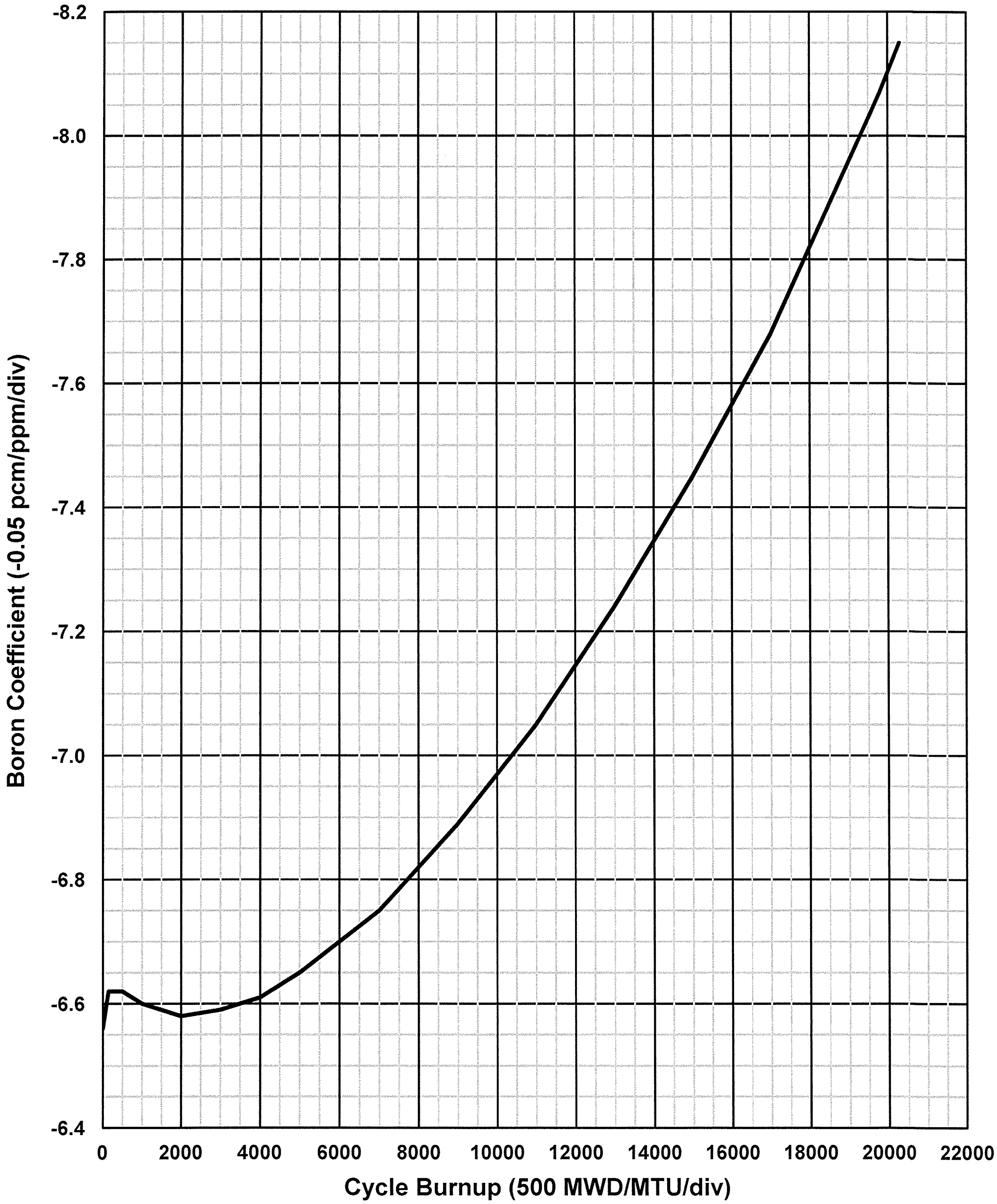
Initial RCS Boron Concentration (ppm)	Gallons PG to Dilute 10 ppm	Gallons PG to Dilute 20ppm	Gallons PG to Dilute 50 ppm	Gallons PG to Dilute 100 ppm	Gallons PG to Dilute 200 ppm	Gallons PG to Dilute 500 ppm	Gallons PG to Dilute 1000 ppm
100	5282	11186	34748				
200	2571	5282	14422	34748			
300	1699	3459	9140	20326	55074		
400	1269	2571	6694	14422	34748		
500	1013	2046	5282	11186	25608		
600	843	1699	4362	9140	20326	89821	
700	721	1453	3715	7728	16867	62801	
800	631	1269	3235	6694	14422	49169	
900	560	1127	2865	5904	12598	40652	
1000	504	1013	2571	5282	11186	34748	
1100	458	920	2332	4778	10060	30386	120207
1200	420	843	2134	4362	9140	27020	89821
1300	387	777	1966	4013	8374	24339	73508
1400	359	721	1823	3715	7728	22149	62801
1500	335	673	1699	3459	7174	20326	55073
1600	314	631	1592	3235	6694	18783	49169
1700	296	593	1497	3039	6274	17461	44481
1800	279	560	1412	2865	5904	16313	40652
1900	265	530	1337	2710	5576	15309	37458
2000	251	504	1269	2571	5282	14422	34748
2100	239	480	1208	2446	5017	13632	32415
2200	228	458	1152	2332	4778	12925	30386
2300	218	438	1102	2228	4560	12288	28601
2400	209	420	1055	2134	4362	11711	27020
2500	201	403	1013	2046	4180	11186	25608
2600	193	387	973	1966	4013	10706	24339
2700	186	373	937	1892	3858	10266	23191
2800	179	359	903	1823	3715	9861	22149
2900	173	347	872	1759	3582	9487	21198
3000	167	335	843	1699	3459	9140	20326

BORON DILUTION

Dilution Table for RCS Temperature of 580.8°F, RCS Pressure of 2250 psia
Pressurizer is Assumed to be 64.5%

Initial RCS Boron Concentration (ppm)	Gallons PG to Dilute 10 ppm	Gallons PG to Dilute 20ppm	Gallons PG to Dilute 50 ppm	Gallons PG to Dilute 100 ppm	Gallons PG to Dilute 200 ppm	Gallons PG to Dilute 500 ppm	Gallons PG to Dilute 1000 ppm
100	5316	11259	34973				
200	2588	5316	14515	34973			
300	1711	3481	9199	20458	55431		
400	1277	2588	6737	14515	34973		
500	1019	2060	5316	11259	25774		
600	848	1711	4390	9199	20458	90404	
700	726	1463	3739	7778	16977	63209	
800	635	1277	3256	6737	14515	49488	
900	564	1134	2884	5943	12680	40916	
1000	507	1019	2588	5316	11259	34973	
1100	461	926	2347	4809	10125	30583	120987
1200	422	848	2147	4390	9199	27195	90404
1300	390	782	1979	4039	8429	24496	73985
1400	362	726	1835	3739	7778	22293	63209
1500	337	677	1711	3481	7220	20458	55431
1600	316	635	1602	3256	6737	18905	49488
1700	298	597	1506	3059	6315	17574	44769
1800	281	564	1421	2884	5943	16419	40916
1900	266	534	1346	2728	5612	15408	37701
2000	253	507	1277	2588	5316	14515	34973
2100	241	483	1216	2462	5050	13721	32626
2200	230	461	1160	2347	4809	13009	30583
2300	220	441	1109	2243	4590	12368	28787
2400	211	422	1062	2147	4390	11787	27195
2500	202	405	1019	2060	4207	11259	25774
2600	194	390	980	1979	4039	10776	24496
2700	187	375	943	1904	3883	10333	23342
2800	181	362	909	1835	3739	9925	22293
2900	174	349	878	1771	3605	9548	21335
3000	168	337	848	1711	3481	9199	20458

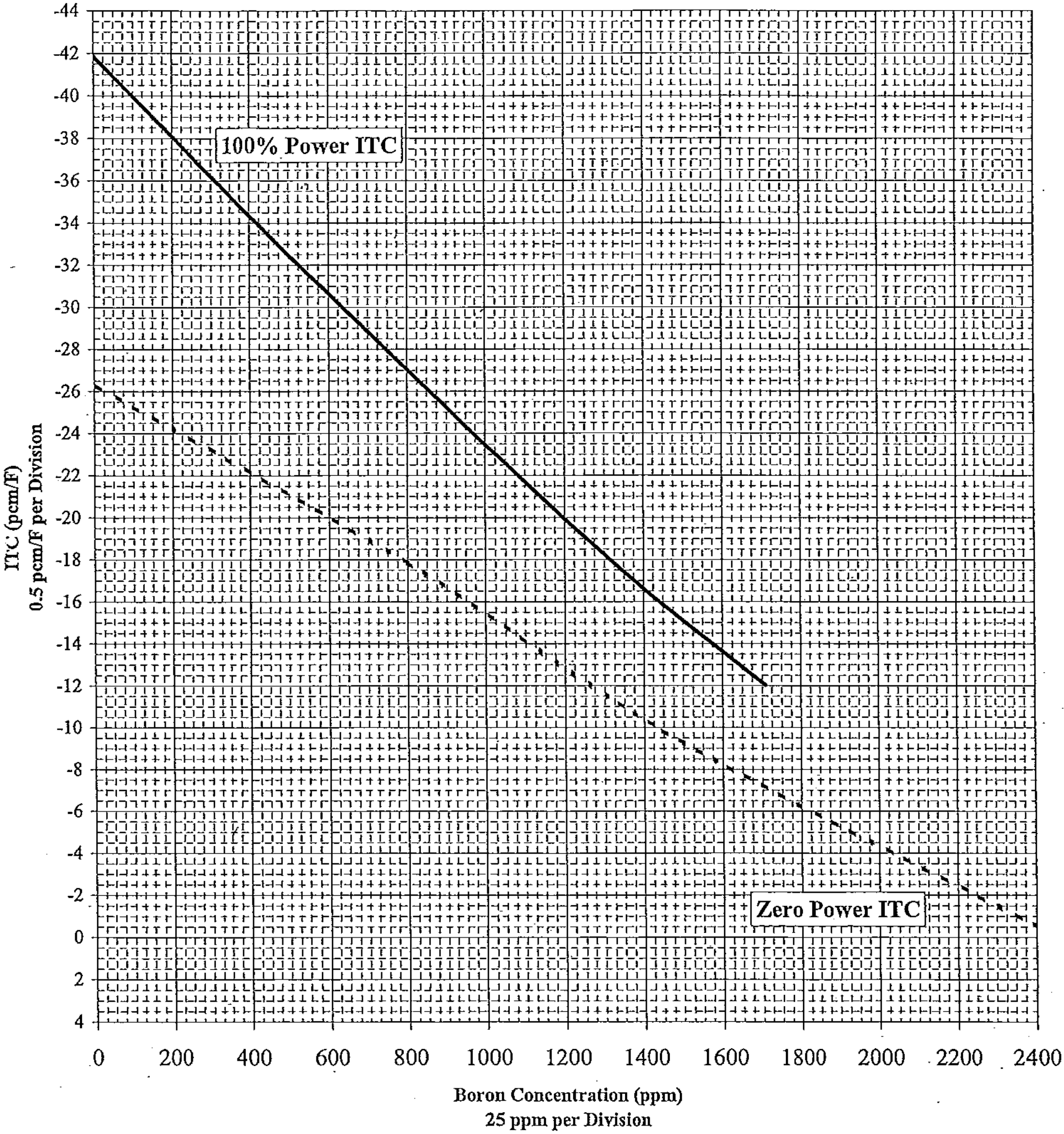
NORTH ANNA UNIT 1 - CYCLE 24
BORON COEFFICIENT VS. BURNUP
FOR ZERO PERCENT POWER



APPROVED BY: RCArrington

DATE: 9/21/13

North Anna Unit 1 Cycle 21 ITC
Data for 100% Power and Zero Power



Prepared By: [signature] Date: 3-27-09

Reviewed By: [signature] Date: 3-27-09

N1C21ITC.xls

Dominion

NORTH ANNA POWER STATION

PROCEDURE NO:	**1-OP-8.3.4**
REVISION NO:	**3**
PROCEDURE TYPE: **OPERATING PROCEDURE**	UNIT NO: **1**

PROCEDURE TITLE:

PLACING THE BLENDER IN THE BORATE MODE OF OPERATION

					REACT MGT	

REVISION SUMMARY:

FrameMaker Template Rev. 030.

Incorporated Requirements of CA176493:

- Added CA176493 to Commitment Section Step 2.4.3.
- Added new Step 5.1.13.e and 5.1.13.f to verify VCT level trend is unchanged, IF NOT THEN evaluate 1-AP-16 entrance.

Administrative Change:

- Added signature lines to Sections 3.0 and 4.0.

PROBLEMS ENCOUNTERED: ☐ NO ☐ YES **Note:** If YES, note problems in remarks.

REMARKS: ____

____ (Use back for additional remarks.)

SRO: DATE:

CONTINUOUS USE

TABLE OF CONTENTS

1.0 PURPOSE

1.1 To provide instructions for placing the blender in the borate mode of operation.

The following synopsis is designed as an aid to understanding the procedure, and is not intended to alter or take the place of the actual purpose, instructions, or text of the procedure itself.

This procedure gives detailed instructions to operate the Blender system in the Borate Mode of operation permits the addition of a pre-selected quantity of concentrated boric acid solution at a pre-selected flow rate to the Reactor Coolant System.

2.0 REFERENCES

2.1 **Source Documents**

2.1.1 UFSAR, Section 9.3, Process Auxiliaries

2.2 **Technical Specifications**

2.2.1 TRM 3.1.1

2.2.2 TRM 3.1.2

2.2.3 TRM 3.1.4

2.2.4 Tech Spec 3.1.8, Primary Grade Water Flow Path Isolation Valves

2.2.5 Tech Spec 3.9.2, Primary Grade Water Flow Path Isolation Valves - Mode 6

2.3 **Technical References**

2.3.1 11715-FM-95A, Chemical and Volume Control System

2.3.2 11715-FM-95B, Chemical and Volume Control System

2.3.3 1-LOG-2A, RCS Makeup Log

2.3.4 1-OP-3.2, Unit Shutdown From Mode 3 To Mode 4

2.3.5 DCP 94-300, Wiring Modification to Instrument Loop

2.3.6 DCP 94-155-1, Perform Wiring Mod to the Instrument Loop

2.3.7 1-OP-8.1A, Valve Checkoff - Chemical and Volume Control, Auxiliary Building

2.3.8 1-OP-8.1, Chemical and Volume Control System

2.3.9 0-OP-9A, Valve Checkoff - Primary Grade Water

2.3.10 1-OP-12.1, Sampling System

2.3.11 0-OP-8.7, Batching Boric Acid

2.3.12 DCP 02-156, PG Water to Boric Acid Blender Flow Modification / Unit 1

2.4 Commitment Documents

2.4.1 CTS Assignment 02-91-1800 Commitment 001, Technical Specifications Change Number 244

2.4.2 CTS Assignment 02-95-1202, Commitment 006, Boron Dilution Events in Pressurized Water Reactors

2.4.3 CA176493, CR390485; Daily Critical Observation 8-4-2010 |

Init **Verif**

3.0 INITIAL CONDITIONS

______ IF Unit 1 is in Mode 3, 4, 5, or 6, THEN 1-LOG-2A, RCS Makeup Log, has been initiated.

4.0 PRECAUTIONS AND LIMITATIONS

______ 4.1 To minimize the risk of an unexpected RCS boron dilution, closely monitor blender Boric Acid and/or PG flows to ensure that the desired flows are maintained. **(Reference 2.4.2)**

______ 4.2 1-GOP-8.3.4, Placing The Blender In The Borate Mode Of Operation, may be used in place of 1-OP-8.3.4.

______ 4.3 Peer checking is required for the performance of this procedure.

Init **Verif**

5.0 INSTRUCTIONS

5.1 Placing The Blender In The Borate Mode Of Operation

______ 5.1.1 Verify Initial Conditions are satisfied.

______ 5.1.2 Review Precautions and Limitations.

NOTE: The Station Curve book should be referenced, as required. IF desired, THEN Attachment 1, Reactivity Worksheet, may be used.

NOTE: Peer checking is required for the performance of this procedure.

______ 5.1.3 Determine the rate and magnitude of Boration required to obtain the desired boration.

______ 5.1.4 Obtain Unit Supervisor concurrence that the rate and magnitude is proper for current plant conditions.

______ 5.1.5 Place the BLENDER CONTROL switch in STOP.

______ 5.1.6 Place the BLENDER MODE switch in BORATE.

______ 5.1.7 Ensure 1-CH-FCV-1113A, BORIC ACID TO BLENDER, is in AUTO and OPEN.

CAUTION

To minimize the risk of an unexpected RCS boron dilution, closely monitor PG flow to ensure that the desired flow is maintained. **(Reference 2.4.2)**

______ 5.1.8 Place BLENDER CONTROL switch to START.

______ 5.1.9 Adjust 1-CH-LCV-1112C controller to allow sufficient volume in VCT as required.

______ 5.1.10 IF Boric Acid flow adjustment is required, THEN adjust 1-CH-FC-1113A, BORIC ACID TO BLENDER FLOW CONTROLLER to the desired flowrate.

______ 5.1.11 WHEN the boration is complete, THEN place the BLENDER CONTROL switch in STOP.

______ 5.1.12 WHEN desired VCT level is obtained, THEN place controller for 1-CH-LCV-1112C at 7.1.

5.1.13 IF blender flush is NOT required, THEN do the following:

a. Ensure the Blender valves are in AUTO and positioned, as follows:

______ • 1-CH-FCV-1114B, BLENDER MAKEUP TO VCT — CLOSED

______ • 1-CH-FCV-1114A, PG TO BLENDER FLOW CONTROL VALVE — CLOSED

______ • 1-CH-FCV-1113B, BLENDER MAKEUP TO CHG PP SUCTION HDR — CLOSED

______ b. Place the BLENDER MODE switch in AUTO.

______ c. Ensure 1-CH-FCV-1113A, BORIC ACID TO BLENDER VALVE, is in AUTO and OPEN.

CAUTION

To minimize the risk of an unexpected RCS boron dilution, closely monitor blender Boric Acid and/or PG flows to ensure that the desired flows are maintained. **(Reference 2.4.2)**

______ d. Place the BLENDER CONTROL switch in START.

______ e. Verify that the VCT level trend is unchanged. **(Reference 2.4.3)**

______ f. IF undesirable downward VCT level trend, THEN evaluate entering 1-AP-16, Increasing Primary Plant Leakage. **(Reference 2.4.3)**

______ 5.1.14 IF blender flush is required, THEN do one of the following:

- 1-GOP-8.3.3, Placing The Blender In The Manual Make-up Mode Of Operation
- 1-OP-8.3.3, Placing The Blender In The Manual Make-up Mode Of Operation

Completed by: ______________________________ Date: __________

Peer Check by: ______________________________ Date: __________

(Page 1 of 2)

Attachment 1
Reactivity Worksheet

NOTE: Perform Section I and/or II, then perform Section III.

I	**Power Change Only** (also accounts for temperature change if Tavg is maintained on program) (N/A if not applicable)	
1.	Rod Worth (A)	
	Present Rod Worth (pw) - from 1-SC-3.5	pw = ____________ pcm
	Desired Rod Worth (dw) - from 1-SC-3.5	dw = ____________ pcm
	Change in Rod Worth PCM (A) A = pw - dw	A = (+/-) ________ pcm
2.	Power Defect (B)	
	Present Power Defect (pd) - from 1-SC-3.8	pd = ____________ pcm
	Desired Power Defect (dd) - from 1-SC-3.8	dd = ____________ pcm
	Change in Power Defect PCM (B) B = pd - dd	B = (+/-) ________ pcm
3.	Change In PCM (C)	
	Desired Change in PCM (C) C = A + B	C = (+/-) ________ pcm

II	**Temperature Change Only** (N/A if not applicable)	
1.	Desired Temperature Change	T = (+/-) _________ °F
2.	Determine the current temperature coefficient from the Temperature Coefficient Data in the Reactor Data Book:	TC = (+/-) _____ pcm/°F
3.	Change In PCM (D)	
	Desired Change in PCM (D) D = -(T)(TC)	D = (+/-) ________ pcm

(Page 2 of 2)

Attachment 1
Reactivity Worksheet

III	Final Calculations	
1.	Final Boron Concentration (H)	
	Current Core Burnup in MWD/MTU from the Reactor Data Book (E)	E = ____________
	Current RCS C_B Coefficient (F) – from 1-SC-3.4	F = (-)________ pcm/ppm
	Desired Change in C_B (G) G = (C and/or D) ÷ F	G = ____________ ppm
	Current RCS C_B (H)	H = ____________ ppm
	Desired Final C_B (I) I = G + H	I = ____________ ppm
2.	Record the following data to be used for boration or dilution calculation:	
	Volume of RCS (vr) **Note 1**	vr = ____________ ft^3
	Density of RCS Water (dr) **Note 2**	dr = __________ lbm/ft^3
3.	For boration only - Record In-Service BAST C_B (J)	J = ____________ ppm
4.	IF desired C_B is less than current C_B, THEN calculate DILUTION as follows:	

$$PG = \frac{(vr)(dr)}{8.22\ lbm/gal} \times \ln\left(\frac{H}{I}\right)$$

$$PG = \frac{(\qquad\qquad)(\qquad\qquad)}{8.22\ lbm/gal} \times \ln\left(\frac{\qquad\qquad}{\qquad\qquad}\right)$$

PG = ____________ Gallons

5.	IF desired C_B is greater than current C_B, THEN calculate BORATION as follows:	

$$BA = \frac{(vr)(dr)}{8.22\ lbm/gal} \times \ln\left(\frac{(J-H)}{(J-I)}\right)$$

$$BA = \frac{(\qquad\qquad)(\qquad\qquad)}{8.22\ lbm/gal} \times \ln\left(\frac{(\qquad - \qquad)}{(\qquad - \qquad)}\right)$$

BA = ____________ Gallons

Note 1: IF Reactor power is 100%, THEN enter 9262 ft^3. IF Reactor power is <100%, THEN obtain volume from 1-SC-2.4.

Note 2: IF Reactor power is 100%, THEN enter 44.779 lbm / ft^3. IF Reactor power is <100%, THEN obtain density from 1-SC-2.4.

Dominion
North Anna Power Station
ADMINISTRATIVE JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

INITIAL CONDITIONS

Unit 1 entered Mode 3 for a refueling outage at 1230 on April 8.

It is now 0030 on April 15.

The last refueling outage was completed more than six months ago.

PCS computer point T0615A indicates component cooling temperature has exceeded the maximum allowable CC temperature for core off-load.

CC temperature is 88°F.

Fuel movement has been suspended.

110 fuel assemblies remain in the core.

An operator has been sent to the Aux Building basement to increase SW flow to the CC heat exchangers.

INITIATING CUE

You are requested to perform Attachment 13 of 1-OP-4.1 to determine the new maximum allowable CC temperature, and to determine if core off-load may recommence.

Record the answers below.

New Maximum Allowable CC Temperature _______________

May core off-load recommence? _______________

Once you have determined the new maximum allowable CC temperature, another operator will initiate a 1-LOG-14 to check CC temperature on PCS.

Dominion
North Anna Power Station
ADMINISTRATIVE JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

TASK

Determine the new maximum allowable CC temperature for core off-load.

Determine if core off-load may recommence.

TASK STANDARDS

Given a copy of 1-OP-4.1, Attachment 13, determine the new maximum allowable CC temperature and determine if core off-load may recommence.

K/A REFERENCE:

G2.1.42 (2.5/3.4)

ALTERNATE PATH:

N/A

TASK COMPLETION TIMES

Validation Time =	10 minutes	Start Time =	______
Actual Time =	_______ minutes	Stop Time =	_______

PERFORMANCE EVALUATION

Rating	[ ] SATISFACTORY	[ ] UNSATISFACTORY
Candidate (Print)	______________________________	
Evaluator (Print)	______________________________	
Evaluator's Signature / Date	______________________________	

EVALUATOR'S COMMENTS

____

____

____

Dominion
North Anna Power Station
ADMINISTRATIVE JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

READ THE APPLICABLE INSTRUCTIONS TO THE CANDIDATE

Instructions for Simulator JPMs

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

Instructions for In-Plant JPMs

I will explain the initial conditions, and state the task to be performed. All steps, including any required communications, shall be simulated for this JPM. Under no circumstances are you to operate any plant equipment. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS

Unit 1 entered Mode 3 for a refueling outage at 1230 on April 8.

It is now 0030 on April 15.

The last refueling outage was completed more than six months ago.

PCS computer point T0615A indicates component cooling temperature has exceeded the maximum allowable CC temperature for core off-load.

CC temperature is 88°F.

Fuel movement has been suspended.

110 fuel assemblies remain in the core.

An operator has been sent to the Aux Building basement to increase SW flow to the CC heat exchangers.

INITIATING CUE

You are requested to perform Attachment 13 of 1-OP-4.1 to determine the new maximum allowable CC temperature, and to determine if core off-load may recommence.

Record the answers below.

New Maximum Allowable CC Temperature _______________

May core off-load recommence? _______________

Once you have determined the new maximum allowable CC temperature, another operator will initiate a 1-LOG-14 to check CC temperature on PCS.

EVALUATION METHOD

Perform if conducted in the simulator or in a laboratory (use Performance Cue(s))

Simulate if conducted in the station or on a dead simulator (use Simulation Cue(s))

TOOLS AND EQUIPMENT

Copy of 1-OP-4.1 or just Attachments 11, 12, and 13

Calculator

PERFORMANCE STEPS

START TIME ___________

1	Immediately initiate actions to reduce CC temperature to within limit.	Procedure Step 1

SAT [ ]   UNSAT [ ]

Standards	Initials step (Initial Condition states that an operator is adjusting SW flow to the CC heat exchangers).

Notes/Comments:

2	Complete calculation to determine "decay time."	Procedure Step 2

Critical Step	SAT [ ] UNSAT [ ]

Standards	Enters "110" (# of assemblies in reactor) Divides 110 by 7.8 Adds 156 (Tcurrent) (6.5 days x 24 hours/day = 156 hours) Subtracts 20 Determines decay time (Tcalc) is 150.1 hours (may be rounded to 150)

Notes/Comments

3	Determine the new maximum CC temperature by plotting the "Decay Time" calculated above on the applicable attachment.	Procedure Step 3

Critical Step	SAT [ ] UNSAT [ ]

Standards	Determines Attachment 12 (Non-Back-to-Back) is applicable because it has been greater than 120 days since the previous outage. Using Attachment 12, marks 150 hours on the X-axis, then moves up to intersect the curve, then moves left to find the new maximum allowable CC temperature of 86.5°F (± 0.5°F) on the Y-axis.

Notes/Comments

4	If CC temperature is less than or equal to the new maximum CC temperature, then core off-load may recommence.	Procedure Step 4

Critical Step	SAT [ ] UNSAT [ ]

Standards	Compares actual CC temperature (88°F) to the new maximum allowable CC temperature (86.5 ± 0.5°F). Determines core off-load may NOT recommence.

Notes/Comments

5	Initiate a 1-LOG-14 to check CC temperature.	Procedure Step 5

SAT [ ]   UNSAT [ ]

Standards	Initials step (Initiating Cue).

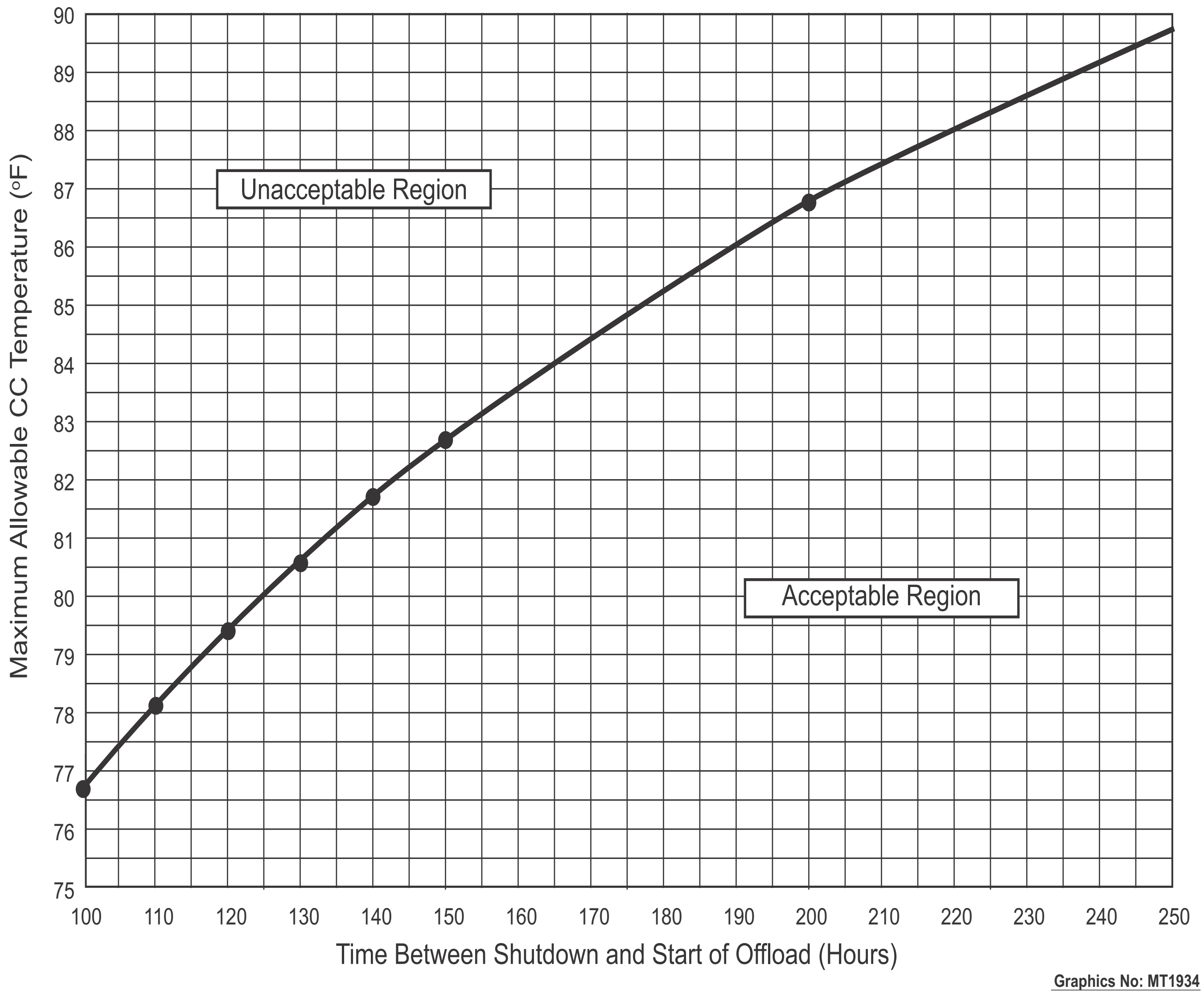
Notes/Comments

>>>>> END OF EVALUATION <<<<<

STOP TIME __________

Attachment 11

Back-To-Back Refueling CC Supply Temperature vs Decay Time



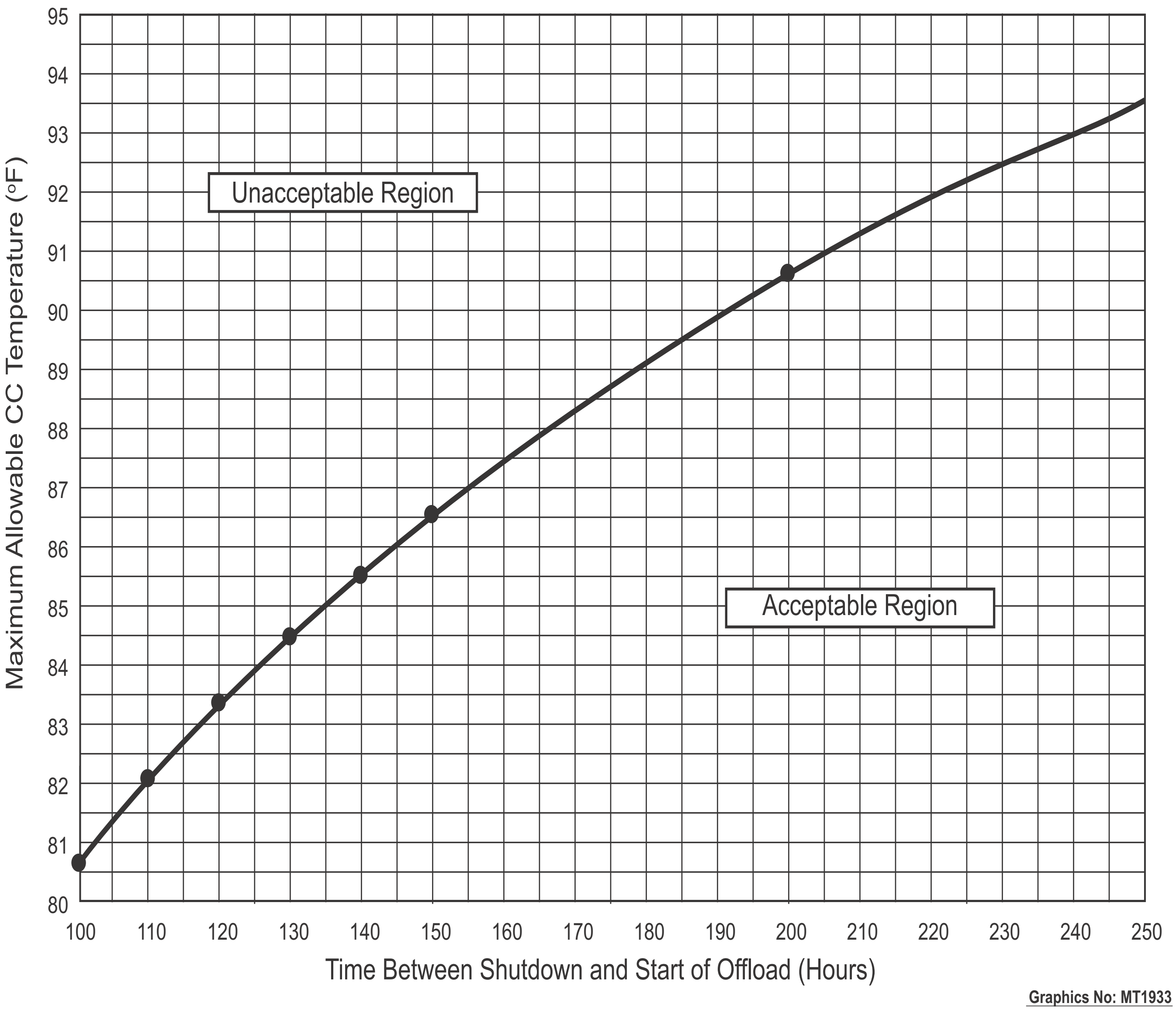
MAXIMUM CC SUPPLY TEMPERATURE VS. DECAY TIME BACK-TO-BACK OFFLOAD (1 PUMP, 2 COOLER)

NOTE: Back-to-Back outage is defined as an outage that has occurred at less than or equal to 120 days since the previous outage unit became subcritical.

(Page 1 of 1)

Attachment 12

Non Back-To-Back Refueling CC Supply Temperature vs Decay Time



MAXIMUM CC SUPPLY TEMPERATURE VS. DECAY TIME
NON-BACK-TO-BACK OFFLOAD (1 PUMP, 2 COOLER)

NOTE: Non Back-to-Back outage is defined as an outage that has occurred at greater than 120 days since the previous outage unit became subcritical.

(Page 1 of 2)

Attachment 13
Determine New Maximum CC Temperature vs Decay Time

______ 1. Immediately initiate actions to reduce CC temperature to within limit.

______ 2. Immediately complete the following calculation to determine the "Decay Time" to be used to find the new maximum CC temperature for the current time since entry into Mode 3:

$$T_{calc} = \left[\frac{\text{\# of assemblies in reactor}}{\text{7.8 assemblies per hour}} + T_{current} \right] - 20$$

T_{calc} = Time to be used on the appropriate attachment (Attachment 11 or Attachment 12) to determine the new maximum CC temperature

$T_{current}$ = Time since entry into Mode 3.

Tcurrent = ____________ (time since entry into Mode 3) (Line 1)

Assemblies = ____________ (number of assemblies in reactor vessel) (Line 2)

Line 2 divided by 7.8 = ____________ (Line 3)

Line 3 + Line 1 = ____________ (Line 4)

Line 4 - 20 = ____________ (Line 5)

Line 5 is the "Decay Time" to read on the appropriate attachment (Attachment 11 or Attachment 12)

______ 3. Determine the new maximum CC temperature by plotting the "Decay Time" calculated above on the applicable attachment listed below:

- Attachment 11, Back-To-Back Refueling CC Supply Temperature vs Decay Time
- Attachment 12, Non Back-To-Back Refueling CC Supply Temperature vs Decay Time

(Page 2 of 2)

Attachment 13

Determine New Maximum CC Temperature vs Decay Time

______ 4. IF the CC supply temperature, as indicated by Unit 1 PCS Computer Point T0615A, is less than or equal to the new maximum CC temperature, THEN core offload may recommence.

______ 5. Initiate a 1-LOG-14 to check CC temperature, as monitored by Unit 1 PCS Computer Point T0615A, is less than or equal to the new maximum CC temperature in accordance with TRM TSR 3.9.7.1. Step 4.57 provides actions to be taken in the event CC temperature increases above the new maximum CC temperature.

Dominion
North Anna Power Station
ADMINISTRATIVE JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

INITIAL CONDITIONS

Both units are in mode 1.

Annunciator H-D1, GENERATOR BREAKER TROUBLE, was received 10 minutes ago on unit 1.

The turbine building operator was sent to investigate.

The BLOCK TRIP annunciator is lit at the local G12 alarm panel.

INITIATING CUE

You are requested to perform 0-PT-80.

Dominion
North Anna Power Station
ADMINISTRATIVE JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

TASK

Complete 0-PT-80.

TASK STANDARDS

Given a copy of 0-PT-80, determine that the 1H emergency bus is not operable.

K/A REFERENCE:

G2.2.12 (3.4/4.1)

ALTERNATE PATH:

N/A

TASK COMPLETION TIMES

Validation Time = 15 minutes   Start Time = ______

Actual Time = _______ minutes   Stop Time = _______

PERFORMANCE EVALUATION

Rating   [ ] SATISFACTORY   [ ] UNSATISFACTORY

Candidate (Print) ____________________

Evaluator (Print) ____________________

Evaluator's Signature / Date ____________________

EVALUATOR'S COMMENTS

____

____

____

Dominion
North Anna Power Station
ADMINISTRATIVE JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

READ THE APPLICABLE INSTRUCTIONS TO THE CANDIDATE

Instructions for Simulator JPMs

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

Instructions for In-Plant JPMs

I will explain the initial conditions, and state the task to be performed. All steps, including any required communications, shall be simulated for this JPM. Under no circumstances are you to operate any plant equipment. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS

Both units are in mode 1.

Annunciator H-D1, GENERATOR BREAKER TROUBLE, was received 10 minutes ago on unit 1.

The turbine building operator was sent to investigate.

The BLOCK TRIP annunciator is lit at the local G12 alarm panel.

INITIATING CUE

You are requested to perform 0-PT-80.

EVALUATION METHOD

Perform if conducted in the simulator or in a laboratory (use Performance Cue(s))

Simulate if conducted in the station or on a dead simulator (use Simulation Cue(s))

TOOLS AND EQUIPMENT

0-PT-80, with Attachment 2 completed

1-AR-H-D1

1-AR-30, pages 3 - 5

PERFORMANCE STEPS

START TIME ________________

1	Review Initial Conditions / Precautions and Limitations.	Procedure Step 3.0 to 4.4
		SAT [ ] UNSAT [ ]

Standards	Reads and initials each step.

Notes/Comments

The "G12 Operable" definition, provided in step 4.4, will be needed to accurately complete attachment 1.

2	If 1H emergency bus frequency meter is inoperable, then do the following:	Procedure Step 6.1
		SAT [ ] UNSAT [ ]

Performance Cue(s)	If asked: "The 1H frequency meter is operable."

Standards	Checks 1H emergency bus frequency meter indicating properly. N/A's steps 6.1.1, 6.1.2, and 6.1.3.

Notes/Comments

This step was added due to the 1H frequency meter reading higher than actual bus frequency. The meter was recently replaced.

3	Complete the following attachments by marking the appropriate box as the required condition is verified. If the condition cannot be verified, then do not mark the box. When an entire branch can be marked, then that condition is sat.	Procedure Step 6.2

	SAT [ ] UNSAT [ ]

Standards	Reads step and goes to attachments 1, 2, and 3.

Notes/Comments

Because there is a bullet before each of these attachments, they may be performed in any order.

4	Attachment 1, 1H flow chart.	Procedure Step Attachment 1

Critical Step	SAT [ ] UNSAT [ ]

Performance Cue(s)	If the operator stops and informs the SRO that 1H is not operable: "Continue with the procedure."

Standards	Records frequency. Marks boxes for 1H on alternate feed (lower branch). Records voltage. Does NOT mark boxes for G12 (G12 is closed and not operable). Does NOT mark box for 1H operable.

Notes/Comments

The "critical step" is NOT marking the boxes for "G12 Operable", "G12 Open", and "1H Operable".

G12 is not operable because a block trip condition exists (step 4.4).

5	Attachment 1, 1J flow chart.	Procedure Step Attachment 1

	SAT [ ] UNSAT [ ]

Standards	Records frequency and voltage. Verifies each condition in a branch and marks the appropriate boxes to complete an entire branch. Determines 1J operable.

Notes/Comments

6	Attachment 2	Procedure Step Attachment 2

SAT [ ]   UNSAT [ ]

Performance Cue(s)	Hand a completed copy of Attachment 2 to the candidate and state: “Attachment 2 has been completed.”

Standards	Goes to Attachment 2.

Notes/Comments A completed copy of Attachment 2 must be provided because the simulator does not model unit 2.

7	Attachment 3, T342 flow chart.	Procedure Step Attachment 3

SAT [ ]   UNSAT [ ]

Standards	Verifies each condition in a branch and marks the appropriate box to complete an entire branch. Determines T342 sat.

Notes/Comments

8	Attachment 3, 500 KV lines flow chart.	Procedure Step Attachment 3

SAT [ ]   UNSAT [ ]

Standards	Verifies each condition in a branch and marks the appropriate box to complete an entire branch for at least 2 of the 3 transmission lines. Determines 500 KV line conditions sat.

Notes/Comments

9	Attachment 3, Bus 5 flow chart.	Procedure Step Attachment 3

SAT [ ]   UNSAT [ ]

Standards	Verifies each condition in a branch and marks appropriate box to complete an entire branch for bus 5. Determines 230 KV to 34.5 KV bus 5 line conditions sat.

Notes/Comments

10	Returns to step 6.2	Procedure Step 6.2

SAT [ ]   UNSAT [ ]

Standards	Initials steps for completing attachments 1, 2, and 3.

Notes/Comments

11	Follow-On	Procedure Step 7.0

Critical Step	SAT [ ] UNSAT [ ]

Standards	Operator determines that 1H emergency bus is NOT operable.

Performance Cue(s)	When the operator reports 1H emergency bus is NOT operable: “Assume another operator will complete this task.”

Notes/Comments 1H emergency bus is inoperable due to being on the alternate feed with G12 inoperable and closed.

>>>>> END OF EVALUATION <<<<<

STOP TIME ___________

SIMULATOR, LABORATORY, IN--PLANT SETUP
(If Required)

SIMULATOR SETUP

JOB PERFORMANCE MEASURE

TASK

Complete 0-PT-80.

CHECKLIST

_____ Ensure “B” MFP in PTL

_____

Dominion

NORTH ANNA POWER STATION

PROCEDURE NO: **0-PT-80**	REVISION NO: **20**
PROCEDURE TYPE: **OPERATIONS PERIODIC TEST**	UNIT NO: **1 & 2**

PROCEDURE TITLE:

AC SOURCES OPERABILITY VERIFICATION

TEST FREQUENCY:	UNIT CONDITIONS REQUIRING TEST:
Weekly surveillance in Modes 1 through 6, during movement of recently irradiated fuel assemblies, OR within 1 hour after entering Action of Tech Spec 3.8.1, and every 8 hours thereafter	All Modes or if EDG or normal offsite supply to emergency bus is inoperable

SPECIAL CONDITIONS: None

SURV REQ						

REVISION SUMMARY: FrameMaker Template Rev. 030.

Incorporated EPAR {P1} adding Step 6.1 to address 1H Emergency Bus frequency meter being inoperable and recording on Attachment 1. Changed Step 6.1.1 to delete obtaining frequency using a multi-meter and address obtaining 1H Emergency Bus frequency from other Emergency Buses. Added Step 6.1.2 to obtain frequency using a 1H EDG rpm calculation, if 1H EDG is the sole source of power.

REASON FOR TEST (CHECK APPROPRIATE BOX):
☐ Surveillance ☐ Post-Maintenance Work Order Number: ____________________

TEST PERFORMED BY (SIGNATURE):	DATE STARTED:	DATE COMPLETED:

TEST RESULT (CHECK APPROPRIATE BOX):
☐ Satisfactory ☐ Unsatisfactory ☐ Partial

CONDITION REPORT NUMBER(S) AND DATE:

THE FOLLOWING PROBLEM(S) WERE ENCOUNTERED AND CORRECTIVE ACTIONS TAKEN:

____

____

____

____

____ (Use back for additional remarks.)

COGNIZANT SUPERVISOR or DESIGNEE:	DATE:
ADDITIONAL REVIEWS:	DATE:

CONTINUOUS USE

TABLE OF CONTENTS

1.0 PURPOSE

To provide instructions for verifying two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Distribution System as required by Surveillance Requirements 3.8.1.1 and 3.8.2.1.

2.0 REFERENCES

2.1 Source Documents

None

2.2 Technical Specifications

2.2.1 Tech Spec 3.8.1

2.2.2 Tech Spec 3.8.2

2.2.3 SR 3.8.1.1

2.2.4 SR 3.8.2.1

2.3 Technical References

2.3.1 11715-FE-1BB, One Line Diagram Electrical Distribution System

2.3.2 DCP-88-05, GDC-17 Third Station Reserve Transformer

2.3.3 Memo from K. S. Berger to J. A. Stall, dated 5-5-1989, North Anna Power Station Operation of 230-36.5KV Transformer

2.3.4 ET CEE 02-0005, Rev. 3, Voltage Specification For The Emergency AC Buses, North Anna Power Station, Unit 1 and 2

2.3.5 DCP 03-001, Installation Of 500/230 KV Transformer 6 and 230 KV Bus Rearrangement Project

2.3.6 DCP 05-143, Relocation of Switchyard Breaker H502, and Replacement of Switchyard Breakers G1TH5 and G102 / NAPS / Unit 1

2.3.7 DCP 06-005, Rebuild of Switchyard 34.5 KV Buses #3 and #4 / North Anna / 1 & 2

2.3.8 0-GOP-5.8, LCO Tracking

2.3.9 DCP NA-11-00004, Installation of Indicating Lights on Panel 1-EI-CB-09 for 230 KV Switchyard Breaker Bay Addition in Support of a Variable Shunt Reactor

2.4 **Commitment Documents**

2.4.1 Plant Issue N-2006-3520, During a post-examination review of a Technical Specification Mode Change Evaluation JPM administered to the License Class SRO Candidates by the NRC Examination Team on 6/05/2006, a potential discrepancy was identified between the TS 3.8.1 (AC Sources - Operating) Bases and 0-PT-80 (AC Sources Operability Verification)

2.4.2 CR020405, Procedural weakness In Maintaining Electrical Source TS Operability

2.4.3 CA135119, CR332963, Permanent procedures need to be enhanced for loss of B and C RSST

2.4.4 CA136920, CR332636: NANN - The B RSST tripped and locked out on a B phase fault

2.4.5 CA180447, Review Needed for Offsite Power Source Operability Verification Procedures

Init **Verif**

3.0 INITIAL CONDITIONS

______ Notify the SRO of this test.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 Comply with the following guidelines when marking steps N/A:

______ • IF the conditional requirements of a step do not require the action to be performed, THEN mark the step N/A.

______ • IF this test is being performed as a Partial PT or Post-Maintenance Test, THEN mark inappropriate steps N/A.

______ • IF any other step is marked N/A, THEN have the SRO approve the N/A and submit a Procedure Action Request (PAR).

4.2 The following Tech Specs apply:

______ • Tech Spec 3.8.1, AC Sources — Operating

______ • Tech Spec 3.8.2, AC Sources — Shutdown

______ 4.3 This procedure is not written to verify separation of offsite power when both 1H and 1J Emergency Busses are on alternate feed. IF required to place both 1H and 1J Emergency Busses on alternate feed, THEN this procedure will require revision to verify that separation of offsite power to 1H and 1J Emergency Busses would be maintained in the event of a dual Unit trip. **(Reference 2.4.1)**

4.4 To aid in the evaluation of certain items contained in Attachment 1, the following definitions are provided below:

______ • GI2 Operable - means that the G12 Breaker is capable of automatic and manual tripping when required (e.g. no Block Trip condition).

______ • GI2 Open - means that the G12 Breaker is open on all three poles (phases), or the Unit 1 Main Generator Links are removed.

______ • 25B1 Operable - means that the breaker is racked to connect and has control power and circuitry available to support its automatic closure feature to transfer its associated B Station Service Bus power supply to the B Reserve Station Service Transformer when required.

______ • B RSST Operable - means that the transformer is in-service and capable of providing power to its designed loads. The associated Load Tap Changer must also be in-service and capable of automatic adjustment of voltage within design assumptions when required.

5.0 SPECIAL TOOLS AND EQUIPMENT

None

6.0 INSTRUCTIONS

6.1 IF 1H Emergency Bus frequency meter is inoperable, THEN do the following:

6.1.1 IF 1H Emergency Bus is being supplied from a RSST, THEN obtain 1H Emergency Bus frequency from one of the following Emergency Buses being supplied from a RSST. Mark buses not used N/A:

______ • 1J Emergency Bus Frequency meter

______ • 2H Emergency Bus Frequency meter

______ • 2J Emergency Bus Frequency meter

NOTE: 1H Emergency Bus Frequency to EDG RPM correlation:

- 59.5 Hz = 892.5 RPM
- 60.0 Hz = 900 RPM
- 60.5 Hz = 907.5 RPM

6.1.2 IF 1H EDG is the sole source of power to 1H Emergency Bus, THEN calculate 1H Emergency Bus frequency, as follows:

______ ______ IV

(____________ x 8) ÷ 120 = ______________ Hz
1H EDG RPM — 1H Bus Frequency

______ 6.1.3 Record 1H Emergency Bus frequency on Attachment 1, Unit 1 Emergency Buses.

NOTE: Attachment 1, Unit 1 Emergency Buses, Attachment 2, Unit 2 Emergency Buses, and Attachment 3, Switchyard, may be performed in any order and in conjunction with other attachments since Section 6.0 only gathers information, therefore order is not important.

NOTE: Verifying that a breaker is racked to test will satisfy the requirement for an open breaker.

NOTE: Verifying that a breakers' disconnects are open will satisfy the requirement for an open breaker.

6.2 Complete the following attachments by marking the appropriate box as the required condition is verified. IF the condition cannot be verified, THEN do not mark the box. WHEN an entire branch can be marked, THEN that condition is SAT.

______ • Attachment 1, Unit 1 Emergency Buses

______ • Attachment 2, Unit 2 Emergency Buses

______ • Attachment 3, Switchyard

7.0 FOLLOW-ON

7.1 **Acceptance Criteria**

NOTE: Utilize Attachment 1, Unit 1 Emergency Buses, Attachment 2, Unit 2 Emergency Buses, and Attachment 3, Switchyard to determine the following conditions.

______ 7.1.1 IF Unit 1 is in Mode 1 - 4, THEN the 1H Emergency Bus AND 1J Emergency Bus are operable.

______ 7.1.2 IF Unit 1 is in Mode 5, 6, Defueled, or during movement of recently irradiated fuel assemblies, THEN at least 1H Emergency Bus OR 1J Emergency Bus is operable.

______ 7.1.3 IF Unit 2 is in Mode 1 - 4, THEN the 2H Emergency Bus AND 2J Emergency Bus are operable.

______ 7.1.4 IF Unit 2 is in Mode 5, 6, Defueled, or during movement of recently irradiated fuel assemblies, THEN at least 2H Emergency Bus OR 2J Emergency Bus is operable.

______ 7.1.5 T342 conditions are SAT.

______ 7.1.6 500 KV line conditions are SAT.

______ 7.1.7 230 KV to 34.5 KV Bus 5 line conditions are SAT.

______ 7.1.8 4160 Volt 2J bus off site power supply is NOT aligned to A RSST.

7.2 **Follow-On Tasks**

______ 7.2.1 IF the requirements of Steps 7.1.1 through 7.1.8 are satisfied, THEN mark the cover sheet SAT.

7.2.2 IF the requirements of Step 7.1.1, 7.1.2, 7.1.3, 7.1.4, OR 7.1.5 are NOT satisfied, THEN do the following:

______ a. Declare the affected Bus inoperable. IF Step 7.1.5 is NOT satisfied, THEN both Busses are inoperable.

______ b. Enter the applicable Action of Tech Spec 3.8.1 or Tech Spec 3.8.2.

______ c. Record the inoperable Bus on the cover sheet.

______ d. Mark the procedure cover sheet UNSAT.

7.2.3 IF the requirement of Steps 7.1.6 or 7.1.7 are NOT satisfied, THEN do the following:

______ a. Notify the SRO to determine, with Engineering Department assistance, if the necessary off-site power supply criteria is met.

______ b. IF the SRO determines off-site power supply criteria is met, THEN mark the cover sheet SAT.

______ c. IF the SRO determines off-site power supply criteria is NOT met, THEN mark the cover sheet UNSAT.

______ U1 SRO ______ U2 SRO

7.2.4 IF any Emergency Bus (Unit 1 or Unit 2) is inoperable, THEN evaluate any Unit in Mode 1 - 4 for Operable Shared Equipment. Refer to 0-GOP-5.8, LCO Tracking, as necessary. **(Reference 2.4.2)**

7.2.5 IF the requirement of Step 7.1.8 is NOT satisfied, THEN do the following:

______ a. Ensure the 72 hour Action Statement of Tech Spec 3.8.1 was entered during performance of 1-MOP-26.79, C RSS Transformer and F Transfer Bus, which established this alternate lineup.

______ b. Mark the cover sheet UNSAT; the off-site power supply criteria is NOT met.

______ 7.2.6 IF this procedure is NOT being performed for a weekly surveillance, THEN note the reason for procedure performance on Cover Page.

7.3 **Completion Notification**

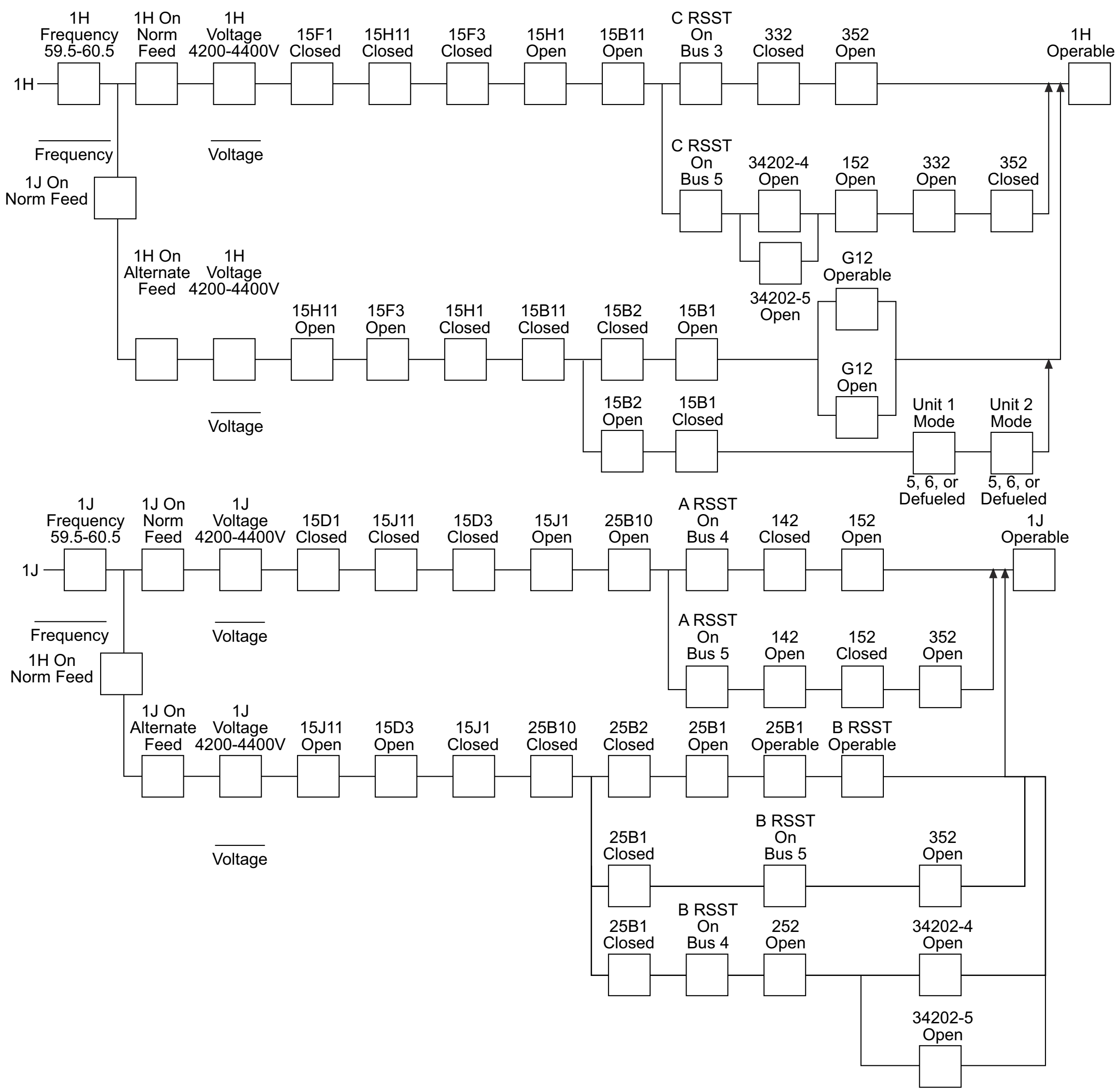
______ Notify the SRO that this test is complete.

Completed by: ______________________________ Date: __________

(Page 1 of 1)

Attachment 1

Unit 1 Emergency Buses

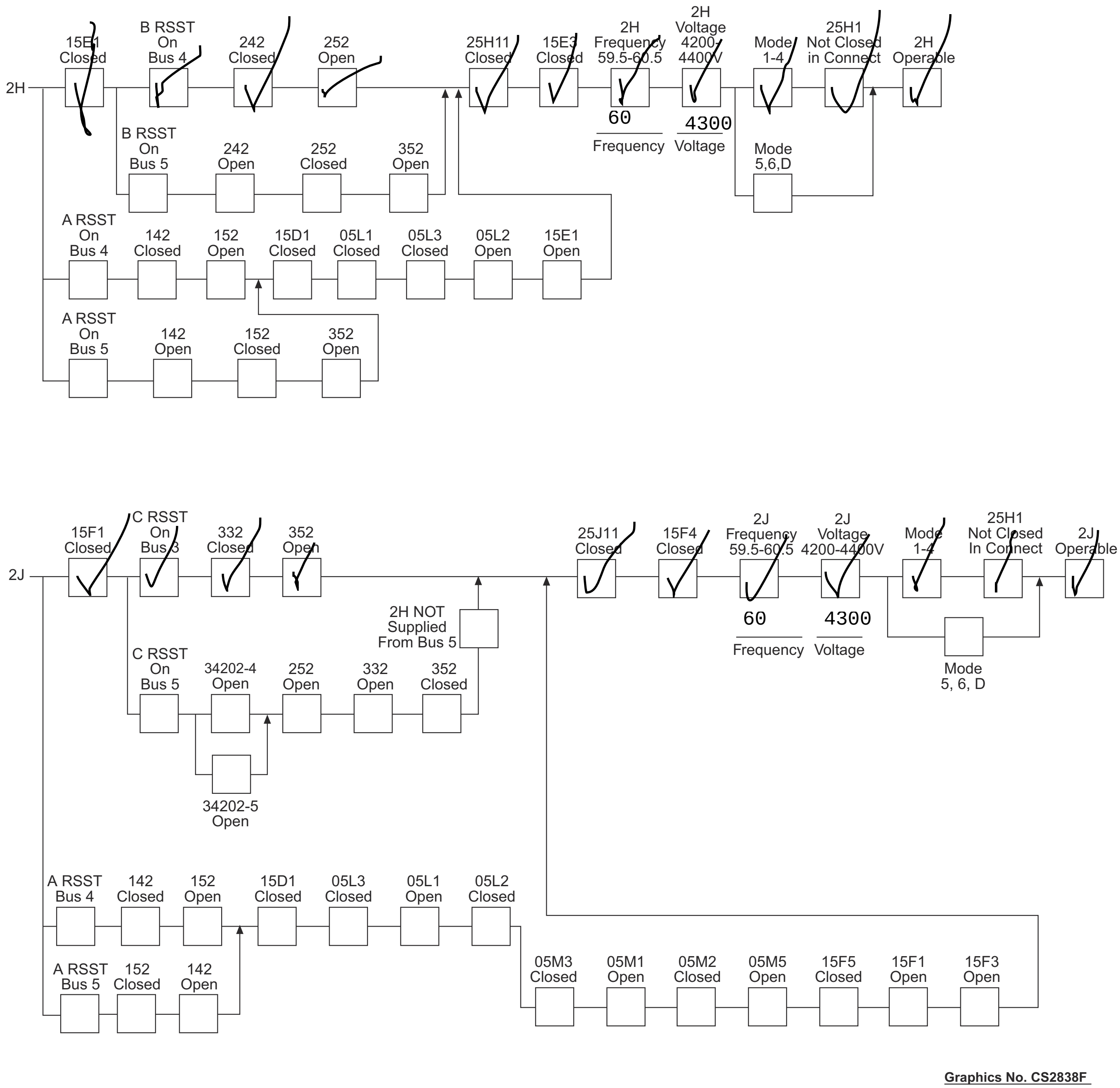


UNIT 1 EMERGENCY BUSES

Graphics No. CS2837G

(Page 1 of 1)

Attachment 2
Unit 2 Emergency Buses



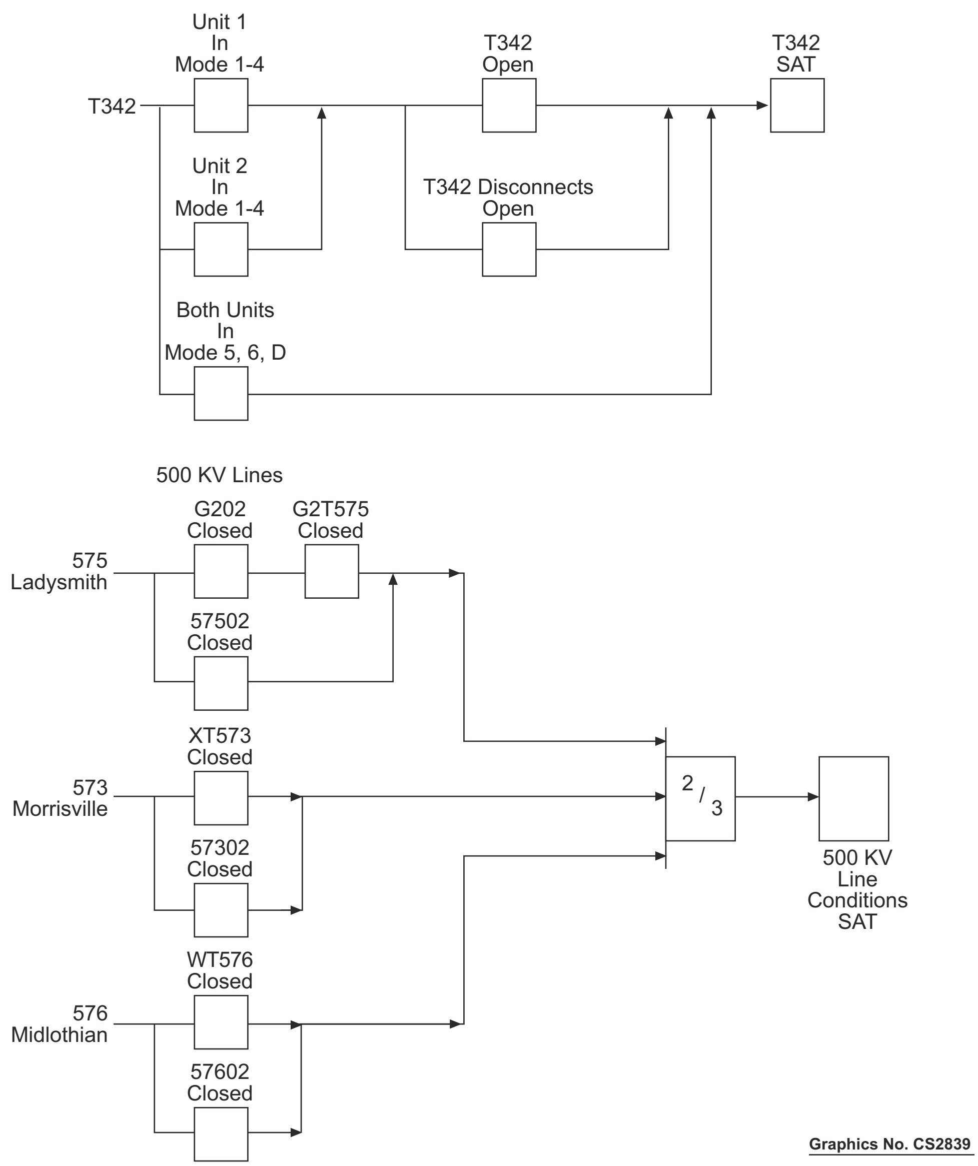
UNIT 2 EMERGENCY BUSES

NOTE: <u>IF</u> "A" RSST is supplying 2J Bus, <u>THEN</u> Acceptance Criteria 7.1.8 is applicable.

(Page 1 of 2)

Attachment 3

Switchyard

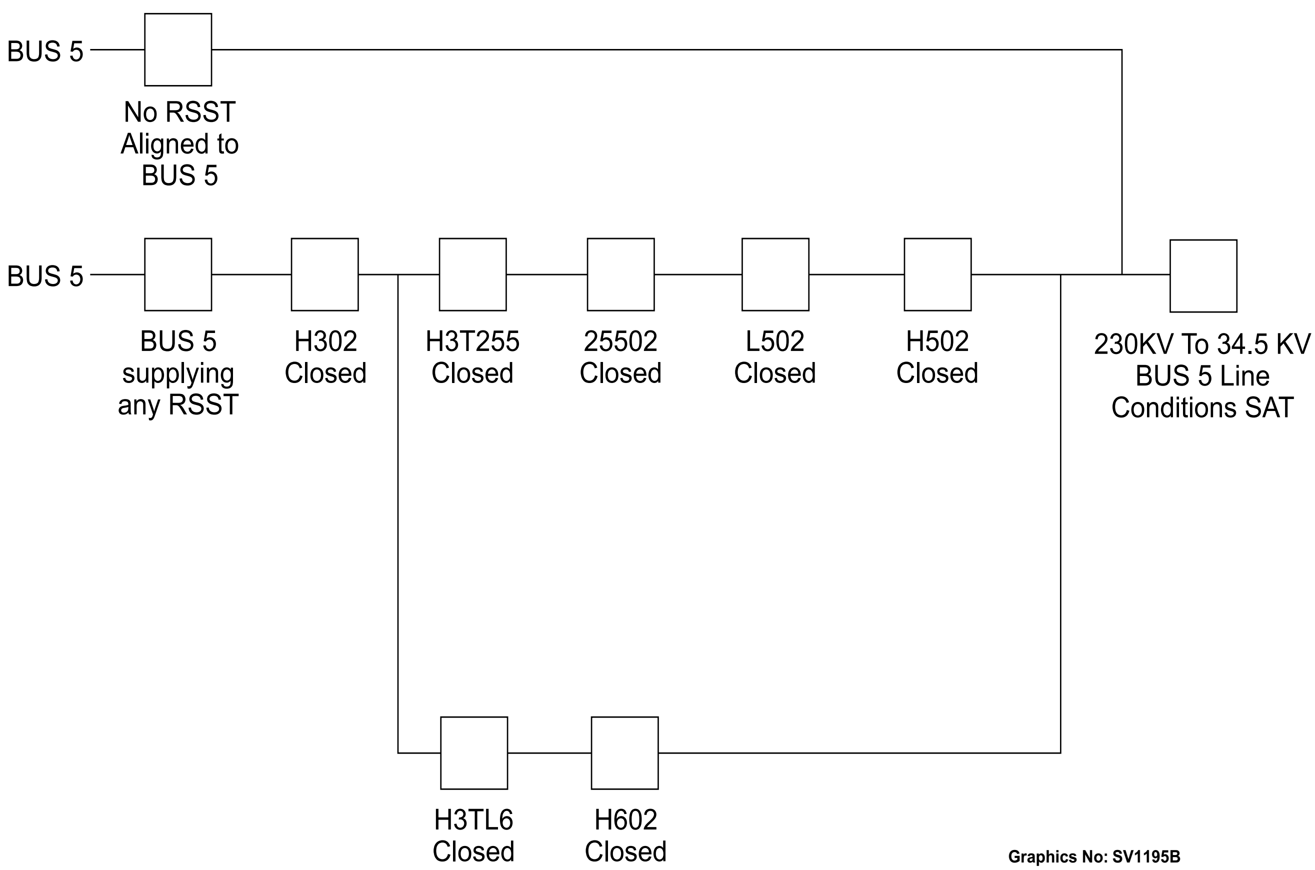


SWITCHYARD

(Page 2 of 2)

Attachment 3
Switchyard

NOTE: Breakers 25502 AND L502 may NOT have indication on the Back Boards Switchyard Mimic Panel. Breaker position may be determined by System Operations OR sending an Operator to switch yard to locally check breaker position.



SWITCHYARD

(Page 1 of 3)

Attachment 1

Window A1 - Block Trip

Unit 1 G12 Breaker

1100 **BLOCK TRIP**	LOW PRESS < 313 psig RESET 323 psig HI PRESS < 2050 psig RESET 2080 psig

NOTE: Breaker will not open.

1. **Probable Cause**

    1.1 Low air pressure on HP or LP system due to rapid cycling of breaker due to fault or by control switch.

    1.2 Failure of make up valve to replenish LP air system.

    1.3 Leakage in HP and/or LP air systems.

2. **Operator Action**

    2.1 Determine which system has low pressure:

    - 1-GB-PI-603, HP system
    - 1-GB-PI-604, LP System

    2.2 IF HP System pressure is low, THEN refer to response for Window B2.

    2.3 IF LP System pressure is low, THEN refer to response for Window C1.

    2.4 Investigate air system to determine if any leakage is apparent and enter Condition Report as applicable.

Attachment 1
Window A1 - Block Trip

2.5 IF 1H Emergency Bus is being supplied from its Alternate Offsite Power Source (1B Station Service Bus), THEN immediately perform 0-PT-80, AC Sources Operability Determination, to determine operability of Offsite Power Sources (adequate source separation). **(References 3.8 and 3.9)**

2.6 IF air system pressure cannot be restored AND it is desired to perform Unit shutdown, THEN do the following:

a. Initiate Unit shutdown using 1-OP-2.2, Unit Shutdown From Mode 1 to Mode 2.

b. Ensure Load Shed is in service using 0-OP-26.7, Load Shed.

c. Obtain a switching order from the System Operator to open G102-1 and G102-2. Operator should be dispatched to the Switchyard to open breakers at minimum Generator load.

d. WHEN Generator load decreases to approx. 80 MWE, THEN swap Station Service Busses to Reserve Station Service supply using 1-OP-26.1, Transferring 4160-Volt Busses.

e. WHEN Generator is at minimum load OR the White Light UNIT 1 GENERATOR MOTORING INITIATED, located above the Load Recorder, is LIT, THEN have the Operator in the Switchyard open G102-1 AND G102-2.

f. Secure Main Generator by turning OFF the Voltage Regulator and reducing Base Adjuster to minimum output.

3. **References**

3.1 B.B.C. Print EI-021-1-B S&W file no. 11715 - 1.44-4B

3.2 B.B.C. Manual CH-A 089 156E

3.3 Loop Book page GB-013, GB-014

Attachment 1

Window A1 - Block Trip

3.4 11715-FM-111B

3.5 ET No. EE 97-039, Rev. 0, Setpoint Evaluation /Determination For the G-12 Breaker High Pressure Air System

3.6 DCP 02-181, G-12 Breaker Compressed Air Control System Modification

3.7 DCP 05-143, Relocation of Switchyard Breaker H502, and Replacement of Breakers G1TH5 and G102 / NAPS / Unit 1

3.8 ET CEE 02-0005, Rev. 3, Voltage Specification For The Emergency AC Buses, North Anna Power Station, Unit 1 and 2

3.9 CA180447, Review Needed for Offsite Power Source Operability Verification Procedures

4. **Actuation**

4.1 1-GB-PS-615, relay LP-MTP1X

4.2 1-GB-PS-605, relay HP-MTP1X

4.3 1-GB-PS-616, relay LP-MTP2X

4.4 1-GB-PS-608, relay HP-MTP2X

GENERATOR
BREAKER
TROUBLE

1.0 Probable Cause

1.1 Any alarm on Generator Breaker Annunciator panel on turbine deck.

2.0 Operator Action

2.1 Send operator to panel at breaker on turbine deck to investigate alarm.

2.2 Refer to 1-AR-30, Main Generator Output Breaker.

2.3 IF a Block Trip condition exists (G12 Breaker will not open automatically or manually), OR G12 Breaker should have opened and did not (Pole Disagreement), AND 1H Emergency Bus is being supplied from its Alternate Offsite Power Source (1B Station Service Bus),THEN immediately perform 0-PT-80, AC Sources Operability Determination, to determine operability of Offsite Power Sources (adequate source separation).

3.0 References

3.1 11715-ESK-10H Sheet 2

4.0 Actuation

4.1 74 alarm relay in annunciator cabinet.

Dominion
North Anna Power Station
JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

INITIAL CONDITIONS

Unit 1 is in mode 1.

1-CH-217, PG Supply to Boric Acid Blender Isolation Valve, diaphragm must be replaced.

INITIATING CUE

You are requested to review the 1-CH-217 tagging record package for accuracy and completeness using the Tag Out Preparation and Review Check List.

A component configuration is NOT required by the tagout.

Circle the results of your review. If not approved, record the reason below.

Approved Not Approved

Dominion
North Anna Power Station
JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

RXXX

TASK

Review a tagging record for accuracy and completeness.

TASK STANDARDS

Tagging record review identifies inadequate boundary because 1-PG-118 is not included as a boundary isolation valve.

K/A REFERENCE:

G2.2.13 (4.1 / 4.3)

ALTERNATE PATH:

N/A

TASK COMPLETION TIMES

Validation Time = 30 minutes
Actual Time = ________ minutes

Start Time = ________
Stop Time = ________

PERFORMANCE EVALUATION

Rating [ ] SATISFACTORY [ ] UNSATISFACTORY

Candidate (Print) ______________________________

Evaluator (Print) ______________________________

Evaluator's Signature / Date ______________________________

EVALUATOR'S COMMENTS

____

____

____

Dominion
North Anna Power Station

JOB PERFORMANCE MEASURE
(Evaluation)

OPERATOR PROGRAM

RXXX

READ THE APPLICABLE INSTRUCTIONS TO THE CANDIDATE

Instructions for Simulator JPMs

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

Instructions for In-Plant JPMs

I will explain the initial conditions, and state the task to be performed. All steps, including any required communications, shall be simulated for this JPM. Under no circumstances are you to operate any plant equipment. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS

Unit 1 is in mode 1.

1-CH-217, PG Supply to Boric Acid Blender Isolation Valve, diaphragm must be replaced.

INITIATING CUE

You are requested to review the 1-CH-217 tagging record package for accuracy and completeness using the Tag Out Preparation and Review Check List.

A component configuration is NOT required by the tagout.

Circle the results of your review. If not approved, record the reason below.

Approved Not Approved

EVALUATION METHOD

Perform if conducted in the simulator or in a laboratory (use Performance Cue(s))

Simulate if conducted in the station or on a dead simulator (use Simulation Cue(s))

TOOLS AND EQUIPMENT

Tagging record package for 1-CH-217 which includes:

- Tagout Preparation and Review Checklist
- Tagout coversheet
- OP-AP-300, Attachment 7, Reactivity Management Screening
- Tagout Request
- Valve Lineup sheets
- 11715-FM-095B, sheet 1
- 11715-FK-095B, sheet 1
- 11715-FM-086D, sheet 1
- Highlighters

PERFORMANCE STEPS

START TIME ___________

1	Work scope defined and understood …	Check List Item 1

	SAT [ ] UNSAT [ ]

Standards	Checks JPM initial conditions to define work scope (replace diaphragm).

Notes/Comments

2	Boundaries established for tasks and hazards … **NO**	Check List Item 2

Critical Step	SAT [ ] UNSAT [ ]

Standards	Reviews tagging record and reference material. Determines 1-PG-118 was excluded as a boundary isolation valve (11715-FM-086D, sheet 1)

Notes/Comments

3	Review for heat trace, IA, seal water, seal cooling fuses …	Check List Item 3

	SAT [ ] UNSAT [ ]

Standards	Reviews tagging record and reference material to determine no heat trace, IA, seal water, seal cooling, fuses …

Notes/Comments

4	If a component configuration is required by the tagout …	Check List Item 4

	SAT [ ] UNSAT [ ]

Standards	Notes initiating cue that component configuration is not required and previously NA'd step on the check list.

Notes/Comments

5	Applicable procedures and controlled documents are used to create tagout. If using non-priority drawings …	Check List Item 5

	SAT [ ] UNSAT [ ]

Standards	Determines applicable procedures and controlled documents were used. Determines non-priority drawings were not used.

Notes/Comments

6	Tagout is sequenced as directed by controlling procedure, or per OP-AA-200.	Check List Item 6

	SAT [ ] UNSAT [ ]

Standards	Reviews tagout and determines sequence is correct.

Notes/Comments

7	“Return To” positions reflect OP-1A, controlling procedure …	Check List Item 7

	SAT [ ] UNSAT [ ]

Standards	Reviews tagout and valve lineup sheets and determines “return to” positions are correct.

Notes/Comments

8	Consider effects on alarms, indications, instrumentation, and controls.	Check List Item 8

	SAT [ ] UNSAT [ ]

Standards	Reviews tagout and determines no effects on alarms, indications, instrumentation, or controls.

Notes/Comments

9	Compliance with TS & TRM (maintain redundant equipment operable).	Check List Item 9

	SAT [ ] UNSAT [ ]

Standards	Determines tagout is in compliance with TS and TRM.

Notes/Comments

10	Reference documents and special notes … are on cover sheet.	Check List Item 10

	SAT [ ] UNSAT [ ]

Standards	Reviews tagout to confirm reference documents and special notes are on cover sheet.

Notes/Comments

11	Tagout Request reviewed as applicable.	Check List Item 11

	SAT [ ] UNSAT [ ]

Standards	Reviews tagout request.

Notes/Comments

12	Consider the effects on reactivity management.	Check List Item 12

	SAT [ ] UNSAT [ ]

Standards	Reviews Reactivity Screening attachment. Reviews notes 1, 2 and 4 on the tagout coversheet.

Notes/Comments

13	Pre-maintenance and post-maintenance requirements are understood.	Check List Item 13

	SAT [ ] UNSAT [ ]

Standards	Reviews tagging record package for pre-maintenance and post-maintenance requirements.

Notes/Comments

14	Comments or problems found.	Procedure Step N/A

	SAT [ ] UNSAT [ ]

Standards	Notes failure to include 1-PG-118 as a boundary isolation. Circles “not-approved” on the JPM cover sheet.

Notes/Comments

>>>>> END OF EVALUATION <<<<<

STOP TIME ___________

Dominion

North Anna Operations Tag-Out Preparation and Review Check List
Tagging is a Safety Process, Tagging is NOT a production process.

Tagout Number 1-14-CH-0001 (1-CH-217)

Prep	Review	SRO1	SRO2	Personnel, Equipment and Administrative Review Items
[✓]	[✓]	[ ]	[ ]	Work scope defined and understood using the following, as applicable: work order, P3E, Craft, Planner and Work Week Coordinator.
[✓]	[✓]	[ ]	[ ]	Boundaries established for task and hazards, all energy sources removed or noted on the tagout. Initiate WM-AA-301, Operational Risk Assessment, for all tagouts containing non-standard isolation boundaries or no available vent/drain path.
[✓]	[✓]	[ ]	[ ]	Review for heat trace, IA, seal water, seal cooling, fuses, patch cords, motor/MOV heaters, "hidden power" supplies.
N/A	N/A	[ ]	[ ]	IF a component configuration is required by the tagout, it's NOT controlled by an MOP, and it is a component/condition controlled by Operations THEN use a Plant Requirement tag to ensure the required status.
[✓]	[✓]	[ ]	[ ]	Applicable procedures and controlled documents are used to create tagout. If using Non-Priority Drawings then verify either no pending changes or the changes are reviewed and noted. If not, refer to OP-AA-200.
[✓]	[✓]	[ ]	[ ]	Tagout is sequenced as directed by controlling procedure, or per OP-AA-200.
[✓]	[✓]	[ ]	[ ]	"Return To" positions reflect OP-1A, Controlling Procedure, or Equipment Status positions. At a minimum, "Return To" position conflicts are to be noted on tagout.
[✓]	[✓]	[ ]	[ ]	Consider effects on alarms, indications, instrumentation, and controls.
[✓]	[✓]	[ ]	[ ]	Compliance with TS & TRM (Maintain Redundant Equipment Operable).
[✓]	[✓]	[ ]	[ ]	References documents and special notes, as required by OP-AA-200, are on cover sheet.
[✓]	[✓]	[ ]	[ ]	Tagout Request reviewed as applicable.
[✓]	[✓]	[ ]	[ ]	Consider the effects on Reactivity Management. If affected, then note on the tagout. OP-AP-300 Attachment 7 can be referenced to aid in determination.
		[ ]	[ ]	Pre-maintenance and Post-maintenance requirements are understood.
				Walkdown to evaluate the following: tagout covers the work scope, labeling, components in overhead, work area hazards, work requests on components to be manipulated, "return to" positions match as found position, potential for trapped water in boundary, alternate lockout device required, potential Status Control bump hazards and inform lead craft of

Preparer: CRO (initials) Walkdown: ℰ (Initials) Review: ℰ (initials) SRO: ______ (initials) SRO: ______ (initials)

Comments or Problems Found:

____

____

____

*If modifications are required, evaluate affects to all associated tagouts.

Component to be Worked:

1-CH -   -217

PG Supply To Boric Acid Blender Isolation Valve

AB 274` BLENDER AREA

NOTES

****** FOR TRAINING USE ONLY ******

Effects Reactivity. Refer to OP-AP-300 Att. 7

1) This tagout removes the ability to supply PG to Unit 1. Ensure no makeups or dilutions are required before hanging this tagout.

2) 1-CH-220 IS "LOCKED CLOSED" to comply with T.S. 3.1.8 in the event of a Unit Trip. If Unit 1 trips while this tagout is hanging, 1-CH-241, 1-CH-FCV-1114B, and 1-CH-FCV-1113B will also need to be secured in the Closed position within 15 minutes, unless a boration is in progress, IAW T.S. 3.1.8 basis. Ensure Unit 1 OATC and Aux Bldg watchstander are briefed on this requirement before hanging tagout.

3) Ensure craft is ready to work prior to tagging.

4) Return to position for 1-CH-217 is for Unit 1 in Modes 1 or 2. If Unit 1 enters modes 3, 4, 5, or 6, then change the return to position of 1-CH-217 to "LOCKED CLOSED".

REASON

Replace diaphragm for 1-CH-217 due to leak.

INSTRUCTIONS / HAZARDS

CAUTION CRAFT: EXPECT WATER (PG).

ADDITIONAL INFORMATION

REF: 11715-FM-095B(1), FM-086D(1), FK-095B(1), 0-OP-9A, 1-OP-8.1A

Tagout Attributes:

Attribute Description	Attribute Value
Section Holder	SHIFT MANAGER
POD Work Week	N/A
POD Schedule	Emergent
Tag On Request	No

Work Order List:

Number / Equipment ID	Description
59102158672 - - - - - - - - - - - - **42** - - - - - - - - - - - - **1-CH - -217**	Replace diaphragm due to leakage

Tagout Verification:

Status	Description	Name	Verification Date
Prepared	Prepared	ESOMS CRO TEST USER	05/15/2014 09:47:17
Walkdown	Walkdown		
1 SRO Review	1 SRO Review		

Status	Description	Name	Verification Date
2 SRO Review	2 SRO Review		
Approved to Hang	Approved to Hang		
Tags Verified Hung	Tags Verified Hung		
Restoration SRO review	Restoration Changed		
Approved to Remove	Approved to Remove		
Tags Verified Removed	Tags Verified Removed		

Tag Type ------ Serial No.	Equipment -------------------- * Equipment Description * Equipment Location	Ver Req	Pla Seq	Placement Configuration ------------- * Notes	Place. 1st Verif Date/Time	Place. 2nd Verif Date/Time	Ver Req	Rest Seq	Rest. Config. *As Left (If Diff.) ------------- * Notes	Rest. 1st Verif Date/Time	Rest. 2nd Verif Date/Time
Mech. Danger ------	1-CH - -220 -------------------- * PG Supply To Chem Mixing Tk And Manual Dilute Isol * BLENDER AREA	IV	1	LOCKED CLOSED -------------			IV	5	CLOSED -------------		
Mech. Danger ------	1-CH - -218 -------------------- * PG Supply To Boric Acid Blender Isolation Valve * AUX BLDG 274` BLENDER AREA	IV	2	CLOSED -------------			IV	4	OPEN -------------		
Mech. Danger ------	1-CH - -216 -------------------- * PG To Boric Acid Blender 1-CH-PC-1114 Isol Valve * BLENDER AREA	IV	3	OPEN -------------			IV	3	OPEN -------------		
Mech. Danger ------	1-CH -ICV -3212 -------------------- * 1-CH-PC-1114 INLET LINE VENT VALVE * AB 274' AT UNIT 1 BLENDER	IV	4	OPEN -------------			IV	1	CLOSED -------------		
Maintenar item ------	1-CH - -217 -------------------- * PG Supply To Boric Acid Blender Isolation Valve * AB 274` BLENDER AREA	IV		-------------			IV	2	OPEN -------------		

ATTACHMENT 7

(Page 1 of 1)

Reactivity Management Screening

Does the activity:

1.	Affect nuclear fuel?	Yes	No
2.	Affect CRD mechanisms or position indication?	Yes	No
3.	Change input instrumentation or software providing heat balance calculations, core thermal calculations, or core power distribution calculations?	Yes	No
4.	Affect reactivity calculations such as shutdown margin or ECP?	Yes	No
5.	Affect incore or excore nuclear instrumentation?	Yes	No
6.	Place Rods in Manual?	Yes	No
7.	Trip Reactor Protection components? (SSPS, RPS, EIF, AMSAC)	Yes	No
8.	Affect boration/dilution system?	Yes	No
9.	Affect main steam or feedwater flow or indication?	Yes	No
10.	Change T hot or T cold indications?	Yes	No
11.	Affect TAVG control circuits?	Yes	No
12.	Affect EH controls?	Yes	No
13.	Affect blowdown flows or indications?	Yes	No
14.	Affect Steam Dump operation?	Yes	No
15.	Place RCS pressure or controls to Manual?	Yes	No
16.	Affect feedwater pre-heating systems (SD, ES)?	Yes	No

If any answer is Yes, then notify the Shift Manager and STA. A formal pre-job brief shall be conducted with the control room team and all individuals involved in the activity. Consideration should be given to establishing additional controls, oversight and monitoring to preclude a reactivity event.

Dominion

Tag-Out Request

OP-AA-200 - Attachment 5 **Page 1 of 1**

Issued To (Name)	Extension	Department
Tag-Out Requested On Unit: ☑ 1 ☐ 2 ☐ 3 ☐ Common		

Reason For Tag-Out	Initiating Document
☑ Corrective Maintenance	Work Order Number 59102158672
☐ Preventive Maintenance	Design Change Package Number
☐ Trouble Shooting	Other
☐ Testing	
☐ Engineering Work Package	
☐ Other	

Equipment Affected 1-CH-217

Work To Be Done Replace diaphragm due to leakage

Assist in Defining Work Scope and establishing safe work boundaries, review the following and check all that apply:

- ☐ "NO ROTATION" Tags (Equipment de-energized and prevented from turning; Non-intrusive maintenance.)
- ☐ "NO FLOW" Tags (System isolated to allow equipment to be manipulated without affecting the plant. Non-intrusive maintenance.)
- ☑ "FULL TAGOUT" Tags (Equipment isolated from all energy sources and depressurized. Intrusive maintenance.)
- ☐ "ELECTRICAL ONLY" (Equipment electrically isolated, the craft will use LOCKOUT, if required, for all other energy sources.)
- ☐ Controlling Procedure, which defines energy sources (OP, MOP, PT, ICP, MCM, etc. List in remarks.)
- ☐ Personnel entering a piping system or plant equipment (List in remarks.) (Operations will assist in determining applicable OPS procedures.)
- ☐ Auxiliary Components associated with equipment:
  - ☐ Seal Water
  - ☐ Oil sub-system
  - ☐ Cooling Water (i.e., BC, SW, CD, CC)
- ☐ Heat Trace (consider if removing insulation)
- ☐ Purge Path Required (Describe Below)
- ☐ System in-service for Freon removal
- ☐ Steam removed from air handler
- ☐ Hazardous Chemicals involved
- ☐ Control Power fuses removal required
- ☐ Grounds required
- ☐ Motor Heater fuses
- ☐ MOV motor/grease heater fuse
- ☐ MOV internal power supplies on LS/Rotor Contacts
- ☐ PMT requires Danger Tags (MOV testing, Flow Scan)

	Date	Time
Tag-Out Request Submitted By (Name) Planner	4/24/14	11:00
Tag-Out Request Verified By (Name) Mechanic	4/24/14	11:30

Recommended Isolations or Remarks (If possible, include tag type and position.)

Step: 5.10 PG HEADER SOUTH OF CATALYTIC RECOM ==> PG HEADER SOUTH OF CATALYTIC RECOMBINER CUBICLE

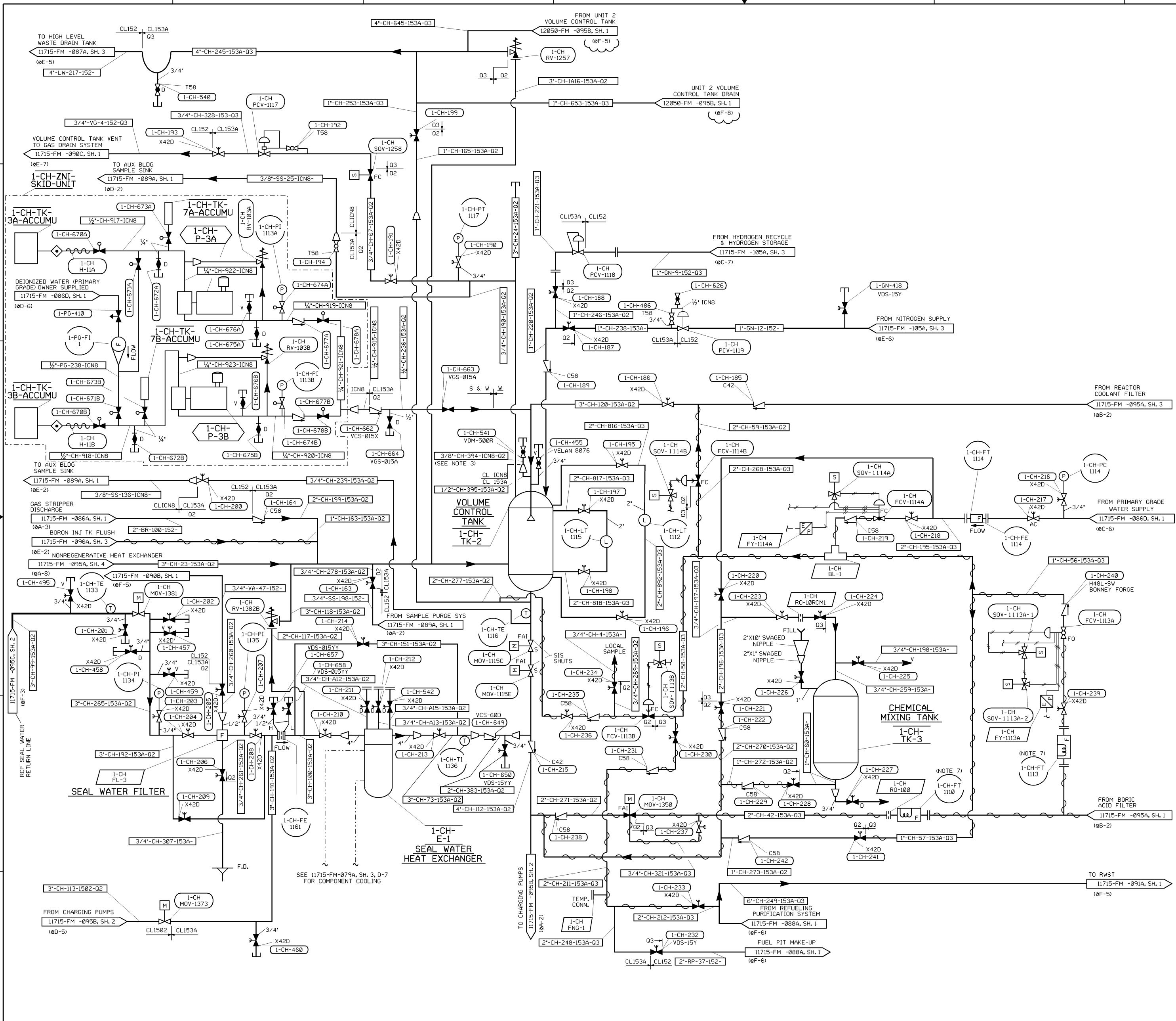
Equipment:	Description/Instruction:	Location:	Required Config:	Actual Config:	Verif:	Initials	
1-PG - -104	Gas Waste Recombiner Flushwater Sply Isol Vv (10 feet overhead outside Recombiner Cube)	Aux Bldg elev.256	CLOSED		IV	Group: 0	Required:
Instruction:							
Notes:							
Prints:							
1-PG - -106	PG Supply to Recombiner Constant Temperature Bath (10 feet overhead outside Recombiner Cube)	259' East Outside Recombiner Cube	CLOSED		IV	Group: 0	Required:
Instruction:							
Notes:							
Prints:							
1-PG - -118	Boric Acid Blender Flushwater Supply Isol Valve (10 feet overhead outside Recombiner Cube)	AB 259`;12`NORTH 1-HC-H2A-101 OVHD	OPEN		IV	Group: 0	Required:
Instruction:							
Notes:							
Prints:							
1-PG - -210	Boric Acid Blender Flush Water Supply Isol Valve (above 1-IA-C-1)	AB 259` IN OVHD ABOVE U-1 AIR COMP.	OPEN		IV	Group: 0	Required:
Instruction:							
Notes:							
Prints:							
1-PG - -300	Primary Grade Water Hose Conn Isol Valve	Aux Bldg 259' at 2-IA-C-1	CLOSED		IV	Group: 0	Required:
Instruction:							
Notes:							
Prints:							
1-PG - -143	Primary Grade Water Branch Header Isol Valve (West of 2-IA-C-1)	IN AUX. BLDG.	OPEN		IV	Group: 0	Required:
Instruction:							
Notes:							
Prints:							
1-PG - -323	Primary Grade Water Supply Header Isol Valve (West wall of 2-IA-C-1)	West wall of 2-IA-C-1	OPEN		IV	Group: 0	Required:
Instruction:							
Notes:							
Prints:							

Step: 5.21 LETDOWN - BLENDER AREA (RT TO LFT) ==> BLENDER AREA (RIGHT TO LEFT)

Equipment:	Description/Instruction:	Location:	Required Config:	Actual Config:	Verif:	Initials	
1-CH - -225	Chemical Mixing Tank Vent Valve	BLENDER AREA	CLOSED		IV		
Instruction:							
Notes:						Group: 0	Required:
Prints:							
1-CH - -226	Chemical Mixing Tank Funnel Isolation Valve	U-1 BLENDER AREA	CLOSED		IV		
Instruction:						Group: 0	Required:
Notes:							
Prints:							
1-CH - -227	Chemical Mixing Tank Drain Valve	BLENDER AREA	CLOSED		IV		
Instruction:							
Notes:						Group: 0	Required:
Prints:							
1-CH - -216	PG To Boric Acid Blender 1-CH-PC-1114 Isol Valve	BLENDER AREA	OPEN		IV		
Instruction:						Group: 0	Required:
Notes:							
Prints:							
1-CH - -217	PG Supply To Boric Acid Blender Isolation Valve	AB 274` BLENDER AREA	OPEN		IV		
Instruction: Modes 1 and 2						Group: 0	Required:
Notes: 4							
Prints:							
1-CH - -217	PG Supply To Boric Acid Blender Isolation Valve	AB 274` BLENDER AREA	LOCKED CLOSED		IV		
Instruction: Modes 3, 4, 5, and 6						Group: 0	Required:
Notes: 4							
Prints:							
1-CH - -228	Chemical Mixing Tank Outlet Isolation Valve	AB 274` BLENDER AREA	CLOSED		IV		
Instruction:						Group: 0	Required:
Notes:							
Prints:							
1-CH - -224	Chemical Mixing Tank PG Supply Header Isol Valve	BLENDER AREA	CLOSED		IV		
Instruction:						Group: 0	Required:
Notes:							
Prints:							

Step: 5.21 LETDOWN - BLENDER AREA (RT TO LFT) ==> BLENDER AREA (RIGHT TO LEFT)

Equipment:	Description/Instruction:	Location:	Required Config:	Actual Config:	Verif:	Initials	
1-CH - -223	Chemical Mixing Tank PG Supply Header Isol Valve	BLENDER AREA	OPEN		IV		
Instruction:						Group: 0	Required:
Notes:							
Prints:							
1-CH - -221	Charging Pumps Manual Dilute Isolation Valve	AB 274` BLENDER AREA	CLOSED		IV		
Instruction:						Group: 0	Required:
Notes:							
Prints:							
1-CH - -220	PG Supply To Chem Mixing Tk And Manual Dilute Isol	BLENDER AREA	CLOSED		IV		
Instruction:						Group: 0	Required:
Notes:							
Prints:							
1-CH - -218	PG Supply To Boric Acid Blender Isolation Valve	AUX BLDG 274` BLENDER AREA	OPEN		IV		
Instruction:						Group: 0	Required:
Notes:							
Prints:							
1-CH - -236	Blender Discharge To Dilute Header (Reach Rod)	VCT CUBE/REACH ROD IN BLENDER AREA	OPEN		IV		
Instruction:						Group: 0	Required:
Notes:							
Prints:							
1-CH - -186	Reactor Coolant Filter To Vct Inlet Hdr Isol Valve (Chain Operated Reach Rod)	AUX BLDG 274` U-1 BLENDER AREA	OPEN		IV	Group: 0	Required:
Instruction:							
Notes:							
Prints:							
1-CH - -191	Vct To Stripper Vent Cond Isolation Valve (Reach Rod)	AB 274` BLENDER AREA (REACHROD)	OPEN		IV		
Instruction:						Group: 0	Required:
Notes:							
Prints:							
1-CH - -230	Boric Acid Blender To Rwst And Fuel Pit Isol Vv	AB Unit 1 Blender Area	CLOSED		IV		
Instruction:						Group: 0	Required:
Notes:							
Prints:							



NOTES

1. FOR ADDITIONAL NOTES, SEE SHEET 1 OF 11715-FM-095A.
2. DEVELOPED FROM W684J903
3. 3/8" CONNECTIONS FOR FUTURE BORON CONCENTRATION MONITORING SYSTEM.
4. FIELD TO PURCHASE ROTAMETER, WHITEY #IRS 8 1/2" OR EQUAL. ROTAMETER WILL BE REPLACED WITH SPOOLPIECE AFTER CALIBRATION OF VOM-500W.
5. TO PROVIDE ACCEPTABLE MOTOR OPERATED VALVE (MOV) CLOSURE THRUST MARGIN, CHARGING PUMP MINIFLOW RECIRCULATION MOVS 1-CH-MOV-1275A/B/C HAVE BEEN INSTALLED WITH FLOW REVERSED THROUGH THE VALVE (FLOW-OVER-THE DISC) IN LIEU OF NORMAL FLOW-UNDER-THE-DISC.

THIS CAD DRAWING SHEETS 1 & 2 OF 2 SUPERSEDES THE MANUAL FLOW AND VALVE OPERATING NUMBERS DIAGRAMS, 11715-FM-040B REV-18 AND 11715-FM-095B REV-13 RESPECTIVELY.

REFERENCE DRAWINGS

01. FOR ADDITIONAL REFERENCE DRAWINGS, SEE SHEET 1 OF 11715-FM-095A.

NUMBER	TITLE
02. 11715-FM-079A	FLOW/VON DIAGRAM,COMPONENT COOLING WATER SYSTEM
03. 11715-FM-086A	FLOW/VON DIAGRAM,BORON RECOVERY SYSTEM
04. 11715-FM-086D	FLOW/VON DIAGRAM,BORON RECOVERY SYSTEM
05. 11715-FM-087A	FLOW/VON DIAGRAM,WASTE DISPOSAL SYSTEM
06. 11715-FM-088A	FLOW/VON DIAGRAM,FUEL PIT CLNG & REF PURIF SYSTEM
07. 11715-FM-089A	FLOW/VON DIAGRAM,SAMPLING SYSTEM
08. 11715-FM-090B	FLOW/VON DIAGRAM,VENT & DRAIN SYSTEM
09. 11715-FM-090C	FLOW/VON DIAGRAM,VENT & DRAIN SYSTEM
10. 11715-FM-095A	FLOW/VON DIAGRAM,CHEM & VOLUME CONTROL SYSTEM
11. 11715-FM-095B	FLOW/VON DIAGRAM,CHEM & VOLUME CONTROL SYSTEM
12. 11715-FM-095C	FLOW/VON DIAGRAM,CHEM & VOLUME CONTROL SYSTEM
13. 11715-FM-096A	FLOW/VON DIAGRAM,SAFETY INJECTION SYSTEM
14. 11715-FM-105A	FLOW/VON DIAGRAM,PRIMARY PLANT GAS SUPPLY SYSTEM
15. 12050-FM-095B	FLOW/VON DIAGRAM,CHEM & VOLUME CONTROL SYSTEM

BYPRODUCT DRAWINGS

1. 11715-CBM-095B, SH. 1 — ISI CLASSIFICATION BOUNDARY DRAWING
2. 11715-DAR-095B, SH.1 — APPENDIX R FLOWPATH
3. 11715-SPM-095B, SH. 1 — ISI SYSTEM PRESSURE TESTING DRAWING
4. 11715-LRM-095B, SH. 1 — LICENSE RENEWAL

BYPRODUCT NOTES

REVISION DESCRIPTION

Dominion — PRIORITY DRAWING

NUCLEAR ENGINEERING
RICHMOND, VIRGINIA

FLOW/VALVE OPERATING NUMBERS DIAGRAM CHEMICAL AND VOLUME CONTROL SYSTEM NORTH ANNA POWER STATION UNIT 1 VIRGINIA POWER

REVISION DESCRIPTION

REVISED PER DC NA-12-00035/ DUR 14-700186. THIS DWG SUPERSEDES REV 42.

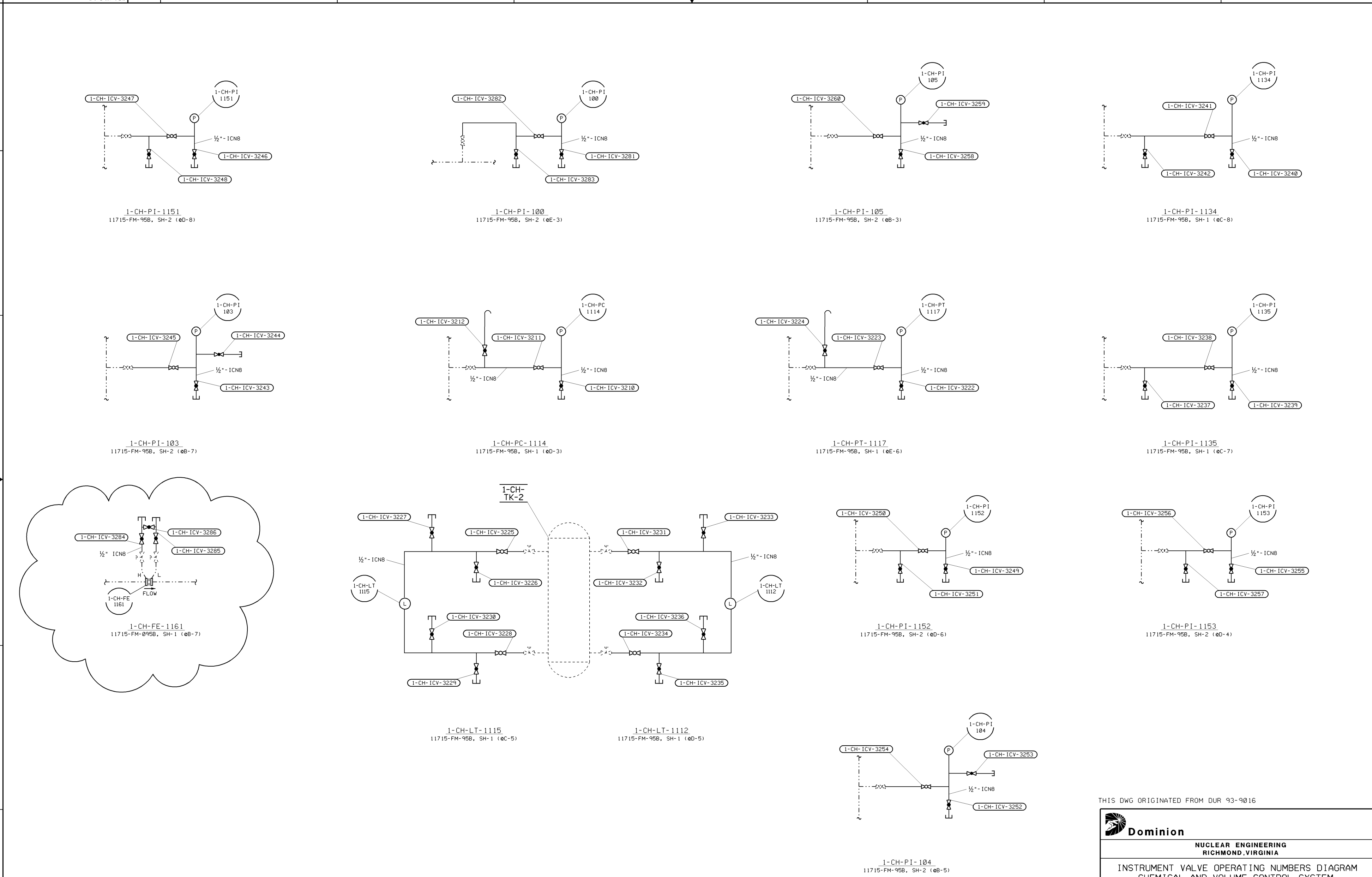
CAD NO: N1095B1M.DGN

DRAWING NO: 11715-FM-095B — REV 43

DSGN ANM — SCALE: NONE — UNLESS OTHERWISE NOTED — SH 1 OF 2

PC=MF3

adam015
09-APR-2014 07:15
N1095B1M.DGN



THIS DWG ORIGINATED FROM DUR 93-9016

Dominion

NUCLEAR ENGINEERING
RICHMOND, VIRGINIA

INSTRUMENT VALVE OPERATING NUMBERS DIAGRAM
CHEMICAL AND VOLUME CONTROL SYSTEM
NORTH ANNA POWER STATION - UNIT 1
VIRGINIA POWER

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C:\usr\dgn\N1095B1DGNs31	
REVISED PER DCP 99-147/DCR 2003-1285 THIS DWG SUPERSEDES REV 6 ORIGINAL	DRAWING NO: 11715-FK-095B	REV 7
DSGN GTW	SCALE: NONE	UNLESS OTHERWISE NOTED SH 1 OF 1

PC=MF2

PRIOR TO USING FOR DESIGN WORK CHECK DMIS FOR WORK PENDING

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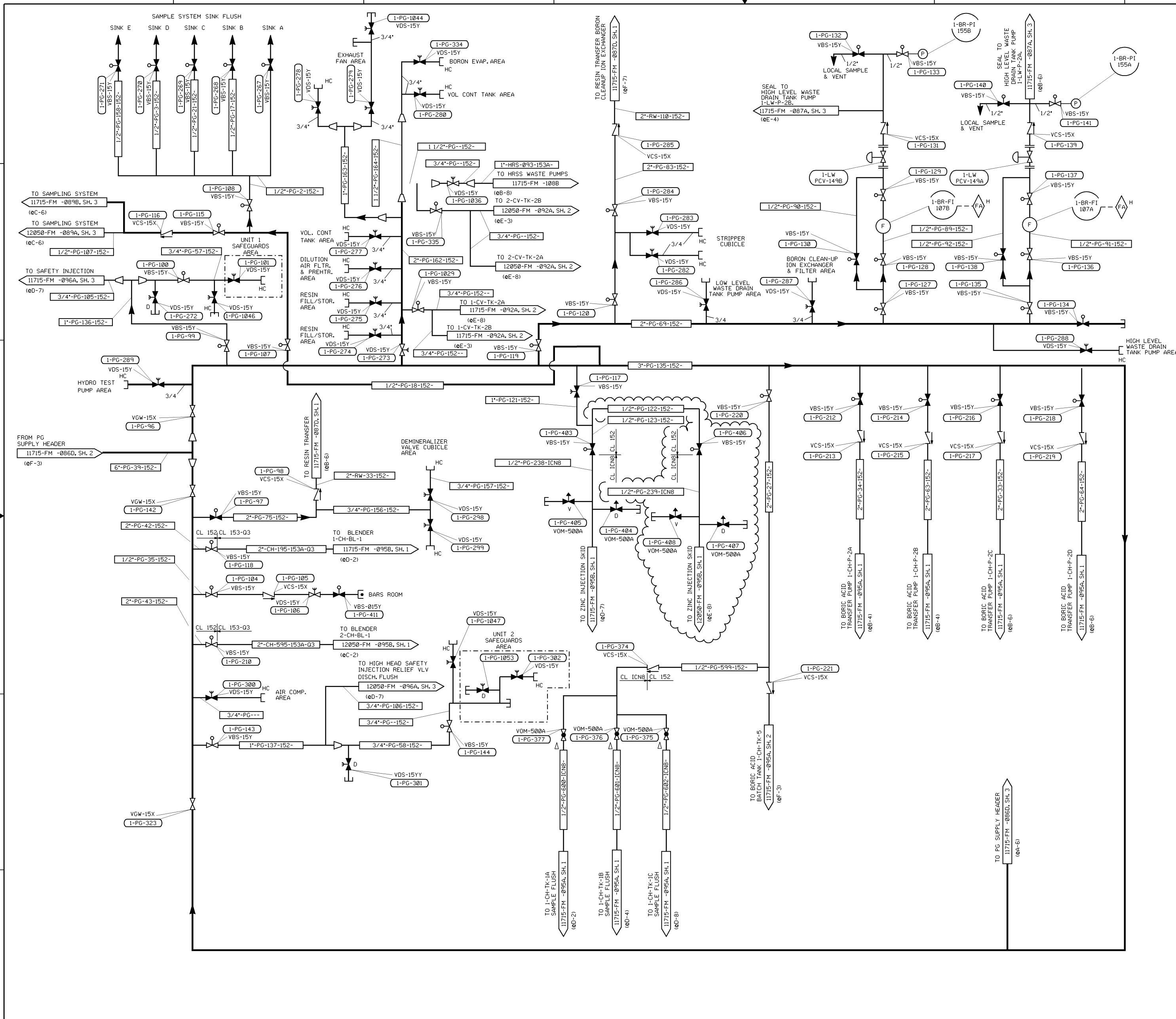
27-MAR-2003 13:36
C:\usr\dgn\N1095B1DGNs31



NOTES

1. FOR NOTES, SEE SHEET 1 OF 11715-FM-082A.

THIS CAD DRAWING SHEETS 1,2 & 3 OF 3 SUPERSEDES THE MANUAL FLOW AND VALVE OPERATING NUMBERS DIAGRAMS, 11715-FM-030D REV-14 AND 11715-FM-086D REV-12 RESPECTIVELY.

REFERENCE DRAWINGS

NUMBER	TITLE
01.	FOR ADDITIONAL REFERENCE DRAWINGS, SEE SHEET 1 OF 11715-FM-082A.
02. 12050-FM-082C	FLOW/VON DIAGRAM,COMPRESSED AIR SYSTEM
03. 11715-FM-086D	FLOW/VON DIAGRAM,BORON RECOVERY SYSTEM
04. 11715-FM-087A	FLOW/VON DIAGRAM,WASTE DISPOSAL SYSTEM
05. 11715-FM-087D	FLOW/VON DIAGRAM,WASTE DISPOSAL SYSTEM
06. 12050-FM-089A	FLOW/VON DIAGRAM,SAMPLING SYSTEM
07. 11715-FM-089B	FLOW/VON DIAGRAM,SAMPLING SYSTEM
08. 11715-FM-095A	FLOW/VON DIAGRAM,CHEM & VOLUME CONTROL SYSTEM
09. 11715-FM-097B	FLOW/VON DIAGRAM,GASEOUS WASTE DISPOSAL SYSTEM
10. 11715-FM-095B	FLOW/VON DIAGRAM,CHEM & VOLUME CONTROL SYSTEM
11. 12050-FM-095B	FLOW/VON DIAGRAM,CHEM & VOLUME CONTROL SYSTEM
12. 12050-FM-096A	FLOW/VON DIAGRAM,SAFETY INJECTION SYSTEM
13. 11715-FM-096A	FLOW/VON DIAGRAM,SAFETY INJECTION SYSTEM
14. 12050-FM-092A	FLOW/VON DIAGRAM,CONT VAC & LEAKAGE MONITOR SYS
15. 12050-FM-095A	FLOW/VON DIAGRAM,CHEM & VOLUME CONTROL SYSTEM

BYPRODUCT DRAWINGS

BYPRODUCT NOTES

REVISION DESCRIPTION

Dominion

PRIORITY DRAWING

NUCLEAR ENGINEERING
RICHMOND,VIRGINIA

FLOW/VALVE OPERATING NUMBERS DIAGRAM BORON RECOVERY SYSTEM NORTH ANNA POWER STATION UNIT 1 VIRGINIA POWER

REVISION DESCRIPTION

REVISED PER DC NA-12-00035/ DUR 14-700186. THIS DWG SUPERSEDES REV 37.

CAD NO: N1086D1M.DGN

DRAWING NO: 11715-FM-086D REV 38

DSGN ANM | SCALE: NONE | UNLESS OTHERWISE NOTED | SH 1 OF 3

PC=MF3

adam015
09-APR-2014 07:15
N1086D1M.DGN

Dominion
North Anna Power Station
ADMINISTRATIVE JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

INITIAL CONDITIONS

An HP technician has reported possible damage on 1-CH-TCV-100 (letdown header bypass to VCT temperature control valve) in the Unit 1 non-regen heat exchanger cube.

Your total dose received this year is 25 mRem.

INITIATING CUE

You are directed to perform the following as the operator who will inspect 1-CH-TCV-100. Record your answers below.

- Select the appropriate RWP task number.

- Determine the required protective clothing for the job.

- Determine the required dosimetry for the job.

- Determine the dose alarm setpoint **AND** the dose rate alarm setpoint.

- Determine the maximum stay time based on reaching the dose alarm setpoint (do not use the 80% value).

- Assume you avoid the areas surveyed for “contact and 30 cm” readings.
- Assume no dose is received during travel to and from 1-CH-TCV-100.

Dominion
North Anna Power Station
ADMINISTRATIVE JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

TASK

Determine correct RWP task number, protective clothing, dosimetry, dose alarm setpoint, dose rate alarm setpoint, and maximum stay-time.

TASK STANDARDS

Correct RWP task, protective clothing, dosimetry, dose alarm and dose rate setpoints are identified from the RWP/Task.

Correct maximum stay time is calculated based on the alarm setpoint.

K/A REFERENCE:

G2.3.7 (3.5/3.6)

ALTERNATE PATH:

N/A

TASK COMPLETION TIMES

Validation Time = 15 minutes
Actual Time = _______ minutes

Start Time = _______
Stop Time = _______

PERFORMANCE EVALUATION

Rating	[ ] SATISFACTORY	[ ] UNSATISFACTORY
Candidate (Print)	______________________	
Evaluator (Print)	______________________	
Evaluator's Signature / Date	______________________	

EVALUATOR'S COMMENTS

____

____

Dominion
North Anna Power Station

ADMINISTRATIVE JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

READ THE APPLICABLE INSTRUCTIONS TO THE CANDIDATE

Instructions for Simulator JPMs

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

Instructions for In-Plant JPMs

I will explain the initial conditions, and state the task to be performed. All steps, including any required communications, shall be simulated for this JPM. Under no circumstances are you to operate any plant equipment. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS

An HP technician has reported possible damage on 1-CH-TCV-100 (letdown header bypass to VCT temperature control valve) in the Unit 1 non-regen heat exchanger cube.

Your total dose received this year is 25 mRem.

INITIATING CUE

You are directed to perform the following as the operator who will inspect 1-CH-TCV-100. Record your answers below.

- Select the appropriate RWP task number.
- Determine the required protective clothing for the job.
- Determine the required dosimetry for the job.
- Determine the dose alarm setpoint **AND** the dose rate alarm setpoint.
- Determine the maximum stay time based on reaching the dose alarm setpoint (do not use the 80% value).
- Assume you avoid the areas surveyed for “contact and 30 cm” readings.
- Assume no dose is received during travel to and from 1-CH-TCV-100.

EVALUATION METHOD

Demonstration if conducted in the simulator or in a laboratory (use DEMONSTRATION cues)
Verbal-visual if conducted in the station or on a dead simulator (use VERBAL-VISUAL cues)

TOOLS AND EQUIPMENT

RWP, Survey Map, Calculator

PERFORMANCE STEPS

START TIME ___________

1	From the RWP provided, determine which task number is applicable for the job.	Procedure Step N/A

Critical Step	SAT [ ] UNSAT [ ]

Standards	Operator determines task 1 is the correct task.

Notes/Comments:

2	From the RWP provided, determine the protective clothing requirements.	Procedure Step: N/A

Critical Step	SAT [ ] UNSAT [ ]

Standards	Operator determines that the area is not contaminated; therefore, no protective clothing is required. **Protective Clothing Requirements:** None

Notes/Comments:

3	From the RWP provided, determine the dosimetry required.	Procedure Step: N/A

Critical Step	SAT [ ] UNSAT [ ]

Standards	Operator identifies that ED/SRD & TLD are required.

Notes/Comments:

4	From the RWP provided, determine the dose alarm setpoint and dose rate alarm setpoint.	Procedure Step: N/A

Critical Step	SAT [ ] UNSAT [ ]

Standards	Operator determines the dose alarm is 5 mRem and the dose rate alarm is 90 mRem/hr.

Notes/Comments:

5	Determine the maximum stay time based on reaching the RWP dose alarm setpoint.	Procedure Step: N/A

Critical Step	SAT [ ] UNSAT [ ]

Standards	Operator uses RWP to determine dose alarm is set at 5 mRem. Operator uses survey map to determine highest general area dose rate is 35 mrem/hr. Operator performs the following calculation to determine stay time: 5 mR divided by 35 mR/hr x 60 min/hr = 8.57 minutes = 8 minutes and 34.2 seconds

Notes/Comments:

Acceptable range = 8.5 to 8.6 minutes (8 minutes, 30 seconds to 8 minutes, 36 seconds)

END OF EVALUATION

STOP TIME ____________

RADIATION WORK PERMIT SUMMARY 14-1207

RP-AA-274 ATT. 2

PLANT CODE	YEAR	RWP NUMBER	REV.	RWP START	TYPE	CATEGORY	RWP EXPIRATION
0	14	14-1207	0	01-JAN-2014 00:00	S	RM	31-DEC-2014 23:59

RWP DESCRIPTION

Routine Duties, walkdown, and valve line-ups by Operations personnel.

TASK SUMMARY

		DOSE ALARM	DOSE RATE ALARM	TIME INTERVAL
1	Routine Duties, walkdown, and valve line-ups by Operations personnel. Access to Reactor Containments allowed for containment closure team only.	5	90	
2	Perform change out of 1-DC-FL-2 (Radwaste Dewatering Filter).	10	90	
3	BARS Operation, Sampling & 1-RP-FL-2 Change Out	5	90	

ALARA INFORMATION

ALARA Review No.	Hours-Estimated	Person-mrem
14-001	2099997	1580

SPECIAL INSTRUCTIONS:

Requirements for High Radiation Area or Locked High Radiation Area Entry:
-Notify RP PRIOR to entry (Discuss work to be performed)
-Signed in on an RWP that ALLOWS entry into the High Radiation Area or Locked High Radiation Area
-Understand the ED Alarm set-points and your response to a dose or dose rate alarm
-Understand the radological conditions (dose rates) in your work area
-Monitor Electronic Dosimetry frequently (Approximately every 15 minutes)
-Notify RP upon exit

Prepared By	Mark Stokes	DATE:	05-OCT-2013 00:0(	Approved By	DATE:
Revised By		DATE:		Approved By	DATE:
Terminated By		DATE:		Approved By	DATE:

VALID FROM 01-JAN-2014 00:00	TO 31-DEC-2014 23:59	RWP 14-1207-1	REV. NO 0

DOSE RATE ALARM: 90 mrem/Hr	BUDGETED DOSE: 1533 mrem
DOSE LIMIT ALARM: 5 mrem	ALARA EVALUATION NO: 14-001

JOB LOCATIONS:
ALL RCAs EXCEPT UNIT 1 AND UNIT 2 REACTOR CONTAINMENTS

JOB DESCRIPTION: Routine Duties, walkdown, and valve line-ups by Operations personnel. Access to Reactor Containments allowed for containment closure team only.

THE MAXIMUM POSTED AREA THAT CAN BE ENTERED:
Locked High Radiation Area

RADIOLOGICAL CONDITIONS: *Indicates estimated value for RWP Preparation. See survey forms for details

GENERAL AREA RADIATION LEVELS (mrem/hr):
See Current RCA Surveys.

CONTACT/HOT SPOT RADIATION LEVELS (mrem/hr):
See Current RCA Surveys.

CONTAMINATION LEVELS (dpm/100cm2):
See Current RCA Surveys.

AIRBORNE RADIOACTIVITY (DAC):
0.3*

REQUIRED JOB COVERAGE:
ROUTINE

DOSIMETRY REQUIREMENTS:
ED/SRD TLD

DOSIMETRY COMMENTS:
1.0)Caution: If PAM(ED) is utilized, ensure PAM(ED) is secured so that individual can feel the vibration if PAM(ED) alarms.
2.0)Notify HP prior to entry into a "Neutron Exposure Area" for additional dosimetry requirements.

PROTECTIVE CLOTHING REQUIREMENTS:
LAB COAT, SHOE COVERS & GLOVES MAY BE USED FOR ANY NON-PHYSICAL ACTIVITIES (I.E. OBSERVATIONS OR EQUIPMENT CHECKS) AND CONTAMINATION LEVELS ARE < 10,000 DPM/100CM2.

1.0)Required Protective Clothing:
  Surgeons Hood
  One Pair Coveralls
  Rubber Boots
  High Top Shoe Covers
  Cotton Inserts
  One Pair Rubber Gloves
2.0)Protective Clothing requirements as stated are for entry into "Contaminated Areas" only.
3.0)Double set of PC's are required for work in areas with contamination levels greater than 100,000 dpm/100cm2
4.0)Protective Clothing requirements for a "Hot Particle Area" (HPA) [in addition to those stated above] are:
  4.1)Hood, gloves, coveralls, high top shoe covers and rubber boots.
  4.2)Workers interfacing with individuals/equipment in a HPA - Gloves and face shield.

A RWP PRE-JOB BRIEFING IS REQUIRED:

HRA BRIEF BY AN HP TECHNICIAN AND SIGN ATTENDANCE SHEET.

WORKER INSTRUCTIONS:

1.0)ED Alarms:
   1.1)If ED dose rate alarm occurs, THEN leave area immediately and notify HP (unless authorized by HP Supervisor and have been briefed on proper responses).
   1.2)If ED dose alarm occurs, THEN leave area immediately and report to the Health Physics office.
2.0)Unless Continuous Health Physics Coverage is provided, workers shall read their SRD/ED approximately every 15 minutes.
3.0)When 80% of dose alarm setpoint is reached, leave work area in a safe condition and exit the RCA.
4.0)Do not enter High Radiation Areas without HP coverage or a ED with knowledge of work area dose rates.
5.0)Workers are responsible for notifying the HP Supervisor or Lead Technician prior to venting\draining systems, that may affect Radiological Conditions in an area, to ensure proper Health Physics Monitoring.
6.0)HPA exit instructions:
   6.1)Use extreme care in removing PCs and frisking.
   6.2)Workers are to be monitored by HP upon exiting the HPA.
   6.3)Workers are to be monitored by HP after interfacing with workers/equipment in a HPA.
7.0)Do not remove any items from a posted "Hot Particle Area" until authorized by HP-Ops.
8.0)Prior to returning Electronic Dosimetry (ED), all individuals shall process through the PDA for whole body monitoring. This is not required if returning ED solely for the purpose of changing RWPs.
9.0)Notify HP prior to entry in overhead areas greater than 8 feet.

HEALTH PHYSICS INSTRUCTIONS:

1.0)Stop work and leave area if whole body dose rates exceed 2000 mRem/hr.
2.0)When the ED dose rate alarm is set >1000 mRem/hr then:
   2.1)Stay times and continuous HP coverage is required.
   2.2)If worker ED alarms, then have worker immediately exit area and report to HP.
3.0)"Hot Particle Area" (HPA) monitoring requirements:
   3.1)Workers exposed skin shall be monitored by HP-Ops every four hours while the workers are in the HPA.

VALID FROM	01-JAN-2014 00:00	TO	31-DEC-2014 23:59	RWP	14-1207-2	REV. NO	0

DOSE RATE ALARM: 90 mrem/Hr	BUDGETED DOSE: 30 mrem
DOSE LIMIT ALARM: 10 mrem	ALARA EVALUATION NO: 14-001

JOB LOCATIONS:
DECON BUILDING AND WASTE SOLIDS

JOB DESCRIPTION: Perform change out of 1-DC-FL-2 (Radwaste Dewatering Filter).

THE MAXIMUM POSTED AREA THAT CAN BE ENTERED:
Locked High Radiation Area

RADIOLOGICAL CONDITIONS: *Indicates estimated value for RWP Preparation. See survey forms for details

GENERAL AREA RADIATION LEVELS (mrem/hr):
2 to 10, up to 500 at 1 foot from exposed filters*

CONTACT/HOT SPOT RADIATION LEVELS (mrem/hr):
Up to 2500 on exposed filters*

CONTAMINATION LEVELS (dpm/100cm2):
<1,000, up to 200,000 on exposed filters*

AIRBORNE RADIOACTIVITY (DAC):
<0.30*

REQUIRED JOB COVERAGE:
ROUTINE

DOSIMETRY REQUIREMENTS:
ED/SRD TLD

DOSIMETRY COMMENTS:
1.0)Caution: If PAM(ED) is utilized, ensure PAM(ED) is secured so that individual can feel the vibration if PAM(ED) alarms.

PROTECTIVE CLOTHING REQUIREMENTS:
LAB COAT, SHOE COVERS & GLOVES MAY BE USED FOR ANY NON-PHYSICAL ACTIVITIES (I.E. OBSERVATIONS OR EQUIPMENT CHECKS) AND CONTAMINATION LEVELS ARE < 10,000 DPM/100CM2.

1.0)Required Protective Clothing:
Surgeons Hood
One Pair Coveralls
Rubber Boots
High Top Shoe Covers
Cotton Inserts
One Pair Rubber Gloves
2.0)Double PC's are required for work in areas with contamination levels greater than 100,000 dpm/100cm2

A RWP PRE-JOB BRIEFING IS REQUIRED:
HRA BRIEF BY AN HP TECHNICIAN AND SIGN ATTENDANCE SHEET.

WORKER INSTRUCTIONS:

1.0)ED Alarms:
   1.1)If ED dose rate alarm occurs, THEN leave area immediately and notify HP (unless authorized by HP Supervisor and have been briefed on proper responses).
   1.2)If ED dose alarm occurs, THEN leave area immediately and report to the Health Physics office.
2.0)Unless Continuous Health Physics Coverage is provided, workers shall read their SRD/ED approximately every 15 minutes.
3.0)When 80% of dose alarm setpoint is reached, leave work area in a safe condition and exit the RCA.
4.0)Do not enter High Radiation Areas without HP coverage or a ED with knowledge of work area dose rates.
5.0)Spent filters are to be stored according to the direction of the Supervisor RMC
6.0)Prior to returning Electronic Dosimetry (ED), all individuals shall process through the PDA for whole body monitoring. This is not required if returning ED solely for the purpose of changing RWPs.
7.0)Notify HP prior to entry in overhead areas greater than 8 feet.

HEALTH PHYSICS INSTRUCTIONS:

1.0)Stop work and leave area if whole body dose rates exceed 7500 mrem/hr.
2.0)CAUTION: Filter dose rates are typically four times the filter housing dose rates. Set the ED Dose Rate Alarm accordingly.
3.0)When the ED dose rate alarm is set >1000 mRem/hr then:
   3.1)Stay times and continuous HP coverage is required.
   3.2)If worker ED alarms, then have worker immediately exit area and report to HP.
3.0)Radiation survey requirements:
   3.1)Prejob filter housing survey.
   3.2)Of filters.
4.0)Contamination Survey requirements:
   4.1)Post Job Survey
5.0)HP-Ops to emphasize precautions associated with opening contaminated systems.
6.0)If there is no DP across filters(See Operations), Strainer cleaning may be necessary. Additional contamination controls should be implemented.
7.0)Filter change-out criteria:
   7.1)Filters are to be drained free of standing water.
   7.2)Filters are to be bagged, with an amount of absorbent material (preferably mopheads) placed in the bag equal to the filter volume.
   7.3)In NO case, should there be standing water in the bottom of the bag when placed for disposal, whether in the liner or the sea-land.
   7.4)Filters can be placed on the DAW sea-land if contact dose rates are < 150 mrem/hr ; otherwise place in the High Radiation Trash DAW liner in Waste Solids.

VALID FROM 01-JAN-2014 00:00 TO 31-DEC-2014 23:59 RWP 14-1207-3 REV. NO 0

DOSE RATE ALARM: 90 mrem/Hr	BUDGETED DOSE: 17 mrem
DOSE LIMIT ALARM: 5 mrem	ALARA EVALUATION NO: 14-001

JOB LOCATIONS:
Aux Building 259' BARS Skid

JOB DESCRIPTION: BARS Operation, Sampling & 1-RP-FL-2 Change Out

THE MAXIMUM POSTED AREA THAT CAN BE ENTERED:
Locked High Radiation Area

RADIOLOGICAL CONDITIONS: *Indicates estimated value for RWP Preparation. See survey forms for details

GENERAL AREA RADIATION LEVELS (mrem/hr):
See Current RCA Surveys

CONTACT/HOT SPOT RADIATION LEVELS (mrem/hr):
See Current RCA Surveys

CONTAMINATION LEVELS (dpm/100cm2):
See Current RCA Surveys

AIRBORNE RADIOACTIVITY (DAC):
0.3*

REQUIRED JOB COVERAGE:
ROUTINE

DOSIMETRY REQUIREMENTS:
ED/SRD TLD

PROTECTIVE CLOTHING REQUIREMENTS:
LAB COAT, SHOE COVERS & GLOVES MAY BE USED FOR ANY NON-PHYSICAL ACTIVITIES (I.E. OBSERVATIONS OR EQUIPMENT CHECKS) AND CONTAMINATION LEVELS ARE < 10,000 DPM/100CM2.

1.0) Required Protective Clothing:
- Surgeons Hood
- One Pair Coveralls
- Rubber Boots
- High Top Shoe Covers
- Cotton Inserts
- One Pair Rubber Gloves

2.0) Protective Clothing requirements as stated are for entry into "Contaminated Areas" only.

3.0) Double set of PC's are required for work in areas with contamination levels greater than 100,000 dpm/100cm2

4.0) Protective Clothing requirements for a "Hot Particle Area" (HPA) [in addition to those stated above] are:
- 4.1) Hood, gloves, coveralls, high top shoe covers and rubber boots.
- 4.2) Workers interfacing with individuals/equipment in a HPA - Gloves and face shield.

A RWP PRE-JOB BRIEFING IS REQUIRED:
HRA BRIEF BY AN HP TECHNICIAN AND SIGN ATTENDANCE SHEET.

WORKER INSTRUCTIONS:

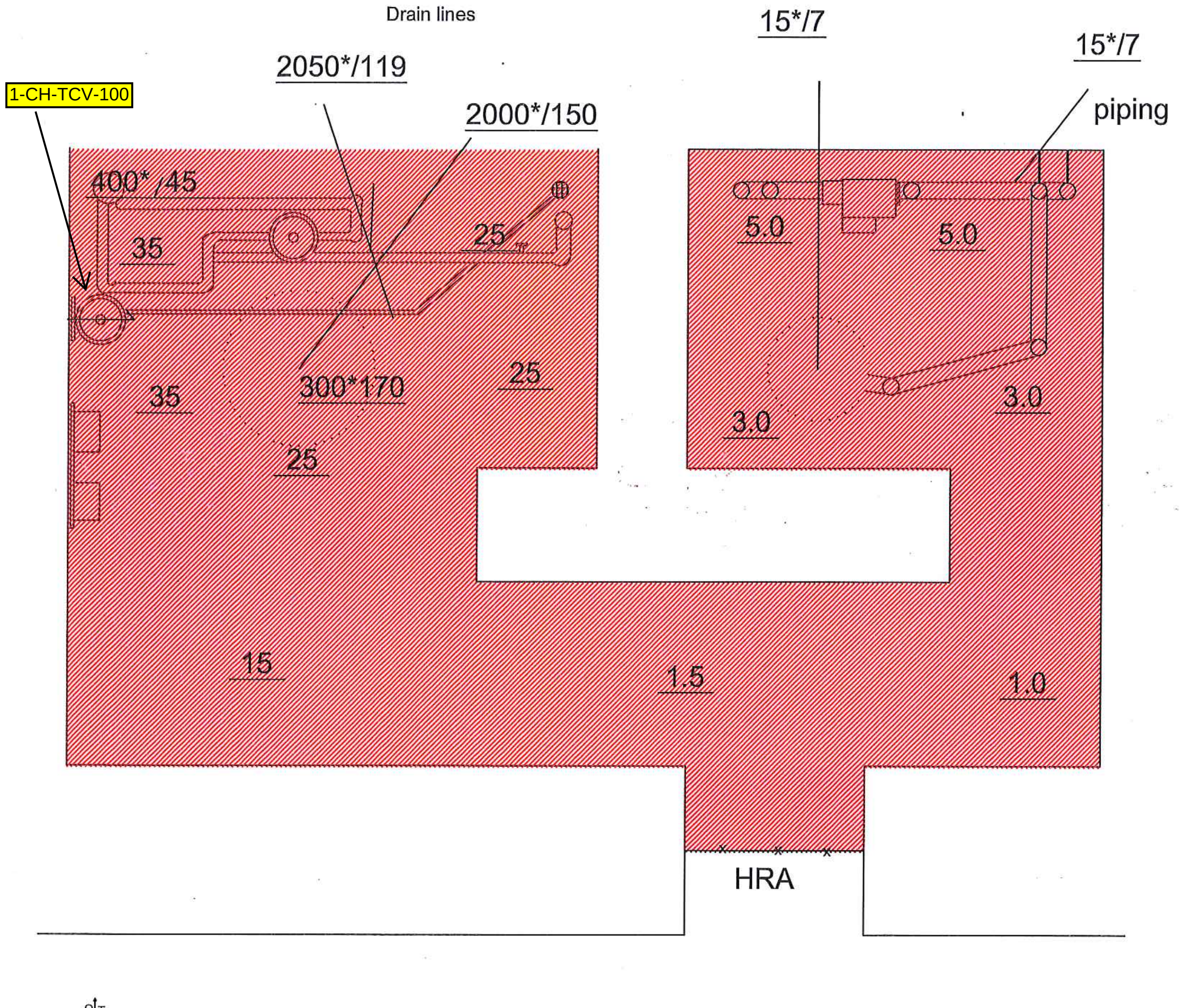
1.0)ED Alarms:
  1.1)If ED dose rate alarm occurs, THEN leave area immediately and notify HP (unless authorized by HP Supervisor and have been briefed on proper responses).
  1.2)If ED dose alarm occurs, THEN leave area immediately and report to the Health Physics office.
2.0)Unless Continuous Health Physics Coverage is provided, workers shall read their SRD/ED approximately every 15 minutes.
3.0)When 80% of dose alarm setpoint is reached, leave work area in a safe condition and exit the RCA.
4.0)Do not enter High Radiation Areas without HP coverage or a ED with knowledge of work area dose rates.
5.0)Workers are responsible for notifying the HP Supervisor or Lead Technician prior to venting\draining systems, that may affect Radiological Conditions in an area, to ensure proper Health Physics Monitoring.
6.0)HPA exit instructions:
  6.1)Use extreme care in removing PCs and frisking.
  6.2)Workers are to be monitored by HP upon exiting the HPA.
  6.3)Workers are to be monitored by HP after interfacing with workers/equipment in a HPA.
7.0)Do not remove any items from a posted "Hot Particle Area" until authorized by HP-Ops.
8.0)Prior to returning Electronic Dosimetry (ED), all individuals shall process through the PDA for whole body monitoring. This is not required if returning ED solely for the purpose of changing RWPs.
9.0)Notify HP prior to entry in overhead areas greater than 8 feet.

HEALTH PHYSICS INSTRUCTIONS:

1.0)Stop work and leave area if whole body dose rates exceed 1500 mrem/hr.
2.0)When the ED dose rate alarm is set >1000 mRem/hr then:
  2.1)Stay times and continuous HP coverage is required.
  2.2)If worker ED alarms, then have worker immediately exit area and report to HP.
3.0)Radiation survey requirements:
  3.1)Prejob filter housing survey.
  3.2)Of filter.
4.0)Contamination Survey requirements:
  4.1)Post Job Survey
5.0)HP-Ops to emphasize precautions associated with opening contaminated systems.
6.0)Filter change-out criteria:
  6.1)Filters are to be drained free of standing water.
  6.2)Filters are to be bagged, with an amount of absorbent material (preferably mopheads) placed in the bag equal to the filter volume.
  6.3)In NO case, should there be standing water in the bottom of the bag when placed for disposal, whether in the liner or the sea-land.
  6.4)Filters can be placed on the DAW sea-land if contact dose rates are < 150 mrem/hr ; otherwise place in the High Radiation Trash DAW liner in Waste Solids.

Verified Current By	Date
MO	4/5/14

Auxiliary Building, 244' Elevation
Unit 1 Non Regen Cube



VHRA = Very High Radiation Area
LHRA = Locked High Radiation Area
HRA = High Radiation Area
RA = Radiation Area
RCA = Radiological Control Area
_#*_ / _#_ Contact and 30 cm Gamma Dose Rate Readings

HPA = Hot Particle Area
CA = Contaminated Area
ARA = Airborne Radioactivity Area
RMA = Radioactive Material Area
-xxx- Radiological Boundary

CAM = Continuous Air Monitor
SOP = Step Off Pad
LDWA = Low Dose Waiting Area
NEA = Neutron Exposure Area
# N = Neutron Dose Rate
# β = Corrected Beta Dose Rate

Dominion
North Anna Power Station
ADMINISTRATIVE JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

INITIAL CONDITIONS

Both units were at 100% power.

High winds from a severe thunderstorm caused a loss of all off-site power.

All emergency diesel generators started and loaded as designed.

Unit 2 is in 2-ES-0.1 and is stable.

A tube rupture occurred on the unit 1 “A” steam generator.

Safety injection automatically initiated on unit 1.

The unit 1 “A” SG PORV is open and cannot be closed from the control room or locally.

INITIATING CUE

You are requested to classify the emergency event in accordance with EPIP-1.01.

Do NOT use SEM Judgment to classify the event.

This is a time critical JPM.

Record your results in the spaces provided.

Emergency Classification ____

EAL Identifier ____

Dominion
North Anna Power Station
ADMINISTRATIVE JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

TASK

Classify an emergency event.

TASK STANDARDS

Event is classified as a Site Area Emergency (FS1.1) in accordance with EPIP-1.01 and the applicable Emergency Level Action Matrix (Hot Conditions > 200°F). Classification is made within 15 minutes of starting the evaluation.

K/A REFERENCE:

G2.4.41 (4.6)

ALTERNATE PATH:

N/A

TASK COMPLETION TIMES

Validation Time = 12 minutes   Start Time = ________
Actual Time = ________ minutes   Stop Time = ________

PERFORMANCE EVALUATION

Rating	[ ] SATISFACTORY	[ ] UNSATISFACTORY
Candidate (Print)	________________________	
Evaluator (Print)	________________________	
Evaluator's Signature / Date	________________________	

EVALUATOR'S COMMENTS

____

____

____

Dominion
North Anna Power Station
ADMINISTRATIVE JOB PERFORMANCE MEASURE EVALUATION

OPERATOR PROGRAM

READ THE APPLICABLE INSTRUCTIONS TO THE CANDIDATE

Instructions for Simulator JPMs

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

Instructions for In-Plant JPMs

I will explain the initial conditions, and state the task to be performed. All steps, including any required communications, shall be simulated for this JPM. Under no circumstances are you to operate any plant equipment. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS

Both units were at 100% power.

High winds from a severe thunderstorm caused a loss of all off-site power.

All emergency diesel generators started and loaded as designed.

Unit 2 is in 2-ES-0.1 and is stable.

A tube rupture occurred on the unit 1 "A" steam generator.

Safety injection automatically initiated on unit 1.

The unit 1 "A" SG PORV is open and cannot be closed from the control room or locally.

INITIATING CUE

You are requested to classify the emergency event in accordance with EPIP-1.01.

Do NOT use SEM Judgment to classify the event.

This is a time critical JPM.

Record your results in the spaces provided.

Emergency Classification ____

EAL Identifier ____

EVALUATION METHOD

Demonstration if conducted in the simulator or in a laboratory (use DEMONSTRATION cues)

Verbal-visual if conducted in the station or on a dead simulator (use VERBAL-VISUAL cues)

TOOLS AND EQUIPMENT

EPIP-1.01

EAL Matrix

EAL Basis Document

PERFORMANCE STEPS

START TIME ___________

START TIME IS CRITICAL: Start time begins the 15 minute clock for event classification (clock stops when classification is made – element 4 of JPM).

1	Operator determines the Event Category on the applicable Emergency Action Level Matrix.	Procedure Step N/A
	Critical Step	SAT [ ] UNSAT [ ]

Standards	Event is identified as Fission Product Barriers, Category F, of the Hot Conditions > 200°F Matrix.

Notes/Comments

2	Operator reviews the Emergency Action Level Matrix associated with the Event Category.	Procedure Step N/A

SAT [ ] UNSAT [ ]

Standards	Category F reviewed to determine highest level of classification.

Notes/Comments

3	Operator uses available resources to obtain indications of emergency conditions.	Procedure Step N/A

SAT [ ] UNSAT [ ]

Standards	Category F emergency condition descriptions are compared to the Initial Conditions. • Table F-1 is reviewed to determine that RCS Barrier D.3 and Containment Barrier D.3 apply (loss of two barriers). • Category F is reviewed to determine that FS1.1 applies (loss or potential loss of any two barriers).

Notes/Comments SU1.1 may be noted as a lower level classification.

4	Operator classifies event based on emergency action level being exceeded.	Procedure Step N/A

Critical Step	SAT [ ] UNSAT [ ]

Standards	Event is classified as a Site Area Emergency IAW EAL Identifier FS1.1 within 15 minutes of the start of the evaluation.

Notes/Comments

>>>>> END OF EVALUATION <<<<<

STOP TIME ___________

Dominion

NORTH ANNA POWER STATION

EMERGENCY PLAN IMPLEMENTING PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
EPIP-1.01	EMERGENCY MANAGER CONTROLLING PROCEDURE (WITH 2 ATTACHMENTS)	49
		PAGE 1 of 9

PURPOSE

To assess potential emergency conditions and initiate corrective actions.

ENTRY CONDITIONS

Any of the following.

1) Another station procedure directs initiation of this procedure.

2) A potential emergency condition is reported to the Shift Manager.

COMMON

REFERENCE USE

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: Declaration of the highest emergency class for which an Initiating Condition and Emergency Action Level is met shall be made.

* *

NOTE: The PCS is potentially unreliable in the event of an earthquake. Therefore, PCS parameters should be evaluated for accuracy should this situation occur.

____ 1 EVALUATE EMERGENCY ACTION LEVELS:

a) Determine event category using the applicable Emergency Action Level Matrix:

- ☐ • Hot Conditions (RCS > 200 °F)
- ☐ • Cold Conditions (RCS ≤ 200 °F)

b) Review both of the following:

- ☐ • Initiating Condition (IC)

AND

- ☐ • EAL associated with the event category

☐ c) Use Control Room monitors, PCS, and outside reports to get indications of emergency conditions listed in the EAL Matrix

d) Check both of the following CURRENTLY MET:

- ☐ • Initiating Condition (IC)

AND

- ☐ • EAL

d) IF IC AND EAL - NOT MET, THEN do the following:

- ☐ • RETURN TO procedure in effect.
- ☐ • GO TO VPAP-2802, NOTIFICATIONS AND REPORTS, to make one-hour, non-emergency reports for classification without declaration.
- ☐ • GO TO Step 11.

☐ e) Initiate a chronological log of events

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

____ 2 RECORD EAL IDENTIFIER, TIME EMERGENCY DECLARED AND SM/SEM NAME:

Emergency Classification	**EAL Identifier**	**Time Declared**	**SM / SEM Name**
Notification of Unusual Event			
Alert			
Site Area Emergency			
General Emergency			

____ 3 ANNOUNCE THE FOLLOWING DECLARATIONS:

- ☐ • Station Emergency Manager position
- ☐ • Emergency Classification
- ☐ • EAL
- ☐ • Time Declared

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Assembly, accountability and/or initiation of facility staffing may not be desired during certain situations (e.g., security event or condition, severe weather, anticipated grid disturbance) or may have already been completed. These activities should be implemented as quickly as achievable given the specific situation.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 4	CHECK - CONDITIONS ALLOW FOR NORMAL IMPLEMENTATION OF EMERGENCY RESPONSE ACTIONS	<u>IF</u> deviation from normal emergency response actions warranted, <u>THEN</u> do the following: ☐ a) Refer to Attachment 1, Considerations for Operations Response Under Abnormal Conditions. ☐ b) Consider applicability of 50.54(x). ☐ c) <u>IF</u> classification/assembly announcement deferred, <u>THEN</u> GO TO Step 6.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 5	NOTIFY PLANT STAFF OF ALERT OR HIGHER CLASSIFICATION:	
	☐ a) Check classification - ALERT OR HIGHER	☐ a) GO TO Step 6.
	☐ b) Check if emergency assembly and accountability - PREVIOUSLY CONDUCTED	b) Do the following: ☐ 1) Have Control Room sound EMERGENCY alarm and make announcement on station Gai-Tronics system as follows: "(Emergency classification) has been declared as the results of ______________________" (event) "All Emergency Response personnel report to your assigned stations" "All supplemental personnel not responding to the emergency and all visitors report to the Security Building" "All other personnel report to your Emergency Assembly Areas" ☐ 2) Repeat RNO Step 5.b. ☐ 3) GO TO Step 6.
	☐ c) Have Control Room sound EMERGENCY alarm and make announcement on station Gai-Tronics system as follows: "(Emergency classification) has been declared as the result of ______________________" (event)	
	☐ d) Repeat Step 5.c	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: Continue this and all further instructions unless otherwise directed to hold.

* *

____ 6 INITIATE SUPPORTING PROCEDURES:

a) Determine if a radiological release is in progress:

☐ Radioactive material not attributable to normal plant operations detected beyond the protected area

OR

☐ Radioactive material not attributable to normal plant operations suspected of migrating beyond the protected area

RESPONSE NOT OBTAINED:

☐ a) IF radiological release NOT in progress, THEN GO TO Step 6.b.

b) Inform Emergency Communicators of the following:

☐ • Emergency Classification

☐ • Emergency Action Level

☐ • Time of Declaration

☐ • Radiological release status

☐ • PARs, if applicable

c) Direct Emergency Communicators to initiate the following procedures:

☐ 1) EPIP-2.01, NOTIFICATION OF STATE AND LOCAL GOVERNMENTS

☐ 2) EPIP-2.02, NOTIFICATION OF NRC

(STEP 6 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	INITIATE SUPPORTING PROCEDURES: (Continued)	
☐	d) Direct HP to initiate EPIP-4.01, RADIOLOGICAL ASSESSMENT DIRECTOR CONTROLLING PROCEDURE	
	e) Establish communications with Security Team Leader:	
☐	1) Provide Security with current emergency classification	
☐	2) Notify Security which Operations Shift is designated for coverage	
☐	3) Direct Security to initiate EPIP-5.09, SECURITY TEAM LEADER CONTROLLING PROCEDURE	
____ 7	CHECK TSC - ACTIVATED	<u>IF</u> TSC - NOT ACTIVATED, <u>THEN</u> do the following: ☐ a) Have STA report to the Control Room. ☐ b) Notify Manager Nuclear Operations or Operations Manager On Call. ☐ c) Consider having Radiological Assessment Director report to the Control Room. ☐ d) <u>WHEN</u> relief SEM arrives, <u>THEN</u> perform turnover using EPIP-1.01, Attachment 2, Turnover Checklist.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

____ 8 IMPLEMENT EPIP FOR EMERGENCY CLASSIFICATION IN EFFECT:

- ☐ • **Notification of Unusual Event** - GO TO EPIP-1.02, RESPONSE TO NOTIFICATION OF UNUSUAL EVENT
- ☐ • **Alert** - GO TO EPIP-1.03, RESPONSE TO ALERT
- ☐ • **Site Area Emergency** - GO TO EPIP-1.04, RESPONSE TO SITE AREA EMERGENCY
- ☐ • **General Emergency** - GO TO EPIP-1.05, RESPONSE TO GENERAL EMERGENCY

____ 9 NOTIFY OFF-SITE AUTHORITIES OF EMERGENCY TERMINATION:

- ☐ a) State and local governments (made by LEOF or CEOF when activated)
- ☐ b) NRC

____10 NOTIFY STATION PERSONNEL ABOUT THE FOLLOWING:

- ☐ • Emergency termination
- ☐ • Facility de-activation
- ☐ • Selective release of personnel
- ☐ • Completion and collection of procedures
- ☐ • Recovery

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 11	TERMINATE EPIP-1.01:	
	☐ • Give completed EPIPs, forms and other applicable records to Nuclear Emergency Preparedness (TSC Emergency Procedures Coordinator if TSC activated)	☐ • Give to STA AND ☐ • Notify Nuclear Document Management that used EPIPs require replacement.
	☐ • Completed By: ____________________ Date: _____________ Time: _____________	

- END -

This attachment provides procedural guidance for controlling selected emergency response actions when their implementation would have adverse results. Station Emergency Manager (SEM) approval is required before any required action is postponed, suspended or modified. The guidance below is not all-inclusive.

UNANTICIPATED HAZARD EXISTS (e.g., security event, tornado or toxic release):

IF notifying off-duty augmentation could create a safety hazard for personnel coming to the station, THEN consider the following alternatives:

- Station Security (if available) can be directed to notify off-duty personnel to report to the remote mustering area (Louisa Fire Training Center).
- Corporate Security, at 804-771-3161 (Tie Line 8-736-3161) or 804-771-3158 (Tie Line 8-736-3158), can be directed to notify off-duty personnel to report to the remote mustering area (Louisa Fire Training Center).
- Corporate Security, at 804-771-3161 (Tie Line 8-736-3161) or 804-771-3158 (Tie Line 8-736-3158), can be directed to notify corporate emergency response organization only using CPIP-3.4, CORPORATE SECURITY SUPPORT.
- Notifications can be deferred until hazardous conditions are resolved.

IF implementation of emergency response actions could compromise Security Plan response strategies, THEN consider postponing or suspending emergency response actions until threat has been resolved, e.g., on-site announcement directing assembly and emergency response facility activation, pager activation and call-out per EPIP-3.05, AUGMENTATION OF EMERGENCY RESPONSE ORGANIZATION and implementation of EPIP-5.04, ACCESS CONTROL.

IF assembling on-site personnel for accountability or activation of emergency response facilities could endanger plant personnel, THEN consider postponing emergency assembly until hazardous conditions are resolved. Corporate Security, at 804-771-3161 (Tie Line 8-736-3161) or 804-771-3158 (Tie Line 8-736-3158), can be directed to notify corporate emergency response organization only using CPIP-3.4, CORPORATE SECURITY SUPPORT. Personnel in unaffected areas on-site can be notified selectively.

IF primary ingress/egress route - NOT AVAILABLE, THEN evaluate alternate route for use during site evacuation or off-duty augmentation (e.g., access via Dyke 1).

ANTICIPATED SITUATION (e.g., forecasted severe weather or grid disturbance):

IF all or part of the ERO has been staged in anticipation of a predicted event, THEN notify Security to omit performance of augmentation notification (as described in EPIP-3.05, AUGMENTATION OF EMERGENCY RESPONSE ORGANIZATION).

IF adequate controls have been established to continually account for personnel staged in anticipation of a predicted event, THEN notify Security to omit performance of initial accountability (as described in EPIP-5.03, PERSONNEL ACCOUNTABILITY).

IF environmental conditions are hazardous, THEN consult with Security Team Leader about suspending procedural requirements for implementing EPIP-5.04, ACCESS CONTROL.

Conduct a turnover between the onshift and relief SEM in accordance with the following checklist. Use place-keeping aid at left of item, "_____________", to track completion.

____ 1. Determine the status of primary responder notification.

____ 2. Determine the status of "Report of Emergency to State and Local Governments," EPIP-2.01. Get completed copies if available.

____ 3. Determine status of the "Report of Radiological Conditions to the State," EPIP-2.01, Attachment 3. Get completed copy if available.

____ 4. Determine status of Emergency Notification System (ENS) communications and completion status of NRC Event Notification Worksheet (EPIP-2.02, Attachment 1).

____ 5. Review classification and initial PAR status.

____ 6. Review present plant conditions and status. Get copy of Critical Safety Functions form.

____ 7. Review status of station firewatches and re-establish if conditions allow.

____ 8. Determine readiness of TSC for activation.

____ 9. After all information is obtained, transfer location to TSC.

IF TSC - FUNCTIONAL, THEN the State and Local Communicator in the Control Room will relocate to TSC with the SEM.

IF TSC - NON-FUNCTIONAL, THEN the responsibilities may be transferred to relief in another facility, e.g., LEOF/CEOF.

____10. Call the Control Room and assess any changes that may have occurred during transition to the TSC.

____11. When sufficient personnel are available, the relief SEM is to assume the following responsibilities from the onshift Station Emergency Manager:

a. Reclassification.

b. Protective Action Recommendations until LEOF activated.

c. Notifications (i.e., state, local, & NRC). Upon LEOF activation, transfer notification responsibilities except for the NRC ENS.

d. Site evacuation authorization.

e. Emergency exposure authorization.

f. Command/control of on-site response.

____12. Direct the Shift Manager to notify the TSC of any personnel dispatched by the Control Room (including name, destination and purpose).

____13. Formally relieve the Interim SEM and assume control in the TSC. Announce name and facility activation status to facility.

North Anna Power Station Emergency Action Level Matrix — Hot Conditions (RCS > 200°F)

Columns: **GENERAL EMERGENCY** | **SITE AREA EMERGENCY** | **ALERT** | Notification of **UNUSUAL EVENT**

R — Abnorm. Rad Release / Rad Effluent

1 — Offsite Rad Conditions

General Emergency

RG1 Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RG1.1
Valid reading on **any** gaseous radiation monitors that exceeds or is expected to exceed Table R-1 column "GE" for ≥ 15 min. (Note 1 & 2)

RG1.2
Dose assessment using actual meteorology indicates doses at or beyond the Site Boundary of **EITHER**:
> 1000 mRem TEDE
OR
> 5000 mRem Thyroid CDE

RG1.3
Field survey indicates closed window dose rate > 1,000 mRem/hr that is expected to continue for > 1 hr at or beyond the site boundary
OR
Field survey sample analysis indicates Thyroid CDE of > 5,000 mRem for 1 hr of inhalation at or beyond the site boundary

Site Area Emergency

RS1 Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RS1.1
Valid reading on **any** gaseous radiation monitors that exceeds or is expected to exceed Table R-1 column "SAE" for ≥ 15 min. (Note 1& 2)

RS1.2
Dose assessment using actual meteorology indicates doses at or beyond the Site Boundary of **EITHER**:
> 100 mRem TEDE
OR
> 500 mRem Thyroid CDE

RS1.3
Field survey indicates closed window dose rate > 100 mRem/hr that is expected to continue for > 1 hr at or beyond the site boundary
OR
Field survey sample analysis indicates Thyroid CDE of > 500 mRem for 1 hr of inhalation at or beyond the site boundary

Alert

RA1a **Any** unplanned release of gaseous or liquid radioactivity to the environment that exceeds 200 times the radiological effluent Technical Specifications for 15 minutes or longer

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RA1.1
Valid reading on Circulating Water Discharge Tunnel Monitor SW-RM-130 (SW-RM-230) > 200 times the Hi-Hi alarm setpoint or offscale high for ≥ 15 min. (Note 2 & 6)

RA1.2
Valid reading on **any** gaseous monitors > Table R-1 column "Alert" for ≥ 15 min. (Note 2)

RA1.3
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x Technical Specification limits for ≥ 15 min. (Note 2)

Unusual Event

RU1a **Any** unplanned release of liquid radioactivity to the environment that exceeds two times the radiological effluent Technical Specifications for 60 minutes or longer

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RU1.1
Valid reading on Circulating Water Discharge Tunnel Monitor SW-RM-130 (SW-RM-230) > 2 times the Hi-Hi alarm setpoint for ≥ 60 min. (Note 2 & 6)

RU1.2
Confirmed sample analyses for liquid releases indicate concentrations or release rates > 2 x Technical Specification limits for ≥ 60 min. (Note 2)

RU1b **Any** unplanned release of gaseous radioactivity to the environment that exceeds two times the allocated radiological effluent ODCM limits for 60 minutes or longer

RU1.3
Valid reading on **any** gaseous monitors > Table R-1 column "NOUE" for ≥ 60 min. (Note 2)

RU1.4
Confirmed sample analyses for gaseous releases indicate concentrations or release rates > 2 x the allocated ODCM limits for ≥ 60 min. (Note 2)

2 — Onsite Rad Conditions

General Emergency: None

Site Area Emergency: None

Table R-1 Gaseous Effluent Monitor Classification Thresholds

Release Point	Monitor	GE	SAE	Alert	NOUE
Vent Stack A	VG-RI-179-1 or 2	4.00E+08 μCi/sec	4.00E+07 μCi/sec	4.56E+06 μCi/sec	3.60E+05 μCi/sec
Vent Stack B	VG-RI-180-1 or 2	3.57E+08 μCi/sec	3.57E+07 μCi/sec	4.07E+06 μCi/sec	3.60E+05 μCi/sec
Process Vent	GW-RI-178-1 or 2	3.70E+08 μCi/sec	3.70E+07 μCi/sec	4.22E+06 μCi/sec	2.80E+05 μCi/sec
Main Steam (Steam Safety) (Note 8)	MS-RM-170 (270) MS-RM-171 (271) MS-RM-172 (272)	8.62E+02 mR/hr	8.62E+01 mR/hr	9.81E+00 mR/hr	N/A
AFWPT Exhaust (Note 8)	MS-RM-176 (276)	2.84E+02 mR/hr	2.84E+01 mR/hr	3.24E+00 mR/hr	N/A

Alert

RA2a Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RA2.1
Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel resulting in a valid Hi-Hi alarm on **any** of the following radiation monitors:
- RM-RMS-153 Fuel Pit Bridge
- RM-RMS-152 New Fuel Storage Area
- RM-RMS-162 (RM-RMS-262) Manipulator Crane Area
- RM-RMS-163 (RM-RMS-263) Containment Area
- RM-RMS-159 (RM-RMS-259) Containment Particulate
- RM-RMS-160 (RM-RMS-260) Containment Gaseous
- GW-RI-178-1 Process Vent Normal Range

RA2.2
A water level drop in the reactor refueling cavity, spent fuel pit or fuel transfer canal that will result in irradiated fuel becoming uncovered

RA2b Release of radioactive material or increases in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain cold shutdown

RA2.3
Valid radiation monitor or survey reading > 1.50E-02 R/hr (15 mR/hr) in areas requiring continuous occupancy to maintain plant safety functions:
Control Room RM-RMS-157
OR
Central Alarm Station (CAS)

Unusual Event

RU2 Unexpected Increase in plant radiation

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RU2.1
Valid low water level alarm or visual observation indicating uncontrolled water level decrease in the refueling cavity, spent fuel pit or fuel transfer canal with **all** irradiated fuel assemblies remaining covered by water
AND
Unplanned valid direct area radiation reading increases resulting in a valid Hi alarm on **any** of the following radiation monitors:
- RM-RMS-162 (RM-RMS-262) - Manipulator Crane Area
- RM-RMS-152 - New Fuel Storage Area
- RM-RMS-153 - Fuel Pit Bridge

RU2.2
Unplanned valid direct area radiation monitor reading increases by a factor of 1000 over normal* levels

* Normal levels can be considered as the highest reading in the past 24 hours excluding the current peak value

H — Hazards

1 — Natural & Destructive Phenomena

General Emergency: None

Site Area Emergency: None

Table H-1 Safe Shutdown Areas
- Cable Vaults & Tunnels
- Emergency Switchgear Rooms
- Emergency Diesel Generators Rooms
- Reactor Containment
- Quench Spray Pump Houses
- Safeguards Areas
- Main Steam Valve House
- Cable Spreading Rooms
- Control Room
- CR Chiller Rooms
- Auxiliary / Fuel / Decontamination Buildings
- Fuel Oil Pump House Room A or B
- Service Water Pump and Valve House
- Intake Structure Control House
- Auxiliary Service Water Pump House
- Turbine Building
- Auxiliary Feedwater Pump House

Alert

HA1 Natural or destructive phenomena affecting a plant safe shutdown area

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA1.1
"OBE EXCEEDED" indicator illuminated on the SYSCOM Network Control Center (NCC)
AND
Earthquake confirmed by **any** of the following:
- Earthquake felt in plant
- National Earthquake Information Center (NEIC)
- Control Room indication of degraded performance of any safety-related structure, system, or component

HA1.2
Tornado or high winds > 80 mph resulting in **EITHER**:
Visible damage to any safety-related structure, system, or component within **any** Table H-1 Area
OR
Control Room indication of degraded performance of any safety-related structure, system, or component

HA1.3
Turbine failure-generated missiles resulting in **EITHER**:
Any visible damage to **any** safety-related structure, system, or component within **any** Table H-1 area
OR
Control Room indications of degraded performance of those safety systems resulting from turbine failure-generated missiles

HA1.4
Uncontrolled flooding resulting in **EITHER**:
Control Room indications of degraded performance of safety-related structure, system or component within **any** Table H-1 area
OR
Creating an industrial safety hazard (e.g. electric shock) in **any** Table H-1 area that precludes access necessary to operate or monitor **any** safety-related structure, system or component

HA1.5
Service Water Reservoir (Service Water Pump House) level < 309 ft.
OR
Lake level > 271 ft. (station finished ground grade)
OR
Class I Dam Emergency

HA1.6
Vehicle crash resulting in **EITHER**:
Visible damage to **any** safety-related structure, system, or component within **any** Table H-1 area
OR
Control Room indication of degraded performance of any safety-related structure, system, or component

Unusual Event

HU1 Natural or destructive phenomena affecting the Protected Area or Main Dam

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU1.1
Seismic event identified by any **TWO** of the following:
- Earthquake felt in the plant
- "SYSTEM TRIGGER" indicator illuminated on the SYSCOM Network Control Center (NCC)
- National Earthquake Information Center (NEIC)

HU1.2
Report by plant personnel of tornado or high winds > 80 mph striking within Protected Area boundary

HU1.3
Report of turbine failure resulting in casing penetration or damage to turbine seals or turbine generator seals

HU1.4
Uncontrolled flooding in **any** Table H-1 area that has the potential to affect safety related equipment needed for the current operating mode

HU1.5
Lake level < 242 ft. **AND** actions required in TRM 3.7.4 not completed
OR
Lake level > 264 ft.
OR
Class II Dam Emergency

2 — Fire or Explosion

General Emergency: None

Site Area Emergency: None

Alert

HA2 Fire or explosion affecting the operability of plant safety-related structures, systems or components required to establish or maintain safe shutdown

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA2.1
Fire or explosion in **any** Table H-1 area
AND EITHER:
Plant personnel report visible damage to **any** safety-related structure, system, or component within the area
OR
Affected system parameter indications show degraded performance

Unusual Event

HU2 Fire or explosion within the Protected Area boundary

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU2.1
Fire in or restricting access to **any** Table H-1 area **not** extinguished within 15 minutes of Control Room notification or verification of a Control Room alarm (Note 3)

HU2.2
Report by plant personnel of an unanticipated explosion within Protected Area boundary, SWPH, SWVH or Auxiliary SWPH resulting in visible damage to permanent structure or equipment

3 — Toxic, Corrosive, Asphyxiant & Flammable Gas

General Emergency: None

Site Area Emergency: None

Alert

HA3 Access to a safe shutdown area is prohibited due to release of toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA3.1
Access to a Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor (Note 9)

Unusual Event

HU3 Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal operation of the plant

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU3.1
Report or detection of toxic, corrosive, asphyxiant or flammable gases that have or could enter the Owner Controlled Area in amounts that can affect normal plant operations

HU3.2
Recommendation by local, county or state officials to evacuate or shelter site personnel based on offsite event involving toxic, corrosive, asphyxiant or flammable gas

4 — Security

General Emergency

HG4 Hostile action resulting in loss of physical control of the facility

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HG4.1
A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions
OR
A hostile action has caused failure of Spent Fuel Cooling Systems and imminent fuel damage is likely for a freshly off-loaded reactor core in pool (i.e. freshly off-loaded reactor core means **any** spent fuel in the Spent Fuel Pit)

Site Area Emergency

HS4 Hostile action within the Protected Area

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HS4.1
A hostile action is occurring or has occurred within the Protected Area as reported by the Security Shift Supervisor

Alert

HA4 Hostile action within the Owner Controlled Area or airborne attack threat

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA4.1
A hostile action is occurring or has occurred within the Owner Controlled Area as reported by the Security Shift Supervisor
OR
A validated notification from NRC of an airliner attack threat < 30 min. away

Unusual Event

HU4 Confirmed security condition or threat which indicates a potential degradation in the level of safety of the plant

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU4.1
A security condition that does **not** involve a hostile action as reported by the Security Shift Supervisor
OR
A credible site-specific security threat notification
OR
A validated notification from NRC providing information of an aircraft threat

5 — Control Room Evacuation

General Emergency: None

Site Area Emergency

HS5 Control Room evacuation has been initiated and plant control **cannot** be established

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HS5.1
Control Room evacuation has been initiated
AND
Control of the plant **cannot** be established from the Auxiliary Shutdown Panel within 15 min. (Note 3)

Alert

HA5 Control Room evacuation has been initiated

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA5.1
Control Room evacuation has been initiated

Unusual Event: None

6 — Judgment

General Emergency

HG6 Other conditions existing which in the judgment of the SEM warrant declaration of General Emergency

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HG6.1
Other conditions exist which in the judgment of the SEM indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or hostile action that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) offsite for more than the immediate site area.

Site Area Emergency

HS6 Other conditions existing which in the judgment of the SEM warrant declaration of Site Area Emergency

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HS6.1
Other conditions exist which in the judgment of the SEM indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or hostile action that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. **Any** releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the site boundary.

Alert

HA6 Other conditions existing which in the judgment of the SEM warrant declaration of an Alert

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA6.1
Other conditions exist which in the judgment of the SEM indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of hostile action. **Any** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE).

Unusual Event

HU6 Other conditions existing which in the judgment of the SEM warrant declaration of a NOUE

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU6.1
Other conditions exist which in the judgment of the SEM indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety-related structures, systems or components occurs.

E — ISFSI

General Emergency: None

Site Area Emergency: None

Alert: None

Unusual Event

EU1 Damage to a loaded cask confinement boundary

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

EU1.1
Damage to a loaded cask/canister confinement boundary

Modes:

1	2	3	4	5	6
Power Operation	Startup	Hot Standby ≥ 350 °F	Hot Shutdown > 200 °F	Cold Shutdown ≤ 200 °F	Refueling

Dominion

North Anna Power Station
Emergency Action Level Matrix
Revision 5

S — System Malfunct.

1 — Loss of Power

General Emergency

SG1 Prolonged loss of **all** offsite power and prolonged loss of **all** onsite AC power to emergency busses

Modes: 1 | 2 | 3 | 4

SG1.1
Loss of **all** offsite and onsite AC power to Unit 1 (Unit 2) 4160-Volt emergency busses H and J
AND EITHER:
Restoration of **any** 4160-Volt emergency bus within 4 hours is **not** likely
OR
CSFST Core Cooling-RED or ORANGE path (Note 7)

Site Area Emergency

SS1a Loss of **all** offsite power and loss of **all** onsite AC power to emergency busses

Modes: 1 | 2 | 3 | 4

SS1.1
Loss of **all** offsite and onsite AC power to Unit 1 (Unit 2) 4160-Volt emergency busses H and J for > 15 min. (Note 3)

SS1b Loss of **all** vital DC power

SS1.2
Loss of **all** vital DC power based on < 105-Volt DC bus voltage indications for > 15 min. (Note 3)

Alert

SA1 AC power capability to emergency busses reduced to a single power source for greater than 15 minutes such that **any** additional single failure would result in loss of **all** AC power to emergency busses

Modes: 1 | 2 | 3 | 4

SA1.1
AC power capability to Unit 1 (Unit 2) 4160-Volt emergency busses H and J reduced to a single power source for > 15 min. (**any** additional single failure would result in loss of **all** AC power to emergency busses) (Note 3)

Unusual Event

SU1 Loss of **all** offsite power to emergency busses for greater than 15 minutes

Modes: 1 | 2 | 3 | 4

SU1.1
Loss of **all** offsite AC power to Unit 1 (Unit 2) 4160-Volt emergency busses H and J for >15 min. (Note 3)

2 — RPS Failure

General Emergency

SG2 Automatic trip and **all** manual actions fail to shutdown the reactor and indication of an extreme challenge to the ability to cool the core exists

Modes: 1 | 2

SG2.1
An automatic trip failed to shutdown the reactor and **all** manual actions **do not** shutdown the reactor as indicated by reactor power ≥ 5%
AND EITHER:
CSFST Core Cooling-RED
OR
CSFST Heat Sink-RED

Site Area Emergency

SS2 Automatic Trip fails to shutdown the reactor and manual actions taken from the reactor control console are **not** successful in shutting down the reactor

Modes: 1 | 2

SS2.1
An automatic trip failed to shutdown the reactor and manual actions taken at the Main Control Room (MCR) Bench Board **do not** shutdown the reactor as indicated by reactor power ≥ 5%

Alert

SA2 Automatic trip fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor

Modes: 1 | 2

SA2.1
An automatic trip failed to shutdown the reactor and manual actions taken at the Main Control Room (MCR) Bench Board successfully shutdown the reactor as indicated by reactor power < 5%

Unusual Event: None

3 — Inability to Reach or Maintain Shutdown Conditions

General Emergency: None

Site Area Emergency: None

Alert: None

Unusual Event

SU3 Inability to reach required shutdown within Technical Specification limits

Modes: 1 | 2 | 3 | 4

SU3.1
Plant is **not** brought to required operating mode within Technical Specifications LCO action statement time

4 — Instrumentation / Communications

General Emergency: None

Table S-1 Significant Transient
- Automatic turbine runback > 25% thermal reactor power
- Electrical load rejection > 25% full electrical load
- Reactor trip
- Safety injection activation
- Thermal power oscillations of > 10%

Site Area Emergency

SS4 Inability to monitor a significant transient in progress

Modes: 1 | 2 | 3 | 4

SS4.1
Loss of **most** (~75%) or **all** annunciators (Panels "A" thru "N") associated with safety-related structures, systems and components on Unit 1 (Unit 2) MCR Bench Board and Vertical Board
AND
PCS is unavailable
AND
Complete loss of ability to monitor **any** critical safety function status
AND
Significant transient is in progress (Table S-1)

Alert

SA4 Unplanned loss of **most** or **all** safety-related structures, systems and components annunciation or indication in Control Room with **EITHER** (1) a significant transient in progress, **OR** (2) compensatory non-alarming indicators are unavailable

Modes: 1 | 2 | 3 | 4

SA4.1
Unplanned loss of **most** (~75%) or **all** of **EITHER**:
- Annunciators (Panels "A" thru "N")
- Indicators

associated with safety-related structures, systems and components on Unit 1 (Unit 2) MCR Bench Board and Vertical Board for > 15 min. (Note 3)
AND EITHER:
A significant transient is in progress (Table S-1)
OR
PCS is unavailable

Unusual Event

SU4a Unplanned loss of **most** or **all** safety-related structures, systems and components annunciation or indication in the Control Room for greater than 15 minutes

Modes: 1 | 2 | 3 | 4

SU4.1
Unplanned loss of **most** (~75%) or **all** of **EITHER**:
- Annunciators (Panels "A" thru "N")
- Indicators

associated with safety-related structures, systems and components on Unit 1 (Unit 2) MCR Bench Board and Vertical Board for > 15 min. (Note 3)

SU4b Unplanned loss of **all** onsite or offsite communications capabilities

SU4.2
Loss of **all** Table S-2 onsite (internal) communications capability affecting the ability to perform routine operations
OR
Loss of **all** Table S-2 offsite (external) communications capability

Table S-2 Communications Systems

System	Onsite (internal)	Offsite (external)
Radio Communications System	X	
Public Address and Intercom System	X	
Private Branch Telephone Exchange (PBX)	X	
Sound Powered Telephone System	X	
Commercial Telephone		X
Dedicated NRC Communications		X

5 — Fuel Clad Degradation

General Emergency: None

Site Area Emergency: None

Alert: None

Unusual Event

SU5 Fuel clad degradation

Modes: 1 | 2 | 3 | 4

SU5.1
Dose rate at one foot (Note 5) from **EITHER**:
1 ml RCS sample ≥ 2.3 mR/hr
OR
120 ml RCS sample ≥ 234 mR/hr

SU5.2
With letdown in service, Reactor Coolant Letdown radiation monitor CH-RI-128 (CH-RI-228) > 1.5 x 10^4 mR/hr (Note 5)

SU5.3
Coolant activity >1.0 μCi/gm Dose Equivalent I-131 for > 48 hrs.
OR
Coolant activity > 60 μCi/gm Dose Equivalent I-131

6 — RCS Leakage

General Emergency: None

Site Area Emergency: None

Alert: None

Unusual Event

SU6 RCS leakage for 15 minutes or longer

Modes: 1 | 2 | 3 | 4

SU6.1
Unidentified or pressure boundary leakage > 10 gpm for 15 minutes or longer (Note 3)
OR
Identified leakage > 25 gpm for 15 minutes or longer (Note 3)

7 — Inadvertent Criticality

General Emergency: None

Site Area Emergency: None

Alert: None

Unusual Event

SU7 Inadvertent criticality

Modes: 3 | 4

SU7.1
An unplanned sustained positive startup rate observed on nuclear instrumentation

F — Fission Product Barriers

General Emergency

Loss of **any** two barriers
AND
Loss or potential loss of third barrier

Modes: 1 | 2 | 3 | 4

FG1.1
Loss of **any** two barriers
AND
Loss or potential loss of third barrier (Table F-1)

Site Area Emergency

Loss or potential loss of **any** two barriers

Modes: 1 | 2 | 3 | 4

FS1.1
Loss or potential loss of **any** two barriers (Table F-1)

Alert

Any loss or **any** potential loss of either Fuel Clad or RCS

Modes: 1 | 2 | 3 | 4

FA1.1
Any loss or **any** potential loss of either Fuel Clad or RCS (Table F-1)

Unusual Event

Any loss or **any** potential loss of Containment

Modes: 1 | 2 | 3 | 4

FU1.1
Any loss or **any** potential loss of Containment (Table F-1)

Table F-1 Fission Product Barrier Matrix

	Fuel Cladding Barrier		Reactor Coolant System Barrier		Containment Barrier	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A.CSFST	1. CSFST Core Cooling-RED	1. CSFST Core Cooling-ORANGE **OR** CSFST Heat Sink-RED and heat sink required	None	1. CSFST RCS Integrity-RED **OR** CSFST Heat Sink-RED and heat sink required	None	1. CSFST Containment-RED
B.Core Exit TCs	2. Core exit TCs > 1,200°F	2. Core exit TCs > 700°F	None	None	* Restoration procedures are considered effective if Core exit TCs readings are lowering or Reactor Vessel water level is rising within 15 min. after restoration procedure entry (Note 3) None	2. Core exit TCs > 1,200°F **AND** Restoration procedures **not** effective* within 15 min. 3. **All** of the following: - Core exit TCs > 700°F - Reactor Vessel water level < Table F-2 thresholds - Restoration procedures **not** effective* within 15 min.
C.Radiation	3. Containment High Range Radiation Monitor > Table F-3 Fuel Clad Loss threshold 4. Dose rate at one foot from **EITHER**: 1 ml RCS sample ≥ 16 mR/hr **OR** 120 ml RCS sample ≥ 1,550 mR/hr 5. With letdown in service, Reactor Coolant Letdown radiation monitor CH-RI-128 (CH-RI-228) > 7.5 x 10^4 mR/hr	None	1. Containment High Range Radiation Monitor > Table F-3 RCS Loss threshold	None	None	4. Containment High Range Radiation Monitor > Table F-3 Containment Potential Loss threshold
D.Inventory	None	3. Reactor Vessel water level < Table F-2 thresholds	2. RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling: < 25ºF [75ºF] 3. SGTR that requires ECCS (SI) actuation	2. Unisolable RCS leak exceeding the capacity of one charging pump (150 gpm) in the normal charging mode	1. A Containment pressure rise followed by a rapid unexplained drop in Containment pressure 2. Following LOCA, Containment pressure or sump level response **not** consistent with LOCA conditions 3. Ruptured SG is also faulted outside of Containment 4. Primary-to-secondary leakage > 10 gpm with non-isolable steam release from affected SG to the environment	5. Containment pressure 60 psia and increasing 6. Containment hydrogen concentration ≥ 4% 7. Containment pressure > 28 psia with < one full train of depressurization equipment operating Note: One train of QS System and one train of RS System comprise one full train of depressurization equipment as designed
E.Other	6. Coolant activity > 300 μCi/gm Dose Equivalent I-131	None	None	None	5. CNTMT Isolation valve(s) **not** closed after **any** required CNTMT isolation **AND** Downstream pathway to the environment exists	None
F. Judgment	7. **Any** condition in the opinion of the SEM that indicates loss of the Fuel Clad barrier	4. **Any** condition in the opinion of the SEM that indicates potential loss of the Fuel Clad barrier	4. **Any** condition in the opinion of the SEM that indicates loss of the RCS barrier	3. **Any** condition in the opinion of the SEM that indicates potential loss of the RCS barrier	6. **Any** condition in the opinion of the SEM that indicates loss of the Containment barrier	8. **Any** condition in the opinion of the SEM that indicates potential loss of the Containment barrier

Notes

Note 1: If dose assessment results are available at the time of declaration, the classification should be based on dose assessment instead of radiation monitor readings. While necessary declarations should **not** be delayed awaiting results, the dose assessment should be initiated / completed in order to determine the proper classification (See EAL RS1.2/RG1.2).

Note 2: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Note 4: Not applicable to this matrix.

Note 5: If actual reactor coolant activity samples are available at the time of the radiation measurement that indicate coolant activity levels are below that specified in EAL SU5.3 (Technical Specification coolant activity limit), declaration of the NOUE is not required under this EAL threshold.

Note 6: The read value shall be rounded down to the nearest tenth digit to facilitate reading the logarithmic meter in the Control Room.

Note 7: CSFSTs should be monitored for information only.

Note 8: Due to digital display limitations for these monitors, classification must be made based on the nearest hundredth of a mR/hr read on the monitor display.

Note 9: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

Table F-2 Reactor Vessel Water Level Thresholds

RVLIS	No. RCPs	Threshold
Full Range	None	48%
Dynamic Range	3	65%
	2	41%
	1	30%

Table F-3 Containment High Range Radiation Monitor Thresholds
RM-RMS-165 (RM-RMS-265) or RM-RMS-166 (RM-RMS-266)

Time After Shutdown (hrs)	Fuel Clad Loss (R/hr)	RCS Loss (R/hr)	Containment Potential Loss (R/hr)
≤ 2	300	5	1300
>2 to 4	200	5	800
>4 to 8	100	5	500
>8 to 12	60	5	200
>12	40	5	100

EAL Identifier

XXX.X

Category (R, H, E, S, F, C)
Emergency classification (G, S, A, U)
Sequential number within subcategory/classification
Subcategory number (1 if no subcategory)

HOT CONDITIONS (RCS > 200°F)

Dominion

North Anna Power Station
Emergency Action Level Matrix
Revision 5

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | Notification of UNUSUAL EVENT

R — Abnorm. Rad Release / Rad Effluent

1 — Offsite Rad Conditions

General Emergency

RG1 Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RG1.1
Valid reading on **any** gaseous radiation monitors that exceeds or is expected to exceed Table R-1 column "GE" for ≥ 15 min. (Note 1& 2)

RG1.2
Dose assessment using actual meteorology indicates doses at or beyond the Site Boundary of **EITHER**:
> 1000 mRem TEDE
OR
> 5000 mRemThyroid CDE

RG1.3
Field survey indicates closed window dose rate > 1,000 mRem/hr that is expected to continue for > 1 hr at or beyond the site boundary
OR
Field survey sample analysis indicates Thyroid CDE of > 5000 mRem for 1 hr of inhalation at or beyond the site boundary

Site Area Emergency

RS1 Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RS1.1
Valid reading on **any** gaseous radiation monitors that exceeds or is expected to exceed Table R-1 column "SAE" for ≥ 15 min. (Note 1 & 2)

RS1.2
Dose assessment using actual meteorology indicates doses at or beyond the Site Boundary of **EITHER**:
> 100 mRem TEDE
OR
> 500 mRem Thyroid CDE

RS1.3
Field survey indicates closed window dose rate > 100 mRem/hr that is expected to continue for > 1 hr at or beyond the site boundary
OR
Field survey sample analysis indicates Thyroid CDE of > 500 mRem for 1 hr of inhalation at or beyond the site boundary

Alert

RA1a **Any** unplanned release of liquid or gaseous radioactivity to the environment that exceeds 200 times the radiological effluent Technical Specifications for 15 minutes or longer

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RA1.1
Valid reading on Circulating Water Discharge Tunnel Monitor SW-RM-130 (SW-RM-230) > 200 times the Hi-Hi alarm setpoint or offscale high for ≥ 15 min. (Note 2 & 6)

RA1.2
Valid reading on **any** gaseous monitors > Table R-1 column "Alert" for ≥ 15 min. (Note 2)

RA1.3
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x Technical Specification limits for ≥ 15 min. (Note 2)

Unusual Event

RU1a **Any** unplanned release of liquid radioactivity to the environment that exceeds two times the radiological effluent Technical Specifications for 60 minutes or longer

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RU1.1
Valid reading on Circulating Water Discharge Tunnel Monitor SW-RM-130 (SW-RM-230) > 2 times the Hi-Hi alarm setpoint for ≥ 60 min. (Note 2 & 6)

RU1.2
Confirmed sample analyses for liquid releases indicate concentrations or release rates > 2 x Technical Specification limits for ≥ 60 min. (Note 2)

RU1b **Any** unplanned release of gaseous radioactivity to the environment that exceeds two times the allocated radiological effluent ODCM limits for 60 minutes or longer

RU1.3
Valid reading on **any** gaseous monitors > Table R-1 column "NOUE" for ≥ 60 min. (Note 2)

RU1.4
Confirmed sample analyses for gaseous releases indicate concentrations or release rates > 2 x the allocated ODCM limits for ≥ 60 min. (Note 2)

2 — Onsite Rad Conditions

General Emergency: None

Site Area Emergency: None

Table R-1 Gaseous Effluent Monitor Classification Thresholds

Release Point	Monitor	GE	SAE	Alert	NOUE
Vent Stack A	VG-RI-179-1 or 2	4.00E+08 µCi/sec	4.00E+07 µCi/sec	4.56E+06 µCi/sec	3.60E+05 µCi/sec
Vent Stack B	VG-RI-180-1 or 2	3.57E+08 µCi/sec	3.57E+07 µCi/sec	4.07E+06 µCi/sec	3.60E+05 µCi/sec
Process Vent	GW-RI-178-1 or 2	3.70E+08 µCi/sec	3.70E+07 µCi/sec	4.22E+06 µCi/sec	2.80E+05 µCi/sec
Main Steam (Steam Safety) (Note 8)	MS-RM-170 (270) MS-RM-171 (271) MS-RM-172 (272)	8.62E+02 mR/hr	8.62E+01 mR/hr	9.81E+00 mR/hr	N/A
AFWPT Exhaust (Note 8)	MS-RM-176 (276)	2.84E+02 mR/hr	2.84E+01 mR/hr	3.24E+00 mR/hr	N/A

Alert

RA2a Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RA2.1
Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel resulting in a valid Hi-Hi alarm on **any** of the following radiation monitors:
- RM-RMS-153 Fuel Pit Bridge
- RM-RMS-152 New Fuel Storage Area
- RM-RMS-162 (RM-RMS-262) Manipulator Crane Area
- RM-RMS-163 (RM-RMS-263) Containment Area
- RM-RMS-159 (RM-RMS-259) Containment Particulate
- RM-RMS-160 (RM-RMS-260) Containment Gaseous
- GW-RI-178-1 Process Vent Normal Range

RA2.2
A water level drop in the reactor refueling cavity, spent fuel pit or fuel transfer canal that will result in irradiated fuel becoming uncovered

RA2b Release of radioactive material or increases in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain cold shutdown

RA2.3
Valid radiation monitor or survey reading > 1.50E-02 R/hr (15 mR/hr) in areas requiring continuous occupancy to maintain plant safety functions:
Control Room RM-RMS-157
OR
Central Alarm Station (CAS)

Unusual Event

RU2 Unexpected Increase in plant radiation

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RU2.1
Valid low water level alarm or visual observation indicating uncontrolled water level decrease in the refueling cavity, spent fuel pit or fuel transfer canal with **all** irradiated fuel assemblies remaining covered by water
AND
Unplanned valid direct area radiation reading increases resulting in a valid Hi alarm on **any** of the following radiation monitors:
- RM-RMS-162 (RM-RMS-262) - Manipulator Crane Area
- RM-RMS-152 - New Fuel Storage Area
- RM-RMS-153 - Fuel Pit Bridge

RU2.2
Unplanned valid direct area radiation monitor reading increases by a factor of 1000 over normal* levels

* Normal levels can be considered as the highest reading in the past 24 hours excluding the current peak value

H — Hazards

1 — Natural & Destructive Phenomena

General Emergency: None

Site Area Emergency: None

Table H-1 Safe Shutdown Areas
- Cable Vaults & Tunnels
- Emergency Switchgear Rooms
- Emergency Diesel Generators Rooms
- Reactor Containment
- Quench Spray Pump Houses
- Safeguards Areas
- Main Steam Valve House
- Cable Spreading Rooms
- Control Room
- CR Chiller Rooms
- Auxiliary / Fuel / Decontamination Buildings
- Fuel Oil Pump House Room A or B
- Service Water Pump and Valve House
- Intake Structure Control House
- Auxiliary Service Water Pump House
- Turbine Building
- Auxiliary Feedwater Pump House

Alert

HA1 Natural or destructive phenomena affecting a plant safe shutdown area

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA1.1
"OBE EXCEEDED" indicator illuminated on the SYSCOM Network Control Center (NCC)
AND
Earthquake confirmed by **any** of the following:
- Earthquake felt in plant
- National Earthquake Information Center (NEIC)
- Control Room indication of degraded performance of any safety-related structure, system, or component

HA1.2
Tornado or high winds > 80 mph resulting in **EITHER**:
Visible damage to any safety-related structure, system, or component within **any** Table H-1 Area
OR
Control Room indication of degraded performance of any safety-related structure, system, or component

HA1.3
Turbine failure-generated missiles resulting in **EITHER**:
Any visible damage to **any** safety-related structure, system, or component within **any** Table H-1 area
OR
Control Room indications of degraded performance of those safety systems resulting from turbine failure-generated missiles

HA1.4
Uncontrolled flooding resulting in **EITHER**:
Control Room indications of degraded performance of safety-related structure, system or component within **any** Table H-1 area
OR
Creating an industrial safety hazard (e.g. electric shock) in **any** Table H-1 area that precludes access necessary to operate or monitor **any** safety-related structure, system or component

HA1.5
Service Water Reservoir (Service Water Pump House) level < 309 ft.
OR
Lake level > 271 ft. (station finished ground grade)
OR
Class I Dam Emergency

HA1.6
Vehicle crash resulting in **EITHER**:
Visible damage to **any** safety-related structure, system, or component within **any** Table H-1 area
OR
Control Room indication of degraded performance of any safety-related structure, system, or component

Unusual Event

HU1 Natural or destructive phenomena affecting the Protected Area or Main Dam

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU1.1
Seismic event identified by any **TWO** of the following:
- Earthquake felt in the plant
- "SYSTEM TRIGGER" indicator illuminated on the SYSCOM Network Control Center (NCC)
- National Earthquake Information Center (NEIC)

HU1.2
Report by plant personnel of tornado or high winds > 80 mph striking within Protected Area boundary

HU1.3
Report of turbine failure resulting in casing penetration or damage to turbine seals or turbine generator seals

HU1.4
Uncontrolled flooding in **any** Table H-1 area that has the potential to affect safety related equipment needed for the current operating mode

HU1.5
Lake level < 242 ft. **AND** actions required in TRM 3.7.4 not completed
OR
Lake level > 264 ft.
OR
Class II Dam Emergency

2 — Fire or Explosion

General Emergency: None

Site Area Emergency: None

Alert

HA2 Fire or explosion affecting the operability of plant safety-related structures, systems or components required to establish or maintain safe shutdown

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA2.1
Fire or explosion in **any** Table H-1 area
AND EITHER:
Plant personnel report visible damage to **any** safety-related structure, system, or component within the area
OR
Affected system parameter indications show degraded performance

Unusual Event

HU2 Fire or explosion within the Protected Area boundary

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU2.1
Fire in or restricting access to **any** Table H-1 area **not** extinguished within 15 minutes of Control Room notification or verification of a Control Room alarm (Note 3)

HU2.2
Report by plant personnel of an unanticipated explosion within Protected Area boundary, SWPH, SWVH or Auxiliary SWPH resulting in visible damage to permanent structure or equipment

3 — Toxic, Corrosive, Asphyxiant & Flammable Gas

General Emergency: None

Site Area Emergency: None

Alert

HA3 Access to a safe shutdown area is prohibited due to release of toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA3.1
Access to a Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor (Note 9)

Unusual Event

HU3 Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal operation of the plant

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU3.1
Report or detection of toxic, corrosive, asphyxiant or flammable gases that have or could enter the Owner Controlled Area in amounts that can affect normal plant operations

HU3.2
Recommendation by local, county or state officials to evacuate or shelter site personnel based on offsite event involving toxic, corrosive, asphyxiant or flammable gas

4 — Security

General Emergency

HG4 Hostile action resulting in loss of physical control of the facility

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HG4.1
A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions
OR
A hostile action has caused failure of Spent Fuel Cooling Systems and imminent fuel damage is likely for a freshly off-loaded reactor core in pool (i.e. freshly off-loaded reactor core means **any** spent fuel in the Spent Fuel Pit)

Site Area Emergency

HS4 Hostile action within the Protected Area

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HS4.1
A hostile action is occurring or has occurred within the Protected Area as reported by the Security Shift Supervisor

Alert

HA4 Hostile action within the Owner Controlled Area or airborne attack threat

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA4.1
A hostile action is occurring or has occurred within the Owner Controlled Area as reported by the Security Shift Supervisor
OR
A validated notification from NRC of an airliner attack threat < 30 min. away

Unusual Event

HU4 Confirmed security condition or threat which indicates a potential degradation in the level of safety of the plant

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU4.1
A security condition that does **not** involve a hostile action as reported by the Security Shift Supervisor
OR
A credible site-specific security threat notification
OR
A validated notification from NRC providing information of an aircraft threat

5 — Control Room Evacuation

General Emergency: None

Site Area Emergency

HS5 Control Room evacuation has been initiated and plant control **cannot** be established

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HS5.1
Control Room evacuation has been initiated
AND
Control of the plant **cannot** be established from the Auxiliary Shutdown Panel within 15 min. (Note 3)

Alert

HA5 Control Room evacuation has been initiated

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA5.1
Control Room evacuation has been initiated

Unusual Event: None

6 — Judgment

General Emergency

HG6 Other conditions existing which in the judgment of the SEM warrant declaration of General Emergency

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HG6.1
Other conditions exist which in the judgment of the SEM indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or hostile action that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) offsite for more than the immediate site area.

Site Area Emergency

HS6 Other conditions existing which in the judgment of the SEM warrant declaration of Site Area Emergency

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HS6.1
Other conditions exist which in the judgment of the SEM indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or hostile action that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. **Any** releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the site boundary.

Alert

HA6 Other conditions existing which in the judgment of the SEM warrant declaration of an Alert

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA6.1
Other conditions exist which in the judgment of the SEM indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of hostile action. **Any** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE).

Unusual Event

HU6 Other conditions existing which in the judgment of the SEM warrant declaration of a NOUE

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU6.1
Other conditions exist which in the judgment of the SEM indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety-related structures, systems or components occurs.

E — ISFSI

General Emergency: None

Site Area Emergency: None

Alert: None

Unusual Event

EU1 Damage to a loaded cask confinement boundary

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

EU1.1
Damage to a loaded cask/canister confinement boundary

Modes:

1	2	3	4	5	6	DEF
Power Operation	Startup	Hot Standby ≥ 350 °F	Hot Shutdown > 200 °F	Cold Shutdown ≤ 200 °F	Refueling	Defueled

Dominion

North Anna Power Station
Emergency Action Level Matrix
Revision 5

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | Notification of UNUSUAL EVENT

C — Cold SD/ Refuel System Malfunct.

1 — Loss of Power

General Emergency: None

Site Area Emergency: None

Alert

CA1 Loss of **all** offsite power and loss of **all** onsite AC power to emergency busses

Modes: 5 | 6 | DEF

CA1.1
Loss of **all** offsite and onsite AC power to Unit 1 (Unit 2) 4160-Volt emergency busses H and J for > 15 min. (Note 3)

Unusual Event

CU1a AC power capability to emergency busses reduced to a single power source for greater than 15 minutes such that **any** additional single failure would result in loss of **all** AC power to emergency busses

Modes: 5 | 6

CU1.1
AC power capability to Unit 1 (Unit 2) 4160-Volt emergency busses H and J reduced to a single power source for > 15 min. (**any** additional single failure would result in loss of **all** AC power to emergency busses) (Note 3)

CU1b Unplanned loss of required DC power for greater than 15 minutes

CU1.2
Unplanned loss of vital DC power to required DC busses based on < 105-Volt DC bus voltage indications
AND
Failure to restore power to at least one required DC bus within 15 min. from the time of loss (Note 3)

2 — RCS Level

General Emergency

CG2 Loss of Reactor Vessel inventory affecting fuel clad integrity with Containment challenged and irradiated fuel in the Reactor Vessel

Modes: 5 | 6

CG2.1
Core uncovery for > 30 min. (Note 3) as indicated by confirmed loss of inventory, Table C-4, **AND** RVLIS full range < 61%
AND
Containment challenged as indicated by **any** of the following:
- Containment closure **not** established (Note 4)
- Containment hydrogen concentration ≥ 4%
- Unplanned rise in containment pressure

CG2.2
Reactor Vessel level **cannot** be monitored for ≥ 30 min. (Note 3) with a loss of RCS inventory as indicated by **any** of the following:
- Manipulator Crane Area RM-RMS-162 (RM-RMS-262) > 5 R/hr
- Erratic source range monitor indication
- Unexplained increase in **any** Table C-1 sump/tank level

AND
Containment challenged as indicated by **any** of the following:
- Containment closure **not** established (Note 4)
- Containment hydrogen concentration ≥ 4%
- Unplanned rise in containment pressure

Site Area Emergency

CS2 Loss of Reactor Vessel inventory affecting core decay heat removal capability

Modes: 5 | 6

CS2.1
Containment closure **not** established (Note 4) with confirmed loss of inventory, Table C-4, **AND** RVLIS full range < 62%

CS2.2
Containment closure established (Note 4) with confirmed loss of inventory, Table C-4, **AND** RVLIS full range < 61%

CS2.3
Reactor Vessel level **cannot** be monitored for ≥ 30 min. (Note 3) with a loss of RCS inventory as indicated **any** of the following:
- Manipulator Crane Area RM-RMS-162 (RM-RMS-262) > 5 R/hr
- Erratic source range monitor indication
- Unexplained increase in **any** Table C-1 sump/tank level

Alert

CA2 Loss of RCS inventory

Modes: 5 | 6

CA2.1
Confirmed loss of inventory, Table C-4, **AND** Reactor Vessel level < bottom of the RCS hot leg as indicated by RVLIS full range < 63%

CA2.2
Reactor Vessel level **cannot** be monitored for ≥ 15 min. **AND** an unexplained increase in **any** Table C-1 sump/tank level (Note 3)

Table C-4 Inventory Loss Confirmatory Indicators

ANY of the following:
- Standpipe level indication decreasing/bottomed out
- RHR pump amp fluctuations
- Decreasing RVLIS trend
- Ultrasonic indications (if in service)

Unusual Event

CU2 Unplanned loss of RCS inventory with irradiated fuel in the Reactor Vessel

Modes: 6

CU2.1
Unplanned RCS level decreasing below the Reactor Vessel flange for ≥ 15 min. (Note 3)

CU2.2
Loss of inventory as indicated by unexplained increase in **any** Table C-1 sump/tank level
AND
Reactor Vessel water level **cannot** be monitored

Table C-1 Sumps/Tanks
- Reactor Containment Sump
- Pressurizer Relief Tank (PRT)
- Primary Drain Transfer Tank (PDTT)
- Component Cooling (CC) Surge Tank
- Refueling Water Storage Tank (RWST)

3 — RCS Temp.

General Emergency: None

Table C-3 RCS Reheat Duration Thresholds

RCS	Containment Closure (Note 4)	Duration
Intact and **not** Reduced/Decreased Inventory	N/A	60 min.*
Not Intact OR Reduced/Decreased Inventory	Established	20 min.*
	Not Established	0 min.

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable

Alert

CA3 Inability to maintain plant in cold shutdown with irradiated fuel in the Reactor Vessel

Modes: 5 | 6

CA3.1
An unplanned event results in RCS temperature > 200°F for > Table C-3 duration (Note 3)
OR
RCS pressure increase of > 10 psig due to a loss of RCS cooling (does not apply in solid plant operations)

Unusual Event

CU3 Unplanned loss of decay heat removal capability with irradiated fuel in the Reactor Vessel

Modes: 5 | 6

CU3.1
An unplanned event results in RCS temperature > 200°F

CU3.2
Loss of **all** RCS temperature and Reactor Vessel level indication for > 15 min. (Note 3)

4 — Communications

General Emergency: None

Site Area Emergency: None

Alert: None

Unusual Event

CU4 Unplanned loss of **all** onsite or offsite communications capabilities

Modes: 5 | 6

CU4.1
Loss of **all** Table C-2 onsite (internal) communications capability affecting the ability to perform routine operations
OR
Loss of **all** Table C-2 offsite (external) communications capability

5 — RCS Leakage

General Emergency: None

Site Area Emergency: None

Alert: None

Unusual Event

CU5 RCS leakage

Modes: 5

CU5.1
Inability to establish or maintain pressurizer level > 15% **OR** RCS target level band (if pressurizer level was intentionally being controlled below the low level setpoint) due to RCS leakage

6 — Inadvertent Criticality

General Emergency: None

Site Area Emergency: None

Alert: None

Unusual Event

CU6 Inadvertent criticality

Modes: 5 | 6

CU6.1
An unplanned sustained positive startup rate observed on nuclear instrumentation

Notes

Note 1: If dose assessment results are available at the time of declaration, the classification should be based on dose assessment instead of radiation monitor readings. While necessary declarations should **not** be delayed awaiting results, the dose assessment should be initiated / completed in order to determine the proper classification (See EAL RS1.2/RG1.2).

Note 2: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Note 4: Containment closure established means potential escape paths for fission product radioactivity within containment are closed, preventing release to the environment.

Note 5: Not applicable to this matrix.

Note 6: The read value shall be rounded down to the nearest tenth digit to facilitate reading the logarithmic meter in the Control Room.

Note 7: Not applicable to this matrix.

Note 8: Due to digital display limitations for these monitors, classification must be made based on the nearest hundredth of a mR/hr read on the monitor display.

Note 9: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

Table C-2 Communications Systems

System	Onsite (internal)	Offsite (external)
Radio Communications System	X	
Public Address and Intercom System	X	
Private Branch Telephone Exchange (PBX)	X	
Sound Powered Telephone System	X	
Commercial Telephone		X
Dedicated NRC Communications		X

EAL Identifier

XXX.X

Category (R, H, E, S, F, C)
Emergency classification (G, S, A, U)
Sequential number within subcategory/classification
Subcategory number (1 if no subcategory)

COLD CONDITIONS (RCS ≤ 200°F)

Dominion

North Anna Power Station
Emergency Action Level Matrix
Revision 5

Dominion® North Anna Power Station

Title: Emergency Action Level Technical Bases Document

Revision Number:	**Effective Date:**
5	December 10, 2013

Revision Summary:

1. EAL SU6.1 Initiating Condition "RCS leakage" changed to "RCS leakage for 15 minutes or longer". This change is being implemented per NRC letter dated September 25, 2013 (SER #13-563). Both conditions (located in box for EAL SU6.1) also revised to include "15 minutes or longer (Note 3)". "Note 3" added below box which states "The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time."
2. Initiating Condition for EALs HA1.1, HA1.2, HA1.3, HA1.4, HA1.5 and HA1.6 were revised to change "Natural and destructive …" to "Natural or destructive …".
3. EALs HU1.4 and HA2.1, added new NAPS Basis Reference "NAPS Appendix R Report, Section 4.4 Attachment to Table 4-1."
4. Attachment 2 – Fission Product Barrier Loss/Potential Loss Matrix and Bases, Containment Potential Barrier Loss #2 and #3 updated to include asterisk (*) and 'Note 3' for Restoration procedures.
5. EAL SU5.2 basis and Attachment 2 – Fission Product Barrier Loss/Potential Loss Matrix and Bases, Fuel Clad Barrier Loss #5 updated to remove the following sentence which stated "The monitors utilize an alarming dosimeter enclosed within a lead-shielded housing to detect ionizing radiation and are located next to the 2-inch letdown lines in the non-regenerative heat exchanger room." [No Change Bar]
   Attachment 2 – Fission Product Barrier Loss/Potential Loss Matrix and Bases, #5 administrative changes to replace colon with period in last sentence of bases and remove duplicate reference to HP-3010.040, Radiation Monitoring System Setpoint Determination. [No Change Bar]
6. Definitions for Available, Containment Closure, Decreased Inventory and Reduced Inventory updated per OU-AA-200, Shutdown Risk Management.
7. VPAP-2805, Shutdown Risk Program references updated to OU-AA-200, Shutdown Risk Management and OU-NA-201, Shutdown Safety Assessment Checklist, where applicable.
8. VPAP-2401, Fire Protection Program references updated to CM-AA-FPA-100, Fire Protection/Appendix R (Fire Safe Shutdown) Program for EALs HU1.4 and HA1.4.
9. EAL RU1.3, basis, editorial change to correct misspelling, 'threshoulds' to 'thresholds'.

Approvals on File

Table of Contents

1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Program for North Anna Power Station (NAPS). It should be used as a technical reference to facilitate review of the NAPS EALs, provide historical documentation for future reference and as an aid for training. Decision-makers responsible for implementation of EP-1.01, Emergency Manager Controlling Procedure, may use this document as a technical reference and an aid in EAL interpretation.

The expectation is that emergency classifications are to be made as soon as conditions are present and recognizable for the classification, but within 15 minutes or less in all cases of conditions present. Use of this document for assistance is not intended to delay the classification.

2.0 DISCUSSION

2.1 Background

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the NAPS Emergency Plan.

In 1992, the NRC endorsed NUMARC/NESP-007 “Methodology for Development of Emergency Action Levels” as an alternative to NUREG-0654 EAL guidance.

NEI 99-01 (NUMARC/NESP-007) Revision 4 and Draft Revision 5 represents the most recently accepted methodology. Enhancements over earlier revisions included:

- Consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur during plant shutdown conditions.
- Initiating conditions and example emergency action levels that fully address conditions that may be postulated to occur at permanently Defueled Stations and Independent Spent Fuel Storage Installations (ISFSIs).
- Simplifying the fission product barrier EAL threshold for a Site Area Emergency.

NRC Bulletin 2005-02: Emergency Preparedness and Response Actions for Security-Based Events (July 2005) was issued directing utilities to modify the Security based EALs.

In response to the NRC Bulletin, NEI issued a white paper titled “Enhancements to Emergency Preparedness Programs for Hostile Action” May 2005 (Revised November 18, 2005) that provides guidance on development of security based EALs.

In July of 2006, the NRC issued Regulatory Issue Summary 2006-12, Endorsement of Nuclear Energy Institute Guidance “Enhancements to Emergency Preparedness Programs for Hostile Action”.

Using NEI 99-01 Rev. 4 and Draft Rev. 5, NAPS conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

Many of the EALs derived from the NEI methodology are fission product barrier based. That is, the conditions that define the EALs are based upon loss or potential loss of one or more of the three *fission product barriers*. “Loss” and “Potential Loss” signify the relative damage and threat of damage to the barrier. “Loss” means the barrier no longer assures containment of radioactive materials; “potential loss” infers an increased probability of barrier loss and decreased certainty of maintaining the barrier.

The primary *fission product barriers* are:

A. <u>Reactor Fuel Clad (FC):</u> Zirconium tubes which house the ceramic uranium oxide pellets along with the end plugs which are welded into each end of the fuel rods comprise the FC barrier.

B. <u>Reactor Coolant System (RCS):</u> The reactor pressure vessel shell, vessel head, vessel nozzles and penetrations and all *primary systems* directly connected to the reactor vessel up to the first containment isolation valve comprise the RCS barrier.

C. <u>Containment (CNTMT):</u> The vapor containment pressure vessel and all isolation valves required to maintain containment integrity under accident conditions comprise the CNTMT barrier.

2.3 Emergency Classification Based on Fission Product Barrier Degradation

The following criteria are the bases for event classification related to fission product barrier loss or potential loss:

Notification of Unusual Event (NOUE):

Any loss or any potential loss of Containment

Alert:

Any loss or any potential loss of either Fuel Clad or RCS

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

Loss of any two barriers and loss or potential loss of third barrier

2.4 EAL Relationship to EOPs and Critical Safety Function Status

Where possible, the EALs have been made consistent with and utilize the conditions defined in the NAPS Emergency Operating Procedure (EOP) network. While the symptoms that drive operator actions specified in the EOPs are not indicative of all possible conditions which warrant emergency classification, they define the symptoms, independent of initiating events, for which reactor plant safety and/or fission product barrier integrity are threatened. When these symptoms are clearly representative of one of the NEI Initiating Conditions, they have been utilized as an EAL. This permits rapid classification of emergency situations based on plant conditions without the need for additional evaluation or event diagnosis. Although some of the EALs presented here are based on conditions defined in the EOPs, classification of emergencies using these EALs is not dependent upon EOP entry or execution. The EALs can be utilized independently or in conjunction with the EOPs.

2.5 Symptom-Based vs. Event-Based Approach

To the extent possible, the EALs are symptom-based. That is, the action level threshold is defined by values of key plant operating parameters that identify emergency or potential emergency conditions. This approach is appropriate because it allows the full scope of variations in the types of events to be classified as emergencies. However, a purely symptom-based approach is not sufficient to address all events for which emergency classification is appropriate. Particular events to which no predetermined symptoms can be ascribed have also been utilized as EALs since they may be indicative of potentially more serious conditions not yet fully realized.

2.6 EAL Organization

The NAPS EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
    - EALs applicable under <u>all</u> plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
    - EALs applicable only under <u>hot</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Hot Standby or Power Operation mode.
    - EALs applicable only under <u>cold</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling or Defueled mode.

  The purpose of the groups is to avoid review of hot condition EALs when the plant is in a cold condition and avoid review of cold condition EALs when the plant is in a hot condition. This approach significantly minimizes the total number of EALs that must be reviewed by the EAL-user for a given plant condition, reduces EAL-user reading burden and, thereby, speeds identification of the EAL that applies to the emergency.

- Within each of the above three groups, assignment of EALs to categories/subcategories – Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. Subcategories are used as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The NAPS EAL categories/subcategories are listed below.

NAPS EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
R – Abnormal Rad Release / Rad Effluent	1 – Offsite Rad Conditions 2 – Onsite Rad Conditions
H – Hazards	1 – Natural & Destructive Phenomena 2 – Fire or Explosion 3 – Toxic, Corrosive, Asphyxiant & Flammable Gas 4 – Security 5 – Control Room Evacuation 6 - Judgment
E – ISFSI	None
Hot Conditions:	
S – System Malfunction	1 – Loss of Power 2 – RPS Failure 3 – Inability to Reach or Maintain Shutdown Conditions 4 – Instrumentation / Communications 5 – Fuel Clad Degradation 6 – RCS Leakage 7 – Inadvertent Criticality
F – Fission Product Barriers	None
Cold Conditions:	
C – Cold Shutdown / Refuel System Malfunction	1 – Loss of Power 2 – RCS Level 3 – RCS Temperature 4 – Communications 5 – Fuel Clad Degradation 6 – RCS Leakage 7 – Inadvertent Criticality

The primary tool for determining the emergency classification level is the EAL Matrix. The user of the EAL Matrix may (but is not required to) consult the EAL Technical Bases Document in order to obtain additional information concerning the EALs under classification consideration. The user should consult Sections 2.7 and 2.8, and Attachments 1 and 2 of this document for such information.

2.7 Technical Bases Information

EAL technical bases are provided in Attachment 1 for each EAL according to EAL group (Any, Hot, Cold), EAL category (R, C, E, H, S and F) and EAL subcategory. A summary explanation of each category and subcategory is given at the beginning of the technical bases discussions of the EALs included in the category. For each EAL, the following information is provided:

Initiating Condition (IC)

Site-specific description of the generic IC given in NEI 99-01, Revision 4.

EAL Identifier

Each EAL is assigned a unique identifier to support accurate communication of the emergency classification to onsite and offsite personnel. Four characters define each EAL identifier:

1. First character (letter): Corresponds to the EAL category as described above (R, C, E, H, S or F)
2. Second character (letter): The emergency classification (U, A, S or G)
3. Third character (number): The numerical sequence of the subcategories given in the EAL matrix. If the category has only one subcategory, it is given the number one (1).
4. Fourth character (number): The numerical sequence of the EAL within the EAL matrix subcategory. If the subcategory has only one EAL, it is given the number one (1).

Classification:

Notification of Unusual Event (U), Alert (A), Site Area Emergency (S) or General Emergency (G)

EAL (enclosed in rectangle)

Exact wording of the EAL as it appears in the EAL Matrix

Mode Applicability

One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown, 5 - Cold Shutdown, 6 - Refueling, D - Defueled, All - All modes, N/A - Not Applicable. (See Section 2.8 for operating mode definitions.)

Basis:

Description of the rationale for the EAL

NAPS Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.8 Operating Mode Applicability

Technical Specifications Table 1.1-1 “Modes” defines the following operating modes. Defueled is defined in OU-NA-201, Shutdown Safety Assessment Checklist, definition 5.3.1.

1 Power Operations

$K_{eff} \geq 0.99$ and rated thermal power > 5%.

2 Startup

$K_{eff} \geq 0.99$ and rated thermal power ≤ 5%.

3 Hot Standby

$K_{eff} < 0.99$ and average reactor coolant temperature $T_{avg} \geq 350°F$.

4 Hot Shutdown

$K_{eff} < 0.99$ and average reactor coolant temperature $350°F > T_{avg} > 200°F$ with all reactor vessel head closure bolts fully tensioned.

5 <u>Cold Shutdown</u>

K_{eff} < 0.99 and average reactor coolant temperature $T_{avg} \leq 200°F$ with all reactor vessel head closure bolts fully tensioned.

6 <u>Refueling</u>

One or more reactor vessel head closure bolts less than fully tensioned.

D <u>Defueled</u>

All fuel assemblies have been removed from Containment.

The plant operating mode that exists at the time that the event occurs (prior to any protective system or operator action is initiated in response to the condition) should be compared to the mode applicability of the EALs. If a lower or higher plant operating mode is reached before the emergency classification is made, the declaration shall be based on the mode that existed at the time the event occurred. If additional events occur, the declaration shall be based on the mode that existed at the time the new event occurred. This logic is applied to determine the plant operation mode, EAL categories (C, F, S) and EAL chart (hot or cold) applicability.

Example:

While in Cold Shutdown, an event occurs that results in the RCS temperature exceeding 200 °F. Evaluation is performed using Cold EALs and a Notification of Unusual Event is declared per EAL CU3.1. If an additional event occurs, such as the plant experiences a loss of offsite power (new event), it would be evaluated under the System Malfunction (Hot) series of EALs.

2.9 Classifying Transient Events

For some events, the condition may be corrected before a declaration has been made. For example, an emergency classification is warranted when automatic and manual actions taken within the control room do not result in a required reactor trip. However, it is likely that actions taken outside of the control room will be successful, probably before the Station Emergency Manager (SEM) classifies the event. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined; in other situations, further analyses (e.g., coolant sampling) may be necessary.

In general, observe the following guidance: Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met. For example, a momentary event, such as an ATWS or an earthquake, requires declaration even though the condition may have been resolved by the time the declaration is made.

- An ATWS represents a *failure* of a front-line safety-related structure, system or component (RPS) designed to protect the health and safety of the public.
- The effect of an earthquake on plant equipment and structures may not be readily apparent until investigations are conducted.

There may be cases in which a plant condition that exceeded an EAL threshold was not recognized at the time of occurrence, but is identified well after the condition has occurred (e.g., as a result of routine log or record review) and the condition no longer exists. In these cases, an emergency shall not be declared. Reporting requirements of 10 CFR 50.72 are applicable and the guidance of NUREG-1022, Rev. 2, Section 3 should be applied.

2.10 Imminent EAL Thresholds

Although the majority of the EALs provide very specific thresholds, the SEM must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the SEM, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded. While this is particularly prudent at the higher emergency classes (the early classification may permit more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.

Additionally, for those EAL thresholds with a time criteria, in the absence of information to the contrary, assume that the applicable time criteria has already been exceeded if the length of time which the EAL threshold conditions have existed is unknown.

2.11 Anticipated/Planned Actions

There may be a condition in which an EAL threshold is anticipated to be exceeded as part of an approved and planned evolution and for which compensatory actions are taken. It is not expected that an emergency be declared for these evolutions where an EAL threshold is intentionally exceeded as part of the approved and planned evolution.

2.12 Unit-Specific Equipment/Component Designation

NAPS is a dual-unit PWR. The EALs are written to apply to both units. When equipment or components are specified within the text of an EAL, the Unit 1 designator is provided followed by the Unit 2 designator in parentheses.

3.0 REFERENCES

3.1 Developmental Documents

A. NEI 99-01 Revision 4, Methodology for Development of Emergency Action Levels

B. NUMARC/NESP-007 Rev. 2, Methodology for Development of Emergency Action Levels, “Questions and Answers”

C. NRC Regulatory Issue Summary (RIS) 2003-18, Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels, Revision 4, Dated January 2003, including Supplement 1 (July 13, 2004) and Supplement 2 (draft April 27, 2005)

D. NRC Bulletin 2005-02 “Emergency Preparedness and Response Actions for Security-Based Events”

E. NEI Guidance “Enhancements to Emergency Preparedness Programs for Hostile Action May 2005” (Revised November 18, 2005)

F. NRC Regulatory Issue Summary 2006-12, Endorsement of Nuclear Energy Institute Guidance “Enhancements to Emergency Preparedness Programs for Hostile Action” (July 2006)

3.2 Interface Documents

A. EPIP-1.01 Emergency Manager Controlling Procedure

B. NAPS EAL Matrix

3.3 Commitments

None

4.0 DEFINITIONS & ACRONYMS

<u>Definitions</u>

Adversary

As applied to security EALs, an armed or suspected-to-be-armed *intruder* whose intent is to commit *sabotage*, disrupt station operations or otherwise commit a crime on station property.

Affecting Safe Shutdown

Event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable hot or cold shutdown condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

> Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in hot shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is not "affecting safe shutdown."
>
> Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in cold shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is "affecting safe shutdown."

Aircraft

A machine or device, such as an airplane, helicopter, glider, or lighter-than-air craft, that is capable of atmospheric flight.

Airliner

A large *aircraft* with the potential for causing significant damage to the plant.

Alert

Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of *hostile action*. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Available

The status of a system, structure, or component (SSC) that is capable of performing its intended function and is in service, or can be placed in service, in a Functional or Operable state via prompt manual or automatic actuation. The SSC must either be in-service, or can be placed in-service or realigned in sufficient time for it to perform its intended function (OU-AA-200, Shutdown Risk Management).

Bomb

Refers to an explosive device suspected of having sufficient force to damage plant systems or structures.

Civil Disturbance

A group of unexpected or unauthorized individuals violently protesting station operations or activities at the site.

Close

To position a valve or damper so as to prevent flow of the process fluid. To make an electrical connection to supply power.

Confinement Boundary

Is the barrier(s) between areas containing radioactive substances and the environment.

Confirm

To prove to be true, exact, or accurate by observation of a condition or characteristic for comparison with an original or procedural requirement.

Containment Closure

The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions. Closure is ensured before Time to Core Boiling or compensatory actions are taken (OU-AA-200, Shutdown Risk Management).

Establishment of Containment closure means that all potential escape paths are *closed*. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required (Technical Specifications Basis 3.9.4).

Contiguous

Being in actual contact; touching along a boundary or at a point.

Control

To perform manual operations of equipment to satisfy some predetermined requirements.

Decreased Inventory

A condition with fuel in the reactor vessel and any RCS Loop Stop valve closed or RCS water level <5% in the Pressurizer (OU-AA-200, Shutdown Risk Management).

EPA PAGs

Environment Protection Agency Protective Action Guidelines. The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem Thyroid CDE. Actual or projected offsite exposures in excess of the EPA PAGs requires NAPS to recommend protective actions for the general public to offsite planning agencies.

Exceeds

To go or be beyond a stated or implied limit, measure, or degree.

Explosion

Is a rapid, violent, unconfined combustion, or catastrophic *failure* of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

Extortion

Is an attempt to cause an action at the station by threat of force.

Faulted

In a steam generator, the existence of secondary side leakage that results in an *uncontrolled* decrease in steam generator pressure or the steam generator being completely depressurized.

Failure

A state of inability to perform a normal function.

Fire

Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

Fission Product Barriers (FPB)

Multiple physical barriers any one of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment. The FPBs are the Reactor Fuel Clad (FC), Reactor Coolant System (RCS) and Containment (CNTMT).

Flooding

A condition where water from an uncontrolled source is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room.

Freshly Off-loaded Core

For the purpose of this EAL scheme, freshly off-loaded core means any spent fuel in the Spent Fuel Pit.

General Emergency

Events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or *hostile action* that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Hostage

Person(s) held as leverage against the station to ensure that demands will be met by the station.

Hostile Action

An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, takes *hostages*, and /or intimidates the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the Nuclear Power Plant. Nonterrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area.)

Hostile Force

One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

Immediately Dangerous to Life and Health (IDLH)

A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

Intruder

Person(s) present in a specified area without authorization.

Intrusion

The act of entering without authorization. Discovery of a *bomb* in a specified area is indication of intrusion into that area by a *hostile force*.

Lower Flammability Limit (LFL)

The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

Maintain

Take action, as necessary, to keep the value of the specified parameter within the applicable limits.

Missile

An object thrown or projected usually so as to strike something at a distance. For the purposes of the EALs a missile is any non-HOSTILE object which travels through the air and damages plant equipment.

Normal Plant Operations

Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from Normal Plant Operations.

Notification of Unusual Event (NOUE)

Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of *safety-related structures, systems or components* occurs.

Owner Controlled Area

The entire area contiguous to the *Protected Area*, owned by the Company and designated to be controlled for security reasons (Nuclear Security Fleet Procedure SY-AA-101).

Primary System

The pipes, valves, and other equipment which connect directly to the Reactor Vessel or reactor coolant system such that a reduction in Reactor Vessel pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

Protected Area

An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

Reduced Inventory

A condition with fuel in the reactor vessel and water level lower than three feet (Kewaunee - < 17% Vessel Level; Millstone Unit 3 - 5 feet) below the reactor vessel flange (OU-AA-200, Shutdown Risk Management).

Restore

To return a parameter or component to the desired state.

Ruptured

In a steam generator, existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

Sabotage

A *hostile action* of deliberate damage, misalignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. Equipment found tampered with or damaged due to malicious mischief may not meet the definition of Sabotage until this determination is made by security supervision.

Safe Plant Shutdown

Hot or cold shutdown (reactor subcritical) with control of coolant inventory and decay heat removal.

Safety Function

Reactivity control (ability to shutdown the reactor and *maintain* shutdown), RCS inventory control (ability to cool the core), secondary heat removal (ability to *maintain* a heat sink) and Spent Fuel Pit (Pool) cooling (ability to remove decay heat from irradiated fuel in storage).

Safety-Related Structures, Systems and Components (as defined in 10CFR50.2)

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary

(2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition; or

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Security Condition

Any security event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A security condition does not involve a hostile action.

Significant Transient

An *unplanned* event involving any of the following:

- Automatic turbine runback > 25% thermal reactor power
- Electrical load rejection > 25% full electrical load
- Reactor trip
- Safety injection activation
- Thermal power oscillations > 10%

Site Area Emergency

Events are in progress or have occurred which involve an actual or likely major *failures* of plant functions needed for protection of the public or *hostile action* that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the *site boundary*.

Site Boundary

The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment (UFSAR Section 2.1.1.3).

Strike Action

Work stoppage within the *Protected Area* by a body of workers to enforce compliance with demands made on NAPS. The strike action must threaten to interrupt *normal plant operations*.

Sustained

Prolonged. Not intermittent or of a transitory nature.

Trip

(1) Automatic function, a device trips on overload undervoltage, high voltage/runout/undercurrent/thermal overload; (2) To manually activate a function.

Unavailable

Not able to perform its intended function.

Uncontrolled

An evolution lacking control, but is not the result of operator action.

Unisolable

A breach or leak that cannot be promptly isolated.

Unplanned

A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

Valid

An indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

Visible Damage

Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

Vital Area

Any plant area which contains vital equipment. Any area, normally within the *protected area*, which contains equipment, systems, components, or material, the *failure*, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

Acronyms

AC .... Alternating Current
ATWS .... Anticipated Transient Without Scram
BWR .... Boiling Water Reactor
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
CR .... Control Room
CTMT .... Containment
CSF .... Critical Safety Function
CSFST .... Critical Safety Function Status Tree
DC .... Direct Current
DE .... Dose Equivalent
DSC .... Dry Storage Canister
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
EPIP .... Emergency Plan Implementing Procedure
ESF .... Engineered Safeguard Feature
ESW .... Emergency Service Water
FAA .... Federal Aviation Administration
FBI .... Federal Bureau of Investigation
FEMA .... Federal Emergency Management Agency
FSAR .... Final Safety Analysis Report
GE .... General Emergency
HOO .... Headquarters Operations Officer
IC .... Initiating Condition
IPEEE .... Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI .... Independent Spent Fuel Storage Installation
Keff .... Effective Neutron Multiplication Factor
LCO .... Limiting Condition of Operation

LER .... Licensee Event Report
LOCA .... Loss of Coolant Accident
LWR .... Light Water Reactor
MSTV .... Main Steam Trip Valve
MSL .... Mean Sea Level
mR .... milliRoentgen
MW .... Megawatt
NEI .... Nuclear Energy Institute
NPP .... Nuclear Power Plant
NRC .... Nuclear Regulatory Commission
NSSS .... Nuclear Steam Supply System
NORAD .... North American Aerospace Defense Command
NOUE .... Notification of Unusual Event
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM .... Off-site Dose Calculation Manual
ORO .... Off-site Response Organization
PA .... Protected Area
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR .... Pressurized Water Reactor
PSIG .... Pounds per Square Inch Gauge
R .... Roentgen
RCC .... Reactor Control Console
RCS .... Reactor Coolant System
rem .... Roentgen Equivalent Man
RETS .... Radiological Effluent Technical Specifications
RPS .... Reactor Protection System
RPV .... Reactor Pressure Vessel
RVLIS .... Reactor Vessel Level Indicating System
SBO .... Station Blackout
SFP .... Spent Fuel Pit
SG .... Steam Generator

SI .... Safety Injection
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
SSSC .... Sealed Surface Storage Cask
SWPH .... Service Water Pump House
SWVH .... Service Water Valve House
TEDE .... Total Effective Dose Equivalent
TOAF .... Top of Active Fuel
TSC .... Technical Support Center
UFSAR .... Updated Final Safety Analysis Report

5.0 ATTACHMENTS

5.1 Attachment 1, Emergency Action Level Technical Bases

5.2 Attachment 2, Fission Product Barrier Loss / Potential Loss Matrix and Basis

<u>Category R – Abnormal Rad Release / Rad Effluent</u>

EAL Group: ANY (EALs in this category are applicable to any plant condition, hot or cold.)

Many EALs are based on actual or potential degradation of *fission product barriers* because of the elevated potential for offsite radioactivity release. Degradation of *fission product barriers* though is not always apparent via non-radiological symptoms. Therefore, direct indication of elevated radiological effluents or area radiation levels are appropriate symptoms for emergency classification.

At lower levels, abnormal radioactivity releases may be indicative of a *failure* of containment systems or precursors to more significant releases. At higher release rates, offsite radiological conditions may result which require offsite protective actions. Elevated area radiation levels in plant may also be indicative of the *failure* of containment systems or preclude access to plant vital equipment necessary to ensure plant safety.

Events of this category pertain to the following subcategories:

<u>1. Offsite Rad Conditions</u>

Direct indication of effluent radiation monitoring systems provides a rapid assessment mechanism to determine releases in excess of classifiable limits. Projected offsite doses, actual offsite field measurements or measured release rates via sampling indicate doses or dose rates above classifiable limits.

<u>2. Onsite Rad Conditions</u>

Sustained general area radiation levels in excess of those indicating loss of control of radioactive materials or those levels which may preclude access to vital plant areas also warrant emergency classification.

RU1.1

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: **Any** *unplanned* release of liquid radioactivity to the environment that *exceeds* two times the radiological effluent Technical Specifications for 60 minutes or longer

EAL:

RU1.1	**Notification of Unusual Event**

Valid reading on Circulating Water Discharge Tunnel Monitor SW-RM-130 (SW-RM-230)

> 2 times the Hi-Hi alarm setpoint for ≥ 60 min. (Note 2 & 6)

Note 2: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.
Note 6: The read value shall be rounded down to the nearest tenth digit to facilitate reading the logarithmic meter in the Control Room.

Mode Applicability:

All

Basis:

Unplanned liquid releases in excess of two times the site Technical Specifications, as identified in the Offsite Dose Calculation Manual (ODCM), that continue for 60 minutes or longer represent an *uncontrolled* situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Notification of Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times the Technical Specification limit for 30 minutes does not exceed this initiating condition. Further, the SEM should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time

RU1.1 (cont)

is unknown.

The effluent monitor alarm setpoints are established to ensure the Technical Specification release limits are not exceeded. Using a value of >2 times the Hi-Hi alarm point provides a recognizable threshold value.

NAPS Basis Reference(s):

1. VPAP-2103N Offsite Dose Calculation Manual (North Anna)

RU1.2

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: **Any** *unplanned* release of liquid radioactivity to the environment that *exceeds* two times the radiological effluent Technical Specifications for 60 minutes or longer

EAL:

RU1.2 Notification of Unusual Event

Confirmed sample analyses for liquid releases indicate concentrations or release rates > 2 x Technical Specifications limits for ≥ 60 min. (Note 2)

Note 2: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Mode Applicability:

All

Basis:

Unplanned releases in excess of two times the site Technical Specification limits, as identified in the Offsite Dose Calculation Manual (ODCM), that continue for 60 minutes or longer represent an *uncontrolled* situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Notification of Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times the Technical Specification limit for 30 minutes does not exceed this initiating condition. Further, the SEM should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

RU1.2 (cont)

NAPS Basis Reference(s):

1. VPAP-2103N Offsite Dose Calculation Manual (North Anna)

RU1.3

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: **Any** *unplanned* release of gaseous radioactivity to the environment that *exceeds* two times the allocated radiological effluent ODCM limits for 60 minutes or longer

EAL:

RU1.3 Notification of Unusual Event

Valid reading on **any** gaseous monitors > Table R-1 column "NOUE" for ≥ 60 min. (Note 2)

Note 2: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Table R-1 Gaseous Effluent Monitor Classification Thresholds

Release Point	Monitor	GE	SAE	Alert	NOUE
Vent Stack A	VG-RI-179-1 or 2	4.00E+08 μCi/sec	4.00E+07 μCi/sec	4.56E+06 μCi/sec	3.60E+05 μCi/sec
Vent Stack B	VG-RI-180-1 or 2	3.57E+08 μCi/sec	3.57E+07 μCi/sec	4.07E+06 μCi/sec	3.60E+05 μCi/sec
Process Vent	GW-RI-178-1 or 2	3.70E+08 μCi/sec	3.70E+07 μCi/sec	4.22E+06 μCi/sec	2.80E+05 μCi/sec
Main Steam (Steam Safety) (Note 8)	MS-RM-170 (270) MS-RM-171 (271) MS-RM-172 (272)	8.62E+02 mR/hr	8.62E+01 mR/hr	9.81E+00 mR/hr	N/A
AFWPT Exhaust (Note 8)	MS-RM-176 (276)	2.84E+02 mR/hr	2.84E+01 mR/hr	3.24E+00 mR/hr	N/A

Note 8: Due to digital display limitations for these monitors, classification must be made based on the nearest hundredth of a mR/hr read on the monitor display.

Mode Applicability:

All

RU1.3 (cont)

Basis:

Unplanned releases in excess of two times the allocated radiological effluent ODCM limits, that continue for 60 minutes or longer represent an *uncontrolled* situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Notification of Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times the allocated ODCM limit for 30 minutes does not exceed this initiating condition. Further, the SEM should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

The basis for the NOUE values correspond to 2 x the allocated Offsite Dose Calculation Manual (ODCM) limit, which is determined using annual average meteorological dispersion. The ODCM limit is applicable to total releases from the site at any point in time (i.e., “instantaneous release rate limit”). This limit is used to calculate the release rate (μCi/sec) for each release pathway which would yield 500 mrem in a year. An allocation factor is applied to each pathway to determine the allocated ODCM limit for that pathway. The allocation factor applied for the Process Vent is 10% or 0.1, and for Vent A and Vent B is 100% or 1.0. The EAL values for the NOUE were calculated as 2 x the allocated ODCM limit for each pathway. This method follows the guidance from NEI 99-01 and provides a justifiable basis for increased NOUE thresholds based on established methods and setpoints provided in the facility ODCM. Due to the fact that there are no ODCM limits on steam safeties or auxiliary feedwater exhausts and the limited ability for these respective radiation monitors to detect low level activity in these steam line configurations, the NOUE classification thresholds for the steam safeties and auxiliary feedwater exhaust are being labeled N/A (not applicable).

RU1.3 (cont)

NAPS Basis Reference(s):

1. VPAP-2103N Offsite Dose Calculation Manual (North Anna)
2. Calc. PA-0225, Rev 0, Addendum 00B, North Anna Radiation Monitor Conversion Factors and EAL Readings
3. Technical Requirement Manual, Table 3.3.7-1

RU1.4

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: **Any** *unplanned* release of gaseous radioactivity to the environment that *exceeds* two times the allocated radiological effluent ODCM limits for 60 minutes or longer

EAL:

RU1.4	**Notification of Unusual Event**
Confirmed sample analyses for gaseous releases indicate concentrations or release rates $>$ 2 x the allocated ODCM limits for ≥ 60 min. (Note 2)	

Note 2: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Mode Applicability:

All

Basis:

Unplanned releases in excess of two times the allocated radiological effluent ODCM limits, that continue for 60 minutes or longer represent an *uncontrolled* situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Notification of Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times the allocated ODCM limit for 30 minutes does not exceed this initiating condition. Further, the SEM should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

RU1.4 (cont)

NAPS Basis Reference(s):

1. VPAP-2103N Offsite Dose Calculation Manual (North Anna)

RA1.1

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: **Any** *unplanned* release of gaseous or liquid radioactivity to the environment that *exceeds* 200 times the radiological effluent Technical Specifications for 15 minutes or longer

EAL:

RA1.1	**Alert**
Valid reading on Circulating Water Discharge Tunnel Monitor SW-RM-130 (SW-RM-230) > 200 times the Hi-Hi alarm setpoint or offscale high for ≥ 15 min. (Note 2 & 6)	

Note 2: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.
Note 6: The read value shall be rounded down to the nearest tenth digit to facilitate reading the logarithmic meter in the Control Room.

Mode Applicability:

All

Basis:

This EAL addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that *exceeds*, by a factor of 200, regulatory commitments for an extended period of time. NAPS incorporates features intended to control the release of radioactive effluents to the environment. Additionally, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Offsite Dose Calculation Manual (ODCM). The occurrence of extended, *uncontrolled* radioactive releases to the environment is indicative of a degradation in these features and/or controls.

RA1.1 (cont)

The SEM should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the SEM should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

The Alert liquid release value is 200 times the monitor alarm setpoint value or offscale high. This event escalates from the Notification of Unusual Event by escalating the magnitude of the release by a factor of 100. Using a value of >200 times the Hi-Hi alarm point or offscale high provides a recognizable threshold value.

The setpoints are established to ensure the Technical Specification release limits are not exceeded.

NAPS Basis Reference(s):

1. VPAP-2103N Offsite Dose Calculation Manual (North Anna)

RA1.2

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: **Any** *unplanned* release of gaseous or liquid radioactivity to the environment that *exceeds* 200 times the radiological effluent Technical Specifications for 15 minutes or longer

EAL:

RA1.2 Alert

Valid reading on **any** gaseous monitors > Table R-1 column “Alert” for ≥ 15 min. (Note 2)

Note 2: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Table R-1 Gaseous Effluent Monitor Classification Thresholds

Release Point	Monitor	GE	SAE	Alert	NOUE
Vent Stack A	VG-RI-179-1 or 2	4.00E+08 µCi/sec	4.00E+07 µCi/sec	4.56E+06 µCi/sec	3.60E+05 µCi/sec
Vent Stack B	VG-RI-180-1 or 2	3.57E+08 µCi/sec	3.57E+07 µCi/sec	4.07E+06 µCi/sec	3.60E+05 µCi/sec
Process Vent	GW-RI-178-1 or 2	3.70E+08 µCi/sec	3.70E+07 µCi/sec	4.22E+06 µCi/sec	2.80E+05 µCi/sec
Main Steam (Steam Safety) (Note 8)	MS-RM-170 (270) MS-RM-171 (271) MS-RM-172 (272)	8.62E+02 mR/hr	8.62E+01 mR/hr	9.81E+00 mR/hr	N/A
AFWPT Exhaust (Note 8)	MS-RM-176 (276)	2.84E+02 mR/hr	2.84E+01 mR/hr	3.24E+00 mR/hr	N/A

Note 8: Due to digital display limitations for these monitors, classification must be made based on the nearest hundredth of a mR/hr read on the monitor display.

Mode Applicability:

All

RA1.2 (cont)

Basis:

This EAL addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that *exceeds*, by a factor of 200, regulatory commitments for an extended period of time. NAPS incorporates features intended to control the release of radioactive effluents to the environment. Additionally, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Offsite Dose Calculation Manual (ODCM). The occurrence of extended, *uncontrolled* radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The SEM should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the SEM should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

The calculated values for the Auxiliary Feedwater Pump Turbine (AFWPT) Exhaust are slightly higher for Unit 1 vs. Unit 2. Unit 2 values are used in Table R-1. This eliminates a possibility of human error, reading wrong unit value and simplifies table. A Unit 1 event would be classified at a slightly lower value than calculated but within the error margin of the radiological calculation.

Releases should not be prorated or averaged. For example, a release exceeding 3 x Alert threshold for 5 minutes does not meet the threshold for declaration.

NAPS Basis Reference(s):

1. VPAP-2103N Offsite Dose Calculation Manual (North Anna)
2. Calc. PA-0225, Rev 0, Addendum 00B, North Anna Radiation Monitor Conversion Factors and EAL Readings

RA1.3

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: **Any** *unplanned* release of gaseous or liquid radioactivity to the environment that *exceeds* 200 times the radiological effluent Technical Specification for 15 minutes or longer

EAL:

RA1.3	**Alert**
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x Technical Specification limits for ≥ 15 min. (Note 2)	

Note 2: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Mode Applicability:

All

Basis:

Confirmed sample analyses in excess of two hundred times the site Technical Specification limits, as identified in the Offsite Dose Calculation Manual (ODCM), that continue for 15 minutes or longer represent an *uncontrolled* situation and hence, a potential degradation in the level of safety. This event escalates from the Notification of Unusual Event by raising the magnitude of the release by a factor of 100 over the Notification of Unusual Event level (i.e., 200 times Technical Specifications). Releases should not be prorated or averaged. For example, a release exceeding 3 x Alert threshold for 5 minutes does not meet the threshold for declaration.

RA1.3 (cont)

The SEM should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the SEM should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

NAPS Basis Reference(s):

1. VPAP-2103N Offsite Dose Calculation Manual (North Anna)

RS1.1

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity *exceeds* 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release

EAL:

RS1.1 Site Area Emergency

Valid reading on **any** gaseous radiation monitors that *exceeds* or is expected to exceed Table R-1 column "SAE" for ≥ 15 min. (Note 1 & 2)

Note 1: If dose assessment results are available at the time of declaration, the classification should be based on dose assessment instead of radiation monitor readings. While necessary declarations should **not** be delayed awaiting results, the dose assessment should be initiated / completed in order to determine the proper classification (See EAL RS1.2/RG1.2).

Note 2: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Table R-1 Gaseous Effluent Monitor Classification Thresholds

Release Point	Monitor	GE	SAE	Alert	NOUE
Vent Stack A	VG-RI-179-1 or 2	4.00E+08 μCi/sec	4.00E+07 μCi/sec	4.56E+06 μCi/sec	3.60E+05 μCi/sec
Vent Stack B	VG-RI-180-1 or 2	3.57E+08 μCi/sec	3.57E+07 μCi/sec	4.07E+06 μCi/sec	3.60E+05 μCi/sec
Process Vent	GW-RI-178-1 or 2	3.70E+08 μCi/sec	3.70E+07 μCi/sec	4.22E+06 μCi/sec	2.80E+05 μCi/sec
Main Steam (Steam Safety) (Note 8)	MS-RM-170 (270) MS-RM-171 (271) MS-RM-172 (272)	8.62E+02 mR/hr	8.62E+01 mR/hr	9.81E+00 mR/hr	N/A
AFWPT Exhaust (Note 8)	MS-RM-176 (276)	2.84E+02 mR/hr	2.84E+01 mR/hr	3.24E+00 mR/hr	N/A

Note 8: Due to digital display limitations for these monitors, classification must be made based on the nearest hundredth of a mR/hr read on the monitor display.

RS1.1 (cont)

Mode Applicability:

All

Basis:

This EAL addresses radioactivity releases that can result in doses at or beyond the *Site Boundary* that exceed a fraction (10%) of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the *failure* of plant systems needed for the protection of the public. While these *failures* are addressed by other EALs, this EAL provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone (e.g., fuel handling accident in the Fuel Building).

The SEM should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the SEM should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

The calculated values for the Auxiliary Feedwater Pump Turbine (AFWPT) Exhaust are slightly higher for Unit 1 vs. Unit 2. Unit 2 values are used in Table R-1. This eliminates a possibility of human error, reading wrong unit value and simplifies table. A Unit 1 event would be classified at a slightly lower value than calculated but within the error margin of the radiological calculation.

RS1.1 (cont)

Since dose assessment is based on actual meteorology whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1.

NAPS Basis Reference(s):

1. EPIP-4.01, Radiological Assessment Director Controlling Procedure
2. EPIP-4.03, Dose Assessment Team Controlling Procedure
3. Calc. PA-0225, Rev 0, Addendum 00B, North Anna Radiation Monitor Conversion Factors and EAL Readings

RS1.2

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity *exceeds* 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release

EAL:

RS1.2 Site Area Emergency

Dose assessment using actual meteorology indicates doses at or beyond the *Site Boundary* of **EITHER**:

> 100 mRem TEDE

OR

> 500 mRem Thyroid CDE

Mode Applicability:

All

Basis:

The 100 mRem integrated TEDE dose in this EAL is based on the 10 CFR 20 average member of the public exposure. This value also provides a desirable gradient (one order of magnitude) between the Alert, Site Area Emergency, and General Emergency classes. It is deemed that exposures less than this limit are not consistent with the Site Area Emergency class description. The 500 mRem integrated Thyroid CDE dose was established in consideration of the 1:5 ratio of the EPA Protective Action Guidelines for TEDE and thyroid exposure. In establishing the dose rate emergency action levels (e.g., EAL RS1.1, etc.), a duration of one hour has been assumed. Therefore, the dose rate EALs are based on a *Site Boundary* dose rate of 100 mRem/hr TEDE or 500 mRem/hr Thyroid CDE, whichever is more limiting. Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible.

NAPS Basis Reference(s):

1. EPIP-4.01, Radiological Assessment Director Controlling Procedure
2. EPIP-4.03, Dose Assessment Team Controlling Procedure

RS1.3

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity *exceeds* 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release

EAL:

RS1.3 Site Area Emergency

Field survey indicates closed window dose rate > 100 mRem/hr that is expected to continue for > 1 hr at or beyond the *site boundary*

OR

Field survey sample analysis indicates Thyroid CDE of > 500 mRem for 1 hr of inhalation at or beyond the *site boundary*

Mode Applicability:

All

Basis:

The 100 mRem integrated TEDE dose in this EAL is based on the 10 CFR average member of the public exposure. This value also provides a desirable gradient (one order of magnitude) between the Alert, Site Area Emergency, and General Emergency classes. It is deemed that exposures less than this limit are not consistent with the Site Area Emergency class description. The 500 mRem integrated Thyroid CDE dose was established in consideration of the 1:5 ratio of the EPA Protective Action Guidelines for TEDE and thyroid exposure. In establishing the dose rate emergency action levels, a duration of one hour is assumed. Therefore, the dose rate EALs are based on a *Site Boundary* dose rate of 100 mRem/hr TEDE or 500 mRem/hr Thyroid CDE, whichever is more limiting.

NAPS Basis Reference(s):

1. UFSAR Figure 2.1-3, Site Boundary
2. NAPS Emergency Plan Definitions
3. EPIP-4.16, Offsite Monitoring

RG1.1

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity *exceeds* 1000 mRem TEDE or 5000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RG1.1 General Emergency

Valid reading on **any** gaseous radiation monitors that *exceeds* or is expected to exceed Table R-1 column "GE" for ≥ 15 min. (Note 1 & 2)

Note 1: If dose assessment results are available at the time of declaration, the classification should be based on dose assessment instead of radiation monitor readings. While necessary declarations should **not** be delayed awaiting results, the dose assessment should be initiated / completed in order to determine the proper classification (See EAL RS1.2/RG1.2).

Note 2: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Table R-1 Gaseous Effluent Monitor Classification Thresholds

Release Point	Monitor	GE	SAE	Alert	NOUE
Vent Stack A	VG-RI-179-1 or 2	4.00E+08 µCi/sec	4.00E+07 µCi/sec	4.56E+06 µCi/sec	3.60E+05 µCi/sec
Vent Stack B	VG-RI-180-1 or 2	3.57E+08 µCi/sec	3.57E+07 µCi/sec	4.07E+06 µCi/sec	3.60E+05 µCi/sec
Process Vent	GW-RI-178-1 or 2	3.70E+08 µCi/sec	3.70E+07 µCi/sec	4.22E+06 µCi/sec	2.80E+05 µCi/sec
Main Steam (Steam Safety) (Note 8)	MS-RM-170 (270) MS-RM-171 (271) MS-RM-172 (272)	8.62E+02 mR/hr	8.62E+01 mR/hr	9.81E+00 mR/hr	N/A
AFWPT Exhaust (Note 8)	MS-RM-176 (276)	2.84E+02 mR/hr	2.84E+01 mR/hr	3.24E+00 mR/hr	N/A

Note 8: Due to digital display limitations for these monitors, classification must be made based on the nearest hundredth of a mR/hr read on the monitor display.

RG1.1 (cont)

Mode Applicability:

All

Basis:

This EAL addresses radioactivity releases that can result in doses at or beyond the *Site Boundary* that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the *failure* of plant systems needed for the protection of the public and likely involve fuel damage. While these *failures* are addressed by other EALs, this EAL provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.

The calculated values for the Auxiliary Feedwater Pump Turbine (AFWPT) Exhaust are slightly higher for Unit 1 vs. Unit 2. Unit 2 values are used in Table R-1. This eliminates a possibility of human error, reading wrong unit value and simplifies table. A Unit 1 event would be classified at a slightly lower value than calculated but within the error margin of the radiological calculation.

The SEM should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the SEM should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

The Table R-1 column “GE” effluent monitor readings are one decade greater than the “SAE” values.

RG1.1 (cont)

Since dose assessment is based on actual meteorology, whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1.

NAPS Basis Reference(s):

1. EPIP-4.01, Radiological Assessment Director Controlling Procedure
2. EPIP-4.03, Dose Assessment Team Controlling Procedure
3. Calc. PA-0225, Rev 0, Addendum 00B, North Anna Radiation Monitor Conversion Factors and EAL Readings

RG1.2

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity *exceeds* 1000 mRem TEDE or 5000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RG1.2 General Emergency

Dose assessment using actual meteorology indicates doses at or beyond the *Site Boundary* of **EITHER**:

> 1000 mRem TEDE

OR

> 5000 mRem Thyroid CDE

Mode Applicability:

All

Basis:

The General Emergency values are based on the boundary dose resulting from an actual or imminent release of gaseous radioactivity that *exceeds* 1000 mRem TEDE or 5000 mRem Thyroid CDE for the actual or projected duration of the release. The 1000 mRem TEDE and the 5000 mRem Thyroid CDE integrated dose are based on the EPA Protective Action Guidelines (PAGs) which indicates that public protective actions are warranted if the dose *exceeds* 1 Rem TEDE or 5 Rem Thyroid CDE. This is consistent with the emergency class description for a General Emergency. This level constitutes the upper level of the desirable gradient for the Site Area Emergency. Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible. In establishing the dose rate emergency action levels, a duration of one hour is assumed. Therefore, the dose rate EALs are based on a *Site Boundary* dose rate of 1000 mRem/hr TEDE or 5000 mRem/hr Thyroid CDE, whichever is more limiting.

RG1.2 (cont)

NAPS Basis Reference(s):

1. EPIP-4.01, Radiological Assessment Director Controlling Procedure
2. EPIP-4.03, Dose Assessment Team Controlling Procedure

RG1.3

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity *exceeds* 1000 mRem TEDE or 5000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RG1.3 General Emergency

Field survey indicates closed window dose rate > 1,000 mRem/hr that is expected to continue for > 1 hr at or beyond the *site boundary*

OR

Field survey sample analysis indicates Thyroid CDE of > 5,000 mRem for 1 hr of inhalation at or beyond the *site boundary*

Mode Applicability:

All

Basis:

The 1000 mrem TEDE integrated dose in this EAL is based on the EPA Protective Action Guidelines (PAGs) which indicate that public protective actions are warranted if the dose exceeds 1 Rem TEDE. This value also provides a desirable gradient (one order of magnitude) between the Site Area Emergency and General Emergency classes. It is deemed that exposures less than this limit are not consistent with the General Emergency class description. The 5,000 mRem integrated Thyroid CDE dose was established in consideration of the 1:5 ratio of the EPA Protective Action Guidelines for TEDE and thyroid exposure. In establishing the dose rate emergency action levels, a duration of one hour is assumed.

RG1.3 (cont)

NAPS Basis Reference(s):

1. UFSAR Figure 2.1-3, Site Boundary
2. NAPS Emergency Plan Definitions
3. EPIP-4.16 Offsite Monitoring

RU2.1

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 2 – Onsite Rad Conditions

Initiating Condition: Unexpected increase in plant radiation

EAL:

RU2.1 Notification of Unusual Event

Valid low water level alarm or visual observation indicating *uncontrolled* water level decrease in the refueling cavity, spent fuel pit or fuel transfer canal with **all** irradiated fuel assemblies remaining covered by water

AND

Unplanned valid direct area radiation reading increases resulting in a *valid* Hi alarm on **any** of the following radiation monitors:

- RM-RMS-162 (RM-RMS-262) Manipulator Crane Area
- RM-RMS-152 - New Fuel Storage Area
- RM-RMS-153 - Fuel Pit Bridge

Mode Applicability:

All

Basis:

In light of Reactor Cavity Seal *failure* incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR, explicit coverage of these types of events via this EAL is appropriate given their potential for elevated doses to plant workers. Loss of inventory from the refueling cavity, spent fuel pit (pool) or fuel transfer canal may reduce water shielding above spent fuel and cause unexpected increases in plant radiation. Classification as a Notification of Unusual Event is warranted as a precursor to a more serious event.

RU2.1 (cont)

The low water level alarm in this EAL refers to the Spent Fuel Pit (SFP) low level alarm. The fuel transfer canal is normally in communication with the spent fuel pit. During refueling operations, the refueling cavity in the Containment is filled and is in communication with the fuel transfer canal when the fuel transfer tube is opened. A decrease in water level in the SFP, fuel transfer canal or refueling cavity is therefore sensed by the SFP low level alarm. Neither the refueling cavity nor the fuel transfer canal is equipped with a low level alarm.

The SFP level is remotely monitored by level switch LS-FC-100. The level switch initiates high and low level annunciators. The SFP WATER LEVEL LOW alarm (window 1E-C6) actuates if SFP level decreases to the 289 ft 4 in. el. (ref. 1). Local level indication is provided by a ruled scale mounted on the east side of the counterfort. Normal level is indicated by the 0 mark on the scale and corresponds to 289 ft 10 in. el. or normal SFP level. Level is normally maintained between the 0 in. mark and the +3 in. mark. The low level alarm corresponds to the -6 in. mark. If makeup to the SFP is required, it takes about 1300 gallons of water to increase SFP level one inch. Refueling cavity level is remotely monitored by pressurizer level calibrated for cold conditions. Normal refueling cavity water level is 2 ft below the refueling deck or 289 ft 10 in. el., which is approximately 55 percent cold cal pressurizer level indicated on 1-RC-LI-1462 (2-RC-LI-1462). The relationship of the low level alarm setpoint to the elevation of normal water level, the spent fuel assemblies and spent fuel storage racks is illustrated in Figures R-1 and R-2 (ref. 3, 4, 5).

RU2.1 (cont)

The probable causes of a loss of SFP or refueling cavity inventory and receipt of the low level alarm are:

- The rupture of a cooling line
- Leakage past the transfer gate
- An improper valve lineup
- Loss of cooling and subsequent inventory boil-off

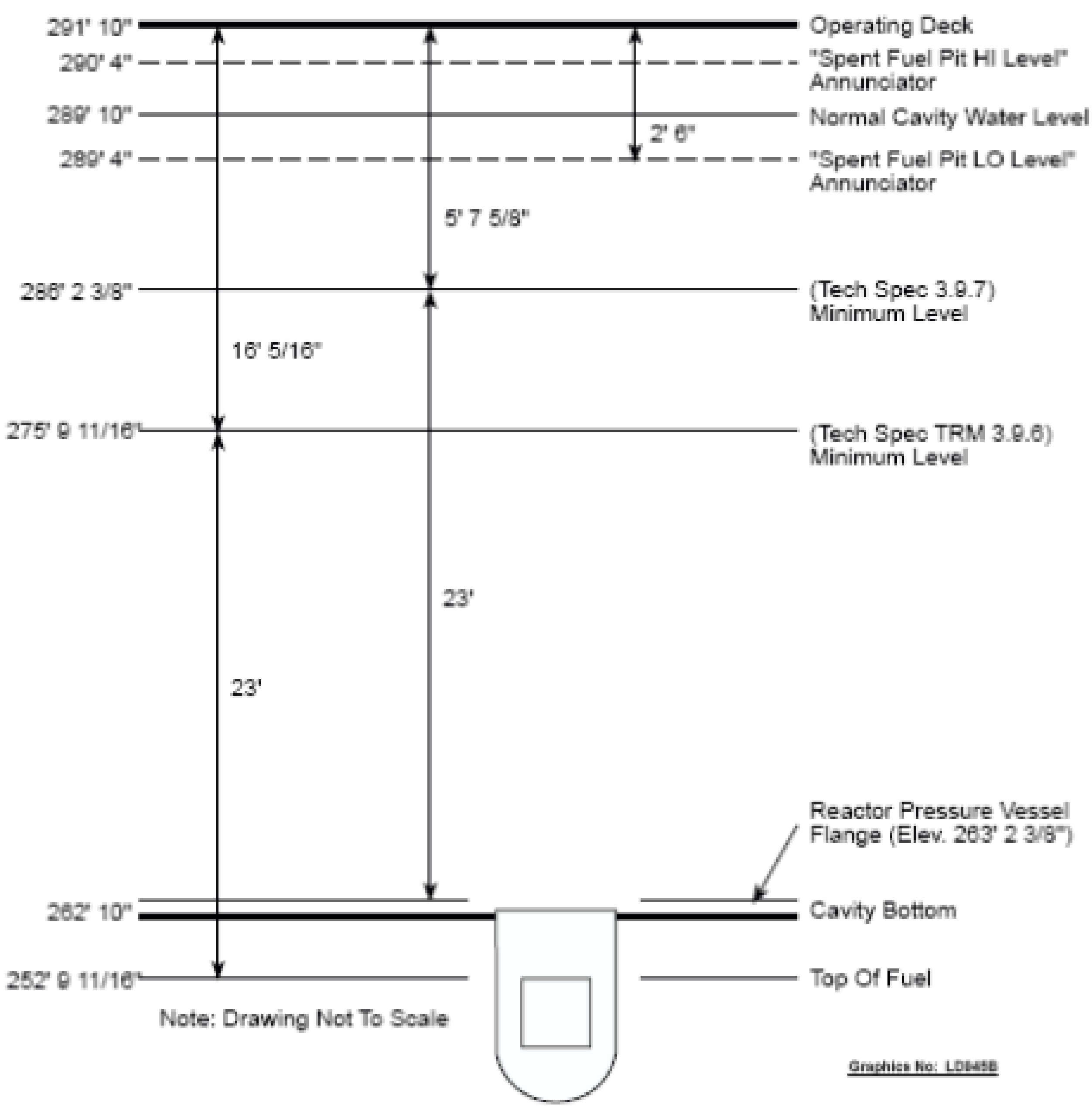
Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment. Technical Specification LCO 3.7.16 (ref. 6) requires at least 23 ft of water above irradiated fuel seated in the Spent Fuel Pit storage racks. If level decreases more than 4 ft 2 in. below the 0 mark, the LCO is not met (ref. 7). Technical Specification LCO 3.9.7 (ref. 8) requires at least 23 ft of water above the Reactor Vessel flange during movement of fuel assemblies. During refueling, this maintains sufficient water level in the fuel transfer canal, refueling cavity, and SFP to retain iodine fission product activity in the water in the event of a fuel handling accident. Therefore, this EAL is applicable for conditions in which irradiated fuel is being transferred to and from the Reactor Vessel and the SFP.

While a radiation monitor (e.g., RM-RMS-152 New Fuel Storage Area Radiation Monitor, RM-RMS-153 Fuel Pit Bridge Radiation Monitor, RM-RMS-162 (RM-RMS-262) Manipulator Crane Area could detect an increase in dose due to a drop in the water level, it might not be a reliable indication, in and of itself, of whether or not inventory is being lost. For example, the reading on an area radiation monitor (permanently installed or temporary) located near the refueling cavity may increase due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Generally, elevated radiation monitor indications need to be combined with another indicator (or personnel report) of water loss.

RU2.1 (cont)

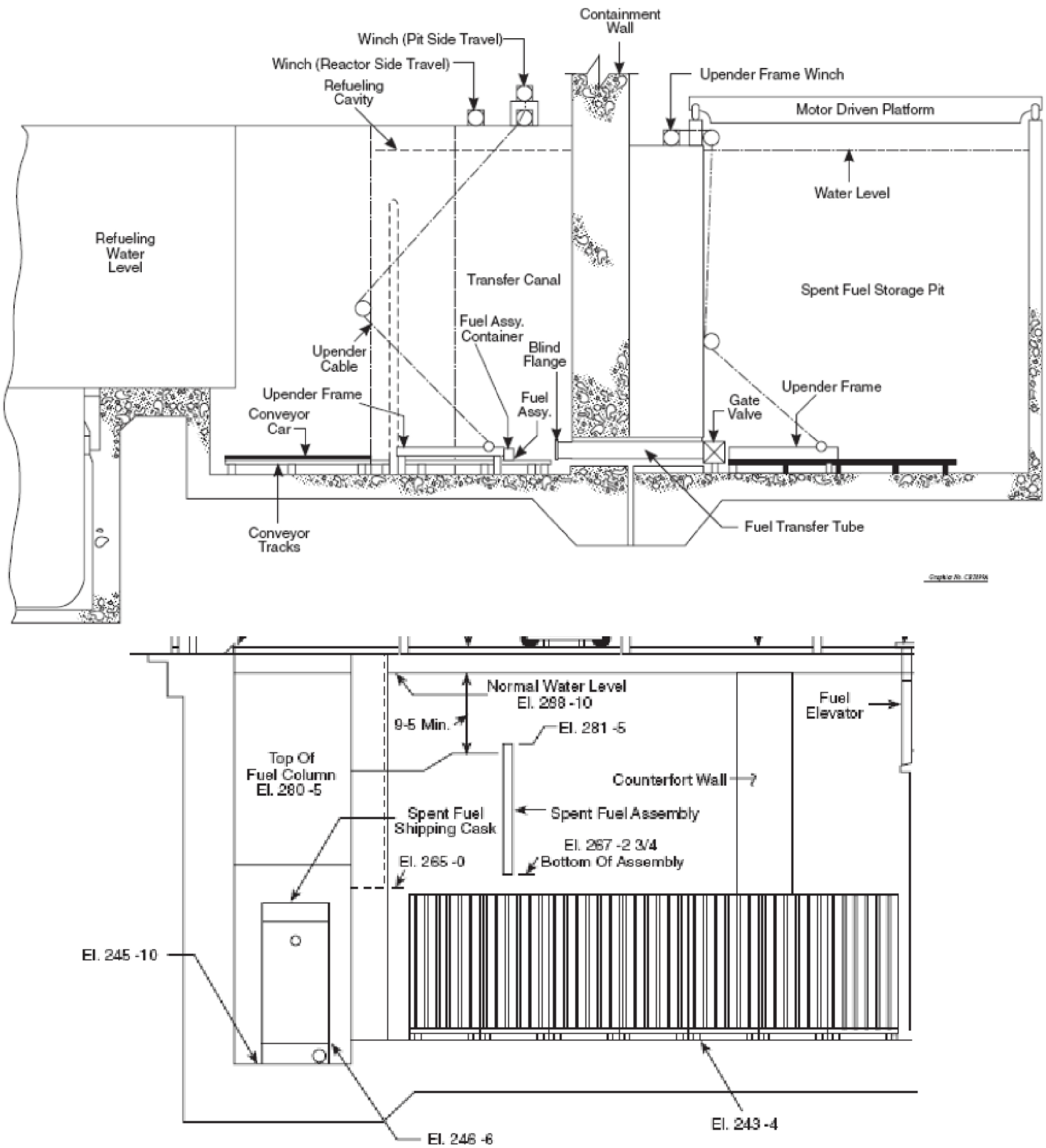
For refueling events where the water level drops below the vessel flange, classification would be via CU2.1. This event escalates to an Alert per RA2.1 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Table for events in operating modes 1-4.

Figure R-1: Spent Fuel Pit (Pool) & Refueling Cavity Levels



RU2.1 (cont)

Figure R-2: Spent Fuel Pit (Pool), Fuel Transfer Canal and Refueling Cavity Arrangement



RU2.1 (cont)

NAPS Basis Reference(s):

1. AR 1-E-C6 Spent Fuel Pit Lo Level
2. 0-AP-27 Malfunction of Spent Fuel Pit System
3. 1-PT-93 (2-PT-93) Reactor Vessel Water Level Determination
4. UFSAR Figure 9.1-3
5. UFSAR Figure 9.1-9
6. Technical Specification LCO 3.7.16
7. 1-AP-52 (2-AP-52) Loss of Refueling Cavity Level During Refueling
8. Technical Specification LCO 3.9.7
9. Technical Specifications B.3.9.7
10. UFSAR Section 9.1
11. 0-GOP-13.3 Assessment of Loss of Inventory
12. 0-AP-5.1 Common Unit Radiation Monitoring System
13. 11715-FM-3A, Sh.1 Arrangement Fuel Building
14. 11715-FM-3B, Sh. 2 Arrangement Fuel Building

RU2.2

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 2 – Onsite Rad Conditions

Initiating Condition: Unexpected increase in plant radiation

EAL:

RU2.2 Notification of Unusual Event

Unplanned valid direct area radiation monitor reading increases by a factor of 1000 over normal* levels

* Normal levels can be considered as the highest reading in the past 24 hours excluding the current peak value

Mode Applicability:

All

Basis:

This EAL addresses *unplanned* increases in radiation levels inside the plant. These radiation levels represent degradation in the control of radioactive material and a potential degradation in the level of safety of the plant. Radiation levels may be determined via installed area monitors, surveys, sampling or other means of obtaining an acceptable radiation reading. This EAL escalates to an Alert per RA2.1 if the elevated radiation levels impair the level of safe plant operation.

NAPS Basis Reference(s):

1. VPAP-2101 Radiation Protection Program

RA2.1

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 2 – Onsite Rad Conditions

Initiating Condition: Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel

EAL:

RA2.1 Alert

Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel resulting in a *valid* Hi-Hi alarm on **any** of the following radiation monitors:

- RM-RMS-153 Fuel Pit Bridge
- RM-RMS-152 New Fuel Storage Area
- RM-RMS-162 (RM-RMS-262) Manipulator Crane Area
- RM-RMS-163 (RM-RMS-263) Containment Area
- RM-RMS-159 (RM-RMS-259) Containment Particulate
- RM-RMS-160 (RM-RMS-260) Containment Gaseous
- GW-RI-178-1 Process Vent Normal Range

Mode Applicability:

All

Basis:

This EAL addresses specific events that have resulted, or may result, in unexpected increases in radiation dose rates within plant buildings and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent degradation in the level of safety of the plant. These events escalate from RU2.1 in that fuel activity has been released or is anticipated due to fuel heatup. This EAL applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

RA2.1 (cont)

When considering escalation, information may come from:

- Radiation monitor readings
- Sampling and surveys
- Dose projections/calculations
- Reports from the scene regarding the extent of damage (e.g., refueling crew, radiation protection technicians)

This EAL is defined by the specific areas where irradiated fuel is located, such as the refueling cavity or Spent Fuel Pit (SFP).

The bases for the SFP area radiation high-high alarms and containment area and ventilation radiation high alarms are a spent fuel handling accident and are, therefore, appropriate for this EAL. In the Fuel Building, a fuel assembly could be dropped in the fuel transfer canal or in the SFP. Should a fuel assembly be dropped in the fuel transfer canal or in the SFP and release radioactivity above a prescribed level, the area radiation monitors sound an alarm, alerting personnel to the problem. If *valid* high indication on RM-RMS-152 or RM-RMS-153 (new fuel area or spent fuel bridge crane area) is received while handling fuel in the Fuel Building, then placing in service the Auxiliary Building Iodine Filter Banks per 0-AP-30 is required (ref. 4). If Fuel Building area radiation reaches 1 R/hr, personnel are evacuated from the Fuel Building (ref. 1).

RA2.1 (cont)

Elevated background at the monitor due to decreasing water level may mask elevated ventilation exhaust airborne activity and needs to be considered. However, while radiation monitors may detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the monitor could in fact be properly responding to a known event involving transfer or relocation of a source stored in or near the SFP or responding to a planned evolution such as removal of the Reactor Vessel head. Interpretation of these EAL thresholds requires some understanding of the actual radiological conditions present in the vicinity of the monitors.

This event escalates to a Site Area or General Emergency via radiological effluent EALs.

NAPS Basis Reference(s):

1. 0-AP-27 Malfunction of Spent Fuel Pit System
2. 0-AP-5.2 MGP Radiation Monitoring System
3. 0-AP-5.1 Common Unit Radiation Monitoring System
4. 0-AP-30 Fuel Failure During Handling
5. 1-AP-5 (2-AP-5) Radiation Monitoring System
6. 1-AP-52 (2-AP-52) Loss of Refueling Cavity Level During Refueling
7. NRC EAL FAQ 2006-013

RA2.2

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 2 – Onsite Rad Conditions

Initiating Condition: Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel

EAL:

RA2.2	**Alert**
A water level drop in the reactor refueling cavity, spent fuel pit or fuel transfer canal that will result in irradiated fuel becoming uncovered	

Mode Applicability:

All

Basis:

This EAL addresses specific events that have resulted, or may result, in unexpected increases in radiation dose rates within plant buildings and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and degradation in the level of safety of the plant. These events escalate from EAL RU2.1 in that fuel activity has been released or is anticipated due to fuel heatup. This EAL applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

When considering escalation, information may come from:

- Radiation monitor readings
- Sampling and surveys
- Dose projections/calculations
- Reports from the scene regarding the extent of damage (e.g., refueling crew, radiation protection technicians)

RA2.2 (cont)

This EAL is defined by the specific areas where irradiated fuel is located such as the refueling cavity, Reactor Vessel or SFP.

There is no remote level indication that level in the SFP or refueling cavity has dropped to the level of the irradiated fuel. Depending on *available* level indication, the declared threshold may need to be based on indications of makeup rate or decrease in refueling water storage tank level.

The movement of irradiated fuel assemblies within Containment requires a minimum water level of 23 ft above the top of the Reactor Vessel flange (ref. 1). During refueling activities, this maintains sufficient water level in the refueling cavity, fuel transfer canal and SFP. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident.

The probable causes of a loss of SFP inventory are:

- The rupture of a cooling line
- Leakage past the transfer gate
- An improper valve lineup
- Loss of cooling and subsequent inventory boil-off

This event escalates to a Site Area or General Emergency via radiological effluent EALs.

RA2.2 (cont)

NAPS Basis Reference(s):

1. Technical Specifications LCO 3.9.5
2. 0-AP-27 Malfunction of Spent Fuel Pit System
3. 0-AP-5.2 MGP Radiation Monitoring System
4. 0-AP-5.1 Common Unit Radiation Monitoring System
5. 0-AP-30 Fuel Failure During Handling
6. 1-AP-5 (2-AP-5) Radiation Monitoring System
7. 1-AP-52 (2-AP-52) Loss of Refueling Cavity Level During Refueling

RA2.3

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 2 – Onsite Rad Conditions

Initiating Condition: Release of radioactive material or increases in radiation levels within the facility that impedes operation of systems required to *maintain* safe operations or to establish or *maintain* cold shutdown

EAL:

RA2.3 **Alert**

Valid radiation monitor or survey reading > 1.50E-02 R/hr (15 mR/hr) in areas requiring continuous occupancy to *maintain* plant *safety functions*:

Control Room RM-RMS-157

OR

Central Alarm Station (CAS)

Mode Applicability:

All

Basis:

This EAL addresses elevated radiation levels in areas requiring continuous occupancy to *maintain* safe plant operation or perform a *safe plant shutdown*. The areas that meet this threshold are the Control Room and the Central Alarm Station (CAS). The security alarm station is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations.

The value of 15 mR/hr is derived from the GDC 19 value of 5 Rem in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737, Clarification of TMI Action Plan Requirements, provides that the 15 mR/hr value can be averaged over the 30 days, the value is used here without averaging. The 30-day duration implies an event potentially more significant than an Alert.

RA2.3 (cont)

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the increase in radiation levels is not a concern of this EAL. The SEM must consider the source or cause of the elevated radiation levels and determine if any other EALs may be involved. For example, a Control Room dose rate of 15 mR/hr may be a problem in itself. However, the increase may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

The Control Room radiation monitor, RM-RMS-157, reads R/hr in scientific notation.

NAPS Basis Reference(s):

1. 0-AP-5.1 Common Unit Radiation Monitoring System

<u>Category C – Cold Shutdown / Refueling System Malfunction</u>

EAL Group: Cold Conditions (RCS temperature ≤ 200ºF); EALs in this category are applicable only in one or more cold operating modes.

Category C EALs are directly associated with cold shutdown, refueling or defueled system *safety functions*. Given the variability of plant configurations (e.g., systems out-of-service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The cold shutdown and refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, *containment closure*, and fuel clad integrity for the applicable operating modes (5 - Cold Shutdown, 6 - Refueling, D – Defueled).

The events of this category pertain to the following subcategories:

<u>1. Loss of Power</u>

Loss of emergency plant electrical power can compromise plant safety-related structures, systems and component operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite sources for 4160-Volt AC emergency busses and loss of vital 125-Volt DC power sources.

<u>2. RCS Level</u>

Reactor Vessel or RCS water level is directly related to the status of adequate core cooling and, therefore, fuel clad integrity.

<u>3. RCS Temperature</u>

Uncontrolled or inadvertent temperature or pressure increases are indicative of a potential loss of *safety functions*.

4. Communications

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

5. RCS Leakage

The Reactor Vessel provides a volume for the coolant that covers the reactor core. The Reactor Vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail.

Excessive RCS leakage greater than Technical Specification limits is utilized to indicate potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and containment integrity.

6. Inadvertent Criticality

Inadvertent criticalities pose potential personnel safety hazards as well being indicative of losses of reactivity control.

CU1.1

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: AC power capability to emergency busses reduced to a single power source for greater than 15 minutes such that **any** additional single failure would result in loss of **all** AC power to emergency busses

EAL:

CU1.1 Notification of Unusual Event

AC power capability to Unit 1 (Unit 2) 4160-Volt emergency busses H and J reduced to a single power source for > 15 min. (**any** additional single failure would result in loss of **all** AC power to the emergency busses) (Note 3)

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

The condition indicated by this EAL is the degradation of the offsite and onsite power sources such that any additional single *failure* would result in a loss of all AC power to the unit emergency 4160-Volt emergency busses. Unit 1 (Unit 2) 4160-Volt emergency busses H and J are the essential busses (ref. 1).

CU1.1 (cont)

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker which would allow automatic backfeed for the SSTs. When a unit is shut down, the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs).

The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). The switchyard voltage is reduced from 500 kV and 230 kV to 34.5 kV. The RSSTs step down the 34.5 kV power to 4160-Volt AC for distribution as the preferred source of power to the emergency electrical distribution System.

The additional bus ties for Unit 1 between the emergency bus 1H and normal bus 1B and emergency bus 1J and normal bus 2B provide two independent offsite power sources to each emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions.

If the SBO diesel generator is supplying power to an emergency bus of a unit that has lost all other sources of emergency offsite AC power, the unit has not lost all 4160-Volt AC power.

CU1.1 (cont)

Several combinations of power *failures* could therefore satisfy this EAL. Consideration should be given to operable loads necessary to remove decay heat or provide RCS makeup capability when evaluating loss of AC power to the 4160-Volt emergency busses. Even though a unit 4160-Volt emergency bus may be energized, if all necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RCS makeup capability) are not functional on the energized bus, the bus should not be considered available.

The 15-minute interval was selected as a threshold to exclude transient power losses. If multiple sources fail to energize the unit 4160-Volt emergency busses within 15 minutes, a Notification of Unusual Event is declared under this EAL.

NAPS Basis Reference(s):

1. 11715-FE-1A Main One Line Diagram (Unit 1)
2. UFSAR Section 8.3
3. 1-OP-26.1 (2-OP-26.1) Transferring 4160-Volt Busses
4. 0-AP-10 Loss of Electrical Power
4. 1-ECA-0.0 (2-ECA-0.0) Loss of All AC Power
5. NRC EAL FAQ 2006-017
6. 12050-FE-1A Main One Line Diagram (Unit 2)

CU1.2

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: *Unplanned* loss of required DC power for greater than 15 minutes

EAL:

CU1.2 Notification of Unusual Event

Unplanned loss of vital DC power to required DC busses based on < 105-Volt DC bus voltage indications

AND

Failure to *restore* power to at least one required DC bus within 15 min. from the time of loss (Note 3)

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during cold shutdown or refueling operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss. The fifteen minute interval is intended to exclude transient or momentary power losses.

There are four independent 125-Volt DC systems. DC power is supplied for:

- Control power to 4160 and 480-Volt AC breakers
- Emergency lighting
- DC motor-driven pumps

CU1.2 (cont)

- AC inverters that supply the 120V AC vital buses

Each 125-Volt DC system is normally powered through its respective battery charger. Each system consists of a 125-Volt DC distribution panel and a respective battery. The battery chargers convert 480-Volt AC power to a 12-Volt DC regulated output, which powers the associated 125-Volt DC busses and maintains a floating charge on the batteries connected to the busses. Each unit has six battery chargers. Four of the battery chargers are called normal battery chargers and are normally used to provide 125-Volt DC to their respective dc bus and battery. Two of the battery chargers are swing battery chargers and can be used as installed spares for either of the two normal battery chargers in the respective safeguards train. The batteries supply power only if the battery chargers fail or if the demand *exceeds* the capacity of the chargers. Each battery consists of 60 cells connected in series and is located in individual battery rooms. A battery terminal voltage of 105 volts DC is the minimum required to ensure proper operation of equipment connected to the DC bus (ref. 1).

"*Unplanned*" is included in this EAL to preclude the declaration of an emergency as a result of planned maintenance activities such as maintenance on a train during shutdown periods. This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS1.2.

If this loss results in the inability to maintain Cold Shutdown, escalation to an Alert will be per CA3.1.

NAPS Basis Reference(s):

1. 0-OP-6.4 Operation of the SBO Diesel (SBO Event)
2. UFSAR Section 8.3.2
3. 0-AP-10 Loss of Electrical Power

CA1.1

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: Loss of **all** offsite power and loss of **all** onsite AC power to emergency busses

EAL:

CA1.1 Alert

Loss of **all** offsite and onsite AC power to Unit 1 (Unit 2) 4160-Volt emergency busses H and J for > 15 min. (Note 3)

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, D - Defueled

Basis:

Loss of all AC power compromises all plant *safety-related structures, systems and components* requiring electrical power. This EAL is indicated by the loss of all offsite and onsite AC power to the Unit 1(Unit 2) 4160-Volt emergency busses H and J (ref. 1).

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs).

CA1.1 (cont)

The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). The switchyard voltage is reduced from 500 kV and 230 kV to 34.5 kV. The RSSTs step down the 34.5 kV power to 4160-Volt AC for distribution as the preferred source of power to the emergency electrical distribution System.

The additional bus ties for Unit 1 between the emergency bus 1H and normal bus 1B and emergency bus 1J and normal bus 2B provide two independent offsite power sources to each emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions. If the SBO diesel generator is supplying power to an emergency bus of a unit that has lost all other sources of emergency offsite AC power, the unit has not lost all 4160-Volt AC power.

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of all AC power to the emergency busses. Even though a unit emergency bus may be energized, if all necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not functional on the energized bus then the bus should not be considered available.

The 15-minute interval was selected as a threshold to exclude transient power losses.

This EAL is the cold and defueled condition equivalent of the hot condition loss of all AC power EAL SS1.1.

CA1.1 (cont)

NAPS Basis Reference(s):

1. 11715-FE-1A Main One Line Diagram (Unit 1)
2. UFSAR Section 8.3
3. 1-OP-26.1 (2-OP-26.1) Transferring 4160-Volt Busses
4. 0-AP-10 Loss of Electrical Power
5. 1-ECA-0.0 (2-ECA-0.0) Loss of All AC Power
6. 12050-FE-1A Main One Line Diagram (Unit 2)

CU2.1

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 2 – RCS Level

Initiating Condition: *Unplanned* loss of RCS inventory with irradiated fuel in the Reactor Vessel

EAL:

CU2.1	**Notification of Unusual Event**
Unplanned RCS level decreasing below the Reactor Vessel flange for ≥ 15 min. (Note 3)	

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

6 - Refueling

Basis:

The Reactor Vessel flange is at 263 ft 2-3/8 in. el. (ref. 1) and can be monitored by:

- RVLIS Train A/B Full Range: Indication is *available* in the Control Room along with strip chart trending of Train A. The upper tap is connected to the vessel manual head vent line. The lower taps connect to the incore detector penetration seal table. Indication is not affected by closure of loop stop valves. If the upper/lower taps are connected, indication is reliable whether the RCS is boiling or subcooled. The upper tap is disconnected fairly early in an outage to support vessel head removal. Once the upper tap is removed, RVLIS still provides for trending of vessel level as long as the RCS is subcooled. RCS pressurization or voiding due to boiling causes a false high level indication. Indicated level is slightly higher than actual level if RCS temperatures are less than 300°F.

CU2.1 (cont)

- RVLIS Train A/B Upper Range, if loops are not isolated: Indication is *available* in the Control Room along with strip chart trending of Train A. The upper tap is connected to the vessel manual head vent line. The lower taps are on the SG side of the loop stop valves, Train A from C hot leg and Train B from A hot leg. Indication is not *available* when the RCS loops are isolated. If the upper/lower taps are connected, indication is reliable whether the RCS is boiling or subcooled. The upper tap is disconnected fairly early in the outage to support vessel head removal. Once the upper tap is removed, RVLIS still provides for trending of vessel level as long as the RCS is subcooled. RCS pressurization or voiding due to boiling causes a false high level indication. Indicated level is slightly higher than actual level if RCS temperatures are less than 300°F.
- 1-RC-LI-102 (2-RC-LI-202) Level Standpipe: This indicator is a magnetic flapper type sharing the same taps as 1-RC-LI-103 (2-RC-LI-203) and measures level from RCS centerline. Indicated level can be monitored locally and in the Control Room via a camera in containment. The upper tap is attached to the top of the Pressurizer via the Pressurizer spray line. The lower tap is attached to the vessel side of the C Tc loop stop valve. Indication is affected by Pressurizer surge line flooding and RHR return flow through 1-RH-MOV-1720B (2-RH-MOV-2720B) to C cold leg.

CU2.1 (cont)

- 1-RC-LI-103 (2-RC-LI-203) Cold Shutdown RCS Level Indicator: - The level transmitter shares the same taps as 1-RC-LI-102 (2-RC-LI-202) and measures level from RCS centerline. Indicated level is *available* in the Control Room with strip chart trending. The upper tap is attached to the top of the Pressurizer via the Pressurizer spray line. The lower tap is attached to the vessel side of the C Tc loop stop valve. The indicator must be manually enabled with a key switch located behind the Unit 1 Vertical Board. Annunciator 1E-B8, RX CLNT DRAIN DOWN LO LVL, alarms at level ≤ 9 in. above centerline to alert the operator of decreasing vessel level. Indication is affected by Pressurizer surge line flooding and RHR return flow through 1-RH-MOV-1720B (2-RH-MOV-2720B) to C cold leg.

This EAL is an Notification of Unusual Event because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. Refueling operations that lower RCS water level below the Reactor Vessel flange are carefully planned and procedurally controlled. An *unplanned* event that results in water level decreasing below the Reactor Vessel flange warrants declaration of an Notification of Unusual Event due to the reduced RCS inventory that is available to keep the core covered. The fifteen-minute interval was chosen because it is reasonable to assume that level can be *restored* within this time frame using one or more of the redundant means of refill that should be available. If level cannot be *restored* in this time frame, a more serious condition may exist.

CU2.1 (cont)

This EAL is not applicable to drops in flooded refueling cavity water level (covered by decreasing Spent Fuel Pit water level in EAL RU2.1) until such time as the level decreases to the level of the vessel flange. If level continues to decrease and reaches the bottom inside diameter of the RCS hot leg penetration, escalation to the Alert level under EAL CA2.1 would be appropriate. If the decreasing level is accompanied by RCS heatup, escalation to the Alert level under EAL CA3.1 may also be appropriate.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refueling mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

NAPS Basis Reference(s):

1. 1-OP-5.4 (2-OP-5.4) Draining the Reactor Coolant System
2. 1-OP-4.1 (2-OP-4.1) Controlling Procedure for Refueling
3. 1-AP-17 (2-AP-17) Shutdown LOCA
4. 1-GOP-13.0 (2-GOP-13.0) Alternate Core Cooling Method
5. 1-AP-11 (2-AP-11) Loss of RHR
6. 1-AP-52 (2-AP-52) Loss of Refueling Cavity Level During Refueling

CU2.2

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 2 – RCS Level

Initiating Condition: *Unplanned* loss of RCS inventory with irradiated fuel in the Reactor Vessel

EAL:

CU2.2 Notification of Unusual Event

Loss of inventory as indicated by unexplained increase in **any** Table C-1 sump/tank level

AND

Reactor Vessel water level **cannot** be monitored

Table C-1 Sumps / Tanks
Reactor Containment Sump Pressurizer Relief Tank (PRT) Primary Drain Transfer Tank (PDTT) Component Cooling (CC) Surge Tank Refueling Water Storage Tank (RWST)

Mode Applicability:

6 - Refueling

Basis:

This EAL is an Notification of Unusual Event because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

CU2.2 (cont)

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode, the RCS is not intact and Reactor Vessel water level and inventory are monitored by different means. In the Refueling mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. Reactor Vessel water level is normally monitored using the following instruments:

- RVLIS Train A/B Full Range: Indication is available in the Control Room along with strip chart trending of Train A. The upper tap is connected to the vessel manual head vent line. The lower taps connect to the incore detector penetration seal table. Indication is not affected by closure of loop stop valves. If the upper/lower taps are connected, indication is reliable whether the RCS is boiling or subcooled. The upper tap is disconnected fairly early in an outage to support vessel head removal. Once the upper tap is removed, RVLIS still provides accurate trending of vessel level as long as the RCS is subcooled. RCS pressurization or voiding due to boiling causes a false high level indication. Indicated level is slightly higher than actual level if RCS temperatures are less than 300°F.

CU2.2 (cont)

- RVLIS Train A/B Upper Range, if loops are not isolated: Indication is available in the Control Room along with strip chart trending of Train A. The upper tap is connected to the vessel manual head vent line. The lower taps are on the SG side of the loop stop valves, Train A from C hot leg and Train B from A hot leg. Indication is not available when the RCS loops are isolated. If the upper/lower taps are connected, indication is reliable whether the RCS is boiling or subcooled. The upper tap is disconnected fairly early in the outage to support vessel head removal. Once the upper tap is removed, RVLIS still provides accurate trending of vessel level as long as the RCS is subcooled. RCS pressurization or voiding due to boiling causes a false high level indication. Indicated level is slightly higher than actual level if RCS temperatures are less than 300°F.
- 1-RC-LI-102 (2-RC-LI-202) Level Standpipe: This indicator is a magnetic flapper type sharing the same taps as 1-RC-LI-103 (2-RC-LI-203) and measures level from RCS centerline. Indicated level can be monitored locally and in the Control Room via a camera in containment. The upper tap is attached to the top of the Pressurizer via the Pressurizer spray line. The lower tap is attached to the vessel side of the C Tc loop stop valve. Indication is affected by Pressurizer surge line flooding and RHR return flow through 1-RH-MOV-1720B (2-RH-MOV-2720B) to C cold leg.

CU2.2 (cont)

- 1-RC-LI-103 (2-RC-LI-203) Cold Shutdown RCS Level Indicator: - The level transmitter shares the same taps as 1-RC-LI-102 (2-RC-LI-202) and measures level from RCS centerline. Indicated level is available in the Control Room with strip chart trending. The upper tap is attached to the top of the Pressurizer via the Pressurizer spray line. The lower tap is attached to the vessel side of the C Tc loop stop valve. The indicator must be manually enabled with a key switch located behind the Unit 1 Vertical Board. Annunciator 1E-B8, RX CLNT DRAIN DOWN LO LVL, alarms at level ≤ 9 in. above centerline to alert the operator of decreasing vessel level. Indication is affected by Pressurizer surge line flooding and RHR return flow through 1-RH-MOV-1720B (2-RH-MOV-2720B) to C cold leg.

In this EAL, all water level indication is *unavailable*, and the Reactor Vessel inventory loss must be detected by sump or tank level changes. Surveillance procedures provide instructions for calculating *primary system* leak rate by manual or computer-based water inventory balances (Ref. 11, 12). Sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Containment to ensure they are indicative of RCS leakage.

If the decreasing level is accompanied by RCS heatup, escalation to the Alert level under EAL CA3.1 may also be appropriate.

CU2.2 (cont)

NAPS Basis Reference(s):

1. 1-OP-5.4 (2-OP-5.4) Draining the Reactor Coolant System
2. UFSAR Section 6.2
3. UFSAR Figure 6.2-57
4. UFSAR Section 9.2
5. UFSAR Section 9.3
6. 1-OP-4.1 (2-OP-4) Controlling Procedure for Refueling
7. 1-AP-17 (2-AP-17) Shutdown LOCA
8. 1-GOP-13.0 (2-GOP-13.0) Alternate Core Cooling Method
9. 1-AP-11 (2-AP-11) Loss of RHR
10. 1-AP-52 (2-AP-52) Loss of Refueling Cavity Level During Refueling
11. 1-PT-52.2 (2-PT-52.2) Reactor Coolant System Leak Rate (Hand Calculation)
12. 1-PT-52.2A (2-PT-52.2A) Reactor Coolant System Leak Rate (Computer Calculation)

CA2.1

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 2 – RCS Level

Initiating Condition: Loss of RCS inventory

EAL:

CA2.1 **Alert**
Confirmed loss of inventory, Table C-4, **AND** Reactor Vessel level < bottom of the RCS hot leg as indicated by RVLIS full range < 63%

Table C-4 Inventory Loss Confirmatory Indicators
ANY of the following: - Standpipe level indication decreasing/bottomed out - RHR pump amp fluctuations - Decreasing RVLIS trend - Ultrasonic indications (if in service)

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

When Reactor Vessel water level decreases to 255.125 ft el., the bottom of the RCS hot leg penetration is uncovered. The elevation of the bottom of the RCS hot leg penetration can be monitored only by RVLIS full range (62.3%). Level monitoring instruments 1-RC-LI-102 (2-RC-LI-202), 1-RC-LI-103, (2-RC-LI-203) 1-RC-LI-105 (2-RC-LI-205) and RVLIS upper range are offscale low when level is below the elevation of the centerline of the RCS loop hot leg penetration (256.333 ft el.).

CA2.1 (cont)

Table C-4 provides a list of confirmatory indicators for RCS inventory loss. Due to the variability of accuracy and usability of RVLIS while in Cold Shutdown or Refueling Mode, the use of RVLIS for emergency classification purposes is contingent on one or more of the listed confirmatory indications.

The RVLIS full range threshold has been determined as follows (ref. 1):

Component	Elevation (ft)	Radius (in.)	RVLIS Full Range (%)
RCS hot leg centerline	256.333	14.5	63.0
Bottom of RCS hot leg	255.125	NA	A
6 in. below bottom of hot leg	254.625	NA	B
Top of fuel	252.807	NA	61.0

RVLIS span %/ft = 0.56721

A = 61.0% + (Bottom of RCS hot leg - Top of fuel) x RVLIS span
= 62.3%

B = 61.0% + (6 in. below bottom of hot leg - Top of fuel) x RVLIS span
= 62.0%

EAL RVLIS values rounded up to the nearest whole percentage point.

This EAL serves as a precursor to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further Reactor Vessel water level decrease and potential core uncovery. The inability to *restore* and *maintain* level after reaching this setpoint infers a *failure* of the RCS barrier.

CA2.1 (cont)

In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the fuel clad may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode, the RCS is not intact and Reactor Vessel water level and inventory are monitored by different means. In the Refueling mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel water level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

CA2.1 (cont)

NAPS Basis Reference(s):

1. 1-OP-5.4 (2-OP-5.4) Draining the Reactor Coolant System
2. UFSAR Section 6.2
3. UFSAR Figure 6.2-57
4. UFSAR Section 9.2
5. UFSAR Section 9.3
6. 1-OP-4.1 (2-OP-4.1) Controlling Procedure for Refueling
7. 1-AP-17 (2-AP-17) Shutdown LOCA
8. 1-GOP-13.0 (2-GOP-13.0) Alternate Core Cooling Method
9. 1-AP-11 (2-AP-11) Loss of RHR
10. 1-AP-52 (2-AP-52) Loss of Refueling Cavity Level During Refueling
11. NRC EAL FAQ 2006-001
12. NRC EAL FAQ 2006-008
13. 1-PT-52.2 (2-PT-52.2) Reactor Coolant System Leak Rate (Hand Calculation)
14. 1-PT-52.2A (2-PT-52.2A) Reactor Coolant System Leak Rate (Computer Calculation)

CA2.2

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 2 – RCS Level

Initiating Condition: Loss of RCS inventory

EAL:

CA2.2	**Alert**
Reactor Vessel level **cannot** be monitored for ≥ 15 min. **AND** an unexplained increase in **any** Table C-1 sump/tank level (Note 3)	

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table C-1 Sumps / Tanks
Reactor Containment Sump
Pressurizer Relief Tank (PRT)
Primary Drain Transfer Tank (PDTT)
Component Cooling (CC) Surge Tank
Refueling Water Storage Tank (RWST)

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

This EAL serves as a precursor to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further Reactor Vessel water level decrease and potential core uncovery. The inability to *restore* and *maintain* level after reaching this setpoint infers a *failure* of the RCS barrier.

CA2.2 (cont)

In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the fuel clad may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode, the RCS is not intact and Reactor Vessel water level and inventory are monitored by different means. In the Refueling mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel water level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

In this EAL, all water level indication would be *unavailable*, and the Reactor Vessel inventory loss must be detected by sump or tank level changes. Surveillance procedures provide instructions for calculating *primary system* leak rate by manual or computer-based water inventory balances (ref. 13, 14). Sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Containment to ensure they are indicative of RCS leakage.

CA2.2 (cont)

The 15-minute interval for the loss of level indication was chosen because it is half of the Site Area Emergency EAL duration. The interval allows this EAL to be an effective precursor to the Site Area Emergency EAL CS2.3. Significant fuel damage is not expected to occur until the core has been uncovered for greater than one hour. Therefore this EAL meets the definition for an Alert emergency.

NAPS Basis Reference(s):

1. 1-OP-5.4 (2-OP-5.4) Draining the Reactor Coolant System
2. UFSAR Section 6.2
3. UFSAR Figure 6.2-57
4. UFSAR Section 9.2
5. UFSAR Section 9.3
6. 1-OP-4.1 (2-OP-4.1) Controlling Procedure for Refueling
7. 1-AP-17 (2-AP-17) Shutdown LOCA
8. 1-GOP-13.0 (2-GOP-13.0) Alternate Core Cooling Method
9. 1-AP-11 (2-AP-11) Loss of RHR
10. 1-AP-52 (2-AP-52) Loss of Refueling Cavity Level During Refueling
11. NRC EAL FAQ 2006-001
12. NRC EAL FAQ 2006-008
13. 1-PT-52.2 (2-PT-52.2) Reactor Coolant System Leak Rate (Hand Calculation)
14. 1-PT-52.2A (2-PT-52.2A) Reactor Coolant System Leak Rate (Computer Calculation)

CS2.1

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 2 – RCS Level

Initiating Condition: Loss of Reactor Vessel inventory affecting core decay heat removal capability

EAL:

CS2.1 Site Area Emergency
Containment closure **not** established (Note 4) with confirmed loss of inventory, Table C-4, **AND** RVLIS full range < 62%

Note 4: Containment closure established means potential escape paths for fission product radioactivity within containment are closed, preventing release to the environment.

Table C-4 Inventory Loss Confirmatory Indicators
ANY of the following: - Standpipe level indication decreasing/bottomed out - RHR pump amp fluctuations - Decreasing RVLIS trend - Ultrasonic indications (if in service)

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

CS2.1 (cont)

Basis:

When Reactor Vessel water level decreases to 254.625 ft el., water level is six inches below the elevation of the bottom of the RCS hot leg penetration. When Reactor Vessel water level drops significantly below the elevation of the bottom of the RCS hot leg penetration, all sources of RCS injection have failed or are incapable of making up for the inventory loss. Six inches below the elevation of the bottom of the RCS hot leg penetration can be monitored only by RVLIS full range (62.0%). Level monitoring instruments 1-RC-LI-102 (2-RC-LI-202), 1-RC-LI-103, (2-RC-LI-203) 1-RC-LI-105 (2-RC-LI-205) and RVLIS upper range are offscale low when level is below the elevation of the centerline of the RCS loop hot leg penetration (256.333 ft el.).

Table C-4 provides a list of confirmatory indicators for RCS inventory loss. Due to the variability of accuracy and usability of RVLIS while in Cold Shutdown or Refueling Mode, the use of RVLIS for emergency classification purposes is contingent on one or more of the listed confirmatory indications.

The RVLIS full range threshold has been determined as follows (ref. 1):

CS2.1 (cont)

Component	Elevation (ft)	Radius (in.)	RVLIS Full Range (%)
RCS hot leg centerline	256.333	14.5	63.0
Bottom of RCS hot leg	255.125	NA	A
6 in. below bottom of hot leg	254.625	NA	B
Top of fuel	252.807	NA	61.0

RVLIS span %/ft = 0.56721

A = 61.0% + (Bottom of RCS hot leg - Top of fuel) x RVLIS span
= 62.3%

B = 61.0% + (6 in. below bottom of hot leg - Top of fuel) x RVLIS span
= 62.0%

Under the conditions specified by this EAL, continued decrease in Reactor Vessel water level is indicative of a loss of inventory control. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the Reactor Vessel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS or Reactor Vessel water level decrease and potential core uncovery. The inability to *restore* and *maintain* level after reaching this setpoint infers a *failure* of the RCS barrier and potential loss of the Fuel Clad barrier.

Containment closure is the action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. Potential escape paths for fission product radioactivity within containment are required to be *closed* to prevent the release to the environment (ref. 2). The status of *Containment closure* is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal.

CS2.1 (cont)

In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the fuel clad may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel.

This EAL escalates to a General Emergency via CG2.1 or based on radiological effluents.

NAPS Basis Reference(s):

1. 1-OP-5.4 (2-OP-5.4) Draining the Reactor Coolant System
2. OU-AA-200, Shutdown Risk Management
3. UFSAR Section 6.2
4. UFSAR Figure 6.2-57
5. UFSAR Section 9.2
6. UFSAR Section 9.3
7. 1-OP-4.1 (2-OP-4.1) Controlling Procedure for Refueling
8. 1-AP-17 (2-AP-17) Shutdown LOCA
9. 1-GOP-13.0 (2-GOP-13.0) Alternate Core Cooling Method
10. 1-AP-11 (2-AP-11) Loss of RHR
11. 1-AP-52 (2-AP-52) Loss of Refueling Cavity Level During Refueling
12. 1-PT-91 (2-PT-91) Containment Penetrations
13. NRC EAL FAQ 2006-001
14. 1-PT-52.2 (2-PT-52.2) Reactor Coolant System Leak Rate (Hand Calculation)
15. 1-PT-52.2A (2-PT-52.2A) Reactor Coolant System Leak Rate (Computer Calculation)
16. 1-OP-1.1 (2-OP-1.1) Unit Startup from Mode 5 Less than 140 F to Mode 5 at Less than 200 F

CS2.2

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 2 – RCS Level

Initiating Condition: Loss of Reactor Vessel inventory affecting core decay heat removal capability

EAL:

CS2.2 Site Area Emergency

Containment closure established (Note 4) with confirmed loss of inventory, Table C-4,

AND RVLIS full range < 61%

Note 4: Containment closure established means potential escape paths for fission product radioactivity within containment are closed, preventing release to the environment.

Table C-4 Inventory Loss Confirmatory Indicators
ANY of the following: - Standpipe level indication decreasing/bottomed out - RHR pump amp fluctuations - Decreasing RVLIS trend - Ultrasonic indications (if in service)

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

CS2.2 (cont)

Basis:

When Reactor Vessel water level drops significantly below the elevation of the bottom of the RCS hot leg penetration, all sources of RCS injection have failed or are incapable of making up for the inventory loss. This level drop can only be remotely monitored by Reactor Vessel Level Instrumentation System (RVLIS). When Reactor Vessel water level drops below RVLIS full range setpoint of 61.0% (ref. 1), core uncovery is about to occur. Table C-4 provides a list of confirmatory indicators for RCS inventory loss. Due to the variability of accuracy and usability of RVLIS while in Cold Shutdown or Refueling Mode, the use of RVLIS for emergency classification purposes is contingent on one or more of the listed confirmatory indications.

Under the conditions specified by this EAL, continued decrease in Reactor Vessel water level is indicative of a loss of inventory control. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the Reactor Vessel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS or Reactor Vessel water level decrease and potential core uncovery. The inability to *restore* and *maintain* level after reaching this setpoint infers a *failure* of the RCS barrier and Potential Loss of the Fuel Clad barrier.

Containment closure is the action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. Potential escape paths for fission product radioactivity within containment are required to be *closed* to prevent the release to the environment (ref. 2). The status of *Containment closure* is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal.

CS2.2 (cont)

In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed.

Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the fuel clad may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling.

This EAL escalates to a General Emergency via CG2.1 or based on radiological effluents.

NAPS Basis Reference(s):

1. 1-OP-5.4 (2-OP-5.4) Draining the Reactor Coolant System
2. OU-AA-200, Shutdown Risk Management
3. UFSAR Section 6.2
4. UFSAR Figure 6.2-57
5. UFSAR Section 7.5
6. UFSAR Table 7.5-3
7. UFSAR Section 9.2
8. UFSAR Section 9.3
9. 1-OP-4.1 (2-OP-4.1) Controlling Procedure for Refueling
10. 1-AP-17 (2-AP-17) Shutdown LOCA
11. 1-GOP-13.0 (2-GOP-13.0) Alternate Core Cooling Method
12. 1-AP-11 (2-AP-11) Loss of RHR
13. 1-AP-52 (2-AP-52) Loss of Refueling Cavity Level During Refueling
14. 1-PT-91 (2-PT-91) Containment Penetrations
15. NRC EAL FAQ 2006-001

CS2.3

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 2 – RCS Level

Initiating Condition: Loss of Reactor Vessel inventory affecting core decay heat removal capability

EAL:

CS2.3 Site Area Emergency

Reactor Vessel level **cannot** be monitored for $\geq$ 30 min. (Note 3) with a loss of RCS inventory as indicated **any** of the following:

- Manipulator Crane Area RM-RMS-162 (RM-RMS-262) > 5 R/hr
- Erratic source range monitor indication
- Unexplained increase in any Table C-1 sump/tank level

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table C-1 Sumps / Tanks
Reactor Containment Sump
Pressurizer Relief Tank (PRT)
Primary Drain Transfer Tank (PDTT)
Component Cooling (CC) Surge Tank
Refueling Water Storage Tank (RWST)

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

CS2.3 (cont)

Bases:

In the Refueling mode, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly less than in the Cold Shutdown mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. The heatup and the threat to damaging the fuel clad thus may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel than for events that occur in the Cold Shutdown mode. The reduced RCS heatup rate lowers boil-off and may slow the loss of vessel inventory.

This EAL is based on concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal, SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues, NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States, and, NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management. A number of variables, (e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining, etc.) can have a significant impact on heat removal capability challenging the Fuel Clad barrier. Analysis indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30 minutes was chosen.

In Refueling mode, Reactor Vessel water level indication from RVLIS is likely *unavailable* but alternate means of level indication are normally installed (including visual observation) to assure that the ability to monitor water level will not be interrupted. The Reactor Vessel inventory loss may be detected by the Manipulator Crane Area Radiation Monitor RM-RMS-162 (RM-RMS-262) or erratic Source Range Monitor indication. Manipulator Crane Area Radiation Monitor RM-RMS-162 (RM-RMS-262) > 5 R/hr is the area radiation monitor reading indicative of core uncovery (ref. 3).

CS2.3 (cont)

Surveillance procedures provide instructions for calculating *primary system* leak rate by manual or computer-based water inventory balances (ref. 12, 13). Sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Containment to ensure they are indicative of RCS leakage. Post-TMI studies indicate that the installed nuclear instrumentation will operates erratically when the core is uncovered and Source Range Monitors (SRM) can be used as a tool for making such determinations. SRM count rate can be indicated in the Control Room by (ref. 14):

- NI-31 and NI-32 source range meters on the MCR Bench Board
- Source range detector output displayed on NR-45)
- Source range meters on the NIS rack

This EAL escalates to a General Emergency via CG2.1 or based on radiological effluents**.**

NAPS Basis Reference(s):

1. 1-OP-5.4 (2-OP-5.4) Draining the Reactor Coolant System
2. OU-AA-200, Shutdown Risk Management
3. Engineering Transmittal ET-NAF-06-0114, Dose Rate at the Containment Manipulator Crane due to a Draindown Event Including Scatter from the Air and Containment Dome
4. 1-OP-4.1 (2-OP-4.1) Controlling Procedure for Refueling
5. 1-AP-17 (2-AP-17) Shutdown LOCA
6. 1-GOP-13.0 (2-GOP-13.0) Alternate Core Cooling Method
7. 1-AP-11 (2-AP-11) Loss of RHR
8. 1-AP-52 (2-AP-52) Loss of Refueling Cavity Level During Refueling
9. NRC EAL FAQ 2006-005, 009, 010, 011
10. 1-PT-91 (2-PT-91) Containment Penetrations
11. CALC-RAD PA-0227, Rev. 0, Dose Rate at the Containment Manipulator Crane Monitor due to a Draindown Event at North Anna or Surry.
12. 1-PT-52.2 (2-PT-52.2) Reactor Coolant System Leak Rate (Hand Calculation)
13. 1-PT-52.2A (2-PT-52.2A) Reactor Coolant System Leak Rate (Computer Calculation)
14. 1-OP-1.1(2-OP-1.1) Unit Startup from Mode 5 Less than 140 F to Mode 5 at Less than 200 F

CG2.1

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 2 – RCS Level

Initiating Condition: Loss of Reactor Vessel inventory affecting fuel clad integrity with Containment challenged and irradiated fuel in the Reactor Vessel

EAL:

CG2.1 General Emergency

Core uncovery for > 30 min. (Note 3) as indicated by confirmed loss of inventory, Table C-4, **AND** RVLIS full range < 61%

AND

Containment challenged as indicated by **any** of the following:

- Containment closure **not** established (Note 4)
- Containment hydrogen concentration ≥ 4%
- Unplanned rise in containment pressure

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Note 4: Containment closure established means potential escape paths for fission product radioactivity within containment are closed, preventing release to the environment.

Table C-4 Inventory Loss Confirmatory Indicators
ANY of the following: - Standpipe level indication decreasing/bottomed out - RHR pump amp fluctuations - Decreasing RVLIS trend - Ultrasonic indications (if in service)

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

CG2.1 (cont)

Basis:

Three conditions are associated with a challenge to Containment integrity:

- *Containment closure* is the action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. Potential escape paths for fission product radioactivity within Containment are required to be *closed* to prevent the release to the environment (ref. 1). The status of *Containment closure* is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal.
- The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations (ref. 2). To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers are likely to have occurred. Containment hydrogen can be monitored in the Control Room on 1-HC-H2A-101 (2-HC-H2A-201).
- Per T.S. (ref. 20) the containment pressure is not expected to rise when the unit is in the Cold Shutdown or Refueling mode. The requirement for *Containment closure* in these modes does not specify design pressure, only that all paths to the environment are *closed*. Therefore any rise in pressure, in combination with loss of inventory or the inability to monitor inventory, can be considered a challenge to *Containment closure*.

When Reactor Vessel water level drops below the RVLIS full range setpoint of 61% (ref. 4), core uncovery is about to occur. RVLIS is the only remotely indicating level monitoring system capable of indicating water level in the Reactor Vessel between the bottom of the RCS hot leg and the top of active fuel. In Refueling mode, however, RVLIS is usually inoperable. Table C-4 provides a list of confirmatory indicators for RCS inventory loss. Due to the variability of accuracy and usability of RVLIS while in Cold Shutdown or Refueling Mode, the use of RVLIS for emergency classification purposes is contingent on one or more of the listed confirmatory indications.

CG2.1 (cont)

This EAL is based on concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal, SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues, NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States, and NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management. A number of variables (e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining, etc.) can have a significant impact on heat removal capability challenging the Fuel Clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the 30-minute interval was conservatively chosen.

The General Emergency is declared on the occurrence of the loss or potential loss of the function of all three *fission product barriers*. Based on the above discussion, RCS barrier *failure* resulting in core uncovery for 30 minutes or more may cause fuel clad *failure*. With the Containment breached or challenged, the potential for unmonitored fission product release to the environment is high. This is consistent with the definition of a General Emergency.

CG2.1 (cont)

NAPS Basis Reference(s):

1. OU-AA-200, Shutdown Risk Management
2. 1-FR-C.1 (2-FR-C.1) Response to Inadequate Core Cooling
3. UFSAR Section 6.2.1
4. 1-OP-5.4 (2-OP-5.4) Draining the Reactor Coolant System
5. Engineering Transmittal ET-NAF-06-0114, Dose Rate at the Containment Manipulator Crane due to a Draindown Event Including Scatter from the Air and Containment Dome
6. UFSAR Section 6.2
7. UFSAR Figure 6.2-57
8. UFSAR Section 7.5
9. UFSAR Table 7.5-3
10. UFSAR Section 9.2
11. UFSAR Section 9.3
12. 1-OP-4.1 (2-OP-4.1) Controlling Procedure for Refueling
13. 1-AP-17 (2-AP-17) Shutdown LOCA
14. 1-GOP-13.0 (2-GOP-13.0) Alternate Core Cooling Method
15. 1-AP-11 (2-AP-11) Loss of RHR
16. 1-AP-52 (2-AP-52) Loss of Refueling Cavity Level During Refueling
17. 1-OP-63.2 (2-OP-63.2) Containment Hydrogen Analyzer
18. 1-PT-91 (2-PT-91) Containment Penetrations
19. Calc PA-0227, Rev. 0, Dose Rate at the Containment Manipulator Crane Monitor due to a Draindown Event at North Anna or Surry.
20. Technical Specification Bases B.3.9.4
21. NRC EAL FAQ 2006-001
22. NRC EAL FAQ 2006-019
23. 1-PT-52.2 (2-PT-52.2) Reactor Coolant System Leak Rate (Hand Calculation)
24. 1-PT-52.2A (2-PT-52.2A) Reactor Coolant System Leak Rate (Computer Calculation)
25. 1-OP-1.1(2-OP-1.1) Unit Startup from Mode 5 Less than 140 F to Mode 5 at Less than 200 F

CG2.2

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 2 – RCS Level

Initiating Condition: Loss of Reactor Vessel inventory affecting fuel clad integrity with Containment challenged and irradiated fuel in the Reactor Vessel

EAL:

CG2.2 General Emergency

Reactor Vessel level **cannot** be monitored for ≧ 30 min. (Note 3) with a loss of RCS inventory as indicated by **any** of the following:

- Manipulator Crane Area RM-RMS-162 (RM-RMS-262) > 5 R/hr
- Erratic source range monitor indication
- Unexplained increase in **any** Table C-1 sump/tank level

AND

Containment challenged as indicated by **any** of the following:

- Containment closure **not** established (Note 4)
- Containment hydrogen concentration ≥ 4%
- Unplanned rise in containment pressure

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Note 4: Containment closure established means potential escape paths for fission product radioactivity within containment are closed, preventing release to the environment.

Table C-1 Sumps / Tanks
Reactor Containment Sump
Pressurizer Relief Tank (PRT)
Primary Drain Transfer Tank (PDTT)
Component Cooling (CC) Surge Tank
Refueling Water Storage Tank (RWST)

CG2.2 (cont)

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

Three conditions are associated with a challenge to Containment integrity:

- *Containment closure* is the action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. Potential escape paths for fission product radioactivity within containment are required to be *closed* to prevent the release to the environment (ref. 1). The status of *Containment closure* is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal.
- The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations (ref. 2). To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers are likely to have occurred. Containment hydrogen can be monitored in the Control Room on 1-HC-H2A-101 (2-HC-H2A-201).
- Per T.S. (ref. 20) the containment pressure is not expected to rise when the unit is in the Cold Shutdown or Refueling mode. The requirement for *containment closure* in these modes does not specify design pressure, only that all paths to the environment are *closed*. Therefore any rise in pressure, in combination with loss of inventory or the inability to monitor inventory, can be considered a challenge to *containment closure*.

CG2.2 (cont)

This EAL is based on concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal, SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues, NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States, and NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management. A number of variables (e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining, etc.) can have a significant impact on heat removal capability challenging the Fuel Clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the 30-minute interval was conservatively chosen.

If all means of level monitoring are not available, the Reactor Vessel inventory loss may be detected by the following indirect methods:

- Manipulator Crane Area Radiation Monitor RM-RMS-162 (RM-RMS-262) > 5 R/hr (ref. 5) is the area radiation monitor reading indicative of core uncovery.
- Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and Source Range Monitors (SRM) can be used as a tool for making such determinations. SRM count rate can be indicated in the Control Room by (ref. 25):
    - NI-31 and NI-32 source range meters on the MCR Bench Board
    - Source range detector output displayed on recorder NR-45
    - Source range meters on the NIS rack

CG2.2 (cont)

- Sump or tank level changes may be indicative of a loss of RCS inventory. Surveillance procedures provide instructions for calculating *primary system* leak rate by manual or computer-based water inventory balances (ref. 23, 24). Containment Sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Containment to ensure they are indicative of RCS leakage.

The General Emergency is declared on the occurrence of the loss or potential loss of the function of all three *fission product barriers*. Based on the above discussion, RCS barrier *failure* resulting in core uncovery for 30 minutes or more may cause fuel clad *failure*. With the Containment breached or challenged, the potential for unmonitored fission product release to the environment is high. This is consistent with the definition of a General Emergency.

CG2.2 (cont)

NAPS Basis Reference(s):

1. OU-AA-200, Shutdown Risk Management
2. 1-FR-C.1 (2-FR-C.1) Response to Inadequate Core Cooling
3. UFSAR Section 6.2.1
4. 1-OP-5.4 (2-OP-5.4) Draining the Reactor Coolant System
5. Engineering Transmittal ET-NAF-06-0114, Dose Rate at the Containment Manipulator Crane due to a Draindown Event Including Scatter from the Air and Containment Dome
6. UFSAR Section 6.2
7. UFSAR Figure 6.2-57
8. UFSAR Section 7.5
9. UFSAR Table 7.5-3
10. UFSAR Section 9.2
11. UFSAR Section 9.3
12. 1-OP-4.1 (2-OP-4.1) Controlling Procedure for Refueling
13. 1-AP-17 (2-AP-17) Shutdown LOCA
14. 1-GOP-13.0 (2-GOP-13.0) Alternate Core Cooling Method
15. 1-AP-11 (2-AP-11) Loss of RHR
16. 1-AP-52 (2-AP-52) Loss of Refueling Cavity Level During Refueling
17. 1-OP-63.2 (2-OP-63.2) Containment Hydrogen Analyzer
18. 1-PT-91 (2-PT-91) Containment Penetrations
19. Calc PA-0227, Rev. 0, Dose Rate at the Containment Manipulator Crane Monitor due to a Draindown Event at North Anna or Surry.
20. Technical Specification Bases B.3.9.4
21. NRC EAL FAQ 2006-001
22. NRC EAL FAQ 2006-019
23. 1-PT-52.2 (2-PT-52.2) Reactor Coolant System Leak Rate (Hand Calculation)
24. 1-PT-52.2A (2-PT-52.2A) Reactor Coolant System Leak Rate (Computer Calculation)
25. 1-OP-1.1(2-OP-1.1) Unit Startup from Mode 5 Less than 140 F to Mode 5 at Less than 200 F

CU3.1

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 3 – RCS Temperature

Initiating Condition: *Unplanned* loss of decay heat removal capability with irradiated fuel in the Reactor Vessel

EAL:

CU3.1	**Notification of Unusual Event**
An *unplanned* event results in RCS temperature > 200°F	

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

This EAL is an Notification of Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In Cold Shutdown mode, the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the Cold Shutdown mode, a large inventory of water is available to keep the core covered. In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown conditions may be attained within hours of operating at power. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shut down. Thus, the heatup threat and the threat to damaging the fuel clad may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling. In addition, the operators should be able to monitor RCS temperature and Reactor Vessel level so that escalation to the Alert under EAL CA2.1, CA2.2 or CA3.1 will occur if required.

CU3.1 (cont)

During refueling operations, the level in the Reactor Vessel will normally be maintained above the vessel flange. Refueling operations that lower water level below the vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RCS/Reactor Vessel temperatures depending on the time since shutdown. Escalation directly to the Alert under EAL CA3.1 is provided should an *unplanned* event result in RCS temperature exceeding the Technical Specification cold shutdown temperature limit with *containment closure* not established.

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1). These include (ref. 2):

	Unit 1	Unit 2
• RCS C Loop WR T_H	T0459A	T0459A
• RHR P_P Disch T	T0630A	T0630A
• RCS Loop A T_C	T0406A	T0406A
• RCS Loop C WR T_C	T0446A	T0446A
• RHR Return Temp	TIRH001A	T2RH001A
• PRZR Surge Temp	T0482A	T0482A
• PRZR Liquid Temp	T0480A or Y9015A	T0480A or Y9015A
• Core Exit Thermocouples		

The SEM must remain attentive to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the SEM, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

NAPS Basis Reference(s):

1. Technical Specifications Table 1.1-1, Modes Definition for Cold Shutdown
2. 1-OP-1.1 (2-OP-1.1) Unit Startup from Mode 5
3. NRC EAL FAQ 2006-012

CU3.2

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 3 – RCS Temperature

Initiating Condition: *Unplanned* loss of decay heat removal capability with irradiated fuel in the Reactor Vessel

EAL:

CU3.2	**Notification of Unusual Event**
Loss of **all** RCS temperature and Reactor Vessel level indication for > 15 min. (Note 3)	

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

This EAL is an Notification of Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In Cold Shutdown mode, the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the Cold Shutdown mode, a large inventory of water is available to keep the core covered. In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown conditions may be attained within hours of operating at power. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shut down. Thus, the heatup threat and the threat to damaging the fuel clad may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling. In addition, the operators should be able to monitor RCS temperature and Reactor Vessel level so that escalation to the Alert under EAL CA2.1, CA2.2 or CA3.1 will occur if required.

CU3.2 (cont)

During refueling operations, the level in the Reactor Vessel will normally be maintained above the vessel flange. Refueling operations that lower water level below the vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RCS/Reactor Vessel temperatures depending on the time since shutdown. Escalation directly to the Alert under EAL CA3.1 is provided should an *unplanned* event result in RCS temperature exceeding the Technical Specification cold shutdown temperature limit with *containment closure* not established.

Unlike the Cold Shutdown mode, normal means of RCS temperature indication and Reactor Vessel level indication may not be available in the Refueling mode. Redundant means of Reactor Vessel level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the Cold Shutdown or Refueling modes, this EAL would result in declaration of an Notification of Unusual Event if either temperature or level indication cannot be *restored* within 15 minutes from the loss of both means of indication. Escalation to Alert under EAL CA3.1 would be based on an increase of RCS pressure or exceeding the temperature criterion (200°F, ref. 1).

Reactor Vessel water level is normally monitored using the following instruments (ref. 2):

- RVLIS Train A/B Full Range
- RVLIS Train A/B Upper Range, if loops are not isolated
- 1-RC-LI-102 (2-RC-LI-202) Level Standpipe via camera in Containment
- 1-RC-LI-103 (2-RC-LI-203) Cold Shutdown RCS Level Indicator
- 1-RC-LI-105 (2-RC-LI-205) Independent RCS Level Indicator

CU3.2 (cont)

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1). These include (ref. 3):

	Unit 1	Unit 2
• RCS C Loop WR T_H	T0459A	T0459A
• RHR P_P Disch T	T0630A	T0630A
• RCS Loop A T_C	T0406A	T0406A
• RCS Loop C WR T_C	T0446A	T0446A
• RHR Return Temp	TIRH001A	T2RH001A
• PRZR Surge Temp	T0482A	T0482A
• PRZR Liquid Temp	T0480A or Y9015A	T0480A or Y9015A

The SEM must remain attentive to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the SEM, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

NAPS Basis Reference(s):

1. Technical Specifications Table 1.1-1, Modes Definition for Cold Shutdown
2. 1-OP-5.4 (2-OP-5.4) Draining the Reactor Coolant System
3. 1-OP-1.1 (2-OP-1.1) Unit Startup from Mode 5
4. 1-OP-4.1 (2-OP-4.1) Controlling Procedure for Refueling
5. 1-AP-17 (2-AP-17) Shutdown LOCA
6. 1-GOP-13.0 (2-GOP-13.0) Alternate Core Cooling Method
7. 1-AP-11 (2-AP-11) Loss of RHR
8. 1-AP-52 (2-AP-52) Loss of Refueling Cavity Level During Refueling
9. NRC EAL FAQ 2006-012

CA3.1

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 3 – RCS Temperature

Initiating Condition: Inability to *maintain* plant in cold shutdown with irradiated fuel in the Reactor Vessel

EAL:

CA3.1 Alert

An *unplanned* event results in RCS temperature > 200°F for > Table C-3 duration (Note 3)

OR

RCS pressure increase of > 10 psig due to a loss of RCS cooling (does not apply in solid plant operations)

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table C-3 RCS Reheat Duration Thresholds

RCS	Containment Closure (Note 4)	Duration
Intact and **not** Reduced/Decreased Inventory	N/A	60 minutes*
Not Intact **OR** Reduced / Decreased Inventory	Established	20 minutes*
	Not Established	0 minutes

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, this EAL is **not** applicable.

Note 4: Containment closure established means potential escape paths for fission product radioactivity within containment are closed, preventing release to the environment.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

CA3.1 (cont)

Basis:

This EAL is based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." A number of phenomena such as pressurization, vortexing, steam generator U-tube draining, RCS level differences when operating at a mid-loop condition, decay heat removal system design and level instrumentation problems can lead to conditions in which decay heat removal is lost and core uncovery can occur. NRC analyses show that some event sequences can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1). These include (ref. 2):

	Unit 1	**Unit 2**
• RCS C Loop WR T_H	T0459A	T0459A
• RHR P_P Disch T	T0630A	T0630A
• RCS Loop A T_C	T0406A	T0406A
• RCS Loop C WR T_C	T0446A	T0446A
• RHR Return Temp	TIRH001A	T2RH001A
• PRZR Surge Temp	T0482A	T0482A
• PRZR Liquid Temp	T0480A or Y9015A	T0480A or Y9015A
• Core Exit Thermocouples		

The first threshold in Table C-3 addresses complete loss of functions required for core cooling for greater than 60 minutes during Refueling and Cold Shutdown modes when RCS integrity is established (irrespective of the status of *containment closure*). RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals). The status of *containment closure* in this threshold is immaterial given that the RCS is providing a high-pressure barrier to fission product release to the environment.

CA3.1 (cont)

The 60-minute interval should allow sufficient time to *restore* cooling without a substantial degradation in plant safety. The asterisk highlights the note at the bottom of the table.

Containment closure is the action to secure Containment as a functional barrier to fission product release during plant shutdown conditions (ref. 3). The status of *Containment closure* is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal.

The second and third thresholds in Table C-3 address the complete loss of functions required for core cooling during Refueling and Cold Shutdown modes for either:

- Greater than 20 minutes when *Containment closure* is established, but RCS integrity is either not established or at *Reduced/Decreased Inventory*.
- Immediately when neither *Containment closure* nor RCS integrity are established (or at Reduced/Decreased Inventory).

Decreased Inventory is defined as a condition with fuel in the Reactor Vessel and any RCS Loop Stop Valve *closed*, or RCS water level less than five percent (5%) in the pressurizer. With the Reactor Vessel Head removed and the Reactor Cavity filled to at least 23 feet above the Reactor Vessel Flange, the RCS is not considered to be in a decreased inventory condition. (VPAP-2805, Shutdown Risk Program).

Reduced Inventory Condition is defined as a condition with fuel in the Reactor Vessel and water level lower than three feet below the Reactor Vessel flange. This corresponds to 42 inches above RCS Hot Leg center-line (VPAP-2805, Shutdown Risk Program).

CA3.1 (cont)

RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals). With the Pressurizer PORV(s) blocked open, the RCS is considered not intact. The allowed 20-minute interval is included to allow operator action to *restore* the heat removal function, if possible. The allowed time frame is consistent with the guidance provided by Generic Letter 88-17, "Loss of Decay Heat Removal" (discussed later in this basis) and is believed to be conservative given that a low pressure Containment barrier to fission product release is established. The asterisk highlights the note at the bottom of the table.

The note indicates that the second threshold is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the 20-minute interval.

No delay time is allowed with *Containment closure* not established and the RCS not intact or at Reduced/Decreased Inventory because of the evaporated reactor coolant that may be released into the Containment during this heatup condition could also be directly released to the environment.

Escalation to a Site Area Emergency would be under EAL CS2.1, CS2.2 or CS2.3 should boiling result in significant Reactor Vessel water level loss leading to core uncovery.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary *unplanned* excursion above 200°F (ref. 1) when the heat removal function is available.

1-RC-PI-1403B (2-RC-PI-2403B) and 1-RC-PI-1402B (2-RC-PI-2402B) are capable of measuring pressure to less than 10 psig (ref. 4, 5).

CA3.1 (cont)

The SEM must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the SEM, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

NAPS Basis Reference(s):

1. Technical Specifications Table 1.1-1, Modes Definition for Cold Shutdown
2. 1-OP-1.1 (2-OP-1.1) Unit Startup from Mode 5
3. OU-AA-200, Shutdown Risk Management
4. 1-ICP-RC-P1403 (2-ICP-RC-P2403) Reactor Coolant System Pressure (Wide and Narrow Range) Protection Channel IV Calibration
5. 1-ICP-RC-P1402 (2-ICP-RC-P2402) Reactor Coolant System Pressure (Wide and Narrow Range) Protection Channel I Calibration
6. 1-OP-4.1 (2-OP-4.1) Controlling Procedure for Refueling
7. 1-OP-5.4 (2-OP-5.4) Draining the Reactor Coolant System
8. 1-PT-91 (2-PT-91) Containment Penetrations
9. 1-LOG-18 (2-LOG-18) Containment Boundary Breach Log

CU4.1

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 4 – Communications

Initiating Condition: *Unplanned* loss of **all** onsite or offsite communications capabilities

EAL:

CU4.1 Notification of Unusual Event

Loss of **all** Table C-2 onsite (internal) communications capability affecting the ability to perform routine operations

OR

Loss of **all** Table C-2 offsite (external) communications capability

Table C-2 Communications Systems		
System	**Onsite (internal)**	**Offsite (external)**
Radio Communications System	**X**	
Public Address and Intercom System	**X**	
Private Branch Telephone Exchange (PBX)	**X**	
Sound Powered Telephone System	**X**	
Commercial Telephone		**X**
Dedicated NRC Communications		**X**

CU4.1 (cont)

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

This EAL addresses loss of communications capability that either prevents the plant operations staff from performing routine tasks necessary for onsite plant operations or inhibits the ability to communicate problems externally to offsite authorities from the Control Room. The loss of offsite communications ability encompasses the loss of all means of communications with offsite authorities and is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This should include Commercial Telephone System (CTS), FAX transmissions and dedicated phone systems. This EAL is applicable only when extraordinary means are being utilized to make communications possible (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.).

The station communications system is designed to provide redundant means to communicate with all essential areas of the station associated with Units 1 and 2 and to essential locations remote from the station during normal operation and under accident conditions. Communication systems vital to operation and safety are designed so that *failure* of one component would not impair the reliability of the total communications system. Onsite/offsite communications include one or more of the systems listed in Table C-2 (ref. 1).

CU4.1 (cont)

Onsite/offsite communications include one or more of the systems listed in Table C-2 (ref. 1).

- Radio Communications System (Onsite & Offsite)

  An Ultra-High Frequency (UHF) two-way radio trunking system is provided at the Station consisting of base stations/repeaters, mobile units installed in emergency vehicles, and hand-held portable radios. The radio trunking system provides redundancy and independent emergency backup equipment for designated station functions.

  The same UHF two-way radio trunking system that provides onsite communications also provides for communications within a ten mile radius of the Station. During an emergency, this system allows direct contact with Radiation Monitoring Teams, Security vehicles, and a separate channel (Talk Group) between the Security Alarm Stations and the Louisa County Sheriff's Department.

- Public Address and Intercom System (Onsite)

  A five channel public address and intercom system (Gai-Tronics System) is installed in the Station. The system power is supplied from a power supply which will *maintain* the system in an operational condition in the event of a normal station service power *failure*. Zones are provided within that Station to insure operability of a major portion of the system should equipment in a zone become inoperative. In the event of an emergency, the system is used to alert Station personnel of any emergency situation and to direct emergency response actions required of on-site personnel.

CU4.1 (cont)

- Private Branch Telephone Exchange, PBX (Onsite)

  The PBX switching equipment is physically located in the PBX Building and is connected to a commercial telephone exchange in Mineral, Virginia. Backup battery power is provided to *maintain* the system operable 6 to 8 hours following the loss of AC power.

- Sound Powered Telephone System (Onsite)

  This system is a multiple channel system connecting selected operating areas of the plant. Headsets consisting of an earphone and microphone are connected to a two wire channel for direct communication between persons in different areas. Operation of this system is not dependent on the availability of the electrical power system. During an emergency, the system would provide an alternate means of relaying messages.

- Commercial Telephone (Offsite)

  Commercial telephone lines are provided between the Station and a commercial telephone exchange in Mineral, Virginia. These lines are connected into the Station PBX. In addition, lines are provided for communications between the Station and the commercial telephone network which are independent of the Station PBX.

CU4.1 (cont)

- <u>Dedicated NRC Communications (Offsite)</u>

  Separate telephone lines are dedicated to the NRC and include the following:

  - Emergency Notification System (ENS): The system on which initial notifications, as well as ongoing information about plant systems, status and parameters, are provided to the NRC. ENS lines are located in the Control Room, TSC and LEOF.
  - Health Physics Network (HPN): Provides for communications regarding radiological and meteorological conditions, assessments, trends, and protective measures. HPN lines are located in the TSC and LEOF.
  - Reactor Safety Counterpart Link (RSCL): Allows for internal NRC discussions regarding plant and equipment conditions. RSCL lines are located in the TSC and LEOF.
  - Protective Measures Counterpart Link (PMCL): Allows for the conduct of internal NRC discussions on radiological releases, meteorological conditions, and protective measures. PMCL lines are located in the TSC and LEOF.
  - Emergency Response Data System (ERDS) Channel: Allows transmittal of reactor parametric data from the site to the NRC. ERDS data is transmitted from the PCS computer, via modem, to the NRC Operations Center.
  - Management Counterpart Link (MCL): This system has been established for internal discussions between the NRC Executive Team Director/members and the NRC Director of Site Operations or licensee management. MCL lines are located in the TSC and LEOF.

CU4.1 (cont)

- o Local Area Network (LAN) Access: Provides access to the NRC local area network. Telephone jacks are provided in the TSC and LEOF for NRC LAN access.

NAPS Basis Reference(s):

1. North Anna Power Station Emergency Plan, Section 7.2
2. UFSAR Section 9.5.2

CU5.1

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 5 – RCS leakage

Initiating Condition: RCS leakage

EAL:

CU5.1 Notification of Unusual Event
Inability to establish or *maintain* pressurizer level > 15% **OR** RCS target level band (if pressurizer level was intentionally being controlled below the low level setpoint) due to RCS leakage

Mode Applicability:

5 - Cold Shutdown

Basis:

The conditions of this EAL may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. The inability to establish or *maintain* pressurizer level above the low pressurizer level setpoint or RCS target level band (if RCS level was intentionally being maintained below the pressurizer low level setpoint) is indicative of the initiating condition. A level transmitter provides a signal that will actuate an alarm when the pressurizer liquid level falls to a fixed level setpoint (15%). The same signal will *trip* the pressurizer heaters "off" and close the letdown line isolation valves.

The phrase "…due to RCS leakage" signifies that classification under this EAL is only required if the low pressurizer level is due to undesired loss of coolant from the RCS.

CU5.1 (cont)

Analysis shows that the operator has 44 minutes after the initial alarm to take any appropriate action to ensure core immersion. The analysis further established that (1) one charging/safety injection pump will provide adequate flow to sustain the system in a safe condition and (2) an initial alarm signal low-pressurizer-level deviation alarm conservatively assumed at 16.4% will occur within 30 seconds of the event initiation, followed by another alarm (low-level heater cutoff) at 15% (ref. 2).

Other EALs (CU2.1 and CU2.2) address the Refueling mode. In cold shutdown, the RCS will normally be intact and RCS inventory and level monitoring means such as pressurizer level indication and makeup volume control tank levels are normally available. In the Refueling mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means.

NAPS Basis Reference(s):

1. UFSAR Section 5.6.2.4
2. UFSAR Section 5.5.4.3.1
3. NRC EAL FAQ 2006-014

CU6.1

Category: C – Cold Shutdown / Refueling System Malfunction

Sub-category: 6 – Inadvertent Criticality

Initiating Condition: Inadvertent criticality

EAL:

CU6.1	**Notification of Unusual Event**
An *unplanned sustained* positive startup rate observed on nuclear instrumentation	

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

Sustained is defined as prolonged, not intermittent or of transitory nature.

This EAL addresses criticality events that occur in Cold Shutdown or Refueling modes (NUREG1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States) such as fuel misloading events and inadvertent dilution events. This EAL indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification.

The term "*sustained*" is used in order to allow exclusion of expected short-term positive startup rates from planned fuel bundle or control rod movements during core alteration. These short-term positive startup rates are the result of the increase in neutron population due to subcritical multiplication.

NAPS Basis Reference(s):

1. FSAR Table 7.5-3

<u>Category E – ISFSI</u>

EAL Group: ANY (EALs in this category are applicable to any plant condition, hot or cold.)

An independent spent fuel storage installation (ISFSI) is a complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage. A significant amount of the radioactive material contained within a cask/canister must escape its packaging and enter the biosphere for there to be a significant environmental effect resulting from an accident involving the dry storage of spent nuclear fuel. Formal offsite planning is not required because the postulated worst-case accident involving an ISFSI has insignificant consequences to the public health and safety.

A Notification of Unusual Event is declared on the basis of the occurrence of an event of sufficient magnitude that a loaded cask *confinement boundary* is damaged or violated. This includes classification based on a loaded fuel storage cask/canister *confinement boundary* loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

A hostile security event that leads to a potential loss in the level of safety of the ISFSI is a classifiable event under Security category EAL HA4.1.

Minor surface damage that does not affect storage cask/canister boundary is excluded from the scope of these EALs.

EU1.1

Category: ISFSI

Sub-category: None

Initiating Condition: Damage to a loaded cask confinement boundary

EAL:

EU1.1	**Notification of Unusual Event**
Damage to a loaded cask/canister confinement boundary	

Mode Applicability:

All

Basis:

A NOUE in this EAL is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded Sealed Surface Storage Cask (SSSC) or NUHOMS Dry Storage Canister (DSC) confinement boundary is damaged or violated. This includes classification based on a loaded fuel storage cask confinement boundary loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

Confinement Boundary is defined as the barrier(s) between areas containing radioactive substances and the environment. Therefore, damage to a confinement boundary must be a confirmed physical breach between the spent fuel and the environment for either the SSSCs or the DSCs. For the SSSCs, this would also include a confirmed loss of both cask seals.

EU1.1 (cont)

NAPS Basis Reference(s):

1. NAPS ISFSI SAR Sections 1, 5 and 8
2. Transnuclear FSAR “NUHOMS Horizontal Modular Storage System for Irradiated Nuclear Fuel”
3. 0-OP-4.54 “Transfer Cask/Dry Sheilded Canister Transfer to ISFSI and Dry Shielded Canister Transfer from Transfer Cask to Horizontal Storage Module”

Category H – Hazards

EAL Group: ANY (EALs in this category are applicable to any plant condition, hot or cold.)

Hazards are non-plant, system-related events that can directly or indirectly affect plant operation, reactor plant safety or personnel safety.

The events of this category pertain to the following subcategories:

1. Natural & Destructive Phenomena

Natural events include hurricanes, earthquakes or tornados that have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety. Non-naturally occurring events that can cause damage to plant facilities and include *aircraft* crashes, *missile* impacts, etc.

2. *Fire* or *Explosion*

Fires can pose significant hazards to personnel and reactor safety. Appropriate for classification are *fires* within the site *Protected Area* or which may affect operability of vital equipment.

3. Toxic, Asphyxiant & Flammable Gas

Non-naturally occurring events that can cause damage to plant facilities and include toxic, asphyxiant or flammable gas leaks.

4. Security

Unauthorized entry attempts into the *Protected Area*, *bomb* threats, *sabotage* attempts, and actual security compromises threatening loss of physical control of the plant.

5. Control Room Evacuation

Events that are indicative of loss of Control Room habitability. If the Control Room must be evacuated, additional support for monitoring and controlling plant functions is necessary through the emergency response facilities.

6. Judgment

The EALs defined in other categories specify the predetermined symptoms or events that are indicative of emergency or potential emergency conditions and thus warrant classification. While these EALs have been developed to address the full spectrum of possible emergency conditions which may warrant classification and subsequent implementation of the Emergency Plan, a provision for classification of emergencies based on operator/management experience and judgment is still necessary. The EALs of this category provide the SEM the latitude to classify emergency conditions consistent with the established classification criteria based upon their judgment.

HU1.1

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the *Protected Area* or Main Dam

EAL:

HU1.1 Notification of Unusual Event

Seismic event identified by any **TWO** of the following:

- Earthquake felt in the plant
- "SYSTEM TRIGGER" indicator illuminated on the SYSCOM Network Control Center (NCC)
- National Earthquake Information Center (NEIC)

Mode Applicability:

All

Basis:

The method of detection with respect to emergency classification relies on the agreement of the shift operators on-duty in the Control Room that the suspected ground motion is a "felt earthquake" as well as the actuation of the NAPS seismic instrumentation. Consensus of the Control Room operators with respect to ground motion helps avoid unnecessary classification if the seismic instrumentation inadvertently *trip* or detect vibrations not related to an earthquake. Ground motion of sufficient acceleration will trigger the SYSCOM seismic instruments and illuminates the "SYSTEM TRIGGER" indicator on the SYSCOM NCC. Illumination of the "SYSTEM TRIGGER" indicates that an earthquake has been detected, which activates the Annunciator B-4, EARTHQUAKE SYSTEM TRIGGERED (Unit 1 Annunciator Panel A). When Annunciator B-4 activates, operators take action according to 0-AP-36, Seismic Event.

HU1.1 (cont)

As defined in the EPRI-sponsored “Guidelines for Nuclear Plant Response to an Earthquake”, dated October 1989, a “felt earthquake” is:

> “An earthquake of sufficient intensity such that: (a) the inventory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of Control Room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated. For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01g.”

Damage to some portions of the site may occur as a result of the felt earthquake but it should not affect the ability of *safety functions* to operate. This event escalates to an Alert under EAL HA1.1 if the earthquake *exceeds* Operating Basis Earthquake (OBE) levels.

NAPS Basis Reference(s):

1. 0-AP-36 Seismic Event
2. AR 1A-B4 Earthquake System Triggered
3. UFSAR Section 2.5.2.6
4. DC NA-11-01213

HU1.2

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the *Protected Area* or Main Dam

EAL:

HU1.2	**Notification of Unusual Event**
Report by plant personnel of tornado or high winds > 80 mph striking within *Protected Area* boundary	

Mode Applicability:

All

Basis:

This EAL is based on the assumption that a tornado striking (touching down) or design force winds (> 80 mph, ref. 1) within the *Protected Area* may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. If such damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under EAL HA1.2.

Wind speed recorders and plant computer points indicate wind speeds up to 100 mph (ref. 2). All station structures are designed, however, to withstand a basic wind speed of 80 mph.

The *Protected Area* is within the security isolation zone and is given in Dwg. 11715-FC-48A, Site Plan Vehicle Barrier System.

It is recognized that the wind speed instruments are located outside the Protected Area. For the purpose of this EAL, these wind speed recordings are assumed to be representative of wind speeds within the Protected Area boundary.

HU1.2 (cont)

NAPS Basis Reference(s):

1. UFSAR Section 3.3.1
2. MODULE NCRODP-68-NA Meteorological Monitoring System
3. Dwg. 11715-FC-48A Site Plan Vehicle Barrier System
4. NAPS Emergency Plan, Section 1.0 Definitions
5. 0-AP-41 Severe Weather Conditions

HU1.3

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the *Protected Area* or Main Dam

EAL:

HU1.3	**Notification of Unusual Event**
Report of turbine *failure* resulting in casing penetration or damage to turbine seals or turbine generator seals	

Mode Applicability:

All

Basis:

This EAL is intended to address main turbine rotating component *failures* of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. For example, a turbine *missile* can be caused by brittle fracture of a rotating turbine part at or near turbine operating speed, or by ductile fracture upon runaway after extensive, highly improbable, control system *failures*. In the event of *missile* ejection, the probability of a strike on a plant region is a function of the energy and direction of an ejected *missile* and of the orientation of the turbine with respect to the plant region. The turbine spins about the axis depicted in Figure H-1 and Figure H-2 (ref. 1). Of major concern is the potential for significant leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. It is not the intent of this EAL to classify minor operational leakage. Actual *fires* and flammable gas build up are appropriately classified through other EALs. This EAL is consistent with the definition of an Notification of Unusual Event while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of this emergency classification level, if appropriate, would be via HA1.3 based on damage done by projectiles generated by the failure.

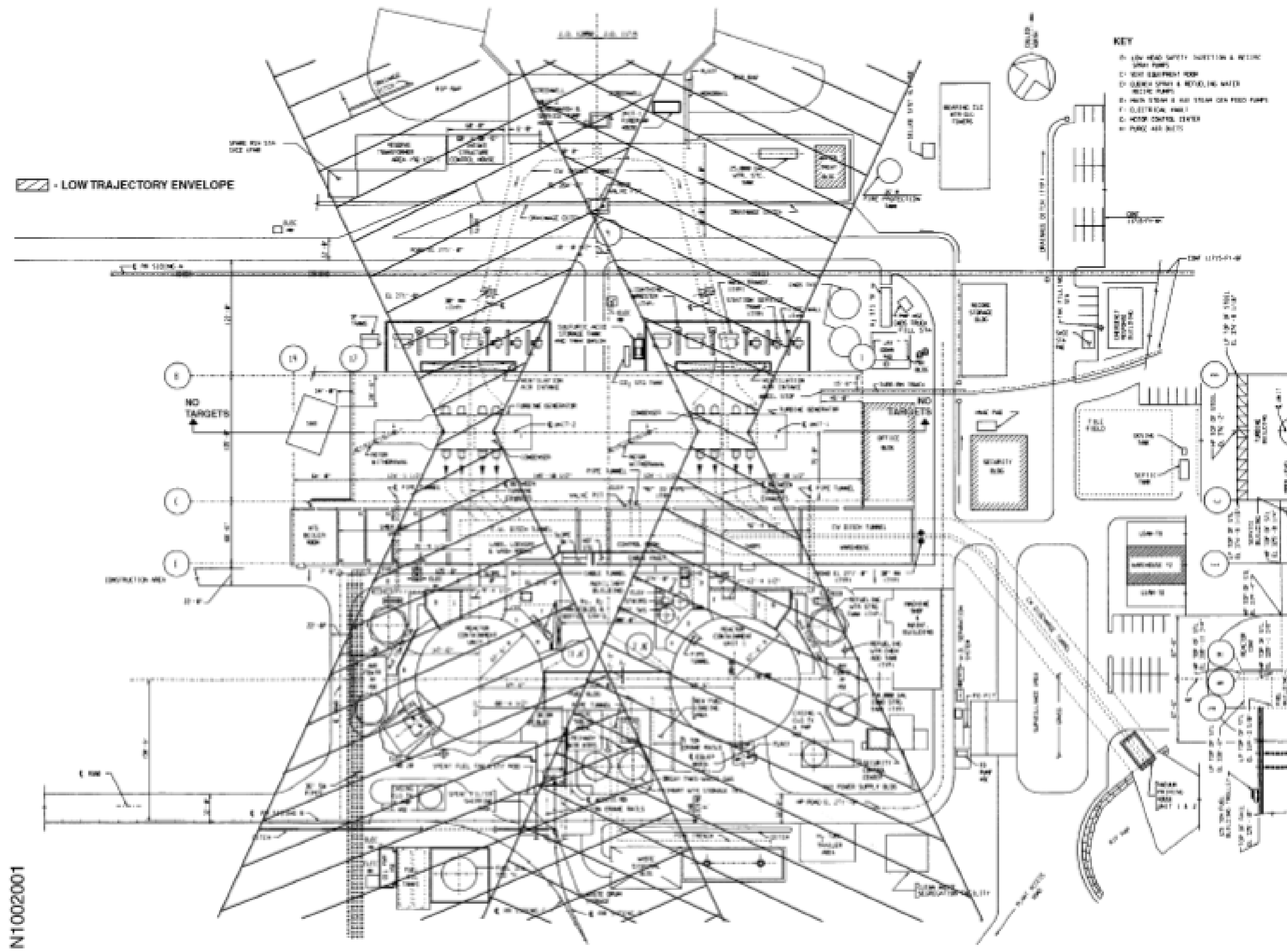
HU1.3 (cont)

NAPS Basis Reference(s):

1. UFSAR Section 10.2

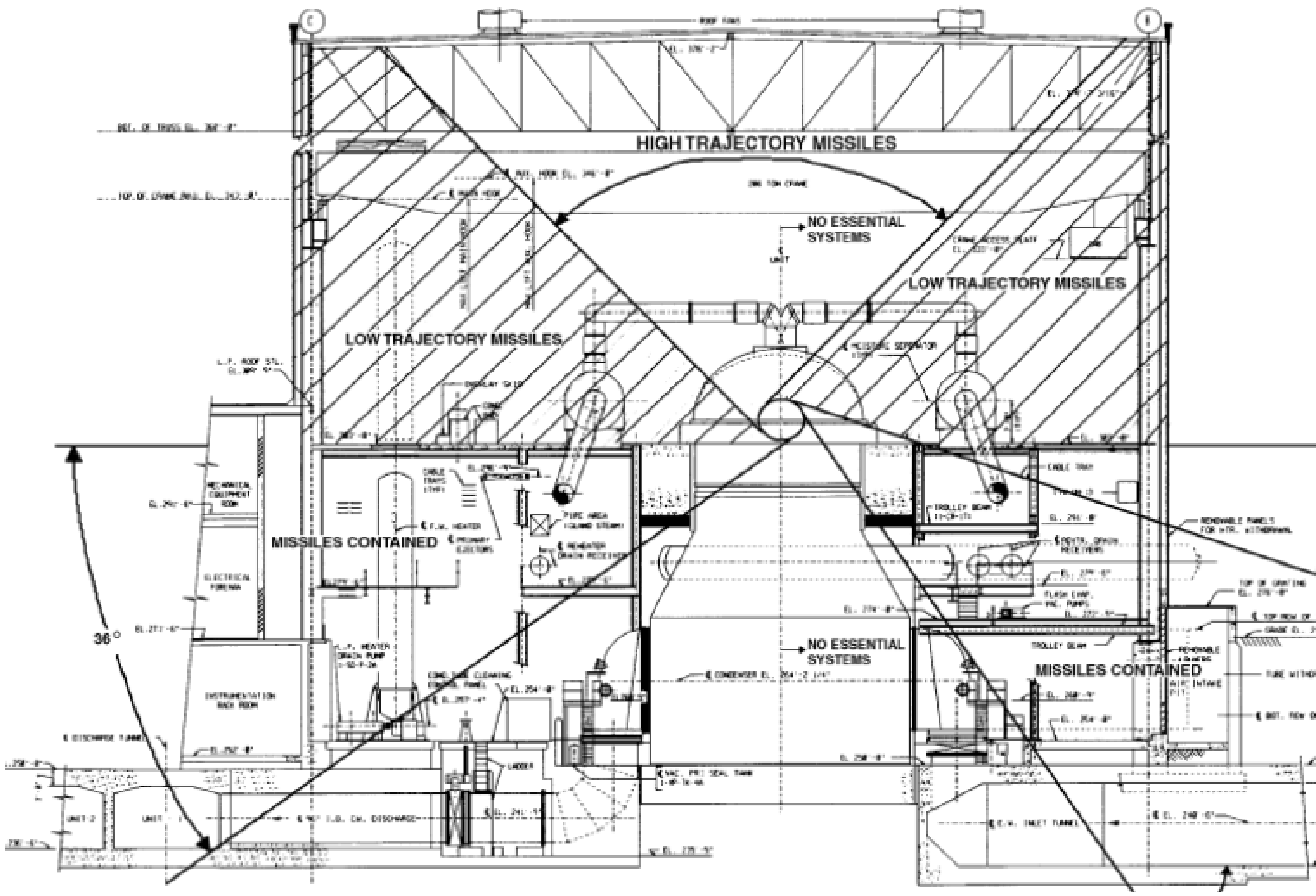
HU1.3 (cont)

Figure H-1: Turbine Axis and Postulated Missile Trajectories



HU1.3 (cont)

Figure H-2: Turbine Axis and Postulated Missile Trajectories (cont'd)



HU1.4

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the *Protected Area* or Main Dam

EAL:

HU1.4 Notification of Unusual Event

Uncontrolled flooding in **any** Table H-1 area that has the potential to affect safety related equipment needed for the current operating mode

Table H-1 Safe Shutdown Areas
• Cable Vaults & Tunnels • Emergency Switchgear Rooms • Emergency Diesel Generators Rooms • Reactor Containment • Quench Spray Pump Houses • Safeguards Areas • Main Steam Valve House • Cable Spreading Rooms • Control Room • CR Chiller Rooms • Auxiliary / Fuel / Decontamination Buildings • Fuel Oil Pump House Room A or B • Service Water Pump and Valve House • Intake Structure Control House • Auxiliary Service Water Pump House • Turbine Building • Auxiliary Feedwater Pump House

HU1.4 (cont)

Mode Applicability:

All

Basis:

This EAL addresses flooding caused by internal events (e.g., component *failures*, circulating water, component cooling or service water line ruptures, equipment misalignment, fire suppression system actuation, outage activity mishaps, etc.) that results in the potential to affect safety related equipment. *Uncontrolled* internal flooding that degrades safety-related equipment or creates a safety hazard precluding access necessary for the safe operation or monitoring of safety equipment warrants escalation to an Alert emergency classification under EAL HA1.4. The internal flooding areas are important drainage areas and typically contain systems that are:

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

Internal flooding in the Turbine building is alarmed by the following annunciators:

- Annunciator "D" Panel F-8, COND TUBE CLEAN PIT HI LEVEL
- Annunciator "D" Panel G-8, COND TUBE CLEAN PIT HI-HI LEVEL
- Annunciator "D" Panel G-7, TURB BLDG FLOOD ALARM TROUBLE

Turbine Building flooding could carry over into the Auxiliary Building through the Service Water Tunnel from the Valve Pit. Conversely, Auxiliary Building flooding could carry over into the Turbine Building through the Service Water Tunnel into the Turbine Building Valve Pit. Auxiliary Building Flooding could cause a loss of Charging Pumps (ref. 1, 2).

HU1.4 (cont)

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

NAPS Basis Reference(s):

1. 0-AP-39.1 Turbine Building Flooding
2. 0-AP-39.2 Auxiliary Building Flooding
3. NAPS IPE Section 3.3.7.1, Screening of Flood Areas
4. UFSAR Section 9.5.1
5. UFSAR Section 10.4
6. UFSAR Section 16.2
7. CM-AA-FPA-100, Fire Protection/Appendix R (Fire Safe Shutdown) Program
8. NAPS Appendix R Report, Section 4.4 Attachment to Table 4-1

HU1.5

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the *Protected Area* or Main Dam

EAL:

HU1.5 Notification of Unusual Event

Lake level < 242 ft. **AND** actions required in TRM 3.7.4 not completed

OR

Lake level > 264 ft.

OR

Class II Dam Emergency

Mode Applicability:

All

Basis:

A lake level < 242 ft. requires the plant to be placed in Mode 5 (ref. 6, 7) and all pumps taking suction on lake sources, including Circulating Water and Fire Water pumps, be secured (ref. 4).

Lake level of 264 ft. corresponds to the Probable Maximum Flood. All station facilities are capable of withstanding the Probable Maximum Flood (ref. 3, 4). If lake level rises to greater than 271 ft., the event may be escalated to an Alert under EAL HA1.5.

When a Class II Dam Emergency exists a slowly developing condition that could develop into a more serious emergency is in progress (ref. 5).

HU1.5 (cont)

NAPS Basis Reference(s):

1. UFSAR Section 2.4.2 Floods
2. Technical Specifications Section 3.7.9
3. UFSAR Section 2.4.3 Probable Maximum Flood on Streams and Rivers
4. 0-AP-40 Abnormal Level in North Anna Reservoir (Lake)
5. 0-AP-40.2 Dam Failure Assessment and Notification
6. UFSAR Section 2.4.11 Low Water Considerations
7. Technical Requirements Manual Section 3.7.4

HA1.1

Category: H – Hazards

Sub-category: Natural & Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting a plant safe shutdown area

EAL:

HA1.1 Alert

"OBE EXCEEDED" indicator illuminated on the SYSCOM Network Control Center (NCC)

AND

Earthquake confirmed by **any** of the following:

- Earthquake felt in plant
- National Earthquake Information Center (NEIC)
- Control Room indication of degraded performance of any safety-related structure, system, or component

Mode Applicability:

All

Basis:

This EAL addresses events that may have resulted in a plant safe shutdown area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant *safety-related structures, systems and components*. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage.

The maximum ground acceleration for the Operating Basis Earthquake (OBE) for structures founded on rock is 0.06g for horizontal ground motion and two-thirds that value for vertical ground motion. As discussed in the UFSAR (Ref. 2), analyses for earthquake motion for structures founded on rock were made using response spectra that were based on past earthquakes normalized to 0.06g for the horizontal direction and 0.04g for the vertical direction for the OBE.

HA1.1 (cont)

The maximum ground acceleration for the OBE for structures founded on soil is 0.09g for horizontal ground motion and two-thirds that value for vertical ground motion. The analyses for earthquake motion for structures founded on soil were made using response spectra that were based on past earthquakes normalized to 0.09g for the horizontal direction and 0.06g for the vertical direction for the OBE.

An earthquake that generates motion that exceeds the OBE will be a “felt” earthquake and will illuminate the “OBE EXCEEDED” indicator on the SYSCOM NCC. Annuciator B-4, EARTHQUAKE SYSTEM TRIGGERED (Unit 1 Annunicator Panel A) will be activated under this circumstance. Operator action in 0-AP-36, Seismic Event, specifies shutdown of both units if an earthquake that exceeds OBE occurs.

General Design Criterion 2 of Appendix A to 10 CFR 50 requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform their *safety functions*. Part V(a)(2) of Appendix A to Part 100, Seismic and Geologic Siting Criteria for Nuclear Power Plants, requires that if vibratory ground motion exceeding that of the OBE occurs, shutdown of the nuclear power plant is required.

Should seismic instrumentation become inoperable, 0-AP-36, Seismic Event, contains compensatory measures.

NAPS Basis Reference(s):

1. 0-AP-36 Seismic Event
2. UFSAR Section 2.5.2.6
3. 10 CFR 50, Appendix A, General Design Criterion 2
4. DC NA-11-01213

HA1.2

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting a plant safe shutdown area |

EAL:

HA1.2 Alert
Tornado or high winds > 80 mph resulting in **EITHER**:

- *Visible damage* to any *safety-related structure, system, or component* within **any** Table H-1 Area
  OR
- Control Room indication of degraded performance of any *safety-related structure, system, or component*

Table H-1 Safe Shutdown Areas
• Cable Vaults & Tunnels • Emergency Switchgear Rooms • Emergency Diesel Generators Rooms • Reactor Containment • Quench Spray Pump Houses • Safeguards Areas • Main Steam Valve House • Cable Spreading Rooms • Control Room • CR Chiller Rooms • Auxiliary / Fuel / Decontamination Buildings • Fuel Oil Pump House Room A or B • Service Water Pump and Valve House • Intake Structure Control House • Auxiliary Service Water Pump House • Turbine Building • Auxiliary Feedwater Pump House

HA1.2 (cont)

Mode Applicability:

All

Basis:

Safety-related structures, systems and components (as defined in 10CFR50.2) are those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary
(2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition; or
(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

This threshold addresses events that may have resulted in a safe shutdown area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant *safety-related structures, systems and components*. Table H-1 safe shutdown areas house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in its lowest energy state (ref. 1). Personnel access to safe shutdown areas may be an important factor in monitoring and controlling equipment operability. Safe shutdown areas include structures that are in contact with or immediately adjacent to the areas that actually contains the equipment of concern. The Alert classification is appropriate if relevant plant parameters indicate that the performance of *safety-related structures, systems and components* in the affected safe shutdown areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety-related structure, system and component performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the SEM with the resources needed to perform detailed damage assessments.

HA1.2 (cont)

This EAL is based on the design basis wind speed of 80 mph (ref. 2). Wind speed recorders measure wind speeds up to 100 mph (ref. 3). All station structures are designed, however, to withstand a basic wind speed of 80 mph. *Sustained* wind loads above this magnitude can cause damage to *safety functions*. Wind speed recorders and plant computer points indicate wind speeds up to 100 mph (ref. 3). All station structures are designed, however, to withstand a basic wind speed of 80 mph.

It is recognized that the wind speed instruments are located outside the Protected Area. For the purpose of this EAL, these wind speed recordings are assumed to be representative of wind speeds within the Protected Area boundary.

NAPS Basis Reference(s):

1. NAPS Appendix R Report, Section 4.4 Attachment to Table 4-1
2. UFSAR Section 3.3.1
3. NCRODP-68-NA Meteorological Monitoring System
4. 0-AP-41 Severe Weather Conditions
5. Dwg. 11715-FC-48A Site Plan Vehicle Barrier System

HA1.3

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting a plant safe shutdown area

EAL:

HA1.3 Alert

Turbine failure-generated *missiles* resulting in **EITHER**:

Any *visible damage* to **any** *safety-related structure, system, or component* within **any** Table H-1 area

OR

Control Room indications of degraded performance of those safety systems resulting from turbine failure-generated missiles

Table H-1 Safe Shutdown Areas
• Cable Vaults & Tunnels • Emergency Switchgear Rooms • Emergency Diesel Generators Rooms • Reactor Containment • Quench Spray Pump Houses • Safeguards Areas • Main Steam Valve House • Cable Spreading Rooms • Control Room • CR Chiller Rooms • Auxiliary / Fuel / Decontamination Buildings • Fuel Oil Pump House Room A or B • Service Water Pump and Valve House • Intake Structure Control House • Auxiliary Service Water Pump House • Turbine Building • Auxiliary Feedwater Pump House

HA1.3 (cont)

Mode Applicability:

All

Basis:

Safety-related structures, systems and components (as defined in 10CFR50.2) are those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary
(2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition; or
(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

This EAL is intended to address the threat to safety-related equipment imposed by *missiles* generated by main turbine rotating component *failures*. For example, a turbine *missile* can be caused by brittle fracture of a rotating turbine part at or near turbine operating speed, or by ductile fracture upon runaway after extensive, highly improbable, control system *failures*. In the event of *missile* ejection, the probability of a strike on a plant region is a function of the energy and direction of an ejected *missile* and of the orientation of the turbine with respect to the plant region. The turbine spins about the axis depicted in Figure H-1 and Figure H-2 (ref. 1).

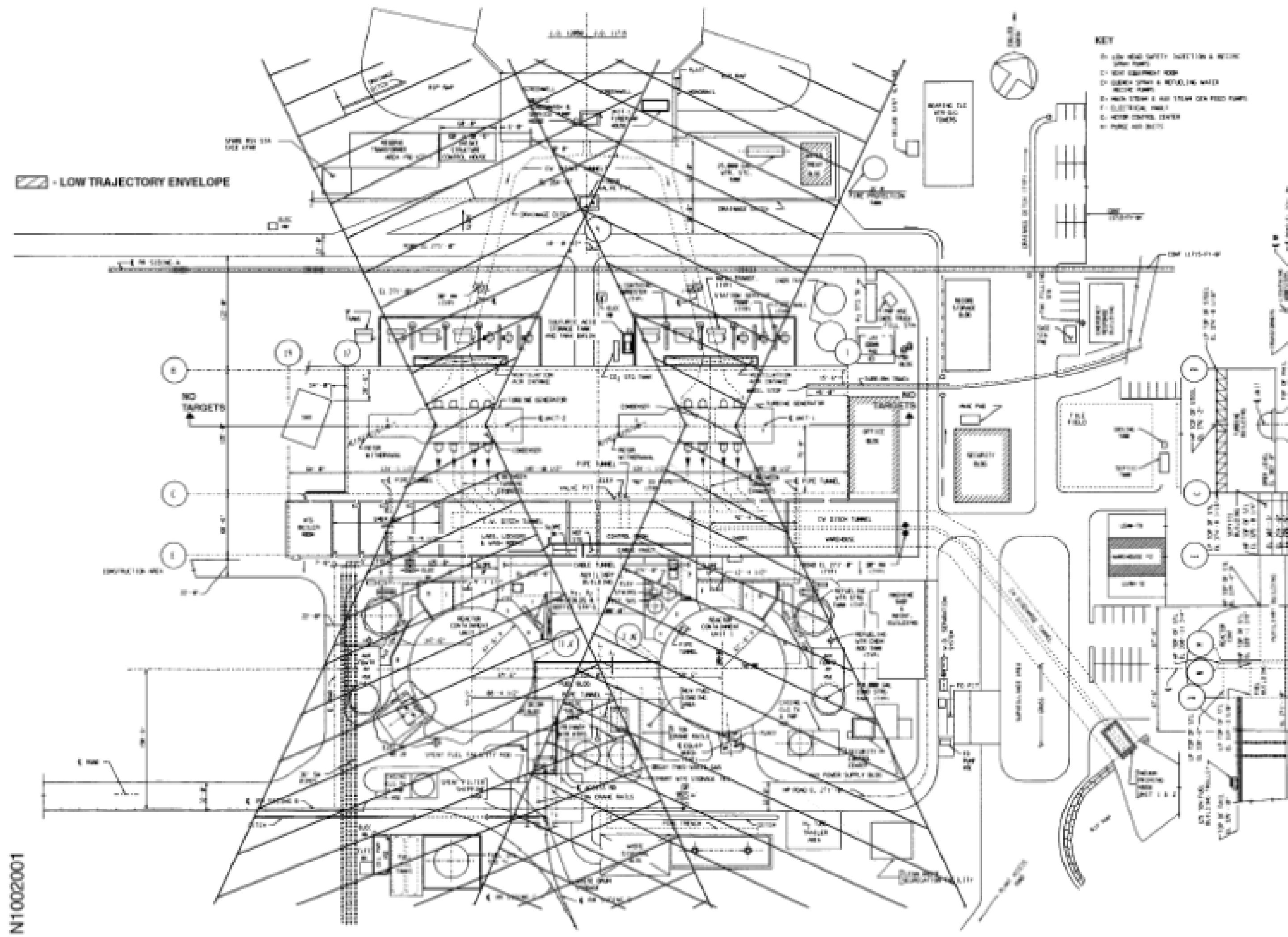
The list of Table H-1 areas includes all areas containing safety-related equipment, their controls, and their power supplies (ref. 2). This EAL is, therefore, consistent with the definition of an ALERT in that if *missiles* have damaged or penetrated areas containing safety-related equipment, the potential exists for substantial degradation of the level of safety of the plant.

NAPS Basis Reference(s):

1. UFSAR Section 10.2
2. NAPS Appendix R Report, Section 4.4 Attachment to Table 4-1

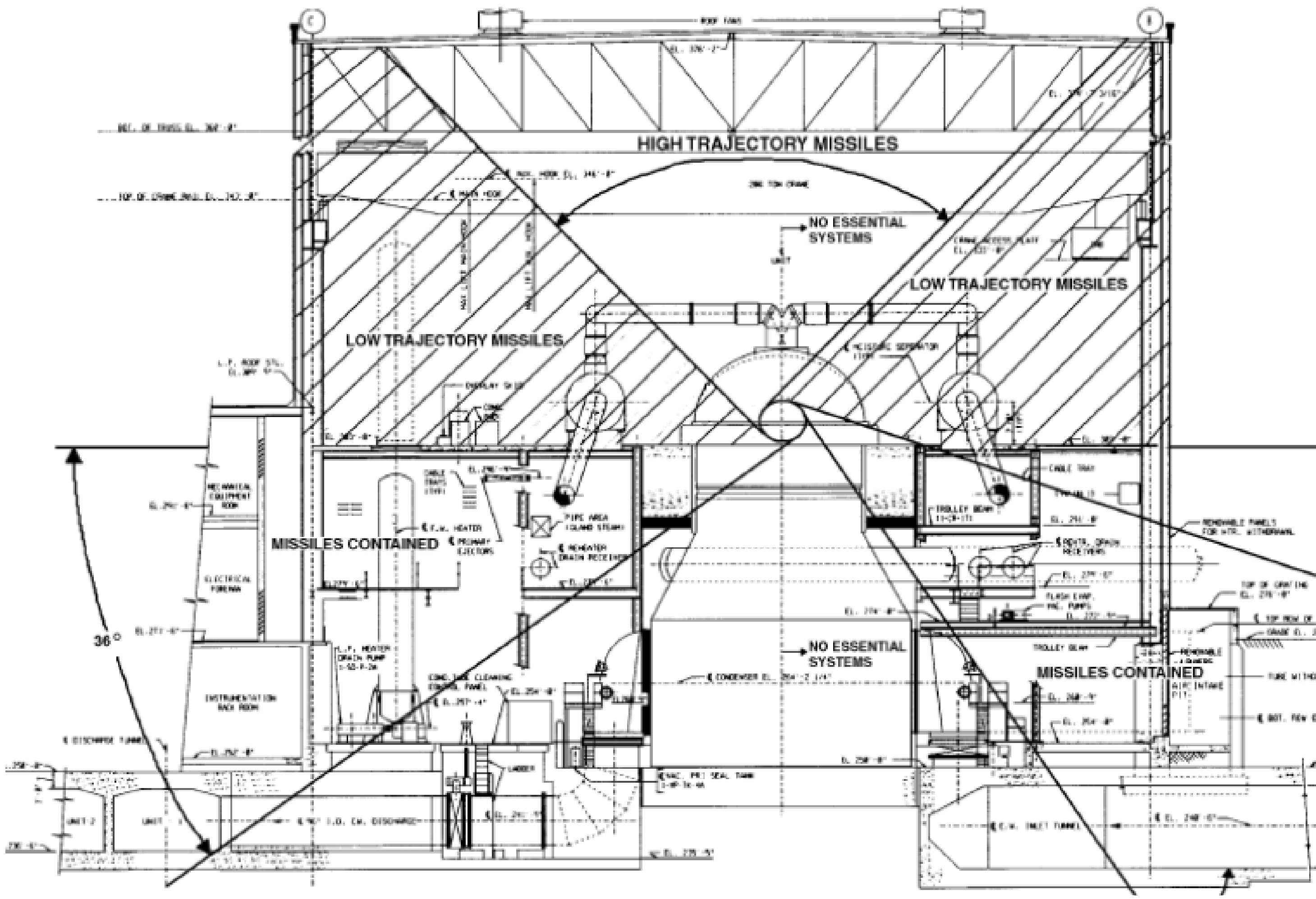
HA1.3 (cont)

Figure H-1: Turbine Axis and Postulated Missile Trajectories



HA1.3 (cont)

Figure H-2: Turbine Axis and Postulated Missile Trajectories (cont'd)



HA1.4

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting a plant safe shutdown area

EAL:

HA1.4 Alert

Uncontrolled flooding resulting in **EITHER**:

Control Room indications of degraded performance of safety-related structure, system or component within **any** Table H-1 area

OR

Creating an industrial safety hazard (e.g. electric shock) in **any** Table H-1 area that precludes access necessary to operate or monitor any safety-related structure, system or component

Table H-1 Safe Shutdown Areas
• Cable Vaults & Tunnels • Emergency Switchgear Rooms • Emergency Diesel Generators Rooms • Reactor Containment • Quench Spray Pump Houses • Safeguards Areas • Main Steam Valve House • Cable Spreading Rooms • Control Room • CR Chiller Rooms • Auxiliary / Fuel / Decontamination Buildings • Fuel Oil Pump House Room A or B • Service Water Pump and Valve House • Intake Structure Control House • Auxiliary Service Water Pump House • Turbine Building • Auxiliary Feedwater Pump House

HA1.4 (cont)

Mode Applicability:

All

Basis:

This EAL addresses flooding caused by internal events (e.g., component *failures*, circulating water, component cooling or service water line ruptures, equipment misalignment, fire suppression system actuation, outage activity mishaps, etc.) that results in degraded safety-related structure, system and component performance. The internal flooding areas are important drainage areas and typically contain systems that are:

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

Internal flooding in the Turbine building is alarmed by the following annunciators:

- Annunciator "D" Panel F-8, COND TUBE CLEAN PIT HI LEVEL
- Annunciator "D" Panel G-8, COND TUBE CLEAN PIT HI-HI LEVEL
- Annunciator "D" Panel G-7, TURB BLDG FLOOD ALARM TROUBLE

Turbine Building flooding could carry over into the Auxiliary Building through the Service Water Tunnel from the Valve Pit. Conversely, Auxiliary Building flooding could carry over into the Turbine Building through the Service Water Tunnel into the Turbine Building Valve Pit. Auxiliary Building Flooding could cause a loss of Charging Pumps (ref. 1, 2).

Uncontrolled internal flooding that has degraded safety-related equipment in Table H-1 (ref. 3) safe shutdown areas or created a safety hazard precluding access necessary for the safe operation or monitoring of safety equipment warrants declaration of an Alert.

HA1.4 (cont)

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

NAPS Basis Reference(s):

1. 0-AP-39.1 Turbine Building Flooding
2. 0-AP-39.2 Auxiliary Building Flooding
3. NAPS Appendix R Report, Section 4.4 Attachment to Table 4-1
4. NAPS IPE Section 3.3.7.1, Screening of Flood Areas
5. UFSAR Section 9.5.1
6. UFSAR Section 10.4
7. UFSAR Section 16.2
8. CM-AA-FPA-100, Fire Protection/Appendix R (Fire Safe Shutdown) Program

HA1.5

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the *Protected Area*

EAL:

HA1.5	**Alert**

Service Water Reservoir (Service Water Pump House) level < 309 ft.

OR

Lake level > 271 ft. (station finished ground grade)

OR

Class I Dam Emergency

Mode Applicability:

All

Basis:

This EAL addresses events that may have resulted in a plant safe shutdown area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant *safety-related structures, systems and components*.

A Service Water Reservoir level of 309 ft. is the minimum required to ensure required suction head for the Service Water Pumps (ref. 1). The Service Water Reservoir serves as the NAPS Ultimate Heat Sink (ref. 2).

Lake level of 271 ft. corresponds to plant grade (ref. 3). This elevation is 6.8 ft. above the probable maximum flood of 264.2 ft. External flooding above plant grade poses a direct threat to plant safe shutdown areas.

HA1.5 (cont)

Actual dam failure or a Class I Dam Emergency per 0-AP-40.2 represents a situation where no time is available to attempt corrective measures that will prevent failure and thus poses a direct threat to surrounding downstream areas (ref. 5).

NAPS Basis Reference(s):

1. UFSAR Section 9.2.1.2.4 Service Water Pump House Design
2. UFSAR Section 9.2.5 Ultimate Heat Sink
3. UFSAR Section 3.4 Water Level (Flood) Design Criteria
4. 0-AP-40 Abnormal Level in North Anna Reservoir (Lake)
5. 0-AP-40.2 Dam Failure Assessment and Notification
6. UFSAR Section 9.2.1.2.2 Service Water Reservoir Design

HA1.6

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting a plant safe shutdown area

EAL:

HA1.6 Alert

Vehicle crash resulting in **EITHER**:

- *Visible damage* to **any** *safety-related structure, system, or component* within **any** Table H-1 area

  OR
- Control Room indication of degraded performance of any *safety-related structure, system, or component*

Table H-1 Safe Shutdown Areas
• Cable Vaults & Tunnels • Emergency Switchgear Rooms • Emergency Diesel Generators Rooms • Reactor Containment • Quench Spray Pump Houses • Safeguards Areas • Main Steam Valve House • Cable Spreading Rooms • Control Room • CR Chiller Rooms • Auxiliary / Fuel / Decontamination Buildings • Fuel Oil Pump House Room A or B • Service Water Pump and Valve House • Intake Structure Control House • Auxiliary Service Water Pump House • Turbine Building • Auxiliary Feedwater Pump House

HA1.6 (cont)

Mode Applicability:

All

Basis:

Safety-related structures, systems and components (as defined in 10CFR50.2) are those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary
(2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition; or
(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Table H-1 safe shutdown areas house equipment the operation of which may be needed to ensure the reactor reaches and is maintained in its lowest energy state (ref. 1). Personnel access to safe shutdown areas may be an important factor in monitoring and controlling equipment operability. This EAL addresses vehicle crashes that preclude personnel access to safe shutdown areas or may have resulted in the area being subjected to forces beyond design limits. It is therefore assumed that equipment operability has been challenged or damage has occurred to plant systems necessary for safe shutdown of the plant. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage.

HA1.6 (cont)

This EAL addresses events such as plane, helicopter, barge, crane, car or truck crashes, or impact of projectiles into a safe shutdown area.

The *Protected Area* is within the security isolation zone and is given in Dwg. 11715-FC-48A, Site Plan Vehicle Barrier System.

If the vehicle crash is determined to be hostile in nature, the event is classified under EAL HA4.1.

NAPS Basis Reference(s):

1. NAPS Appendix R Report, Section 4.4 Attachment to Table 4-1
2. Dwg. 11715-FC-48A Site Plan Vehicle Barrier System

HU2.1

Category: H – Hazards

Sub-category: 2 – Fire or Explosion

Initiating Condition: *Fire* or *Explosion* within the *Protected Area* boundary.

EAL:

HU2.1 Notification of Unusual Event

Fire in or restricting access to **any** Table H-1 area **not** extinguished within 15 minutes of Control Room notification or verification of a Control Room alarm (Note 3)

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table H-1 Safe Shutdown Areas
• Cable Vaults & Tunnels • Emergency Switchgear Rooms • Emergency Diesel Generators Rooms • Reactor Containment • Quench Spray Pump Houses • Safeguards Areas • Main Steam Valve House • Cable Spreading Rooms • Control Room • CR Chiller Rooms • Auxiliary / Fuel / Decontamination Buildings • Fuel Oil Pump House Room A or B • Service Water Pump and Valve House • Intake Structure Control House • Auxiliary Service Water Pump House • Turbine Building • Auxiliary Feedwater Pump House

HU2.1 (cont)

Mode Applicability:

All

Basis:

The purpose of this EAL is to address the magnitude and extent of *fires* that may be potentially significant precursors to damage to *safety-related structures, systems and components*. As used here, a *confirmed fire* is a *fire* that has been identified through visual observation and report by plant personnel or sensor alarm indication. The 15-minute period begins when a credible report is received that a *fire* is occurring or a *valid* fire detection system alarm is received. Validation of a fire detection system alarm includes actions that can be taken within the Control Room or other nearby location to ensure that the alarm is not spurious. A validated alarm is assumed to be an indication of a *fire* unless personnel dispatched to the scene disprove the alarm within the 15-minute period. In other words, a personnel report from the scene may be used to disprove a validated alarm if the report is received within 15 minutes of the alarm. The report, however, shall not be required to validate the alarm or used to justify the start time for the 15-minute period.

The intent of the 15-minute period is to size the *fire* and discriminate against small *fires* that are readily extinguished (e.g., smoldering waste paper basket). The area lists are limited and apply to buildings and areas in actual contact with or immediately adjacent to safe shutdown areas or other significant buildings or areas. The intent of this EAL is not to include buildings or areas that are not immediately adjacent to Table H-1 safe shutdown areas (ref. 1). This excludes *fires* within free-standing support buildings, waste paper basket *fires* and other small *fires* of no safety consequence.

EAL HA2.1 provides escalation to the Alert classification.

NAPS Basis Reference(s):

1. NAPS Appendix R Report, Section 4.4 Attachment to Table 4-1

HU2.2

Category: H – Hazards

Sub-category: 2 – Fire or Explosion

Initiating Condition: *Fire* or *Explosion* within the *Protected Area* boundary.

EAL:

HU2.2	**Notification of Unusual Event**
Report by plant personnel of an unanticipated *explosion* within *Protected Area* boundary, SWPH, SWVH or Auxiliary SWPH resulting in *visible damage* to permanent structure or equipment	

Mode Applicability:

All

Basis:

The *Protected Area* is within the security isolation zone and is given in Dwg. 11715-FC-48A, Site Plan Vehicle Barrier System.

For this EAL, only those unanticipated *explosions* within the *Protected Area, SWPH, SWVH or Auxiliary SWPH* should be considered. As used here, an *explosion* is a rapid, violent, unconfined combustion, or catastrophic *failure* of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the *explosion* with reports of evidence of damage (e.g., deformation, scorching, etc.) is sufficient for declaration. The SEM also needs to consider any security aspects of the *explosion*.

Escalation of this emergency classification level, if appropriate, would be based on HA2.1.

HU2.2 (cont)

A steam line break or steam *explosion* that damages surrounding permanent structures or equipment would be classified under this EAL. This does not mean the emergency is classified simply because the steam line break occurred. The method of damage is not as important as the degradation of plant structures or equipment. The need to classify the steam line break itself is considered in fission product barrier degradation monitoring (EAL Category F).

If the *explosion* is determined to be hostile in nature, the event is classified under the Security EALs. On October 3, 2002, Susquehanna Unit 2 was in startup with the reactor critical at approximately two percent power, when the startup transformer (T-20) failed (*explosion* and *fire*). The recirculation pumps tripped due to a power loss caused by the transformer *failure*. The Unit 2 reactor was manually shutdown. The initial assessment of plant conditions was not very extensive. Several plant personnel heard two *explosions* and witnessed a *fire* at T-20. Personnel at the scene attempted to report the information to the main control room, but were unsuccessful because the telephone they tried to use had lost power due to the event. A maintenance technician announced the *fire* via the plant public address system, which was overheard by control room personnel. Although the control room operators immediately responded to the overheard *fire* announcement, they did not attempt to obtain more details of the T-20 *failure* by talking with the technician or any other personnel who had been near the transformer. The transformer *fire* was evaluated against the criteria in emergency action level [SSES EAL] 14.1, "Fire or Explosion," which stated, in part, that the criteria for classification was "a *fire* lasting more than 15 minutes OR an *explosion* inside the security *protected area* with no significant damage to station facilities." Since the control room was unaware that an *explosion* had also occurred, it was concluded no emergency condition existed because the *fire* lasted for 10 minutes only. Although information related to the transformer *failure* (*explosion* and *fire*) was available from security and other plant personnel, it was approximately 34 minutes before the Shift Manager became aware that an *explosion* which required an Notification of Unusual Event classification had occurred (ref. 2).

HU2.2 (cont)

NAPS Basis Reference(s):

1. Dwg. 11715-FC-48A, Site Plan Vehicle Barrier System
2. Susquehanna Steam Electric Station - NRC Integrated Inspection Report 50-387, -388/02-06 dated January 24, 2003, Green Non-Cited Violation for late classification

HA2.1

Category: H – Hazards

Sub-category: 2 – Fire or Explosion

Initiating Condition: *Fire* or *explosion* affecting the operability of plant *safety-related structures, systems or components* required to establish or *maintain* safe shutdown

EAL:

HA2.1 **Alert**

Fire or *explosion* in **any** Table H-1 area

AND EITHER:

- Plant personnel report *visible damage* to **any** *safety-related structure, system, or component* within the area

  OR
- Affected system parameter indications show degraded performance

Table H-1 Safe Shutdown Areas
• Cable Vaults & Tunnels • Emergency Switchgear Rooms • Emergency Diesel Generators Rooms • Reactor Containment • Quench Spray Pump Houses • Safeguards Areas • Main Steam Valve House • Cable Spreading Rooms • Control Room • CR Chiller Rooms • Auxiliary / Fuel / Decontamination Buildings • Fuel Oil Pump House Room A or B • Service Water Pump and Valve House • Intake Structure Control House • Auxiliary Service Water Pump House • Turbine Building • Auxiliary Feedwater Pump House

HA2.1 (cont)

Mode Applicability:

All

Basis:

Safety-related structures, systems and components (as defined in 10CFR50.2) are those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary
(2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition; or
(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

The listed areas contain functions and systems required for the safe shutdown of the plant. The NAPS Appendix R report was consulted for equipment and plant areas required for the applicable mode (ref. 1).

The only *explosion*s that should be considered are those of sufficient force to: damage permanent structures or equipment required for safe operation, or result in degraded performance of *safety-related structures, systems and components* within the identified plant areas. An *explosion* is a rapid, violent, unconfined combustion, or catastrophic *failure* of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components. No attempt is made to assess the actual magnitude of the damage. The wording of this EAL does not imply that an assessment of safety-related structure, system and component performance should be performed; rather that safety-related structure, system and component parameter symptoms are degraded as a result of the event. The declaration of an Alert and the activation of the TSC provide the SEM with the resources needed to perform damage assessments. The SEM also needs to consider the security aspects of the *explosions*.

HA2.1 (cont)

A steam line break or steam *explosion* that damages permanent structures or equipment would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

NAPS Basis Reference(s):

1. Dwg. 11715-FC-48A, Site Plan Vehicle Barrier System
2. NAPS Appendix R Report, Section 4.4 Attachment to Table 4-1

HU3.1

Category: H – Hazards

Sub-category: 3 – Toxic, Corrosive, Asphyxiant & Flammable Gas

Initiating Condition: Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal operation of the plant

EAL:

HU3.1 Notification of Unusual Event
Report or detection of toxic, corrosive, asphyxiant or flammable gases that have or could enter the Owner Controlled Area in amounts that can affect *normal plant operations*

Mode Applicability:

All

Basis:

Entry into 0-AP-23 Oil and Hazardous Substances Spill Response does not, in and of itself, constitute having affected normal plant operations, as defined.

This EAL is based on the existence of *uncontrolled* releases of toxic, corrosive, asphyxiant or flammable gas affecting *normal plant operations* or the health of plant personnel. The release may have originated within the *Owner Controlled Area*, or it may have originated offsite and subsequently drifted inside the *Owner Controlled Area*. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) resulting in the plant being within the evacuation area should also be considered in this EAL because of the adverse effect on *normal plant operations*.

HU3.1 (cont)

It is intended that releases of toxic, corrosive, asphyxiant or flammable gases are of sufficient quantity and the release point of such gases is such that *normal plant operations* would be affected. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation. Some gases are toxic by their very nature. Others, like carbon dioxide, can be lethal if they reduce oxygen to low concentrations (asphyxiant) that are *immediately dangerous to life and health* (IDLH).

NRC position is that anytime carbon dioxide is discharged in plant areas such that the area becomes uninhabitable, regardless of whether anyone is in the areas, conditions for classification exist. The EAL is not intended to require significant assessment or quantification. The EAL assumes an *uncontrolled* process that has the potential to affect plant operations or personnel safety. Releases occurring during planned surveillance activities or planned maintenance/tag-out activities, therefore, are excluded. Similarly, a carbon dioxide release prompted by a *valid* indication (e.g., smoke detector) or intentional actuation of a fire extinguisher is considered a planned release.

0-AP-23, Oil or Hazardous Substance Spill Response, and Regulatory Guide 1.78, Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, Table 1, Toxicity Limits (IDLH Limits) for Some Hazardous Chemicals, provide additional information on hazardous substances and spills.

On 6/27/2003, an *unplanned* release of carbon dioxide in the Cable Spreading Room resulted in oxygen level within the lower level of the Service Building stairwell dropping below that necessary for habitability. No Emergency Plan classification was declared. North Anna received a Green non-cited violation of 10 CFR 50.54(q), 50.47(b)(4), and Section IV.B of Appendix E of 10 CFR 50, for failure to classify (ref. 1). NRC stated that their position on the declaration of an Unusual Event was that anytime carbon dioxide was discharged in plant areas such that the area became uninhabitable, regardless of whether anyone was in the areas, the Unusual Event should be declared (ref. 2).

HU3.1 (cont)

On 5/14/2000, Ionics personnel were overcome by gasses resulting from mixing two separate chemicals together which produced chlorine gas (ref. 3). No Emergency Plan classification was declared in this instance even though chlorine gas is toxic and safety of personnel within the Owner Controlled Area was affected (ref. 4).

Examples of toxic, asphyxiant or flammable gas releases classified at other stations include:

- On November 10, 2003, a 1/8 inch sealant injection plug for the Main Generator at Seabrook was discovered out of its location. This allowed the leakage of hydrogen gas into the Turbine Building. The fire brigade leader responded and observed a flammable concentration of hydrogen.
- On April 2, 2002, a hose reel on the turbine building fire suppression CO_2 system at Nine Mile Point was broken during work activities. The turbine building, reactor building, radioactive waste building, and control building were evacuated. The control room was not evacuated.
- On March 4, 2002, a non-isolable leak on a propane tank occurred at Point Beach. The tank was approximately 500 to 750 gallons and was 55% full. The tank was leaking at the rate of 5% decrease every 20 minutes. The tank was 25 to 30 yards out in the Owner Controlled Area and a local area evacuation had been performed. The local fire department was on the scene.
- On June 18, 2001, during chemical unloading of sulfuric acid from a truck trailer at Salem, excessive acid fumes came from the tank, which was located in the turbine building. The chemistry technician supporting the delivery experienced throat and respiratory irritation and was transported to the hospital for examination.
- On December 21, 2000, an inadvertent (personnel error) discharge in a localized area (switchgear room) of the reactor building occurred at Nine Mile Point.

HU3.1 (cont)

- On October 9, 2000, *failure* of a chemical transfer pump at Salem resulted in leakage of 25 to 50 gallons of hydrazine from a chemical addition tank. The odor of ammonia was detected in Unit 1 Turbine Building. The area was evacuated; no personnel were working in the area. No safeguards activated or were needed. All safeguards equipment was available. Hydrazine was contained in a dike and directed to chemical waste.

Should the release affect safe operations or the ability to obtain or *maintain* safe shutdown, escalation to an Alert would be based on EAL HA3.1 or HA3.2. Should an *explosion* or *fire* occur due to flammable gas within an affected plant area, an Alert may be appropriate based on EAL HA2.1.

Steam is not considered a gas applicable under this EAL. Depending on the event, potential steam impacts might be considered under explosions or judgment EALs.

NAPS Basis Reference(s):

1. NRC Integrated Inspection Report No. 50-338, 339/2003-04, dated 10/27/2003
2. Plant Issue N-2000-1377
3. Plant Issue Resolution N-2003-2539-R10
4. 0-AP-23, Oil or Hazardous Substance Spill Response
5. Regulatory Guide 1.78, Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, Table 1, Toxicity Limits (IDLH Limits) for Some Hazardous Chemicals
6. NRC EAL FAQ 2006-023

HU3.2

Category: H – Hazards

Sub-category: 3 – Toxic, Corrosive, Asphyxiant & Flammable Gas

Initiating Condition: Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal operation of the plant

EAL:

HU3.2	**Notification of Unusual Event**
Recommendation by local, county or state officials to evacuate or shelter site personnel based on offsite event involving toxic, corrosive, asphyxiant or flammable gas	

Mode Applicability:

All

Basis:

This EAL is based on the existence of *uncontrolled* releases of toxic, corrosive, asphyxiants or flammable gas affecting *normal plant operations* or the health of plant personnel. The release originated offsite and local, county or state officials have reported the need for evacuation or sheltering of site personnel. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) are considered in this EAL because they may adversely affect *normal plant operations*.

The EAL is not intended to require significant assessment or quantification. The EAL assumes an *uncontrolled* process that has the potential to affect plant operations or personnel safety

State officials may determine the evacuation area for offsite spills by using the Department of Transportation (DOT) Evacuation Tables for Selected Hazardous Materials in the DOT Emergency Response Guide for Hazardous Materials.

HU3.2 (cont)

Should the release affect safe operations or the ability to obtain or *maintain* safe shutdown, escalation to an Alert would be based on EAL HA3.1 or HA3.2. Should an *explosion* or *fire* occur due to flammable gas within an affected plant area, an Alert may be appropriate based on EAL HA2.1.

NAPS Basis Reference(s):

None

HA3.1

Category: H – Hazards

Sub-category: 3 – Toxic, Corrosive, Asphyxiant & Flammable Gas

Initiating Condition: Access to a safe shutdown area is prohibited due to release of toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor

EAL:

HA3.1 Alert

Access to a Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor (Note 9)

Note 9: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

Table H-1 Safe Shutdown Areas
• Cable Vaults & Tunnels • Emergency Switchgear Rooms • Emergency Diesel Generators Rooms • Reactor Containment • Quench Spray Pump Houses • Safeguards Areas • Main Steam Valve House • Cable Spreading Rooms • Control Room • CR Chiller Rooms • Auxiliary / Fuel / Decontamination Buildings • Fuel Oil Pump House Room A or B • Service Water Pump and Valve House • Intake Structure Control House • Auxiliary Service Water Pump House • Turbine Building • Auxiliary Feedwater Pump House

HA3.1 (cont)

Mode Applicability:

All

Basis:

This EAL is based on gases that have entered a plant structure in concentrations that are unsafe for plant personnel and, therefore, preclude access to equipment necessary for the safe operation of the plant.

Gases in a safe shutdown area can affect the ability to safely operate or safely shutdown the reactor.

The fact that SCBA may be worn does not eliminate the need to declare the event.

Declaration should not be delayed for confirmation from atmospheric testing if it is reasonable to conclude that the atmosphere poses an immediate threat to life and health or an immediate threat of severe exposure to gases. This could be based upon documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

HA3.1 (cont)

An uncontrolled release of flammable gases within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Flammable gases, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This EAL assumes concentrations of flammable gases which can ignite/support combustion.

Examples of toxic, asphyxiant or flammable gas releases classified at other stations include:

- On June 2, 2002, a CO_2 system injection occurred during diesel testing at Peach Bottom. Levels of CO_2 *immediately dangerous to life and health (IDLH)* were detected within a *vital area*.
- On August 8, 2007, Peach Bottom declared an Alert because a portable, wall mounted CO2 fire extinguisher in an Emergency Diesel Generator (EDG) room had rapidly discharged its contents. The Shift Manager declared the Alert for an IDLH atmosphere in a vital area. Site engineering analysis of this event indicated that the resulting CO2 concentration from the extinguisher discharge was well below the IDLH limit. An assessment of this event as applied to NAPS indicated that NAPS does not have any extinguishers in areas small enough for IDLH concentrations to occur (ref. 3).

Steam is not considered a gas applicable under this EAL. Depending on the event, potential steam impacts might be considered under explosions or judgment EALs.

NAPS Basis Reference(s):

1. NAPS Appendix R Report, Section 4.4 Attachment to Table 4-1
2. NRC EAL FAQ 2006-024
3. INPO OE25324 (North Anna OEE000249)

HU4.1

Category: **H – Hazards**

Sub-category: 4 – Security

Initiating Condition: Confirmed security condition or threat which indicates a potential degradation in the level of safety of the plant

EAL:

HU4.1 Notification of Unusual Event

A security condition that does **not** involve a hostile action as reported by the Security Shift Supervisor

OR

A credible site-specific security threat notification

OR

A validated notification from NRC providing information of an aircraft threat

Mode Applicability:

All

Basis:

Security conditions which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as hostile actions are classifiable under HA4.1, HS4.1 and HG4.1.

A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. Consideration should be given to upgrading the emergency response status and emergency classification level in accordance with Dominion’s “Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program”.

HU4.1 (cont)

<u>1st Threshold</u>

Reference is made to the Security Shift Supervisor because these individuals are the designated personnel on-site qualified and trained to confirm that a security condition is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Safeguards Contingency Plan.

This threshold is based on the Dominion's "Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program". Safeguards Contingency Plans are based on guidance provided by NEI 03-12.

<u>2nd Threshold</u>

The second threshold is included to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the Notification of Unusual Event.

The determination of "credible" is made through use of information found in Dominion's "Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program".

<u>3rd Threshold</u>

The intent of this EAL threshold is to ensure that notifications for the aircraft threat are made in a timely manner and that off-site response organizations and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

HU4.1 (cont)

This EAL is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Notification of Unusual Event.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Escalation to Alert emergency classification level via HA4.1 would be appropriate if the threat involves an airliner less than 30 minutes from of the plant.

NAPS Basis Reference(s):

1. NRC Safeguards Advisory 10/6/01
2. Dominion's "Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program"
3. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
4. NRC Bulletin 2005-02 "Emergency Preparedness and Response Actions for Security-Based Events"
5. Enhancements to Emergency Preparedness Programs for Hostile Action May 2005 (Revised November 18, 2005)
6. NRC Regulatory Issue Summary 2006-12, Endorsement of the Nuclear Energy Institute Guidance "Enhancements to Emergency Preparedness Programs for Hostile Action," dated 7/19/06"

HA4.1

Category: H – Hazards

Sub-category: 4 – Security

Initiating Condition: Hostile action within the Owner Controlled Area or airborne attack threat.

EAL:

HA4.1	**Alert**

A hostile action is occurring or has occurred within the Owner Controlled Area as reported by the Security Shift Supervisor

OR

A validated notification from NRC of an airliner attack threat < 30 min. away

Mode Applicability:

All

Basis:

These EAL thresholds address the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering).

HA4.1 (cont)

1st Threshold

This EAL threshold addresses the potential for a very rapid progression of events due to a hostile action. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the Owner Controlled Area. Those events are adequately addressed by other EALs.

Although nuclear plant security officers are well trained and prepared to protect against hostile action, it is appropriate for OROs (off-site response organizations) to be notified and encouraged to begin activation (if they do not normally) to be better prepared should it be necessary to consider further actions.

2nd Threshold

This EAL threshold addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time.

The intent of this EAL threshold is to ensure that notifications for the airliner attack threat are made in a timely manner and that off-site response organizations and plant personnel are at a state of heightened awareness regarding the credible threat. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.

This EAL threshold is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Alert.

HA4.1 (cont)

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

If not previously notified by the NRC that the airborne hostile action was intentional, then it would be expected, although not certain, that notification by an appropriate Federal agency would follow. In this case, appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. However, the declaration should not be unduly delayed awaiting Federal notification.

NAPS Basis Reference(s):

1. NRC Safeguards Advisory 10/6/01
2. Dominion's "Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program"
3. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
4. NRC Bulletin 2005-02 "Emergency Preparedness and Response Actions for Security-Based Events"
5. Enhancements to Emergency Preparedness Programs for Hostile Action May 2005 (Revised November 18, 2005)
6. NRC Regulatory Issue Summary 2006-12, Endorsement of the Nuclear Energy Institute Guidance "Enhancements to Emergency Preparedness Programs for Hostile Action," dated 7/19/06"

HS4.1

Category: H – Hazards

Sub-category: 4 – Security

Initiating Condition: Hostile action within the Protected Area

EAL:

HS4.1	**Site Area Emergency**
A hostile action is occurring or has occurred within the Protected Area as reported by the Security Shift Supervisor	

Mode Applicability:

All

Basis:

This condition represents an escalated threat to plant safety above that contained in the Alert in that a hostile force has progressed from the Owner Controlled Area to the Protected Area.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires off-site response organizations readiness and preparation for the implementation of protective measures.

HS4.1(cont)

This EAL addresses the potential for a very rapid progression of events due to a hostile action. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the Protected Area.

Although nuclear plant security officers are well trained and prepared to protect against hostile action, it is appropriate for off-site response organizations to be notified and encouraged to begin preparations for public protective actions (if they do not normally) to be better prepared should it be necessary to consider further actions.

If not previously notified by NRC that the airborne hostile action was intentional, then it would be expected, although not certain, that notification by an appropriate Federal agency would follow. In this case, appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. However, the declaration should not be unduly delayed awaiting Federal notification.

Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack.

NAPS Basis Reference(s):

1. NRC Safeguards Advisory 10/6/01
2. Dominion's "Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program"
3. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
4. NRC Bulletin 2005-02 "Emergency Preparedness and Response Actions for Security-Based Events"
5. Enhancements to Emergency Preparedness Programs for Hostile Action May 2005 (Revised November 18, 2005)
6. NRC Regulatory Issue Summary 2006-12, Endorsement of the Nuclear Energy Institute Guidance "Enhancements to Emergency Preparedness Programs for Hostile Action," dated 7/19/06"

HG4.1

Category: H – Hazards

Sub-category: 4 – Security

Initiating Condition: Hostile action resulting in loss of physical control of the facility

EAL:

HG4.1	**General Emergency**

A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions

<u>OR</u>

A hostile action has caused failure of Spent Fuel Cooling Systems and imminent fuel damage is likely for a freshly off-loaded reactor core in pool (i.e. freshly off-loaded reactor core means **any** spent fuel in the Spent Fuel Pit)

Mode Applicability:

All

Basis:

<u>1st Threshold</u>

This EAL threshold encompasses conditions under which a hostile action has resulted in a loss of physical control of *Vital Areas* (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location.

HG4.1 (cont)

These safety functions are reactivity control (ability to shut down the reactor and keep it shutdown) reactor water level (ability to cool the core), decay heat removal (ability to maintain a heat sink) and Spent Fuel Pit (Pool) cooling (ability to remove decay heat from irradiated fuel in storage).

Loss of physical control of the Control Room or remote shutdown capability alone may not prevent the ability to maintain safety functions per se. Design of the remote shutdown capability and the location of the transfer switches should be taken into account. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions.

If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the threshold is not met.

2nd Threshold

This EAL threshold addresses failure of spent fuel cooling systems as a result of hostile action if imminent fuel damage is likely. Dominion has decided to conservatively apply this threshold when **any** spent fuel is in the Spent Fuel Pit.

NAPS Basis Reference(s):

1. NRC Safeguards Advisory 10/6/01
2. Dominion's "Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program"
3. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated 2/4/02
4. NRC Bulletin 2005-02 "Emergency Preparedness and Response Actions for Security-Based Events"
5. Enhancements to Emergency Preparedness Programs for Hostile Action May 2005 (Revised November 18, 2005)
6. NRC Regulatory Issue Summary 2006-12, Endorsement of the Nuclear Energy Institute Guidance "Enhancements to Emergency Preparedness Programs for Hostile Action," dated 7/19/06"

HA5.1

Category: H – Hazards

Sub-category: 5 – Control Room Evacuation

Initiating Condition: Control Room evacuation has been initiated

EAL:

HA5.1	**Alert**
Control Room evacuation has been initiated	

Mode Applicability:

All

Basis:

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency operations centers are necessary. ( )-AP-20, Operation from the Auxiliary Shutdown Panel, provides specific instructions for evacuating the Control Room and establishing plant *control* at the Unit 1 and Unit 2 Auxiliary Shutdown Panels and other areas of the station (e.g., Service Water Valve House, Emergency Switchgear Rooms, Emergency Diesel Generator Rooms, etc.). The Shift Manager determines if the Control Room is inoperable and requires evacuation of both units. Control Room inhabitability may be caused by *fire*, dense smoke, noxious fumes, etc. Inability to establish plant *control* from outside the Control Room escalates this event to a Site Area Emergency.

NAPS Basis Reference(s):

1. 1-AP-20 (2-AP-20), Operation from the Auxiliary Shutdown Panel

HS5.1

Category: H – Hazards

Sub-category: 5 – Control Room Evacuation

Initiating Condition: Control Room evacuation has been initiated and plant *control* **cannot** be established

EAL:

HS5.1 Site Area Emergency

Control Room evacuation has been initiated

AND

Control of the plant **cannot** be established from the Auxiliary Shutdown Panel within 15 min. (Note 3)

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

All

Basis:

This EAL indicates that expeditious transfer of *safety-related structures, systems and components* has not occurred, but fission product barrier damage may not yet be indicated. The intent of this EAL is to capture events in which *control* of the plant cannot be reestablished in a timely manner. The 15 minute time for transfer starts when the Control Room begins to be evacuated (not when ( )-AP-20 is entered). The time interval is based on how quickly *control* must be reestablished without core uncovery and/or core damage. The determination of whether or not *control* is established from outside the Control Room is based on SEM judgment. The SEM is expected to make a reasonable, informed judgment that *control* of the plant from the Unit 1 and Unit 2 Auxiliary Shutdown Panels cannot be established within the 15 minute interval.

HS5.1 (cont)

Once the Control Room is evacuated, the objective is to establish *control* of important plant equipment and *maintain* knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on components and instruments that supply protection for and information about *safety functions*. Typically, these *safety functions* are reactivity control (ability to shutdown the reactor and *maintain* it shutdown), RCS inventory (ability to cool the core), secondary heat removal (ability to *maintain* a heat sink) and Spent Fuel Pit (Pool) cooling (ability to remove decay heat from irradiated fuel in storage). In Cold Shutdown and Refueling modes, operator concern is directed toward maintaining core cooling such as is discussed in Generic Letter 88-17, "Loss of Decay Heat Removal." In Operating, and Hot Standby modes, operator concern is primarily directed toward maintaining critical *safety functions* and thereby assuring fission product barrier integrity.

( )-AP-20, Operation from the Auxiliary Shutdown Panel, provides specific instructions for evacuating the Control Room and establishing plant *control* at the Unit 1 and Unit 2 Auxiliary Shutdown Panels and other areas of the station (e.g., Service Water Valve House, Emergency Switchgear Rooms, Emergency Diesel Generator Rooms, etc.).

NAPS Basis Reference(s):

1. 1-AP-20 (2-AP-20), Operation from the Auxiliary Shutdown Panel

HU6.1

Category: H – Hazards

Sub-category: 6 – Judgment

Initiating Condition: Other conditions existing which in the judgment of the SEM warrant declaration of a NOUE

EAL:

HU6.1 Notification of Unusual Event

Other conditions exist which in the judgment of the SEM indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of *safety-related structures, systems or components* occurs.

Mode Applicability:

All

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Station Emergency Manager (SEM) to fall under the Notification of Unusual Event emergency class. This may include a security threat to facility protection. The SEM is the designated onsite individual having the responsibility and authority for implementing the North Anna Emergency Plan. (Ref. 1)

HU6.1 (cont)

From a broad perspective, one area that may warrant SEM judgment is related to likely or actual breakdown of site-specific event mitigating actions. Examples to consider include:

- Inadequate emergency response procedures
- Transient response either unexpected or not understood
- *Failure* or unavailability of emergency systems during an accident in excess of that assumed in accident analysis
- Insufficient availability of equipment and/or support personnel

NAPS Basis Reference(s):

1. North Anna Power Station Emergency Plan
2. NRC Bulletin 2005-02 “Emergency Preparedness and Response Actions for Security-Based Events”
3. Enhancements to Emergency Preparedness Programs for Hostile Action May 2005 (Revised November 18, 2005)

HA6.1

Category: H – Hazards
Sub-category: 6 – Judgment
Initiating Condition: Other conditions existing which in the judgment of the SEM warrant declaration of an Alert

EAL:

HA6.1 Alert

Other conditions exist which in the judgment of the SEM indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of *hostile action*. **Any** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE).

Mode Applicability:

All

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Station Emergency Manager (SEM) to fall under the Alert emergency class. This may include a security event that involves probable life threatening risk to site personnel or damage to site equipment because of intentional malicious dedicated efforts of a hostile act.

HA6.1 (cont)

The SEM is the designated onsite individual having the responsibility and authority for implementing the North Anna Emergency Plan. The Shift Manager or Unit Supervisor initially acts in the capacity of the SEM and takes actions as outlined in the EPIPs. If required by the emergency classification, or if deemed appropriate by the SEM, emergency response personnel are notified and instructed to report to their emergency response locations. (Ref. 1)

NAPS Basis Reference(s):

1. North Anna Power Station Emergency Plan
2. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security-Based Events
3. Enhancements to Emergency Preparedness Programs for Hostile Action May 2005 (Revised November 18, 2005)

HS6.1

Category: H – Hazards

Sub-category: 6 – Judgment

Initiating Condition: Other conditions existing which in the judgment of the SEM warrant declaration of Site Area Emergency

EAL:

HS6.1	**Site Area Emergency**

Other conditions exist which in the judgment of the SEM indicate that events are in progress or have occurred which involve an actual or likely major *failures* of plant functions needed for protection of the public or *hostile action* that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely *failure* of or; 2) that prevent effective access to equipment needed for the protection of the public. **Any** releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the *site boundary*.

Mode Applicability:

All

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere, but that warrant declaration of an emergency because conditions exist which are believed by the SEM to fall under the emergency class description for Site Area Emergency. This may include security events that result in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely *failure* of or; (2) prevents effective access to equipment needed for the protection of the public. The SEM is the designated onsite individual having the responsibility and authority for implementing the North Anna Emergency Plan. (Ref. 1)

HS6.1 (cont)

EPA PAGs stands for Environmental Protection Agency Protective Action Guidelines.

NAPS Basis Reference(s):

1. North Anna Power Station Emergency Plan
2. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security-Based Events
3. Enhancements to Emergency Preparedness Programs for Hostile Action May 2005 (Revised November 18, 2005)

HG6.1

Category: H – Hazards

Sub-category: 6 – Judgment

Initiating Condition: Other conditions existing which in the judgment of the SEM warrant declaration of General Emergency

EAL:

HG6.1	**General Emergency**
Other conditions exist which in the judgment of the SEM indicate that events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or *hostile action* that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) offsite for more than the immediate site area.	

Mode Applicability:

All

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere, but that warrant declaration of an emergency because conditions exist which are believed by the SEM to fall under the General Emergency class. This may include security events that result in an actual loss of physical *control* of the facility. The SEM is the designated onsite individual having the responsibility and authority for implementing the North Anna Emergency Plan. The Shift Manager or Unit Supervisor initially acts in the capacity of the SEM and takes actions as outlined in the EPIPs. If required by the emergency classification, or if deemed appropriate by the SEM, emergency response personnel are notified and instructed to report to their emergency response locations. (Ref. 1)

HG6.1 (cont)

Releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the Exclusion Area Boundary. *EPA PAGs* stands for Environmental Protection Agency Protective Action Guidelines.

NAPS Basis Reference(s):

1. North Anna Power Station Emergency Plan
2. NRC Bulletin 2005-02 Emergency Preparedness and Response Actions for Security-Based Events
3. Enhancements to Emergency Preparedness Programs for Hostile Action May 2005 (Revised November 18, 2005)

<u>Category S – System Malfunction</u>

EAL Group: Hot Conditions (RCS temperature > 200ºF); EALs in this category are applicable only in one or more hot operating modes.

Numerous system-related equipment *failure* events that warrant emergency classification have been identified in this category. They may pose actual or potential threats to plant safety.

The events of this category pertain to the following subcategories:

<u>1. Loss of Power</u>

Loss of emergency plant electrical AC or DC power can compromise plant safety-related structure, system and component operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity.

<u>2. RPS *Failure*</u>

Events related to *failure* of the Reactor Protection System (RPS) to initiate and complete reactor trips. In the plant licensing basis, postulated *failures* of the RPS to complete a reactor *trip* comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification however, ATWS is intended to mean any *trip failure* event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive *control* of reactivity is at risk and could cause a threat to fuel clad, RCS and Containment integrity.

<u>3. Inability to Reach or *Maintain* Shutdown Conditions</u>

Only one EAL falls into this subcategory. It is related to the *failure* of the plant to be brought to the required plant operating condition required by Technical Specifications if a limiting condition for operation (LCO) is not met.

<u>4. Instrumentation / Communications</u>

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of annunciators are in this subcategory.

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

<u>5. Fuel Clad Degradation</u>

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant increase from these base-line levels (2% - 5% clad *failures*) is indicative of fuel *failures* and is covered under the *Fission Product Barriers* category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

<u>6. RCS Leakage</u>

The Reactor Vessel provides a volume for the coolant that covers the reactor core. The Reactor Vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail.

Excessive RCS leakage greater than Technical Specification limits are utilized to indicate potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and Containment integrity.

<u>7. Inadvertent Criticality</u>

Inadvertent criticalities pose potential personnel safety hazards as well being indicative of losses of reactivity *control*.

SU1.1

Category: S – System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: Loss of **all** offsite power to emergency busses for greater than 15 minutes

EAL:

SU1.1	**Notification of Unusual Event**
Loss of **all** offsite AC power to Unit 1 (Unit 2) 4160-Volt emergency busses H and J for > 15 min. (Note 3)	

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Prolonged loss of all offsite AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of AC power (station blackout). Unit 1 (Unit 2) 4160-Volt emergency busses H and J are the essential busses (ref. 1).

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs).

SU1.1 (cont)

The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). The switchyard voltage is reduced from 500 kV and 230 kV to 34.5 kV. The RSSTs step down the 34.5 kV power to 4160-Volt AC for

distribution as the preferred source of power to the emergency electrical distribution System.

The additional bus ties for Unit 1 between the emergency bus 1H and normal bus 1B and emergency bus 1J and normal bus 2B provide two independent offsite power sources to each emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions.

The 15-minute interval was selected as a threshold to exclude transient power losses. If neither unit 4160-Volt emergency bus is energized by an offsite source within 15 minutes, an Notification of Unusual Event is declared under this EAL.

SU1.1 (cont)

NAPS Basis Reference(s):

1. 11715-FE-1A Main One Line Diagram
2. UFSAR Section 8.3
3. 1-OP-26.1 (2-OP-26.1) Transferring 4160-Volt Busses
4. 0-AP-10 Loss of Electrical Power
5. 1-ECA-0.0 (2-ECA-0.0) Loss of All AC Power
6. NRC EAL FAQ 2006-002

SA1.1

Category: S – System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: AC power capability to emergency busses reduced to a single power source for greater than 15 minutes such that any additional single *failure* would result in loss of all AC power to emergency busses

EAL:

SA1.1	**Alert**
AC power capability to Unit 1 (Unit 2) 4160-Volt emergency busses H and J reduced to a single power source for > 15 min. (**any** additional single *failure* would result in loss of **all** AC power to the emergency busses) (Note 3)	

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

The condition indicated by this EAL is the degradation of the offsite and onsite power sources such that any additional single *failure* would result in a loss of all AC power to the unit emergency 4160-Volt emergency busses. Unit 1 (Unit 2) 4160-Volt emergency busses H and J are the essential busses (ref. 1).

The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). The switchyard voltage is reduced from 500 kV and 230 kV to 34.5 kV. The RSSTs step down the 34.5 kV power to 4160-Volt AC for distribution as the preferred source of power to the emergency electrical distribution System.

SA1.1 (cont)

The additional bus ties for Unit 1 between the emergency bus 1H and normal bus 1B and emergency bus 1J and normal bus 2B provide two independent offsite power sources to each emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions.

If the SBO diesel generator is supplying power to an emergency bus of a unit that has lost all other sources of emergency offsite AC power, the unit has not lost all 4160-Volt AC power.

Several combinations of power *failures* could therefore satisfy this EAL. The subsequent loss of the remaining power source escalates the event to a Site Area Emergency under EAL SS1.1.

Consideration should be given to operable loads necessary to remove decay heat or provide RCS makeup capability when evaluating loss of AC power to the 4160-Volt emergency busses. Even though a unit 4160-Volt emergency bus may be energized, if all necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RCS makeup capability) are not functional on the energized bus, the bus should not be considered available.

The 15-minute interval was selected as a threshold to exclude transient power losses. If multiple sources fail to energize the unit 4160-Volt emergency busses within 15 minutes, an Alert is declared under this EAL.

SA1.1 (cont)

NAPS Basis Reference(s):

1. FE-1A Main One Line Diagram
2. UFSAR Section 8.3
3. 1-OP-26.1 (2-OP-26.1) Transferring 4160-Volt Busses
4. 0-AP-10 Loss of Electrical Power
5. 1-ECA-0.0 (2-ECA-0.0) Loss of All AC Power
6. NRC EAL FAQ 2006-002

SS1.1

Category: S – System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: Loss of **all** offsite power and loss of **all** onsite AC power to emergency busses

EAL

SS1.1	**Site Area Emergency**
Loss of **all** offsite and onsite AC power to Unit 1 (Unit 2) 4160-Volt emergency busses H and J for > 15 min. (Note 3)	

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Loss of offsite and onsite AC power compromises all plant *safety-related structures, systems and components* requiring electrical power. Unit 1 (Unit 2) 4160-Volt emergency busses H and J are the essential busses (ref. 1).

The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). The switchyard voltage is reduced from 500 kV and 230 kV to 34.5 kV. The RSSTs step down the 34.5 kV power to 4160-Volt AC for distribution as the preferred source of power to the emergency electrical distribution System.

The additional bus ties for Unit 1 between the emergency bus 1H and normal bus 1B and emergency bus 1J and normal bus 2B provide two independent offsite power sources to each emergency bus.

SS1.1 (cont)

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions.

If the SBO diesel generator is supplying power to an emergency bus of a unit that has lost all other sources of emergency offsite AC power, the unit has not lost all 4160-Volt AC power.

Consideration should be given to operable loads necessary to remove decay heat or provide RCS makeup capability when evaluating loss of AC power to the 4160-Volt emergency busses. Even though a unit 4160-Volt emergency bus may be energized, if all necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RCS makeup capability) are not functional on the energized bus, the bus should not be considered available.

Prolonged loss of all AC power will cause core uncovery and loss of containment integrity; thus, this event can escalate to a General Emergency under EAL SG1.1 or Fission Product Barrier matrix. The 15-minute interval was selected as a threshold to exclude transient power losses.

SS1.1 (cont)

NAPS Basis Reference(s):

1. 11715-FE-1A Main One Line Diagram (Unit 1)
2. UFSAR Section 8.3
3. 1-OP-26.1 (2-OP-26.1) Transferring 4160-Volt Busses
4. 0-AP-10 Loss of Electrical Power
5. 1-ECA-0.0 (2-ECA-0.0) Loss of All AC Power
6. NRC EAL FAQ 2006-003
7. 12050-FE-1A Main One Line Diagram (Unit 2)

SS1.2

Category: S – System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: Loss of **all** vital DC power

EAL:

SS1.2	**Site Area Emergency**
Loss of **all** vital DC power based on < 105-Volt DC bus voltage indications for > 15 min. (Note 3)	

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Loss of all DC power compromises the ability to monitor and *control* plant *safety functions*. Prolonged loss of all DC power will cause core uncovery and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

There are four independent 125-Volt DC systems. DC power is supplied for:

- Control power to 4160 and 480-Volt AC breakers
- Emergency lighting
- DC motor-driven pumps
- AC inverters that supply the 120V ac vital buses

Each 125-Volt DC system is normally powered through its respective battery charger. Each system consists of a 125-Volt DC distribution panel and a respective battery. The battery chargers convert 480-Volt AC power to a 12-Volt DC regulated output, which powers the associated 125-Volt DC busses and maintains a floating charge on the batteries connected to the busses.

SS1.2 (cont)

Each unit has six battery chargers. Four of the battery chargers are called normal battery chargers and are normally used to provide 125-Volt DC to their respective dc bus and battery. Two of the battery chargers are swing battery chargers and can be used as installed spares for either of the two normal battery chargers in the respective safeguards train. The batteries supply power only if the battery chargers fail or if the demand *exceeds* the capacity of the chargers. Each battery consists of 60 cells connected in series and is located in individual battery rooms. A battery terminal voltage of 105 volts DC is the minimum required to ensure proper operation of equipment connected to the DC bus (ref. 1).

This EAL is the hot condition equivalent of the cold condition loss of DC power
EAL CU1.2.

NAPS Basis Reference(s):

1. 0-OP-6.4 Operation of the SBO Diesel (SBO Event)
2. UFSAR Section 8.3.2
3. 0-AP-10 Loss of Electrical Power

SG1.1

Category: S – System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: Prolonged loss of **all** offsite power and prolonged loss of **all** onsite AC power to emergency busses

EAL:

SG1.1 General Emergency

Loss of **all** offsite and onsite AC power to Unit 1 (Unit 2) 4160-Volt emergency busses H and J

AND EITHER:

Restoration of **any** 4160-Volt emergency bus within 4 hours is **not** likely

OR

CSFST Core Cooling-RED or ORANGE path (Note 7)

Note 7: CSFSTs should be monitored for information only.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Loss of all AC power compromises all plant *safety-related structures, systems and components* requiring electrical power including RHR, ECCS, Containment heat removal and secondary heat removal. Prolonged loss of all AC power leads to loss of Fuel Clad, RCS and Containment barriers. The four-hour interval (ref. 1) to *restore* AC power is based on the blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout."

Although this EAL may be viewed as redundant to the Fission Product Barrier EALs, its inclusion is necessary to better assure timely recognition and emergency response.

SG1.1 (cont)

The likelihood of restoring at least one 4160-Volt emergency bus to a unit should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions. Unit 1 (Unit 2) 4160-Volt emergency busses H and J are the essential busses (ref. 2).

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs).

The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). The switchyard voltage is reduced from 500 kV and 230 kV to 34.5 kV. The RSSTs step down the 34.5 kV power to 4160-Volt AC for distribution as the preferred source of power to the emergency electrical distribution System.

The additional bus ties for Unit 1 between the emergency bus 1H and normal bus 1B and emergency bus 1J and normal bus 2B provide two independent offsite power sources to each emergency bus.

SG1.1 (cont)

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions. If the SBO diesel generator is supplying power to an emergency bus of a unit that has lost all offsite AC power, the unit has not lost all 4160-Volt AC power.

Consideration should be given to operable loads necessary to remove decay heat or provide RCS makeup capability when evaluating loss of AC power to the 4160-Volt emergency busses. Even though a unit 4160-Volt emergency bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RCS makeup capability) are not operable on the energized bus, the bus should not be considered operable.

In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be *restored*, it is necessary to give the SEM a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of *fission product barriers* is imminent?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be *restored* in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

SG1.1 (cont)

Thus, indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on SEM judgment as it relates to imminent loss or potential loss of *fission product barriers* and degraded ability to monitor *fission product barriers*. Indication of continuing core cooling degradation is manifested by entry to Critical Safety Function Status Tree Core Cooling-ORANGE or RED paths (ref. 3).

NAPS Basis Reference(s):

1. Calculation CA-022198, Determination Of Station Blackout Coping Duration Category North Anna Power Station Units 1 & 2.
2. 11715-FE-1A Main One Line Diagram (Unit 1)
3. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 2 Core Cooling
4. 1-FR-C.1 (2-FR-C.1) Response to Inadequate Core Cooling
5. 1-FR-C.2 (2-FR-C.2) Response to Degraded Core Cooling
6. UFSAR Section 8.3
7. 1-OP-26.1 (2-OP-26.1) Transferring 4160-Volt Busses
8. 0-AP-10 Loss of Electrical Power
9. 1-ECA-0.0 (2-ECA-0.0) Loss of All AC Power
10. NRC EAL FAQ 2006-016
11. 12050-FE-1A Main One Line Diagram (Unit 2)

SA2.1

Category: S – System Malfunction

Sub-category: 2 – RPS Failure

Initiating Condition: Automatic trip fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor

EAL:

SA2.1 Alert

An automatic trip failed to shutdown the reactor and manual actions taken at the Main Control Room (MCR) Bench Board successfully shutdown the reactor as indicated by reactor power < 5%

Mode Applicability:

1 - Power Operation, 2 - Startup

Basis:

A reactor trip is automatically initiated by the Reactor Protection System (RPS) when certain continuously monitored parameters exceed predetermined setpoints. A reactor trip may be the result of manual or automatic action (ref. 1):

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a few percent of the original power level and then decays to a level some 8 decades less at a startup rate of about -1/3 DPM. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from an automatic reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a negative startup rate as nuclear power drops into the source range.

SA2.1 (cont)

Per 1-E-0 (2-E-0), Reactor Trip or Safety Injection, the operator ensures that the reactor has tripped by (ref. 2):

- Manually tripping the reactor
- Checking the reactor trip and bypass breakers are open and rod bottom lights are lit
- Observing neutron flux is decreasing

If these responses cannot be verified, as part of contingency actions, the operator enters 1-FR-S.1 (2-FR-S.1) Response to Nuclear Power Generation / ATWS and the reactor is tripped locally from the Rod Drive room or the 307 Switchgear. Local opening of the reactor trip breakers requires actions outside of the Control Room; rapid control rod insertion by these methods is therefore **not** considered a “successful” manual reactor trip. For purposes of emergency classification, a “successful” manual reactor trip, therefore, includes only those immediate actions taken by the reactor operator at the Main Control Room (MCR) Bench Board to open the reactor trip and bypass breakers (ref. 3).

A reactor trip resulting from actuation of the AMSAC logic is not considered a successful reactor trip for the purposes of this EAL.

The Alert emergency classification is required whenever the Shift Manager determines that a required automatic reactor trip did not occur. It is recognized that 1-E-0 (2-E-0) instructs the operator to insert a manual reactor trip whether or not a required automatic reactor trip actually occurred. However, the *failure* of the automatic RPS trip signal to complete a reactor trip following receipt of an automatic trip signal meets the Alert classification threshold of potential substantial degradation in the level of safety of the plant. This is true even if no radiation alarms indicate fuel problems.

SA2.1 (cont)

In the event that the operator identifies a reactor trip is imminent and successfully initiates a manual reactor trip before the automatic trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel *failures* does not lead to a potential fission product barrier loss. If manual reactor trip actions at the main control boards (following an unsuccessful automatic reactor trip) fail to reduce reactor power below 5% (ref. 5), the event escalates to the Site Area Emergency under EAL SS2.1.

In the event that the operator identifies that a manual reactor trip is warranted and no automatic trip setpoint has been reached, and attempts to trip the unit but the trip is unsuccessful, and subsequently an automatic trip signal is received (e.g., the operator manually trips the main turbine or any other auto trip signal is received) and the auto trip signal successfully trips the reactor, then no classification is required. This is true even though the manual trip signal goes to the UV coil as does all the automatic trip signals. The manual trip is not considered an automatic trip for the purposes of EAL classification in the RPS subcategory.

This EAL also applies in Startup Mode when an automatic trip fails to shutdown the reactor when power was less than 5% at the time of the automatic trip failure.

NAPS Basis Reference(s):

1. UFSAR Table 7.2-1
2. 1-E-0 (2-E-0) Reactor Trip or Safety Injection
3. 1-FR-S.1 (2-FR-S.1) Response to Nuclear Power Generation / ATWS
4. UFSAR Figure 7.2-13
5. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 1 Subcriticality

SS2.1

Category: S – System Malfunction

Sub-category: 2 – RPS Failure

Initiating Condition: Automatic Trip fails to shutdown the reactor and manual actions taken from the reactor control console are **not** successful in shutting down the reactor

EAL:

SS2.1	**Site Area Emergency**

An automatic trip failed to shutdown the reactor and manual actions taken at the Main Control Room (MCR) Bench Board **do not** shutdown the reactor as indicated by reactor power $\geq$5%

Mode Applicability:

1 - Power Operation, 2 - Startup

Basis:

This EAL addresses any automatic reactor trip signal followed by a manual trip that fails to shut down the reactor to an extent the reactor is producing energy (≥ 5%, ref. 1) in excess of the heat load for which the *safety-related structures, systems and components* were designed. A manual reactor trip is any set of actions taken by the operator(s) at the Main Control Bench Board for the purpose of rapidly inserting control rods into the core.

A reactor trip resulting from actuation of the AMSAC logic is not considered a successful reactor trip.

SS2.1 (cont)

Automatic and manual trips are not considered successful if action away from the Control Room is required to trip the reactor. Local operator action to open the reactor trip and bypass breakers in the Rod Drive room and 307 Switchgear is not considered a "successful" manual reactor trip. If any of the alternate recovery actions for emergency boration of the RCS listed in EOPs are required to reduce reactor power below 5%, the reactor trips have been unsuccessful. Start up rate (SUR) on the Intermediate Range and Gamma-Metrics Wide-Range power are used as indicators of decreasing power and should be observed following any reactor trip from power (ref. 3).

The combination of *failure* of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat poses a direct threat to the Fuel Clad and RCS barriers and warrants declaration of a Site Area Emergency.

Escalation of this event to a General Emergency would be under EAL SG2.1 or SEM judgment.

NAPS Basis Reference(s):

1. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 1 Subcriticality
2. UFSAR Figure 7.2-13
3. 1-FR-S.1 (2-FR-S.1) Response to Nuclear Power Generation / ATWS
4. UFSAR Table 7.2-1
5. 1-E-0 (2-E-0) Reactor Trip or Safety Injection

SG2.1

Category: S – System Malfunction

Sub-category: 2 – RPS Failure

Initiating Condition: Automatic trip and **all** manual actions fail to shutdown the reactor and indication of an extreme challenge to the ability to cool the core exists

EAL:

SG2.1 General Emergency

An automatic trip failed to shutdown the reactor and **all** manual actions **do not** shutdown the reactor as indicated by reactor power $\geq$ 5%

AND EITHER:

CSFST Core Cooling-RED

OR

CSFST Heat Sink-RED

Mode Applicability:

1 - Power Operation, 2 - Startup

Basis:

This EAL addresses any automatic reactor trip signal followed by all manual trips failing to shut down the reactor to an extent the reactor is producing energy (≥ 5%, ref. 1) in excess of the heat load for which the *safety-related structures, systems and components* were designed.

SG2.1 (cont)

The combination of *failure* of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS barriers.

Indication that core cooling is extremely challenged is manifested by entry to Critical Safety Function Status Tree (CSFST) Core Cooling-RED path. (ref. 4, 5):

Indication that heat removal is extremely challenged is manifested by entry to CSFST Heat Sink-RED path (ref. 6, 7).

In the event the challenge to either core cooling or heat removal occurs at a time when the reactor has not been brought below the power associated with safety-related structure, system and component design power (5%, ref. 1), a core melt sequence may exist and rapid degradation of the fuel clad could begin. To permit maximum offsite intervention time, the General Emergency declaration is therefore appropriate in anticipation of an inevitable General Emergency declaration due to Loss and Potential Loss of *fission product barriers*.

NAPS Basis Reference(s):

1. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 1 Subcriticality
2. UFSAR Figure 7.2-13
3. 1-FR-S.1 (2-FR-S.1) Response to Nuclear Power Generation / ATWS
4. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 2 Core Cooling
5. 1-FR-C.1 (2-FR-C.1) Response to Inadequate Core Cooling
6. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 3 Heat Sink
7. 1-FR-H.1 (2-FR-H.1) Response to Loss of Secondary Heat Sink
8. UFSAR Table 7.2-1
9. 1-E-0 (2-E-0) Reactor Trip or Safety Injection

SU3.1

Category: S – System Malfunction

Sub-category: 3 – Inability to Reach or Maintain Shutdown Conditions

Initiating Condition: Inability to reach required shutdown within Technical Specification limits

EAL:

SU3.1	**Notification of Unusual Event**
Plant is **not** brought to required operating mode within Technical Specifications LCO action statement time	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a prescribed shutdown mode when the Technical Specification configuration cannot be *restored*. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the Technical Specification requires a four-hour report under 10 CFR 50.72 (b) non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate declaration of an Unusual Event is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of an Notification of Unusual Event is based on the time at which the LCO-specified action completion period elapses under Technical Specifications and is not related to how long a condition may have existed. Other Technical Specification shutdowns that involve precursors to more serious events are addressed by other EALs.

SU3.1 (cont)

NAPS Basis Reference(s):

1. Technical Specifications for North Anna Units 1 and 2
2. UFSAR Chapter 16 Technical Specifications and Requirements

SU4.1

Category: S – System Malfunction

Sub-category: 4 – Instrumentation / Communications

Initiating Condition: *Unplanned* loss of **most** or **all** safety-related structures, systems and components annunciation or indication in the Control Room for greater than 15 minutes

EAL:

SU4.1 Notification of Unusual Event

Unplanned loss of **most** (~75%) or **all** of **EITHER**:

- Annunciators (Panels “A” thru “N”)
- Indicators

associated with *safety-related structures, systems and components* on Unit 1 (Unit 2) MCR Bench Board and Vertical Board for > 15 min. (Note 3)

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

The Main Control Room (MCR) Bench Board and Vertical Board are the main panels in the Main Control Room (MCR) (ref. 1). Most of the essential instruments and controls for power operation, and protective equipment that is immediately needed in cases of emergency, are mounted on the bench board in functional groupings. Recorders, indicators and annunciators are mounted on the vertical board in agreement, wherever appropriate, with the functional groupings of the bench board. Annunciator Panels 1-EI-CB-21A (2-EI-CB-21A) through -21H, -21J through -21N are located at the top of the vertical boards. Annunciators or indicators for this EAL include those identified in the Abnormal Operating Instructions, the Emergency Operating Instructions and other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

SU4.1 (cont)

UFSAR Tables 7.5-1 and 7.5-2 (ref. 2, 3) list the information readouts provided to the operator to enable him to perform required manual *safety functions* and to determine the effect of manual actions taken following a reactor trip due to a Condition II, III, or IV event. Table 7.5-2 also contains the minimum set of parameters classified as Type A for Condition IV events as analyzed by Regulatory Guide 1.97 (ref. 4). The tables list the information readouts required to *maintain* the plant in a hot shutdown condition or to proceed to a cold shutdown condition within the limits of the Technical Specifications. UFSAR Table 7.5-3 (ref. 5) lists the information available to the operator for monitoring conditions in the reactor, the reactor coolant system, the containment, and process systems throughout all normal operating conditions of the plant, including anticipated operational occurrences.

This EAL recognizes the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment. The availability of computer-based monitoring capability (i.e., PCS or SPDS) is not a factor at the Notification of Unusual Event emergency classification level.

"*Unplanned*" loss of annunciators or indicators excludes scheduled maintenance and testing activities.

Quantification of "most" is arbitrary. If approximately 75% of the safety-related structure, system or component annunciators or indications are lost, an elevated risk exists that a degraded plant condition may be undetected. A detailed count of the lost instrumentation is not required. The judgment of the Shift Manager, however, should be used as the threshold for determining the severity of the plant conditions.

SU4.1 (cont)

Plant design provides redundant safety-related structure, system and component indication powered from separate uninterruptible power supplies. While *failure* of a large portion of annunciators is more likely than a *failure* of a large portion of indications, *failure* of indications is included in this EAL due to difficulty associated with assessment of plant conditions when indications are not available. The loss of several safety-related structure, system or component indicators should remain a function of the specific system or component operability status and is addressed by the applicable Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to instrument loss must be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action time, EAL SU3.1 ensures declaration of a Notification of Unusual Event.

The 15-minute interval offers time to recover from transient or momentary power losses. Due to the limited number of *safety-related structures, systems and components* in operation during Cold Shutdown, Refueling and Defueled modes, this EAL is not applicable during these modes of operation. If all computer-driven monitoring capability is *unavailable* or a *significant transient* is in progress during the loss of annunciation or indication, the event escalates to an Alert classification under EAL SA4.1.

SU4.1 (cont)

NAPS Basis Reference(s):

1. UFSAR Figure 1.2-3
2. UFSAR Table 7.5-1
3. UFSAR Table 7.5-2
4. Technical Report PE-0013 North Anna Power Station Response to Regulatory Guide 1.97
5. UFSAR Table 7.5-3
6. 1-AP-6 (2-AP-6) Loss of Main Control Room Annunciators
7. 1-AP-3 (2-AP-3) Loss of Vital Instrumentation
8. AR 1F-H6 UNIT #2 ANN SYS POWER SUPPLY FAILURE
9. AR 2F-H6 UNIT #1 ANN SYS POWER SUPPLY FAILURE
10. Dwg. 11715-FE-27B, Arrgt - Main Control Room El. 276' - 9"
11. Dwg. 11715-ESK-10A thru AH
12. Dwg. 11715-ESK-10B thru 10Q

SU4.2

Category: S – System Malfunction

Sub-category: 4 – Instrumentation / Communications

Initiating Condition: *Unplanned* loss of **all** onsite or offsite communications capabilities

EAL:

SU4.2 Notification of Unusual Event

Loss of **all** Table S-2 onsite (internal) communications capability affecting the ability to perform routine operations

OR

Loss of **all** Table S-2 offsite (external) communications capability

Table S-2 Communications Systems		
System	**Onsite (internal)**	**Offsite (external)**
Radio Communications System	**X**	
Public Address and Intercom System	**X**	
Private Branch Telephone Exchange (PBX)	**X**	
Sound Powered Telephone System	**X**	
Commercial Telephone		**X**
Dedicated NRC Communications		**X**

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

SU4.2 (cont)

Basis:

This EAL addresses loss of communications capability that either prevents the plant operations staff from performing routine tasks necessary for onsite plant operations or inhibits the ability to communicate problems externally to offsite authorities from the Control Room. The loss of offsite communications ability encompasses the loss of all means of communications with offsite authorities and is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This should include Commercial Telephone System (CTS), FAX transmissions and dedicated phone systems. This EAL is applicable only when extraordinary means are being utilized to make communications possible (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.).

The station communications system is designed to provide redundant means to communicate with all essential areas of the station associated with Units 1 and 2 and to essential locations remote from the station during normal operation and under accident conditions. Communication systems vital to operation and safety are designed so that *failure* of one component would not impair the reliability of the total communications system. Onsite/offsite communications include one or more of the systems listed in Table S-2 (ref. 1).

- <u>Radio Communications System (Onsite & Offsite)</u>

  An Ultra-High Frequency (UHF) two-way radio trunking system is provided at the Station consisting of base stations/repeaters, mobile units installed in emergency vehicles, and hand-held portable radios. The radio trunking system provides redundancy and independent emergency backup equipment for designated station functions.

SU4.2 (cont)

The same UHF two-way radio trunking system that provides onsite communications also provides for communications within a ten mile radius of the Station. During an emergency, this system allows direct contact with Radiation Monitoring Teams, Security vehicles, and a separate channel (Talk Group) between the Security Alarm Stations and the Louisa County Sheriff's Department.

- Public Address and Intercom System (Onsite)

  A five channel public address and intercom system (Gai-Tronics System) is installed in the Station. The system power is supplied from a power supply which will *maintain* the system in an operational condition in the event of a normal station service power *failure*. Zones are provided within that Station to insure operability of a major portion of the system should equipment in a zone become inoperative. In the event of an emergency, the system is used to alert Station personnel of any emergency situation and to direct emergency response actions required of on-site personnel.

- Private Branch Telephone Exchange, PBX (Onsite)

  The PBX switching equipment is physically located in the PBX Building and is connected to a commercial telephone exchange in Mineral, Virginia. Backup battery power is provided to *maintain* the system operable 6 to 8 hours following the loss of AC power.

SU4.2 (cont)

- <u>Sound Powered Telephone System (Onsite)</u>

  This system is a multiple channel system connecting selected operating areas of the plant. Headsets consisting of an earphone and microphone are connected to a two wire channel for direct communication between persons in different areas. Operation of this system is not dependent on the availability of the electrical power system. During an emergency, the system would provide an alternate means of relaying messages.

- <u>Commercial Telephone (Offsite)</u>

  Commercial telephone lines are provided between the Station and a commercial telephone exchange in Mineral, Virginia. These lines are connected into the Station PBX. In addition, lines are provided for communications between the Station and the commercial telephone network which are independent of the Station PBX.

- <u>Dedicated NRC Communications (Offsite)</u>

  Separate telephone lines are dedicated to the NRC and include the following:

SU4.2 (cont)

- Emergency Notification System (ENS): The system on which initial notifications, as well as ongoing information about plant systems, status and parameters, are provided to the NRC. ENS lines are located in the Control Room, TSC and LEOF.
- Health Physics Network (HPN): Provides for communications regarding radiological and meteorological conditions, assessments, trends, and protective measures. HPN lines are located in the TSC and LEOF.
- Reactor Safety Counterpart Link (RSCL): Allows for internal NRC discussions regarding plant and equipment conditions. RSCL lines are located in the TSC and LEOF.
- Protective Measures Counterpart Link (PMCL): Allows for the conduct of internal NRC discussions on radiological releases, meteorological conditions, and protective measures. PMCL lines are located in the TSC and LEOF.
- Emergency Response Data System (ERDS) Channel: Allows transmittal of reactor parametric data from the site to the NRC. ERDS data is transmitted from the PCS computer, via modem, to the NRC Operations Center.
- Management Counterpart Link (MCL): This system has been established for internal discussions between the NRC Executive Team Director/members and the NRC Director of Site Operations or licensee management. MCL lines are located in the TSC and LEOF.
- Local Area Network (LAN) Access: Provides access to the NRC local area network. Telephone jacks are provided in the TSC and LEOF for NRC LAN access.

SU4.2 (cont)

NAPS Basis Reference(s):

1. North Anna Power Station Emergency Plan, Section 7.2
2. UFSAR Section 9.5.2

SA4.1

Category: S – System Malfunction

Sub-category: 4 – Instrumentation / Communications

Initiating Condition: *Unplanned* loss of **most** or **all** safety-related structures, systems and components annunciation or indication in Control Room with **EITHER** (1) a *significant transient* in progress, **OR** (2) compensatory non-alarming indicators are *unavailable*

EAL:

SA4.1 Alert

Unplanned loss of **most** (~75%) or **all** of **EITHER**:

- Annunciators (Panels "A" thru "N")
- Indicators

associated with *safety-related structures, systems and components* on Unit 1 (Unit 2) MCR Bench Board and Vertical Board for > 15 min. (Note 3)

AND EITHER:

A *significant transient* is in progress (Table S-1)

OR

PCS is *unavailable*

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table S-1 Significant Transient
- Automatic turbine runback > 25% thermal reactor power
- Electrical load rejection > 25% full electrical load
- Reactor trip
- Safety injection activation
- Thermal power oscillations of > 10%

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

SA4.1 (cont)

Basis:

The Main Control Room (MCR) Bench Board and Vertical Board are the main panels in the Main Control Room (MCR) (ref. 1). Most of the essential instruments and controls for power operation, and protective equipment that is immediately needed in cases of emergency, are mounted on the bench board in functional groupings. Recorders, indicators and annunciators are mounted on the vertical board in agreement, wherever appropriate, with the functional groupings of the bench board. Annunciator Panels 1-EI-CB-21A (2-EI-CB-21A) through -21H, -21J through -21N are located at the top of the vertical boards.

Annunciators or indicators for this EAL include those identified in the Abnormal Operating Instructions, the Emergency Operating Instructions and other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

UFSAR Tables 7.5-1 and 7.5-2 (ref. 2, 3) list the information readouts provided to the operator to enable him to perform required manual *safety functions* and to determine the effect of manual actions taken following a reactor trip due to a Condition II, III, or IV event. Table 7.5-2 also contains the minimum set of parameters classified as Type A for Condition IV events as analyzed by Regulatory Guide 1.97 (ref. 4). The tables list the information readouts required to *maintain* the plant in a hot shutdown condition or to proceed to a cold shutdown condition within the limits of the Technical Specifications. UFSAR Table 7.5-3 (ref. 5) lists the information available to the operator for monitoring conditions in the reactor, the reactor coolant system, the containment, and process systems throughout all normal operating conditions of the plant, including anticipated operational occurrences.

This EAL recognizes the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a transient.

SA4.1 (cont)

"*Unplanned*" loss of annunciators or indicators does not include scheduled maintenance and testing activities.

Quantification of "most" is arbitrary. If approximately 75% of the safety-related structure, system and component annunciators or indications are lost, an elevated risk exists that a degraded plant condition may be undetected. A detailed count of the lost instrumentation is not required. The judgment of the Shift Manager, however, should be used as the threshold for determining the severity of the plant conditions.

Plant design provides redundant safety-related structure, system and component indication powered from separate uninterruptible power supplies. While *failure* of a large portion of annunciators is more likely than a *failure* of a large portion of indications, *failure* of indications is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of several safety-related structure, system and component indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to instrument loss must be reported via 10 CFR 50.72.

Table S-1, Significant Transients, includes response to automatic or manually initiated functions such as trips, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

If both a major portion of the annunciation system and all computer monitoring capability (i.e., PCS which includes SPDS) are *unavailable* to the extent that additional operating personnel are required to monitor indications, the Alert declaration is required. The compensatory indications include:

SA4.1 (cont)

- Plant Computer System (PCS, ref. 5): PCS is provided in each unit to assist the operator in the efficient operation of the plant. The primary function is to provide the operator with additional information regarding the condition of the nuclear steam supply system. It also has the capability to monitor inputs from the balance of plant systems and to alarm and log various off-normal conditions. There is no direct reactor *control* or protection action taken by the computer.
- Safety Parameter Display System (SPDS): The SPDS/ERG Validyne System (post-TMI data) receives inputs from SPDS and the ERG (Emergency Response Guidelines) System and converts this data to usable information for the PCS primary/backup server. The principle purpose of the SPDS system is to provide a display of plant parameters from which the safety status of operation may be assessed. It aids the Control Room personnel during abnormal and emergency conditions in determining the safety status of the plant and assessing whether abnormal conditions warrant corrective actions (ref. 6, 7, 8).

Due to the limited number of *safety-related structures, systems and components* in operation during Cold Shutdown, Refueling and Defueled modes, this EAL is not applicable during these modes of operation. If the operating crew cannot monitor the transient in progress, the Alert escalates to a Site Area Emergency under EAL SS4.1.

SA4.1 (cont)

NAPS Basis Reference(s):

1. UFSAR Figure 1.2-3
2. UFSAR Table 7.5-1
3. UFSAR Table 7.5-2
4. Technical Report PE-0013 North Anna Power Station Response to Regulatory Guide 1.97
5. UFSAR Table 7.5-3
6. UFSAR Section 7.7.1.10
7. UFSAR Section 8
8. VPAP-2606 Safety Parameter Display System (SPDS) (North Anna)
9. 1-AP-6 (2-AP-6) Loss of Main Control Room Annunciators
10. 1-AP-3 (2-AP-3) Loss of Vital Instrumentation
11. AR 1F-H6 UNIT #2 ANN SYS POWER SUPPLY FAILURE
12. AR 2F-H6 UNIT #1 ANN SYS POWER SUPPLY FAILURE
13. Dwg. 11715-FE-27B, Arrgt - Main Control Room El. 276' - 9"
14. Dwg. 11715-ESK-10A thru AH
15. Dwg. 11715-ESK-10B thru 10Q

SS4.1

Category: S – System Malfunction

Sub-category: 4 – Instrumentation / Communications

Initiating Condition: Inability to monitor a *significant transient* in progress

EAL:

SS4.1 Site Area Emergency

Loss of **most** or **all** (~75%) annunciators (Panels "A" thru "N") associated with *safety-related structures, systems and components* on Unit 1 (Unit 2) MCR Bench Board and Vertical Board

AND

PCS is *unavailable*

AND

Complete loss of ability to monitor **any** critical *safety function* status

AND

Significant transient is in progress (Table S-1)

Table S-1 Significant Transient
- Automatic turbine runback > 25% thermal reactor power - Electrical load rejection > 25% full electrical load - Reactor trip - Safety injection activation - Thermal power oscillations of > 10%

SS4.1 (cont)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

The Main Control Room (MCR) Bench Board and Vertical Board are the main panels in the Main Control Room (MCR) (ref. 1). Most of the essential instruments and controls for power operation, and protective equipment that is immediately needed in cases of emergency, are mounted on the bench board in functional groupings. Recorders, indicators and annunciators are mounted on the vertical board in agreement, wherever appropriate, with the functional groupings of the bench board. Annunciator Panels 1-EI-CB-21A (2-EI-CB-21A) through -21H, -21J through -21N are located at the top of the vertical boards. Annunciators or indicators for this EAL include those identified in the Abnormal Operating Instructions, the Emergency Operating Instructions and other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

UFSAR Tables 7.5-1 and 7.5-2 (ref. 2, 3) list the information readouts provided to the operator to enable him to perform required manual *safety functions* and to determine the effect of manual actions taken following a reactor trip due to a Condition II, III, or IV event. Table 7.5-2 also contains the minimum set of parameters classified as Type A for Condition IV events as analyzed by Regulatory Guide 1.97 (ref. 4). The tables list the information readouts required to *maintain* the plant in a hot shutdown condition or to proceed to a cold shutdown condition within the limits of the Technical Specifications. UFSAR Table 7.5-3 (ref. 4) lists the information available to the operator for monitoring conditions in the reactor, the reactor coolant system, the containment, and process systems throughout all normal operating conditions of the plant, including anticipated operational occurrences.

This EAL recognizes the inability of the Control Room staff to monitor the plant response to a *significant transient*. A Site Area Emergency exists if the Control Room staff cannot monitor *safety functions* needed for protection of the public.

SS4.1 (cont)

Quantification of “most” is arbitrary. If approximately 75% of the safety-related structure, system and component annunciators or indications are lost, an elevated risk exists that a degraded plant condition may be undetected. A detailed count of the lost instrumentation is not required. The judgment of the Shift Manager, however, should be used as the threshold for determining the severity of the plant conditions.

EOPs are entered if a *significant transient* is in progress. The hierarchy on controlling and maintaining *safety functions* within acceptance criteria are specified therein and include the following:

- Reactivity *control* (ability to shut down the reactor and keep it shutdown)
- RCS inventory (ability to cool the core)
- Secondary heat removal (ability to *maintain* a heat sink)
- Spent Fuel Pit cooling (ability to remove decay heat from irradiated fuel in storage)

Table S-1, Significant transients, includes response to automatic or manually initiated functions such as trips, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

Indications needed to monitor *safety functions* necessary for protection of the public must include Control Room indications, computer generated indications (i.e., PCS and SPDS) and dedicated annunciation capability. The specific indications should be those used to determine such functions as the ability to shut down the reactor, *maintain* the core cooled and in a coolable geometry, remove heat from the core, and *maintain* the reactor coolant system and containment intact.

Planned actions are included in the EAL since a loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

SS4.1 (cont)

NAPS Basis Reference(s):

1. UFSAR Figure 1.2-3
2. UFSAR Table 7.5-1
3. UFSAR Table 7.5-2
4. Technical Report PE-0013 North Anna Power Station Response to Regulatory Guide 1.97
5. UFSAR Table 7.5-3
6. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachments 1 thru 5

SU5.1

Category: S – System Malfunction

Sub-category: 5 – Fuel Clad Degradation

Initiating Condition: Fuel clad degradation

EAL:

SU5.1 Notification of Unusual Event

Dose rate at one foot (Note 5) from **EITHER**:

1 ml RCS sample $\geq$ 2.3 mR/hr

OR

120 ml RCS sample $\geq$ 234 mR/hr

Note 5: If actual reactor coolant activity samples are available at the time of the radiation measurement that indicate coolant activity levels are below that specified in EAL SU5.3 (Technical Specification coolant activity limit), declaration of the NOUE is not required under this EAL threshold.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

RCS sample one foot dose rates in excess of 2.3 mR/hr for a 1 ml sample or in excess of 234 mR/hr for a 120 ml sample are indicative of coolant activity in excess of 60 μCi/gm DEI-131. Dose rate is assumed to result from radioactive iodines (I-131 thru I-135) in RCS in concentrations corresponding to 60 μCi/gm DEI-131. This value corresponds to the NAPS Technical Specification coolant activity limit for iodine spike at full power operations.

SU5.1 (cont)

This EAL is included as an Notification of Unusual Event because it is considered to be a potential degradation of the level of safety of the plant and a potential precursor of more serious problems. Escalation of this EAL to the Alert level is via the Fission Product Barriers matrix.

NAPS Basis Reference(s):

1. NAF-07-002 Memo from Claude Flory to Brian McBride “NAPS/SPS EAL Upgrade Project – RCS Specific Activity” dated 1/29/07

SU5.2

Category: S – System Malfunction

Sub-category: 5 – Fuel Clad Degradation

Initiating Condition: Fuel clad degradation

EAL:

SU5.2	**Notification of Unusual Event**
With letdown in service, Reactor Coolant Letdown radiation monitor CH-RI-128 (CH-RI-228) > 1.5 x 10^4 mR/hr (Note 5)	

Note 5: If actual reactor coolant activity samples are available at the time of the radiation measurement that indicate coolant activity levels are below that specified in EAL SU5.3 (Technical Specification coolant activity limit), declaration of the NOUE is not required under this EAL threshold.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

The normal charging and letdown flow path allows purification of the reactor coolant and *control* of the RCS volume. Hot (547°F) reactor coolant from the cold leg of loop A passes through the regenerative heat exchanger. The regenerative heat exchanger cools the letdown stream to approximately 290°F. The discharge of the regenerative heat exchanger then passes through the non-regenerative heat exchanger, where its temperature is further reduced to approximately 105°F. A letdown filter removes suspended solids from the stream before entering one of five demineralizers. The letdown stream then flows through radiation monitor CH-RI-128 (CH-RI-228) to detect fission product activity in the reactor coolant and warn of a potential fuel element *failure*.

SU5.2 (cont)

The monitors detect gross fission product activity in the reactor coolant and are provided to warn of possible fuel element *failure*. When letdown is in service, a monitor reading above the threshold value is indicative of more than 60 μCi/gm DEI-131 accident mix (ref. 1) after 1 hour of decay. A monitor reading in excess of the threshold value (1.5E4 mR/hr, equivalent to 60 μCi/gm) indicates a challenge to the Technical Specification allowable limits for fuel clad degradation (iodine spike) requiring a reactor shutdown and be cooled down below 500 ºF within 6 hours. Escalation of this EAL to the Alert level is via the Fission Product Barriers matrix.

NAPS Basis Reference(s):

1. Calculation No. PA-0234, Rev. 1 Post Accident Letdown Radiation Monitor Response for North Anna
2. HP-3010.040 Radiation Monitoring System Setpoint Determination
3. UFSAR Section 9.3.4
4. 1-AP-5 (2-AP-5) Unit 1 Radiation Monitoring System

SU5.3

Category: S – System Malfunction

Sub-category: 5 – Fuel Clad Degradation

Initiating Condition: Fuel clad degradation

EAL:

SU5.3	**Notification of Unusual Event**
Coolant activity > 1.0 µCi/gm Dose Equivalent I-131 for > 48 hrs.	
OR	
Coolant activity > 60 µCi/gm Dose Equivalent I-131	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Coolant activity in excess of 1.0 µCi/gm indicates a challenge to the Technical Specification allowable limits for fuel clad degradation, with 48 hours to restore coolant activity below the limit. If coolant activity exceeds 60 µCi/gm, the reactor shall be shut down and cooled to 500 °F or less within 6 hours after detection. This EAL addresses reactor coolant samples exceeding coolant Technical Specifications. This EAL addresses reactor coolant samples exceeding Technical Specification LCO 3.4.16.a and 3.4.16.b, which are applicable for Modes 1, 2 and 3 with RCS average temperature (T_{avg}) ≥ 500°F (ref. 1). The Technical Specification limits accommodate an iodine spike phenomenon that may occur following changes in thermal power. The Technical Specification LCO limits are established to minimize the offsite radiological dose consequences in the event of a steam generator tube rupture (SGTR) accident. Escalation of this EAL to the Alert level is via the Fission Product Barriers matrix.

NAPS Basis Reference(s):

1. Technical Specifications 3.4.16
2. Technical Specifications Figure 3.4.16-1

SU6.1

Category: S – System Malfunction

Sub-category: 6 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU6.1 Notification of Unusual Event

Unidentified or pressure boundary leakage > 10 gpm for 15 minutes or longer (Note 3)

OR

Identified leakage > 25 gpm for 15 minutes or longer (Note 3)

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

The conditions of this EAL may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. Manual or computer-calculated water balance inventory methods are normally used to determine RCS leakage.

Identified leakage is defined in Technical Specifications (ref. 2) as:

- Leakage from pump seals or valve packing (except reactor coolant pump seal water injection or leakoff) that is captured and conducted to collection systems or a sump or collecting tank.
- Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage.
- RCS leakage through a steam generator to the Secondary System.

SU6.1 (cont)

Unidentified leakage is all leakage (except RCP seal water injection or leakoff) that is not identified leakage. Pressure Boundary leakage is leakage (except SG leakage) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

Generally, leakage into *closed* systems, or leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the unidentified leakage monitoring systems or not to be from a fault in the reactor coolant pressure boundary, are called identified leakages.

Uncontained leakage to the containment atmosphere may be the result of a variety of possible leakages that are generally classified as unidentified leakages. Unidentified leakage is eventually collected in tanks or sumps where the flowrate can be established and monitored during operation.

The 10 gpm value for the unidentified leakage and pressure boundary leakage was selected because it is quantifiable with normal Control Room leak detection methods. 1-PT-52.2 (2-PT-52.2), Reactor Coolant System Leak Rate (Hand Calculation), and 1-PT-52.2A (2-PT-52.2A), Reactor Coolant System Leak Rate (Computer Calculation), are performed to determine the source and flowrate of the leakage. Steam Generator leakage is also considered when evaluating unidentified leakage. The 25 gpm value for identified leakage is set at a higher value because of the significance of identified leakage in comparison to unidentified or pressure boundary leakage. Escalation of this EAL to the Alert level is via the Fission Product Barriers matrix.

SU6.1 (cont)

NAPS Basis Reference(s):

1. 1-AP-16 (2-AP-16) Increasing Primary Plant Leakage
2. Technical Specifications 1.1
3. 1-PT-52.2 (2-PT-52.2) Reactor Coolant System Leak Rate (Hand Calculation)
4. 1-PT-52.2A (2-PT-52.2A) Reactor Coolant System Leak Rate (Computer Calculation)
5. 1-AP-24 (2-AP-24) Steam Generator Tube Leak
6. 1-AP-24.1 (2-AP-24.1) Shutdown Steam Generator Tube Leak
7. 1-AP-52 (2-AP-52) Loss of Refueling Cavity Level During Refueling
8. UFSAR Section 5.2.4
9. Technical Specifications SR 3.4.13.1

SU7.1

Category: S – System Malfunction

Sub-category: 7 – Inadvertent Criticality

Initiating Condition: Inadvertent criticality

EAL:

SU7.1	**Notification of Unusual Event**
An *unplanned sustained* positive startup rate observed on nuclear instrumentation	

Mode Applicability:

3 - Hot Standby, 4 - Hot Shutdown

Basis:

Sustained is defined as prolonged, not intermittent or of transitory nature.

This EAL addresses inadvertent criticality events. While the primary concern of this EAL is criticality events that occur in Cold Shutdown or Refueling modes (see EAL CU6.1) (NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States), the EAL is applicable in other modes in which inadvertent criticalities are possible. This EAL indicates a potential degradation of the level of safety of the plant, warranting an Notification of Unusual Event classification. This EAL excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated). The Cold Shutdown/Refueling EAL is CU6.1. Escalation would be by the Fission Product Barrier matrix, as appropriate to the operating mode at the time of the event.

The term "*sustained*" is used in order to allow exclusion of expected short-term positive startup rates from planned fuel bundle or control rod movements during core alteration. These short-term positive startup rates are the result of the increase in neutron population due to subcritical multiplication.

SU7.1 (cont)

NAPS Basis Reference(s):

1. FSAR Table 7.5-3

<u>Category F – *Fission Product Barriers*</u>

EAL Group: Hot Conditions (RCS temperature > 200ºF); EALs in this category are applicable only in the hot operating modes.

EALs in this category represent threats to the defense in depth design concept that precludes the release of highly radioactive fission products to the environment. This concept relies on multiple physical barriers any one of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment. The primary *fission product barriers* are:

A. <u>Reactor Fuel Clad (FC):</u> The zirconium tubes which house the ceramic uranium oxide pellets along with the end plugs which are welded into each end of the fuel rods comprise the fuel clad.

B. <u>Reactor Coolant System (RCS):</u> The Reactor Vessel shell, vessel head, vessel nozzles and penetrations and all *primary systems* directly connected to the Reactor Vessel up to the first isolation valve comprise the RCS.

C. <u>Containment (CNTMT):</u> The vapor Containment structure and all isolation valves required to *maintain* Containment integrity under accident conditions comprise the Containment barrier.

The EALs in this category require evaluation of the loss and Potential Loss thresholds listed in the fission product barrier matrix (Attachment 2). "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

<u>Notification of Unusual Event:</u>
Any loss or any potential loss of Containment

<u>Alert:</u>
Any loss or any potential loss of either Fuel Clad or RCS

<u>Site Area Emergency:</u>

Loss or potential loss of any two barriers

<u>General Emergency:</u>
Loss of any two barriers and loss or potential loss of third barrier

The logic used for these initiating conditions reflects the following considerations:

- The Fuel Clad barrier and the RCS barrier are weighted more heavily than the Containment barrier. NOUE EALs associated with RCS and Fuel Clad barriers are addressed under System Malfunction EALs.
- At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS barrier loss EALs existed, that, in addition to offsite dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS barrier “Potential Loss” EALs existed, the SEM would have more assurance that there was **no** immediate need to escalate to a General Emergency.
- The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.

Fission Product Barrier EALs must be capable of addressing event dynamics. Imminent Loss or Potential Loss should result in a classification as if the affected threshold(s) are already exceeded, particularly for the higher emergency classes.

Determine which combination of the three barriers are lost or have a potential loss and use FU1.1, FA1.1, FS1.1 and FG1.1 to classify the event. Also an event or multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent. In this imminent loss situation, use judgment and classify as if the thresholds are exceeded.

FU1.1

Category: Fission Product Barriers

Sub-category: N/A

Initiating Condition: **Any** loss or **any** potential loss of Containment

EAL:

FU1.1	**Notification of Unusual Event**
Any loss or **any** potential loss of Containment (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Fuel Clad, RCS and Containment comprise the *fission product barriers*. Attachment 2 lists the fission product barrier thresholds, bases and references.

Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Fuel Clad and RCS barriers, the loss of either of which results in an Alert (EAL FA1.1), loss of the Containment barrier in and of itself does not result in the relocation of radioactive materials or the potential for degradation of core cooling capability. However, loss or potential loss of the Containment barrier in combination with the loss or potential loss of either the Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1.

NAPS Basis Reference(s):

None

FA1.1

Category: Fission Product Barriers

Sub-category: N/A

Initiating Condition: **Any** loss or **any** potential loss of either Fuel Clad or RCS

EAL:

FA1.1	**Alert**
Any loss or **any** potential loss of either Fuel Clad or RCS (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Fuel Clad, RCS and Containment comprise the *fission product barriers*. Attachment 2 lists the fission product barrier thresholds, bases and references.

At the Alert classification level, Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1.

NAPS Basis Reference(s):

None

FS1.1

Category: Fission Product Barriers

Sub-category: N/A

Initiating Condition: Loss or potential loss of **any** two barriers

EAL:

FS1.1	**Site Area Emergency**
Loss or potential loss of **any** two barriers (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Fuel Clad, RCS and Containment comprise the *fission product barriers*. Attachment 2 lists the fission product barrier thresholds, bases and references.

At the Site Area Emergency classification level, each barrier is weighted equally. A Site Area Emergency is therefore appropriate for any combination of the following conditions:

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

FS1.1 (cont.)

At the Site Area Emergency classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, the SEM would have greater assurance that escalation to a General Emergency is less imminent.

NAPS Basis Reference(s):

None

FG1.1

Category: Fission Product Barriers
Sub-category: N/A
Initiating Condition: Loss of **any** two barriers **AND** loss or potential loss of third barrier
EAL:

FG1.1 General Emergency Loss of **any** two barriers **AND** Loss or potential loss of third barrier (Table F-1)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Fuel Clad, RCS and Containment comprise the *fission product barriers*. Attachment 2 lists the fission product barrier thresholds, bases and references.

At the General Emergency classification level each barrier is weighted equally. A General Emergency is therefore appropriate for any combination of the following conditions:

- Loss of Fuel Clad, RCS and Containment barriers
- Loss of Fuel Clad and RCS barriers with potential loss of Containment barrier
- Loss of RCS and Containment barriers with potential loss of Fuel Clad barrier
- Loss of Fuel Clad and Containment barriers with potential loss of RCS barrier

NAPS Basis Reference(s):

None

Fuel Clad Barrier Potential Loss

1. **CSFST Core Cooling-ORANGE**

   OR

   CSFST Heat Sink-Red and heat sink required

   Critical Safety Function Status Tree (CSFST) Core Cooling-ORANGE path is entered if core exit thermocouples (TCs) are less than 1,200ºF, RCS core exit TCs SCM is less than or equal to 25ºF [75ºF], and any of the following (ref. 1, 2):

   - No RCPs are running and either: core exit TCs are greater than or equal to 700ºF and RVLIS full range is greater than 48%, or core exit TCs are less than 700ºF and RVLIS full range is less than or equal to 48%.
   - At least one RCP is running and Reactor Vessel water level is less than or equal to RVLIS dynamic head readings in Table F-2.

Table F-2 Reactor Vessel Water Level Thresholds		
RVLIS	**No. RCPs**	**Threshold**
Full Range	None	48%
Dynamic Range	3	65%
	2	41%
	1	30%

   These conditions indicate subcooling has been lost and that some fuel clad damage may potentially occur.

Indication that heat removal is extremely challenged is manifested by entry to CSFST Heat Sink-RED path (ref. 3, 4). CSFST Heat Sink-RED path is entered if all SG NR LVLs are less than or equal to 11% [22%] and total FW flow is less than or equal to 340 gpm. The combination of these conditions when heat sink is required indicates the ultimate heat sink function is under extreme challenge. The phrase “and heat sink required” precludes over-classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created by operator action directed by an Emergency Operating Procedure. This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus is a challenge of the Fuel Clad barrier.

CSFST values enclosed in brackets apply under Adverse Containment Atmosphere conditions:

- Greater than or equal to 20 psia Containment pressure, or
- Containment Radiation has reached or exceeded 10^5 R/hr or 70% on the High Range Recorder.

<u>NAPS References</u>:

1. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 2 Core Cooling
2. 1-FR-C.2 (2-FR-C.2) Response to Degraded Core Cooling
3. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 3 Heat Sink
4. 1-FR-H.1 (2-FR-H.1) Response to Loss of Secondary Heat Sink

2. **Core exit TCs > 700ºF**

The core exit TC value corresponds to the temperature in the Core Cooling Critical Safety Function Status Tree (CSFST) ORANGE path but is evaluated separately because the CSFST considers the degree of subcooling prior to status determination. This threshold is an explicit Fuel Clad potential loss to address conditions when the CSFSTs may not be in use (initiation after SI is blocked). This temperature indicates subcooling has been lost and that some fuel clad damage may occur.

NAPS References:

1. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 2 Core Cooling
2. 1-FR-C.2 (2-FR-C.2) Response to Degraded Core Cooling

3. **Reactor Vessel water level < Table F-2 thresholds**

Table F-2 Reactor Vessel Water Level Thresholds		
RVLIS	**No. RCPs**	**Threshold**
Full Range	None	48%
Dynamic Range	3	65%
	2	41%
	1	30%

The Reactor Vessel water levels listed in Table F-2 (ref. 1) are used in the EOPs to signal core uncovery and are, therefore, indications of inadequate coolant inventory. According to the Core Cooling-ORANGE path, these water levels indicate subcooling has been lost and that some fuel clad damage may occur.

NAPS References:

1. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 2 Core Cooling
2. 1-FR-C.1 (2-FR-C.1) Response to Inadequate Core Cooling

4. **Any condition in the opinion of the SEM that indicates potential loss of the Fuel Clad barrier**

The Station Emergency Manager (SEM) judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur based on a projection of current safety-related structure, system and component performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all *fission product barriers* and likely entry to the EOPs. The SEM should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

NAPS References:

None

Fuel Clad Barrier Loss

1. **CSFST Core Cooling-RED**

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path is entered if either:

- Core exit TCs are greater than or equal to 1,200ºF, or
- Core exit TCs are greater than or equal to 700ºF with RCS subcooling margin less than or equal to 25ºF [75ºF], no RCPs are running, and RVLIS full range is less than or equal to 48%.

CSFST values enclosed in brackets apply under Adverse Containment Atmosphere conditions:

- Greater than or equal to 20 psia Containment pressure, or
- Containment Radiation has reached or exceeded 10^5 R/hr or 70% on the High Range Recorder.

Either set of conditions indicates significant core exit superheating and core uncovery. This is considered a loss of the Fuel Clad barrier.

NAPS References:

1. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 2 Core Cooling
2. 1-FR-C.1 (2-FR-C.1) Response to Inadequate Core Cooling

2. **Core exit TCs > 1,200ºF**

Core exit TC readings greater than 1200°F indicate significant core exit superheating and core uncovery. This is considered a loss of the Fuel Clad barrier.

NAPS References:

1. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 2 Core Cooling
2. 1-FR-C.1 (2-FR-C.1) Response to Inadequate Core Cooling

3. **Containment High Range Radiation Monitor > Table F-3 Fuel Clad Loss threshold**

Containment radiation monitor readings greater than the Table F-3 (ref. 1) Fuel Clad Loss threshold indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the Containment. The reading is derived assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I-131 into the Containment atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within Technical Specifications and are therefore indicative of fuel damage (approximately 5 % clad *failure* depending on core inventory and RCS volume). This value is higher than that specified for RCS Barrier Loss #1.

The values used in Table F-3 were derived using the DAMAGE program. The following graph shows the expected monitor response to 5% clad *failure* for a given time after shutdown. Values were rounded to instrument significant values and placed in a table to simplify use for individuals classifying and event.

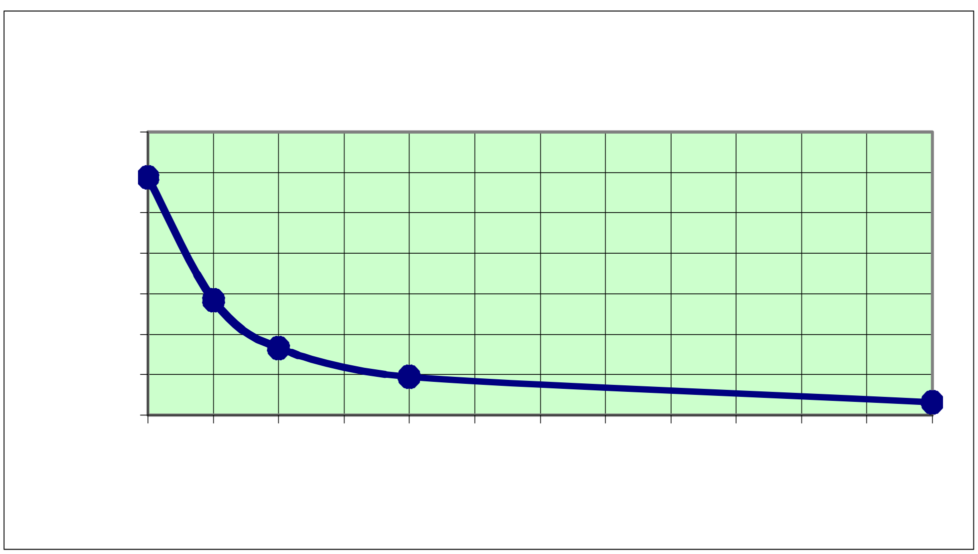


Table F-3 Containment High Range Radiation Monitor Thresholds RM-RMS-165 (265) or RM-RMS-166 (266)			
Time After Shutdown (hrs)	Fuel Clad Loss (R/hr)	RCS Loss (R/hr)	Containment Potential Loss (R/hr)
$\leq$ 2	300	5	1300
>2 to 4	200	5	800
>4 to 8	100	5	500
> 8 to 12	60	5	200
>12	40	5	100

It is important to recognize that the radiation monitor may be sensitive to shine from the Reactor Vessel or RCS piping.

Monitors used for this fission product barrier loss threshold are RM-RMS-165 (265) and RM-RMS-166 (266). These monitors provide indication in the Control Room with a range of 10^0 to 10^7 R/hr (ref. 2).

<u>NAPS References</u>:

1. NAPS Core Damage Computer Program
2. UFSAR Table 7.5.2
3. Calc-PA-0186 Rev. 000, Containment High Range Monitor Response Curves for North Anna and Surry

4. **Dose rate at one foot from <u>EITHER</u>:**
1 ml RCS sample $\geq$ 16 mR/hr
<u>OR</u>
120 ml RCS sample $\geq$ 1,550 mR/hr

An alternative method used to determine 5% failed fuel (~300 µCi/gm Dose Equivalent I-131) is to take a survey meter reading of either a one or 120 milliliter sample of RCS at one foot distance. Dose rate is assumed to result from radioactive iodines (I-131 thru I-135) in RCS in concentrations corresponding to a loss of 5% of gap radioactivity. For 5% loss of gap it is assumed that 2% of core inventory of radioactive iodines are in the gap. Core inventory was obtained from Calculation PA-0186 and RCS volume from DAMAGE code to yield iodine radioactivity concentrations in the RCS sample. The code MICROSHIELD was then used to determine dose rate at one foot from various volumes of RCS sample.

<u>NAPS References</u>:

1. NAF-07-002 Memo from Claude Flory to Brian McBride “NAPS/SPS EAL Upgrade Project – RCS Specific Activity” dated 1/29/07

5. **With letdown in service, Reactor Coolant Letdown radiation monitor CH-RI-128 (CH-RI-228) > 7.5 x 10^4 mR/hr**

The normal charging and letdown flow path allows purification of the reactor coolant and *control* of the RCS volume. Hot (547°F) reactor coolant from the cold leg of loop A passes through the regenerative heat exchanger. The regenerative heat exchanger cools the letdown stream to approximately 290°F. The discharge of the regenerative heat exchanger then passes through the non-regenerative heat exchanger, where its temperature is further reduced to approximately 105°F. A letdown filter removes suspended solids from the stream before entering one of five demineralizers. The letdown stream then flows past the radiation monitor CH-RI-128 (CH-RI-228).

The monitor detects gross fission product activity in the reactor coolant and is provided to warn of possible fuel element *failure*.

The letdown monitor reading of 7.5 x 10^4 mR/hr is equivalent to 300 µCi/gm dose equivalent I-131 (ref. 1).

NAPS References:

1. Calculation No. PA-0234, Rev. 1 Post Accident Letdown Radiation Monitor Response for North Anna
2. HP-3010.040 Radiation Monitoring System Setpoint Determination
3. UFSAR Section 9.3.4
4. 1-AP-5 (2-AP-5) Unit 1 Radiation Monitoring System

6. **Coolant activity > 300 µCi/gm Dose Equivalent I-131**

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The threshold Dose Equivalent (DEQ) I-131 concentration is well above that expected for iodine spikes and corresponds to about 2-5% fuel clad damage. When reactor coolant activity reaches this level, significant clad heating has occurred and thus the Fuel Clad barrier is considered lost.

NAPS References:

1. Calculation No. PA-0234, Rev. 0 Post Accident Letdown Radiation Monitor Response for North Anna

7. **Any condition in the opinion of the SEM that indicates loss of the Fuel Clad barrier**

The Station Emergency Manager (SEM) judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur based on a projection of current safety-related structure, system and component performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all *fission product barriers* and likely entry to the EOPs. The SEM should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

NAPS References:

None

Reactor Coolant System Barrier Potential Loss

1. **CSFST RCS Integrity-RED**
   OR

   CSFST Heat Sink-RED and heat sink required

   Critical Safety Function Status Tree (CSFST) RCS Integrity-RED path is entered if a temperature decrease in any RCS cold leg is greater than or equal to 100ºF in last 60 minutes and any RCS pressure-cold leg temperature is to the right of Limit A in Figure F-1 (ref. 1).

   CSFST Heat Sink-RED path is entered if all SG NR LVLs are less than or equal to 11% [22%] and total FW flow is less than or equal to 340 gpm (ref. 2). The phrase “and heat sink required” precludes over-classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created by operator action directed by an Emergency Operating Procedure.

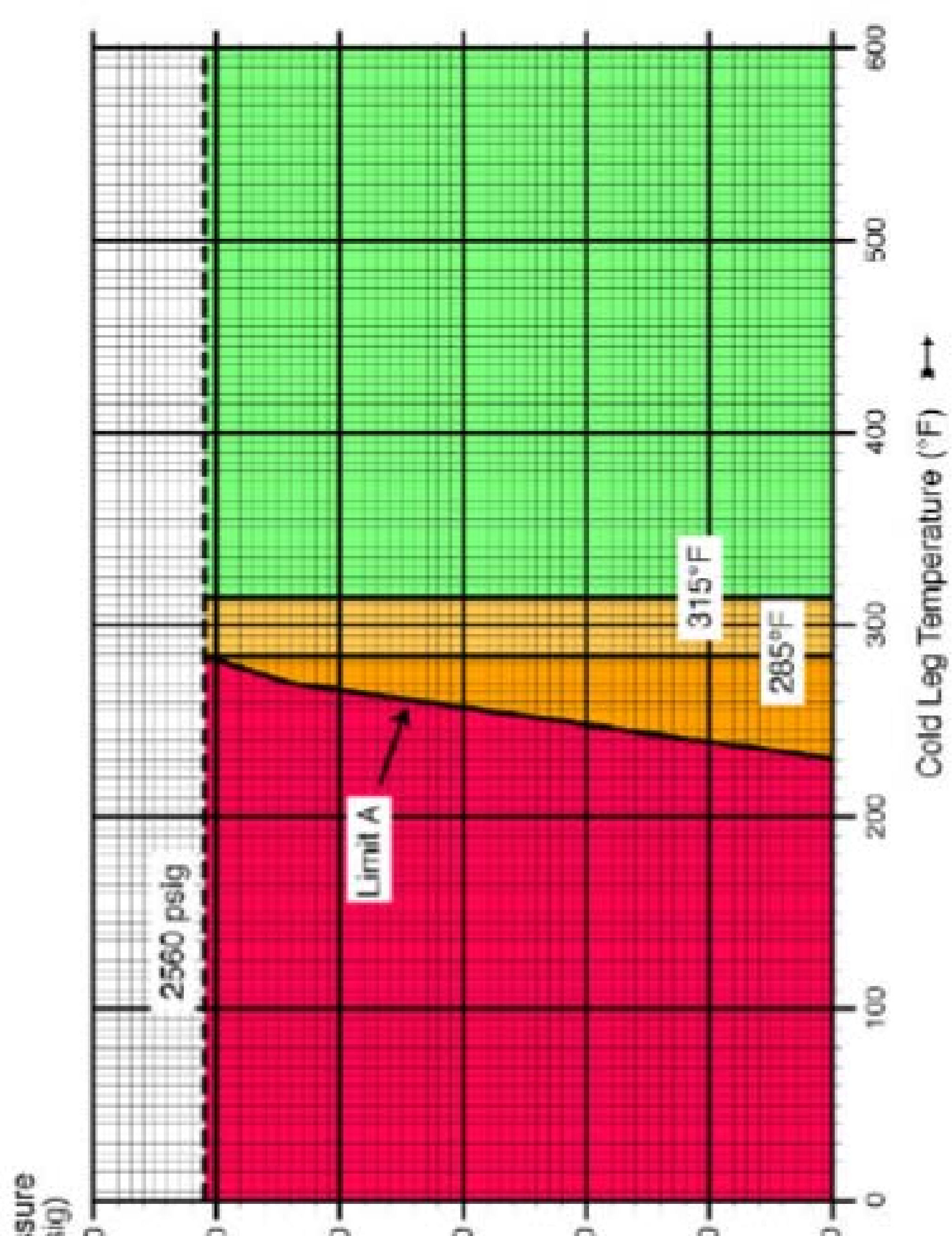
   The combination of these conditions indicates the RCS barrier is under significant challenge. CSFST values enclosed in brackets apply under Adverse Containment Atmosphere conditions (ref. 3):

   - Greater than or equal to 20 psia Containment pressure, or
   - Containment Radiation has reached or exceeded 10^5 R/hr or 70% on the High Range Recorder.

   NAPS References:

   1. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 4 Integrity
   2. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 3 Heat Sink
   3. 1-FR-H.1 (2-FR-H.1) Response to Loss of Secondary Heat Sink

Figure F-1: Integrity CSF Operational Limits Curve



2. **Unisolable RCS leak exceeding the capacity of one charging pump (150 gpm) in the normal charging mode**

This threshold is based on the inability to *maintain* liquid inventory within the RCS by normal operation of the Chemical and Volume Control System (CVCS). The CVCS includes three centrifugal charging pumps each with a nominal flow capacity of 150 gpm (ref. 1). The primary purpose of the charging pumps is to provide the motive force for injecting charging water into the RCS. They also provide seal water injection flow to the RCPs, the means to fill isolated RCS loops, and flow for auxiliary spray to the pressurizer. Additionally, the charging pumps provide high head safety injection. Upon actuation of the Safety Injection (SI) System, the charging pumps automatically take suction on the RWSTs and inject the contents of the BIT into the RCS. Charging flowrate is determined from a pressurizer level signal. The Unit 1 and 2 charging pumps have the ability to be cross connected should all charging pumps of either unit become *unavailable* such that, either unit can provide charging water to the other unit (ref. 2).

<u>NAPS References</u>:

1. UFSAR Table 6.3-1
2. UFSAR Section 9.3.4

3. **Any condition in the opinion of the SEM that indicates potential loss of the RCS barrier**

The SEM judgment threshold addresses any other factors relevant to determining if the RCS barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- <u>Imminent barrier degradation</u> exists if the degradation will likely occur based on a projection of current safety-related structure, system and component performance. The term "imminent" refers to the inability to reach final safety acceptance criteria before completing all checks.
- <u>Barrier monitoring</u> capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- <u>Dominant accident sequences</u> lead to degradation of all *fission product barriers* and likely entry to the EOPs. The SEM should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

<u>NAPS References</u>:

None

Reactor Coolant System Barrier Loss

1. **Containment High Range Radiation Monitor > Table F-3 RCS Loss threshold**

Containment radiation monitor readings greater than the Table F-3 (ref. 1) RCS Loss threshold indicate the release of reactor coolant to the Containment. The readings assume the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the Containment atmosphere. Because of the very high fuel clad integrity, only small amounts of noble gases would be dissolved in the primary coolant. The readings are less than those specified for Fuel Clad barrier Loss #4 because no damage to the fuel clad is assumed. Only leakage from the RCS is assumed for this barrier loss threshold.

Conservative estimates (high RCS μCi/gm) indicated that the readings from release of the normal RCS inventory would be below normal readings on the monitor while the station was operating. Therefore, a value 5 times the normal Containment Radiation Monitor Reading of ~ 1 R/hr is used. The reading is less than that specified for Fuel Cladding barrier Loss because no damage to the fuel cladding is assumed. Only leakage from the RCS is assumed for this barrier loss threshold. The value is high enough to preclude erroneous classification of barrier loss due to *normal plant operations* and is the lowest readable value on the monitors.

Table F-3 Containment High Range Radiation Monitor Thresholds RM-RMS-165 (265) or RM-RMS-166 (266)			
Time After Shutdown (hrs)	Fuel Clad Loss (R/hr)	RCS Loss (R/hr)	Containment Potential Loss (R/hr)
< 2	300	5	1300
>2 to 4	200	5	800
>4 to 8	100	5	500
> 8 to 12	60	5	200
>12	40	5	100

It is important to recognize that the radiation monitor may be sensitive to shine from the Reactor Vessel or RCS piping.

Monitors used for this fission product barrier loss threshold are RM-RMS-165 (265) and RM-RMS-166 (266). These monitors provide indication in the Control Room with a range of 10^0 to 10^7 R/hr (ref. 2).

NAPS References:

1. NAPS Damage Computer Program
2. UFSAR Table 7.5.2
3. Calc-000-PA-0186 Rev. 000, Containment High Range Monitor Response Curves for North Anna and Surry

2. **RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling: < 25ºF [75ºF]**

1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 2 Core Cooling, indicates that if subcooling margin based on core exit TCs is less than or equal to 25°F [75ºF], a loss of RCS subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak. This threshold addresses conditions in which leakage from the RCS is greater than available inventory *control* capacity such that a loss of subcooling has occurred. 1-AP-16 (2-AP-16), Increasing Primary Plant Leakage, provides a list of conditions that may be observed when excessive RCS leakage occurs and provides appropriate actions to prevent and mitigate the consequences of RCS leakage.

Following an uncomplicated reactor trip, subcooling margin should be greater than 50°F. Subcooling margin greater than 25°F [75ºF] ensures the fluid surrounding the core is sufficiently cooled and provides margin for reestablishing flow should subcooling deteriorate when SI flow is secured. The loss of subcooling is therefore the fundamental indication that the inventory control systems are incapable of counteracting the mass loss through the leak in the RCS.

CSFST values enclosed in brackets apply under Adverse Containment Atmosphere conditions:

- Greater than or equal to 20 psia Containment pressure, or
- Containment Radiation has reached or exceeded 10^5 R/hr or 70% on the High Range Recorder.

The loss of subcooling as a result of inability to establish RCS heat transfer to the ultimate heat sink is indicative of Potential Losses of the Fuel Clad and RCS barriers.

<u>NAPS References</u>:

1. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 2 Core Cooling
2. 1-FR-C.2 (2-FR-C.2) Response to Degraded Core Cooling

3. 1-AP-16 (2-AP-16) Increasing Primary Plant Leakage

3. **SGTR that requires ECCS (SI) actuation**

In conjunction with Containment barrier Loss #3 and the Fuel Clad barrier thresholds, this threshold addresses the full spectrum of Steam Generator Tube Rupture (SGTR) events. To meet this threshold, the leakage must be large enough to require actuation of ECCS (SI). ECCS (SI) actuation is caused by:

- Low-low pressurizer pressure
- High steam-line pressure differential between the steam generators
- High steam-line flow in two out of three steam lines, coincident with either low steam-line pressure or low-low T_{avg} in two out of three loops
- High containment pressure

Technical Specifications Table 3.3.2-1 lists allowable values for ECCS (SI) actuation setpoints.

NAPS References:

1. UFSAR Section 7.3.1.3.3.1
2. UFSAR Figure 7.2-9
3. Technical Specifications Table 3.3.2-1

4. **Any condition in the opinion of the SEM that indicates loss of the RCS barrier**

The SEM judgment threshold addresses any other factors relevant to determining if the RCS barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur based on a projection of current safety-related structure, system and component performance. The term “imminent” refers to the recognition of the inability to reach safety acceptance criteria before completion of all checks.

- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all *fission product barriers* and likely entry to the EOPs. The SEM should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

NAPS References:

None

Containment Barrier Potential Loss

1. **CSFST Containment-RED**

Critical Safety Function Status Tree (CSFST) Containment-Red path is entered if Containment pressure is equal to or greater than 60 psia. This pressure is the containment design pressure and is in excess of that expected from the design basis loss of coolant accident. This threshold is indicative of a loss of both RCS and Fuel Clad barriers in that it is not possible to reach this condition without severe core degradation (metal-water reaction) or *failure* to trip in combination with RCS breach. This combination of conditions would be expected to require the declaration of a General Emergency.

NAPS References:

1. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 5 Containment
2. UFSAR Section 6.2.1

2. **Core exit TCs > 1,200°F**
AND
Restoration procedures not effective* within 15 min.

*Restoration procedures are considered effective if Core exit TCs readings are lowering or Reactor Vessel water level is rising within 15 min. after restoration procedure entry (Note 3).

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This threshold indicates significant core exit superheating and core uncovery. If core exit thermocouple (TC) readings are greater than 1,200°F, Fuel Clad barrier is lost. Core exit TCs provide an indirect indication of fuel clad temperature by measuring the temperature of the primary coolant that leaves the core region. Although clad rupture due to high temperature is not expected for core exit TC readings less than the threshold, temperatures of this magnitude signal significant superheating of the reactor coolant and core uncovery. Events that result in core exit TC readings above the loss threshold are severe accidents and are a severe accident management “Badly Damaged (BD)” condition.

The BD descriptor signifies possible core overheating to the point that clad ballooning/collapse may occur and portions of the core may have melted.

It must also be assumed the loss of RCS inventory is a result of a loss of the RCS barrier. These conditions, if not mitigated, can lead to core melt which in turn may result in a loss of Containment. Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the Reactor Vessel in a significant fraction of the core damage scenarios, and the likelihood of Containment *failure* is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. The 15-minute period allows implementation of procedural guidance to *restore* RCS inventory.

NAPS References:

1. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 2 Core Cooling
2. 1-FR-C.1 (2-FR-C.1) Response to Inadequate Core Cooling

3. All of the following:

- **Core exit TCs > 700°F**
- **Reactor Vessel water level < Table F-2 thresholds**
- **Restoration procedures not effective* within 15 min.**

Table F-2 Reactor Vessel Water Level Thresholds		
RVLIS	**No. RCPs**	**Threshold**
Full Range	None	48%
Dynamic Range	3	65%
	2	41%
	1	30%

*Restoration procedures are considered effective if Core exit TCs readings are lowering or Reactor Vessel water level is rising within 15 min. after restoration procedure entry (Note 3).

Note 3: The SEM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This threshold indicates significant core exit superheating (core exit TC readings >700°F) and core uncovery. It must be assumed that the loss of RCS inventory is a result of a loss of the RCS barrier. If RVLIS is reading greater than or equal to the Table F-2 thresholds, safety injection has been successful in restoring RCS inventory and core cooling. In the event that RVLIS reads less than Table F-2 thresholds, core cooling continues to be degraded. These conditions, if not mitigated, will likely lead to core melt which will in turn result in a challenge of Containment.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the Reactor Vessel in a significant fraction of the core damage scenarios, and that the likelihood of Containment *failure* is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. Whether or not procedures will be effective should be apparent within 15 minutes.

NAPS References:

1. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 2 Core Cooling
2. 1-FR-C.2 (2-FR-C.2) Response to Degraded Core Cooling

4. **Containment High Range Radiation Monitor > Table F-3 Containment Potential Loss threshold**

Containment radiation monitor readings greater than the Table F-3 Containment Potential Loss threshold values indicate significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier. NUREG-1228 “Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents” states that such readings do not exist when the amount of clad damage is less than 20%. A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major *failure* into the reactor coolant has occurred.

Regardless of whether the Containment barrier itself is challenged, this amount of activity in containment could have severe consequences if released. It is, therefore, prudent to treat this as a Potential Loss of the Containment barrier.

The values used in Table F-3 were derived using the DAMAGE program. The following graph shows the expected monitor response to 5% clad *failure* for a given time after shutdown. Values were rounded to instrument significant values and placed in a table to simplify use for individuals classifying and event.

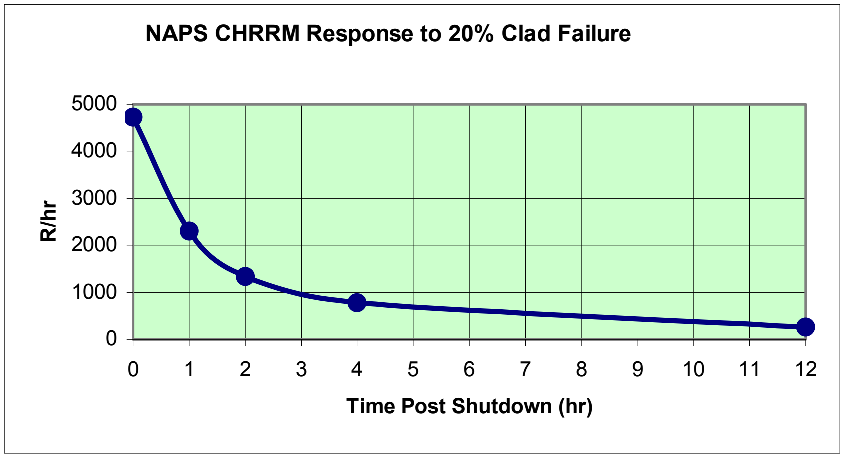


Table F-3 Containment High Range Radiation Monitor Thresholds RM-RMS-165 (RM-RMS-265) or RM-RMS-166 (RM-RMS-266)			
Time After Shutdown (hrs)	Fuel Clad Loss (R/hr)	RCS Loss (R/hr)	Containment Potential Loss (R/hr)
< 2	300	5	1300
>2 to 4	200	5	800
>4 to 8	100	5	500
> 8 to 12	60	5	200
>12	40	5	100

The readings are higher than that specified for Fuel Clad barrier Loss #3 and RCS barrier Loss #3. Containment radiation readings at or above the Containment barrier Potential Loss threshold, therefore, signify a loss of two *fission product barriers* and Potential Loss of a third, indicating the need to upgrade the emergency classification to a General Emergency.

It is important to recognize that the radiation monitor may be sensitive to shine from the Reactor Vessel or RCS piping.

Monitors used for this fission product barrier loss threshold are RM-RMS-165 (265) and RM-RMS-166 (266). These monitors provide indication in the Control Room with a range of 10^0 to 10^7 R/hr (ref. 2).

<u>NAPS References</u>:

1. NAPS Damage Computer Program
2. UFSAR Table 7.5.2
3. Calc-000-PA-0186 Rev. 000, Containment High Range Monitor Response Curves for North Anna and Surry

5. **Containment pressure 60 psia and increasing**

This threshold is the containment design pressure and is in excess of that expected from the design basis loss of coolant accident (LOCA). Proper actuation and operation of the Containment heat removal system when required should *maintain* containment pressure well below the design pressure. The Containment response for the spectrum of LOCAs considered in the plant design basis is described in Section 6 of the UFSAR. The threshold is therefore indicative of a loss of both RCS and Fuel Clad barriers in that it should not be reached without severe core degradation (metal-water reaction) or *failure* to trip in combination with RCS breach. This condition would be expected to require the declaration of a General Emergency.

<u>NAPS References</u>:

1. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 5 Containment
2. UFSAR Section 6.2.1

6. **Containment hydrogen concentration ≥ 4%**

After a LOCA, the containment atmosphere is a homogeneous mixture of steam, air, solid and gaseous fission products, hydrogen, and water droplets. During and following a LOCA, the hydrogen concentration in the containment results from radiolytic decomposition of water and metal-water reaction. If hydrogen concentration reaches or *exceeds* the *lower flammability limit* of 4%, (ref. 1) in an oxygen rich environment, a potentially explosive mixture exists. Operation of the Containment Hydrogen Recombiner with Containment hydrogen concentrations at or above 4% could result in ignition of the hydrogen. If the combustible mixture ignites inside containment, loss of the Containment barrier could occur. To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers must also have occurred. Since this threshold is also indicative of loss of both Fuel Clad and RCS barriers with the Potential Loss of the Containment barrier, it therefore will likely warrant declaration of a General Emergency.

Containment hydrogen analyzers 1-HC-H2A-101 and 2-HC-H2A-201 display hydrogen concentration on PAMC-1 and PAMC-2 with a range of 0 - 10% (ref. 2).

NAPS References:

1. 1-FR-C.1 (2-FR-C.1) Response to Inadequate Core Cooling
2. UFSAR Table 7.5-2
3. UFSAR Section 7.5.2
4. 1-OP-63.2 (2-OP-63.2) Containment Hydrogen Analyzer

7. **Containment pressure > 28 psia with less than one full train of depressurization equipment operating**

This threshold represents a Potential Loss of the Containment barrier because the Containment heat removal and depressurization equipment (but not including Containment venting strategies) is either lost or degraded. The Quench Spray (QS) System, operating in conjunction with the Recirculation Spray (RS) System, is designed to cool and depressurize the containment structure to subatmospheric pressure in less than 60 minutes following a Design Basis Accident (ref. 1, 2).

The QS System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a spray pump, a dedicated spray header, nozzles, valves, and piping. Each train is powered from a separate Engineered Safety Features (ESF) bus. The refueling water storage tank (RWST) supplies borated water to the QS System (ref. 1).

The RS System consists of two separate trains of equal capacity, each capable of meeting the design and accident analysis bases. Each train includes one RS subsystem outside containment and one RS subsystem inside containment. Each subsystem consists of one approximately 50% capacity spray pump, one spray cooler, one 180° coverage spray header, nozzles, valves, piping, instrumentation, and controls. Each outside RS subsystem also includes a casing cooling pump with its own valves, piping, instrumentation, and controls. The two outside RS subsystem spray pumps are located outside containment and the two inside RS subsystems spray pumps are located inside containment.

Each RS train (one inside and one outside RS subsystem) is powered from a separate ESF bus. Each train of the RS System provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission product removal. Two spray pumps are required to provide 360° of Containment spray coverage assumed in the accident analysis. One train of RS or two outside RS subsystems will provide the Containment spray coverage and required flow (ref. 2).

The high-high Containment pressure setpoint 28 psia (ref. 4) is the pressure at which the equipment should actuate and begin performing its function. The design basis accident analyses and evaluations assume the operation of one QS System train and one RS System train, which is therefore defined to be “one full train of depressurization equipment.” If less than this equipment is operating per design and Containment pressure is above the actuation setpoint, the threshold is met.

NAPS References:

1. Technical Specifications B 3.6.6
2. Technical Specifications B 3.6.7
3. 1-F-0 (2-F-0) Critical Safety Function Status Trees, Attachment 6 Containment
4. 1-FR-Z.1 (2-FR-Z.1) Response to High Containment Pressure
5. Technical Specifications Table 3.3.2-1

8. **Any condition in the opinion of the SEM that indicates potential loss of the Containment barrier**

The SEM judgment threshold addresses any other factors relevant to determining if the Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur based on a projection of current safety-related structure, system and component performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all *fission product barriers* and likely entry to the EOPs. The SEM should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

NAPS References:

None

<u>Containment Barrier Loss</u>

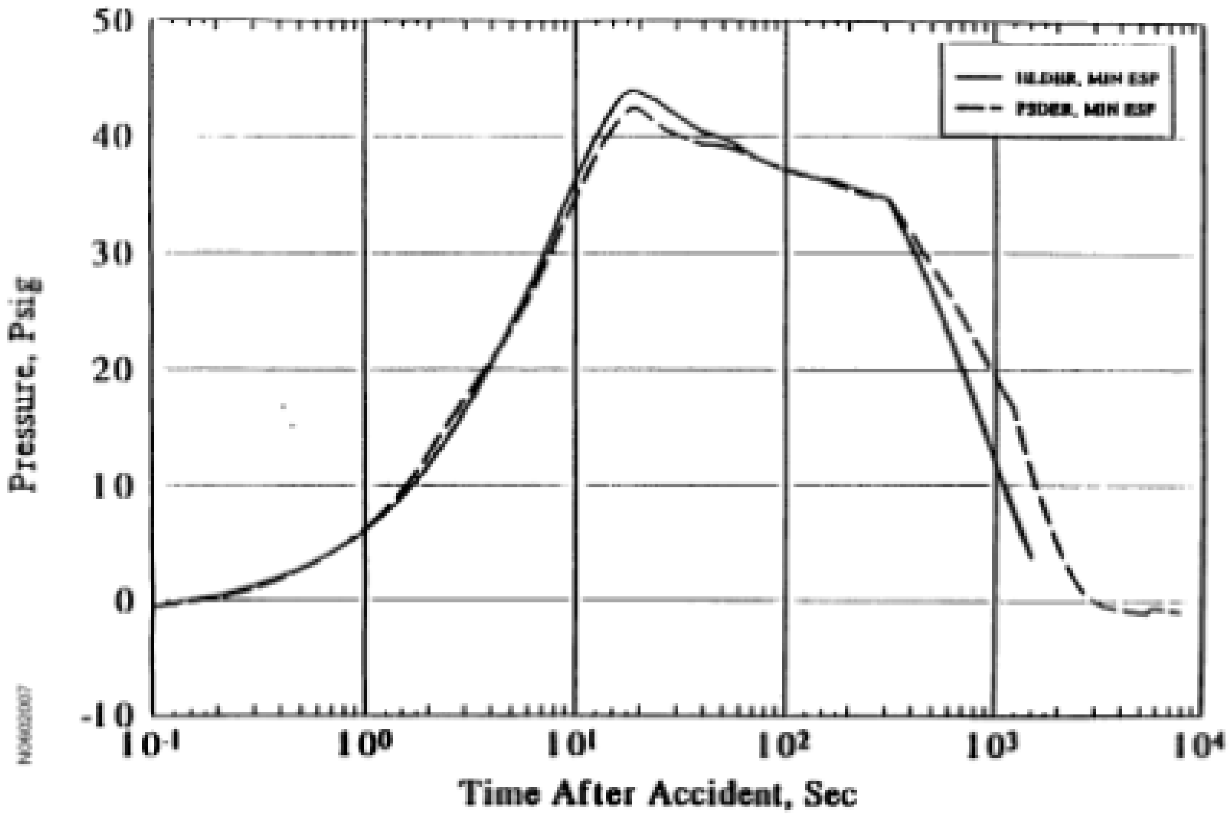
1. **A Containment pressure rise followed by a rapid unexplained drop in containment pressure**

Rapid unexplained loss of pressure (i.e., not attributable to Containment spray operation, running Containment Fan Cooling Units or condensation effects) following an initial pressure increase indicates a loss of both RCS and Containment integrity. UFSAR Section 6 describes Containment pressure response under accident conditions. Figure F-2 illustrates the Containment pressure trend for typical LOCA events.

<u>NAPS References</u>:

1. UFSAR Figure 6.2-6
2. UFSAR Section 6.2

Figure F-2: Containment Pressure Trend for LOCA Events (ref. 1)



2. **Following LOCA, Containment pressure or sump level response not consistent with LOCA conditions**

This threshold addresses unexpected changes occurring in Containment pressure or sump level that are not explainable due to operator actions or automatic system actions. Containment pressure and sump levels should increase as a result of the mass and energy release into Containment from a LOCA. Thus, Containment pressure or sump levels not increasing indicate Containment bypass and a loss of Containment integrity. UFSAR Section 6 describes containment pressure response for LOCA events.

<u>NAPS References</u>:

1. UFSAR Section 6.2

3. ***Ruptured* SG is also *faulted* outside of Containment**

Steam Generator (SG) tube leakage can represent a bypass of the Containment barrier as well as a loss of the RCS barrier. This threshold addresses the condition in which a *ruptured* SG is also *faulted* and represents a bypass of the RCS and Containment barriers. A *faulted* SG means the existence of secondary side leakage that results in an *uncontrolled* decrease in steam generator pressure or the steam generator being completely depressurized. A *ruptured* SG means the existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection. In conjunction with RCS barrier Loss #2, this threshold would always result in the declaration of a Site Area Emergency.

<u>NAPS References</u>:

1. 1-E-2 (2-E-2) Faulted Steam Generator Isolation
2. 1-E-3 (2-E-3) Steam Generator Tube Rupture

4. **Primary-to-secondary leakage > 10 gpm with non-isolable steam release from affected SG to the environment**

Steam Generator (SG) tube leakage can represent a bypass of the Containment barrier as well as a loss of the RCS barrier. This threshold represents a bypass of the RCS and Containment barriers. In conjunction with RCS barrier Loss #2, this would always result in the declaration of a Site Area Emergency.

The threshold for establishing the non-isolable secondary side release is intended to be a prolonged release of radioactivity from the affected steam generator directly to the environment. This could be expected to occur when the main condenser is *unavailable* to accept the contaminated steam (i.e., SGTR with concurrent loss of offsite power and the *ruptured* steam generator is required for plant cooldown or has a stuck open relief valve). If the main condenser is available, there may be releases through the air ejectors, gland seal exhausters and other similarly controlled and monitored pathways. These pathways do not meet the intent of a non-isolable release path to the environment. These minor releases are assessed using radiological effluent EAL thresholds.

A pressure boundary leakage of 10 gpm is also used as the threshold in RCS Leakage EAL SU6.1. For smaller breaks, not exceeding the normal charging capacity threshold in RCS barrier Potential Loss #2 or not resulting in ECCS actuation in RCS barrier Loss #2, this threshold results in the declaration of a Notification of Unusual Event. For larger breaks, RCS barrier Potential Loss #2 and RCS barrier Loss #2 would result in an Alert. For SG tube ruptures (SGTRs) which may involve more than one steam generator or unisolable secondary line breaks, this threshold would occur in conjunction with RCS barrier Loss #2 and would result in a Site Area Emergency. Escalation to General Emergency would be based on the Potential Loss of the Fuel Clad barrier.

There is some redundancy in the Containment Loss thresholds #3 and #4. This was recognized during the NEI EAL development process.

NAPS References:

1. 1-E-2 (2-E-2) Faulted Steam Generator Isolation
2. 1-E-3 (2-E-3) Steam Generator Tube Rupture

5. **CNTMT Isolation valve(s) not closed after any required CNTMT isolation**

   AND

   Downstream pathway to the environment exists

This threshold addresses incomplete Containment (CNTMT) isolation that allows direct release to the environment. It represents a loss of both the RCS and Containment barriers and therefore warrants declaration of a Site Area Emergency. *Failure* of Containment isolation or Containment ventilation isolation valves to isolate when required addresses incomplete Containment isolation that allows direct release to the environment.

An unisolable leak upstream of an outboard isolation valve would meet the intent of this threshold.

NAPS References:

1. 1-ECA-1 (2-ECA-1) LOCA Outside Containment

6. **Any condition in the opinion of the SEM that indicates loss of the Containment barrier**

The SEM judgment threshold addresses any other factors relevant to determining if the Containment barrier lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur based on a projection of current safety-related structure, system and component performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all *fission product barriers* and likely entry to the EOPs. The SEM should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

NAPS References:

None