Draft Submittal

NORTH ANNA JUNE EXAM
50-338 & 50-339/2004-301

JUNE 17 - 25,2004

1. Reactor Operator Operator Written Exam

(1 OF 2)

001K5 09 001

Unit 1 is currently at 50% power. Power escalation was stopped several hours ago due to a problem with governor valve #1. The reactor engineer has requested that rods be withdrawn to 110 steps on D bank to place them further above the rod insertion limits. The following plant conditions exist:

•

Which ONE of the following indicates the number of gallons of boric acid that will have to be added in order to maintain RCS Tave and reactor power stable at 50% power with rods at 110 steps on D bank?

A. **10.8**
B✓ 15.3
C. 26.6
D. 100.8

A. Correct. Using At-power rod worth of 634.2 (interpolated) for 105 steps and **607.6** for 110 steps and a boron worth of –6.73 pcm/ppm

634.2 - 607.6 = 26.6 pcm/ -6.73 = -3.95 ppm

Using equation 50455.4 * ln[(14000 - 975)/(14000 - 978.95)] = 15.3 gallons

Using rounder numbers 634 - 608 = 26 pcm/-6.7pcm/ppm = -3.88 ppm

Using equation 50455.4 * ln[(14000 - 975)/(14000 – 979)] = 15.5

B. Incorrect. This is the approximate answer you would get if you use HZP rod worth values.

C. Incorrect. This is the pcm needed. If candidate does not finish equation they could choose this answer.

D. Incorrect. This is the approximate answer you would get if you do not divide by the boron worth number, i.e. use **1001** as your final boron number versus 979.

Knowledge of the operational implications **of** the following concepts as they apply to the Control Rod drive system: Relationships between reactivity due to boron and reactivity due to **control** ***rod***
(CFR: 41.5 / ***45.7)***

New question

References used and needed: rod worth curve tables (3.5 - give both HZP and full power tables), boration table/equation (2.2), boron coefficient curve (3.4)

Level (RO/SRO):	RO	Tier:	**2**
Group:	**2**	Importance Rating:	3.5/3.7
Type (Bank/Mod/New):	**NEW**	Cog (Knowledge/Comp):	COMPREHENSION
Reference (Y/N):	***Y***	Last Exam(Y/N):	N

BORON ADDITION

~~LEVEL 3 CONTROLLED COPY~~

The general equation for tke 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:

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

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: E/Wk 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

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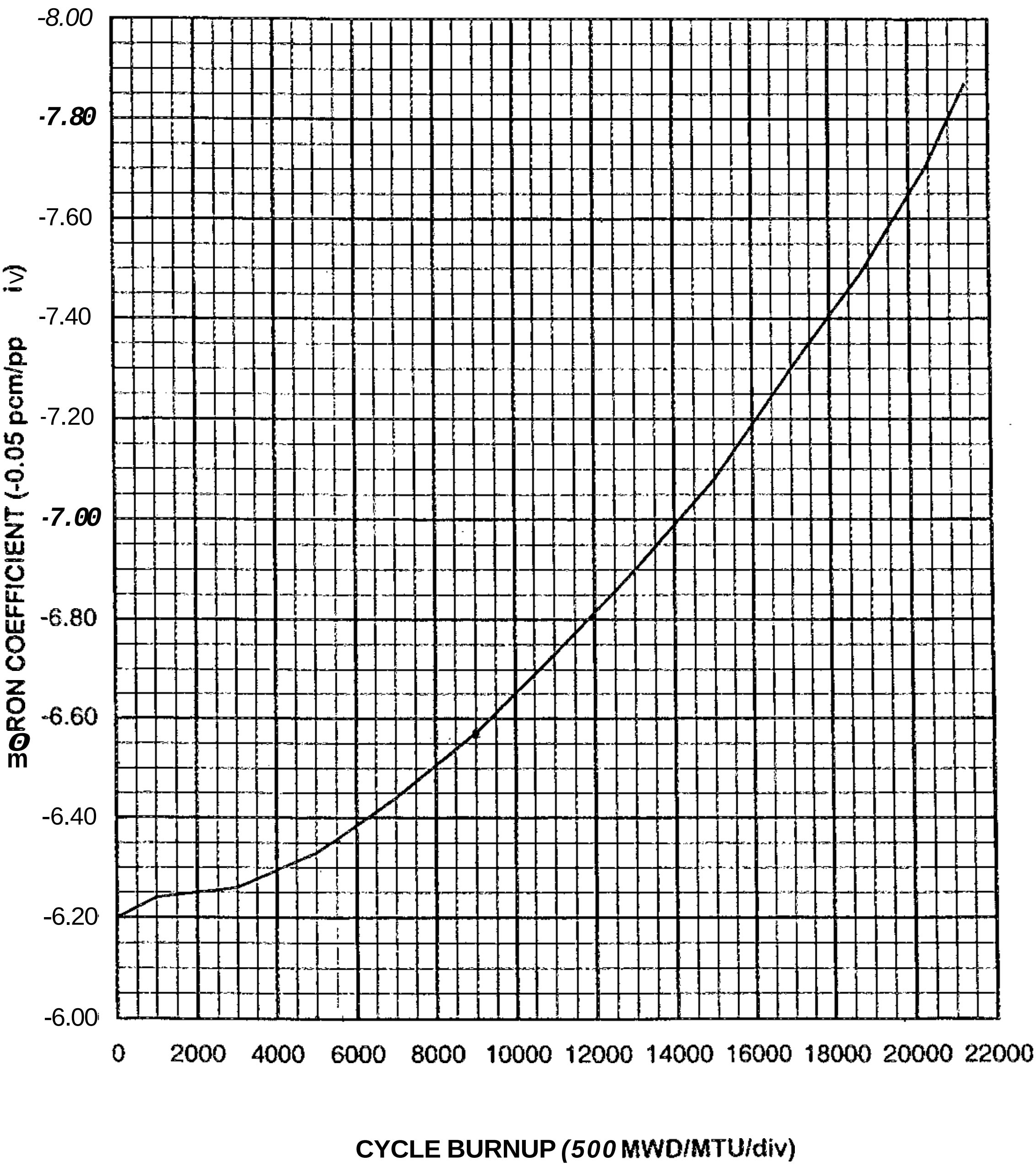
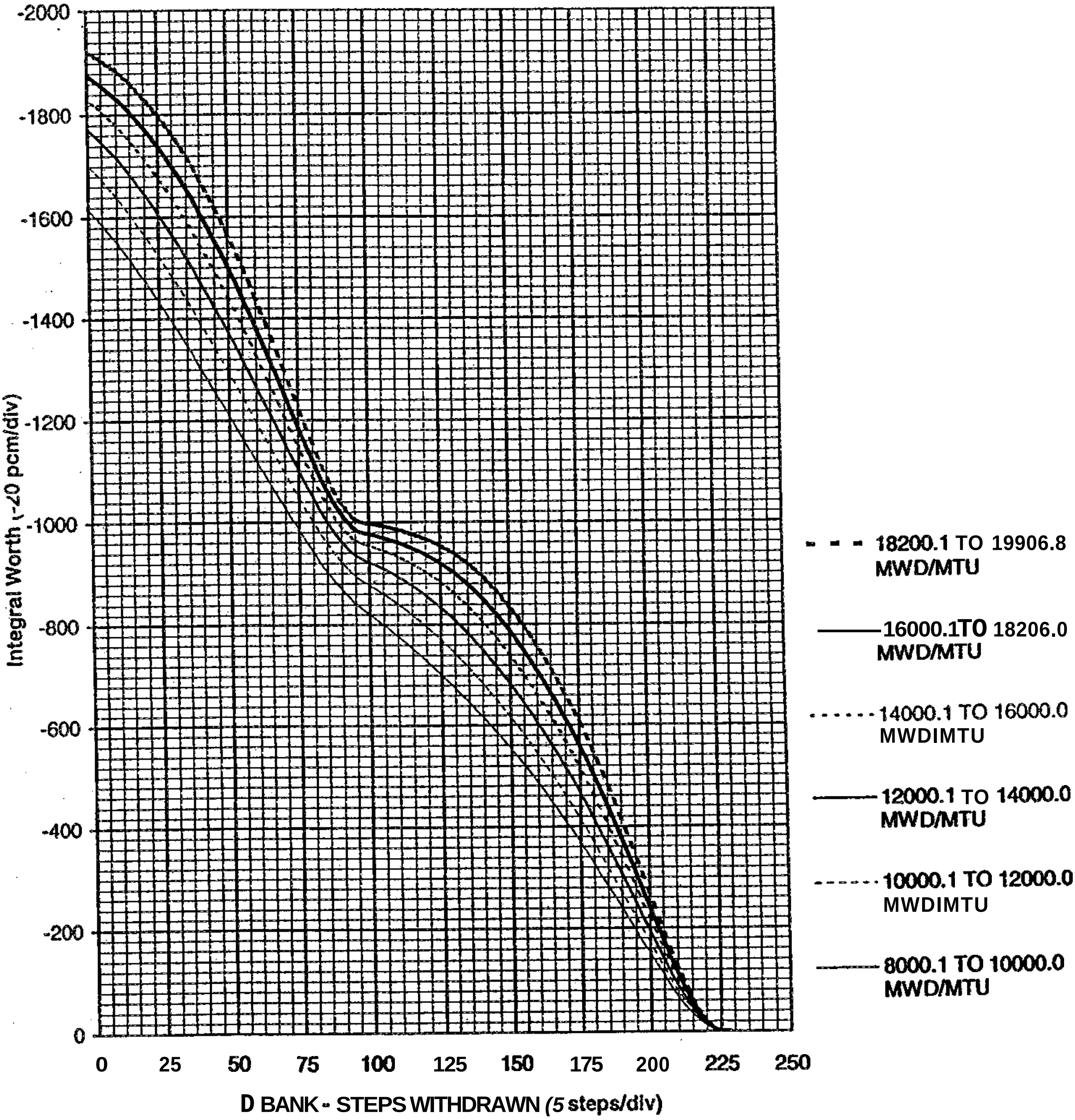
2000	5.991	5.815	5.218	4.975	4.671	4.304	4.080	4.107
2100	6.041	5.862	5.261	5.016	4.710	4.340	4.114	4.141
2200	6.090	5.911	5.304	5.058	4.749	4.376	4.148	4.175
2300	6.141	5.961	5.349	5.100	4.788	4.412	4.182	4.209
2400	6.192	6.011	5.394	5.143	4.829	4.449	4.218	4.245
2500	6.245	6.061	5.439	5.187	4.870	4.487	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

NORTH ANNA UNIT 1 -CYCLE 17
BORON COEFFICIENT Vs. BURNUP
FOR ZERO PERCENT POWER



APPROVED BY W. [signature] DATE 4/3/03

NORTH ANNA UNIT 1 - CYCLE 17
HZP INTEGRAL WORTH PLOT FOR
CONTROL BANKS C AND D IN OVERLAP



APPROVED BY: W. [signature]

DATE 12/19/03

NORTH ANNA UNIT 1 - CYCLE 17
HZP INTEGRAL ROD WORTH TABLE FOR
CONTROL BANKS C AND D IN OVERLAP

		CYCLE BURNUP RANGE (MWD/MTU)					
D-BANK POS STEPS	C-BANK POS STEPS	8000.1 TO 10000.0	10000.1 TO 12000.0	12000.1 TO 14000.0	14000.1 TO 16000.0	16000.1 TO 18200.0	18200.1 TO 19900.0
228	228	0.0	0.0	0.0	0.0	0.0	0.0
224	228	-0.6	-0.9	-1.0	-1.2	-1.3	-1.5
222	228	-4.2	-5.2	-5.7	-6.5	-7.3	-8.4
220	228	-10.4	-12.5	-13.9	-15.8	-17.8	-20.4
218	228	-17.9	-21.3	-23.7	-26.9	-30.2	-34.4
216	228	-28.0	-33.2	-37.3	-42.3	-47.4	-54.0
214	228	-39.6	-46.8	-52.8	-59.9	-67.0	-76.1
212	228	-52.0	-61.3	-69.4	-78.6	-87.9	-99.6
210	228	-65.7	-76.9	-87.3	-98.7	-110.2	-124.6
208	228	-82.3	-94.7	-107.5	-121.4	-135.2	-152.4
206	228	-99.3	-113.0	-128.3	-144.8	-160.9	-181.0
204	228	-116.5	-131.5	-149.4	-168.4	-186.9	-209.8
202	228	-133.4	-150.8	-171.2	-192.8	-213.6	-239.3
200	228	-150.1	-170.5	-193.6	-217.7	-240.9	-269.2
198	228	-166.9	-190.3	-216.1	-242.7	-268.1	-299.1
196	228	-183.8	-210.0	-238.4	-267.5	-295.1	-328.5
194	228	-202.4	-230.2	-261.1	-292.6	-322.3	-357.9
192	228	-221.0	-250.3	-283.8	-317.5	-349.2	-387.0
190	228	-239.4	-270.2	-306.1	-342.1	-375.7	-415.5

NORTH ANNA UNIT 1 - CYCLE 17
HZP INTEGRAL ROD WORTH TABLE FOR CONTROL BANKS C AND D IN OVERLAP

		CYCLE BURNUP RANGE (MWD/MTU)					
D-BANK POS STEPS	C-BANK POS STEPS	8000.1 TO 10000.0	10000.1 TO 12000.0	12000.1 TO 14000.0	14000.1 TO 16000.0	16000.1 TO 18200.0	18200.1 TO 19900.0
188	228	-257.1	-290.0	-328.3	-366.4	-401.8	-443.4
186	228	-274.1	-309.8	-350.4	-390.5	-427.4	-470.7
184	228	-290.9	-329.3	-372.2	-414.1	-452.6	-497.4
182	228	-307.5	-348.6	-393.6	-437.4	-477.3	-523.4
180	228	-325.0	-367.6	-414.6	-460.1	-501.3	-548.5
178	228	-342.6	-386.4	-435.3	-482.4	-524.7	-573.0
176	228	-359.8	-404.8	-455.7	-504.2	-547.6	-596.8
174	228	-376.6	-423.0	-475.6	-525.5	-570.0	-619.9
172	228	-392.3	-440.9	-495.1	-546.3	-591.5	-642.0
170	228	-407.7	-458.4	-514.3	-566.6	-612.5	-663.4
168	228	-422.8	-475.7	-533.1	-586.4	-633.0	-684.2
166	228	-438.4	-492.6	-551.4	-605.6	-652.7	-704.1
164	228	-454.2	-509.2	-569.2	-624.3	-671.8	-723.2
162	228	-469.7	-525.5	-586.7	-642.5	-690.3	-741.6
160	228	-485.0	-541.5	-603.9	-660.3	-708.3	-759.5
158	228	-499.1	-557.1	-620.4	-677.3	-725.4	-776.2
156	228	-512.8	-572.4	-636.6	-693.8	-741.9	-792.3
154	228	-526.4	-587.4	-652.4	-709.9	-757.9	-807.7
152	228	-539.7	-602.2	-667.8	-725.4	-773.2	-822.5

NORTH ANNA UNIT 1 - CYCLE 17
HZP INTEGRAL ROB WORTH TABLE FOR
CONTROL BANKS C AND D IN OVERLAP

		CYCLE BURNUP RANGE (MWD/MTU)					
D-BANK POS STEPS	C-BANK POS STEPS	8000.1 TO 10000.0	10000.1 TO 12000.0	12000.1 TO 14000.0	14000.1 TO 16000.0	16000.1 TO 18200.0	18200.1 TO 19900.0
150	228	-552.9	-616.5	-682.6	-740.3	-787.9	-836.3
148	228	-565.8	-630.6	-697.1	-754.8	-801.9	-849.4
146	228	-578.6	-644.3	-711.3	-768.8	-815.4	-862.0
144	228	-591.1	-657.7	-724.9	-782.1	-828.2	-873.6
142	228	-603.5	-670.9	-738.1	-795.0	-840.2	-884.6
140	228	-615.7	-683.8	-750.9	-807.3	-851.8	-895.0
138	228	-627.7	-696.3	-763.4	-819.2	-862.8	-904.8
136	228	-639.4	-708.5	-775.4	-830.3	-873.0	-913.8
134	228	-651.0	-720.5	-786.9	-841.0	-882.7	-922.2
132	228	-662.5	-732.1	-798.1	-851.2	-892.0	-930.2
130	228	-673.7	-743.4	-808.8	-860.9	-900.6	-937.5
128	228	-684.8	-754.5	-819.1	-870.1	-908.6	-944.1
126	228	-695.7	-765.3	-828.9	-878.8	-916.1	-950.4
124	228	-706.4	-775.7	-838.5	-887.1	-923.2	-956.2
122	228	-716.9	-785.9	-847.5	-894.7	-929.7	-961.4
120	228	-727.2	-795.7	-856.1	-902.0	-935.8	-966.3
118	228	-737.4	-805.2	-864.3	-908.9	-941.4	-970.7
116	228	-747.4	-814.4	-872.2	-915.3	-946.7	-974.8
114	228	-757.2	-823.3	-879.6	-921.2	-951.4	-978.5

NORTH ANNA UNIT 1 - CYCLE 17
HZP INTEGRAL ROD WORTH TABLE FOR
CONTROL BANKS C AND D IN OVERLAP

D-BANK POS STEPS	C-BANK POS STEPS	CYCLE BURNUP RANGE (MWD/MTU)					
		8000.1 TO 10000.0	10000.1 TO 12000.0	12000.1 TO 14000.0	14000.1 TO 16000.0	16000.1 TO 18200.0	18200.1 TO 19900.0
112	228	-766.8	-831.9	-886.6	-926.7	-955.8	-981.9
110	228	-776.2	-840.2	-893.3	-932.0	-959.9	-985.0
108	228	-785.3	-848.1	-899.5	-936.8	-963.7	-987.7
106	228	-794.3	-855.7	-905.4	-941.2	-967.1	-990.2
104	228	-803.1	-863.0	-911.0	-945.3	-970.2	-992.5
102	228	-811.6	-870.1	-916.3	-949.2	-973.1	-994.6
100	228	-819.9	-876.7	-921.1	-952.7	-975.7	-996.5
98	226	-827.9	-883.0	-925.7	-955.9	-978.1	-998.1
96	224	-836.2	-889.8	-930.8	-960.0	-981.5	-1001.1
94	222	-845.9	-898.6	-938.7	-967.6	-989.1	-1009.1
92	220	-857.8	-909.7	-949.3	-978.5	-1000.8	-1021.8
90	218	-868.7	-921.7	-961.1	-991.0	-1014.0	-1036.5
88	216	-88[illegible]6	-935.5	-975.8	-1007.1	-1031.7	-1056.4
86	214	-895.4	-950.5	-992.2	-1025.1	-1051.6	-1078.6
84	212	-909.7	-966.2	-1009.5	-1044.1	-1072.5	-1102.1
82	210	-924.7	-982.7	-1027.7	-1064.4	-1094.8	-1127.0
80	208	-941.5	-1000.7	-1047.9	-1086.9	-1119.5	-1154.4
78	206	-958.5	-1019.1	-1068.6	-1109.9	-1144.9	-1182.6
76	204	-975.8	-1037.7	-1089.5	-1133.2	-1170.4	-1210.8

NORTH ANNA UNIT 1 - CYCLE 17
HZP INTEGRAL ROD WORTH TABLE FOR
CONTROL BANKS C AND D IN OVERLAP

		CYCLE BURNUP RANGE (MWD/MTU)					
D-BANK POS STEPS	C-BANK POS STEPS	8000.1 TO 10000.0	10000.1 TO 12000.0	12000.1 TO 14000.0	14000.1 TO 16000.0	16000.1 TO 18200.0	18200.1 TO 19900.0
74	202	-992.8	-1056.8	-1110.9	-1157.1	-1196.6	-1239.7
72	200	-1009.8	-1076.2	-1132.8	-1181.4	-1223.2	-1268.9
70	198	-1026.9	-1095.7	-1154.7	-1205.7	-1249.7	-1298.1
68	196	-1044.1	-1115.2	-1176.5	-1229.8	-1276.0	-1326.8
66	194	-1062.4	-1135.2	-1198.7	-1254.3	-1302.5	-1355.5
64	192	-1080.8	-1155.1	-1220.7	-1278.5	-1328.6	-1383.8
62	190	-1099.2	-1174.9	-1242.6	-1302.4	-1354.4	-1411.5
60	188	-1117.4	-1194.8	-1264.3	-1326.1	-1379.8	-1438.7
58	186	-1135.2	-1214.6	-1286.1	-1349.6	-1404.8	-1465.2
56	184	-1153.1	-1234.4	-1307.6	-1372.7	-1429.3	-1491.1
54	182	-1171.0	-1254.2	-1328.8	-1395.5	-1453.4	-1516.5
52	180	-1189.6	-1273.8	-1349.9	-1417.9	-1476.8	-1540.9
50	178	-1208.3	-1293.5	-1370.7	-1439.9	-1499.7	-1564.7
48	176	-1226.9	-1312.9	-1391.3	-1461.5	-1522.2	-1587.9
46	174	-1245.4	-1332.2	-1411.6	-1482.7	-1544.1	-1610.4
44	172	-1263.4	-1351.4	-1431.7	-1503.5	-1565.3	-1631.9
42	170	-1281.7	-1370.5	-1451.4	-1523.7	-1586.0	-1652.7
40	168	-1299.9	-1389.3	-1470.8	-1543.6	-1606.1	-1672.9
38	166	-1318.4	-1408.1	-1490.0	-1563.1	-1625.6	-1692.3

NORTH ANNA UNIT I - CYCLE 17
HZP INTEGRAL ROD WORTH TABLE FOR
CONTROL BANKS C AND D IN OVERLAP

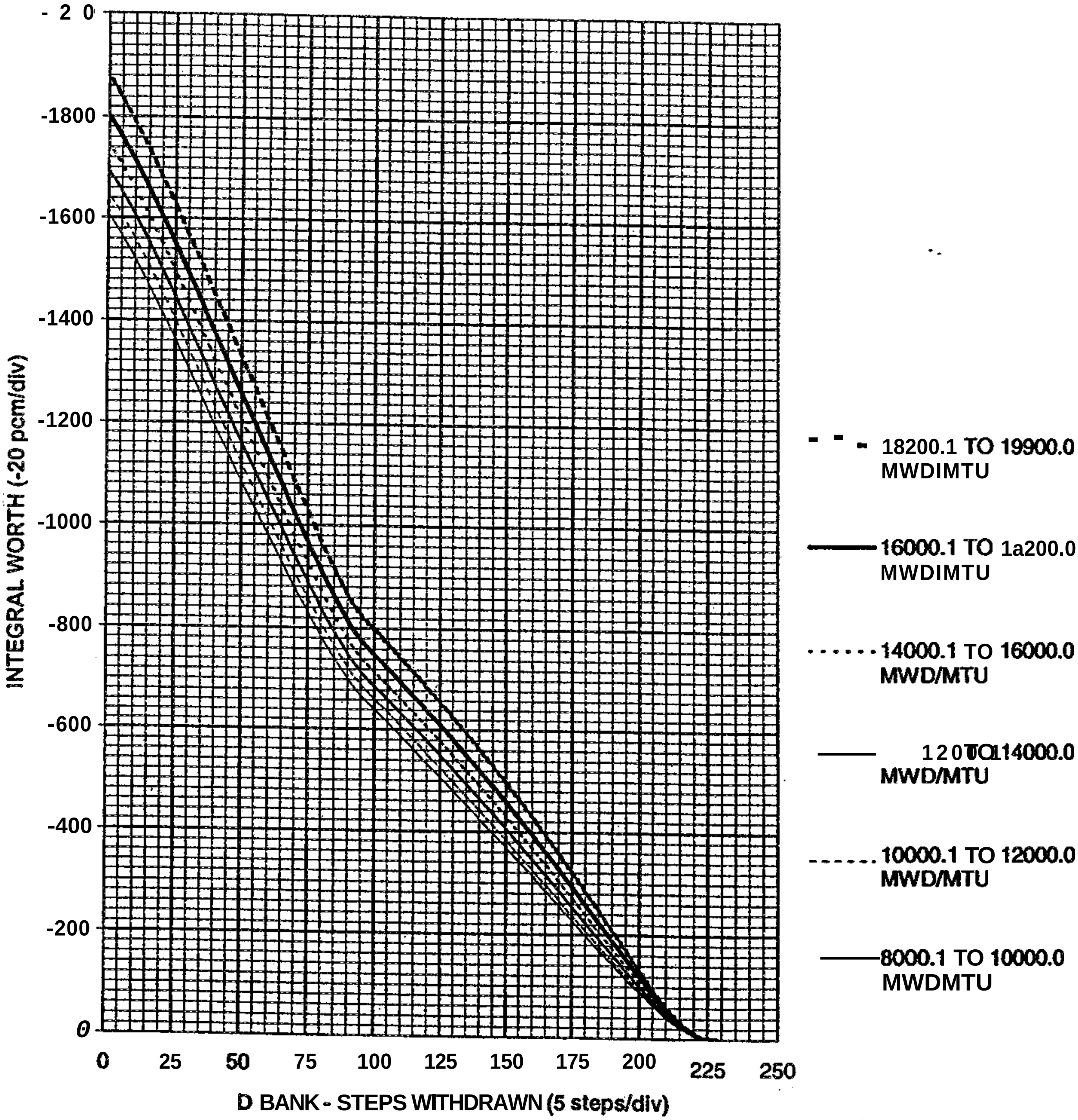
		CYCLE BURNUP RANGE (MWD/MTU)					
D-BANK POS STEPS	C-BANK POS STEPS	8000.1 TO 10000.0	10000.1 TO 12000.0	12000.1 TO 14000.0	14000.1 TO 16000.0	16000.1 TO 18200.0	18200.1 TO 19900.0
36	164	-1336.9	-1426.7	-1508.9	-1582.0	-1644.5	-1710.9
34	162	-1355.1	-1444.9	-1527.4	-1600.5	-1662.9	-1728.9
32	160	-1373.1	-1463.0	-1545.6	-1618.6	-1680.7	-1746.2
30	158	-1390.3	-1480.8	-1563.3	-1636.0	-1697.6	-1762.3
28	156	-1407.6	-1498.2	-1580.6	-1652.9	-1714.0	-1777.8
26	154	-1424.7	-1515.4	-1597.5	-1669.3	-1729.7	-1792.6
24	152	-1441.6	-1532.3	-1614.0	-1685.2	-1744.9	-1806.8
22	150	-1458.2	-1548.8	-1630.0	-1700.4	-1759.2	-1820.0
20	148	-1474.1	-1564.8	-1645.5	-1715.1	-1773.0	-1832.5
18	146	-1489.9	-1580.6	-1660.6	-1729.3	-1786.2	-1844.5
16	144	-1505.3	-1596.0	-1675.2	-1742.8	-1798.6	-1855.6
14	142	-1520.5	-1610.8	-1689.2	-1755.6	-1810.3	-1866.0
12	140	-1535.4	-1625.2	-1702.7	-1767.9	-1821.5	-1875.8
10	138	-1549.9	-1639.3	-1715.7	-1779.8	-1832.1	-1885.0
8	136	-1564.2	-1652.8	-1728.1	-1790.8	-1841.8	-1893.4
6	134	-1577.6	-1665.7	-1739.9	-1801.3	-1851.1	-1901.2
4	132	-1590.6	-1678.2	-1751.2	-1811.2	-1859.8	-1908.6
2	130	-1603.3	-1690.2	-1762.0	-1820.6	-1867.9	-1915.3
0	128	-1615.5	-1701.7	-1772.1	-1829.3	-1875.3	-1921.4

NORTH ANNA UNIT 1 - CYCLE 17
HZP INTEGRAL ROD WORTH TABLE FOR
CONTROL BANKS C AND D IN OVERLAP

		CYCLE BURNUP RANGE (MWD/MTU)					
D-BANK POS STEPS	C-BANK POS STEPS	8000.1 TO 10000.0	10000.1 TO 12000.0	12000.1 TO 14000.0	14000.1 TO 16000.0	16000.1 TO 18200.0	18200.1 TO 19900.0
0	126	-1626.1	-1711.9	-1781.3	-1837.3	-1882.2	-1927.1
0	124	-1636.5	-1721.8	-1790.2	-1844.9	-1888.6	-1932.3
0	122	-1646.7	-1731.3	-1798.5	-1851.9	-1894.5	-1937.0
0	120	-1656.8	-1740.5	-1806.4	-1858.5	-1899.9	-1941.4
0	118	-1666.7	-1749.5	-1814.0	-1864.7	-1905.0	-1945.4
0	116	-1676.5	-1758.1	-1821.3	-1870.5	-1909.7	-1949.0
0	114	-1685.9	-1766.4	-1828.0	-1875.8	-1914.0	-1952.3
0	112	-1695.2	-1774.4	-1834.4	-1880.8	-1917.9	-1955.3
0	110	-1704.3	-1782.1	-1840.4	-1885.5	-1921.6	-1958.1
0	108	-1713.2	-1789.4	-1846.1	-1889.8	-1924.9	-1960.5
0	106	-1721.9	-1796.4	-1851.4	-1893.7	-1927.9	-1962.7
0	104	-1730.3	-1803.1	-1856.4	-1897.4	-1930.7	-1964.8
0	102	-1738.6	-1809.5	-1861.1	-1900.9	-1933.3	-1966.6
0	100	-1746.6	-1815.6	-1865.4	-1904.0	-1935.6	-1968.2
0	98	-1754.3	-1821.3	-1869.5	-1906.8	-1937.7	-1969.7

NORTH ANNA UNIT 1 - CYCLE 17
AT-POWER INTEGRAL WORTH PLOT FOR
CONTROL BANKS C AND D IN OVERLAP

NOTE WORTH AT NOMINAL HFP CONDITIONS



APPROVED BY: A [signature] DATE 12/19/03

NORTH ANNA UNIT 1 - CYCLE 17
AT-POWER INTEGRAL ROD WORTH TABLE FOR CONTROL BANKS C AND D IN OVERLAP

Note: Worth at Nominal HFP Conditions

		CYCLE BURNUP RANGE (MWD/MTU)					
D-BANK POS STEPS	C-BANK POS STEPS	8000.1 TO 10000.0	10000.1 TO 12000.0	12000.1 TO 14000.0	14000.1 TO 16000.0	16000.1 TO 18200.0	18200.1 TO 19900.0
228	228	0.0	0.0	0.0	0.0	0.0	0.0
224	228	-0.4	-0.4	-0.4	-0.5	-0.5	-0.5
222	228	-2.7	-2.9	-3.1	-3.4	-3.8	-4.4
220	228	-6.5	-7.1	-7.5	-8.2	-9.1	-10.5
218	228	-11.1	-12.1	-12.8	-14.0	-15.4	-17.7
216	228	-17.1	-18.5	-19.8	-21.7	-23.9	-27.3
214	228	-23.8	-25.8	-27.7	-30.3	-33.4	-38.1
212	228	-31.0	-33.7	-36.2	-39.6	-43.5	-49.6
210	228	-38.9	-42.1	-45.3	-49.5	-54.4	-61.8
208	228	-48.3	-51.5	-55.5	-60.6	-66.4	-75.3
206	228	-58.1	-61.2	-66.0	-72.0	-78.9	-89.3
204	228	-68.1	-71.3	-76.9	-83.8	-91.7	-103.5
202	228	-78.0	-81.7	-88.1	-95.9	-104.9	-118.2
200	228	-87.9	-92.5	-99.6	-108.5	-118.3	-133.2
198	228	-98.0	-103.4	-111.4	-121.2	-132.0	-148.3
196	228	-108.3	-114.6	-123.5	-134.1	-146.0	-163.6
194	228	-119.4	-125.8	-135.7	-147.1	-159.9	-179.1
192	228	-130.7	-137.3	-147.9	-160.3	-174.0	-194.4
190	228	-142.2	-148.9	-160.3	-173.6	-188.2	-210.0

NORTH ANNA UNIT 1 - CYCLE 17
AT-POWER INTEGRAL ROD WORTH TABLE FOR CONTROL BANKS C AND D IN OVERLAP

Note: **Worth** at Nominal **HFF Conditions**

		CYCLE BURNUP RANGE (MWD/MTU)					
D-BANK POS STEPS	C-BANK POS STEPS	8000.1 TO 10000.0	10000.1 TO 12000.0	12000.1 TO 14000.0	14000.1 TO 16000.0	16000.1 TO 18200.0	18200.1 TO 19900.0
188	228	-153.5	-160.6	-172.7	-186.8	-202.3	-225.4
186	228	-164.3	-172.2	-185.1	-199.9	-216.3	-240.5
184	228	-175.3	-184.0	-197.5	-213.1	-230.4	-255.8
182	228	-186.4	-195.9	-210.2	-226.5	-244.7	-271.1
180	228	-198.0	-207.7	-222.6	-239.6	-258.5	-286.2
178	228	-209.8	-219.5	-235.0	-252.8	-272.4	-301.2
176	228	-221.7	-231.4	-247.6	-266.0	-286.4	-316.2
174	228	-233.6	-243.4	-260.2	-279.3	-300.3	-331.1
172	228	-244.6	-255.1	-272.5	-292.2	-313.9	-345.6
170	228	-255.7	-266.9	-284.9	-305.1	-327.5	-360.1
168	228	-267.0	-278.9	-297.4	-318.1	-341.2	-374.7
166	228	-278.5	-290.6	-309.6	-331.0	-354.5	-389.0
164	228	-290.3	-302.3	-321.8	-343.6	-367.7	-402.9
162	228	-302.2	-314.0	-334.0	-356.3	-381.0	-417.0
160	228	-314.3	-326.0	-346.4	-369.3	-394.6	-431.3
158	228	-325.2	-337.5	-358.3	-381.6	-407.4	-445.0
156	228	-336.3	-349.1	-370.3	-394.1	-420.3	-458.7
154	228	-347.5	-360.8	-382.5	-406.7	-433.4	-472.5
152	228	-358.6	-372.5	-394.6	-419.2	-446.4	-486.2

NORTH ANNA UNIT 1 - CYCLE 17
AT-POWER INTEGRAL ROD WORTH TABLE FOR CONTROL BANKS C AND D IN OVERLAP

Note: Worth at Nominal HFP Conditions

		CYCLE BURNUP RANGE (MWD/MTU)					
D-BANK POS STEPS	C-BANK POS STEPS	8000.1 TO [illegible]	10000.1 TO 12000.0	12000.1 TO [illegible]	14000.1 TO [illegible]	16000.1 TO 18200	18200.1 TO 19900.0
150	228	-369.6	-383.9	-406.3	-431.2	-458.8	-499.3
148	228	-380.6	-395.4	-418.1	-443.4	-471.5	-512.6
146	228	-391.9	407.1	-430.2	-455.9	-484.3	-526.1
144	228	-402.8	-418.5	-441.9	-467.8	-496.6	-538.9
142	228	-413.7	-429.8	-453.3	-479.6	-508.9	-551.6
140	228	-424.8	-441.2	-465.1	-491.7	-521.3	-564.6
138	228	-436.0	-452.8	-476.9	-503.9	-533.8	-577.7
136	228	-446.7	-463.8	-488.2	-515.3	-545.5	-589.9
134	228	-457.6	-475.0	-499.6	-526.9	-557.5	-602.4
132	228	-468.7	-486.5	-511.2	-538.9	-569.7	-614.9
130	228	-479.6	-497.7	-522.5	-550.4	-581.4	-627.3
128	228	-490.4	-508.7	-533.6	-561.8	-593.2	-639.5
126	228	-501.2	-519.9	-545.1	-573.3	-604.9	-651.3
124	228	-512.3	-531.2	-556.5	-585.0	-616.8	-663.9
122	228	-522.9	-542.1	-567.5	-596.1	-628.0	-675.4
120	228	-533.6	-553.0	-578.5	-607.1	-639.3	-687.1
118	228	-544.5	-564.1	-589.7	-618.5	-650.9	-699.0
116	228	-555.4	-575.2	-600.8	-629.8	-662.4	-710.7
114	228	-565.9	-585.8	-611.4	-640.4	-673.0	-721.6

NORTH ANNA UNIT 1 - CYCLE 7
AT-POWER INTEGRAL ROD WORTH TABLE FOR CONTROL BANKS C AND D IN OVERLAP

Note: **Worth at** Nominal HFP Conditions

		CYCLE BURNUP RANGE (MWD/MTU)					
D-BANK POS STEPS	C-BANK POS STEPS	8000.1 TO 10000.0	10000.1 TO 12000.0	12000.1 TO 14000.0	14000.1 TO 16000.0	16000.1 TO 18200.0	18200.1 TO 19900.0
112	228	-576.5	-596.6	-622.2	-651.3	-684.1	-732.8
110	228	-587.4	-607.6	-633.4	-662.6	-695.3	-744.3
108	228	-597.8	-618.3	-643.9	-673.3	-706.1	-755.2
106	228	-608.3	-628.8	-654.5	-683.8	-716.8	-765.9
104	228	-618.9	-639.5	-665.2	-694.5	-727.6	-776.9
102	228	-629.8	-650.5	-676.3	-705.5	-738.6	-788.0
100	228	-640.0	-660.8	-686.4	-715.7	-748.8	-798.3
98	226	-650.5	-671.3	-696.8	-726.1	-759.2	-808.7
96	224	-661.3	-682.4	-707.9	-737.2	-770.4	-820.0
94	222	-673.7	-695.0	-720.6	-750.2	-783.7	-833.9
92	220	-687.1	-708.9	-734.6	-764.7	-798.6	-849.6
90	218	-701.3	-723.7	-749.6	-780.1	-814.6	-866.5
88	216	-716.8	-739.9	-766.3	-797.6	-833.0	-886.0
86	214	-732.6	-756.4	-783.4	-815.4	-851.7	-906.0
84	212	-748.8	-773.3	-801.0	-833.8	-871.0	-926.6
82	210	-765.8	-791.0	-819.4	-853.1	-891.3	-948.1
80	208	-784.3	-809.8	-839.0	-873.5	-912.7	-970.9
78	206	-802.6	-828.4	-858.4	-893.8	-934.0	-993.5
76	204	-821.4	-847.6	-878.4	-914.7	-955.8	-1016.6

NORTH ANNA UNIT 1 - CYCLE 17
AT-POWER INTEGRAL ROD WORTH TABLE FOR CONTROL BANKS C AND D IN OVERLAP

Note: Worth at Nominal HFP Conditions

		CYCLE BURNUP RANGE (MWD/MTU)					
D-BANK POS STEPS	C-BANK POS STEPS	8000.1 TO 10000.0	10000.1 TO 12000.0	12000.1 TO 14000.0	14000.1 TO 16000.0	16000.1 TO 18200.0	18200.1 TO 19900.0
74	202	-840.4	-867.3	-898.9	-936.2	-978.2	-1040.4
72	200	-859.3	-887.2	-919.6	-957.8	-1000.8	-1064.3
70	198	-878.2	-907.2	-940.4	-979.4	-1023.4	-1088.0
68	196	-897.8	-927.7	-961.7	-1001.6	-1046.4	-1112.2
66	194	-918.2	-948.6	-983.4	-1024.1	-1069.8	-1136.8
64	192	-938.4	-969.3	-1004.7	-1046.3	-1092.9	-1160.8
62	190	-959.1	-990.3	-1026.5	-1068.9	-1116.2	-1185.1
60	188	-979.8	-1011.8	-1048.7	-1091.7	-1139.8	-1209.7
58	186	-1000.4	-1033.3	-1070.9	-1114.6	-1163.4	-1234.2
56	184	-1020.7	-1054.7	-1092.9	-1137.3	-1186.7	-1258.1
54	182	-1041.7	-1076.5	-1115.4	-1160.4	-1210.4	-1282.6
52	180	-1063.2	-1098.6	-1138.1	-1183.6	-1234.2	-1307.1
50	178	-1084.6	-1120.3	-1160.3	-1206.4	-1257.5	-1331.1
48	176	-1106.2	-1142.3	-1182.8	-1229.4	-1280.9	-1355.0
46	174	-1128.2	-1164.7	-1205.7	-1252.9	-1304.8	-1379.4
44	172	-1149.9	-1187.2	-1228.8	-1276.1	-1328.5	-1403.6
42	170	-1171.9	-1209.3	-1251.3	-1299.1	-1351.8	-1427.1
40	168	-1194.3	-1231.9	-1274.2	-1322.4	-1375.4	-1451.1
38	166	-1217.0	-1254.5	-1297.3	-1345.9	-1399.1	-1475.1

NORTH ANNA UNIT 1 - CYCLE 17
AT-POWER INTEGRAL ROD WORTH TABLE FOR
CONTROL BANKS C AND D IN OVERLAP

Note: Worth at Nominal HFP **Conditions**

		CYCLE BURNUP RANGE (MWD/MTU)					
D-BANK POS STEPS	C-BANK POS STEPS	8000.1 TO 10000.0	10000.1 TO 12000.0	12000.1 TO 14000.0	14000.1 TO 16000.0	16000.1 TO 18200.0	18200.1 TO 19900.0
36	164	-1239.7	-1277.0	-1320.1	-1369.2	-1422.4	-1498.8
34	162	-1261.9	-1299.5	-1342.9	-1392.2	-1445.8	-1522.3
32	160	-1284.6	-1322.3	-1366.1	-1415.6	-1469.5	-1546.2
30	158	-1306.4	-1344.8	-1388.9	-1438.7	-1492.7	-1569.7
28	156	-1328.6	-1367.1	-1411.5	-1461.4	-1515.5	-1592.6
26	154	-1351.0	-1389.4	-1434.2	-1484.3	-1538.6	-1615.9
24	152	-1373.5	-1411.9	-1456.9	-1507.2	-1561.6	-1639.0
22	150	-1395.6	-1433.9	-1499.0	-1529.6	-1584.1	-1661.5
20	148	-1416.7	-1455.5	-1500.9	-1551.6	-1606.1	-1683.7
18	146	-1438.1	-1477.3	-1522.8	-1573.7	-1628.4	-1706.2
16	144	-1458.9	-1498.6	-1544.2	-1595.3	-1650.1	-1727.7
14	142	-1479.5	-1519.1	-1564.9	-1616.1	-1671.0	-1748.8
12	140	-1500.0	-1539.4	-1585.3	-1636.5	-1691.5	-1769.5
10	138	-1520.4	-1559.6	-1605.6	-1657.0	-1712.0	-1790.2
8	136	-1539.7	-1578.7	-1624.9	-1676.2	-1731.4	-1809.7
6	134	-1557.6	-1596.7	-1643.0	-1694.4	-1749.6	-1828.1
4	132	-1575.1	-1614.4	-1660.7	-1712.5	-1767.7	-1846.5
2	130	-1591.9	-1631.3	-1677.7	-1729.5	-1785.2	-1864.0
0	128	-1607.5	-1647.1	-1693.5	-1745.4	-1800.9	-1880.2

NORTH ANNA UNIT 1 - CYCLE 17
AT-POWER INTEGRAL ROB WORTH TABLE FOR CONTROL BANKS C AND D IN OVERLAP

Note: **Worth at** Nominal **HFP Conditions**

D-BANK POS STEPS	C-BANK POS STEPS	CYCLE BURNUP RANGE (MWD/MTU) 8000.1 TO 10000.0	10000.1 TO 12000.0	12000.1 TO 14000.0	14000.1 TO 16000.0	16000.1 TO 18200.0	18200.1 TO 19900.0
0	126	-1618.9	-1658.7	-1705.2	-1757.3	-1813.2	-1892.7
0	124	-1630.4	-1670.3	-1716.9	-1769.4	-1825.5	-1905.4
0	122	-1641.4	-1681.4	-1728.3	-1780.8	-1837.4	-1917.7
0	120	-1652.5	-1692.5	-1739.5	-1792.2	-1849.0	-1929.7
0	118	-1663.8	-1703.8	-1750.9	-1803.8	-1860.9	-1941.9
0	116	-1675.1	-1715.2	-1762.4	-1815.4	-1872.7	-1954.0
0	114	-1685.9	-1726.0	-1773.2	-1826.4	-1883.7	-1965.2
0	112	-1697.0	-1737.1	-1784.3	-1837.6	-1895.1	-1976.8
0	110	-1708.2	-1748.3	-1795.6	-1849.0	-1906.9	-1988.6
0	108	-1719.2	-1759.2	-1806.5	-1860.0	-1917.8	-1999.9
0	106	-1729.9	-1769.9	-1817.3	-1870.7	-1928.8	-2010.9
0	104	-1741.0	-1780.9	-1828.3	-1881.7	-1940.0	-2022.1
0	102	-1752.2	-1792.1	-1839.5	-1893.0	-1951.3	-2033.6
0	100	-1762.8	-1802.6	-1849.8	-1903.4	-1961.9	-2044.2
0	98	-1773.6	-1813.3	-1860.4	-1914.1	-1972.7	-2055.0

002A1.09 001

The following plant conditions exist.

*Unit 1 *is* at 50% power

•MOL conditions

•A saturated mixed-bed ion exchanger is in service

*Work is being performed on 1-CC-TCV-106, CC Return from NRHX

•The OATC notices that the **output** of 1-CC-TCV-106 has drifted down approximately 30%.

Which ONE of the following describes the plant response to this event?

A?' T_{ave} will increase until rods step in
B. T_{ave} will decrease until rods step out
C. 1-CH-PCV-1145, letdown PCV, will begin to throttle open
D. 1-CH-TCV-1143, letdown IX divert valve, diverts flow around the ion exchangers

A. Correct. This in an inverse acting valve. When the output drifts down the valve is actually opening. When the valve opens the letdown stream begins to cool off. This causes the demins to have a greater affinity for boson. Tave will increase until rods step out, or the TCV position is adjusted.

B. Incorrect. Could be chosen if candidate does not realize valve is reverse-acting.

C. Incorrect. Will not affect the pressure of letdown stream, only the temperature.

D. Incorrect. Will cool off the letdown stream, not heat it up. T_{ave}, though, will increase.

Ability to predict and/or monitor changes in parameters associated with operating the RCS controls including: RCS **T-ave**
(CFR: 41.5:45.5)

Modified bank question 50803

References:
Objective 328 from CVCS study guide

Level (RO/SRO):	RO	Tier:	2
Group:	2	Importance Rating:	3.7/3.8
Type (Bank/Mod/New):	MODIFIED	Cog (Knowledge/Comp):	COMPREHENSION
Reference (Y/N):	N	Last Exam(Y/N):	N

Topic 2.16: Mixed-Bed Demineralizer 328

2.16a. Objective

Explain the following concepts associated with the mixed-bed demineralizer.

- Purpose
- How the addition of positive reactivity is prevented when placing the demineralizer in service (SOER-94-2)
- How a change in letdown temperature causes a reactivity increase or decrease (OE-8275-I, CTS-02-97-2158-002, DR-N-97-0016)
- How a demineralizer is verified to contain resin prior to placing it in service

2.16b. Content

1. The mixed-bed demineralizer has three purposes:
   1.1. Purify the reactor coolant.
   1.2. Reduce reactor coolant fission product activity.
   1.3. Assist in maintaining RCS pH.

2. Operators must us6 care to prevent the addition of positive reactivity when placing the demineralizer in service (SOER-94-2) which could most likely occur at one of the following times:
   2.1. Following resin replacement when the mixed bed resin will remove some boron from the letdown flow stream, or
   2.2. Following a refueling outage.
   2.3. The RCS boron concentration is high following refueling and the old mixed bed IX (which is stili full of the boron-free letdown water from the previous cycle's coastdown) is placed in service for the initial clean-up of the RCS.

3. If boration of the mixed bed demineralizer to be placed in sewice is required. letdown flow is passed through the demineralizer and diverted directly to the Boron Recovery System.

3.1. When two samples indicate that the demineralizer is at nearly the same boron concentration as the primary system, letdown system flow is directed to the VCT.

4. A change In letdown temperature can also cause a change in reactivity (OE 8275 I; CTS 02-97-2158802; DR-N-97-0016).

4.1. The amount of boron held on the anion resin beads Is dependent upon the temperature of the resin.

4.2. As the temperature changes, the amount of boron that can be held on the beads changes, such that:

4.2.1. when temperature decreases, the anion beads are capable of holding more boron and will remove it from the fluid passing through the bed and in effect, a dilution takes place

4.2.2. when temperature increases, the anion beads cannot hold as much boron and it is rejected into the fluid passing through the bed and a boration takes place

5. Here at North Anna Unit 2, reactor power and temperature increased when 2-CC-TCV-206 (non-regenerative heat exchanger) was manually failed open.

5.1. This controller had been operating erratically and was manually failed open with the downstream isolation valve throttled to allow maintenance on the positioner.

5.2. The component cooling flow through the heat exchanger increased which cooled letdown temperature from 110°F to approximately 80°f.

5.3. Soon, reactor power and RCS Tavg was noted to have slightly increased. Control rods were manually inserted to control reactor power and temperature.

5.4. Work was completed and 2-CC-TCV-206 was returned to automatic. As ietdown temperature returned to 110°F, the control rods were withdrawn for control.

6. At Sizewell B in June of 1996, a performance test was required foliowing cleaning of a CC heat exchanger required full service water flow to be established causing CC temperature to decrease.

6.1. The decreased CC temperature caused CVCS letdown temperature to exceed the capability of the letdown temperature control valve for the non-regenerative heat exchanger to maintain and letdown temperature decreased from 83°F to 68°F.

6.2. The mixed bed demineralizer began to remove boron from the letdown flow unit a higher saturation point was reached, resulting in a dilution of the primary system.

6.3. RCS temperature increased and the Rod Control System responded to restore RCS temperature.

6.4. When the test was complete, service water flow was returned to normal causing letdown temperature to increase and the effect within the mixed bed demineralizer was reversed, boron was released from the resin beads to the letdown flow causing a RCS boration and temperature decrease.

6.5. Again control rods responded to restore RCS temperature to normal.

7. Operators should avoid any actions which may cause letdown temperature to change significantly to prevent unanticipated reactivity transients such as described above. (OE 8275 I)

8. A mixed-bed demineralizer is verified to contain resin prior to placing it in service by checking the Demin Status program on the Equipment Status System (VPESS).

Topic 2.17: Cation-Bed Demineralizer 330

2.17a. Objective

Explain the following concepts associated with the cation-bed demineralizer.

- Purpose
- How the addition of positive reactivity is prevented when placing the demineralizer in service (SOER-94-2)
- How a demineralizer is verified to contain resin prior to placing it in service

Original Bank Question

1 **ID: 50803** **Points: 1.00**

The following plant conditions exist.

- Unit is at 50% power
- MOL conditions

A saturated mixed-bed ion exchanger is in service

1-CC-TCV-106, CC return from NRHX, drifts 30% in the closed direction. Letdown temperature stabilizes at 130°F. Which ONE of the following plant responses is expected?

A. Rods step out

B. Rods step in

C. 2-CH-PCV-1145, letdown PCV, throttles open

D. 1-CH-TCV-1143, letdown IX divert valve, diverts flow around the ion exchangers

Answer: A

003AK3.09 001

Unit 2 is at 45% power when a control bank rod drops into the core. The reactor did not trip. The procedure for recovering this rod requires that the dropped rod's group step counter reading be recorded.

Which ONE of the following states the reason for this requirement?

A! Phis allows the operator *to* determine final position of the dropped rod at the end recovery.
B. This documents that the rod insertion limits have not been violated during recove
C. This allows the bank overlap unit to be reset to its proper value after recovery.
D. This allows the operator to determine rod withdrawal rate in accordance with Attachment 3, " Calculation of Maximum Rod Withdrawal Rate."

A. Correct. The operator will have to withdraw the dropped rod back to the position of all the other sods in that bank. All other rods in that bank will have their lift coils disconnected. This will allow only the dropped rod to move. The step counter is repositioned to zero and the P/A Converter tracks its movement. The operator uses the recorded step counter reading to know where *to* place the dropped rod before reconnecting all the other rods in that bank.

B. Incorrect. Step 6 of 1-AP-1.2 has the operator verify Rod Insertion Limits met. This is done before the step counter is taken to zero, therefore there is no need to record step counter numbers while performing this step. This step does need to be done per 1-AP-1.2 and the operator does need to know rod heights. This makes this answer plausible.

C. Incorrect. The Bank Overlap function is not affected because individual bank select is used to withdraw control rods. This answer is plausible. Bank overlap will have to be reset if an entire bank did not move when demanded. Attachment V to verify proper bank overlap is performed in 1-AP-1.3, " Control Rod Out of Alignment."

D. Incorrect. Rod withdrawal rate using Attachment 3 will be performed but the withdrawal rate is based on time since the rod dropped. This is plausible due to Attachment 3 needing to be performed. Examinee may figure localized xenon effects are the basis for limiting withdrawal rate. Xenon is a function of power. The examinee may relate this function to rod height.

Knowledge of the reasons for the following responses as they apply Dropped Control Rod:

Recording of group bank position for dropped rod (reference point used to withdraw dropped rod to equal height with other rods in the bank)

(CFR: 41.5 / 41.10 / 45.6 / 45.13)

North **Anna bank question 460**

References:
Objective 15168 from **Abnormal Procedures study guide**

Level (RO/SRO):	RO	**Tier:**	1
Group:	**2**	**Importance Rating:**	3.0/3.5
Type (Bank/Mod/New):	BANK	**Cog** (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	**N**	Last Exam(Y/N):	**N**

...

Topic 1.3.6: Retrieving a Dropped Rod 15168

1.3.6a. Objective

Explain the following concepts concerning dropped rod retrieval in accordance with 1-AP-1.2, "Dropped Rod" (I-At'-I .2).

- Why the reading on the affected group's group step counter is recorded.
- Why the affected banks pulse-to-analog converter is set to zero.
- How proper group 1 – group 2 sequencing is established.
- Which main control board annunciators are expected to alarm, including when each alarm clears.

1.3.6b. Content

1. Prior to retrieving a dropped rod, the group step counter reading for the affected group is recorded.

   1.1. This is done because the affected group step counter will be manually reset to zero prior to rod retrieval, and the operator will need to refer to this value to determine the final position of the dropped rod.

2. Prior to retrieving a dropped rod, the pulse-to-analog (P/A) converter for the affected bank is manually pulsed down to zero using controls in the instrument rack room.

   2.1, The P/A converter must be set to zero because, as the dropped rod is retrieved, the converter will count upward.

3. When the dropped rod is retrieved it is important that proper group 1 – group 2 sequencing be established.

   3.1. The master cycler reversible counter in the rod control system tracks which group was the last group to get a "move' command, and which group should be the next group to get a "move" command.

3.1.1. If a group 1 rod is being retrieved, then the rod should be withdrawn to one step greater than the desired endpoint, then inserted one step.

3.1.2. If a group 2 rod is being retrieved, then the rod should be withdrawn **to,** but not beyond. then desired endpoint.

4. Several annunciators are expected to alarm during the retrieval of a dropped rod.

4.1. The RPI ROD BOT ROD DROP alarm will have alarmed when the rod dropped.

4.1.1. The alarm will clear during retrieval **as** the rod is withdrawn above the rod bottom bistable.

4.2. The affected bank's ROD BANK LO / LO-LO LIMIT will alarm when the PIA converter is pulsed down below the rod insertion limits for the current power level.

4.2.1. The alarm will clear during red retrieval. as the rod is withdrawn above the insertion limits.

4.3. The ROD CONTROL URGENT FAILURE will be received when dropped rod retrieval begins.

4.3.1. The rod control system senses the feedback that a "move" command was sent to the non-affected group, but no rods moved in that group because its lift coil disconnect switches are open.

4.3.2. The alarm will clear when the operator pushes the ALARM RESET pushbutton on the main control board following retrieval of the dropped rod.

1.4: Misaligned Control Rod

Topic 1.4.1: Information Associated with 1-AP-1.3 11028

1.4.1a. Qbjective

List the following information associated with **1-AP-1.3,** "Control Rod Out of Alignment" (SOER-84-2).

- Purpose of the procedure
- Modes of applicability

003GG2.2.22 001

Unit **1** *is* in mode 4
RHR is in service
The following RCS conditions exist:

Loop Cold Leg Temperatures		Steam Generator Temperatures	
Loop A	285°F	SG A	345°F
Loop B	228°F	SG B	250°F
Loop C	269°F	SG C	319°F

Based on the above parameters, which ONE of the following lists the RCPs that can be started?

A. None of the reactor coolant pumps
B. "A" reactor coolant pump
C. "A" "B" and "C" reactor coolant pumps
D. All of the reactor coolant pumps

← Makes it more plausible since it is the loop with temp < 235°F.

A. Correct. Since loop B cold leg temperature is <235, each SG temperature would have to be **less** than or equal to 50 degrees above each cold leg temperature.

B. Incorrect. Candidates may pick this answer if they think the limit **is** > 50 degrees, vice **less** than or equal to, and do not realize that it applies to each SG and cold leg.

C. Incorrect. Candidates may pick this answer if they think it is permissible to start pumps in the loops that are **less** than or equal to 50 degrees below the respective cold leg temperatures.

D. Incorrect. Candidates could pick this answer if they think they 50 degree limit only applies to the loop that has a cold leg temperature that is less than or equal to 235. "B" loop meeds the 50 degree criteria.

RCPs
Knowledge of limiting conditions for operations and safety limits

Modified North Anna bank question 60116

References:
Steam tables (Candidates will have steam tables as a reference.)
Objective 3528 from study guide on RCS
Tech Spec 3.4.6

Level (RO/SRO):	RO	Tier:	2
Group:	1	Importance Rating:	3.4/4.1
Type (Bank/Mod/New):	MODIFIED	Cog (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	*Y*	Last Exam(Y/N):	N

Topic 1.4.4: Reactor Coolant System Technical Specifications Various 3528

1.4.4a. Objective

Explain the following concepts associated with Reactor Coolant System Technical Specifications

- Why the loop stop valve must be maintained open and deenergized in modes 1, 2, 3 & 4 (TS-3.4.17)
- Eases for the limits on primary-to-secondary leakage in mode 1 above 50% power (TRM-3.4.4, SG Monitoring Prog.)
- Why a reactor coolant pump may not be started with Reactor Coolant System (RCS) cold-leg temperatures ≤ 235°F (290°F) on unit 1 (2) unless the secondary temperature of the steam generators is < 50°F above each of the RCS cold-leg temperatures (TS 3.4.6 Bases)
- Why only one charging pump and low-head safety injection pump may be operable when the Reactor Coolant System is < 235°F (270°F) on unit 1 (2) (TS-3.4.12 Bases)

1.4.4b. Content

1. TS-3.4.17 bases for maintaining loop stop valve breakers open with power removed in modes 1, 2, 3 & 4 is to prevent inadvertent closure of a loop stop valve during unit operation.

   1.1. This ensures all reactor coolant loops remain in operation and maintains the DNBR above the design limit during all normal operations and anticipated transients.

2. The bases for limiting primary-to-secondary leakage above 50% power are as follows:

   2.1. These limits ensure that, if a fatigue-induced crack occurs in one or more S/Gs, the resulting leak will be detected in sufficient time to conduct an orderly shutdown prior to catastrophic tube failure.

   2.2. The limits on an increase in leakage of 60 gpd between surveillance intervals and for an increasing trend indicating that 100 gpd would be exceeded in 90 minutes ensure that, if a fatigue-induced crack occurs, power can be reduced to a level below which the crack will not propagate.

3. A reactor coolant pump may not be started when one or more RCS cold-leg temperatures is ≤ 235°F (290°F on unit 2) unless the secondary temperature of the steam generaton is < 50°F above each of the

Topic 1.4.4: Reactor Coolant System Technical Specifications Various 3528

1.4.4a. Objective

Explain the following concepts associated with Reactor Coolant System Technical Specifications.

- Why the loop stop valve must be maintained open and deenergized in modes 1, 2, 3 & 4 (TS-3.4.17)
- Bases for the limits on primary-to-secondary leakage in mode 1 above 50% power (TRM-3.4.4, SG Monitoring Prog.)
- Why a reactor codant pump may not be started with Reactor Coolant System (RCS) cold-leg temperatures ≤ 235°F (270°F) on unit 1 (2) unless the secondary temperature of the steam generators is < 50°F above each of the RCS cold-leg temperatures (TS 3.4.6 Bases)
- Why only one charging pump and low-head safety injection pump may be operable when the Reactor Coolant System is < 235°F (270°F) on unit 1 (2) (TS-3.4.12 Bases)

1.4.4b. Content

1. TS-3.4.17 bases for maintaining loop stop valve breakers open with power removed in modes I, *2,* 3 & 4 is to prevent inadvertent closure of a loop stop valve during unit operation.

   1.1. This ensures all reactor coolant loops remain in operation and maintains the DNBR above the design limit during all normal operations and anticipated transients.

2. The bases for limiting primary-to-secondary leakage above 50% power are as follows:

   2.1. These limits ensure that, if a fatigue-induced crack occurs in one or more S/Gs, the resulting leak will be detected in sufficient time to conduct an orderly shutdown prior to catastrophic tube failure.

   2.2. The limits on an increase in leakage of 60 gpd between Surveillance intervals and fer an increasing trend indicating that 100 gpd would be exceeded in 90 minutes ensure that, if a fatigue-induced crack occurs, power can be reduced to *a level* below which the crack will *not* propagate.

3. A reactor coolant pump may not be started when one or more RCS cold-leg temperatures is ≤ 235°F (270°F on unit 2) unless the secondary temperature of the steam generators is < 50°F above each of the

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops—MODE 4

LCO 3.4.6 Two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation.

------------ NOTE ------------

1. All reactor coolant pumps (RCPs) and RHR pumps may be removed from operation for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. No RCP shall be started with any RCS cold leg temperature ≤ 235°F (Unit 1), 270°F (Unit 2) unless the secondary side water temperature of each steam generator (SG) is ≤ 50°F above each of the RCS cold leg temperatures.

APPLICABILITY: MODE 4.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required loop inoperable.	A.1 Initiate action to restore a second loop	Immediately
	AND	
	A.2 ---------NOTE--------- Only required if RHR loop is OPERABLE. ---------------------- Be in MODE 5.	24 hours

Unit 1 is in the final stages of recovering from a **loss** of station service power. The **loss** of power occurred while an Reactor Coolant System (RCS) cooldown **was** in progress. Presently, all reactor coolant pumps are off, pressurizer level is 61%, and all main steam trip valves are closed. Power has been restored to all plant electrical busses, and it is desired to **start** the "C" reactor coolant pump to continue the plant cooldown.

LOOP COLD LEG TEMPERATURES:	SG PRESSURES:
Loop A = 285°F	A = 90 PSIG
Loop E = 228°F	E = 35 PSIG
Loop C = 269°F	C = 60 PSIG

Based on the above conditions, it is not permissible to start the " C reactor coolant pump because __________

A. "B" loop cold leg temperature is **less** than 235°F and "A" steam generator temperature is >50°F above "B" loop cold leg temperature

E. auctioneered low cold leg temperature must be within 5°F of "C" cold leg temperature

C. RCS pressure **is** less than 280 PSIG as indicated by saturation pressure for "C" loop cold leg temperature

D. auctioneered low cold leg temperature must **be** within 20°F of " C cold leg temperature to justify technical specification requirements

Answer: A

004K2.04 001

Unit 1 has dust restored power to the 1H 4160V bus. Operators are currently restoring power to associated MCCs. The following annunciators are currently lit:

B-C5 BAT 1A HI TEMP CH 1-11
B-C6 BAT 1A LO TEMP CH 1-11

Which **ONE** of the following describes the actions(s) required to clear these annunciators?

A. Press the reset button on the heater's power supply breaker
B. Adjust setpoint on local temperature controllers
C. Take the heater's breaker control switch to PTL then back to AUTO
D. Cycle the heater's power supply breaker to OFF then back to **ON**

A. Correct. When the BAST heaters lose power both the HI and LO temperature annunciators will come in. The heaters must be reset using the local reset button on the heater's power supply breaker.

B. Incorrect. There are TICs located in the auxiliary building near the tanks. These are only adjusted by instrument techs for ICPs. Operators take logs on these once a day. **A** candidate could choose this answer if he/she doesn't realize that adjustment of the TIC will not clear the HI and LO temperature alarms, the undetvoltage signal must be reset.

C. Incorrect. Pressurizer heaters can be reset using this method. **A** charging pump breaker Lockout annunciator is also reset using this method. There are no control switches for the BAST heaters. **A** candidate could choose this answer based on knowledge of how other equipment is reset.

D. Incorrect. The reset button is located on the breaker, turning the breaker OFF then back ON will not reset the heater UV on the breaker. **A** candidate could choose this answer is he/she remembers that the reset involves the breaker, but does not remember that a reset button must be pressed.

Knowledge of **electrical** power supplies to **the** following: BWST tank **heaters**

(CFR: **41.7)**

New question

References:
Attachment 3 of AP-10
Annunciator response **for** B-C6

Level (RO/SRO):	RO	Tier:	2
Group:	1	Importance Rating:	2.6/2.7
Type (Bank/Mod/New):	**NEW**	Cog (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	N	Last Exam(Y/N):	N

004K2.04

NUMBER 0-AP-10	ATTACHMENT TITLE **DEGRADED AND UNDERVOLTAGE EQUIPMENT ACTUATION**	REVISION 44
ATTACHMENT 3		PAGE 1 of 7

THE FOLLOWING EQUIPMENT WILL TRIP DURING A DEGRADED OR UNDERVOLTAGE CONDITION ON 1H 4160-VOLT EMERGENCY BUS:

- 1-CC-P-1A
- 1-CH-P-1C (H Bus) (Will trip if 1-CH-P-1A running)
- 1-CH-E-6A, BAST Htr (manually reset 1H1-2N C2) ←
- 1-CW-P-2B
- 1-FC-P-1A
- 1-FW-P-3A
- 1-HV-1-1A
- 1-HV-F-1C (H Bus)
- 1-HV-F-37A
- 1-HV-F-37B
- 1-HV-F-37C
- 1-HV-F-80A (manually reset in SWVH)
- 1-HV-UH-80B (manually reset in SWVH)
- 1-QS-P-1A
- 1-RH-P-1A
- 1-RS-P-1A
- 1-RS-P-2A
- 1-RS-P-3A
- 1-SI-TK-2A Boron Injection Tank Htr
- 1-SW-P-1A

BAT 1A LO TEMP CH I - II

136°F

1.0 Probable Cause

1.1 Loss of power to heaters
1.2 Lmproper setting or operation of temperature controller

2.0 Operator Action

2.1 Locally verify setting and operation of temperature controllers, 1-CH-TIC-1107 and 1-CH-TIC-1162.

2.2 IF high temp and low temp annunciators are both in alarm, THEN check breakers 1-EE-BKR-1J1-2N B3 and 1-EE-BKR-1H1-2N 62 and reset UV pushbutton at breakers if required.

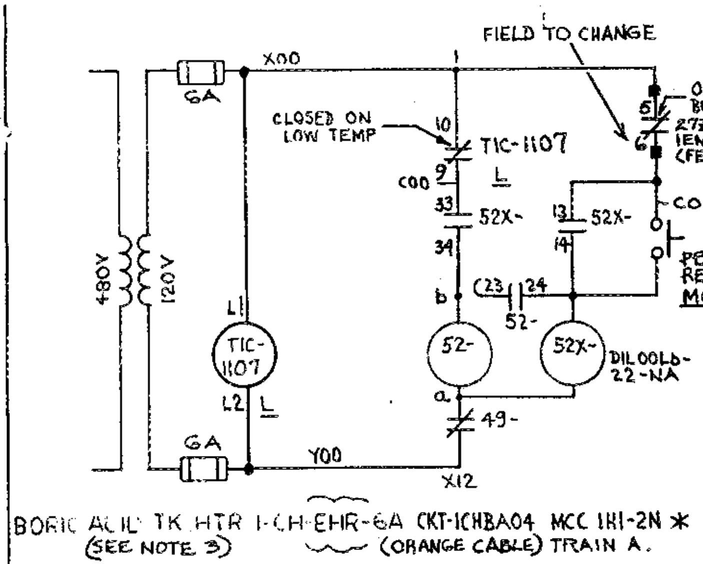
3.0 References

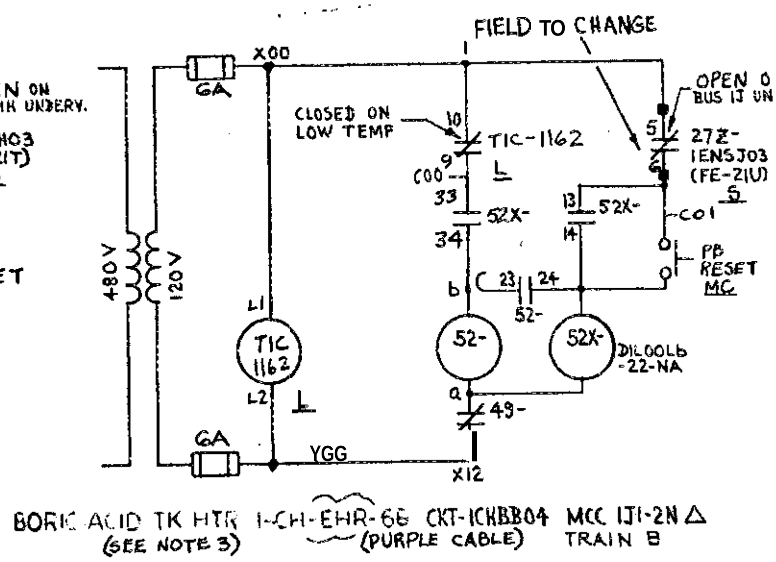
3.1 T.S. 3.1.2.7 and 3.1.2.8 (ITS TR 3.1.1 and 3.1.2)
3.2 Unit 1 Loop Book, pages CH-44 and 45

4.0 Actuation

4.1 1-CH-TIC-1107 and/or 1162

** WARNING: THIS IS ONLY A PARTIAL SECTION OF ENTIRE DRAWING. **





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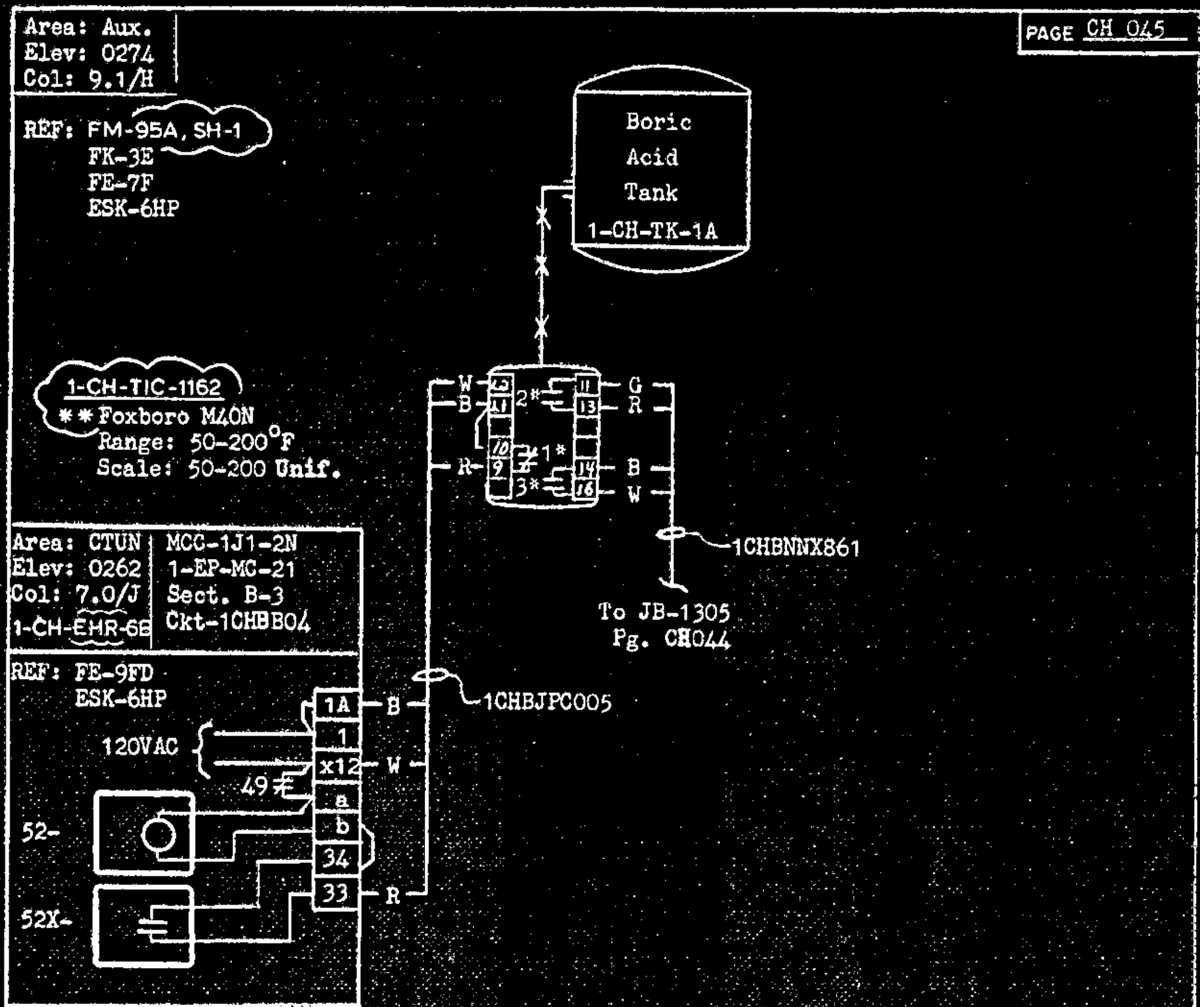
PAGE CH 045



* LOOP ACTION:

1-CH-TIC-1162 measures the temperature in the boric acid tank and contact 1 closes on temp. below low setpoint to start heater provided the bus undervoltage has been reset. Temperature above high setpoint shuts off heater.
Contact 2 opens on further decrease in temperature below low-low setpoint to activate low alarm Channel II.
Contact 3 opens on increasing temperature above high-high setpoint to activate high alarm Channel II. (see pg. CH044).

** FOR REFERENCE ONLY

3 DUR 93-871T

REVISION 3	BY	DATE	TITLE CHEMICAL & VOLUME CONTROL SYSTEM BORIC ACID TANK (1-CH-TK-1A) HEATER (1-CH-EHR-6B) CONTROL AND TEMPERATURE ALARMS CHANNEL II	PROJECT NAPS No. 1
PREPARED	DJS	—		J.O. NO 11715
CHECKED	LBA	—		
APPROVED	RAS	9/16/93		

REVISIONS BY	4	5	6	7	8	NUMBER T-1162
DATE						

THIS DRAWING SUPERSEDES REV 2

- RHR is in service maintaining a stable RCS temperature
- I-CC-MOV-1008 and 1-CC-MOV-10OB (CC to the RHR heat exchangers) are throttled open approximately *5%*
- The RHR flow control valve (1-RH-FCV-1605) is controlling in automatic
- An instrument technician inadvertently isolates air to the RHR heat exchanger outle

A! decrease; decrease
B. increase; increase
C. decrease; increase
D. increase; decrease

A. Correct. 1-RH-HCV-1758 fails open. This will cause full flow through the heat exchanger(s). RHR outlet temperature will decrease. In automatic, 1-RH-FCV-1605 output will decrease to throttle the valve closed in order to control RHR flow at the setpoint.

B. Incorrect. 1-RH-FCV-1605 fails closed. The candidate could mistakenly think that 1-RH-HCV-1758 also fails closed. This assumption would make this answer correct.

C. Incorrect. In automatic, 1-RH-FCV-1605 output will decrease to throttle the valve closed in order to control RHR flow at the setpoint.

D. Incorrect. RHR temperature will decrease since 1758 fails open. In automatic, 1-RH-FCV-1605 output will decrease to throttle the valve closed in order to control RHR flow at the setpoint. Candidate could mistakenly think that 1758 failed closed and that 1605 is a reverse acting valve.

Ability to **manually operate and/or monitor in the control** room: **Heat exchanger bypass** flow **control**

(CFR: **41.7 / 45.5 to 45.8)**

New Question

References:
NCRODP module 40 – Residual Heat Removal (pages 8, 17-18)
Attachment *to 0-AP-10*
Loop Book RH-005

Level (RO/SRO):	RO	Tier:	*2*
Group:	1	Importance Rating:	3.4/3.1
Type (Bank/Mod/New):	**NEW**	Cog (Knowledge/Comp):	COMPREHENSION
Reference (Y/N):	N	Last **Exam(Y/N):**	N

A miniflow recirculation line is provided to ensure that the pump does not overheat or vibrate when the system discharge line is isolated. A locked-open manual isolation valve **(RH-48)** is used to establish the design flow through the line. The design flow rate is 500 gpm; however, in pre-operational testing the estimated flow through the line with the valve full open (eight turns) was only 200 gpm, which is considered satisfactory.

Because the **RHR** pumps have **a** common recirculation line, operating both pumps simultaneously on recirculation may result in one pump operating at shutoff head. Additionally. to preclude a reduction in the pump operating lifetime, the time that an **RHR** pump is on recirculation should be minimized.

The common discharge line of the pumps is equipped with a temperature element (TE-1604) which monitors the inlet temperature to the heat exchangers. The line also has a pressure transmitter **(PIC-1662)** which is used to annunciate a high discharge pressure alarm in the MCR. Before the coolant reaches the heat exchangers there is a **3/8-inch** line which is used to sample the **RHR** coolant at the pump discharge.

Heat Exchangers. The **RHR** pump discharge is directed to two identical heat exchangers arranged in parallel. Each heat exchanger is designed to remove one-half the design heat load of the system (30.5 x **10^6 Btu/hr)**. Each heat exchanger is a shell and tube type, two-pass heat exchanger. Reactor coolant flows through the 559 tubes, and component cooling water fluws through the shell side. The heat exchanger tubes and tubesheet serve as the physical boundary between the two fluid flows, as well as the heat transfer area **(4070 ft^2).**

Each heat exchanger is located on the **RHR** Rat, vertically mounted. and is approximately **16.5** feet tall. The RHR side of the heat exchanger has two drain lines with isolation valves which allow the operator to drain the tube side for maintenance. The heat exchanger has an inlet (RH-19) and an outlet **(RH-24)** isolation valve which allow the operator to remove the heat exchanger from service.

Flow Control Valves. The outlet flow from the heat exchangers is controlled by an air-operated butterfly valve **(HCV-1758).** The valve is physically located on the **WHR** flat and is used to control the rate of change of **RHR** temperature during heatup ***or*** cooldown. The valve position is varied in response to manually controlled signals from the control located in the **MCR.** The valve fails In the full open position.

The total flow rate for the RHR System is controlled by flow control valve **FCV-1605.** The valve controls the heat exchanger bypass flow and is similar in construction to **HCV-1758.** except that **FCV-1605** fails in the shut position. This valve operates in conjunction with a manual-automatic (M/A) station and the output of flow transmitter **FT-1605.** Design Change 88-01-1 relocated the manual/auto controller for **FCV-1605** from the vertical section of the main control board to the benchboard section. It is now located below the station alarm control switch. The controller's new location allows the operator to remain within his normal area while operating the valve manually. The breaker. The breaker also opens automatically due to undervoltage. If automatic breaker opening occurs, it is indicated in the MCR by:

1. the RHR PUMP 1A AUTO **TRIP** alarm (window 1E-A6),
2. the RHR PUMP 1B AUTO TRIP alarm (window 1E-A7), and/or
3. the amber light above the respective pump handswitch.

Flow Control Valves. The RHR coolant flow is controlled by two valves: a) HCV-1758 and b) FCV-1605. The flow through the heat exchangers is controlled by HCV-1758. The valve receives position signals from a manual control station located on benchboard 1-1 in the MCR. By varying the position of **HCV-1758,** the operator varies the rate of change of RHR coolant temperatures. For example. if the operator wants to increase the RHR cooldown rate, he throttles open HVC-1758 using the manual control station. Opening the valve Increases the flow through the heat exchanger. increasing the heat removal from the system. The valve fails in the full open position, ensuring heat exchanger flow in the event of a loss of power or operating air to the valve operator.

Flow control valve FCV-1605 controls the total Row rate through the RHR System. An M/A station controller, located on Section 1-2 of the benchboard beneath the station alarm controller, is used to establish the system flow. In the automatic mode of operation the valve is controlled by the output of flow transmitter FT-1605 and the setpoint established on the M/A station. The flow transmitter provides a system Row signal to primary plant processing cabinet 6. Thls signal provides MCR flow indication and a low flow alarm as previously described. The transmitter output is also sent to flow controller FC-1605B. The controller compares the flow signal against the flow setpoint and provides an output signal to an electro-pneumatic (E/P) converter. The converter modulates the operating air to the valve operator to control the valve position. This valve is designed to fail in the fully shut position.

The manual portion of the controller tracks the automatic signal allowing a bumpless transfer from automatic to manual. In the manual mode the operator uses the raise and lower pushbuttons to position the valve to achieve the desired flow as indicated on FI-1605 at vertical board 1-2. Bumpless transfer is also achieved when transferring from manual to automatic. After a restoration of power to the cabinet following a loss of power, the system is In manual control with the valve control signal calling for the fully shut position.

The operation of HCV-1758 and FCV-1605 are directly associated with each other although there are no circuit connections. Assume that FCV-1605 is set to maintain 4000 gpm of system flow and HCV-1758 is positioned to allow 1000 gpm of heat exchanger flow. The flow rates within the system would be:

1. 1000 gpm through HCV-1758 and the heat exchangers.
2. 3200 gpm through FCV-1605, and

3. 200 gpm through the miniflow circulation line.

There is 4200 gpm from the combined discharge of HCV-1758 and FCV-1605; however, 200 gpm is returned to the suction of the pump. The remaining flow, 4000 gpm, passes out of the system discharge and is monitored by FT-1605. If the operator increases the flow through the heat exchangers, FCV-1605 automatically throttles shut to maintain system flow at 4000 gpm.

The isolation valve (HCV-1142) between the RHR System and CVCS is controlled by a manual control station on benchboard 1-1. The valve is opened to allow letdown flow to the CVCS when RHR is in operation. Flow to the CVCS is controlled by HCV-1142 and PCV-1145 in the CVCS which controls inlet pressure to the non-regenerative heat exchanger. The valve (HCV-1142) fails shut on a loss of operating air or power. The General System Operation section provides more detail on the operation of this valve.

Outlet Isolation Valves. The outlet isolation valves (MOV-1720A, -1720B) are controlled by a two-position (OPEN. CLOSE) handswitch. Their operation is identical to the operation of the inlet isolation valves, except there is no open permissive interlock.

Component Cooling Instrumentation, Alarms, and Controls

Instrumentation and Alarms. The outlet temperature from each heat exchanger (see Figure 40-2-NA) is monitwed by temperature elements which provide indication on vertical board 1-2 and provide an input to an annunciator. The inlet flow to each heat exchanger is also monitored by a flow transmitter which provides indication on the same vertical board as well as an input to the same annunciator. The monitoring devices are:

1. heat exchanger 1A - TE-149A and FT-132A; and
2. heat exchanger 1B - TE-149B and FT-32B.

These four devices ail input to the RHR HEAT EXCHANGER CC OUTLET HIGH TEMPERATURE HIGH FLOW alarm (window 1E-A5). The alarm annunciates if:

1. either temperature elements indicates greater than 150°F, or
2. either flow transmitter indicates greater than 9080 gpm.

The outlet temperature from each seal cooler is monitored by temperature elements which provide computer data logs. The outlet flow from each seal cooler is also monitored by a flow switch which provides indication on vertical board 1-2 as well as an input to an annunciator. The monitoring devices are:

1. seal cooler 1A - TE-150A and FS-131A; and
2. seal cooler 1B - TE-150B and FS-131B.

The flow switches annunciate the following alarms if component cooling flow to the seal cooler fails below 5 gpm:

1. Flow switch FS-131A annunciates the RHR PUMP 1A COOLING WATER LOW FLOW alarm (window E-B5).

<u>NOTE</u>: The following valves in Containment will close on loss of air.

1-SI-HCV-1851A, B, & C	Makeup to Accumulator A. B. & C
1-SI-HCV-1850A & B	Accumulator Test A
1-SI-HCV-1850C & D	Accumulator Test B
1-SI-HCV-1850E & F	Accumulator Test C
1 SI-HCV-1852A, B, & C	Accumulator Drains A. B, & C
1-SI-TV-1842	Accumulator Test Isolations
1-BD-TV-100B & G	Steam Generator "A" Blowdown
1-BD-TV-100D & H	Steam Generator "B" Blowdown
1-BD-TV-100F & J	Steam Generator "C" Blowdown

<u>NOTE</u>: The following valves in Containment will open on loss of air.

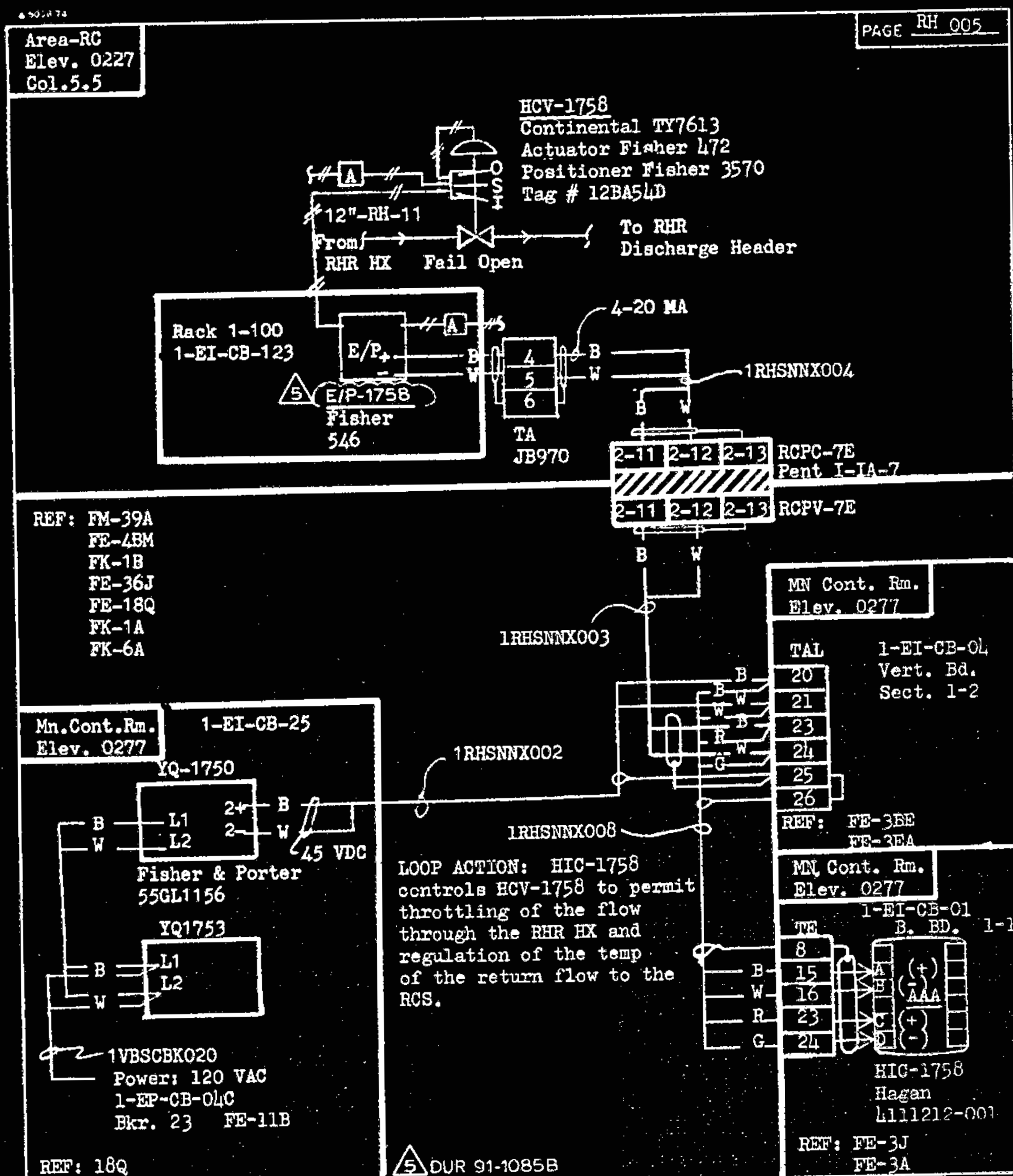
1-CC-TV-108A & B	NS TK HX CC Inlet
1-CC-TV-107A & B	NS TK HX CC Outlet
1-RC-HCV-1544	Vessel Flange Leakoff
1-RH-HCV-1758	RH Hx Outlet
1-CH-HCV-1310	Charging to Loop "B"
1-CH-HCV-1303A, B, & C	RCP Seal Leakoff A. B, & C
1-CH-HCV-1389	Excess Letdown 3-Way Valve Fails to VCT (Seal Water Return)

<u>NOTE</u>: The following valves in the AFPH will open on loss of air and depletion of the seismic air flasks.

1-FW-HCV-100A	AFW HCV Header to A SG
1 FW-HCV-100B	AFW HCV Header to B SG
1-FW-HCV-100C	AFW HCV Header to C SG
1-FW-PCV-159A	AFW Pumps to MOV Hdr Pressure Control Valve
1-FW-PCV-159B	APW Pumps to HCV Hdr Pressure Control Valve

11715

RH-005



REVISION 5			TITLE	PROJECT: NAPS NO. 1
	BY	DATE	RESIDUAL HEAT REMOVAL SYSTEM RHR EXCHANGERS OUTLET FLOW CONTROL	J. O. NO. 11715
PREPARED	WKU	7/17/92		
CHECKED	LBR	7/21/92		NUMBER: HCV-1758
APPROVED	JCL	7/23/92		

REVISION BY	6	7	8	9	10
DATE					

THIS DRAWING SUPERSEDES REV. 4 ORIGINAL

005K1.09 001

Which ONE of the following would be an effect of placing the Residual Heat Removal System in service when its-boron concentration is lower than that of the Reactor Coolant System?

A. A loss of shutdown margin could occur.
B. The decay heat load could increase due to the lower suspended solids in the system.
C. Radiation levels in the RHR system would decrease.
D. RHR system heat exchangers could become fouled due to the sudden pH change.

A. Correct. If the RHR boron concentration is lower than that of the RCS then placing the RHR system in service will allow this water to mix with the RCS, lowering the RCS boron concentration. The new, lower **RCS** boron concentration may not be high enough to meet shutdown margin criteria.

B. Incorrect. The examinee may think lower suspended solids could lead to less neutron absorption from subcritical multiplication leading to an increased decay heat load.

6. Incorrect. The examinee may assume since RHR is at a lower boron concentration that it was **last** in service earlier in core life. Radiation levels in **the** RCS tend to increase over core life due to leaking fuel.

D. Incorrect. The high boron concentration will raise the pH of the RHR system, but this will not cause fouling in the heat exchangers.

Knowledge of the physical connections and/or cause-effect relationships between RHR system and the following: RCS

(CFR: 41.2 to 41.9 / 45.7 to 45.8)

North Anna bank question 2004

References:
1-OP-14.1

Level (RO/SRO):	RO	Tier:	2
Group:	2	Importance Rating:	3.6/3.9
Type (Bank/Mod/New):	BANK	Cog (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	N	Last Exam(Y/N):	

VIRGINIA POWER NORTH ANNA POWER STATION	PROCEDURE NO: 1-OP-14.1	
	UNIT NO: 1	REVISION NO: 49
PROCEDURE TYPE: OPERATING	EFFECTIVE DATE: ON FILE	EXPIRATION DATE: N/A

PROCEDURE TITLE

RESIDUAL MEAT REMOVAL

ITS				EOP AP		

REVISION SUMMARY:

- Incorporated the following for ITS Implementation:
  Changed **ITS TRM 3.7.9** to ITS None in Step 2.2.6 and **4.20 -** 5th Bullet due to deletion of ITS TRM 3.7.9.

Writer: J. Goerge / S. Lacey	**Reviewer:** D. Hawkins

ELECTRONIC DISTRIBUTION — APPROVAL ON FILE

PROBLEMS ENCOUNTERED: ☐ *Yes* ☐ No **NOTE: If yes, note problems in Remarks.**

REMARKS:

____

____

____

____

____ *(use back for additional space)*

SHIFT SUPERVISOR:	DATE:

TABLE OF CONTENTS

1.0 PURPOSE

1.1 To provide instructions for placing RHR in operation.

1.2 To provide instructions for removing RHR from operation.

1.3 To provide instructions for removing A RHR HX from service.

1.4 To provide instructions for removing B RHR HX from service

1.5 To provide instructions far restoring A RHR HX to service.

1.6 To provide instructions for restoring B RHR HX to service.

1.7 To provide instructions for returning RHR to operation following draindown of RHR.

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, as follows:

Section 5.1 places the RHR system in service. System valves are manipulated for startup. A Calculation is performed to determine the boron concentration of the RHR system including the volume of the Accumulator stagnant discharge legs. The calculation assumes the RHR\Accumulator stagnant discharge legs are at the lowest Boron concentration that existed at any time during current fuel cycle.

If at least one RCP is running and the RHR boron concentration is less than that of the RCS, then a calculation is performed to borate the RCS to compensate for the lower RHR boron concentration. When the RHR system is placed in service, with the RCPs running, adequate mixing of the contents of the diluted RHR\Accumulator discharge stagnant legs will occur before reaching the Reactor Core. The diluted RHR\Accumulator discharge stagnant legs can occur from lower RCS boron concentration during operation, if RCS leakage is into the Accumulators. When the RCS is borated for shutdown, then these diluted stagnant legs will remain.

If no RCPs are running and the RHR boron concentration is **less** than the that ofthe RCS, then adequate mixing will NOT occur. The latest 1-PT-10, Shutdown Margin Determination is addressed. If the calculation for **RHR** boron concentration with the volume of the Accumulator stagnant discharge legs is less than shutdown margin determination, then RHR system must be equalized with the RCS. The RHR system boron concentration is equalized using Attachment I by flowing the RCS through RHR using letdown to the Gas Stripper **until** the RHR reaches the RCS boron concentration for cold shutdown based on the shutdown margin determination. The blender or the RWST is used for RCS makeup.

The RHR system is then placed in service flowing one RHR discharge at a time to ensure adequate RHR\Accumulator discharge stagnant leg flushing. Only one discharge MOV is opened at a **time** to ensure adequate flow is maintained to prevent flow induced chatter of discharge check valves.

Section ***5.2*** removes the RHR system from service. The **RHR** letdown isolation is **left** 10% open to compensate for thermal expansion and contraction. To ensure the disc of 1-CH-HCV-1142, RHR Letdown Isol Valve, is off the seat prior to establishing the final position, the valve is initially opened to 50% then positioned at 10%. The system discharges are closed, then the RHR pumps are secured. The system suctions are closed and system valves ***are*** manipulated for the secured alignment.

Sections **5.3** and **5.4** remove individual RHR Heat Exchangers from service by adjusting RHR flow, then isolating the RHR inlet valve. The **RHR** Heat Exchanger CC outlet is then isolated.

Sections 5.5 and 5.6 restore individual **RHR** Heat Exchangers **to** service by throttling RHR Heat Exchanger CC outlet, as required based on **RCS** temperature. The RHR inlet valve **is** unisolated, RHR flow is adjusted as required, and CC is throttled to maintain temperature.

Section **5.7 returns** the RHR system to service following drainage of the RIIR system. The Reactor vessel level is ensured filled adequately to 74 inches above centerline and an RCS makeup source is verified. The RHR inlet isolation is throttled opened based on RCS make **up** rate and the system is thoroughly vented. When the **RCS** is at 74 inches, then RCS make up is secured and **the** RHR inlet and outlet isolations arc fully opened. The RHR system is thoroughly vented. The system valves arc aligned, an RIIR pump is started and **flow is** adjusted. Only one discharge MOV is **left** open to ensure adequate flow *is* maintained to prevent flow induced chatter of discharge check valves. RHR Periodic Tests are then performed, as required.

2.0 REFERENCES

2.1 Source Documents

2.1.1 Letter from **M.** L. Bowling to NRC, dated **3-29-93,** concerning additional information **for** TS **change** deleting RIIR Suction MOV auto-close.

2.2 Technical Specifications

2.2.1 Unit 1 Tech Spec **3.1.1.3.1 (ITS** TRM **3.1.4)**

2.2.2 Unit 1 Tech Spec **3.4.1.3** (ITS **3.4.6, 3.4.7,** 3.4.8, TRM **3.4.7)**

2.2.3 Unit 1 Tech **Spec 3.4.9.1** (ITS **3.4.3)**

2.2.4 Unit 1 Tech Spec **3.9.8.2** (ITS **3.9.6)**

2.2.5 Unit I Tech Spec 3.7.9.1 (ITS TRM 3.7.8)

2.2.6 Unit 1 **Tech** Spec **3.7.9.2** (ITS None)

2.2.7 Unit I Tech Spec **3.9.8.1** (ITS **3.9.5)**

2.2.8 Unit 1 Tech Spec 3.7.4.1 **(ITS3.7.8)**

2.2.9 Unit **2** Tech Spec **3.7.4.1** (ITS Delete **-** Common with Unit I)

2.3 Technical References

2.3.1 UFSAR Section 5, Reactor Coolant System

2.3.2 NAPS **PLS** Document

2.3.3 IEB 88-04, Potential Safety Related **Pump** Loss

2.3.4 Westinghouse VRA/VGB-SU 5.2.2, RHR Functional Test

2.3.5 **11715-FM-79A, CC**

2.3.6 **11715-FM-93A,** RCS

2.3.7 **11715-FM-94A,** RHR

2.3.8 **11715-FM-95C,** CVCS

2.3.9 1-OP-3.3, Unit Shutdown from Mode 4 to Mode **5**

2.3.10 1-OP-8.3, Boron Concentration Control

2.3.11 1-OP-14A, Residual Heat Removal

2.3.12 **I-OP-51.1,** Component Cooling System

2.3.13 1-PT-10, Shutdown Margin Determination

2.3.14 1-PT-78.1, Residual Heat Removal System-Monthly

2.3.15 I-PT-78.3, Residual Heat Removal Pump and Valves Test

2.3.16 **1-PT-212.12,** Valve Inservice Inspection (RHR Control Valves)

2.3.17 1-PT-213.15, Valve Inservice Inspection (Misc)

2.3.18 1-PT-213.16, Valve Inservice Inspection (CC Common Loads Check Valves)

2.3.19 NCRODP-40, Residual Heat Removal System

2.3.20 Westinghouse/Ingersoll-Rand Residual Heat Removal Pump Technical Manual

2.3.21 NA-DW 1082H41, SSPS

2.3.22 0-OP-49.1, Service Water System Normal Operation

2.3.23 1-GOP-13.0, Alternate Core Cooling Method Assessment

2.3.24 This procedure is referenced by the following Emergency/Abnormal Procedures:

a. 1-ES-0.2A, Natural Circulation Cooidown with Shroud Cooling Fans

b. 1-ES-0.2B, Natural Circulation Cooldown without Shroud Cooling Fans

c. 1-ES-0.3, Natural Circulation Cooldown with Steam Void in Vessel (with RVLIS)

d. 1-ES-0.4, Natural Circulation Cooldown with Steam Void in Vessel (without RVLIS)

e. I-ES-3.1, Post-SGTR Cooldown using Backfill

f. 1-ES-3.2, Post-SGTR Cooldown using Blowdown

g. 1-ES-3.3, Post-SGTR Cooldown using Steam Dump

h. I-ECA-2.1, Uncontrolled Depressurization of NE Steam Generators

i. 1-ECA-3.3, SGTR without Pressurizer Pressure Control

j. 1-FR-H.1, Response to Loss of Secondary Heat Sink

k. 1-AP-16, Increasing Primary Plant Leakage

2.3.25 0-OP-26.9, 4160 Volt Breaker Operation

2.3.26 SER 11-87, Water Hammer in Component Cooling System, Alternate Core Cooling Method Assessment, Dated 4-23-87

2.3.27 Memorandum, BK Day - JW Daily, 1/26/96, Revising Procedures for Isolating SW to the RSHXs when RS *is* Secured (Revision 39-P1)

2.3.28 Memorandum, BK Day - JW Daily, 2/27/96, E-PAR to OP-14.1 (Revision 39-PI)

2.3.29 DCP 94-10. SW Piping Replacement

2.3.30 ET No. 97-013, Rev 0 Addressing SI Accumulator Sampling Techniques

2.3.31 Memorandum from R. L. Stevens to P. Bradley, Addressing Calculation Of Shutdown Margin DR N97-494 (Revision 42)

2.3.32 ET No. 97-053, Rev. 0 Addressing Calculation of Shutdown Margin for Placing RHR in Service

2.4 **Commitment Documents**

2.4.1 NE Technical Report No. 865 Rev I, Background and Guidance for Ensuring Adequate Backup Decay Heat Removal Following Loss of RHR

2.4.2 ME Technical Report No. 681 Rev **1,** Safety Evaluation **of** Reduced **Flow** Rate for the North Anna Residual Heat Removal System

2.4.3 Memo **from** K. L. Basehore to M. **D.** Crist, dated 02-16-93, Compensation **for** Coolant Temperatures Less **than** 70°F During Refueling Operations

2.4.4 CTS 02-94-1812-003, License Amendment 189, Tech Spec change No. 287, Heat **Up** and Cooldown operating limits, PORV lift setpoints, LTOPS enable temperature, PORV and PORV Block Valve operating requirements. Changes the following (old) Tech Specs: 3.1.2.2, 3.1.2.4, 3.4.1.3, 3.4.3, 3.4.3.2, Figures 3.4-2 and 3.4-3, 3.4.9.3, 3.5.2, and 3.5.3.

2.4.5 EP 95-1006, Incorporate *SO* 213 into Procedures

2.4.6 Engineering Transmittal No. CEE 96-081, Rev. 0, Addressing Residual Heat Removal Minimum Flow Requirements

2.4.7 DR N-96-2846, Addressing B RHR HX CC Outlet Temperature Inaccuracy With Low CC Flows

2.4.8 DR N-97-1760, Hi pressure letdown relief leaking by approx. 4 gpm

2.4.9 DCP 98-110, Relocation of RHR HX CC Outlet Temperature Element NAPS Unit 1

2.4.10 DCP 98-107, Component Cooling Water Containment Return Cross Tie NAPS Unit 1

2.4.11 Plant Issue N-2000-1881, Addressing RHR CC Return Line Temperature Stress (See Revision 46)

3.0 INITIAL CONDITIONS

3.1 Review the equipment status to verify station configuration supports the performance of this procedure.

3.2 Unit 1 is in Mode 4, 5, 6, or defueled.

3.3 RCS pressure is **less than 418** psig.

3.4 RCS temperature is less than 350°F.

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 any other step **is** marked N/A, THEN have the Shift Supervisor **(or** designee) approve the N/A AND justify the N/A on the Procedure Cover Sheet.

4.2 To avoid thermal shock, flow through the RHR Loop must be initiated slowly. A warmup period of 5 minutes at 50-200 gpm is required before increasing RHR flow.

4.3 To maximize pump life, the time RHR Pumps are on recirc must be minimized,

4.4 IF RCS temperature is above 140°F and NO RCPs *are* in operation, THEN RHR MUST be in operation.

PERFORM ONLY AFTER ITS IMPLEMENTATION

IF ITS Implemented, THEN:

4.5 IF in Mode 5 with any RCS Loops filled, OR in Mode 6 with water **level** ≥ 23 ft above Reactor Vessel flange, THEN at least one RHR Loop MUST be OPERABLE and in operation. IF in Mode 6 with water **level < 23** ft above Reactor Vessel flange, THEN the second RHR Loop must also be OPERABLE.

4.6 **Observe** the following concerning RHR flow rate:

4.6.1 WHEN in Mode 4 or 5 with RHR in operation, OR in Mode 6 with RCS water level **< 23** ft above Reactor Vessel flange:

- IF RCS temperature is at least 140' F OR the Reactor has been shutdown for **less** than 100 hours, THEN RHR flow must be at least 3000 gpm.
- IF RCS temperature **is less** than 140° **F** AND the Reactor has been shutdown **for** at least 100 hours, THEN RIIR flow MUST **be** > 2500 **gpm** to ***ANY*** RCS Loop receiving RHR discharge flow, to prevent potential damage to the Accumulator and RHR discharge check valves, due to flow induced chatter. **(References 2.4.2 and 2.4.6)**

4.6.2 IF in Mode 6 with RCS water Bevel ≥ 23 ft above Reactor Vessel flange, THEN RHR flow must be at least 3000 gpm.

PERFORM ONLY AFTER ITS IMPLEMENTATION

PERFORM With CURRENT, UNIMPROVED TECH SPECS ONLY

IF Current, Unimproved Tech Specs, **THEN:**

4.4 Observe the fallowing concerning RHR flow rate:

- IF RCS temperature is at least 140' F OR the Reactor has been shutdown for less *than* 100 hours, THEN RHR flow must be at least 3000 gpm.
- IF RCS temperature is less than 140' F AND the Reactor has been shutdown for at least 100 hours, THEN RHR flow MUST be > 2500 gpm **to** ANY RCS loop receiving RHR discharge flow, to prevent potential damage to the Accumulator and RHR discharge check valves, due to flow induced chatter. (References **2.4.2** and **2.4.6**)

PERFORM With CURRENT, UNIMPROVED TECH SPECS ONLY

4.8 RHR Pump Motor **Start** Duty is as follows:

- Cold or Hot – one immediate restart.
- Subsequent restarts – After minimum run-time of 15 minutes, *as* needed.
- Subsequent restarts from idle – 45 minutes apart.

4.9 To minimize the potential for loss *of* RHR flow, the RHR Suction MOVs should be de-energized when fuel is in the vessel and all RCPs are secured.

4.10 **1-RH-RV-1721A,** RHR Suction Relief, and **1-RH-RV-1721B,** RHR Suction Relief, WILL lift at 467 psig. RHR pressure must not be allowed to exceed 467 psig. (Reference **2.1.1)**

4.11 IF the RCS is solid **AND** an RHR Pump will be started or stopped, THEN place 1-CB-PCV-1145, LETDOWN PRESSURE CONTROL VALVE, **in** MANUAL.

4.12 RIIR flow in excess of 4000 gpm could result in RHR **Pump** runout.

4.13 Operating both RHR Pumps on recirc at the same time may result in deadheading one **pump.**

4.14 **IF** Reactor Coolant temperature will be **less** than 70°F, THEN the Reactor Engineer MUST be contacted for determination of Shutdown Margin. (Reference **2.4.3)**

4.15 RCS cool down rates MUST be limited to 50°F/hr. (Reference **2.4.4)**

4.16 WHEN filling from the blender, THEN monitor blender **Boric** Acid and PG flows to ensure that the flows do not change.

4.17 WHEN filling RHR, THEN the PDTT and Containment **sump** levels MUST be monitored to ensure that no excessive **leakage** exists.

4.18 WHEN the RHR System is placed in service, THEN an Operator in Containment should walk down the RHR **System** and monitor for leakage.

4.19 Coordination with Health Physics should **be** made to establish Containment integrity as required.

4.20 The following Tech Specs apply:

- Tech Spec **3.1.1.3.1,** Reactivity Control Systems, Boron Dilution, Reactor Coolant Flow (ITS TRM **3.1.4)**
- Tech Spec **3.4.1.3,** Reactor Coolant System Shutdown — Shutdown (ITS **3.4.6, 3.4.7, 3.4.8, TRM 3.4.7)**
- Tech Spec **3.4.9.1,** Reactor Coolant System Pressure and Temperature Limits (ITS **3.4.3)**
- Tech Spec **3.7.9.1,** RMR — Operating (ITS TRM **3.7.8)**
- Tech Spec **3.7.9.2,** RHR — Shutdown **(ITS** None)
- Tech **Spec** 3.9.8.1, RHR, Refueling, Normal Water Level (ITS 3.9.5)
- Tech Spec 3.9.8.2, RHR, Refueling, Low Water Level (ITS **3.9.6)**

4.21 IF less than four SW pumps are running when either Unit has accident conditions, THEN the following precautions apply in order to avoid SW pump cavitation or runout problems during evolutions that increase demand on the SW system:

- Discharge pressures for running SW Pumps must be maintained greater than 43 psig (monitored in the Control Room).
- IF SW Reservoir level drops below 313' elevation, THEN SW Pumps must be monitored closely, and SW flows adjusted as appropriate.
- In order to maintain SW pump discharge pressures above 43 psig, throttling of SW flows to CCHXs or other loads may be necessary.

4.22 WHEN all below conditions are met, THEN a third Service Water Pump must he aligned to the Service Water Header which is supplying the CC Heat Exchangers:

- The Service Water 49-day Action Statement *is* in effect for CCHX Service Water piping repair/replacement.
- The **Unit** is required to be placed in Mode 5.
- Service Water supply temperature exceeds 78.5°F.

4.23 Action Statements **of** numerous Tech Specs require suspension of all activities involving "positive reactivity changes" or "reduction in RCS boron concentration". Flowing water into **the** RCS **with** a iower boron concentration than that **of** the RCS is considered a positive reactivity change and will result in reduction in RCS boron concentration, therefore must not **be** done when prohibited by any Tech Spec Action Statement.

4.24 IF a CC Containment Trip Valve on the **RHR** Outlet lines goes shut, THEN **the** associated RHR HX MOV must be closed to prevent losing the **differential** pressure across the **PDTT** cooler, RHR Pump Seal **Coolers,** Excess Letdown **HX** and NST Coolers.

4.25 **To** avoid possible CC **steam** formation, thermal shock and return line temperature stress of the RHR **HX, 1-CC-TI-149A,** HTX RTN TMP, **and 1-CC-TI-149B,** HTX RTN TMP, should be kept less than 150°F and MUST remain less **than** or equal to **180** °F.

PERFORM ONLY AFTER ITS IMPLEMENTATION

IF ITS Implemented, **THEN:**

4.26 **All** RCP and RHR **pumps may** be removed from **operation** for **1** hour or **less,** per 8 hour period, provided **the** requirements of Tech Spec **3.4.6** are **met.**

PERFORM ONLY AFTER ITS IMPLEMENTATION

Init Verif

5.0 INSTRUCTKONS

5.1 Placing RHR System In Operation

______ 5.1.1 Verify Initial Conditions are satisfied.

5.1.2 Review Precautions and Limitations.

5.1.3 IF CC valve inservice testing is required, THEN refer to 1-PT-213.15, Valve Inservice Inspection (Misc).

5.1.4 Open the following valves:

______ • 1-CC-TV-103A, A RIIR HX CC OUTLET ISOL

• 1-CC-TV-103B, B RIIR HX CC OUTLET ISOL

______ 5.1.5 Open 1-CH-HCV-1142, RHR LETDOWN ISOL VALVE, to 10 percent to begin equalizing RCS and RHR pressure.

______ 5.1.6 Close 1-RH-HCV-1758, RHR HEAT EXCHANGER OUTLET.

5.1.7 In MANUAL, open 1-RH-FCV-1605, RHR HEAT EXCHANGER BYPASS FLOW.

5.1.8 Have Chemistry personnel sample the RCS and RHR System boron concentration (C_B). Record the C_Bs:

______ • RCS C_B: ____________ ppm

• RHR C_B: ____________ ppm

NOTE: The current fuel cycle lowest RCS Boron concentration is the lowest Boron concentration that existed at any **time** during current fuel cycle. This is obtainable **from** the Chemistry Department.

5.1.9 Record the current fuel cycle lowest RCS Boron Concentration: **(References 2.3.30 and 2.3.31)**

_______ • RCS $C_{B\ cycle\ minimum}$: ____________ ppm

NOTE: The following calculation determines **the** RHR Boron Concentration including the volume of B and C Accumulator Discharge **lines.**

5.1.10 Calculate **the** RHR \ Accumulator line C_B (RHR\Accum C_B), as follows: **(References 2.3.30 and 2.3.31)**

_______ a. Perform Attachment 2, Boron Concentration Determination.

_______ b. Perform Independent Verification of the calculations in Attachment 2, Boron Concentration Determination.

_______ _______ c. Record the RHR boron concentration including the volume of B and C accumulator discharge lines calculated in Attachment 2: _________ ppm

5.1.11 IF at least **one** RCP **is running** AND the RHR \ accumulator line C_B (RHR/accum C_B) calculated in Attachment 2, Boron Concentration Determination, is **less than the** RCS C_B recorded in Step **5.1.8,** THEN do **the** following: **(References 2.3.28 and 2.3.29)**

_______ a. **Perform** Attachment **3,** RCS Dilution Determination.

_______ **b.** Perform Independent Verification of the calculations of Attachment 3, RCS Dilution Determination.

_______ _______ c. Record the new RCS C_B resulting from the dilution that would occur due to placing RHR in service calculated in Attachment 3: __________ ppm

NOTE: IF the dilution will NOT reduce RCS C_B below Cold Shutdown C_B requirements, THEN boration is not required.

______ ______ d. Determine **minimum** RCS C_B for cold shutdown from the latest Shutdown Margin PT. Record the **minimum** RCS C_B: __________ ppm

e. IF the minimum required cold shutdown C_B recorded in **Step** 5.1.11 (d) is greater **than** the RCS boron concentration due to RCS dilution recorded in **Step 5.1.11** (c), THEN do the following:

1. Perform Attachment 4, New Required RCS Boron.

2. Ifave a second person independently verify the calculations of Attachment **4,** New Required RCS Boron.

______ ______ 3. Record the new required RCS C_B calculated in Attachment 4: ____________ ppm

4. Borate **the** RCS as required to greater than or equal to the C_B recorded in Step **5.1.11** (e) **3.**

______ 5. Have Chemistry personnel **sample** the RCS (C_B). Record the C_B: RCS C_B: ____________ ppm

6. IF RCS C_B **recorded** in Step **5.1.11** (e) 5 is less than the minimum C_B recorded in Step 5.1.11 **(e) 3,** THEN **repeat** Steps 5.1.11 (e) 4 and **5.1.11 (e) 5 until** RCS C_B **is** greater than or equal to the minimum C_B **recorded** in Step **5.1.11 (e) 3.**

5.1.12 IF NO RCP is running AND the RHR \ accumulator line C_B (RHR/accum C_B) recorded in Step 5.1.10 (c) is less than the KCS C_B recorded in Step 5.1.8, THEN do the following: (Reference **2.3.29**)

a. Calculate the Adjusted Minimum RCS C_B for cold shutdown using the latest 1-PT-IO, Shutdown Margin Determination as follows:

______ ▪ ______ 1. Record the latest minimum RCS C_B from I-PT-IO, Shutdown **Margin** Determination: (A) ________ ppm

______ ______ 2. Add the 100 ppm RCS C_B Adjustment Factor to determine the Adjusted Minimum RCS C_B

[(A) ppm + 100 ppm] = ________ ppm

NOTE: IF performing this procedure in accordance with **an** Emergency OR Abnormal Operating Procedure AND RHR must be placed into service immediately, THEN Step 5.1.12 (b) may be marked N/A with Shift Supervisor concurrence.

______ SS b. IF the RHR/accumulator line C_B (RHR \ accum C_B) recorded in Step 5.1.10 (c) **is** less than the adjusted Minimum RCS C_B recorded in Step 5.1.12 ***(a)*** 2, THEN perform Attachment 1, RIIR System Boration.

______ STA 5.I.13 **Notify** the STA to complete 1-GOP-13.0, Alternate Core Cooling Method Assessment. (Reference **2.4.1)**

5.1.14 Unlock **and** place the following breakers in ON:

______
- 1-EE-BKR-1H1-2S D1, Loop A Hot Leg **to** RH **Pumps** Isol Valve Ckt Bkr 1-RH-MOV-1700
- 1-EE-BKR-1J1-2S **F3,** Loop A Hot Leg to RH Pps Circuit **Breaker** I-**RH-MOV-1701**

5.1.15 Ensure **the** following conditions exist:

______ • RCS pressure is less than 418 psig.

______ • RCS temperature **is** less than 350"F.

NOTE: Increasing Letdown pressure to equalize KCS and RHR pressures will allow 1-RH-MOV-1700 and 1-RH-MOV-1701 to open easily.

5.1.16 Open **the** following **valves:**

______ • 1-RH-MOV-1700, RESIDUAL IIEAT REMOVAL INLET ISOLATION VALVE

______ • 1-RH-MOV-1701, RESIDIJAL **HEAT** REMOVAL INLET ISOLATION VALVE

NOTE: CC flow must **be** established through the RHR Heat Exchanger in order for **the** temperature indicators to properly indicate actual CC temperature.

NOTE: The CC Head **Tank** level should be monitored while CC is adjusted **through** the RHR HX.

5.1.17 To ensure that **CC** return temperature on 1-CC-TI-149.4 and 1-CC-TI-149B remains 180"F or **less** during RHR warmup, throttle open BOTH of the following valves:

______ • 1-CC-MOV-100A, RHR HEAT EXCHANGER CC RETURN VALVE

______ • **1-CC-MOV-100B,** RHR HEAT EXCHANGER CC **RETURN** VALVE

* *

CAUTION: Operating both RHR Pumps on recirc at the same time may result in deadheading one pump.

CAUTION: To maximize pump life, the time RHR **Pumps** are operated on recirc **must** be minimized.

* *

5.1.18 **Start** one of ***the*** following pumps. Mark the remaining pump N/A:

- 1-RH-P-1A, RESIDUAL HT **REMOVAL** PP A
- 1-RH-P-1B, RESIDUAL HT REMOVAL PP B

NOTE: Letdown flow should be maintained at less **than 120** gpm.

NOTE: **RHR** warmup rate **must not** exceed 200° F/hr.

5.1.19 Throttle open 1-CH-HCV-1142, RHR LETDOWN ISOL VALVE.

5.1.20 After verifying proper operation of the running RHR Pump, swap pumps as follows:

a. **Stop the** running RHR Pump.

b. Start **the** other RHR Pump.

* *

CAUTION: IF less than four SW pumps are running when either Unit has accident conditions, THEN the following precautions apply in order to avoid SW pump cavitation or runout problems during evolutions that increase demand on the SW system:

- Discharge pressures for running **SW** Pumps must be maintained greater than **43 psig** (monitored in the Control **Room).**
- IF SW Reservoir level **drops** below **313'** elevation, THEN **SW** Pumps must he monitored closely, and SW flows adjusted as appropriate.
- In order to maintain SW **pump** discharge pressures above **43** psig, throttling of SW **flows** to CCHXs or other loads may he necessary.

* *

NOTE: IF decay heat load is **low,** THEN a second CC Pump may NOT be needed.

5.1.21 IF required, THEN do the following:

_____ • Start mother **CC Pump** using 1-OP-51.1, Component Cooling System.

_____ • Start another **SW Pump** using **0-OP-49.1,** Service Water System Normal Operation.

* *

CAUTION: **A** warm-up period **of** 5 minutes at 50-200 gpm is required before **RHR** flow is increased.

CAUTION: Flow through the **tube** side of the **RHR** HX should not exceed 4000 gpm (2×10^6 lbs/hr).

CAUTION: **RHR** flow MUST be > 2500 **gpm** to **ANY RCS loop** receiving **RHR** discharge flow, to prevent potential damage to the Accumulator **and** RHR discharge check **valves,** due to flow induced chatter. **(References 2.4.2 and 2.4.6)**

* *

_____ 5.1.22 Close 1-RH-FCV-1605, RHR **HEAT EXCHANGER BYPASS FLOW.**

5.1.23 Close the following breakers:

- 1-EE-BKR-1H1-2S C2, Residual Heat Removal to B RCS Loop Hsol Vv CB 1-RH-MOV-1720A
- 1-EE-BKR-1Jl-2S C1, Residual Meat Removal to C RCS Loop Isol Valve CB 1-RH-MOV-1720B

NOTE: Both RHR discharges are flowed to ensure proper boration of each stagnant line. The RHR Discharge Valve flowed first should be the one that will NOT remain in service.

5.1.24 Throttle open one RHR Outlet Discharge, as follows:

a. Press OPEN button to start valve moving open on ONE of the following valves. Mark valve not opened NIA:

- **1-RH-MOV-1720A,** RHR OUTLET ISOL VALVE DISCHARGE TO **E** COLD LEG
- **1-RH-MOV-1720B,** RHR OUTLET ISOLATION DISCHARGE TO **C** COLD LEG

b. WHEN **the** RHR Outlet Discharge is in the desired position, THEN press OPEN AND CLOSE buttons SIMULTANEOUSLY and release to stop the valve.

5.1.25 WHEN RHR HX outlet temperature on 1-RH-TR-1606, RIIR Temperature, *is* within 10° F of RCS temperature, THEN **do** the following:

a. Fully open ONE of the following valves. Mark valve not opened N/A

- **1-RH-MOV-1720A,** RHR OUTLET HSOL VALVE DISCHARGE TO B COLD LEG
- **1-RH-MOV-1720B,** RHR OUTLET ISOLATION DISCHARGE TO C COLD LEG

_____ b. Record time: ____________

* *

CAUTION: RHR flow in excess of4000 gpm could result in RHR Pump runout.

CAUTION: RHR flow MUST be > 2500 gpm to **ANY** RCS loop receiving RHR discharge flow, to prevent potential damage to the Accumulator and RHR discharge check valves, due to **flow** induced chatter. **(References. 2.4.2 and 2.4.6)**

* *

_____ 5.1.26 Slowly open 1-RH-FCV-1605, RHR HEAT EXCHANGER BYPASS FLOW, to equalize temperature between RHR and RCS.

* *

CAUTION: To maximize pump life, the time RHR Pumps are operated on recirc must be **minimized.**

* *

5.1.27 WHEN ten minutes have elapsed since the performance of Step 5.1.25, THEN do the following:

a. Close the valve opened in Step 5.1.25. Mark **valve** not closed N/A:

_____ • **1-RH-MOV-1720A,** RHR OUTLET ISOL VALVE DISCHARGE TO B COLD LEG

_____ • **1-RH-MOV-1720B,** RHR OUTLET ISOLATION DISCHARGE TO C COLD LEG

b. Open the valve NOT opened in Step 5.1.25. Mark the other valve **N/A**

- **1-RH-MOV-1720A**, RHR OUTLET ISOL VALVE DISCHARGE TO B COLD LEG
- **1-RH-MOV-1720B**, RHR OUTLET ISOLATION DISCHARGE TO C COLD LEG

5.1.28 WHEN temperature has equalized, THEN place the M/A station for 1-RH-FCV-1605, RHR HEAT EXCHANGER BYPASS FLOW, in AUTO.

5.1.29 IF required, THEN adjust 1-CH-HCV-1142, RHR LETDOWN ISOL VALVE.

NOTE: The purpose of the next step is to verify the accuracy of the calculations (done in Step 5.1.11) regarding the amount of RCS dilution when RHK was placed in service.

______ 5.1.30 IF the RCS was borated in Step 5.1.11.e, THEN verify RCS C_B is greater than or equal to the minimum required CSD C_B recorded in Step 5.1.11.d.

NOTE: The **purpose of** the next step **is** to **verify** proper operation of 1-RH-FCV-1605, RHR HEAT EXCHANGER BYPASS FLOW, and 1-RH-HCV-1758, RHR HEAT EXCHANGER OUTLET, **and** to verify that RHR Pump discharge check valves seat properly.

______ 5.1.31 Satisfy inservice testing requirements by verifying that approximately 4000 gpm is maintained with the M/A station for 1-RH-FCV-1605, RHR HEAT EXCHANGER BYPASS FLOW, in AUTO.

* *

CAUTION: **A** warm-up period of 5 minutes at 50-200 gpm is required before RHR flow **is** increased.

CAUTION: Flow through the tube side of the RHR HX should not exceed 4000 gpm (2 x 10^6 lbs/hr).

CAUTION: RHR flow <u>MUST</u> be > 2500 gpm to **ANY** RCS loop receiving RIIR discharge flow, to prevent potential damage to **the** Accumulator **and** RHR discharge check valves, due to **flow** induced chatter. **(References 2.4.2 and 2.4.6)**

* *

_______ 5.1.32 Using 1-OP-3.3, Unit Shutdown from Mode **4** to Mode **5**, control the cooldown rate by adjusting 1-RH-HCV-1758, **RIIR HEAT** EXCHANGER OUTLET.

Completed ________________________ Date: ____________

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Unit 4 has experienced a loss of offsite power with a failure of the 1H diesel to start. An SI signal is then generated due to SG delta P.

Under these conditions, which ONE of the following sets of valves will reposition on the SI signal?

A✓ 1-SI-MOV-1867D, BIT outlet
1-CH-MOV-11158, Charging pump suction from RWST
1-CH-MOV-12898, Normal charging outlet

B. 1-SI-MOV-1867C, BIT outlet
1-CH-MOV-1115B, Charging pump suction from RWST
1-CH-MOV-1115E, Charging pump suction from VCT

C. 1-SI-MOV-1867A, BIT inlet
1-CH-MOV-I2898, Normal charging outlet
1-CH-MOV-11158, Charging pump suction from RWST

D. 1-SI-MOV-1867B, BIT inlet
1-CH-MOV-1289A, Normal charging outlet
1-CH-MOV-1115E, Charging pump suction from VCT

A. Correct. 1867D is BIT outlet MOV powered from J bus. 1-CH-MOV-1115B is charging pump suction from RWST powered *from* J ***bus.*** 1-CH-MOV-1289B **is** normal charging outlet powered from 1J bus. (The charging pump suctions from the RWST and VCT are confusing. 1115C and 11 15D are powered from the H bus while 1115B and 1115E are powered from the J bus. If the candidate confuses these power supplies he/she could immediately discard all answers with 1115B in them as being incorrect.)

B. Incorrect. 1-SI-MOV-1867C is powered from H bus.

C. Incorrect. 1-SI-MOV-1867A is powered from H bus.

D. Incorrect. 1-CH-MOV-1289A **is** powered from H bus.

Knowledge of electrical **power supplies to** the **following: ESFAS-operated valves**

(CFR: 41.7)

New question

References:
North **Anna** load **list**

QUESTIONS REPORT
for ROQUESTIONS

Level (RO/SRO):	RO	**Tier:**	*2*
Group:	1	**Importance** Rating:	3.6/3.8
Type (Bank/Mod/New):	NEW	Cog (Knowledge/Comp):	KNOWLEDGE
Reference (YM):	N	**Last** Exam(Y/N):	N

606 K2.04

BUS/MCC MARK NO.: 1-EE -MCC -1J1-2N DESCRIPTION: 1-EP-MC-21

LOCATION: CABLE VAULT UNIT 1 REFERENCE: 11715-FE-1R

OWER SUPPLY: 1-EP-480-1J 14J-5

BREAKER NO.	MARK NUMBER	DESCRIPTION
A1	1-Hv -F -37P	CONTROL ROD DRIVE COOLING FAN
A2	1-PG -P -2B	PG PUMP
A3	1-IA -C -2B	CONTAINMENT INSTR AIR COMP
A4	1-SW -P -6	RAD MONITOR SAMPLE PUMP
Bl	I-IA -C -2B	REDUNDANT IN SERIES - NOT CONNECTED TO BUS********
B2	1-CV -P -3B	CONTAINMENT VACUUM PUMP
B3	1-CH -EHR -6B	BORIC ACID TANK HEATER
B4	1-CH -EHR -7B	BORIC ACID TANK HEATER
B5	1-SW -P -7	RAD MONITOR SAMPLE PUMP
C1	1-SW -MOV -103B	"B" RS HX SUPPLY
C2	1-SW -MOV -103C	"C" RS HX SUPPLY
63	1-SI -MOV -1867B	ISOL INLET MOV TO 1-SI-TK-2 (BIT)
C4	1-RS -MOV -100B	CASING COOLING PUMP DISCHARGE
D1	1-SW -MOV -104B	"B" RS HX RETURN
D2	1-SW -MOV -104C	"C" RS HX RETURN
D3	1-SI -MOV -1867D	BIT OUTLET
D4	1-RS -MOV -101A	CASING COOLING DISCHARGE
El	1-SW -MOV -101B	"A" SW HDR SUPPLY TO RS HX
E2	1-SW -MOV -101D	"B" SW HDR SUPPLY TO RS HX
E3	1-CH -MOV -1115B	CHG PUMP SUCTION FROM RWST
E4L	1-EP -CB -84G 1-CC -P -1B 2-cc -P -lB	MOTOR KEATER CAB 1-EP-CB-84G: 1-cc-P-1B 2-CC-P-1B TRANS 65
E4R	1-EP -CB -13R	HEAT TRACE TRANSFORMER 13R
F1	1-SW -MOV -105B	RS HX RETURN TO "B" SW HDR
F2	1-SW -MOV -105D	RS HX RETURN TO "A" SW KDR
F3	1-CH -MOV -1115E	CHG PUMP SUCTION FROM VCT
F4	1-FW -MOV -100A	MOV HDR PEED TO "A" S/G
G1	1-QS -MOV -101B	"B" QS PUMP DISCHARGE
G2	1-SI -MOV -1862B	"B" LHSI PMP SUCT FROM RSWT
G3	1-QS -MOV -100B	"B" QS PUMP SUCTION
G4	1-FW -MOV -100B	MOV HDR FEED TO "B" S/G
H1	1-RS -MOV -155B	"B" OUTSIDE RS PUMP SUCT
H2	1-RS -MOV -156B	"B" OUTSIDE RS PUMP DISCHARGE
H3	1-CH -MOV -1289B	NORMAL CHARGING HDR ISOL
H4	1-FW -MOV -100C	MOV HDR FEED TO "C" S/G
J1	I-SI -MOV -1869B	NORM CHG KDR DISCH TO TH
J2	1-SI. -MOV -1863B	"B" LKSI DISCH TO CHG PUMP SUCTION
J3	1-SI -MOV -1860B	"B" LHSI PMP SUCT FROM CONT SUMP
J4	1-HV -F -70B	AUX FEED PUMP HOUSE EXHAUST FAN
K1	I-SI -MOV -1864B	"B" LHSI PUMP DISCH TO TC
K2	1-SI -MOV -1890B	"B" LHSI PUMP DISCH TO TH
K3	1-SI -MOV -1890D	LHSI DISCH TO TC
K4	I-SI -MOV -1865C	"C" ACCUM OUTLET

BUS/MCC MARK NO.: 1-EE -MCC ≠1H1-2N DESCRIPTION: 1-EP-MC-19

LOCATION: CABLE VAULT UNIT 1 REFERENCE: 11715-FE-1Q

)WER SUPPLY: 2-EP-480-1H 14H-3

BREAKER NO.	MARK NUMBER	DESCRIPTION
A1	1-HV -F -37A	CONTROL ROB COOLING FAN
A2	1-HV -F -8A	AUX BDLG CENTRAL EXH PAN
A3	1-IA -C -2A	CONT INSTR AIR COMP
A4	1-SW -P -8	RAD MONITOR SAMPLE PUMP
B1	1-HV -F -37B	CONTROL ROD COOLING FAN
B2	1-PG -P -2A	PG PUMP
B3	1-CV -P -3A	CONT VACUUM PUMP
B4	1-HC -F -1	CONT PURGE BLOWER
C1	1-EG -€ -lHB	1H DIESEL FUEL OIL TRANS PUMP (B/U)
C2	1-CH -EHR -6A	BORIC ACID TANK HTR
c3	1-SI -EHR -2A	BORON INJECTION TANK HTR
c4		SPARE
C5	1-SW -P -9A	RAD MONITOR SAMPLE PUMP
D1	1-SI -MOV -1867A	BIT INLET
D2	1-SW -MOV -103A	"A" RS HX INLET
D3	1-SW -MOV -103D	"D" RS HX INLET
D4		SPARE
El	1-SI -MOV -1867C	BIT OUTLET
E2	1-SW -MOV -101A	"A" SW HDR SUPPLY-RS HX
E3	1-SW -MOV -101C	"B" SW HDR SUPPLY-RS HX
E4		SPARE
F1	1-CH -MOV -1115C	CHG PMP SUCT FROM VCT
F2	1-CH -MOV -1115D	CHG PMP SUCT FROM RWST
F3	1-CH -MOV -1275B	"B" CHG PMP RECIRC
F4		SPARE
G1	1-CH -MOV -1289A	NORMAL CHG LINE
G2	1-SI -MOV -1860A	"A" LHSI PMP SUCT CONT SUMP
G3	1-SI -MOV -1869A	ALT CHG HDR DISCH TO TH
64		SPARE
H1	1-HV -F -70A	AUX FEED PUMP HOUSE EXHAUST FAN
H2	1-SI -MOV -1863A	"A" LHSI PUMP DISCH TO CHG PP SUCT
H3	1-SI -MOV -1864A	"A" LHSI PUMP DISCH TO TC
H4	1-HV -F -71A	SFDS EMERG VENT FAN
J1	1-QS -MOV -101A	"A" QS PUMP DISCH
52	1-SI -MOV -1890A	"A" LHSI PUMP BISCH TO TH
J3	1-SI -MOV -1890C	LHSI DPSCH TO TC
54	1-IA -C -2A	CONT INSTR AIR COMP
K1	1-RS -MOV -156A	"A" OUTSIDE RS PUMP DISCH
K2	1-RS -MOV -155A	"A" OUTSIDE RS PUMP SUCT
K3	1-FC -P -lA	SPENT FUEL PIT PUMP
L1	1-SW -MOV -104A	"A" RS HX OUTLET
L2	1-SW -MOV -104D	"D" RS HX OUTLET
L3	1-SI -MOV -1865A	"A" ACCUM OUTLET
L4L	1-EP -CB -84F	MOTOR HEATER FUSE PANEL 1-EP-CB-84F
	1-CC -P -1A	MOTOR FUSES FOR 1-CC-P-1A
	2-CC -P -lA	MOTOR FUSES FOR 2-CC-P-1A
		TRANS 64
L4R	1-EP -CB -13N	HEAT TRACE TRANS 13N
M1	1-SW -MOV -105A	RS HX RETURN-"B" SW HEADER
M2	1-SW -MOV -105C	RS HX RETURN-"A" SW HEADER
M3	1-RC -MOV -1586	"B" LOOP BYPASS
M4		SPARE

007A4.01001

The following conditions exist:

- Unit 1 is at 100% power
- A PRZR safety valve has been leaking by for several hours
- Annunciator B-HI, PRZ RELIEF TK HI TEMP, has just illuminated
- The annunciator response directs the operator to do a feed and bleed on the PRT to reduce temperature.

The operator will open 1-RC-HCV-1519B, PRZR RELIEF TANK MAKEUP WATER SUPPLY ISOL, to fill the PRT using the __________ located on __________

A?' switch; benchboard 1-1
B. switch; benchboard 1-2
C. pushbutton; "H" safeguards panel
D. pushbutton; "J" safeguards panel

A. Correct. The valve is operated by an open/close switch located on benchboard 1-1.

B. Incorrect. The switch is iocated on benchboard 1-1 not benchboard 1-2.

C. Incorrect. While valves located on the safeguards panels are usually operated by pushbutton, this valve is located on benchboard 1-1. 1-RC-HCV-1519A is located on the "H" safeguards panel.

D. Incorrect. While valves located on the safeguards panels are usually operated by pushbutton, this valve is located on benchboard 1-1.

Ability to **manually operate and/or monitor in the control room:** PRT **spray supply valve**
(CFR: **41.7 / 45.5 to 45.8)**

New question

References:
NCRODP module 38, Reactor Coolant System (page 60)

Level (RO/SRO):	RO	Tier:	2
Group:	1	Importance Rating:	2.7/2.7
Type (Bank/Mod/New):	NEW	Cog (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	N	Last Exam(Y/N):	N

007 A4.01

3. annunciation of the PRESSURIZER RELIEF TANK HIGH TEMPERATURE alarm (Window 1B-H1) at 112°F.

Pressure transmitter PT-1472 provides the following outputs:

1. indication at Vertical Board 1-I (PI-1472),
2. computer input, and
3. annunciation of the PRESSURIZER RELIEF TANK HIGH PRESSURE alarm (Window 1B-F1) at 14 psig.

Level Transmitter LT-1470 provides the following outputs:

1. indimtion at Vertical Board 1-1 (LI-1470),
2. computer input, and
3. annunciation of the PRESSURIZER RELIEF TANK HIGH-LOW LEVEL alarm (Window 1B-G1) at 78 percent/66 percent.

PRT Controls. The PRT is penetrated by a nitrogen supply line, a primary grade makeup water supply line, a vent line, and a drain line. The flow through each of these lines is controlled by an air-operated, solenoid-controlled valve. The valves are as foilows:

nitrogen supply valve	HCV-1550
makeup water supply valve	HCV-1519B
drain valve	HCV-1523
vent valve	HCV-1549

Each of the valves has a two-position (CLOSE. OPEN) handswitch on the MCB benchboard section. Above each switch there is a red (green) light which indicates that the valve is open (shut).

Draindown Level Indication System Instrumentation and Alarms. A magnetic flag type, visual level indicator is mounted on the standpipe. This allows local level monitoring between the cold leg center line (256 ft-4 in. elevation, which is 0 ft. on the local scale) and the 269 ft-10 in. elevation, which is 13 ft - 6 in. on the scale. The standpipe has an accuracy of ±1/2 inch.

A Rosemount differential pressure type level transmitter is mounted off the standpipe and provides input to a level indicator and recorder on the 1-2 control board in the control room labeied "Cold Shutdown RCS Level." This indication has a span of 0 to 100 inches with 0 being the centerline of C Loop cold leg. It has an accuracy of ±2 inches.

The transmitter also supplies an input to the RX CLNT DRAIN DOWN LO LVL annunciator which has a setpoint of +9 inches above centerline.

The transmitter and the annunciator receive power from Westinghouse Process Rack F which is powered from Semi-Vital bus 1-A. The control room indicator receives power from I-EQ-CB-24 which is powered from 1C1-2.

Which ONE (1) of the following would explain the inability to reset Train "A" safety injection?

A.✓ Reactor trip breaker RTA failed to open
B. Reactor trip breaker RTB failed to open
C. Time delay for resetting SI has not been met
D. Signal that initiated the SI has not cleared

Knowledge of the interrelations between Reactor Trip - Stabilization and the following: Breakers, relays and disconnects

(CFR: 41.7 / 45.7 / 45.8)

North Anna bank question 5772

References:
NCRODP module 77 - Reactor Protection System (page 50)

Level (RO/SRO):	RO	Tier:	I
Group:	1	Importance Rating:	2.6/2.8
Type (Bank/Mod/New):	BANK	Cog (Knowledge/Comp):	COMPREHENSION
Reference (Y/N):	N	Last Exam (Y/N):	N

OO7 EK2.0

The manual phase A containment isolation trip signal is processed by a lock-in relay. This lock-In relay signal and the one from the ESF actuation circuit (discussed above) are processed by an OR gate. Receipt of either signal generates a phase A containment isolation signal. The manual resets on benchboard 1-1 reset both the manual actuation and ESF actuation lock-in relays for their respective trains.

The intermediate high-high containment pressure trip signal is processed with the high steam line flow trip signal (mentioned above) by an OR gate. Receipt of either trip signal is retained by a lock-in relay. The lock-in relay generates a steam line isolation signal for each of the three loops (only one loop shown in Figure 77-12-NA). The manual resets on the safeguards panels reset the lock-in relays for their respective trains.

Each of the main steam line trip valves can be closed using a pushbutton on either safeguards panel (Train A SOV operated by pushbutton on Train A safeguards panel; Train B SOV operated by pushbutton on Train B safeguards panel). The manual close signal is processed with the steam line isolation signal for each loop by an OR gate. Receipt of either signal generates a signal that is sent to the main steam line trip valves.

The high-high Containment pressure trip signal is sent to both the phase B containment isolation circuits and to the spray actuation circuits. A lock-in relay (with rnr) is used to retain the signal sent to each of the ESF circuits. The manual spray actuation/phase B containment isolation trip signai is also sent to the phase B containment isolation circuits and to the spray actuation circuits. A lock-in relay is used for each of the manual actuation signals.

The lock-in relay signals to the phase B containment isolation circuits are processed by an OR gate. Receipt of either a high-high containment pressure or manual spray actuation signal generates a phase B containment isolation actuation signal and actuates a CONTAINMENT ISOLATION PHASE B alarm. Phase "B" manual reset pushbuttons on benchboard 1-2 reset the lock-in relays for both trip signals.

The lock-in relay signals to the spray actuation circuits are processed by an OR gate. Receipt of either a high-high containment pressure or manual spray actuation signal generates a spray actuation signal and actuates a CONTAINMENT VESSEL DEPRESSURIZATION INITIATE alarm. Spray actuation manual reset switches on benchboard 1-1 reset the lock-in relays for both trip signals.

--> Safety Injection Blocking Circuit. The SI blocking circuit allows the operator to prevent automatic re-initiation of the SI systems by resetting SI on benchboard 1-1. The block remains in effect until the reactor trip breakers are reset. Resetting the reactor trip breakers automatically defeats the blocking circuit and arms the SI automatic initiating circuits.

The SI blocking circuit consists of three AND gates, one OR gate, and a NOT gate. The three AND gates control the operation of the blocking circuit and are referred to as the SI trip AND gate, the manual reset AND gate, and the reactor trip AND gate (see Figure 77-12-NA). The reactor trip signal is derived from permissive interlock P-4 in RPS.

During normal power operations no reactor trip signal is present, so the output of the reactor trip AND gate is 0. The 0 signal is passed to an OR gate and to the NOT gate. The output of the NOT gate is therefore a 1 signal, which is passed to the SI trip AND gate On receipt of a SI trip signal from one of the automatic trip sources, the SI trip AND gate has two 1 signals and therefore passes a 1 signal to the SI lock-in relay. Upon receipt of the SI trip signal, the lock-in relay raises its output to a high signal, which actuates the ESF Systems. The automatic SI trip signal causes a reactor trip. which results in a 1 signal to the reactor trip AND gate.

After a 60-second time delay, the SI actuation signal Is passed to the manual reset AND gate in the blocking circuit. When the operator resets SI, the manual reset AND gate raises its output signal to a 1 signal. The reactor trip AND gate output signai (which is 0) is processed with the manual reset AND gate output signal by an OR gate. The manual reset AND gate 1 output signal causes the OR gate output signal to go to a 1, which is passed to the reactor trip AND gate. The reactor trip AND gate now has two 1 signals as inputs and raises its output to a 1 signal.

The 1 output signal from the reactor trip AND gate has two effects. First, it satisfies the OR gate to keep its output of a 1. This means that the manual reset must be inserted only momentarily, since the reactor trip AND gate then generates the signal that keeps one of its Inputs a 1. Second, the 1 output signal to the NOT gate causes its output to go to 0. Any subsequent <u>automatic</u> SI trip signals are not passed through the SI trip AND gate because the NOT gate input is a 0, thus blocking automatic re-initiation of SI. Note that SI may be <u>manually</u> initiated at any time. The reactor trip AND gate remains satisfied until the P-4 reactor trip signal from ESF is removed, which occurs when the reactor trip breakers are reset.

Note that, if the P-4 signal for either train of SI is <u>not</u> present, that train of SI may still be manually reset after the 60-second time delay. However, if any automatic SI initiating signal is still present, then that train of SI will **re-initiate** after the RESET switch is released. This is because the block function requires the P-4 signal for that train, but the reset function does not require the P-4 signal.

When the operator resets SI, the 1 output signal from tha manual reset AND gate also resets the lock-in relay. Resetting the lock-in relay removes the SI actuation signal from:

1. emergency diesel generators,
2. AFW pumps,
3. the SI System, and
4. the Service Water System.

008AK1.02 001

Which ONE of the following describes how and why the ***rate*** of RCS depressurization changes as RCS pressure decreases from 1000 psig due to a stuck open pressurizer PORV? (Assume SI DOES NOT initiate and **NO** operator action is taken.)

A?' Steam voiding in the vessel head acts like a pressurizer and decreases the rate of RCS depressurization.
B. Reduced restrictions through the PRT rupture disk increase the rate of RCS depressurization
C. As RCS pressure decreases, pressurizer heaters are energized, decreasing the rate of RCS depressurization
D. At low RCS pressure, heat retained in the steam generators is released and increases the rate of RCS depressurization.

A. Correct. The phase change as the RCS hits saturation will slow the rate of depressurization until the amount of inventory loss becomes so great that the phase change does little to change the rate of RCS depressurization.

B. Incorrect. The PRT rupture disk is sized to allow relief through the PORV's without inhibiting their capacity to relieve **RCS** pressure. If the rupture disk were smaller than the PORV's this answer would cause the rate of decrease to slow. This makes this answer plausible.

C. Incorrect. As pressure decreases pressurizer heaters will energize but will not slow depressurization of the RCS from a stuck open PORV. The rate of inventory loss is so great that the contribution of the heaters can't be seen. This answer is plausible because on a slow loss of inventory heaters do affect the rate at which pressure decreases.

D. Incorrect. Heat released from a steam generator is a function of RCS temperature in comparison to steam generator temperature. During cold RCS temperatures RCP starts can cause a pressure spike in the RCS if steam generators are hotter than the RCS. This is mentioned both in procedures and Tee. Specs. Examinee may use this knowledge to choose this answer.

Knowledge of the operational implications of the following concepts as they apply to the Pressurizer Vapor Space Accident: Change in leak rate with change in pressure

(CFR: 41.8 to 41.10 / 45.3)

From INPO bank - Farley 1

Which one of the following describes the reason the rate of RCS pressure decrease changes as RCS pressure decreases from 1000 psig due to a stuck open pressurizer PORV? (Assume SI DOES NOT initiate and NO operator action is taken.

A. Steam voiding in the RCS acts like a Pressurizer and slows down the rate of RCS pressure decrease.

B. RCS pressure decrease rate increases due to reduced restrictions through the PRT rupture disk

C. As RCS pressure decreases, pressurizer heaters are energized, slowing the rate of RCS pressure decrease.

D. At low RCS pressure, heat retained in the steam generators is released which increases the rate of RCS pressure decrease.

Answer: A

Level (RO/SRO):	RO	**Tier:**	1
Group:	1	**Importance Rating:**	3.8/3.7
Type (Bank/Mod/New):	BANK	Cog (Knowledge/Comp):	COMPREHENSION
Reference (Y/N):	N	Last Exam(Y/N):	N

Pressurizer Vapor Space accident

Questions Marked for Collection

..000008.K1.02 10/23/1995

WEC Farley 1

Which ONE of the following describes the reason the rate of RCS pressure decrease changes as RCS pressure decreases from 1000 psig due to a stuck open Pressurizer PORV? (Assume SI DOES NOT initiate and NO operator action is taken.)

Steam voiding in the RCS acts like a pressurizer and slows down the rate of RCS pressure decrease.

RCS pressure decrease rate increases due to reduced restrictions through the PRT rupture disk.

As RCS pressure decreases, pressurizer heaters are energized, slowing the rate of RCS pressure decrease.

At low RCS pressures, heat retained in the Steam Generators is released which increases the rate of RCS pressure decrease.

008K4.01 001

Assume the following conditions.

- Unit 1 is in mode 5
- 1H EDG is tagged out for pre-planned maintenance
- 1-CC-P-1A is in auto-standby
- 1-CC-P-1B is running
- The 1J emergency bus normal feeder trips on overcurrent
- The 1J EDG starts and loads on the 1J bus as designed

Assuming that all systems respond as designed, which ONE of the following responses describes the final component cooling pump configuration after the event?

A.✓ 1-CC-P-1A running; 1-CC-P-1B running
B. 1-CC-P-1A running; 1-CC-P-1B not running
C. 1-CC-P-1A not running; 1-CC-P-1B running
D. 1-CC-P-1A not running; 1-CC-P-1B not running

A. Correct. The "A CC pump will start when the UV occurs on the "J" bus. The "B" CC pump will start 15 seconds after power is restored to the "J" *bus.*

B. Incorrect. The "A" CC pump will start when the UV occurs on the "J" bus. The "B" CC pump will start 15 seconds after power is restored to the "J" bus. A candidate who doesn't remember that the circuit does not look to see if there is another pump running when power is restored to the bus could choose this answer.

C. Incorrect. The " A CC pump will start when the UV occurs on the "J" bus. A candidate who doesn't remember that the "A pump has an auto start on UV on the opposite bus could choose this answer.

D. Incorrect. Both CC pumps will be running. The "A CC pump will start when the UV occurs on the "J" bus. The "B" CC pump will start 15 seconds after power is restored to the "J" bus.

Knowledge of CCW design feature(s) and/or interlock(s) which provide for the following: Automatic start of standby pump

North Anna bank question 3052

References:
Objective 3656 from Self-study guide for CC
ESK 5P, 5Q

QUESTIONS REPORT
for ROQUESTIONS

Level (RO/SRO):	RO	Tier:	2
Group:	1	Importance Rating:	3.113.3
Type (Bank/Mod/New):	RANK	Cog (Knowledge/Comp):	COMPREHENSION
Reference (Y/N):	N	Last Exam(Y/N):	N

2.1.1a. Objective

Explain the following concepts associated with the component cooling water pumps.

- Purpose
- Why a pump's discharge valve is throttled prior to starting the pump

2.1.1b. Content

1. The CC pumps provide the motive force for circulating cooling water through the CC heat exchangers, individual system loads, and back to the pumps suction.

2. When starting the pump, the pump's discharge valve is throttled to 25% open to prevent water hammer.

Topic 2.1.2: CC Pump Starts and Trips
3656

2.1.2a. Objective

List the following information associated with the component cooling water pumps

- Power supply to each pump
- Interlocks associated with manually starting a pump
- Interlocks associated with automatically starting a pump
- Interlocks associated with automatically tripping a pump

2.1.2b. Content

1. The CC pumps are powered from the 4160 volt stub-busses.

2. To manually start CC pump 1-CC-P-1A, the following conditions must exist:

   2.1. No ground or phase overcurrent condition exists

   2.2. Normal voltage on the supply bus for at least 20 seconds

   2.3. No CDA signal present

3. A CC pump will automatically start, provided all of the following conditions **exist:**

   3.1. Selected control switch in AUTO as appropriate (local or remote)

   3.2. ***Ne*** CDA signal

   3.3. **No** ground or phase overcurrent condition

3 3.4. Either:

   3.4.1. Auto-trip signal on opposite pump or undetvoltage condition on the opposite bus, assuming no UV/DV signal on supply bus **for** at least 20 seconds

   3.4.2. Following a UV/DV on supply bus: power restored **for** at least 15 seconds, but no more than 20 seconds.

4. The following conditions will automatically trip a ***CC*** pump:

   4.1. Undervoltage condition on the supply bus

   4.2. CDA signal

   4.3. Ground or phase overcurrent condition

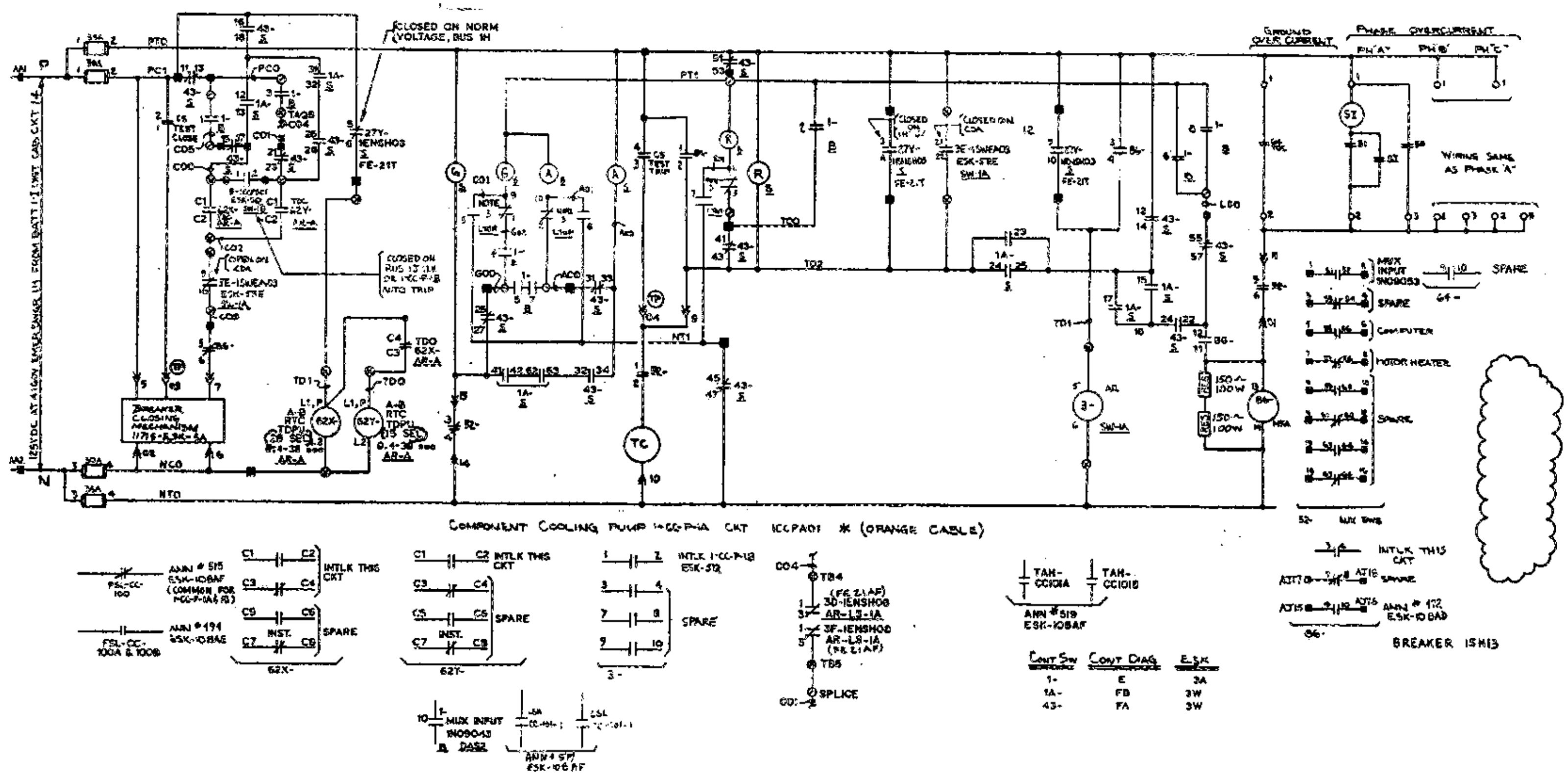
2.2: Component Cooling Heat Exchangers

Topic 2.2.1: Component Cooling Heat Exchangers (CCHX) 3657

2.2.1a. Objective

Explain the following concepts associated with the component cooling heat exchangers.

- **Purpose**
- How component cooling water temperature may be adjusted (1-OP-51.1)
- Why heat exchanger service water outlet valves are administratively controlled in the throttled position



NOTE:
1. PREFIX ALL WIRES WITH 1CCPA01
2. FIELD TO REMOVE INST CONTACTS FROM RELAY 62X- (ESK-5G)
3. FROM RELAY 30-1CE0A05, ESK 11AH

Dominion

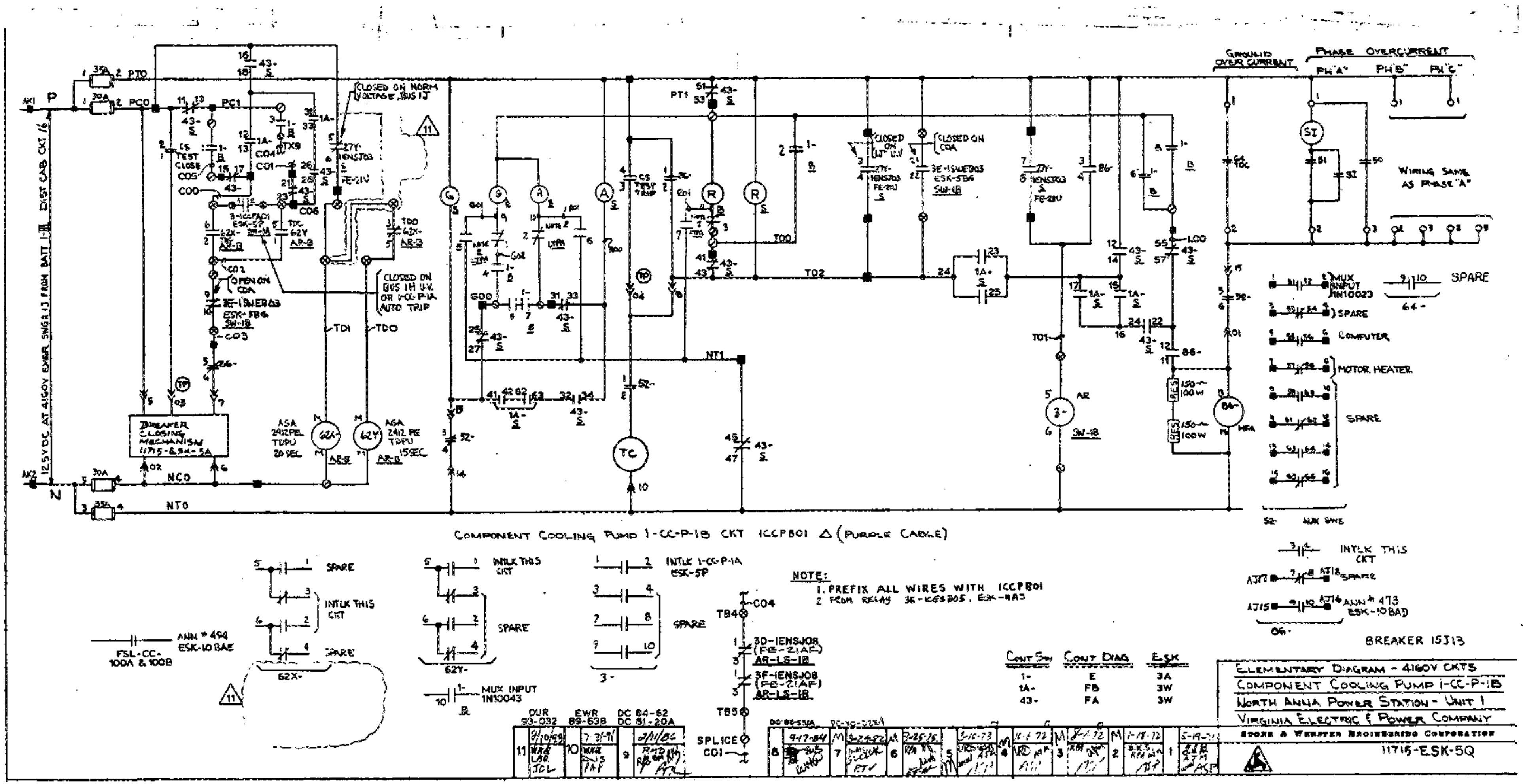
NUCLEAR ENGINEERING
RICHMOND, VIRGINIA

ELEMENTARY DIAGRAM-4160V CKTS
COMPONENT COOLING PUMP 1-CC-P-1A
NORTH ANNA POWER STATION - UNIT 1

REVISION DESCRIPTION	CAD NO: C:\usr\dgn\N1ESK5PNYBwh1	
REVISED & REDRAWN PER DCR 2003-0500 THIS DWG SUPERSEDES REV 15 ORIGINAL	DRAWING NO: 11715-ESK-5P	REV 16
DSGN ELC	SCALE: NONE	UNLESS OTHERWISE NOTED SH 1 OF 1

PC-EC1 & MC1

11715-ESK-5Q



125VDC AT 4160V EMER SWGR 1J FROM BATT 1-III DIST CAB CKT 16
BREAKER CLOSING MECHANISM 11715-ESK-5A
CLOSED ON NORM VOLTAGE BUS 1J
CLOSED ON BUS 1H U.V. OR 1-CC-P-1A AUTO TRIP
OPEN ON CDA
CLOSED ON CDA
GROUND OVERCURRENT
PHASE OVERCURRENT
PH "A"
PH "B"
PH "C"
WIRING SAME AS PHASE "A"
MUX INPUT 1N10023
SPARE
COMPUTER
MOTOR HEATER
52- AUX SWS
INTLK THIS CKT
ANN # 473 ESK-10BAD
ANN # 494 ESK-10BAE
FSL-CC-100A & 100B
COMPONENT COOLING PUMP 1-CC-P-1B CKT 1CCPB01 △ (PURPLE CABLE)
INTLK 1-CC-P-1A ESK-5P
MUX INPUT 1N10043
NOTE:
1. PREFIX ALL WIRES WITH 1CCPB01
2. FROM RELAY 3E-1ESB05, ESK-9A3
BREAKER 15J13
CONT SW | CONT DIAG | ESK
1- | E | 3A
1A- | FB | 3W
43- | FA | 3W
ELEMENTARY DIAGRAM - 4160V CKTS
COMPONENT COOLING PUMP 1-CC-P-1B
NORTH ANNA POWER STATION - UNIT 1
VIRGINIA ELECTRIC & POWER COMPANY
STONE & WEBSTER ENGINEERING CORPORATION
11715-ESK-5Q

009EA1 17 001

Unit 1 is operating at 100% power when a pressurizer safety valve lifts. The crew performs the immediate actions of 1-E-0 and initiates safety injection due to low-low pressurizer pressure. A short time later the SBA announces that containment sump level is increasing. The OATC announces that the PRT rupture disk has blown.

What ONE indication did the OATC use to come to this conclusion?

A?' PRT pressure suddenly decreased
5. Pressurizer level took a step decrease
C. PRT level suddenly decreased
D. Pressurizer pressure took a step decrease

A. Correct. PRT pressure will decrease when the rupture disk blows.

B. Incorrect. Pressurizer level will not take a step decrease when the rupture disk blows.

C. Incorrect. PRT level will remain stable.

D. Incorrect. Pressurizer pressure may be decreasing, but it will not take a step decrease when the PRT rupture disk blows.

Ability to operate and / or monitor the following as they apply to Small Break LOCA: PRT

(CFR: 41.7 / 45.5 / 45.6)

Modified bank question 5447

References:
Objective 75847 from self-study guide on Reactor Coolant System

Level (RO/SRO):	RO	Tier:	1
Group:	1	Importance Rating:	3.4/3.4
Type (Bank/Mod/New):	MODIFIED	Cog (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	N	Last Exam(Y/N):	N

Topic 3.2.3: PRT Rupture Discs
15847

3.2.3a. Objective

Explain the following concepts associated with the pressurizer relief tank rupture discs.

- Purpose of the rupture discs
- Pressure at which the rupture discs are desgned to function
- Indicated PRT pressure at which the rupture disc would be expected to function
- Means available in the control room to determine that the rupture discs have functioned

3.2.3b. Content

1. Two 18-inch rupture discs, located on top of the tank, provide overpressure protection for the PRT.

   1.1. The rupture discs have a combined capacity equal to that of the pressurizer safety valves.

2. The rupture discs are designed to blow at 100 psid.

   2.1 Rupture pressure is twice that expected during a design discharge in order to prevent lesser discharges from deforming the rupture disc.

3. Assuming that containment pressure is at 9.50 psia, the indicated pressure at which **the** rupture discs would blow would be 109.5 psi.

4. The following means are available in the control room to determine that the rupture discs have blown:

   4.1. PRT pressure decreases

   4.2. Containment radiation monitor readings increase.

   4.3. Containment sump level increases.

Which ONE of the following is NOT an indication in the control room that the PRT rupture discs have blown?

A. PRT temperature increases

B. PRT pressure decreases

C. Containment radiation monitor readings increase

D. Containment sump level increases

Answer: A

010GG2.4.31 ***001***

The following conditions exist:

- Unit 1 **is** at 100% power
- Annunciator B-F7, PRZ HI-LO PRESS has just illuminated
- 1-RC-PI-1444 indicates 1700 psig
- 1-RC-PI-1445, 1-RC-PI-1455, 1-RC-PI-1456, and 1-RC-PI-1457 indicate pressure **is** 2315 psig and slowly increasing.

In order to mitigate the consequences of this failure, the OATC should place the master pressure controller in manual and press the __________ pushbutton to __________ the controller output.

A?' lower; increase
B. lower; decrease
C. raise; increase
D. raise, decrease

A. Correct. 1-RC-PI-1444 has failed low. This PI has a range of 1700 to 2500 psig. The master pressure controller is inverse acting. The lower pushbutton should be pressed because pressure is high. This will increase the output of the master pressure controller causing heaters to turn off and sprays to open to lower pressure. Candidate must realize that pressure is indeed **increasing** and that output must be **increased** in order to **decrease** pressure.

B. Incorrect. The lower pushbutton **is** correct; however this will increase, not decrease, the output of the controller.

C. Incorrect. The raise pushbutton **is** incorrect; however the output of the controller does need to be increased.

D. Incorrect. The raise pushbutton **is** incorrect **as is** the desired change in the output of the controller. Candidate must realize that actual pressure is high.

Pressurizer Pressure Control
Knowledge of annunciators alarms and indications and use of the response instructions.

New question

References:
Annunciator response for B-F7
Loop book RC-111

Level (RO/SRO):	RO	Tier:	**2**
Group:	1	Importance Rating:	3.3/3.4
Type (Bank/Mod/New):	**NEW**	Cog (Knowledge/Comp):	COMPREHENSION
Reference (Y/N):	N	**Last Exam(Y/N):**	**N**

PRZ HI-LO PRESS	2310 psi HI 2215 psi LO

1.0 Probable Cause

1.1 Pressure control system malfunction
1.2 Load transient
1.3 Instrument failure
1.4 Cooldown in progress

2.0 Operator Action

2.1 Verify alarm:
2.1.1 Check PZR pressure channels
2.1.2 IF pressure is normal, THEN submit WR

2.2 High Pressure
2.2.1 Verify proper operation of the pressure controller. IF necessary, THEN take manual control and return pressure to normal
2.2.2 Ensure all heaters are off
2.2.3 Ensure both spray valves are fully open
2.2.4 IF due to a plant transient, THEN stop or reduce the rate of the transient to allow pressure to return to normal

2.3 Low Pressure
2.3.1 IF controlled cooldown/depressurization is in progress, THEN return to procedure and step in effect.
2.3.2 Verify proper operation of the pressure controller. IF necessary, THEN take manual control and return pressure to normal.
2.3.3 Ensure all heaters are on
2.3.4 Ensure both spray valves and power operated reliefs are fully closed
2.3.5 IF due to a plant transient, THEN stop or reduce the rate of the transient to allow pressure to return to normal
2.3.6 Go to 1-AP-44, LOSS OF REACTOR COOLANT SYSTEM PRESSURE
2.3.7 IF pressure dropped below 2205 psig, THEN refer to Tech Spec 3.2.5 (ITS 3.4.1)

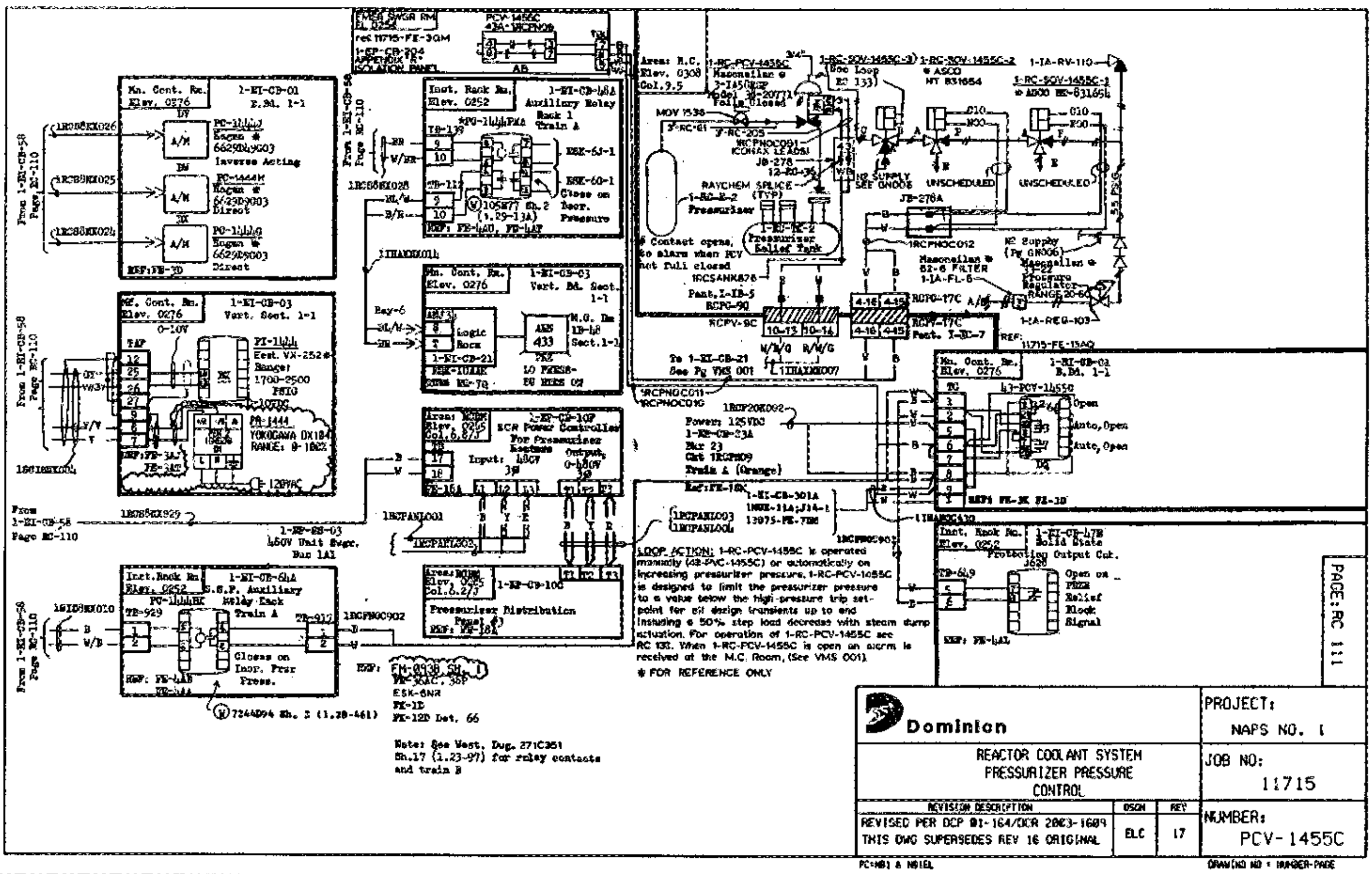
2.4 IF alarm is due to a failure of 1-RC-PT-1445, THEN refer to TS 3.4.3.2 (ITS 3.4.11) due to 1-RC-PCV-1456 inoperable.

3.0 References

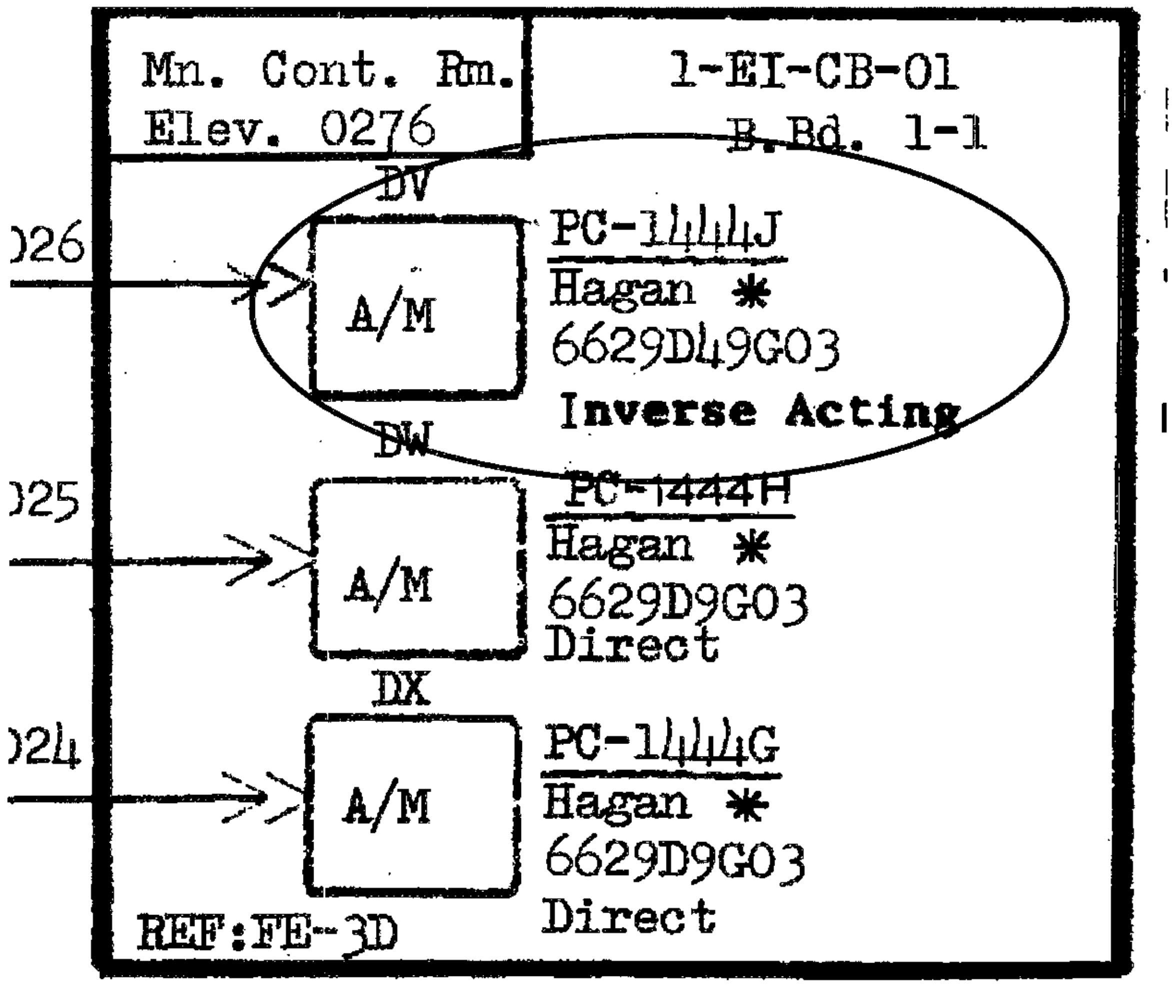
3.1 11715-ESK-10AAG
3.2 PLS Document
3.3 W Drawing 5655D33 Sheet 11
3.4 Unit 1 Loop Book, page RC-108
3.5 EWR 92 - 049A
3.6 Tech Spec 3.2.5 (ITS 3.4.1) and 3.4.3.2 (ITS 3.4.11)

4.0 Actuation

4.1 **High** Pressure-1-RC-PC-1445C
4.2 Low Pressure-1-RC-PC-1445B



PRIOR TO USING FOR DESIGN WORK CHECK DMIS FOR WORK PENDING


Mn. Cont. Rm.
Elev. 0276
1-EI-CB-01
B.Bd. 1-1
26
DV
A/M
PC-1444J
Hagan *
6629D49G03
Inverse Acting
25
DW
A/M
~~PC-1444H~~
Hagan *
6629D9G03
Direct
24
DX
A/M
PC-1444G
Hagan *
6629D9G03
Direct
REF:FE-3D

010K5.02 001

Unit 1 is at 100% power. A Pressurizer PORV is leaking to the PRT. During this process the __________ of the leaking steam stays constant while the moisture content of the steam will __________

A.✓ enthalpy; decrease
B. enthalpy; increase
C. entropy; decrease
D. entropy; increase

A. Correct. The process is isenthalpic and the moisture content of the steam will decrease during the process, possibly becoming superheated.

B. Incorrect. Although the process is isenthalpic, the moisture content of the steam will decrease

C. Incorrect. The process is isenthalpic not isentropic.

D. Incorrect. The process is isenthalpic not isentropic and the moisture content of the stearn will decrease.

Knowledge of the operational implications of the following concepts as they apply to the Pressurizer Pressure control system Constant enthalpy expansion through a valve

(CFR: 41.5 / 45.7)

Mew Question

References:
Objective 3603 ..om Pressurizer Pressure ani Protection Syst m study guide

Level (RO/SRO):	RO	T	2
Group:	1	Importance Rating:	2.6/3.0
Type (Bank/Mod/New):	NEW	Cog (Knowledge/Comp):	COMPREHENSION
Reference (Y/N):	N	Last Exam(Y/N):	N

Topic 1.3.10: Throttling Processes
3603

1.3.10a. Objective

Given the initial conditions of a fluid, calculate its state after it undergoes a throttling process.

1.3.10b. Content

1. A throttling process is an isenthalpic depressurization of a fluid.

   1.1. Valves. orfices, and pipe breaks are modeled as an isenthalpic depressurization.

   1.2. For throttling processes $h_{initial} = h_{final}$ the specific enthalpy initial equals the specific enthalpy final.

   1.3. The state of the fluid immediately after the throttling process can be found by knowing that the specific enthalpy stays constant.

   1.4. Example: 2400 psia, 100% quality steam escapes from an open pressurizer PORV to a tailpipe pressure of 40 psia. What is the state of the steam in the tailpipe?

   $$h_{initial} = h_{final} = 1103.7\ \frac{ETU}{lbm}$$

   1.5. The escaping fluid must be a wet vapor because 1103.7 $\frac{ETU}{lbm}$ is between 236.1 $\frac{BTU}{lbm}$ and 1169.8 $\frac{BTU}{lbm}$ which are h_f and h_g for 40 **psia.**

   1.6. Consequently, the state of the steam may be found from the wet vapor equation.

   $h_{wv} = h_f + xh_{fg}$

   $1103.7 = 236.1 + x(933.6)$

   $x = 92.93\%$

1.7. The temperature of the tailpipe fluid must be 267.25°F, which is the saturation temperature for 40 psia.

1.8. The solution may also be achieved by using the Mollier diagram.

1.9. An isenthalpic process is represented by a horizontal line.

1.10. $1103.7\ \frac{BTU}{lbm}$ intercepts the **40** psia line at 7% moisture

2.11. If the initial condition of the fluid is steam, the exit fluid can be a wet vapor or a superheated vapor.

1.12. Example: 1100 psia, **100%** quality steam escapes from a steam generator safety valve to atmospheric pressure, **14.896** psia. What is the condition of the escaping fluid before it mixes with the atmosphere?

$$h_{initial} = h_{final} = 1189.1\ \frac{BTU}{lbm}$$

1.14. The escaping fluid must be superheated since $1189.1\ \frac{BTU}{lbm}$ is greater than $1150.5\ \frac{BTU}{lbm}$.

1.14. The temperature of the escaping steam may be found by interpolation on the super heat table

$$T_f = 250°F + \left(\frac{1189.1-1168.8}{1192.6-1168.8}\right)50°F = 292.6°F$$

OR

from the Mollier diagram T_f = 292°F.

1.15. If the condition of the Ruid is saturated liquid or subcooled liquid, the fluid after the throttling process is a wet vapor or subcooled liquid.

1.16. Example: Saturated liquid is at the critical point (3208.2 psia, 705.47°F). The liquid undergoes a throttling process to ,08865 psia. What is the condition of the fluid if no ice is formed?

$$h_f = h_i = 906\ \frac{BTU}{lbm}$$

$906 = .003 + x(1075.5)$

$$x = \frac{906 - .0003}{1075.5} = .842$$

1.17. Wet vapor of **84.2%** quality at a temperature of 32.018°F, or get 16% moisture from the Mollier.

2.18. 906 $\frac{BTU}{lbm}$ is most specific enthalpy that liquid water can have, and regardless of the extent of the depressurization, the final fluid is a wet vapor.

1.19. Actually, if you want to get technical, depressurizing below .08865 psia produces an icy vapor.

1.20. Depressurizing a subcooled liquid results in **a** wet vapor if the final pressure is the saturation pressure for the temperature **of** the fluid.

1.21. If the fluid's pressure remains above the saturation pressure, the fluid remains a subcooled liquid.

1.22. A useful approximation for subcooled liquids is that their speafic enthalpy is equal to the specific enthalpy of a saturated liquid at the same TEMPERATURE.

1.23. Example: **2250** psia, **548°F** subcooled liquid is throttled to a downstream pressure of **308.78** psia. **What** is the condition **of** the fluid?

$h_i = h_{sat\ 548°F} = \mathbf{546.9}\ \frac{BTU}{lbm}$,

1.24. It must be **a** wet vapor **at** 308.78 psia **since** 546.9 **is** between 396.9 and **1203.1.**

$h_{initial} = h_{final}$

$x = .186$

1.25. **The** downstream fluid is a wet vapor of **18.6%** quality at a temperature of **420°F.**

1.46. If a heat exchanger is placed upstream **of** the throffling valve, what is the maximum temperature to which the fluid **must be** cooled **so** that **no** flashing occurs during the depressurization to 308.78 psia?

Answer: 419.999°F

01iEG2.4.49 001

The following conditions exist on Unit 1:

- The unit is operating at 100% power
- Annunciator J-A6, RX CONT SUMP HI LEVEL, has just illuminated
- Pressurizer level is 30% and decreasing
- Pressurizer pressure is 1900 psig and decreasing.

Based on these indications, which ONE of the following lists the correct order of the OATC's next actions?

A.✓ Trip the reactor, trip the turbine, verify emergency busses energized, then initiate SI.
B. Trip the turbine, trip the reactor, verify emergency busses energized, then initiate SI.
C. immediately initiate SI, then verify the reactor and turbine are tripped and emergency busses are energized.
D. Trip the reactor, trip the turbine. verify emergency busses energized, an SI will not be required.

A. Correct. These are the immediate actions, in order, for 1-E-0. A reactor trip signal should have been generated on low pressurizer pressure (1870 psig and rate sensitive)

B. Incorrect. The reactor must be tripped first per the immediate actions of 1-E-0

C. Incorrect. The reactor and turbine should be tripped before SI is initiated per the immediate actions of 1-E-0.

D. Incorrect. SI would be required under these conditions. Pressurizer pressure is decreasing rapidly and the SI setpoint for low low pressurizer pressure is 1780 psig, by the time the other actions are performed an SI signal would have been generated.

barge Break LOCA
Ability to **perform without reference** to **procedures those actions that require immediate operation** of **system components and controls.**

Bank question

References:
Objective 12020 from study guide on Emergency Procedures
OPAP-0002 6.4.4e

Level (RO/SRO):	RO	Tier:	1
Group:	1	Importance Rating:	4.0/4.0
Type (Bank/Mod/New):	BANK	Cog (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	N	Last Exam(Y/N):	N

2. Attachments are often used for procedure actions that can be accomplished by an Operator who is not part of the Control Room Shift Team.

6.4.4 Implementation **of EOPs** After Initial Entry

a. Communication for EOP implementation shall occur between the cognizant Shift Team members and the EOP Reader. No other personnel shall become involved in discussions, debates, or questions unless prompted by the cognizant SRO.

b. If an EOP contains **a** Continuous Action Page. it shall be frequently monitored. It contains information that may be applicable at any step of the procedure.

c. The Shift Team members involved in the performance of the EOPs shall accurately convey **to** the EOP Reader the status **of** equipment and conditions.

d. Safety System Resets

1. If the system (e.g., Safety Injection, Containment Isolation) has not been reset once, then the Reactor Operator shall reset the system when directed by procedure with concurrence from the cognizant SRO whether it has actuated or not. ***This*** action **is** not required if the procedure step includes the words **if** necessary.

2. Once. the system has been reset once, it is not necessary to reposition the reset switches every time a procedure directs a reset. Verify that the applicable status light still reflects the reset condition. If it does not, another reset shall be performed.

e. Immediate **Action** Steps

1. EOP Immediate Action Steps should be performed from memory. Immediate Action Steps are designated by brackets around the individual step number in the applicable procedures (e.g., [1.])

2. The first four Immediate Action Steps **of** E-0 and the Inmediate Action Steps **of** FR-S.1 shall be performed in sequence or sequentially. **All** other Immediate Action steps do not have specific step sequence requirements.

3. Immediate Action Steps that have been performed shall be verified when the EOP is entered.

Topic 2.1.1: 1-E-0 (Reactor Trip or Safety Injection) 12020

2.1.1a. Objective

List the following information associated with I-E-0. "Reactor Trip or Safety Injection."

- Purpose
- Modes of applicability
- Entry conditions
- Immediate operator actions
- Major action categories

2.1.1b. Content

1. E-0, Reactor Trip/Safety Injection, verifies proper response of the automatic protection systems following manual or automatic actuation of reactor trip or safety injection
    1.1. Additionally, the procedure guides the operator in the assessment of plant conditions and directs the operator to the optimal recovery guidelines based on the event diagnosis.

2. If the unit is operating in modes 1, 2, or 3 and a reactor trip or safety injection occurs, E-0 will be applicable.

3. E-0 is the entered as a result of any of the following conditions:
    3.1. A reactor trip has occurred or a reactor trip is required.
    3.2. A reactor trip/safety injection has occurred or a reactor trip/safety injection is required.
    3.3. A transition is made from another plant procedure.

4. The first four steps of E-0 are immediate operator action steps.
    4.1 These steps should be performed from memory prior to reading the procedure

4.2. The immediate operator action steps of E-0 are limited to time critical actions that verify automatic response of the reactor protection system.

4.3. Although the immediate operator action steps of E-0 do not need to be memorized verbatim, they do need to be memorized to the extent that the operator can complete the intent of each step.

4.4. It should be noted that the immediate operator action steps of E-0 must be performed in the sequential order in which they are written.

4.5. The first step of E-0 is to "**VERIFY** REACTOR TRIP."

4.5.1. The reactor is manually tripped and the operator verifies the following indications

4.5.1.1. Reactor trip and bypass breakers are open

4.5.1.2. Rod bottom lights are lit

4.5.1.3. Neutron flux is decreasing

4.5.2. **If** the reactor is not tripped the operator is directed to enter **1-FR-S.1,** Response to Nuclear Power Generation/ATWS

4.6. The second step is to "**VERIFY TURBINE** TRIP."

4.6.1. The turbine is manually tripped. the reheaters are reset, and the operator verifies the following indications

4.6.1.1. Turbine stop valves are closed

4.6.1.2. Generator output breaker opens

4.6.2. If the turbine did not trip the operator performs the following actions **as** required

4.6.2.1. Put both EHC pumps in PLT

4.6.2.2. Manually runback the turbine

4.6.2.3. Close the MSTVs and Bypass valves

4.6.3. If the generator output breaker did not open (after **30 sec T.D.)** the operator should manually open the output breaker and the exciter field breaker

4.7. The third step is to "VEKIFY BOTH **AC EMERGENCY** BUSSES ENERGIZED."

4.7.1. **If** no emergency bus is energized, the operator should try to restore one

4.7.2. If no emergency bus can be restored the operator is directed to enter 1-ECA-0.0, Loss of All AC Power

4.8. The fourth step is to "CHECK SI STATUS."

4.8.1. The operator checks if SI is actuated by checking if the **LHSI** pumps are running or if any SI first-out annunciator is lit

4.8.2. If SI has been automatically actuated the operator should perform a manual SI actuation

4.8.3. If SI has not been automatically actuated the operator should determine if SI is required by checking the parameters that would initiate an automatic SI actuation

4.8.4. If SI is required the operator should manually initiate SI

4.8.5. If SI is not required the operator is directed to enter 1-ES-0.1, Reactor Trip Response

5. The major action categories of E-0 are as follows:

5.1. Verify automatic actions as initiated by the protection and safeguards systems.

5.2. Identify appropriate Optimal Recovery Guideline.

5.3. Shut down unnecessary equipment and continue trying to identify appropriate optimal recovery guideline.

Topic 2.1.2: Adverse Containment Criteria 12450

2.1.2a. Objective

Explain the following concepts concerning the use of adverse containment criteria.

- Purpose of the adverse containment criteria
- How adverse cuntainment criteria are denoted within a procedure
- Conditions that required the use of the adverse containment criteria
- When the use of adverse containment criteria can be discontinued

2.1.2b. Content

011K6.04 001

The following conditions exist:

•

Assuming no operator action, which ONE of the following describes the effect this malfunction will have on the pressurizer level control system?

A. The pressurizer will stabilize at 64.5% level.
B. The unit will trip on high pressurizer level.
C. The pressurizer will Bill solid with no reactor trip.

Knowledge of the effect of a **loss** or malfunction of the following will have on the PZR LCS: Operation of PZR level controllers

North Anna bank question 2269

References:
Objectives 10654 and 8839 from Pressurizer Control and Protection study guide
NCRODP module 74 Pressurizer Control and Protection (page 40)

Level (RO/SRO):	KO	**Tier:**	2
Group:	2	**Importance** Rating:	3.1/3.1
Type (Bank/Mod/New):	BANK	**Cog** (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	N	**Last Exam**(Y/N):	N

O11K6.04

Now** have **the students solve the following problem:

Unit 1 is operating at 75% powerwith the following conditions:

RCS Tave is 575°F.

PRZR level is 64.5 %

RCS pressure is 2255 **PSIG** and rising

If **the** OATC were to take manual control of the PRZR pressure master controller and raise its output to 50%, how would PRZR pressure respond?

Answer: Pressure would continue to rise since **the** backup heaters are still energized **due** to PRZR level being 5% > than program.

Topic 2.2.7: PRZR Level Control Response to Median-Select Failure 10654

2.2.7a. Objective

Describe the response of the Pressurizer Levei Control System to a failure of median-select T_{avg}.

2.2.7b. Content

1. Assuming an initial power of **50%,** if **median-select T_{avg}** were to fail high:
   1.1. Charging flow **would** automatically increase.
   1.2. PRZR level would increase to ***64.5%.***
   1.3. Charging flow would return ***to*** the previous value, and level would stabilize at **64.5%.**

2. Assuming an initial power *of* **50%,** if median-select T_{avg} were to fail low:
   2.1. Charging flow would automatically decrease.
   2.2. PRZR level **would** decrease to **28.4%.**
   2.3. Charging flow would return to the previous value, and level would stabilize at 28.4%.

Topic 2.2.2: PRZR Level Control Functions
8839

2.2.2a. Objective

Explain the following concepts associated with pressurizer level control

- Purpose of the pressurizer level control switch **IILM-459**
- How the pressurizer level setpoint is determined

2.2.2b. Content

NA-DW-5655D33, SH 1I may be used to explain.

1. PRZR level control switch **1/LM-459** allows selection of two level transmitters (protection) to feed the control channels **LC-459** and LC-460
   1.1. 461/460, 459/460, 459/461 feeds **LC-459** and 460 respectively

2. The PRZR program level setpoint is based on median/high select T_{avg}.
   2.1. At no-load T_{avg} (547°F) program level is **28.4%.**
   2.2. It increases linearly to 64.5% at full-load T_{avg} (580.8°F).

only input to channel LC-1459, LT-1460 can only input to channel LC-1460, but LT-1461 can input to either channel. Level transmitter LT-1461 serves as a backup to both channels. The following discussion assumes the handswitch is in the 1459-1460 position.

bevel Control Channel LC-1459. The LC-1459 control channel uses an indication of actual pressurizer level from LT-1459 as well as a setpoint signal supplied from the $\Delta T/T_{avg}$ control subsystem. The setpoint signal establishes the programmed pressurizer level as a function of T_{avg}. At a T_{avg} = 547°F the programmed level signal setpoint is 25.6 percent. The programmed ievei setpoint increases linearly to a maximum setpoint of 64.5 percent for a T_{avg} = 580.8°F. Below a T_{avg} = 547°F, the programmed setpoint remains at 25.8 percent: if T_{avg} exceeds its maximum allowed value of 580.8°F, the programmed setpoint remains at 64.5 percent. The setpoint value is supplied to pen 1 of level recorder LR-1459 (LR-2459).

The output of transmitter LT-1459 and the programmed setpoint are compared in the following:

1. Comparator LC-1459D which energizes the backup heaters if actuai level rises more than 5 percent above the programmed setpoint. This feature anticipates the pressure drop caused by the excess insurge of relatively cold loop coolant. The comparator also annunciates the PRESSURIZER HIGH LEVEL BACKUP HEATERS ON alarm (window 1B-G6).
2. Comparator LC-1459E which annunciates the PRESSURIZER LOW LEVEL alarm (window 1B-8) if actual level falls more than 5 percent below the programmed level.
3. Comparator LC-1459F, discussed later in this subsection.

Comparator LC-1459C compares the actuai level signal to an internal setpoint which, when level falls to 15 percent, initiates the following:

1. Deenergizes all the heaters to ensure that they do not operate when they become uncovered due to the low liquid level in the pressurizer. The signal opens the control group breaker, and opens the backup groups main line contactors.
2. Shuts CVCS letdown isolation valves LCV-1460A, HCV-1200A, HCV-1200B, and HCV-1200C which secures letdown flow from the RC System in order to restore pressurizer level.
3. Annunciates the PRESSURIZER LOW LEVEL HEATERS OFF - LETDOWN ISOLATION alarm (window 1B-G7).

Level controller LC-1459F compares the actual level signal and the setpoint signal in order to produce a level demand signal. The controller is a PID-type controller with an associated M/A station (LC-1459G). The station is a standard M/A station except the setpoint dial is not functional. The operator can manually control the controller and produce a level demand signal. As

012K2 01 001

The power to the Reactor Protection System logic bays is supplied from 120-volt AC vital busses. Train "A" is supplied from __________, and train "B" is supplied from __________.

A. busses I and II; busses III and IV
B. busses I and III; busses II and IV
C. bus II ONLY; bus IV ONLY
D. bus I ONLY; bus III ONLY

A. Correct. Power to train A logic bay of RPS is supplied from vital busses I and II. Power to train B logic bay of RPS is supplied from vital busses III and IV.

B. Incorrect. Candidate could think that busses I and III (odd) should be together, as well as II and IV (even).

C. Incorrect. Candidate could think that only the "even" busses are used to supply the RPS logic bays.

D. Incorrect. Candidate could think that only the "odd busses are used to supply the RPS logic bays. The slave relays are supplied from these busses.

Knowledge of electrical power supplies to the following: RPS channels, components and interconnections (CFR: 41.7)

North Anna bank question 1475

References:
Objective 8963 from self-study guide for Reactor Protection

Level (RO/SRO):	RO	Tier:	2
Group:	1	Importance Rating:	3.3/3.7
Type (Bank/Mod/New):	BANK	Cog (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	N	Last Exam(Y/N):	N

Topic 2.6: Power Supplies
8963

2.6a. Objective

List the power supply to each of the following Reactor Protection System components.

- Input relay bays I, II, III, and IV
- Train A logic bay
- Train B logic bay
- Train A slave relays
- Train B slave relays

2.6b. Content

1. Input relay bays I, II, 111, and IV are powered from vital busses I, II, III and IV respectively.

2. The train "A logic bay is powered from Vital busses I and II.

3. The train "B" logic bay is powered from Vital busses III and IV.

4. The train "A" slave relays are powered from Vital bus I.

5. The train "B" slave relays are powered from vital bus III.

3: Reactor Trips and Interlocks

Topic 3.1: Instruments, Coincidences, Setpoints, and Interlocks
8964

3.1a. Objective

012K6.01 001

Unit 4 was operating at 84% power when an "A" loop hot leg RTD failed. The following plant conditions exist:

- The instrument shop requested that the channel be placed in trip immediately for troubleshooting
- The crew performed applicable actions of 1-MOP-55.74, " Delta T/T_{ave} Protection Instrumentation," to prepare for placing the channel in trip
- As the instrument technician correctiy placed 1-RC-TTS-1412B-1 (BS-1) on card C1-421 to TEST the Reactor Operator did NOT receive the expected annunciators or computer alarms.

Based on the above conditions, the crew should __________

A?' be in hot standby within 7 hours due to the possible inoperability of both trains of solid state
B. be in cold shutdown within 84 hours of the time the channel failed because the channel cannot be placed in trip
C. place the redundant bistable for overpower deltaT in the "B" or "C" Loop to TEST to ensure reactor trip reliability, then repair the " A channel within 72 hours
D. continue with 1-MOP-55.74 until completed and write a work request on 1-RC-TTS-1412B-1 (BS-1) test switch; power operations may continue

A. Correct. If the bistable switch is not operating correctly then the operability of solid state is in question and the crew should enter 3.03 per the Precautions and Limitations and Caution of the MOP. The unit must be placed in mode 3 within 7 hours.

B. Incorrect. At this point it is not the operability of the channel that is in question, but the operability of solid state, Thus 3.03 should be entered and actions taken to place the unit in hot standby within 7 hours. However, 84 hours is a plausible time for a mode 4 entry if it were applicable.

C. Incorrect. Placing more than one channel in TEST could possibly cause a unit trip. At this point the operability of the delta T/Tave channel is not the greatest concern. It is the operability of solid state protection.

D. Incorrect. At this point it is not the operability of the channel that is in question, but the operability of solid state protection. Thus 3.03 should be entered. Power operations cannot continue.

QUESTIONS REPORT
for ROQUESTIONS

Knowledge of the effect of a loss or malfunction of the following will have on the WPS: Bistables and bistable test equipment

North **Anna** bank **question 60461**

References:
1-MOP **55.74**

Level (RO/SRO):	KO	**Tier:**	**2**
Group:	**1**	Importance Rating:	2.8/3.3
Type (Bank/Mod/New):	BANK	**Cog** (Knowledge/Comp):	**COMPREIIENSION**
Keference (Y/N):	**N**	**Last** Exam(Y/N):	N

02K6.01

Dominion **NORTH ANNA POWER STATION**	PROCEDURE NO: 1-MOP-55.74 REVISION NO: 8 P-1
PROCEDURE TYPE MAINTENANCE OPERATING	UNIT NO: 1

PROCEDURE TITLE:

ΔT/TAVE PROTECTION INSTRUMENTATION

				EQP AP	

REVISION SUMMARY.

Converted to FrameMaker using Template Rev. 030.

- Incorporated DCP 01-007, Phase 2 PCS Installation and P-250 Removal -- Unit I:
  - Added DCP tu references as Step 2.3.6.
  - Deleted 'and computer alarm prints out' in Step 4.3 since no computer alarm is listed, and added '(P-250) or actuates (Phase 2 PCS)' in Steps 5.2.9, 5.3.9, and 5.4.9.
  - Added Phase 2 PCS Computer Point descriptions and designated original point descriptions for P-250 in substeps (a), (d), and (f) of Steps 5.2.9, 5.3.9, and 5.4.9.
- Made IT§ changes permanent in Section 2.2 and in Steps 4.1, 4.2, 4.3, 5.1.1, 5.1.1.b, 5.1.2, 5.1.2.b, and Cautions before Steps 5.2.3, 5.3.3, and 5.4.3. Removed ITS from Review Bar.
- Deleted Step 2.3.4, 1-AP-3.4, Loss of Vital Instrumentation Loop $\Delta T/T_{avg}$, which was replaced by this procedure. No need to reference a deleted procedure.
- **EPAR {P1}:** Incorporated **Ops** Concern **04-0153 to allow** performance **of section 5.4** in step **5.1.3.**

PROCEDURE USED:	☐ Entirely	☐ Partially	Note: If used partially. note reasons in remarks.
PROBLEMS ENCOUNTERED:	☐ NO	☐ YES	Note: if YES, note problems in remarks.

REMARKS: ______________________________

____

____

____

____

______________________________ (Use back for additional remarks.)

SHIFT SUPERVISOR:	DATE:

TABLE OF CONTENTS

1.0 PURPOSE

To provide instructions to safely place the $\Delta T/T_{ave}$ Protection instrument channels in TEST.

2.6 REFERENCES

2.1 Source Documents

2.1.1 DCP 88-01, Changed Unit 1 Annunciators

2.1.2 QDR N8.24, Weed RTDs

2.2 Technical Specifications

2.2.1 Tech Spec 3.0.3

2.2.2 Tech Spec 3.3.1, Reactor Trip System Instrumentation, Condition "E"

2.2.3 Tech Spec 3.3.1, Table 3.3.1-1, Reactor **Trip** System Instrumentation, Functions 6 **and** *4*

2.2.4 Tech Spec 3.3.2, Engineered Safety Feature Actuation System, Conditions "D" and "J"

2.2.5 Tech Spec 3.3.2, Table 3.3.2-1, Engineered Safety Feature Actuation System, Functions If, 4d, **and** 8c

2.2.6 TRM 4.3, Reactor **Trip** Instrumentation Trip Setpoints, Table 4.3-1, Overtemperature AT and Overpower ΔT

2.3 Technieal References

2.3.1 Westinghouse SSPS Tech Manual

2.3.2 Westinghouse Process Instrumentation Manual and Prints

2.3.3 Instrument Department PTs

2.3.4 1-AP-3, **Loss of** Vital Instrumentation

2.3.5 DCP **89-40-1,** RTD Bypass Removal

2.3.6 DCP 01-007, Phase 2 PCS Installation and P-250 Removal - Unit 1

2.4 Commitment Documents

2.4.1 **CTS** Assignment **02-99-1801-003,** Tech Spec Change **290**

3.0 INITIAL CONDITIONS

The applicable steps of 1-AP-3, **Loss** of Vital Instrumentation, have been completed.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 Tech Spec **3.3.1,** Reactor Trip System Instrumentation, Condition "E":

With the number of **OPERABLE** channels one less than the Total Number of Channels, STARTUP and POWER OPERATION may proceed provided the Following conditions are satisfied

4.1.1 The inoperable channel is placed in the tripped condition within **72** hours.

4.1.2 The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed **for** up to 12 hours for surveillance testing of other channels per Specification **3.3.1,** Reactor **Trip** System Instrumentation, Condition "E".

If the conditions are not satisfied in the time permitted, place the unit in Mode **3** in ***6*** hours.

Applicability: Mode **1 or** 2

4.2 Tech Spec **3.3.2,** Engineered Safety Feature Actuation System, Condition "D":

With the number of OPERABLE channels one less than **the** Total Number of Channels, STARTUP and POWER OPERATION may proceed provided the following conditions are satisfied:

4.2.1 The inoperable channel is placed in the tripped condition within 72 hours.

4.2.2 The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to **12** hours for surveillance testing of other channels per Specification 3.3.2, Engineered Safety Feature Actuation System, Condition "D".

Applicability: Mode **1, 2,** or **3** with trip function not blocked above **P-12.**

→ **4.3** IF an indicated annunciator is NOT LIT AND **the** associated computer point does not print out **(P-250)** or alarm (Phase **2 PCS), THEN** verify the correct switch was placed in TRIP. **IF** the correct switch was in TRIP and no actuations occurred, **THEN** the operability of both trains of the Solid State Protection System is questionable and the unit must **be** placed in the mode required by Tech Spec 3.0.3.

4.4 IF the Unit **is** in **Mode 4, 5,** or **6, THEN** $\Delta T/T_{ave}$ Protection Instrument **MAY be** placed in Test as desired by the Shift Supervisor.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours;

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicabiltty shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specific condition in the Applicability for an unlimited period of time,

(continued)

013K4.10 001

Unit 1 was operating at 95% power when a LOCA occurred, resulting in a reactor trip. The following plant conditions exist:

- Safety injection has actuated due to low-Sow pressurizer pressure
- "B" steam generator level is now 10% NR
- The crew has placed the control switch for the "38" auxiliary feedwater pump in PTL due to severe pump vibration.

Which ONE of the following lists the signal(s) that will prevent the "3B" auxiliary feedwater pump from stopping?

A. Safety injection and ATWS Mitigation System Actuation Circuit (AMSAC) only
B. Safety injection only
C. Low-low steam generator level Only
D. Safety injection, ATWS Mitigation System Actuation Circuit (AMSAC), and low-low steam generator level

Knowledge of ESF design feature(s) and or interlock(s) which provide for the following: Safeguards equipment control reset

(CFR: **41.7)**

North Anna bank question 5752

References:
Objective 6036 from self-study guide for Auxiliary Feedwater

Level (RO/SRO):	RO	**Tier:**	**2**
Group:	**1**	**Importance Rating:**	3.3/3.7
Type (Bank/Mod/New):	BANK	**Cog** (Knowledge/Comp):	**KNOWLEDGE**
Reference (Y/N):	N	**Last** Exam(Y/N):	N

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Topic 2.4: Securing a Motor-Driven AFW Pump Following an Auto Start 6036

2.4a. Objective

Describe the actions associated with manually stopping the motor-driven auxiliary feedwater pump following an automatic start.

2.4b. Content

1. If an SI or AMSAC signal is present, regardless of which signal actually started the pump, then SI and/or AMSAC, as applicable, must be reset before the pump can be manually stopped.

2. If no SI or AMSAC signal is present, then the pump can be manually stopped without clearing the AUTO-START signals, however if the control switch is returned to AUTO, the pump will re-start.

Use ESK-5AA to explain.

Topic 2.5: Stopping a Motor-Driven AFW Pump from the Aux. S/D Panel 10173

2.5a. Objective

List the operator actions required for manually stopping a motor-driven auxiliary feedwater pump from the auxiliary shutdown panel following an automatic start.

2.5b. Content

Display Aux. Shutdown dwg. CB476

Display Aux. Shutdown dwg. CB477

1. If an SI signal is present, regardless of which signal actually started the pump. then SI must be reset before the pump can be manually stopped.

Topic 7.1: ATWS Mitigation System Actuation Circuit 6553

7.1a. Objective

Explain the following concepts associated with the ATWS Mitigation System Actuation Circuit (AMSAC).

- Purpose
- Automatic actuation signals
- How the reactor is tripped when actuated
- Power level when AMSAC is normally placed in service
- Power level when AMSAC is normally removed from sewice

7.1b. Content

1. The purpose of the AMSAC system is to initiate a turbine trip, a reactor trip. and auxiliary feedwater system flow upon detection of an ATWS event and to prevent a loss of heat sink with a failure of the turbine to trip.
   1.1. An ATWS event is described as a postulated operational occurrence or design basis event coincident with a failure of the Reactor Protection System to shutdown the reactor.

2. To prevent RCS overpressurization, the main turbine must be tripped within 30 seconds and AFW delivering flow within 60 seconds.
   2.1. The AMSAC is initiated when:
      2.1.1. 2/2 first stage pressures > 38 percent within the last 6 minutes (C-20)
      2.1.2. 2/3 SG level transmitters in 2/3 steam generators ≤ 13% for > 27 seconds.

3. AMSAC trips the reactor by opening the supply breakers for the rod drive MG sets.

4. AMSAC is placed in service prior to exceeding 40% power.

014K4.06 001

The following plant conditions exist on Unit 1:

- Reactor startup is in progress
- Control bank D rods are at 25 steps
- All other rods are at expected position for this plant condition
- While taking critical data, Control Bank A, Group 2 rods drop to the bottom of the core due to a mechanical failure.

As the rods drop, which ONE (1) of the following annunciators would receive an alarm input signal for this event?

A✓ CMPTR ALARM ROD DEV/SEQ
B. NIS PWR RGE HI Φ RATE RX TRIP
C. ROB BANK A LO/LO-LO LIMIT
D. NIS PR LWR DEB DEV-DEF <50%

A. Correct. **This** annunciator should alarm when rods are ≥24 steps from bank position below 50% power.

B. Incorrect. This annunciator/trip is caused by 2/4 power range channels 4 *5%* in 2 seconds. This reactor is not in the power range at this time. This answer could be chosen if candidate remembers that multiple rod drops are a probable cause for this annunciator. It is a symptom or entry condition of 1-AP-1.2, " Dropped **Rod.**"

C. Incorrect. This annunciator is driven from the P/A converter. This drives the step counters. The alarm comes off the step counters instead of the IRPIs. If the examinee doesn't make this distinction they would choose this answer.

D. Incorrect. Although power is < 50%, the reactor is currently not in the power range. This is an alarm that can be caused by rod misalignment, making this a plausible distracter. It is a symptom or entry condition of 1-AP-1.2, " Dropped Rod."

Knowledge of Rod Position Indication design feature(s) and or interlock(s) which provide for the following:

Individual and group misalignment

(CFR: 41.7)

INPO bank question from Prairie Island 2

The following plant conditions exist on Unit 1:

Reactor startup in progress
Control Bank D rods are at 25 steps
All other **rods** at expected position for given plant conditions.

While taking ICRR data, the Control Bank **A** Group 2 rods drop from their present position to the bottom due to a mechanical failure.

As the rods drop which of the following would be the FIRST annunciator to receive an alarm input signal for this event?

a. "COMPUTER ALARM ROD DEVIATION/SEQUENCING"
b. "FLUX RATE REACTOR TRIP"
c. "CONTROL BANKS LOW LIMIT"
d. "ROB AT BOTTOM"

Answer: a

References:
Annunciator responses for A-FI, D-E4, A-H1, A-C8
Entry conditions for 1-AP-1.2, "Dropped Rod."

Level (RO/SRO):	RO	Tier:	2
Group:	2	Importance Rating:	3.4/3.7
Type (Bank/Mod/New):	BANK	Cog (Knowledge/Comp):	COMPREHENSION
Reference (Y/N):	**N**	Last Exam(Y/N):	N

Questions Marked for Collection

..014000.K4.06 8/16/2002

WEC Prairie Island 2

The following plant conditions exist on Unit 1:

Reactor startup in progress
Control Bank D rods are at 25 steps
All other rods at expected position for given plant conditions.

While taking ICRR data, the Control Bank A Group 2 rods drop from their present position to the bottom due to a mechanical failure.

As the rods drop which of the following would be the FIRST annunciator to receive an alarm input signal for this event?

"COMPUTER ALARM ROD DEVIATION/SEQUENCING"

"FLUX RATE REACTOR TRIP

"CONTROL BANKS LOW LIMIT"

"ROD AT BOTTOM"

CMPTR ALARM
ROD DEV/SEQ

1.0 Probable Cause

1.1 Dropped rod

1.2 Rod position error signal or indication fault as indicated by any of the following alarms:

NOTE: This annunciator has multiple reflash capabilities.

a. Movement of any Control Bank with the Shutdown Banks less than fully withdrawn.

b. Any improper control rod overlap sequence of control rod bank positions NOT 128 steps apart for partially withdrawn banks.

c. Any Rod Position indicating >10 steps deviation from bank.

d. Any Rod Position indicating >12 steps deviation from bank.

e. Any Rod Position indicating >24 steps deviation from bank, below 50% Power.

f. >12 steps deviation for ≥30 minutes total accumulated time in the previous 24 hour period, below 50% Power.

g. >12 steps deviation for ≥60 minutes total accumulated time in the previous 24 hour period, below 50% Power.

1.3 Rod Position Indication System in TEST.

1.4 IRPI drift caused by changes in Containment temperature or ventilation.

1.5 IRPI drift caused by changes in Reactor Coolant System Temperature.

2.0 Operator Action

2.1 IF IRPI drift as described in steps 1.4 or 1.5 is NOT the cause of the Annunciator, **THEN** place Control Rod Mode Selector Switch in MANUAL.

2.2 Determine if rod(s) misposition is causing the alarm by monitoring the following:

- IRPIs
- Unit 1 PCS alarm messages
- Power Range NIs, including computer calculated tilt
- Tavg
- Delta **Flux** indicators

2.3 IF a shutdown is required AND a Flux Map was NOT performed to determine actual rod position, THEN declare the Rod INOPERABLE AND increase the minimum required Shutdown Margin by an amount at least equal to the withdrawn Rod Worth of the Rod. (Tech Spec 3.1.1)

2.4 IF alarm is caused by a mispositioned rod, THEN GO TO the applicable procedure:

. 1-AP-1.2, Dropped Rod
. 1-AP-1.3, Control Rod Out of Alignment

2.5 IF alarm is caused by erroneous IRPI indication, THEN do the following:

a. IF in Mode 2 AND increasing power, THEM step power increase.

b. Refer to Tech Spec 3.1.7 or TRM 3.1.3.

c. Notify the Instrument Department to adjust the IRPI a5 required.

2.6 IF all indications are normal, THEN declare the Unit 1 PCS Computer Rod Monitoring Program inoperable.

2.7 IF the Automatic Rod Position Deviation Alarm program is inoperable, THEN, as a good practice, compare the demand position indicators and the individual rod position indicator channels at least once per 4 hours to ensure that rod position indication is within the tolerance requirements.

2.8 WHEN the abnormal condition(s) have been corrected, THEN do the following:

a. Ensure Rod Control Mode Selector Switch is returned to AUTO.

b. Verify the Unit 1 PCS displays the message ROD MONITORING RETURNED TO NORMAL.

e. IF the Unit 1 PCS does NOT display message ROD MONITORING RETURNED TO NORMAL, THEN verify operability of the control rod position monitoring programs by performing 1-PT-20.3, Unit 1 PCS Rod Deviation Monitor Functional Test.

3.0 References

3.1 11715-ESK-10A, 10AAK
3.2 1-OP-57.1, Incore Movable Detector System
3.3 I-AP-1.3. Control Rod Out Of Alignment
3.4 1-AP-1.2, Dropped Rod
3.5 Tech Spec 3.1.1
3.6 Tech Spec 3.1.4
3.7 Tech Spec SR 3.1.4.1
3.8 Tech Spec 3.1.7
3.9 Tech Spec 3.2.4
3.10 TRM 3.1.3
3.11 CTS 02-95-2127-083, Tech Spec Review
3.12 1-PT-20.3, Unit 1 PCS Rod Deviation Monitor Functional Test
3.13 BCP 01-007, Phase 2 PCS Installation and P-250 Removal - Unit 1

4.0 Actuation

4.1 Computer Rod Supervisor Program

NIS PWR RGE HI ϕ RATE RX TRIP	> ± 5% /2 sec on 2/4 Channels

1.0 Probable Cause

1.1 RCCA ejection

1.2 Excessive load increase

1.3 Multiple dropped rods

2.0 Operator Action

2.1 IF the Reactor is tripped, THEN GO TO 1-E-0, Reactor Trip or Safety Injection.

2.2 IF the Reactor is NOT tripped, THEN verify reactor power level is normal by using alternate and redundant indications.

2.3 IF reactor power level is NOT normal OR unable to determine that the Reactor is in a safe operating condition, THEN trip the Reactor and GO TO 1-E-0, Reactor Trip or Safety Injection.

IF reactor power level is normal AND at least 2/4 NIS PR HI ϕ RATE CHNL I / II / III / IV (Panel L-E1, E2, E3, E4) status alarms are LIT, THEN begin an orderly shutdown and initiate Actions in accordance with Tech Spec 3.0.3 for inoperable Solid State Protection System.

2.5 IF all plant parameters are normal AND the alarm is due to unknown instrument malfunctions, **THEN** immediately contact Instrument Department to determine cause of alarm.

2.6 IF desired to clear the rate alarms, THEN momentarily place the Kate Mode switch in RESET on the affected Power Range drawers.

3.0 References

3.1 11715-ESK-10D, 10AAC
3.2 Westinghouse Logic NA-DW-5655D33
3.3 Tech Spec 3.0.3
3.4 1-E-0, Reactor Trip or Safety Injection

4.0 Actuation

4.1 1-NI-NC-41U
4.2 1-NI-NC-41K
4.3 1-NI-NC-42U
4.4 1-NI-NC-42K
4.5 1-NI-NC-43W
4.6 1-NI-NC-43K
4.7 1-NI-NC-44W

4.8 1-NI-NC-44K

-END-

VIRGINIA POWER
NORTH ANNA POWER STATION
APPROVAL: ON FILE

1-EI-CB-21A ANNUNCIATOR H1

1-AR-A-H1
REV. 1
Effective Date:10/25/01

ROD BANK A LO/LO-LO LIMIT

Control Bank A
c 222 Steps (bo)
< 212 Steps (bo-Lo)

1.0 Probable Cause

1.1 Rx shutdown in progress
1.2 Rod exercise test in progress
1.3 Failure of Rod Control system

2.0 Operator Action

2.1 Verify alarm by checking Group Step Counters and IRPI

2.2 Alarm is normal for reactor shutdown or rod exercise

2.3 IF bank is in due to rod system failure, THEN GO TO 1-AP-1.1

3.0 References

3.1 Precautions, limitations, setpoints document
3.2 Tech Specs 3.1.1.1, 3.1.3.6 (ITS 3.1.1, 3.1.6)
3.3 1-AP-1.1, Continuous Uncontrolled Rod Motion
3.4 11715-FE-07N

4.0 Actuation

4.1 Controller module 1-RC-TC-1409D-1, 1-RC-TC-1409D-2

NIS PR LWR DET DEV -DEF <50%

Flux seen by one lower detector deviates from the average flux >2%
Alarm defeat at <50%

1.0 Probable Cause

1.1 Power range channel lower detector deviation >2% of average of Power flux
1.2 Power level <50%
1.3 Rod misalignment

2.0 Operator Action

2.1 If power is less than 50%, alarm is normal

2.2 IF power is greater than 50%, THEN do the following:
2.2.1 Do 1-PT-23 to determine if QPTR is >1.02.
2.2.2 IF QPTR >1.02, THEN enter ITS Action 3.2.4.A
2.2.3 IF QPTR <1.02 and alarm still lit, THEN declare alarm INOPERABLE and determine QPTR inaccordance with 1-PT-23 or 1-PT-23.1 at least once every 12 hours until the alarm is OPERABLE.
2.2.4 Defeat one channel at a time (at detector current comparator) to determine which channel is out of agreement
2.2.5 IF necessary, THEN readjust N.I.S Power Ranges, IF problem is not corrected, THEN refer to 1-AP-4.3.
2.2.6 If rod misalignment is suspected, THEN refer to 1-AP-1.3.

3.0 References

3.1 Westinghouse manual, Nuclear Instrumentation System
3.2 ITS 3.2.4
3.3 1-AP-4.3, Malfunction of Nuclear Instrumentation (Power Range)
3.4 1-PT-23, Quadrant Power Tilt Ratio Determination
3.5 1-AP-1.3, Control Rod Out of Alignment

4.0 Actuation

4.1 NIS Detector Current Comparator NM 603 in drawer N50

VIRGINIA POWER
NORTH ANNA POWER STATION
ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-AP-1.2	DROPPED ROD (WITH FOUR ATTACHMENTS)	9
		PAGE 1 of 7

PURPOSE

To provide instructions for recovering a dropped rod.

ENTRY CONDITIONS

This procedure is entered when a rod drops, as indicated by any of the following:

- Rod bottom light is LIT,
- Rapid decrease in Tave.
- Rapid decrease in Reactor power level.
- Rapid decrease in Pressurizer pressure and level.
- Annunciator Panel "A" G-2. RPI ROD BOT ROD DROP. is LIT.
- Annunciator Panel "A" D 4, CMPTR ALARM PR TILT, is LIT.
- Annunciator Panel "A" F-1, CMPTR ALARM ROD DEV/SEQ, is LIT.
- Annunciator Panel "A" B 7. NIS PR CHNL AVE FLUX DEVIATION. is LIT.
- Annunciator Panel "A" C-7. NIS PR UP DET DEV-DEF <50%, is LIT.
- Annunciator Panel "A" B-8, NIS PR HI FLUX RATE CH I-II-III-IV, is LIT, or
- Annunciator Panel "A" C-8, NIS PR LWR DET DEV DEF <50%, is LIT.

RECOMMENDED APPROVAL:	DATE	EFFECTIVE DATE
RECOMMENDED APPROVAL - ON FILE		
APPROVAL:	DATE	
APPROVAL ON FILE		

015AA2.08 001

A component cooling water mallfunction **has** caused reduced cooling flow to the Reactor Coolant Pumps (RCPs). The following conditions exist:

- Unit 1 is at 100% power
- RCP temperatures are as follows:

	A	B	C
Motor Bearing	191	185	190
Pump Bearing	195	200	230
Stator	225	285	245

Based on this information the crew should __________

A.✓ Trip the reactor, then trip the "C" RCP due to high pump bearing temperature
B. Trip the reactor, then trip "A" RCP due to high motor bearing temperature
C. Immediately trip the "C" RCP due to high pump bearing temperature
D. Continue to monitor RCP temperatures, no temperatures meet the RCP trip criteria

A. Correct. The "C" RCP pump bearing temperature **is** > 225 degrees. The reactor must be tripped before securing the "C" RCP.

B. Incorrect. The "A" RCP motor bearing temperature, though higher than the other two motors, is not greater than the RCP trip criteria of 195 degrees.

C. Incorrect. Though the "C" WCP should be tripped due to high pump bearing temperature, the reactor should be tripped first.

D. Incorrect. The "C" RCP pump bearing temperature is greater than the RCP trip criteria of 225 degrees.

Ability to determine and interpret the following as they apply to RCP Malfunction: When to secure RCPs on high bearing temperature

(CFR: 41.10 / 43.5 / 45.13)

Modification of Cook question to fit KA

..000015.A2.02 12/9/2002
WEC Cook 1

A Component Cooling water leak inside containment has caused reduced flow to the RCPs. The following conditions exist:

-Unit 1 is at 100% power.

-Temperatures / RCP #

	11	12	13	14
Motor Bearing	196 F	178 F	189 F	173 F
bower Bearing Water	195 F	184 F	201 F	184 F
Seal Leakoff	**187F**	176 F	177 F	179 F

-Ann 107 Drop 52, RCP Vibration High - NOT LIT

Which ONE of the following set of actions must be taken?

Immediately Trip the Reactor, then trip RCP#11.
Open QRV-150, No. 1 Seal Bypass Valve.
Perform a rapid Plant Shutdown and stop RCP#13 within **30** minutes.
Immediately Trip the Reactor, then trip RCP#13.

References:
Objective 11659 from study guide on Abnormal Procedures.
1-AP-15, "**Loss** of Component Cooling."
Precautions and limitations from 1-OP-5.2

Level (RO/SRO):	RO	**Tier:**	1
Group:	1	**Importance** Rating:	3.4/3.5
Type (**Bank/Mod/New**):	MODIFIED	Cog (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	N	**Last** Exam(Y/N):	N

15AA2.08

Questions Marked for Collection

..000015.A2.02 12/9/2002

WEC Cook 1

A Component Cooling water *leak* **inside containment has caused reduced flow to the RCPs. The following conditions exist:**

-Unit 1 is at 100% power.

-Temperatures / RCP #	11	12	13	14
Motor Bearing	196 F	178 F	189 F	173 F
Lower Bearing Water	195 F	184F	201 F	184 F
Seal Leakoff	187 F	176 F	177 F	179 F

-Ann **107 Drop** 52, RC**P** Vibration High - NOT **LIT**

Which ONE **of the following** set of **actions must be taken?**

Immediately Trip the Reactor, then trip RCP#11.

Open QRV-150, No. 1 **Seal** Bypass Valve.

Perform a **rapid Plant Shutdown and stop** RCP#13 within **30 minutes.**

Immediately Trip the Reactor, then trip RCP#13.

4.5 The first RCP started will initiate forced flow in **all** non-isolated RCS Coops, due to reverse flow.

4.6 The RCP Trip **Criteria** are as follows:

- Hot and Cold Leg Isolation Valves for RCP being started open in coincidence with RCP loop flows <u>NOT</u> increasing within **30** seconds after closing breaker
- RCP starting current <u>NOT</u> decreasing within 30 seconds after breaker closure
- RCP proximity vibration greater **than** 20 mils
- RCP seismic vibration greater than **5 mils**
- Number 1 Seal ΔP **less** than 200 psid
- Number I Seal Leakoff flow Iess than allowed by the curve in Attachment 1
- Number 1 Seal Leakoff flow **is** ≥ 5.9 gpm
- RCP Motor Bearing temperatures greater **than** 195°F
- RCP Lower Seal Water Bearing (Pump Bearing) temperature greater **than** 225°F
- RCP Stator Winding temperature greater than 300°F
- Loss of Seal Injection <u>AND</u> CC to RCP Thermal Barrier

4.7 <u>IF</u> after a pump start, Number **1** Seal ΔP **is** rapidly decreasing <u>AND</u> it **is** imminent that Number **1** Seal ΔP will decrease to less than 200 psid. <u>THEN</u> the affected RCP <u>MUST</u> **be** stopped when the Number **1** Seal AP reaches 240 psid. This will ensure enough **seal** flow is availabie during pump coastdown.

4.8 **Start** only ONE RCP **at** a **time.**

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*10.__	MONITOR RCP TEMPERATURES: • Motor hearing temperature LESS THAN 195°F • Pump radial bearing temperature - LESS THAN 225°F • Stator winding temperature - LESS THAN 300°F	Do the following while continuing with this procedure: a) GO TO 1 E 0, REACTOR TRIP OR SAFETY INJECTION. b) WHEN Reactor is tripped. THEN stop affected RCPs.
11._	ISOLATE LETDOWN BY CLOSING THE FOLLOWING VALVES: a) Letdown Orifice Isolation Valves: • 1 CH-HCV-1200A • 1-CH-HCV 1200B • 1-CH HCV-1200C b) Letdown Isolation Valves: • 1-CH-LCV-1460A • 1 CH-LCV-1460B	Close I-CH-LCV-1460A by placing control switch in ISO.
12._	CHECK EXCESS LETDOWN - SECURED	Secure Excess Letdown using 1-OP-8.5. OPERATION OF EXCESS LETDOWN.
13._	CLOSE 1-CK-FCV-1122. CHARGING FLOW CONTROL VALVE	Close 1-CH-MOV-1289A, Normal Charging Line Isolation Valve.
14._	CLOSE 1-CH MOV-1380. SEAL WATER RETURN ISOLATION VALVE	Close 1-CH MOV 1381. Seal Water Retuni Isolation Valve.

Topic 8.6: Information Associated with 1-AP-15 11659

8.6a. Objective

List the following information associated with 1-AP-15, "Loss of Component Cooling Water."

- Purpose of the procedure
- Modes of applicability
- Entry conditions
- Possible causes of component cooling water loss
- Means used to monitor reactor coolant pump temperatures
- Possible effect on containment penetrations

8.6b. Content

1. The purpose of AP-15 is to provide guidance to the operator in response to either a complete or partial loss of the CC System.

2. This procedure is applicable during all modes of plant operation,

3. AP-15 should be entered when the following conditions exist

   3.1. Any of the following alarms are actuated:

   3.1.1. 1G-A1, CC SURGE TK HI-LO LEVEL

   3.1.2. 16-83, CC HX 1A-1B CC OUTLET LO FLOW

   3.1.3. 1G-C3, CC HX OUTLET LO PRESS

   3.1.4. 1G-F5, COMP COOL PP 1A AUTO TRIP

   3.1.5. 1G-E8, COMP COOL PP 1B AUTO TRIP, or

   3.2. Low flow/high temperature is detected on:

   3.2.1. Excess letdown heat exchanger

   3.2.2. Reactor coolant pumps.

3.3. High temperature is detected on non-regenerative heat exchanger.

3.4. boss of the Service Water System has occurred.

4. Possible causes for a loss of component cooling water include:

   4.1. Component Cooling Water System leak.

      4.1.1. Verify that level is indicated in the component cooling water head tank.

      4.1.2. If no level is indicated in the head tank, then place the pumps in PTL, isolate the source of leakage, and refill the head tank.

   4.2. Failure of a running component cooling water pump.

      4.2.1. Start the affected unit's standby component cooling water pump or start the other unit's standby pump and cross-tie the CC Systems.

5. One of the major heat loads supplied with component cooling are the reactor coolant pumps.

   5.1. The reactor coolant pump temperatures are normally monitored using the plant computer.

      5.1.1. P-250

      5.1.2. PCS

   5.2. When a loss of component cooling is identified the operator should select "Reactor Coolant Pumps" for display on the computer's CRT.

      5.2.1. If neither computer system is available then the RCP temperatures should be swapped to the recorder.

   5.3. If any reactor coolant pump temperature limit is exceeded, the operator is directed to initiate E-Q, trip the reactor, turbine. and then the affected RCP.

      5.3.1. This is a continuous action and should be initiated whenever a RCP temperature limit is exceeded while this procedure is in effect.

6. Each unit's hot-pipe containment penetrations are equipped with cooling coils supplied by the CC System

   6.1. Long-term loss of CC could affect the integrity of these penetrations.

6.2. Engineering should be requested to evaluate the impact **of** the **loss** of component cooling on these penetrations.

01562 1.28 001

1 1-RC-P-1B ("B" RCP) amps start fluctuating and the indicated vibrations begin increasing rapidly. The backboards operator reports that "B" RCP proximity vibration indication just pegged high, then the indication went to zero. The amps continue to fluctuate.

With respect to the RCP only, which ONE of the following actions should be taken and why?

A?' Trip the "E" RCP; the vibration monitoring indication was over-ranged and therefore failed low.
5. Do not trip "B" RCP; the vibration instrument has failed and amp indication is known to occasionally fluctuate.
C. Do not trip *the* "B" RCF until predictive analysis confirms that actual vibrations are high.
D. Trip the "B" RCP; vibrations have returned to normal but fluctuating amps are an RCP trip criterion.

A. Correct. Extremely high Vibrations can cause the Bently-Nevada indication to fail either high or low. The fluctuating amps are another indication that there is a problem with the RCP. The RCP should be tripped.

B. Incorrect. Extremely high vibrations can cause the Bently-Nevada indication to fail either high or low. RCP amps are NOT known to occasionally fluctuate.

C. Incorrect. Extremely high vibrations can cause the Bently-Nevada indication to fail either high or low.

D. Incorrect. Fluctuating amps are not an RCP trip criterion. The vibrations have not returned to normal **as** the indication is reading zero.

RCP Malfunctions
Knowledge of the purpose and function of major system components and controls

North Anna bank question 3810

References:
Objective 9840 from self-study guide for Reactor Coolant system

Level (RO/SRO):	RO	Tier:	1
Group:	1	Importance Rating:	3.2/3.3
Type (Bank/Mod/New):	BANK	Cog (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	N	Last Exam(Y/N):	Y(02)

Topic 2.3.7: RCP Vibration Instrumentation
9840

2.3.7a. Objective

List the following information associated with reactor coolant pump vibration instrumentation.

- Two locations on a pump which **are** monitored for vibration
- Means available in the control room to indicate that pump vibration **is** high (SER-2-97)
- Setpoints for the RCP 1A (1B/1C) VIBRATION ALERT/DANGER alarm
- Indications of an over-ranged vibration instrument

2.3.7b. Content

1. Reactor coolant pump vibration is monitored **as** follows:

   1.1. Shaft vibration (proximity) is measured by two relative **shaft** probes mounted on top of the pump **seal** housing.

   1.1.1. The probes, one in line with the pump discharge and the other perpendicular to the pump discharge, are mounted in the same horizontal plane near the pump shaft.

   1.2. Frame vibration (seismic) is measured by two velocity seismoprobes located 90 degrees apart in the same horizontal plane and mounted at the top **of** the motor support stand.

   1.3. Proximeters and converters convert the probe signals to linear output that is displayed on meters in the control room.

   1.4. The meters automatically indicate the highest output from the relative probes and seismoprobes.

   1.4.1. Manual selection allows the monitoring **of** individual probes.

   1.4.2. Indicator lights display alert and danger limits of vibration.

2. The following control room indications can **be** used to determine abnormal reactor coolant pump vibration:

   2.1. Annunciators.

2.2. Indicators on backboards.

3. The setpoints for the RCP 1A (1B/1C) VIBRATION ALERT/DANGER alarm 1A-E5 (E6, E7) are as follows:

   3.1. Proximity—15 mils (alert); 20 mils (danger).

   3.2. Seismic—3 mils (alert); 5 mils (danger).

Discuss INPO SER 2-97, Reactor Coolant Pump Damage From a Separated Component.

4. Extremely high RCP vibrations may over-range the vibration sensor input and cause the indications to fail high or low.

   4.1. If an alarm is received followed by off-scale high or low indications, the crew should consider tripping the RCP.

Topic 2.3.8: RCP Motor and Bearing Temperature Limits 9572

2.3.8a. Objective

List the motor and bearing temperature limits that require tripping the reactor coolant pump.

2.3.8b. Content

1. The motor and bearing temperature limits that require tripping the reactor coolant pump are:

   1.1. RCP motor bearing > 195°F

   1.2. RCP pump bearing > 225°F

   1.3. RCP stator > 300°F

Topic 2.3.9: RCP Trip on Pump Bearing Versus Motor Bearing

016A4 01 001

Unit **1** is at 100% power. A failure has occurred which caused rods to step in and steam generator level controllers to decrease level to 33% in all steam generators. Operators have placed rod control in manual and are manually restoring steam generator levels to **44%.**

In order to return rods and steam generator level control to automatic the operators must first ____________

A?' swap to an operable first stage pressure channel
B. place the failed first stage pressure in trip
C. swap to an operable deltaT/T_{ave} channel
D. place the failed deltaT/T_{ave} channel in trip

A. Correct. The failed channel **is** an input to rod control and SG water level control. An operable channel will need to be selected to allow rods to be placed back in automatic and to allow automatic SG level control to control at 44%. This will be done in 4-AP-3, "Loss of Vital Instrumentation."

B. Incorrect. Examinee may realize the MOP to place failed first-stage pressure channel in trip will require these valves lo be placed in manual and feel that this needs to be performed before the controls are placed in automatic. By tech specs, the channel does not have to be placed in trip for 72 hours.

C. Incorrect. Failure of a Tave/ **Delta** T channel hi would cause rods to step in but median select Tave should **select** out the failed channel if out of range. **If** the examinee fails to realize how median select Tave works, they could choose this answer.

D. Incorrect. If the examinee doesn't realize median select Tave selects out the channel when it failed, they may feel the need to place the bistables in trip before returning controls to auto. There is no way to select out the failed channel.

Ability to **manually operate and/ or monitor in the control room:** NNI **channel select** controls

(CFR: **41.7** / *45.5* to **45.8)**

References:
Objective 12007 in Rod Control study guiide
1-AP-3, "**Loss** of Vital Instrumentation."

Level (RO/SRO):	RO	**Tier:**	2
Group:	2	**Importance Rating:**	2.8/2.9
Type (Bank/Mod/New):	NEW	Cog (Knowledge/Comp):	COMPREHENSION
Reference (Y/N):	N	Last Exam(Y/N):	N

12007

4.3a. Objective

Given a set of plant conditions, evaluate Rod Control System operations in light of the following issues

- Effect of a failure, maifunction. or loss of a related system or component on this system
- Effect of a failure, malfunction, or loss of components in this system on related systems
- Expected plant or system response based on rod control component interlocks or design features
- Impact on the technical specifications
- Response if limits or setpoints associated with this system or its components have been exceeded
- Proper operator response to the condition as stated

CONTROL SYSTEM **FAILURE ANALYSIS**

(NOTE: ALL EFFECTS ASSUME NO OPERATOR ACTION IS TAKEN.)

:OD CONTROL SYSTEM

TAVE circuit failure (Output of median select circuit)	
HIGH	EFFECT: When T_{ave} circuit fails high a large temperature error is sensed between T_{ave} and T_{ref}. Rods move in at *92* spm causing a large power mismatch. Tref decreases-.. slightly as steam pressure decreases (IMP OUT) or remains relatively stable (IMP IN). Power mismatch circuit will be unable to compensate. ACTUAL T_{ave} drops rapidly, as will pressure and pressurizer level. Plant will trip on rate-compensated low pressurizer pressure.
LOW	EFFECT: When T_{ave} fails low a large temperature error is sensed between T_{ave} and T_{ref}. This signal is initially sent to the rods and rods will begin to withdraw at maximum speed. As rods withdraw, ACTUAL Tave increases until the first reactor protection setpoint is reached or the rods are fully withdrawn.
First Stage (Impulse) Pressure (PT-1446 or PT-1447. selected channel)	
HIGH	EFFECT: At full power – T_{ref} is limited to 100% T_{ref} (580.8°). If rods not full out - the power mismatch circuit will sense a sudden increase in turbine power. Rods will withdraw at 72 spm. The rate-compensation of the power mismatch signal will decay to zero over the next several seconds. The unit stabilizes at approximately the same power level since the governor valves ramped closed to maintain a constant impulse pressure as the steam pressure increased with T_{ave}.
LOW	EFFECT: T_{ref} will fail to no-load value (547°) and cause a T_{ave}-T_{ref} mismatch. Rods will drive in at 72 spm. When T_{ave} drops the temperature error will drop to zero. The unit will trip and SI on high steam flow with low-low T_{ave} due to overshoot on rods. 3
N-44	
	EFFECT: Causes a momentary large power mismatch that will slowly decay away. The power mismatch circuit causes rods to drive in at *92* spm. Actual power decreases causing T_{ave} to decrease. The temperature error will cause the automatic rod control circuit to call for control rod withdrawal, however; since N-44 has failed high. a rod withdrawal block exists and rods will remain in the present inserted position.
OW	EFFECT: Power mismatch causes rods to drive out at 72 spm (if not already out.) The power mismatch signal will slowly decay away. Due to the outward rod motion, T_{ave} will increase. The temperature error will cause the automatic rod control circuit to call for control rod insertion, and rods will eventually drive to near the previous position.

Given a set of plant conditions. evaluate Steam Generator Level Control and Protection System operations in light of the following issues.

- Effect of a failure, malfunction, or loss of a related system or component on this system
- Effect of a failure, malfunction. or **loss** of components in this system on related systems
- Expected plant or system response based on steam generator level control and protection component interlocks or design features
- Impact on the technical specifications
- Response if limits or setpoints associated with this system or its components have been exceeded
- Proper operator response to the condition as stated

5.2b. Content

CONTROL **SYSTEM FAILURE ANALYSIS**
(NOTE: ALL EFFECTS ASSUME NO OPERATOR ACTION IS TAKEN.)

STEAM GENERATOR LEVEL CONTROL **(MFRV)**

Steam **Flow** (Selected)	
HIGH	large steam flow-feed flow mismatch is generated which causes the FRV to fully open. Steam generator level rises rapidly. High level causes a turbine trip, reactor trip, feed pump trip, and feedwater isolation.
LOW	EFFECT: If failed channel is selected, **a** large steam flow-feed flow mismatch is generated and the FRV goes closed. Level drops rapidly. Reactor will trip on feed **flow** < steam fluw mismatch coincident with low level on the unaffected channel. (The affected channel will generate a low-low SG level reactor trip.)
Steam Pressure (Density Compensated)	
HIGH	EFFECT: (Same *as* high steam flow, *if* associated steam flow channel is selected.) A **large** steam flow-feed flow mismatch is generated which causes the FRV to **fully** open. **Steam** generator level **rises** rapidly. High level **causes** a turbine trip, reactor trip, feed pump trip, anc feedwater isolation.
LOW	EFFECT: (Same as low steam flow, if associated steam **flow** channel is selected.) **If** failed channel **is** selected, a large steam flow-feed flow mismatch is generated and the **FRV** goes closed. Level drops rapidly. Reactor will trip on feed flow < steam flow mismatch coincident with low level on the unaffected channel. (The affected channel will generate a low-low **SG** level reactor trip.)
Feed Flow (Selected)	
HIGH	EFFECT (Same **as low** steam flow.) If failed channel is selected, a large steam flow-feed flow mismatch is generated and the FRV **goes** closed. Level drops rapidly. Reactor will trip on feed flow < steam flow mismatch (steam fiow **40%** > feed flow) coincident with low level (25%). (The affected channel will generate a low-low SG level reactor trip (18%)
LOW	EFFECT: (Same as high steam flow.) A large steam flow-feed flow mismatch is generated which causes the fully open. level rises rapidly. level (75%) causes a turbine trip, reactor trip, ***feed*** pump trip, and feedwater isolation.

SG Level (Channel III)	
HIGH	EFFECT: Level is a delayed dominant signal. The (indicated) high level will drive the FRV shut and the steam flow-feed flow mismatch circuit will not be able to overcome it. The SG level will rapidly decrease giving a reactor trip on feed flow mismatch (steam flow 40% > feed flow) coincident with low level (25%).
LOW	EFFECT: **Level effect drives FRV full open. SG level increases** rapidly. High-high level (75%) causes a turbine trip, reactor trip, feed pump trip, and feedwater isolation.
First Stage **Pressure (Selected)**	
HIGH	EFFECT: **Level reference limited to 100%** program **(44%) If level is lower than 100% program,** it will **be brought** to 100% **program.**
LOW	EFFECT: **Level decreases** to **0% program (33%).**

STEAM GENERATOR LEVEL CONTROL (BYPASS FRV)

N-44 (5% power)	
HIGH	EFFECT: Anticipatory signal would cause bypass FRV to open. The level dominant feature of the SGWLC will not stop increase in feedwater flow. A turbine trip will occur on high SG level.
LOW	EFFECT: **No feed forward signal present. Bypass FRV would control on level error only** and not be as responsive.

VIRGINIA POWER
NORTH ANNA POWER STATION
ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-AP-3	LOSS OF VITAL INSTRUMENTATION (WITH TWO ATTACHMENTS)	20
		PAGE 1 of 15

PURPOSE

To provide instructions to follow in the event of a loss of vital instrumentation.

ENTRY CONDITIONS

This procedure is entered when a faulty indication occurs on any of the following vital instrumentation channels:

- Reactor Coolant Flow, or
- Pressurizer Level. or
- Pressurizer Pressure Protection. or
- DELTA T/TAVE Protection. or
- Containment Pressure Protection. or
- Steam Generator Level. or
- Turbine Stop Valves Indication. or
- Turbine First Stage Impulse Pressure. or
- Turbine Auto Stop Oil Low Pressure Trip Signal, or
- Steam Flow, or
- Feed Flow. or
- Steam Pressure, or
- Station Service Bus Undervoltage. or
- Station Service Bus Underfrequency.

RECOMMENDED APPROVAL:	DATE	EFFECTIVE DATE
RECOMMENDED APPROVAL - ON FILE		
APPROVAL: APPROVAL - ON FILE	DATE	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1]__	VERIFY REDUNDANT INSTRUMENT CHANNEL INDICATION - NORMAL	IF unable to determine Reactor is in a safe operating condition. THEN GO TO 1-E-0. REACTOR TRIP OR SAFETY INJECTION.
[2]__	VERIFY STEAM GENERATOR LEVEL CONTROL PARAMETERS - NORMAL: • Steam Flow • Feed Flow • Steam Generator Level Ch III • Steam Pressure	Do the following: a) Place the associated valves in MANUAL: • Main Feed Reg Valves • Main Feed Reg Bypass Valves b) Control Steam Generator level.
[3]__	VERIFY TURBINE FIRST STAGE PRESSURE INDICATIONS - NORMAL	IF the controlling channel failed. THEN place Control Rod Mode Selector switch in MANUAL.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.__	VERIFY SYSTEMS AFFECTED BY PRESSURIZER LEVEL CHANNELS - NORMAL	
	a) Verify operable Pressurizer level channels - SELECTED	a) Do the following: 1) Place 1-CH-FCV-1122, Charging Flow Control Valve in MANUAL and control level at program. 2) Select operable Pressurizer level channels for control.
	b) Verify Letdown - IN SERVICE	b) Restore letdown using Attachment 2, LETDOWN RESTORATION.
	c) Verify Pressurizer Level Control - IN AUTO	c) Do the following: 1) Verify level restored to program. 2) Verify expected output of I-RC-LCV-1459G. Pressurizer Level Control. 3) Place 1-CH-FCV-1122. Charging Flow Control Valve in AUTO.
	d) Verify Pressurizer Control Group Heaters - NOT TRIPPED	d) Reset Pressurizer Control Group Heaters by placing control switch to START position.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.__	VERIFY BOTH TURBINE FIRST STAGE PRESSURE CHANNELS - NORMAL	IF Condenser Stem Dumps are available, THEN transfer to Steam Pressure Mode by doing the following: a) Place both STEAM DUMP INTLK switches to OFF/RESET b) Place STEAM DUMP CONTROLLER to MANUAL c) Place MODE SELECTOR switch to STEAM PRESS d) Ensure Steam Dump demand is ZERO e) Return STEAM DUMP CONTROLLER to AUTO f) Verify Steam Dump demand is ZERO g) Place both STEAM DUMP INTLK switches to ON
6.__	VERIFY OPERABLE CHANNELS SELECTED FOR ALL OF THE FOLLOWING SGWLC INSTRUMENTS: • Turbine First Stage Pressure • "A" SG Steam Flow • "B" SG Steam Flow • "C" SG Steam Flow • "A" SG Feed Flow • "B" SG Feed Flow • "C" SG Feed Flow	Do one of the following as directed by the Unit 1 SRO: • IF desired to swap ONLY the failed channel, THEN GO TO Step 8. OR • IF desired to swap ALL SGWLC channels to the same channel, THEN GO TO Step 9.
7.__	GO TO STEP 10	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.__	SWAP ONLY THE FAILED SGWLC CHANNEL AS FOLLOWS:	
	a) Swap of Turbine First Stage Pressure channel - DESIRED	a) GO TO Step 8b.
	1) Verify Rod Control Mode Selector Switch in MANUAL	1) Place Rod Control Mode Selector Switch is in MANUAL
	2) Verify Steam Dumps in one of the following conditions: • Steam Pressure Mode OR • OFF	2) Do one of the following with Unit SRO concurrence: • Place Steam Dumps in - OFF OR • IF Condenser Steam Dumps are available, THEN transfer to Steam Pressure Mode by doing the following: a. Place both STEAM DUMP INTLK switches to OFF/RESET b. Place STEAM DUMP CONTROLLER to MANUAL c. Place MODE SELECTOR switch to STEAM PRESS d. Ensure Steam Dump demand is ZERO e. Place STEAM DUMP CONTROLLER to AUTO f. Verify Steam Dump demand is ZERO g. Place both STEAM DUMP INTLK switches to ON

(STEP 8 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.	SWAP ONLY THE FAILED SGWLC CHANNEL AS FOLLOWS (Continued):	
	3) **Check** ALL Bypass Feed Reg valves in MANUAL	3) Place ALL Bypass Feed Reg valves **are** in MANUAL.
	4) Place ALL Main Feed **Reg** valves in MANUAL	
	5) Select **the** operable Turbine First Stage Pressure channel for control	
	6) Verify ALL Steam Generator channel III levels - OPERABLE	6) GO TO Step 8a7.
	a. Verify Stem Generator Levels are on program	
	b. Return the Main or **Bypass** Feed Reg Valves to AUTO. **as** required	
	7) Verify Condenser Steam **Dumps** - AVAILABLE	7) GO TO Step 8a9.

(STEP 8 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.	SWAP ONLY THE FAILED SGWLC CHANNEL AS FOLLOWS (Continued):	
	8) Place Steam Dumps in Stem Pressure Mode by doing the following with Unit SRO concurrence:	
	a. Place both STEAM DUMP INTLK switches to OFF/RESET	
	b. Place STEAM DUMP CONTROLLER to MANUAL	
	c. Place MODE SELECTOR switch to STEAM PRESS	
	d. Ensure Stem Dump demand is ZERO	
	e. Place STEAM DUMP CONTROLLER to AUTO	
	f. Verify Sream Dump demand is ZERO	
	g. Place both STEAM DUMP INTLK switches to ON	
	9) Auto Rod Control - DESIRED	9) GO TO Step 8b.
	a. Verify Tave and Tref MATCHED	
	b. Return Rod Control Mode Selector switch to AUTO	

(STEP 8 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.	SWAP ONLY THE FAILED SGWLC CHANNEL AS FOLLOWS (Continued):	
	b) Swap of Steam Flow channel - DESIRED	b) GO TO Step Rc.
	1) Verify affected Main Feed Reg valve in MANUAL	1) Place affected Main Feed Reg valve in MANUAL.
	2) Select the operable Steam Flow channel for control.	
	3) Verify affected Steam Generator Level channel III - OPERABLE	3) GO TO Step 8c.
	a. Verify affected Steam Generator Level is on program	
	b. Retuni affected Main Feed Reg Valve to AUTO. as required	
	c) Swap of Feed Flow channel - DESIRED	c) GO TO Step 10.
	1) Verify affected Main Feed Reg valve in MANUAL	1) Place affected Main Peed Reg valve in MANUAL.
	2) Select the operable Feed Flow channel for control	
	3) Verify affected Steam Generator Level channel III - OPERABLE	3) GO TO Step 10
	a. Verify affected Steam Generator Level is on program	
	b. Return affected Main Feed Reg Valve to AUTO, as required	
	d) GO TO Step 10.	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.__	SWAP ALL SGWLC CHANNELS AS FOLLOWS:	
	a) Verify Rod Control Mode Selector Switch in MANUAL	a) Place Rod Control Mode Selector Switch in MANUAL.
	b) Verify Stem Dumps in one of the following conditions: • **Steam** Pressure Mode OR • OFF	b) Do one of the following with Unit SRO concurrence: • Place Steam Dumps in - OFF OR • IF Condenser Steam Dumps are available. THEN transfer to Steam Pressure Mode by doing the following: 1) Place both STEAM DUMP INTLK switches to OFF/RESET 2) Place STEAM DUMP CONTROLLER to MANUAL **3)** Place MODE SELECTOR switch to STEAM PRESS 4) Ensure Stem Dump demand is ZERO 5) Place STEAM DUMP CONTROLLER to AUTO 6) Verify Steam Dump demand *is* ZERO 7) Place both STEAM DUMP INTLK switches to ON
	c) Check ALL Bypass Feed **Reg** valves in MANUAL	c) Place ALL Bypass **Feed** Reg valves are in MANUAL.
	d) Place ALL Main Feed Reg valves in MANUAL	

(STEP 9 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.	SWAP ALL SGWLC **CHANNELS** AS FOLLOWS (Continued):	
	e) Select ALL of the following channels to the same channel: • Steam Flow • Feed Flow • First Stage Pressure	
	f) Verify ALL Steam Ceneratvr channel III levels - OPERABLE	f) GO TO Step 9g.
	1) Verify Steam Generator Levels are on program	
	2) Return the Main or Bypass Feed Reg Valves to AUTO, as required	
	g) Verify Condenser Steam Dumps - AVAILABLE	g) GO TO Step 9i

(STEP 9 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

9. SWAP ALL SGWLC CHANNELS AS **FOLLOWS** (Continued):

h) Do one of the following with Unit SRO concurrence:

- Place Steam Dumps in Steam Pressure Mode by doing the following:

  1) Place both STEAM DUMP INTLK switches to OFF/RESET

  2) Place STEAM DUMP CONTROLLER to MANUAL

  3) Place MODE SELECTOR switch to STEAM PRESS

  4) Ensure **Stem** Dump demand is ZERO

  5) Place STEAM DUMP CONTROLLER to AUTO

  6) Verify Stem Dump demand **is** ZERO

  7) Place both STEAM DUMP INTLK switches to ON

OR

- Place Steam Dumps in Tave Mode by doing the following:

  1) Verify BOTH channels of Turbine First Stage Pressure are operable

  2) Place both STEAM DUMP INTLK switches to OFF/RESET

(STEP 9 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.	SWAP ALL SGWLC CHANNELS AS FOLLOWS (Continued):	
	3) VERIFY ANNUNCIATOR PANEL "P" E-4. C 7 PERM STM DUMP ARMED FROH LOSS OF LOAD - NOT LIT	3) Place Stem Dump Mode Selector switch to RESET.
	4) Place MODE SELECTOR switch to TAVE	
	5) Ensure Steam Dump demand is ZERO	
	6) Place both STEAM DUMP INTLK switches to ON	
	i) Auto Rod Control - DESIRED	i) GO TO Step 10.
	1) Verify Tave and Tref - MATCHED	
	2) Return Rod Control Mode Selector switch to AUTO	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: With one instrument channel lost. operations may continue only if the channel is placed in trip condition within the specified time period and the conditions of the applicable Technical Specification are met.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10.	VEKIFY OPERATION OF THE FOLLOWING INSTRUMENTS:	
	a) Reactor Coolant Flow Instnimentation indication - NORMAL	a) IF unit is in Mode 1, THEN complete 1-MOP-55.71. REACTOR COOLANT FLOW INSTRUMENTATION within 72 hours.
	b) Pressurizer Level Instrumentation indication - NORMAL	b) IF unit is in Mode 1 or 2. THEN complete 1-MOP-55.72. PRESSURIZER LEVEL INSTKUMENTATION within 72 hours.
	c) Pressurizer Pressure Protection Instrumentation indication - NORMAL	c) IF unit is in Mode 1, 2. or 3. THEN complete I-MOP-55.73. PRESSURIZER PRESSURE PROTECTION INSTRUMENTATION. Section 5.1 within one hour.
	d) Loop ΔT/TAVE Protection Instrumentation indication NOKMAL	d) IF unit is in Mode 1. 2. or 3. THEN complete 1-MOP-55.74. LOOP ΔT/TAVE PROTECTION INSTRUMENTATION. Section 5.1 within one hour.
	e) Containment Pressure Protection Instrumentation indication NORMAL	e) IF unit is in Mode 1, 2. 3, or 4. THEN complete 1-MOP-55.75. CONTAINMENT PRESSURE PROTECTION INSTRUMENTATION within 72 hours.
	f) Steam Generator Level Instrumentation indication - NORMAL	f) IF unit is in Mode 1. 2, or 3. THEN complete I-MOP-55.76. STEAM GENERATOR LEVEL INSTRUMENTATION within 72 hours.
	g) Steam Pressure Instrumentation indication - NORMAL	g) IF unit is in Mode 1, 2, or 3. THEN complete 1-MOP-55.79. STEAM PRESSURE INSTRUMENTATION within 72 hours.

(STEP 10 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10.	VERIFY OPERATION OF THE FOLLOWING INSTRUMENTS (Continued):	
	h) Steam Flow Instrumentation indication - NORMAL	h) IF unit is in Mode 1, 2, or 3. THEN complete 1-MOP-55.77. STEAM FLOW INSTRUMENTATION within 72 hours.
	i) Feed Flow Instrumentation indication NORMAL	i) IF unit is in Mode 1 or 2. THEN complete 1-MOP-55.78, FEED FLOW INSTRUMENTATION within 72 hours.
	j) Turbine Stop Valve Closure Signal Instrumentation annunciator indication - NORMAL	j) IF unit is in Mode 1, THEN complete 1-MOP-55.80. TURBINE STOP VALVE CLOSURE SIGNAL INSTRUMENTATION within 72 hours
	k) Turbine First Stage Pressure Instrumentation indication NORMAL	k) IF unit is in Mode 1. 2, or 3. THEN complete I-MOP-55.81. TURBINE FIRST STAGE PRESSURE INSTRUMENTATION. Section 5.1 within one hour.
	l) Turbine Auto Stop Oil Pressure annunciator indication - NORMAL	l) IF unit is in Mode 1. THEN complete 1 MOP-55.82. TURBINE AUTO STOP OIL LOW PRESSURE INSTRUMENTATION within 72 hours
	m) RCP Bus Undervoltage annunciator indication NORMAL	m) IF unit is in Mode 1, THEN complete 1-MOP-55.83. REACTOR PROTECTION SYSTEM INPUT FROM STATION SERVICE BUSES 2A, 2B, AND 2C UNDERVOLTAGE within 72 hours.
	n) RCP Bus Underfrequency annunciator indication NORMAL	n) IF unit is in Mode 1, THEN complete 1-MOP-55.84, REACTOR PROTECTION SYSTEM INPUT FROM STATION SERVICE BUSES 2A, 2B, AND 2C UNDERFREQUENCY within 72 hours.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11.___	VERIFY MAINTENANCE OPERATING PROCEDURE(S) - INITIATED FOR ALL FAIJLTY INSTRUMENT CHANNELS	IF the failed instrument channel was not in the mode specified. THEN do the following: a) Continue operation. b) Enter Action Statement. c) Do either of the following: • Initiate the appropriate MOP specified in Step 10 for the failed channel(s). OR • Have the I&C department place the failed channel(s) in trip. d) Refer to the applicable Technical Specifications as listed in the Reference Section of the associated MOP. e) DO NOT enter mode specified in Technical Specification until all requirements of Technical Specifications for affected channel have been met. f) Notify Instrument Department to repair faulty channel
12.	NOTIFY SUPERINTENDENT OF OPERATIONS OR OPERATIONS MANAGER ON CALL OF FAILURE	
13.___	RETURN TQ PROCEDURE IN EFFECT	

- END -

- Westinghouse SSP Tech Manual
- Westinghouse Process Instrumentation Manual and Prints
- Instrument Department PTs
- Tech Spec 3.3.1
- Tech Spec 3.3.2
- Tech Spec 3.3.4
- Tech Spec 3.3.3
- CTS 02-92-2506-001. from NPES 92-04
- 1-MOP-55.71. REACTOR COOLANT FLOW INSTRUMENTATION
- 1-MOP-55.72. PRESSURIZER LEVEL INSTRUMENTATION
- 1-MOY-55.73. PRESSURIZER PRESSURE PROTECTION INSTRUMENTATION
- 1-MOP-55.74. LOOP ΔT/TAVE PROTECTION INSTRUMENTATION
- 1-MOP-55.75. CONTAINMENT PRESSIJRE PROTECTION INSTRUMENTATION
- 1-MOP-55.76. STEAM GENERATOR LEVEL INSTRUMENTATION
- 1-MOP-55.77. STEAM FLOW INSTRUMENTATION
- 1-MOP-55.78. FEED FLOW INSTRUMENTATION
- 1-MOP-55.79. STEAM PRESSURE INSTRUMENTATION
- 1-MOP-55.80. TURBINE STOP VALVE CLOSURE SIGNAL INSTRUMENTATION
- 1-MOP-55.81. TURBINE FIRST STAGE PRESSURE INSTRUMENTATION
- I-MOP 55.82. AUTO STOP OIL LOW PRESSURE INSTRUMENTATION
- 1-MOP-55.83. REACTOR PROTECTION SYSTEM INPUT FROM STATION SERVICE BUSES 1A, 1B, AND 1C UNDERVOLTAGE
- 1-MOP-55.84. REACTOR PROTECTION SYSTEM INPUT FROM STATION SERVICE BUSES 1A, 1B, AND 1C UNDERFREQUENCY

- CTS Assignment 02-99-1801-003. Tech Spec Change 290

NUMBER	ATTACHMENT TITLE	REVISION
1-AP-3	LETDOWN RESTORATION	20
ATTACHMENT		PAGE
2		1 of 1

_____ 1 Ensure Charging Flow is at least 25 gpm.

2 Ensure the following valves are open:

_____ • 1-CH-LCV-1460A, LETDOWN ISOLATION VALVE

_____ • 1-CH-LCV-1460B, LETDOWN ISOLATION VALVE

_____ • 1-CH-TV-1204A, LETDOWN ISOLATION VALVE

_____ • 1-CH-TV-1204B, LETDOWN ISOLATION VALVE

_____ 3 Place 1-CH-PCV-1145. LETDOWN PRESSURE CONTROL VALVE. in MAN.

_____ 4 Fully open 1-CH-PCV-1145.

NOTE: To prevent potential overheating of Letdown. Charging flow may need to be increased immediately after establishing Letdown flow.

5 Open the desired Letdown Orifice Iso'lation Valve(s):

_____ • 1-CH-HCV-1200A, A LETDOWN ORIFICE ISOLATION VALVE

_____ • 1-CH-HCV-1200B, 8 LETDOWN ORIFICE ISOLATION VALVE

_____ • 1-CH-HCV-1200C, C LETDOWN ORIFICE ISOLATION VALVE

_____ 6 Adjust 1-CH-PCV-1145 to obtain 300 psig Letdown pressure as indicated on 1-CH-PI-1145, NONKEGENERATIVE HEAT EXCH OUTLET PRESS.

_____ 7 Place 1-CH-PCV-1145 in AUTO.

_____ 8 Adjust Charging and Letdown to maintain program PRZR level.

-END-

022K3.01 001

The following conditions exist.

- Unit 1 is at 100% power
- The mechanical chiller has tripped
- Containment temperature is currently 102°F.

The crew has entered 1-AP-35, "Loss of Containment Air Recirculation Cooling" and has reached the step to Check Containment Temperature <105°F. A note prior to the step explains that containment temperature should be kept less than 105°F due to

____

A.✓ EQ concerns relating to equipment life expectancy
B. EQ concerns relating to the containment structure integrity
C. concerns for containment entry stay times
D. the containment partial pressure indicators being unreliable at high ambient temperatures

A. Correct. Per the note there are EQ concerns relating to equipment life expectancy

E. Incorrect. The concern at this temperature is for equipment, not the containment structure. The candidate could choose this answer based on knowledge that the containment structure is one of the fission barriers.

C. Incorrect. While containment stay times would be affected this is not the reason for the concern with temperature. A candidate could choose this answer based on knowledge that stay times would be reduced at this high of a containment temperature.

D. Incorrect. The containment partial pressure is calculated using temperatures taken at the outlet of the containment air recirc fans. Due to the trip of the chiller these indications are declared inoperable; however the 105°F temperature has nothing to do with whether these indicators are reliable. A candidate could choose this question based on knowledge that the partial pressure indications are declared inoperable on a loss of containment cooling.

Knowledge of the effect that a loss or malfunction of the Containment Cooling system will have on the following: Containment equipment subject to damage by high or low temperature, humidity and pressure (CFR: 41.7 / 45.6)

Modified question

References:
1-AP-35

Level (RO/SRO):	**RO**	Tier:	**2**
Group:	**1**	Importance Rating:	2.9/3.2
Type (Bank/Mod/New):	MODIFIED	Cog (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	N	**Last Exam(Y/N):**	Y(00)

QUESTION: 85 (1.0)

With unit **1** at 100%power, chilled water flow **is** lost to the containment air recirc fans and the crew is unable to align **a** backup source of cooling water.

Which ONE of the following is correct concerning the potential effects of this?

a. Failure of safety-related equipment in Containment could result **during normal operation** if containment air temperature reaches 110°F.

b. Failure of safety-related equipment in containment could result **during normal operation** if containment air temperature reaches 125°F.

c. Failure of safety-related equipment in containment could result **following a MSLB if** containment air temperature reaches **110**°F **prior to the MSLB.**

d. Failure of safety-related equipment in containment could result **following a MSLB** if containment air temperature reaches 125°F **prior to the MSLB.**

ANSWER: d

Answer correct: per **TS-3.6.1.5** bases, safety-related equipment in containment could experience temperatures greater than that for which they are qualified if CTMT air temperature is **>120°F** prior to the occurrence **of** a MSLB or LOCA.	Distractors plausible: a & b – common misconception that the limits on CTMT air temperature are in place to ensure safety-related equipment in GTMT doesn't fail due to high temperature **during normal operation.** c – TS-3.6.1.5 lists two limits on **CTMT** temperature (105°F and 120°F); correct if candidate believes the basis for the 105°F limit is to ensure equipment doesn't fail due to high temperature following a MSLB.	Distractors wrong: a & b – basis for limit is to ensure equipment doesn't fail following a MSLB or LOCA. c – limit that applies to failure of equipment following **a** MSLB **or** LOCA is 120°F, no: 105°F.
K/A: 022-K3.01	Objective: 1958	Source: Mew
Reference: 1-AP-35; TS-3.6.1.5 (and bases.)	Level: Knowledge	

OUU F 3.01

VIRGINIA POWER
NORTH ANNA POWER STATION
ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-AP-35	LOSS OF CONTAINMENT AIR RECIRCULATION COOLING (WITH TWO ATTACHMENTS)	15 PAGE 1 of 7

PURPOSE

To provide instnictions for responding to a loss of Containment Recirculation Cooling caused by loss of Chiller Units, Containment Ventilation **Fans,** or Chilled Water flow.

ENTRY CONDITIONS

This procedure is entered **if** Unit **1** is in Mode 1, 2. 3 or 4 and any of tho following conditions exist:

- Increasing Containment temperature,
- Loss of either Mechanical or Steam Chiller.
- Containment Air Recirc Fan auto trip alarm or breaker disagreement indication on ventilation panel, or
- Annunciator Panel "G" B-2. CD TO AIR RECIRC CLRS HI-LO TEMP, is LIT.

RECOMMENDED APPROVAL:	DATE	EFFECTIVE DATE
RECOMMENDED APPROVAL - ON FILE		
APPROVAL:	DATE	
APPROVAL - ON **FILE**		

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE: The 1 of Containment Air Partial Pressure indicator(s) does NOT necessarily mean non-compliance with Tech Spec 3.6.	
1.__	DECLARE THE DIGITAL CONTAINMENT PARTIAL AIR PRESSURE INDICATORS INOPERABLE: • 1-LM-PI-101A • 1-LM-PI-101B	
2.__	INITIATE ATTACHMENT 2, CALCULATION OF CONTAINMENT AIR PARTIAL PRESSURE. WHILE CONTINUING WITH THIS PROCEDURE	
	NOTE: Because of Environmental Qualification concerns relating to equipment life expectancy. Containment Average Air temperature should be kept less than 105°F.	
3.__	CHECK CONTAINMENT AVERAGE AIR TEMPERATURE:	
	a) Less than 120°F	a) Enter Action Statement of Technical Specification 3.6.5.
	b) Less than 105°F	b) Notify Engineering Department.

024AK1.04 001

Unit 1 is operating at 100% power when the auxiliary building operator reports that there is a low temperature alarm locked in on an emergency borate heat trace line.

Per technical specifications lines containing borated water should be kept at a minimum of 115°F in order to __________

A?' maintain boron solubility
B. reduce O_2 to minimize corrosion
C. maintain protective oxide film to minimize corrosion
D. prevent crud bursts

A. Correct. The tech spec minimum temperature of 115 degrees is based on maintaining the boron in solution.

B. Incorrect. Since O_2 levels are a concern in the RCS this answer could be chosen.

C. incorrect. Candidate couid think that maintaining the oxide film is the reason, though this is a concern for high flows.

D. Incorrect. Since crud bursts are usually undesirable this answer could be chosen.

Knowledge of the operational implications of the following concepts as they apply to the Emergency Boration system: Low temperature limits for boron concentration

(CFR: 41.8 to 41.10 / 45.3)

Modified Surry question frorn INPO bank.

References:

TRM 3.1.1
T.S. 3.5.6 Bases
NCRODP module 22 Heat Tracing (page 14)

Level (KOISRO):	RO	Tier:	1
Group:	2	Importance Rating:	3.813.6
Type (Bank/Mod/New):	MODIFIED	Cog (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	N	Last Exam(Y/N):	N

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 Boration Flow Paths–Operating

TR 3.1.1 Two boron injection flow paths shall be OPERABLE.

---------------------------- NOTES ----------------------------

1. For Unit 1, one flow path shall be from the boric acid storage tanks (BASTs) via an OPERABLE boric acid transfer pump and an OPERABLE charging pump to the Reactor Coolant System (RCS), and one flow path shall be from the refueling water storage tank (RWST) to the RCS via a second OPERABLE charging pump.

2. For Unit 2, the flow paths shall be any two of the following: from the BASTs via an OPERABLE boric acid transfer pump and an OPERABLE charging pump to the RCS, or either of two flow paths Prom the RWST to the RCS via OPERABLE charging pumps.

APPLICABILITY: MODES 1, 2, 3 and 4.

---------------------------- NOTES ----------------------------

1. In MODE 4, only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is ≤ 235°F (Unit 1), ≤ 270°F (Unit 2).

2. Only one charging pump is required to be OPERABLE for one hour following heatup > 235°F (Unit 1), > 270°F (Unit 2) and for one hour prior to cooldown ≤ 235°F (Unit 1), ≤ 270°F (Unit 2).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- Only applicable to Unit 1. ---------------------------- Flow path from the RWST inoperable.	A.1 Restore inoperable flow path to OPERABLE status.	1 hour

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.l	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours
C. One required flow path inoperable for reasons other than Condition A.	C.l	Restore inoperable flow path to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition C not met.	B.l	Be in MODE 3.	6 hours
	AND		
	D.2	Borate to a SHUTDOWN MARGIN within limits provided in the COLR.	6 hours
	AND		
	D.3	Restore inoperable flow path to OPERABLE status.	168 hours
E. Required Action and associated Completion Time of Condition D not met.	E.1	Be in MODE 5.	30 hours

TRM SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.1.1.1	Verify required BAST solution temperature $\geq$ 115°F.	7 days
TSR 3.1.1.2	Verify required BAST boron concentration is $\geq$ 12,950 ppm and $\leq$ 15,750 ppm.	7 days

TRM SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.1.1.3	Verify required BAST borated water volume is ≥ 6000 and ≤ 16,280 gallons.	7 days
PSR 3.1.1.4	Verify temperature of the heat traced portion of the flow path from required boric acid tanks is ≥ 115°F.	7 days
TSR 3.1.1.5	Verify that on recirculation flow, the required charging pumps develop a discharge pressure of ≥ 2410 psig.	In accordance with the Inservice Testing Program
TSR 3.1.1.6	--------------------NOTE-------------------- Only applicable to Unit 1. -- Verify that on recirculation flow, the required boric acid transfer pump develops a discharge pressure of ≥ 109 psig.	In accordance with the Inservice Testing Program
TSR 3.1.1.7	Verify each manual, power operated, and automatic valve in the flow path, that is not locked. sealed or otherwise secured in position, is in the correct position.	31 days

BASES

APPLICABLE SAFETY ANALYSES (continued)

The contents of the BIT are not credited for core cooling or immediate boration in the LOCA analysis, but are for post LOCA recovery. The BIT maximum boron concentration of 15,750 ppm is used to determine the minimum time for hot leg recirculation switchover. The minimum boron concentration of 12,950 ppm is used to determine the minimum mixed mean sump boron concentration for post LOCA shutdown requirements.

For the MSLB, the BIT is the primary mechanism for injecting boron into the core to counteract the positive increases in reactivity caused by an RCS cooldown. The MSLB core response analysis conservatively assumes a 0 ppm minimum boron concentration of the BIT, which also affects the departure from nucleate boiling design analysis. The MSLB containment response analysis conservatively assumes a 2000 ppm minimum boron concentration of the BIT. Reference to the LOCA and MSLB analyses is used to assess changes to the BIT to evaluate their effect on the acceptance limits contained in these analyses.

The minimum temperature limit of 115°F for the BIT ensures that the solution does not reach the boric acid precipitation point. The temperature of the solution is monitored and alarmed on the main control board.

The BIT boron concentration limits are established to ensure that the core remains subcritical during post LOCA recovery. The BIT will counteract any positive increases in reactivity caused by an RCS cooldown.

The BIT water volume of 900 gallons is used to ensure that the appropriate quantity of highly borated water with sufficient negative reactivity is injected into the RCS to shut down the core following an MSLB, to determine the hot leg recirculation switchover time, and to safeguard against boron precipitation.

The BIT satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO establishes the minimum requirements for contained volume, boron concentration, and temperature of the BIT inventory. This ensures that an adequate supply of borated water is available in the event of a LOCA or MSLB to maintain the reactor subcritical following these accidents.

(continued)

GENERAL SYSTEM OPERATION

Objectives

Upon completion of this section, you will be able to:

A. State the design basis of the Heat Tracing System.

B. Discuss the theory of operation of the Heat Tracing System.

C. **DESCRIBE THE THEORY AND OPERATION OF THE HEAT TRACING SYSTEM.**

This section presents a discussion of the theory behind the operation and design of the Heat Tracing System.

Design Basis

Heat tracing is installed on piping, valves, and instrumentation lines carrying boric acid solution, and on certain systems which can experience freezing. The design of the Heat Tracing System for those systems that contain 9 percent by weight (15,750 ppm) boric acid is based on the following criteria:

1. Two separate heat tracing systems are provided (normal and redundant).
2. Each heat tracing system is designed to maintain the fluid temperature in piping above 115°F (the minimum temperature to ensure solubility of 15,750 ppm boron is 111°F).
3. Each of the redundant heat tracing systems is capable of being electrically powered from the emergency diesel generator.
4. Only one of the two redundant branch heat tracing circuits is energized at a time.

Theory of Operation

The solubility of boron in water is a function of the temperature of the water. As the temperature of water decreases, the solubility of baron decreases. For any given temperature. water is only capable of having a certain amount of boron dissolved in it. If a sample of water at a given temperature (111°F, for example) already has the maximum amount of borii acid dissolved in it, then reducing the temperature will result in some of the boric acid coming out of solution (precipitating). Conversely, increasing the temperature will result in the water being capable of having more boric acid dissolved in it. Figure 22-8-NA shows a graph of boron solubility as a function of temperature. Heat tracing is installed on specified pipes and components throughout the plant to prevent the solidification of boron at temperatures below 115°F, and to prevent the freezing of water due to excessively low ambient temperatures.

Heat Exchangers. When checking the valve lineup the operator notices that the light bulbs are burned out for 1-SW-MOV-104A, SW Return from "A" RSHX Isolation Valve.

Which ONE of the following indications can the operator use to verify that this valve is open?

A. Indication for " A RSHX heat exchanger SW flow on the "H" Safeguards Panel.
B. Annunciator J-B3, SERV WTR RETURN HDR LO FLOW, **NOT** lit.

Ability to monitor automatic operations of the Containment spray system including: Verification that cooling water is supplied to the containment spray heat exchanger

(CFR: **41.7 / 45.5)**

New question

References:
FM 78B sheet 3
Loop book SW-003
Annunciator response for J-B3, J-E6

Level (RO/SRO):	RO	Tier:	2
Group:	1	Importance Rating:	3.9/4.2
Type (Bank/Mod/New):	NEW	Cog (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	N	Last Exam(Y/N):	N

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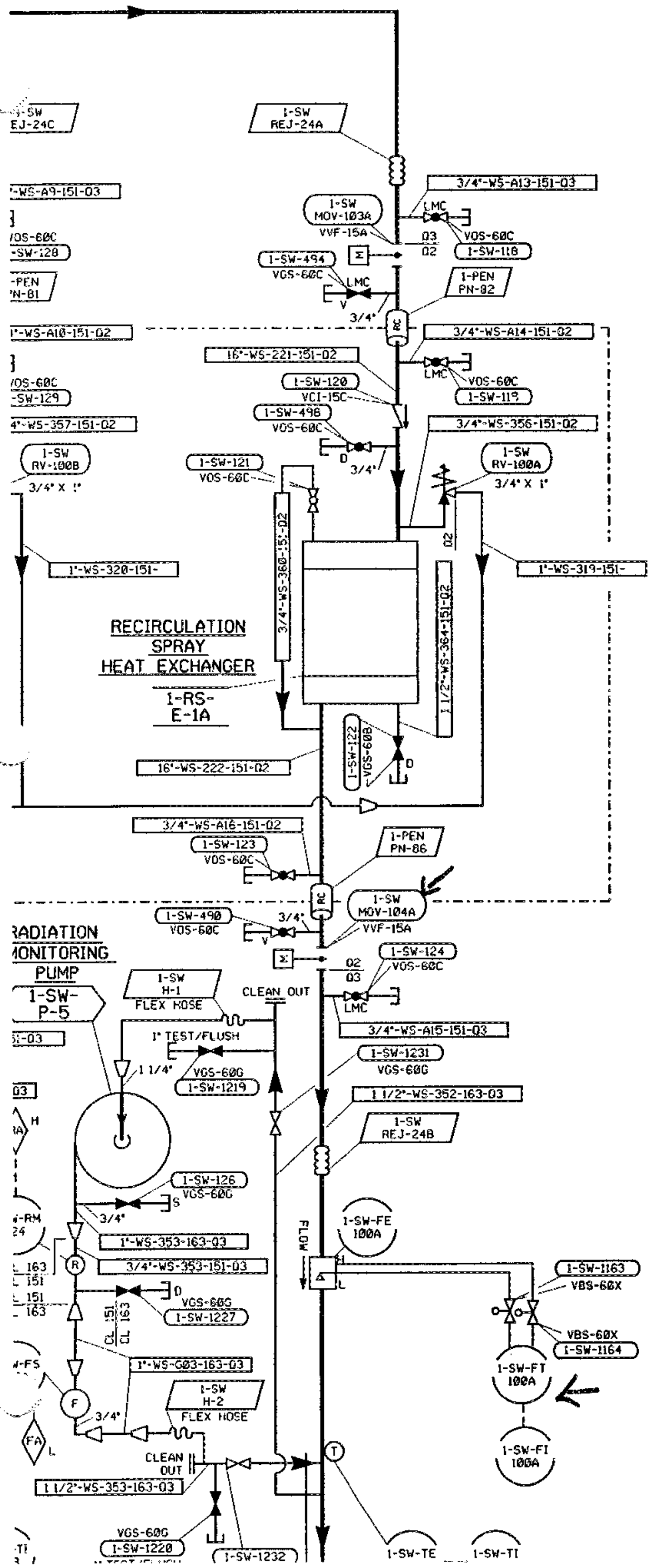
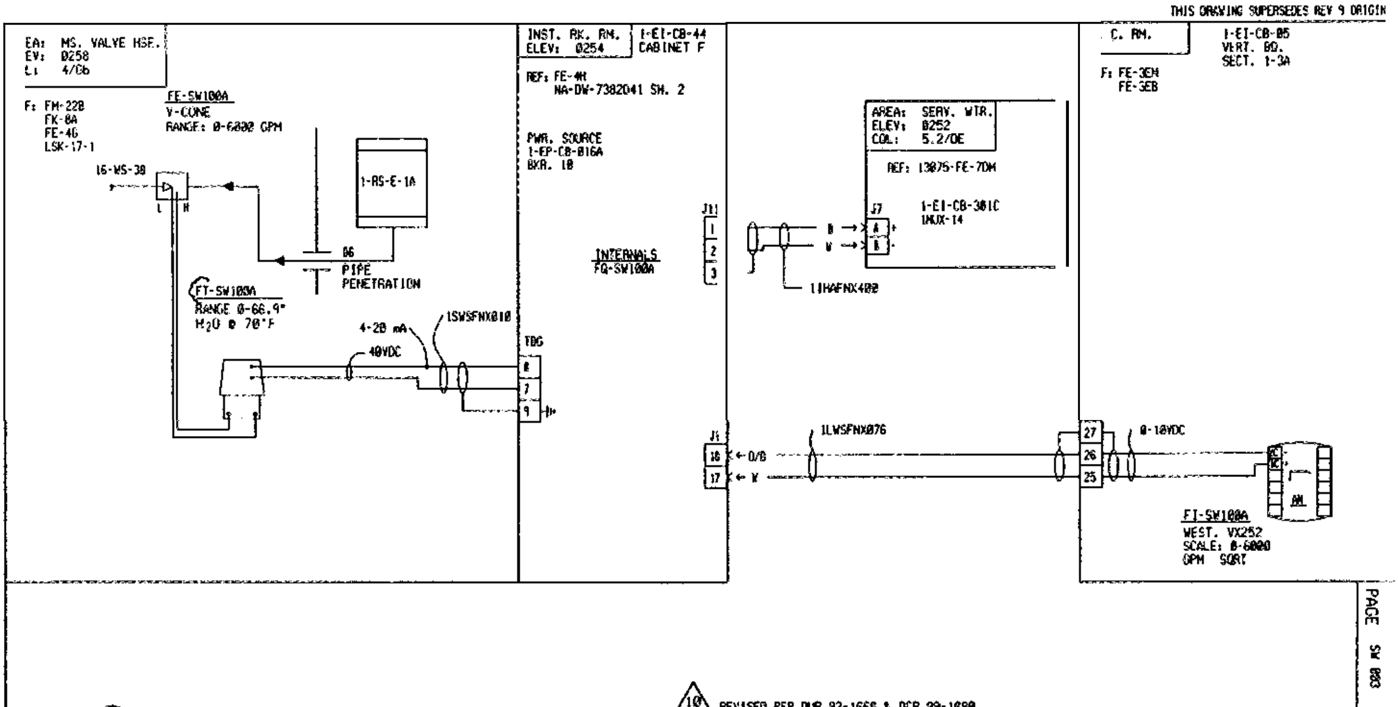
**** WARNING: THIS IS ONLY A PARTIAL SECTION OF ENTIRE DRAWING. ****



FM 78B SI



LOOP ACTION: FT-SW100A SENSES FLOW THROUGH RECIRCULATION SPRAY HEAT EXCHANGER (1-RS-E-1A) AND PROVIDES INDICATION IN MAIN CONTROL ROOM.

10 REVISED PER DUR 92-1660 & DCR 99-1090

REV 10	BY	DATE
DESIGNED	ELC	10-27-99
CHECKED		
ENGINEERED		
APPROVED		
REVISIONS BY		
DATE		

VIRGINIA POWER

SERVICE WATER SYSTEM
RECIRCULATING SPRAY HEAT EXCHANGER
OUTFLOW

PROJECT: NAPS NO. 1

JOB NO: 11715

NUMBER F-SW100A

SERV WTR
RETURN HDR
LO FLOW

e 6000 gpm

NOTE: In extreme cold temperature, this alarm may be a normal condition.

1.0 Probable Cause

1.1 Removal from service of major service water heat exchanger or header
1.2 Reduction of total system flow due to cold Service Water/Component Cooling water temperatures
1.3 Opening of 2-SW-MOV-220A and 220B or 1-SW-MOV-120A and 120B allowing water to return to discharge tunnel
1.4 Broken service water line

2.0 Operator Action

2.1 Check Return Header flow indication as follows:

* "A" Header: 1-SW-FI-103A and 1-SW-FI-110A

* "B" Header: 1-SW-FI-104B and 1-SW-FI-111B

2.2 IF Service Water is operating Reservoir-To-Reservoir, THEN ensure 1-SW-MOV-120A, 1-SW-MOV-120B, 2-SW-MOV-220A, and 2-SW-MOV-220B are closed to stop flow to discharge tunnel.

2.3 Check operating Service Water Pump(s) discharge pressure.

2.4 Check Service Water System for severe leak or rupture.

2.5 IF severe leak or rupture OR loss of Service Water, THEN GO TO 0-AP-12, Loss of Service Water.

3.0 References

3.1 11715-LSK-17-1.1
3.2 11715-FM-78A, Service Mater
3.3 11715-ESK-10BAF
3.4 Instrument Loops 11715-SW-010, -011
3.5 CTS Item 02-94-2229-060, Service Water System Operational Performance Assessment Item 139.0

4.0 Actuations

4.1 1-SW-FT-103 and 104 service water return header flow transmitters

UNIT 1 SW
MODE VALVES
CHANGE POS

NOTE: IF a high volume blowdown of the Service Water Reservoir is in progress, THEN the position input from 1-SW-MOV-120B may be jumpered out by 0-OP-49.7.

1.0 Probable Cause

1.1 Operator changing position of 1-SW-MOV-120A or 120B (Service Water to Discharge Tunnel)

2.0 Operator Action

2.1 Verify 1-SW-MOV-120A and 1-SW-MOV-120B in proper position.

3.0 References

3.1 11715-LSK-17-2D
3.2 11715-ESK-10BAG
3.3 0-OP-49.7, High Volume Blowdown of the Service Water Reservoir

, 0 Actuation

4.1 LS-11 valve position limit switch

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A large break LOCA has occurred on Unit 2. The safety injection system has swapped to cold-leg recirculation mode. The BOP is verifying that 2-SI-MOV-2885A, B,C, and D (Low-head SI Pump Recirc Valves) are all closed.

These valves must be closed in order to ___________

A✓ prevent a radioactivity *release to* atmosphere
B. ensure adequate low-head flow is available to the charging pump suctions
C. ensure adequate low-head flow is available to the cold legs
D. prevent low-head SI pump runout

A. Correct. In recirculation mode the low-heads are taking a suction from the water in the containment sump which will be radioactive. The low head recircs must be closed to keep from putting this sump water into the **RWST** which is vented to atmosphere.

B. Incorrect. Low head flow (design flow is 3000 gpm) is adequate to supply the charging pump suctions even with the recircs open. A candidiate could choose this answer based on knowing the importance of maintaining low-head flow to the charging pumps.

C. Incorrect. Low head flow is adequate (design flow is 3000 gpm) to supply water to the cold legs and the charging pump suctions even with the recircs open.

D. Incorrect. The maximum flow rate for the low heads is 4500 gpm. The low heads can supply the charging pumps, the cold legs, and the recircs without running out the pumps. The recircs are provided to allow a *flow* path **(keep** from dead heading) when the low heads are running with RCS pressure above the discharge pressure of the pumps.

Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: Prevention of path for escape of radioactivity from containment to the outside (interlock on **RWST** isolation after swapover)

New question

References:
NCRODP module *52* (Safety Injection) page 16

Level (RO/SRO):	RO	Tier:	*2*
Group:	1	Importance Rating:	3.714.1
Type (Bank/Mod/New):	**NEW**	Cog (Knowledge/Comp):	COMPREHENSION
Reference (Y/N):	N	Last Exam(Y/N):	N

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from each pump suction area (see Figure 52-4-NA). The air ejectors use the pump discharge as the high pressure source of water to create a suction on the pump suction space. This not only fills the pump suction bell with water, but also increases the flow of water from the sump to the pump suction pit.

A minimum flow bypass line is provided for each pump to recirculate fluid to the RWST to prevent overheating of the pump while operating at shutoff head and for test purposes. Twc motor-operated, isolation valves MOV-1885A & C and MOV-1885B *8* D are piped in series on the recirculation line for each pump. The recirculation line is automatically isolated when the foliowing conditions are satisfied:

1. SI recirculation mode signal is present (from SI, lock-in relay),
2. RWST level is below 19.4 percent, and
3. Either MOV-1863A or B respectively has opened.

During the recirculation mode, the LHSI pumps take a suction on the containment sump. If the recirculation line isolation valves did not shut. radioactive gases from the sump water would be released to the atmosphere through the RWST vent. The valves do not shut until minimal cooling flow is ensured by MOV-1863A or B opening.

The LHSI pumps take a suction on either the RWST or on the containment sump. During normal operations and the injection mode, the LHSI pumps are lined up to receive water from the RWST through motor-operated, isolation valves MOV-1862A and B. During the recirculation mode, these isofation valves are shut and the motor-operated, isolation valves MOV-1860A and B from the containment sump are opened. On receipt of a low-low RWST level. MOV-1860A and 5 will open automatically if an SI recirculation mode signal is present and the respective LHSI pump recirculation valves have shut.

The LHSI pump discharge can be directed to the RCS cold legs. the HHSI pump suction, or the RCS hot legs. During normal plant operations and the injection mode, the discharge of the pumps is lined up to the RCS cold legs through normally open, pump discharge valves MOV-1864A and B and a pair of normally open, isolation valves MOV-1890C and D that are piped in parallel. On initiation of the recirculation mode, the discharge of the LHSI pumps continues to the RCS cold loops with some portion being directed to the suction of the HHSI pumps through normally shut, isolation valves MOV-1863A and B. This lineup ensures net positive suction head to the HHSI pumps, since water is no longer being provided to the HHSI pumps from the RWST. During the recirculation mode, the discharge of the LHSI pumps is periodically lined up to the RCS hot legs through normally shut, isolation valves MOV-1890A and B. On Unit I, the outside recirculation pumps RS-P-2A and B can discharge to the LHSI pump discharge headers in the event of failure of one or both of the LHSI pumps. Each outside recirculation pump is normally isolated from the corresponding LHSI pump by a pair of series manual isolation valves. They are operated from outside the safeguards building with a T-handle wrench inserted into the associated

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Unit 1 is operating at 10% power with all systems in automatic when an outsurge from the pressurizer causes level to drop to 12%. The OATC notices that the pressurizer spray valves have gone closed.

The spray valves closed ___________

A.✓ due to the level decrease causing a corresponding pressure decrease
B. because the pressurizer heaters turned off on low level, which sent a signal to shut the spray valves
C. directly from a pressurizer low-low level signal
D. to prevent them from spraying cold water on the uncovered pressurizer heaters which could burn them out

A. Correct. The pressure decrease caused by a combination of the outsurge and the loss of the pressurizer heaters on low level, would cause the output of the master pressurizer controller to decrease until the spray valves were shut.

B. Incorrect. There are no signals sent from the pressurizer heaters which close the spray valves. This answer could be picked by a candidate that does not understand that such a signal is not needed.

C. Incorrect. There is no signal from low or low-low Pressurizer level which close the spray valves. This answer could be picked by a candidate that does not understand that such a signal is not needed.

D. Incorrect. The heaters are turned off on level <15% to keep them from burning up. A candidate could pick this answer based on this reasoning.

Knowledge of the reasons for the following responses as they apply to Pressurizer Pressure Control system malfunction: Isolation of PZR spray following loss of PZR heaters

(CFR: 41.5 / 41.10 / 45.6 / 45.13)

New question.

References:
NCRODP module 74 - Pressure Control and Protection (page 4, 28)

Level (RO/SRO):	RO	Tier:	1
Group:	1	Importance Rating:	3.5/3.8
Type (Bank/Mod/New):	NEW	Cog (Knowledge/Comp):	COMPREHENSION
Reference (Y/N):	N	Last Exam(Y/N):	N

GENERAL DESCRIPTION

Objectives

Upon completion of this section, you will be able to:

A. Describe, in general terms, the normal operation of the pressurizer and related subsystems.

B. Describe, in general terms, the major components associated with the pressurizer.

C. Describe, in general terms, the major components associated with the pressurizer relief tank.

D. Describe, in general terms, the operation of the pressurizer pressure control and protection subsystem, and the output signals provided by the subsystem.

E. Describe. in general terms, the operation of the pressurizer level control and protection subsystem, and the output signals provided by the subsystem.

F. State the systems that interface with the Pressure Control and Protection System and describe how and why they do so.

G. DESCRIBE THE MAJOR COMPONENTS AND OPERATION OF THE PRESSURE CONTROL AND PROTECTION SYSTEM.

This section presents a general description of the Pressure Control and Protection System operation. major components, and system interrelationships. The purpose of this section is to provide the reader with a general understanding of the system operation.

Normal Operation

The pressurizer is a vertical cylindrical vessel that is equipped with electrical heaters. The pressurizer is slightly over half-full with liquid during normal power operation, with the remainder of the volume filled with steam (steam bubbie). The temperature of the liquid in the pressurizer is greater than that in the reactor coolant loops/vessel. The pressurizer liquid is at saturated condition, with the temperature maintained by the electric heaters. The steam bubble above it is also at saturated condition with a temperature corresponding to the desired pressure for the RC System. The pressure In the pressurizer is transmitted to the remainder of the RC System via the surge line which connects the bottom of the pressurizer with the RC System piping near the outlet of the reactor vessel.

Pressurizer

The pressurizer malntains system pressure during steady-state operation as well as transient conditiins. Load changes cause a change in the temperature of the water in the reactor coolant loops. If the coolant temperature decreases, Ute volume of the codant decreases. causing an outsurge of water from the pressurizer. As the outsurge continues the steam bubble expands causing a drop in RC System pressure, allowing some of the saturated liquid to flash to steam. This does not maintain pressure at the original pressure, but it does mitigate the magnitude of the pressure drop. The

The operator can secure the control group heaters by placing the handswitch in the STOP position, causing the feeder breaker to open The manual opening of the breaker is indicated by a green flag and a green light. The feeder breaker is automatically tripped open by a current overload device or by the level control subsystem if pressurizer level falls to 15 percent. if the iow level heater cutout occurs (LC-1459C or LC-146OC) the PRESSURIZER LOW LEVEL HEATERS OFF - LETDOWN ISOLATION alarm (window 1B-G7) annunciates at the MCB. The low pressurizer level heater cutout ensures that the heaters do not operate when the liquid level in the pressurizer begins to uncover the heaters and expose them to the steam environment. This could lead to heater burnout.

If the overcurrent trip occurs the PRESSURIZER CONTROL GROUP HEATERS OVERLOAD TRIP alarm (window 1B-H7) annunciates at the MCB. The overcurrent trip must be manually reset at the feeder breaker; the low level cutout automatically resets when level rises above 15 percent. The automatic breaker trip is indicated by a green flag and an amber light. The breaker can also be manually tripped at the feeder breaker using the trip pushbutton. In the PULL-TO-LOCK position, the breaker trips if not already open, and all indicating lights are deenergized.

Backup heater groups No. 2 and No. 5 are identical in operation, differing only in power supply (see Table 74-6). These heaters are supplied from non-emergency buses, the significance of which is described later in this section. The heater groups each have a four position (PULL-TO-LOCK, STOP, AUTO, START) handswitch with three indicating lights (green, amber, red) above the handswitch. Using group No. 2 as an example, the heaters receive 480V ac power through a feeder breaker located in MCC-1B1, a set of contactors. and a heater panel. The feeder breaker and the contactors (contained in and operated by a controller located on the south wall of the Rod Drive Room) are both controlled by the same handswitch. With the handswitch in the AUTO position and the breaker dosed, as indicated by the red light and the red flag in the window above the handswitch, power is delivered to the breaker side of the contactors. The contactors are operated by a controller which uses 120V ac power which has been transformed from 480V ac downstream of the feeder breaker.

The controller operates the contactors based on control signal from the pressure and level control subsystems. If pressure falls from the nominal 2235 psig to 2210 psig a pressure comparator in the pressure control subsystem (PC-1444F) energizes the heater group. A white light on the vertical board indicates when the group is energized. The heaters remain energized until pressure rises to 2215 psig or feeder breaker opens. When the backup heaters energize, the PRESSURIZER LOW PRESSURE - BACKUP HEATERS ON alarm (window 1B-H6) annunciates at the MCB. The heaters are also energized if the pressurizer level rises to 5 percent above the programmed level. The level control subsystem (LC-1459D) energizes the heaters in anticipation of a drop in pressure due to the excess insurge of relatively cold water from the RC System. If this feature energizes the

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The following conditions exist on unit 1:

- The unit is in mode 6
- Containment purge is in service
- Core alterations are in progress
- A hi-hi radiation signal is received on 1-RMS-RM-162, Manipulator Crane Radiation Monitor
- The refueling SRO reports that a bag of radioactive waste was carried by the manipulator crane and has now cleared the area
- Radiation reading has returned to normal on 1-RMS-RM-162.

Based on the above, containment purge ____________

A.✓ has isolated and must be manually restored after hi-hi radiation signai is reset
B. has isolated and will automatically restore after hi-hi radiation signal is reset
C. has not isolated since it does not get a signal from 1-RMS-RM-162
D. was isolated, but automatically restored when the radiation reading returned to a normal value

A. Correct. A hi-hi radiation signal on 1-RMS-RM-162 will isolate purge and exhaust and it must be manually restored once the hi-hi signal is reset.

B. Incorrect. The system must be manually restored. A candidate could choose this answer based on the information that the area radiation has returned to normal and knowledge that if both containments are purging the containment that was not the cause of the isolation signal has purge automatically restored.

C. Incorrect. The system does get a signal from 1-RMS-RM-162 when this rad monitor is in service. The containment purge system also gets a signal from 159/160. A candidate could choose this answer if he/she does not remember that the system also gets a signal from 162.

D. Incorrect. The system does not automatically restore, and the hi-hi signal must be reset, as it will not clear when radiation readings return to normal. A candidate could choose this answer based on the information that the area radiation has returned to normal.

Ability to (a) predict the impacts of the following on the Containment purge and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Maintenance or other activity taking place inside containment

(CFR: 41.5 / 43.5 145.3 145.13)

New question

References:
Annunciator response for K-D3, RAD MONITOR SYSTEM FAILURE TEST
NCRODP module 47 Primary ventilation (page 28)
NCRODP Module 46, Radiation Monitoring (page 28)
Note in 1-OP-21.2, "Containment Purge."

Level (RO/SRO):	RO	Tier:	2
Group:	2	Importance Rating:	2.9/3.6
Type (Bank/Mod/New):	NEW	Cog (Knowledge/Comp):	COMPREHENSION
Reference (Y/N):	N	Last Exam(Y/N):	N

RAD MONITOR
SYSTEM
FAILURE
TEST

1.0 Probable Cause

1.1 Blown fuse on UNIT 1 or Common radiation monitor
1.2 Operate switch out of "OPERATE" position
1.3 Loss of power to radiation monitor
1.4 Hi-Hi Alarm cleared but not reset with subsequent Hi Alarm on same radiation monitor

2.0 Operator Action

2.1 Determine which radiation monitor is causing alarm.

2.2 Determine if testing is in progress.

2.3 Verify Rad Monitor in service in accordance with 1-OP-62.1, Operation of Area Radiation Monitors.

2.4 For any failure, intiate 1-AP-5, Unit 1 Radiation Monitor System, or 0-AP-5.1, Common Unit Radiation Monitoring System.

2.5 Refer to Tech Spec 3.3.3.1 (IT$ 3.4.15, TRM 3.3.7) and VPAP-2103, Offsite Dose Calculation Manual.

2.6 Reset Hi-Hi Alarms that have cleared.

3.0 References

3.1 11715-ESK-10-AAW and 11X
3.2 Stone and Webster vendor drawing 11715/12050-1.21-67
3.3 Westinghouse technical manual, Radiation Monitoring, PO NA-200/NA-1200
3.4 Tech Spec 3.3.3.1 (ITS 3.4.15, TRM 3.3.7) (Radiation Monitoring)
3.5 VPAP-2103, Offsite Dose Calculation Manual

4.0 Actuation

4.1 K contacts, in UNIT 1 and Common radiation monitors

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valve, which permits reduced purge flow if required. Motor operated valve MOV-HV-100C is located in the purge exhaust line inside Containment. When iow purge flow rates are required, MOV-HV-101 is opened and MOV-HV-100D is shut. All containment purge system containment isolation MOVs for a unit automatically shut when a Hi Hi radiation condition exists in the Containment Structure for that unit. The valves trip shut on a signal from radiation monitor RM-159, -160, or -162 (refer to module NCRODP-46-NA. Radiation Monitoring System).

These motor-operated valves are driven by 1/7-HP motors. MQV-HV-100A and -100C are powered from MCC 1A1-1. MOV-HV-100B and -100D are powered from MCC 1B1-2.

Containment Pressure Equalizing Valve. Before the purge equipment can be placed in service, containment pressure must be equalized with atmospheric pressure. An 18-inch pressure equalizing valve (MOV-HV-102) provides that capability and is located between the purge supply containment isolation valve and the containment wall. This MOV automatically shuts when a Hi Hi radiation condition in Containment is detected by radiation monitor RM-159, -160, or -162. This motor-operated valve is driven by a 1/7-HP motor and is powered from MCC 1C1-1. A temporary spoolpiece connection is made at MOV-HV-102/202 to allow portable air compressors to pressurize the Containment during Type-A leakage testing of the Containment Structure.

Containment Purge Exhaust Iodine Filtration Equipment Bypass Dampers. Four dampers are used to control the direction of ventilation exhaust air. Exhaust air is normaily discharged directly to the environment from the Containment via two air-operated dampers (AOD-HV-104-1 and 104-2) in line with the exhaust fan suction. If the air becomes contaminated, the exhaust air is redirected through the Auxiiiary Building iodine filter banks by way of iodine filter supply and return dampers AOD-HV-104-4 and 104-3. These dampers can be controlied from either unit's Vent Panel with a handswitch.

Containment Elevator Exhaust Fan. Fan HV-F-53 exhausts air from the elevator machinery room at a rate of 2,800 cfm. The fan is powered from MCC 1C1-1 and starts the elevator fan when the space temperature exceeds 110°F (as sensed by local temperature switch TS-HV-161).

Appendix R Ventilation Subsystem

The Appendix R Ventilation Subsystem is designed to ventilate critical areas in the Auxiliary Building and Fuel Building if the normal ventilation equipment which ventilates these areas is disabled by fire. The system is normally shutdown, and operated only during emergencies addressed by FCA procedures.

The critical areas ventiiated are:

1. Auxiliary Building Component Cooling Area
2. Auxiliary Building Charging Pump Cubicles
3. Fuel Building Auxiliary Monitoring Panel Area

Personnel Hatch area monitor RMS-RM-161 (-261) is located outside the Unit 1 (Unit 2) Containment near the hatch. The detector output is transmitted to the RMS instrument rack 1-EI-CB-49A (-49 in the MCR. There, the radiation intensity is displayed on a log ratemeter RI-RMS-161 (wide range of 10^{-4} to 10^{1} Rlhr and narrow range 10^{-4} to 10^{-1} Whr) and is simultaneously recorded on a multipoint strip chart recorder RR-100 (-200). High radiation intensity is indicated by both audio and visual alarms. In addition, the output is transmitted to a local log indicator RMS-RI-161A (range 10^{-4} to 10^{1} Rlhr) for audio and visual alarms.

The containment area monitors RMS-RM-162 through -164 (listed on Table 46-1) are similar in operation. Only RMS-RM-162 is described. RMS-RM-162 detects gamma radiation in the manipulator crane area of Containment by means of a fixed-position G-M tube. The detector output is transmitted to the RMS instrument rack 1-EI-CB-49A in the MCR. There, the radiation intensity is displayed on a log ratemeter RMS-RI-162 (wide range of 10^{-4} to 10' R/hr and narrow range of 10^{-4} to 10^{-1} R/hr) and simultaneously recorded on a multipoint strip chart recorder RR-100. High radiation intensity is indicated by both audio and visual alarms.

Hi-Hi alarm on RMS-RM-162 will stop the containment purge supply & exhaust fans (if running) and shut all containment purge supply & exhaust MOVs on the effected unit. In addition, the output is transmitted to a local indicator RMS-RI-162A (10^{-4} to 10^{1} Rlhr) for audio and visual alarms. Area monitor controls at the channel drawer consist of the foilowing:

1. POWER ON iight, indicates instrument power applied to this channel;
2. operation selector switch (RESET, CHECK SOURCE, OPERATE, LEVEL CAL, PULSE CAL);
3. range selector switch (WIDE, NARROW, HV (high voltage);
4. CHANNEL TEST light. which is illuminated when the operator selector switch is out of OPERATE;
5. HIGH ALARM light, which indicates high radioactivity;
6. LOW ALARM light. which indicates when RMS input signal has faiied or RMS has maifunctioned; and
7. fuse holder, which indicates when line fuse is blown.

Note that removal of manipulator crane (RMS-RM-162) fuses would cause containment purge supply and exhaust fans to trip and the associated unit's purge valves to shut.

Gaseous and Particulate **Monitors** (see Table 46-6). The meter in the face of the drawer indicates counts per minute (10^{1} to 10^{4} or 10^{1} to 10^{6}) or high voltage applied to the detector (0 to 2.5 kV dc). depending on range selector

NOTE: The following parameters initiate automatic functions which affect the purge and exhaust system.

- High radiation levels in containment (I-RM-RMS-159, **160** or **162)** will close the containment purge system butterfly valves.
- Air temperature leaving the steam heating coils at < 35°F will trip the containment purge supply fans.
- Purge Supply Fans 1-HV-F-4A AND **4B will** trip IF 1-MOV-HV-100A OR 1-MOV-HV-100B are NOT FULL OPEN.
- Purge Exhaust Fans 1-HV-F-5A AND 5B will **trip** IF 1-MOV-HV-100C OR 1-MOV-HV-100D are NOT FULL OPEN AND 1-MOV-HV-101 IS FULL CLOSED.

NOTE: Steam Heating is required if Containment temperature is less than 65°F.

5.1.19 **IF** directed by Health Physics, **THEN** remove containment purge flow from the iodine filter as per 0-OP-21.5, Operation of Auxiliary Building Iodine Filters.

Completed by: ______________________________ Date: __________

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Unit I is in mode **3** when the OATC notes that **N-31** and **N-32** are not within the specified tolerance of one-half decade. I&C determined that the high voltage setting for **N-31** is set too high.

Due to this mis-calibration, **N-31** is reading __________ than it should due to __________

A. higher; the gas amplification effect
B. lower; over-compensation
C. higher; under-compensation
D. lower; the gas amplification effect

A. Correct. If the high voltage setting is set too high the gas amplification effect will cause the detector counts to be high.

B. Incorrect. Both parts of this answer are incorrect. The counts will be higher and there is no compensation voltage for the source ranges.

C. Incorrect. The first part of the answer is correct, however there is no compensating voltage for the source ranges.

D. Incorrect. The second part of the answer is correct, however this will cause the detector counts to be high.

Ability to determine and interpret the following **as** they apply to **Loss** of Source Range: Effect of improper HV setting

(CFR: 41.10 / 43.5 / 45.13)

New question

References:
Objective 7771 from self-study guide on Ex-Core Nuclear Instrumentation

Level (ROISKO):	KO	Tier:	1
Group:	a	Importance Rating:	2.5/2.9
Type (Bank/Mod/New):	NEW	Cog (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	N	Last Exam(Y/N):	N

Topic 2.2.2: Source Range Detector 7771

2.2.2a. Objective

Explain the following concepts concerning the source-range detector.

- Type of detector
- Portion of the reactor core monitored by the detector
- How the detector operates to produce an electrical output from neutron and gamma Interactions
- Setpoint and coincidence of the SOURCE RANGE LOSS OF DETECTOR VOLTAGE alarm
- Purpose of the "crowbar" circuit

2.2.2b. Content

1. Each channel of source range instrumentation uses a BF_3 proportional counter.
   1.1. Each source range detector shares a common housing assembly with an intermediate range detector.
   1.2. The source range detector is located in the bottom of the assembly, at an elevation of approximately one-quarter *of* the core height.
   1.3. Each source range detector tube consists of an aluminum shell (cathode) filled with **boron-**trifluoride gas and an inner tungsten electrode (anode) insulated from the cathode.

2. Due to its location, the source range detector monitors the bottom of the core.

3. The detector operates to produce an electrical output from neutron and gamma interactions.
   3.1. The neutron-sensitive material utilized by the proportional counter is boron (B), enriched with B-10.
   3.2. The high microscopic absorption cross-section for thermal neutrons of this particular isotope increases the probability of neutron interaction.
   3.3. When a B-10 atom absorbs a thermal neutron, the following reaction occurs:

3.3.1.810 + Neutron → B-11 → Li-7 + He-4 + electron + 2.8 Mev

3.4. The 2.8 MeV represents the kinetic energy of the products of this reaction.

3.5. The high voltage potential applied between the anode(+) and cathode(-) operates the detector in the proportional region of the gas ionization curve.

3.6. Positive helium (He) and lithium (Li) ions are attracted to the negative cathode, while the free electrons move toward the positive anode.

3.7. As these charged particles move through the filler gas, they give up their energy to the filler gas molecules.

3.8. The gas molecules ionize, producing more ion pairs.

3.9. The collection of ions on the electrodes creates a negative pulse which is sent from the detector to the **pulse** preamplifier through the triaxial cable.

3.10. In the proportional region of operation, the detector voltage is high enough to accelerate the ion pairs to a sufficient energy that they, in turn, cause subsequent ionizations.

3.11. This effect is known as gas amplification.

3.12. The proportional counters also detect pulses from gamma ray interactions occurring in the detector.

3.13. Gamma rays interact by either Compton scattering, pair production, or photoelectric effect.

3.14. Although gamma pulses are approximately 10 times smaller in size than neutron pulses, they still must **be** considered in the total current produced by the detector.

3.15. The primary source of gamma rays in the source range is the decay of activated materials and is not indicative ***of*** reactor power.

3.16. The current produced by the gamma pulses ***is*** therefore eliminated electronically in the source range circuitry.

4. The LOSS OF DETECTOR VOLTAGE lamp illuminates when a source range detector's high voltage power supply drops 100 volts below normal.

4.1. **A** bistable in each source range drawer actuates the annunciator and a drawer mounted **loss** of detector voltage lamp.

4.2. The annunciator **is** automatically defeated when power level increases above P-10.

4.3. The **loss** of detector voltage lamp is not interlocked with P-10.

5. Each source range high voltage power supply is provided with a "crow-bar circuit, which protects the channel from excessive voltage.

5.1. When a 20% over-voltage condition occurs, a thyristor gates on the high voltage power supply in order to protect the detector from excessively high voltage.

5.2. Reset by removing and reinstalling the instrument power fuses.

Topic 2.2.3: Order of SR Components 7772

2.2.3a. Objective

List the following components in sequential order as they appear in a source-range nuclear instrument.

- Bistables
- High-voltage power supply
- Level amplifier
- Log pulse integrator
- Meter (local)
- Preamplifier
- Pulse amplifier
- Pulse driver
- Pulse shaper

2.2.3b. Content