PSEG Public Service Electric and Gas Company

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Robert L. Mittl General Manager
Nuclear Assurance and Regulation

October 3, 1984

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, MD 20814

Attention: Mr. Albert Schwencer, Chief
Licensing Branch 2
Division of Licensing

Gentlemen:

HOPE CREEK GENERATING STATION
DOCKET NO. 50-354
DRAFT SAFETY EVALUATION REPORT
OPEN ITEM STATUS

Attachment 1 is a current list which provides a status of the open items identified in Section 1.7 of the Draft Safety Evaluation Report (SER). Items identified as "complete" are those for which PSE&G has provided responses and no confirmation of status has been received from the staff. We will consider these items closed unless notified otherwise. In order to permit timely resolution of items identified as "complete" which may not be resolved to the staff's satisfaction, please provide a specific description of the issue which remains to be resolved.

Attachment 2 is a current list which identifies Draft SER Sections not yet provided.

Enclosed for your review and approval (see Attachment 4) are the resolutions to the Draft SER open items listed in Attachment 3 previously submitted on October 1, 1984.

Pursuant to discussions with the Licensee Qualifications Branch, enclosed for your review (see Attachment 5) is a copy of revised FSAR Section 13.2 concerning training programs previously submitted on October 1 and 2, 1984.

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Also enclosed (see Attachment 6) is one copy of "An overview of PSE&G Technical Qualifications and Management Capability in Support of the Operation of Hope Creek Generating Station" previously transmitted on July 18, 1984, in a letter from E. Liden, PSE&G, to F. Allenspach, NRC.

A signed original of the required affidavit is provided to document the submittal of these items.

Should you have any questions or require any additional information on these items, please contact us.

Very truly yours,

R J Writt

Attachments/Enclosure

C D. H. Wagner
USNRC Licensing Project Manager (w/attach.)

W. H. Bateman
USNRC Senior Resident Inspector (w/attach.)

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
DOCKET NO. 50-354

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

Public Service Electric and Gas Company hereby submits the enclosed responses to DSER open items and revised FSAR Section 13.2 for the Hope Creek Generating Station.

The matters set forth in this submittal are true to the best of my knowledge, information, and belief.

Respectfully submitted,

Public Service Electric and Gas Company

By: Thomas J. Martin
Thomas J. Martin
Vice President -
Engineering and Construction

Sworn to and subscribed before me, a Notary Public of New Jersey, this 3rd day of October 1984.

David K. Burd

DAVID K. BURD
NOTARY PUBLIC OF NEW JERSEY
My Comm. Expires 10-23-85

MP84 154/04 4-vw

DATE: 10/3/84

ATTACHMENT 1

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
1	2.3.1	Design-basis temperatures for safety-related auxiliary systems	Complete	8/15/84
2a	2.3.3	Accuracies of meteorological measurements	Complete	8/15/84 (Rev. 1)
2b	2.3.3	Accuracies of meteorological measurements	Complete	8/15/84 (Rev. 1)
2c	2.3.3	Accuracies of meteorological measurements	Complete	8/15/84 (Rev. 2)
2d	2.3.3	Accuracies of meteorological measurements	Complete	8/15/84 (Rev. 2)
3a	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2)	Complete	8/15/84 (Rev. 2)
3b	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2)	Complete	8/15/84 (Rev. 2)
3c	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2)	NRC Action	
4	2.4.2.2	Ponding levels	Complete	8/03/84
5a	2.4.5	Wave impact and runup on service Water Intake Structure	Complete	9/13/84 (Rev. 3)
5b	2.4.5	Wave impact and runup on service water intake structure	Complete	9/13/84 (Rev. 3)
5c	2.4.5	Wave impact and runup on service water intake structure	Complete	7/27/84
5d	2.4.5	Wave impact and runup on service water intake structure	Complete	9/13/84 (Rev. 3)
6a	2.4.10	Stability of erosion protection structures	Complete	8/20/84
6b	2.4.10	Stability of erosion protection structures	Complete	8/20/84
6c	2.4.10	Stability of erosion protection structures	Complete	8/03/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
7a	2.4.11.2	Thermal aspects of ultimate heat sink	Complete	8/3/84
7b	2.4.11.2	Thermal aspects of ultimate heat sink	Complete	8/3/84
8	2.5.2.2	Choice of maximum earthquake for New England - Piedmont Tectonic Province	Complete	8/15/84
9	2.5.4	Soil damping values	Complete	6/1/84
10	2.5.4	Foundation level response spectra	Complete	6/1/84
11	2.5.4	Soil shear moduli variation	Complete	6/1/84
12	2.5.4	Combination of soil layer properties	Complete	6/1/84
13	2.5.4	Lab test shear moduli values	Complete	6/1/84
14	2.5.4	Liquefaction analysis of river bottom sands	Complete	6/1/84
15	2.5.4	Tabulations of shear moduli	Complete	6/1/84
16	2.5.4	Drying and wetting effect on Vincentown	Complete	6/1/84
17	2.5.4	Power block settlement monitoring	Complete	6/1/84
18	2.5.4	Maximum earth at rest pressure coefficient	Complete	6/1/84
19	2.5.4	Liquefaction analysis for service water piping	Complete	6/1/84
20	2.5.4	Explanation of observed power block settlement	Complete	6/1/84
21	2.5.4	Service water pipe settlement records	Complete	6/1/84
22	2.5.4	Cofferdam stability	Complete	6/1/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
23	2.5.4	Clarification of FSAR Tables 2.5.13 and 2.5.14	Complete	6/1/84
24	2.5.4	Soil depth models for intake structure	Complete	6/1/84
25	2.5.4	Intake structure soil modeling	Complete	8/10/84
26	2.5.4.4	Intake structure sliding stability	Complete	8/20/84
27	2.5.5	Slope stability	Complete	6/1/84
28a	3.4.1	Flood protection	Complete	8/30/84 (Rev. 1)
28b	3.4.1	Flood protection	Complete	8/30/84 (Rev. 1)
28c	3.4.1	Flood protection	Complete	8/30/84 (Rev. 1)
28d	3.4.1	Flood protection	Complete	8/30/84 (Rev. 1)
28e	3.4.1	Flood protection	Complete	8/30/84 (Rev. 1)
28f	3.4.1	Flood protection	Complete	7/27/84
28g	3.4.1	Flood protection	Complete	7/27/84
29	3.5.1.1	Internally generated missiles (outside containment)	Complete	8/3/84 (Rev. 1)
30	3.5.1.2	Internally generated missiles (inside containment)	Closed (5/30/84-Aux.Sys.Mtg.)	6/1/84
31	3.5.1.3	Turbine missiles	Complete	7/18/84
32	3.5.1.4	Missiles generated by natural phenomena	Complete	7/27/84
33	3.5.2	Structures, systems, and components to be protected from externally generated missiles	Complete	7/27/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
34	3.6.2	Unrestrained whipping pipe inside containment	Complete	7/18/84
35	3.6.2	ISI program for pipe welds in break exclusion zone	Complete	6/29/84
36	3.6.2	Postulated pipe ruptures	Complete	6/29/84
37	3.6.2	Feedwater isolation check valve operability	Complete	8/20/84
38	3.6.2	Design of pipe rupture restraints	Complete	8/20/84
39	3.7.2.3	SSI analysis results using finite element method and elastic half-space approach for containment structure	Complete	8/3/84
40	3.7.2.3	SSI analysis results using finite element method and elastic half-space approach for intake structure	Complete	8/3/84
41	3.8.2	Steel containment buckling analysis	Complete	6/1/84
42	3.8.2	Steel containment ultimate capacity analysis	Complete	8/20/84 (Rev. 1)
43	3.8.2	SRV/LOCA pool dynamic loads	Complete	6/1/84
44	3.8.3	ACI 349 deviations for internal structures	Complete	6/1/84
45	3.8.4	ACI 349 deviations for Category I structures	Complete	8/20/84 (Rev. 1)
46	3.8.5	ACI 349 deviations for foundations	Complete	8/20/84 (Rev. 1)
47	3.8.6	Base mat response spectra	Complete	8/10/84 (Rev. 1)
48	3.8.6	Rocking time histories	Complete	8/20/84 (Rev. 1)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
49	3.8.6	Gross concrete section	Complete	8/20/84 (Rev. 1)
50	3.8.6	Vertical floor flexibility response spectra	Complete	8/20/84 (Rev. 1)
51	3.8.6	Comparison of Bechtel independent verification results with the design-basis results	Complete	8/20/84 (Rev. 2)
52	3.8.6	Ductility ratios due to pipe break	Complete	8/3/84
53	3.8.6	Design of seismic Category I tanks	Complete	8/20/84 (Rev. 1)
54	3.8.6	Combination of vertical responses	Complete	8/10/84 (Rev. 1)
55	3.8.6	Torsional stiffness calculation	Complete	6/1/84
56	3.8.6	Drywell stick model development	Complete	8/20/84 (Rev. 1)
57	3.8.6	Rotational time history inputs	Complete	6/1/84
58	3.8.6	"O" reference point for auxiliary building model	Complete	6/1/84
59	3.8.6	Overturning moment of reactor building foundation mat	Complete	8/20/84 (Rev. 1)
60	3.8.6	BSAP element size limitations	Complete	8/20/84 (Rev. 1)
61	3.8.6	Seismic modeling of drywell shield wall	Complete	6/1/84
62	3.8.6	Drywell shield wall boundary conditions	Complete	6/1/84
63	3.8.6	Reactor building dome boundary conditions	Complete	6/1/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
64	3.8.6	SSI analysis 12 Hz cutoff frequency	Complete	8/20/84 (Rev. 1)
65	3.8.6	Intake structure crane heavy load drop	Complete	6/1/84
66	3.8.6	Impedance analysis for the intake structure	Complete	8/10/84 (Rev. 1)
67	3.8.6	Critical loads calculation for reactor building dome	Complete	6/1/84
68	3.8.6	Reactor building foundation mat contact pressures	Complete	6/1/84
69	3.8.6	Factors of safety against sliding and overturning of drywell shield wall	Complete	6/1/84
70	3.8.6	Seismic shear force distribution in cylinder wall	Complete	6/1/84
71	3.8.6	Overturning of cylinder wall	Complete	6/1/84
72	3.8.6	Deep beam design of fuel pool walls	Complete	6/1/84
73	3.8.6	ASHSD dome model load inputs	Complete	6/1/84
74	3.8.6	Tornado depressurization	Complete	6/1/84
75	3.8.6	Auxiliary building abnormal pressure	Complete	6/1/84
76	3.8.6	Tangential shear stresses in drywell shield wall and the cylinder wall	Complete	6/1/84
77	3.8.6	Factor of safety against overturning of intake structure	Complete	8/20/84 (Rev. 1)
78	3.8.6	Dead load calculations	Complete	6/1/84
79	3.8.6	Post-modification seismic loads for the torus	Complete	8/20/84 (Rev. 1)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
80	3.8.6	Torus fluid-structure interactions	Complete	6/1/84
81	3.8.6	Seismic displacement of torus	Complete	8/20/84 (Rev. 1)
82	3.8.6	Review of seismic Category I tank design	Complete	8/20/84 (Rev. 1)
83	3.8.6	Factors of safety for drywell buckling evaluation	Complete	6/1/84
84	3.8.6	Ultimate capacity of containment (materials)	Complete	8/20/84 (Rev. 1)
85	3.8.6	Load combination consistency	Complete	6/1/84
86	3.9.1	Computer code validation	Complete	8/20/84
87	3.9.1	Information on transients	Complete	8/20/84
88	3.9.1	Stress analysis and elastic-plastic analysis	Complete	6/29/84
89	3.9.2.1	Vibration levels for NSSS piping systems	Complete	6/29/84
90	3.9.2.1	Vibration monitoring program during testing	Complete	7/18/84
91	3.9.2.2	Piping supports and anchors	Complete	6/29/84
92	3.9.2.2	Triple flued-head containment penetrations	Complete	6/15/84
93	3.9.3.1	Load combinations and allowable stress limits	Complete	6/29/84
94	3.9.3.2	Design of SRVs and SRV discharge piping	Complete	6/29/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
95	3.9.3.2	Fatigue evaluation on SRV piping and LOCA downcomers	Complete	6/15/84
96	3.9.3.3	IE Information Notice 83-80	Complete	8/20/84 (Rev. 1)
97	3.9.3.3	Buckling criteria used for component supports	Complete	6/29/84
98	3.9.3.3	Design of bolts	Complete	6/15/84
99a	3.9.5	Stress categories and limits for core support structures	Complete	6/15/84
99b	3.9.5	Stress categories and limits for core support structures	Complete	6/15/84
100a	3.9.6	10CFR50.55a paragraph (g)	Complete	6/29/84
100b	3.9.6	10CFR50.55a paragraph (g)	Complete	9/12/84 (Rev. 1)
101	3.9.6	PSI and ISI programs for pumps and valves	Complete	9/12/84 (Rev. 1)
102	3.9.6	Leak testing of pressure isolation valves	Complete	9/12/84 (Rev. 1)
103a1	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103a2	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103a3	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103a4	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TC A. SCHWENCER LETTER DATED
103a5	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103a6	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103a7	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103b1	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103b2	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103b3	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103b4	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103b5	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103b6	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103c1	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103c2	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103c3	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103c4	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
104	3.11	Environmental qualification of mechanical and electrical equipment	NRC Action	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
105	4.2	Plant-specific mechanical fracturing analysis	Complete	8/20/84 (Rev. 1)
106	4.2	Applicability of seismic andd LOCA loading evaluation	Complete	8/20/84 (Rev. 1)
107	4.2	Minimal post-irradiation fuel surveillance program	Complete	6/29/84
108	4.2	Gadolina thermal conductivity equation	Complete	6/29/84
109a	4.4.7	TMI-2 Item II.F.2	Complete	8/20/84
109b	4.4.7	TMI-2 Item II.F.2	Complete	8/20/84
110a	4.6	Functional design of reactivity control systems	Complete	8/30/84 (Rev. 1)
110b	4.6	Functional design of reactivity control systems	Complete	8/30/84 (Rev. 1)
111a	5.2.4.3	Preservice inspection program (components within reactor pressure boundary)	Complete	6/29/84
111b	5.2.4.3	Preservice inspection program (components within reactor pressure boundary)	Complete	6/29/84
111c	5.2.4.3	Preservice inspection program (components within reactor pressure boundary)	Complete	6/29/84
112a	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	8/30/84 (Rev. 1)
112b	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	8/30/84 (Rev. 1)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
112c	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	8/30/84 (Rev. 1)
112d	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	8/30/84 (Rev. 1)
112e	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	8/30/84 (Rev. 1)
113	5.3.4	GE procedure applicability	Complete	7/18/84
114	5.3.4	Compliance with NB 2360 of the Summer 1972 Addenda to the 1971 ASME Code	Complete	7/18/84
115	5.3.4	Drop weight and Charpy v-notch tests for closure flange materials	Complete	9/5/84 (Rev. 1)
116	5.3.4	Charpy v-notch test data for base materials as used in shell course No. 1	Complete	7/18/84
117	5.3.4	Compliance with NB 2332 of Winter 1972 Addenda of the ASME Code	Complete	8/20/84
118	5.3.4	Lead factors and neutron fluence for surveillance capsules	Complete	8/20/84
119	6.2	TMI item II.E.4.1	Complete	6/29/84
120a	6.2	TMI Item II.E.4.2	Complete	8/20/84
120b	6.2	TMI Item II.E.4.2	Complete	8/20/84
121	6.2.1.3.3	Use of NUREG-0588	Complete	7/27/84
122	6.2.1.3.3	Temperature profile	Complete	7/27/84
123	6.2.1.4	Butterfly valve operation (post accident)	Complete	6/29/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
124a	6.2.1.5.1	RPV shield annulus analysis	Complete	8/20/84 (Rev. 1)
124b	6.2.1.5.1	RPV shield annulus analysis	Complete	8/20/84 (Rev. 1)
124c	6.2.1.5.1	RPV shield annulus analysis	Complete	8/20/84 (Rev. 1)
125	6.2.1.5.2	Design drywell head differential pressure	Complete	6/15/84
126a	6.2.1.6	Redundant position indicators for vacuum breakers (and control room alarms)	Complete	8/20/84
126b	6.2.1.6	Redundant position indicators for vacuum breakers (and control room alarms)	Complete	8/20/84
127	6.2.1.6	Operability testing of vacuum breakers	Complete	8/20/84 (Rev. 1)
128	6.2.2	Air ingestion	Complete	7/27/84
129	6.2.2	Insulation ingestion	Complete	6/1/84
130	6.2.3	Potential bypass leakage paths	Complete	9/13/84 (Rev. 1)
131	6.2.3	Administration of secondary containment openings	Complete	7/18/84
132	6.2.4	Containment isolation review	Complete	6/15/84
133a	6.2.4.1	Containment purge system	Complete	8/20/84
133b	6.2.4.1	Containment purge system	Completa	8/20/84
133c	6.2.4.1	Containment purge system	Complete	8/20/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
134	6.2.6	Containment leakage testing	Complete	6/15/84
135	6.3.3	LPCS and LPCI injection valve interlocks	Complete	8/20/84
136	6.3.5	Plant-specific LOCA (see Section 15.9.13)	Complete	8/20/84 (Rev. 1)
137a	6.4	Control room habitability	Complete	8/20/84
137b	6.4	Control room habitability	Complete	8/20/84
137c	6.4	Control room habitability	Complete	8/20/84
138	6.6	Preservice inspection program for Class 2 and 3 components	Complete	6/29/84
139	6.7	MSIV leakage control system	Complete	6/29/84
140a	9.1.2	Spent fuel pool storage	Complete	9/7/84 (Rev. 2)
140b	9.1.2	Spent fuel pool storage	Complete	9/7/84 (Rev. 2)
140c	9.1.2	Spent fuel pool storage	Complete	9/7/84 (Rev. 2)
140d	9.1.2	Spent fuel pool storage	Complete	9/7/84 (Rev. 2)
141a	9.1.3	Spent fuel cooling and cleanup system	Complete	8/30/84 (Rev. 1)
141b	9.1.3	Spent fuel cooling and cleanup system	Complete	8/30/84 (Rev. 1)
141c	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/30/84 (Rev. 1)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
141d	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/30/84 (Rev. 1)
141e	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/30/84 (Rev. 1)
141f	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/30/84 (Rev. 1)
141g	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/30/84 (Rev. 1)
142a	9.1.4	Light load handling system (related to refueling)	Complete	8/15/84 (Rev. 1)
142b	9.1.4	Light load handling system (related to refueling)	Complete	8/15/84 (Rev. 1)
143a	9.1.5	Overhead heavy load handling	Complete	9/7/84
143b	9.1.5	Overhead heavy load handling	Complete	9/13/84
144a	9.2.1	Station service water system	Complete	8/15/84 (Rev. 1)
144b	9.2.1	Station service water system	Complete	8/15/84 (Rev. 1)
144c	9.2.1	Station service water system	Complete	8/15/84 (Rev. 1)
145	9.2.2	ISI program and functional testing of safety and turbine auxiliaries cooling systems	Closed (5/30/84-Aux.Sys.Mtg.)	6/15/84
146	9.2.6	Switches and wiring associated with HPCI/RCIC torus suction	Closed (5/30/84-Aux.Sys.Mtg.)	6/15/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
147a	9.3.1	Compressed air systems	Complete	9/21/84 (Rev. 2)
147b	9.3.1	Compressed air systems	Complete	9/21/84 (Rev. 2)
147c	9.3.1	Compressed air systems	Complete	9/21/84 (Rev. 2)
147d	9.3.1	Compressed air systems	Complete	9/21/84 (Rev. 2)
148	9.3.2	Post-accident sampling system (II.B.3)	Complete	9/12/84 (Rev. 1)
149a	9.3.3	Equipment and floor drainage system	Complete	7/27/84
149b	9.3.3	Equipment and floor drainage system	Complete	7/27/84
150	9.3.6	Primary containment instrument gas system	Complete	8/3/84 (Rev. 1)
151a	9.4.1	Control structure ventilation system	Complete	8/30/84 (Rev. 1)
151b	9.4.1	Control structure ventilation system	Complete	8/30/84 (Rev. 1)
152	9.4.4	Radioactivity monitoring elements	Closed (5/30/84-Aux.Sys.Mtg.)	6/1/84
153	9.4.5	Engineered safety features ventilation system	Complete	8/30/84 (Rev 2)
154	9.5.1.4.a	Metal roof deck construction classificiation	Complete	6/1/84
155	9.5.1.4.b	Ongoing review of safe shutdown capability	NRC Action	
156	9.5.1.4.c	Ongoing review of alternate shutdown capability	NRC Action	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
157	9.5.1.4.e	Cable tray protection	Complete	8/20/84
158	9.5.1.5.a	Class B fire detection system	Complete	6/15/84
159	9.5.1.5.a	Primary and secondary power supplies for fire detection system	Complete	6/1/84
160	9.5.1.5.b	Fire water pump capacity	Complete	8/13/84
161	9.5.1.5.b	Fire water valve supervision	Complete	6/1/84
162	9.5.1.5.c	Deluge valves	Complete	6/1/84
163	9.5.1.5.c	Manual hose station pipe sizing	Complete	6/1/84
164	9.5.1.6.e	Remote shutdown panel ventilation	Complete	6/1/84
165	9.5.1.6.g	Emergency diesel generator day tank protection	Complete	6/1/84
166	12.3.4.2	Airborne radioactivity monitor positioning	Complete	9/13/84 (Rev. 2)
167	12.3.4.2	Portable continuous air monitors	Complete	7/18/84
168	12.5.2	Equipment, training, and procedures for inplant iodine instrumentation	Complete	6/29/84
169	12.5.3	Guidance of Division B Regulatory Guides	Complete	7/18/84
170	13.5.2	Procedures generation package submittal	Complete	6/29/84
171	13.5.2	TMI Item I.C.1	Complete	6/29/84
172	13.5.2	PGP Commitment	Complete	6/29/84
173	13.5.2	Procedures covering abnormal releases of radioactivity	Complete	6/29/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
174	13.5.2	Resolution explanation in FSAR of TMI Items I.C.7 and I.C.8	Complete	6/15/84
175	13.6	Physical security	Open	
176a	14.2	Initial plant test program	Complete	8/13/84
176b	14.2	Initial plant test program	Complete	8/13/84
176c	14.2	Initial plant test program	Complete	7/27/84
176d	14.2	Initial plant test program	Complete	8/24/84 (Rev. 2)
176e	14.2	Initial plant test program	Complete	7/27/84
176f	14.2	Initial plant test program	Complete	8/13/84
176g	14.2	Initial plant test program	Complete	8/20/84
176h	14.2	Initial plant test program	Complete	8/13/84
176i	14.2	Initial plant test program	Complete	7/27/84
177	15.1.1	Partial feedwater heating	Complete	8/20/84 (Rev. 1)
178	15.6.5	LOCA resulting from spectrum of postulated piping breaks within RCP	NRC Action	
179	15.7.4	Radiological consequences of fuel handling accidents	NRC Action	
180	15.7.5	Spent fuel cask drop accidents	NRC Action	
181	15.9.5	TMI-2 Item II.K.3.3	Complete	6/29/84
182	15.9.10	TMI-2 Item II.K.3.18	Complete	6/1/84
183	18	Hope Creek DCRDR	Complete	8/15/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
184	7.2.2.1.e	Failures in reactor vessel level sensing lines	Complete	8/1/84 (Rev 1)
185	7.2.2.2	Trip system sensors and cabling in turbine building	Complete	6/1/84
186	7.2.2.3	Testability of plant protection systems at power	Complete	8/13/84 (Rev. 1)
187	7.2.2.4	Lifting of leads to perform surveillance testing	Complete	8/3/84
188	7.2.2.5	Setpoint methodology	Complete	8/1/84
189	7.2.2.6	Isolation devices	Complete	8/1/84
190	7.2.2.7	Regulatory Guide 1.75	Complete	6/1/84
191	7.2.2.8	Scram discharge volume	Complete	6/29/84
192	7.2.2.9	Reactor mode switch	Complete	8/15/84 (Rev. 1)
193	7.3.2.1.10	Manual initiation of safety systems	Complete	8/1/84
194	7.3.2.2	Standard review plan deviations	Complete	8/1/84 (Rev 1)
195a	7.3.2.3	Freeze-protection/water filled instrument and sampling lines and cabinet temperature control	Complete	8/1/84
195b	7.3.2.3	Freeze-protection/water filled instrument and sampling lines and cabinet temperature control	Complete	8/1/84
196	7.3.2.4	Sharing of common instrument taps	Complete	8/1/84
197	7.3.2.5	Microprocessor, multiplexer and computer systems	Complete	8/1/84 (Rev 1)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
198	7.3.2.6	TMI Item II.K.3.18-ADS actuation	Complete	8/20/84
199	7.4.2.1	IE Bulletin 79-27-Loss of non-class IE instrumentation and control power system bus during operation	Complete	8/24/84 (Rev. 1)
200	7.4.2.2	Remote shutdown system	Complete	8/15/84 (Rev 1)
201	7.4.2.3	RCIC/HPCI interactions	Complete	8/3/84
202	7.5.2.1	Level measurement errors as a result of environmental temperature effects on level instrumentation reference leg	Complete	8/3/84
203	7.5.2.2	Regulatory Guide 1.97	Complete	8/3/84
204	7.5.2.3	TMI Item II.F.1 - Accident monitoring	Complete	8/1/84
205	7.5.2.4	Plant process computer system	Complete	6/1/84
206	7.6.2.1	High pressure/low pressure interlocks	Complete	7/27/84
207	7.7.2.1	HELBs and consequential control system failures	Complete	8/24/84 (Rev. 1)
208	7.7.2.2	Multiple control system failures	Complete	8/24/84 (Rev. 1)
209	7.7.2.3	Credit for non-safety related systems in Chapter 15 of the FSAR	Complete	8/1/84 (Rev 1)
210	7.7.2.4	Transient analysis recording system	Complete	7/27/84
211a	4.5.1	Control rod drive structural materials	Complete	7/27/84
211b	4.5.1	Control rod drive structural materials	Complete	7/27/84
211c	4.5.1	Control rod drive structural materials	Complete	7/27/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL T A. SCHWENCER LETTER DATED
211d	4.5.1	Control rod drive structural materials	Complete	7/27/84
211e	4.5.1	Control rod drive structural materials	Complete	7/27/84
212	4.5.2	Reactor internals materials	Complete	7/27/84
213	5.2.3	Reactor coolant pressure boundary material	Complete	7/27/84
214	6.1.1	Engineered safety features materials	Complete	7/27/84
215	10.3.6	Main steam and feedwater system materials	Complete	7/27/84
216a	5.3.1	Reactor vessel materials	Complete	7/27/84
216b	5.3.1	Reactor vessel materials	Complete	7/27/84
217	9.5.1.1	Fire protection organization	Complete	8/15/84
218	9.5.1.1	Fire hazards analysis	Complete	6/1/84
219	9.5.1.2	Fire protection administrative controls	Complete	8/15/84
220	9.5.1.3	Fire brigade and fire brigade training	Complete	8/15/84
221	8.2.2.1	Physical separation of offsite transmission lines	Complete	8/1/84
222	8.2.2.2	Design provisions for re-establishment of an offsite power source	Complete	9/14/84 (Rev. 1)
223	8.2.2.3	Independence of offsite circuits between the switchyard and class IE buses	Complete	9/26/84 (Rev. 3)
224	8.2.2.4	Common failure mode between onsite and offsite power circuits	Complete	9/26/84 (Rev. 2)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
225	8.2.3.1	Testability of automatic transfer of power from the normal to preferred power source	Complete	9/21/84 (Rev. 1)
226	8.2.2.5	Grid stability	Complete	8/13/84 (Rev. 1)
227	8.2.2.6	Capacity and capability of offsite circuits	Complete	8/1/84
228	8.3.1.1(1)	Voltage drop during transient conditions	Complete	8/1/84
229	8.3.1.1(2)	Basis for using bus voltage versus actual connected load voltage in the voltage drop analysis	Complete	8/1/84
230	8.3.1.1(3)	Clarification of Table 8.3-11	Complete	8/1/84
231	8.3.1.1(4)	Undervoltage trip setpoints	Complete	8/1/84
232	8.3.1.1(5)	Load configuration used for the voltage drop analysis	Complete	8/1/84
233	8.3.3.4.1	Periodic system testing	Complete	9/21/84 (Rev. 2)
234	8.3.1.3	Capacity and capability of onsite AC power supplies and use of administrative controls to prevent overloading of the diesel generators	Complete	8/1/84
235	8.3.1.5	Diesel generators load acceptance test	Complete	9/21/84 (Rev. 2)
236	8.3.1.6	Compliance with position C.6 of RG 1.9	Complete	8/1/84
237	8.3.1.7	Decription of the load sequencer	Complete	9/21/84 (Rev. 1)
238	8.2.2.7	Sequencing of loads on the offsite power system	Complete	9/21/84 (Rev. 1)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
239	8.3.1.8	Testing to verify 80% minimum voltage	Complete	8/15/84
240	8.3.1.9	Compliance with BTP-PSB-2	Complete	8/1/84
241	8.3.1.10	Load acceptance test a[illegible] prolonged no load operation of the diesel generator	Complete	9/21/84 (Rev. 3)
242	8.3.2.1	Compliance with position 1 of Regulatory Guide 1.128	Complete	9/13/84 (Rev. 1)
243	8.3.3.1.3	Protection or qualification of Class 1E equipment from the effects of fire suppression systems	Complete	9/13/84 (Rev. 1)
244	8.3.3.3.1	Analysis and test to demonstrate adequacy of less than specified separation	Complete	9/28/84 (Rev. 2A)
245	8.3.3.3.2	The use of 18 versus 36 inches of separation between raceways	Complete	9/28/84 (Rev. 2B)
246	8.3.3.3.3	Specified separation of raceways by analysis and test	Complete	8/1/84
247	8.3.3.5.1	Capability of penetrations to withstand long duration short circuits at less than maximum or worst case short circuit	Complete	9/13/84 (Rev. 1)
248	8.3.3.5.2	Separation of penetration primary and backup protections	Complete	8/1/84
249	8.3.3.5.3	The use of bypassed thermal overload protective devices for penetration protections	Complete	8/1/84
250	8.3.3.5.4	Testing of fuses in accordance with R.G. 1.63	Complete	8/1/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
251	8.3.3.5.5	Fault current analysis for all representative penetration circuits	Complete	9/24/84 (Rev. 3)
252	8.3.3.5.6	The use of a single breaker to provide penetration protection	Complete	9/21/84 (Rev. 2)
253	8.3.[illegible].1.4	Commitment to protect all Class 1E equipment from external hazards versus only class 1E equipment in one division	Complete	9/28/84 (Rev. 3A)
254	8.3.3.1.5	Protection of class 1E power supplies from failure of unqualified class 1E loads	Complete	9/14/84 (Rev. 1)
255	8.3.2.2	Battery capacity	Complete	8/1/84
256	8.3.2.3	Automatic trip of loads to maintain sufficient battery capacity	Complete	9/13/84 (Rev. 1)
257	8.3.2.5	Justification for a 0 to 13 second load cycle	Complete	9/13/83 (Rev. 1)
258	8.3.2.6	Design and qualification of DC system loads to operate between minimum and maximum voltage levels	Complete	8/1/84
259	8.3.3.3.4	Use of an inverter [illegible] an isolation de[illegible]ce	Complete	10/3/84 (Rev. 3)
260	8.3.3.3.5	Use of a sin[illegible]le breaker tripped by a LOCA signal u[illegible]ed as an isolation dev[illegible]ce	Complete	10/3/84 (Rev. 2)
261	8.3.3.3.6	Automatic transfer of loads and interconnection between redundant divisions	Complete	9/13/84 (Rev. 1)
262	11.4.2.d	Solid waste control program	Complete	8/20/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
263	11.4.2.e	Fire protection for solid radwaste storage area	Complete	8/13/84
264	6.2.5	Sources of oxygen	Complete	8/20/84
265	6.8.1.4	ESF Filter Testing	Complete	8/13/84
266	6.8.1.4	Field leak tests	Complete	8/13/84
267	6.4.1	Control room toxic chemical detectors	Complete	8/13/84
268		Air filtration unit drains	Complete	9/13/84 (Rev. 1)
269	5.2.2	Code cases N-242 and N-242-1	Complete	8/20/84
270	5.2.2	Code case N-252	Complete	8/20/84
TS-1	2.4.14	Closure of watertight doors to safety-related structures	Open	
TS-2	4.4.4	Single recirculation loop operation	Open	
TS-3	4.4.5	Core flow monitoring for crud effects	Complete	6/1/84
TS-4	4.4.6	Loose parts monitoring system	Open	
TS-5	4.4.9	Natural circulation in normal operation	Open	
TS-6	6.2.3	Secondary containment negative pressure	Open	
TS-7	6.2.3	Inleakage and drawdown time in secondary containment	Open	
TS-8	6.2.4.1	Leakage integrity testing	Open	
TS-9	6.3.4.2	ECCS subsystem periodic component testing	Open	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TC A. SCHWENCER LETTER DATED
TS-10	6.7	MSIV leakage rate		
TS-11	15.2.2	Availability, setpoints, and testing of turbine bypass system	Open	
TS-12	15.6.4	Primary coolant activity		
LC-1	4.2	Fuel rod internal pressure criteria	Complete	6/1/84
LC-2	4.4.4	Stability analysis submitted before second-cycle operation	Open	

ATTACHMENT 2 DATE: 10/3/84

DRAFT SER SECTIONS AND DATES PROVIDED

SECTION	DATE	SECTION	DATE
3.1			
3.2.1		11.4.1	See Notes 1&5
3.2.2		11.4.2	See Notes 1&5
5.1		11.5.1	See Notes 1&5
5.2.1		11.5.2	See Notes 1&5
6.5.1	See Notes 1&5	13.1.1	See Note 4
8.1	See Note 2	13.1.2	See Note 4
8.2.1	See Note 2	13.2.1	See Note 4
8.2.2	See Note 2	13.2.2	See Note 4
8.2.3	See Note 2	13.3.1	See Note 4
8.2.4	See Note 2	13.3.2	See Note 4
8.3.1	See Note 2	13.3.3	See Note 4
8.3.2	See Note 2	13.3.4	See Note 4
8.4.1	See Note 2	13.4	See Note 4
8.4.2	See Note 2	13.5.1	See Note 4
8.4.3	See Note 2	15.2.3	
8.4.5	See Note 2	15.2.4	
8.4.6	See Note 2	15.2.5	
8.4.7	See Note 2	15.2.6	
8.4.8	See Note 2	15.2.7	
9.5.2	See Note 3	15.2.8	
9.5.3	See Note 3	15.7.3	See Notes 1&5
9.5.7	See Note 3	17.1	8/3/84
9.5.8	See Note 3	17.2	8/3/84
10.1	See Note 3	17.3	8/3/84
10.2	See Note 3	17.4	8/3/84
10.2.3	See Note 3		
10.3.2	See Note 3		
10.4.1	See Note 3		
10.4.2	See Notes 3&5		
10.4.3	See Notes 3&5		
10.4.4	See Note 3		
11.1.1	See Notes 1&5		
11.1.2	See Notes 1&5		
11.2.1	See Notes 1&5		
11.2.2	See Notes 1&5		
11.3.1	See Notes 1&5		
11.3.2	See Notes 1&5		

Notes:

1. Open items provided in letter dated July 24, 1984 (Schwencer to Mittl)
2. Open items provided in June 6, 1984 meeting
3. Open items provided in April 17-18, 1984 meeting
4. Open items provided in May 2. 1984 meting
5. Draft SER Section provided in letter dated August 7, 1984 (Schwencer to Mittl)

CT:db

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Attachment 3

DSER OPEN ITEMS

259	8.3.3.3.4	Use of an inverter as an isolation device
260	8.3.3.3.5	The use of a single breaker tripped by a LOCA signal as an isolation device

ATTACHMENT 4

DSER Open Item No. 259 (DSER Section 8.3.3.3.4)

USE OF AN INVERTER AS AN ISOLATION DEVICE

By Amendment 4 to the FSAR, the applicant indicated that the non-Class 1E public address system distribution panel shown on sheet 2 of Figure 8.3-11 of the FSAR is supplied power from the Class 1E dc system through an inverter. The applicant further stated that this inverter is an acceptable isolation device per IEEE-384-1981, Section 7.1.2.3. The staff does not agree. Test and analysis to demonstrate the adequacy of an inverter as an isolation device will be pursued with the applicant.

RESPONSE

The response to Question 430.33 has been revised to state that the inverter will be tested as an isolation device. In the event that the tests are not successful, the non Class 1E loads will be removed or the cables will be re-routed.

Rev 2

Question 430.33

DSER Open Item No. 260 (DSER Section 8.3.3.3.5)

THE USE OF A SINGLE BREAKER TRIPPED BY A LOCA SIGNAL AS AN ISOLATION DEVICE

Section 8.3.1.1.2 of the FSAR indicates that the Class 1E system provides power to non-Class 1E loads. Non-Class 1E loads are connected to the Class 1E system through a single breaker that is tripped automatically by a LOCA signal. The single breaker tripped by a LOCA signal provides acceptable isolation between Class 1E and non-Class 1E circuits for the design basis accident--LOCA. However, for other design basis accidents or operating occurrences that do not generate a LOCA signal (such as loss of offsite power, design basis exposure fire, seismic events, etc.), it is the staff concern that a single breaker may not provide acceptable isolation.

By Amendment 4 to the FSAR, the applicant indicated that protecttive device coordination studies show that the single breaker time overcurrent trip characteristics will trip to clear a fault prior to initiation of a trip of a upstream breaker. Identification of all non-Class 1E circuits being isolated using a single breaker trip by LOCA signal, periodic testing of breaker coordination, and capability of breaker to trip prior to any versus only upstream breaker and for all versus only circuit faults, will be pursued with the applicant.

RESPONSE

Response to Question 430.33 has been revised to provide the requested information.

QUESTION 430.33 (SECTION 8.3.1 and 8.3.2)

Section 8.3.1.1.2 of the FSAR indicates that the Class 1E system provides power to non-Class 1E loads. Non-Class 1E loads are connected to the Class 1E system through a single breaker that is tripped automatically by a LOCA signal. The single breaker tripped by a LOCA signal provides acceptable isolation between Class 1E and Non-Class 1E circuits for the design basis accident - LOCA. However, for other design basis accidents or operating occurrances that do not generate a LOCA signal (such as loss of offsite power, design basis exposure fire, seismic events, etc), it is the staff concern that a single breaker may not provide acceptable isolation. Provide an analysis, in accordance with the guidelines of Section 4.9 of IEEE Standard 308-1974, that demonstrates that failure of anyone or simultanous combined failure of all non Class 1E loads will not prevent any of the four channels of Class 1E power from performing its safety function. The analysis should consider, but not be limited to, (1) capacity and capability of onsite and offsite power supplies and their associated distribution system to supply power to Class 1E loads within their design ratings for all modes of plant operation, (2) the guidelines of Section 7.1.2.1 of IEEE standard 384-1981, (3) an analysis of diesel generator loadings for loss of offsite power similiar to that presented in Tables 8.3-2 through 8.3-6 of the FSAR, (4) the failure of the Non Class 1E dc system that supplies control power to the subject non Class 1E loads, and (5) a similiar analysis of the Class 1E dc system if non-Class 1E loads are connected.

RESPONSE

The following discussion demonstrates the adequacy of employing a single circuit breaker tripped by a LOCA signal as an isolation device between a Class 1E power bus and a non-Class 1E load for design bases (is) event that do not generate LOCA signals.

X

Figure 430.33-1 shows the two configurations that employ a circuit breaker tripped by a LOCA signal as an isolation device. The two configurations are:

a. A Class 1E unit substation supplies a non-Class 1E motor control center (MCC) or a motor load through Class 1E circuit breaker B.

b. A Class 1E motor control center supplies through Class 1E circuit breaker D, a non-Class 1E distribution panel.

The Class 1E circuit breakers B and D are qualified to operate for HCGS seismic and environmental parameters for all design basis events. These circuit breakers will trip to isolate their

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respective Class 1E power supply buses from the non-Class 1E loads in the event the non-Class 1E loads fail. This applies whether the plant is supplied from an offsite source or an onsite source. Thus, the failure of the non-Class 1E loads supplied from Class 1E power supply buses will not prevent any of the four channels of Class 1E power supplies from performing its safety function. INSERT A FROM PAGE 430.33-2A

COMPLIANCE WITH GUIDELINES OF SECTION 7.1.2.1 OF IEEE 384-1981

Protective device coordination studies for devices shown in Figure 430.33-1 have shown that the time-overcurrent trip characteristics of circuit breakers A, B, C, and D are such that:

a. Circuit breaker B will trip to clear a fault current prior to initiation of a trip of circuit breaker A.

b. Circuit breaker D will trip to clear a fault current prior to initiation of a trip of circuit breaker C.

Both the onsite and offsite powers supply sources are separately capable of supplying the necessary fault current for sufficient time to ensure the proper protective device coordination without loss of function of Class 1E loads. INSERT B FROM PAGE 430.33-2A

STANDBY DIESEL GENERATOR LOADINGS FOR LOSS OF OFFSITE POWER

Table 8.3-1 tabulates the loads, their KW ratings, and loading sequences for design basis accident (DBA) and loss of offsite power (LOP) scenarios. It can be verified by inspecting Table 8.3-1 that DBA loading of the SDGs is the limiting case with respect to the loading capability of the SDGs.

FAILURE OF THE NON-CLASS 1E DC SYSTEM THAT SUPPLIES CONTROL POWER TO THE SUBJECT NON-CLASS 1E LOADS

For configuration (a) (described above) the circuit breaker B supplying a Non-Class 1E MCC or a motor load is controlled by Class 1E 125 V dc control power supply. For a non-Class 1E motor load, a non-Class 1E circuit breaker is provided downstream of circuit breaker B. This non-Class 1E circuit breaker (GE-AKR type) is controlled by a non-Class 1E 125 V dc control power. GE-AKR type circuit breakers are direct acting trip devices and do not require external control power supply for tripping for electrical fault conditions. Therefore, the failure of the control power supply does not prevent the circuit breaker ~~to~~ tripping in response to the failure of non-Class 1E motor load. from

INSERT C FROM PAGES 430.33-2B

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INSERT A

The Class IE onsite ac sources and the offsite power sources and their distribution system are of sufficient capacity and capability to supply power to both Class IE and non-Class IE loads during all plant conditions. In the event of a LOCA the non-Class IE loads are automatically tripped from the class IE buses in accordance with Position C.1 of Regulatory Guide 1.75. IN ADDITION, CABLES FROM THE CLASS IE BUSES TO THE NON-CLASS IE LOADS ARE ROUTED IN RIGID STEEL CONDUITS OR TRAYS. WHERE TRAY ROUTING IS USED, NON-CLASS IE CABLES ASSOCIATED WITH OTHER IE CHANNE ARE NOT RUN TOGETHER IN THE SAME TRA

AN OPERATION DESIGN CHANGE CONTROL PROGRAM WILL BE IN EFFECT AT THE HOPE CREEK PLANT TO ASSURE THAT FUTURE ADDITIONS/MODIFICATIONS WILL COMA WITH THIS REQUIREMENT. ADDITIONALLY, THE PERTINENT DESIGN DOCUMENTS WILL BE PROVIDED WITH A NOTATIONS TO REFLECT THIS REQUIREMENT.

INSERT B

Periodic testing of the breaker time-overcurrent trip characteristics will be performed to demonstrate that the circuit breaker trip function remains within required limits. Table 430.33-1 identifies the non-Class IE loads that are supplied through circuit breakers B and D of Figure 430.33-1.

DSER OPEN ITEM

QUESTION 430.33 Insert "C"

ANALYSIS FOR SUPPLYING NON-CLASS 1E FROM CLASS 1E DC SYSTEMS

Figure 8.3.11 shows non-Class 1E public address system distribution panel 10J496 supplied from a Class 1E dc power bus 10D410 through a Class 1E inverter in UPS unit 10D496. The inverter is an acceptable isolation device per IEEE-384-1981, Section 7.1.2.3. Therefore, a failure in the non-Class 1E distribution panel 10J496 will not degrade Class 1E dc system bus 10D410.

The HCGS UPS system will be tested to demonstrate the adequacy of an inverter being applied as an isolation device. The test will demonstrate that voltage, current, and frequency on the Class 1E side of the UPS are not degraded below acceptable levels when maximum credible voltage or current transient is applied on the non-Class 1E side of the UPS system. The tests to be performed will simulate all operating modes for which the HCGS UPS system is designed. The tests will include the following types of faults at the UPS output location:

a. Phase to ground
b. Neutral to ground
c. Phase to neutral without ground
d. Hot short (460 Vac)

A test plan is submitted separately for the staff's review. The test report and any associated analysis of the test results will be submitted in December 1984.

An analysis has been performed to support the values used for the acceptance criteria for voltages. This analysis shows that the voltages specified will not cause misoperation or loss of any electrical equipment connected to the supply buses.

The results of this analysis for the ac systems is stated in FSAR Section 8.3.1.2.1 and the calculated results are shown in Table 8.3-11. The results of the dc analysis are contained in FSAR Section 8.3.2. These results indicate that the 125 volt dc system has an acceptable operating capability with battery voltage variations of 35 volts (140 volts dc to 105 volts dc). The test acceptance criterion limits the bus voltage variation to 105-135 volts.

In addition, the acceptance values for the test currents are well below the level that would cause the infeed breakers to the UPS supply buses to trip. These values are as follows:

Circuit	Acceptance Current	Infeed breaker Setting
Normal 480 VAC Supply	0-55 amperes continuous with a maximum peak not to exceed 132 amperes and no value above 55 amperes shall persist for longer than 10 mS	600 amperes Pick-up

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Circuit	Acceptance Current	Infeed breaker Setting
Back-up 480 VAC Supply	0-78 amperes continuous with a maximum peak not to exceed 500 amperes and no value above 78 amperes shall persist for longer than 10 mS	600 amperes Pick-up
Alternate 125 VDC Supply	The bus voltage variation of 105-135 volts will hold for the following cases: (1) With the UPS energized but without load the input current should not exceed 56 amperes (2) With the UPS input current at 56 amperes the input current should not exceed the range of 0-56 amperes (3) With the UPS input current at 158 amperes the input current should not exceed the range of of 0-158 amperes	2000 ampere fuse

The following is justification that the above acceptance current values do not adversely effect the Class 1E buses. The 480 volt ac back-up feed is supplied from a 480 volt Class 1E motor control center which in turn is supplied from a 480 volt Class 1E unit substation. The infeed breaker to the MCC is an AKR - 50 which has a 600 ampere pick-up setting for its time delay trip setpoint. This allows the largest motor loads on the MCC, in combination with the maximum acceptable current spike of the UPS acceptance values (500 amperes for not longer than 10 mS), to persist for 25 seconds. Since the 500 ampere spike is completed in 10 mS, the largest motor loads then have 55 seconds to accelerate. This is 48 seconds longer than the time delay for the primary protective device for the largest motor and, therefore, it is not possible for any of the Class 1E loads to be disabled. The inrush current of the normal ac feed is 132 amperes for 10 mS which is less than the 480V ac backup supply. The normal 480V supply breaker is the same type and size as the 480V back-up supply breaker. Therefore the Class 1E loads on the MCC's from the normal and backup 480Vac supply are not affected by any short circuits on the output of the inverter.

The alternate 125V dc supply full load amperes are already included in the 125 volt battery load profiles. The maximum current duty on any of the 125 volt Class 1E batteries is 451.1 amperes (battery 1AD411). The impedance of the conductors from the battery to the 125 volt dc bus is such that the voltage drop for the specified load profiles does not cause the 125 volt bus to drop below 105 volts.

If the testing can not demonstrate adequacy of the UPS as an isolation device, then an isolation transformer will be added between the inverter and the distribution panel. The test plan for the isolation transformer is also submitted separately for the staff's review.

In the event of failure of both tests the non-Class 1E loads associated with the UPS system will be removed from the Class 1E buses or the cables to these loads will be re-routed so as to be separated from Class 1E cables associated with other Class 1E channels or an isolation means acceptable to the staff will be employed.

Rev. 3

TABLE 430.33-1

NON-CLASS IE LOADS CONNECTED TO CLASS IE BUSES THROUGH CIRCUIT BREAKER TRIPPED BY LOCA SIGNAL

LOAD NO.	NON-CLASS IE LOAD DESCRIPTION	CLASS IE BUS	CLASS IE CIRCUIT BREAKER NO.
1	Reactor Auxiliary Cooling System Pump 1AP209	10B410	52-41011
2	Radwaste and Service Area MCC 10B313	10B410	52-41014
3	Reactor Building Supply Air Handling Unit 1BVH300	10B410	52-41024
4	Reactor Auxiliary Cooling System Pump 1BP209	10B420	52-42011
5	Radwaste and Service Area MCC 10B323	10B420	52-42014
6	Reactor Building Exhaust Fan 1BV301	10B420	52-42024
7	Reactor Building Supply Air Handling Unit 1CVH300	10B430	52-43024
8	Control Rod Drive Pump 1AP207	10B430	52-43014
9	Control Rod Drive Pump 1BP207	10B440	52-44014
10	Reactor Building Supply Air Handling Unit 1AVH300	10B440	52-44024
11	Radwaste Area Supply Fan 0BV316	10B440	52-44034
12	Reactor Area MCC 10B252	10B450	52-45011
13	Radwaste Area Exhaust Fan 0AV305	10B450	52-45014
14	Emergency Instrument Air Compressor 10K100	10B450	52-45024
15	Reactor Building Exhaust Fan 1CV301	10B450	52-45034
16	Reactor Area MCC 10B262	10B460	52-46011
17	Radwaste Area Exhaust Fan 0BV305	10B460	52-46014
18	Reactor Area MCC 10B272	10B470	52-47011
19	Radwaste Area Exhaust Fan 0CV305	10B470	52-47014
20	Radwaste Area Supply Fan 0AV316	10B470	52-47024
21	Technical Support Center MCC 00B474	10B470	52-47031

DSER OPEN ITEM 260

TABLE 430.33-1
CONTINUED

22	Reactor Area MCC 1DB282	1DB480	52-48011
23	Reactor Building Exhaust Fan 1AV301	1DB480	52-48029
~~24~~	~~NSSS Computer Inverter 1DD485~~	~~1DB441~~	~~52-441023~~
25	Public Address System Inverter 1DD496	1DB451	52-451023
~~26~~	~~BOP Computer Inverter 1AD492~~	~~1DB461~~	~~52-461023~~
27	Security System Inverter 0AD495	1DB471	52-471023
~~28~~	~~BOP Computer Inverter 1BD492~~	~~1DB481~~	~~52-481023~~

DSER OPEN ITEM 260

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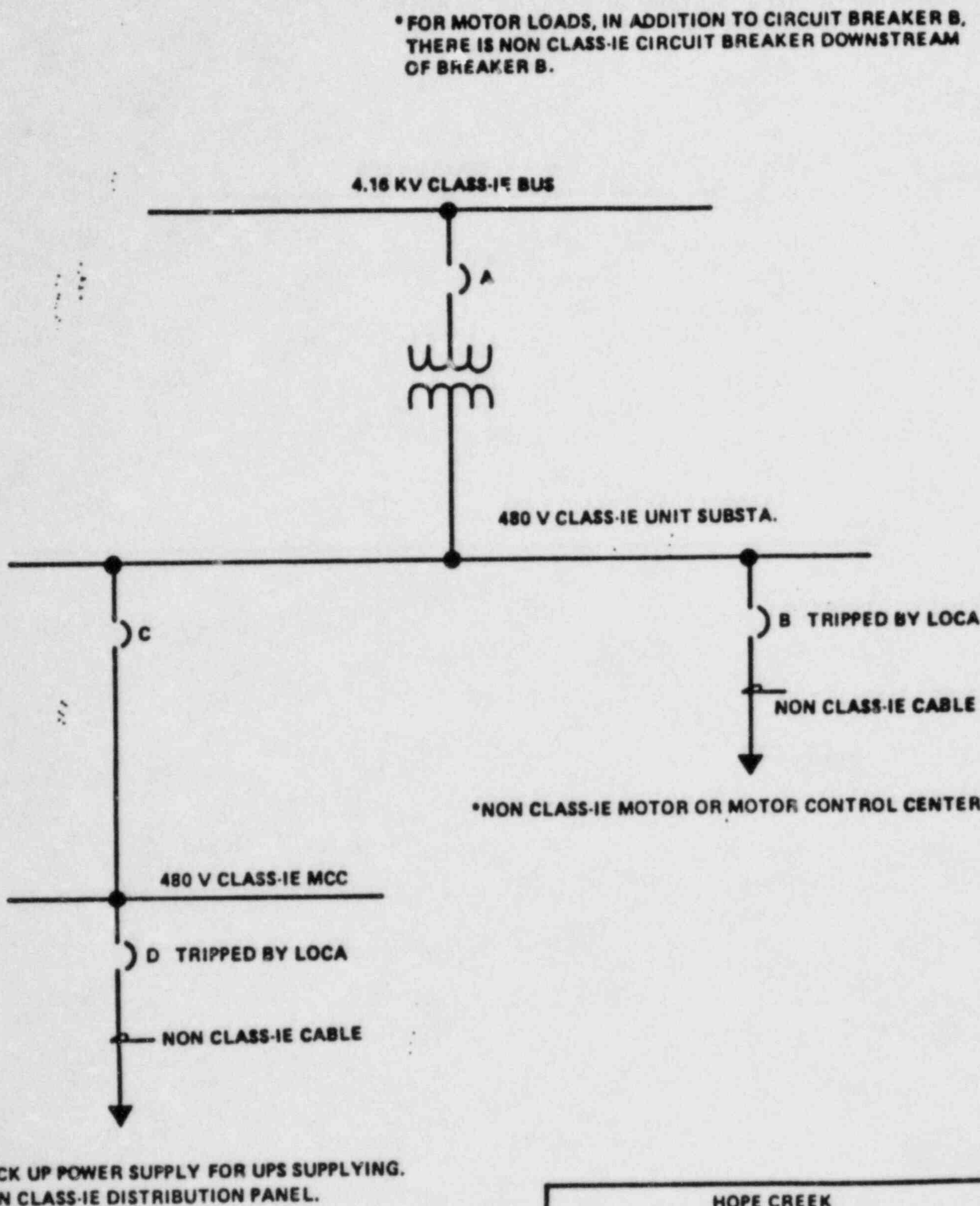


HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

ISOLATION BETWEEN CLASS-IE POWER SUPPLIES AND NON CLASS-IE LOADS— TRIPPING CIRCUIT BREAKER

FIGURE 430.33-1 AMENDMENT 4, 1/84

TEST PROCEDURE, ISOLATION VERIFICATION

S/N 9743 1E 20KVA UPS (INSTRUMENTATION AC POWER SUPPLY)

FOR PUBLIC SERVICE ELECTRIC & GAS CO.
HOPE CREEK GENERATING STATION
PO. 10855-E-154 (Q)-AC

OBJECTIVE:

TESTING TO ESTABLISH THE UPS SYSTEM AS A CIRCUIT ISOLATION SYSTEM.

PASS CRITERIA:

DEFINITION OF ISOLATION DEVICE OR SYSTEM: A DEVICE OR SYSTEM IS CONSIDERED TO BE A CIRCUIT ISOLATION DEVICE IF IT IS APPLIED SUCH THAT THE MAXIMUM CREDIBLE VOLTAGE OR CURRENT TRANSIENT APPLIED TO THE NON CLASS 1E SIDE OF THE DEVICE WILL NOT DEGRADE THE CLASS 1E CIRCUIT ON THE OTHER SIDE OF THAT DEVICE.

CIRCUIT	NORMAL VARIATION
ALT. DC. SUPPLY	105-135 VDC 0-FULL LOAD ADC
NORMAL AC SUPPLY	480+10% V(L-L) 3 PHASE 0-55A, 0-132AP FOR 10MSEC
BACK UP AC SUPPLY	480+10% V 1 PHASE 0-78A, 0-500AP FOR 10MSEC

ANY VARIATIONS OUTSIDE OF NORMAL VARIATIONS SPECIFIED, WILL BE ANALYZED ON A CASE BY CASE BASIS.

FAULT LOCATION AND TYPE

FAULTS WILL BE APPLIED TO UPS SYSTEM OUTPUT TERMINALS BY CLOSING A SWITCH AS REQUIRED.

FAULT TYPES:

1. PHASE (HOT) TO GROUND
2. NEUTRAL TO GROUND
3. PHASE TO NEUTRAL W/O GROUND
4. 480VAC APPLIED ACROSS UPS OUTPUT W/O GROUND (HOT SHORT)

THE CONDITION OF THE THREE CLASS 1E SOURCES WILL BE MONITORED THROUGH SUITABLE SIGNAL CONDITIONERS, BY GOULD INC., 2000W SERIES HIGH FREQUENCY RECORDING SYSTEM.

Rev. 3

TEST PROCEDURES

1.0 GENERAL NOTES

1.1 BEFORE STARTING TEST DETERMINE AND RECORD ALL SIGNAL CONDITIONER TRANSFER RATION (MULTIPLIER) VALUES.

1.2 NORMAL SYSTEM OPERATION DURING EACH TEST

A. CONNECTION PER FIG. 1.
B. THE LOAD ON THE UPS SHALL BE ADJUSTED FOR EACH OF THREE SEPARATE TESTS FOR EACH UPS INPUT SOURCE:
(1) NO LOAD
(2) OUTPUT LOAD AT .08 PF TO ACHIEVE 56 AMPERES INPUT CURRENT WHEN FED FROM 125 VOLT DC. LOAD SHOULD REMAIN THE SAME FOR AC INPUTS
(3) OUTPUT LOAD AT .08 PF TO ACHIEVE 158 AMPERES INPUT CURRENT WHEN FED FROM 125 VOLT DC. LOAD SHOULD REMAIN THE SAME FOR AC INPUTS
C. UPS POWERED BY "ALTERNATE" DC SOURCE (BATTERY) AND ONE OR BOTH AC SOURCES, "NORMAL" & "BACK-UP"
D. STATIC SWITCH IN "PREFERRED" POSITION.
E. ALL BREAKERS & SWITCHES CLOSED, BOTH BYPASS SWITCHES IN "NORMAL" POSITION
"TEST" SWITCH - CENTERED
"RETURN MODE" SWITCH - IN "AUTO" POSITION
"ISOLATION" TOGGLE SWITCHES - ON
"SYNC" TOGGLE SWITCH - ON

1.3 TEST INSTRUMENTATION

A. GOULD INC., MODEL 2800W HIGH FREQUENCY RECORDING SYSTEM. EIGHT CHANNEL, INDEPENDENT SCALE SELECT ±.050 TO ±500 VOLTS FULL SCALE.
B. POTENTIAL TRANSFORMER 480V, 60HZ PRIMARY 120V SECONDARY (4:1 RATIO).
C. CURRENT TRANSFORMER 1000:1 RATIO WITH 10 OHM BURDEN RESISTOR. (.01V/A).
D. WIDEBAND DC ISOLATION AMPLIFIER, GOULD INC. MODEL 13-4615-10 OR EQUIVALENT.

1.4 TEST FACILITY AND EQUIPMENT

A. DC SUPPLY - C&D 4LCW-15 BATTERY (60 CELLS, 80KW FOR 30 MIN.) AND BATTERY CHARGER.
B. AC SUPPLY - 480V, 3 PHASE, 4W, 60 HZ, 1200A GROUNDED NEUTRAL.
C. AC LOAD BANK - 0-30KW OR 0-30KVA @ 0.8PF.
D. FAULT APPLICATION DEVICE - G.E. CIRCUIT BREAKER TJC 36400G 400A, 3P. MAGNETIC ONLY.
E. HOT FAULT SOURCE - TRANSFORMER, 1 PH 480:120V 30KVA OR LARGER.

2.0 TEST PROCEDURE

2.1 BASE LINE DATA

START UP THE UPS WITH ALL SOURCES AVAILABLE. SET UP "NORMAL OPERATION" PER 1.2 AND ALLOW SYSTEM TO WARM UP FOR AT LEAST 30 MINUTES.

A1. METERING AND CONNECTIONS PER FIG. 2 AND "BACKUP SOURCE" BREAKER OPEN. RECORD IN "STORE" MODE AT 20KHZ TIME BASE. COPY MEMORY TO PAPER.
A2. REPEAT A1 EXCEPT USE 500HZ TIME BASE.
B1. WITH METERING AND CONNECTIONS PER FIG. 2 AND "NORMAL SOURCE" BREAKER OPEN. RECORD IN "STORE" MODE AT 20KHZ TIME BASE. COPY MEMORY TO PAPER.
B2. REPEAT B1 EXCEPT STATIC SWITCH TRANSFERRED TO BACKUP.
B3. REPEAT B1 EXCEPT USE 500HZ TIME BASE.
B4. REPEAT B2 EXCEPT USE 500HZ TIME BASE.

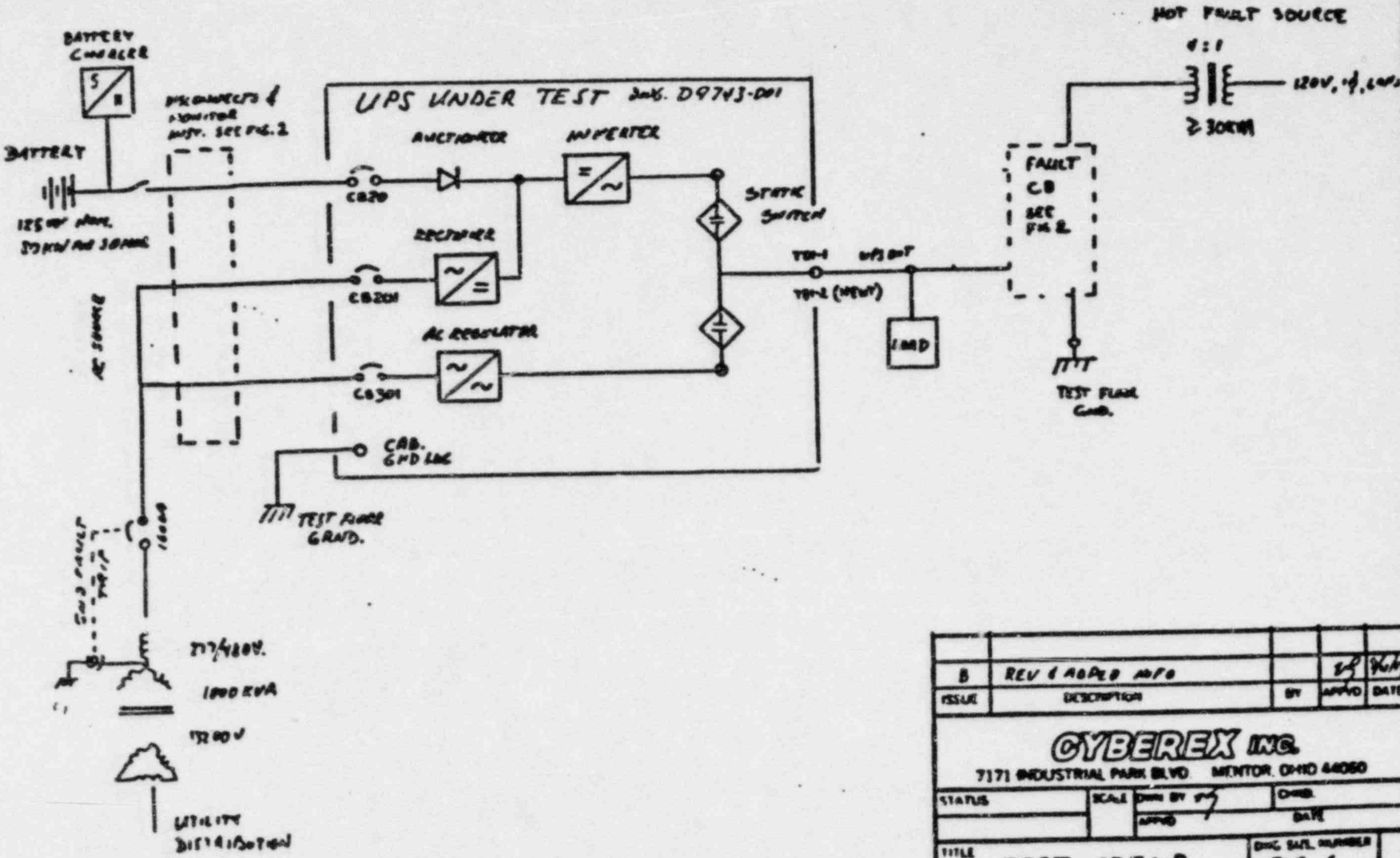
Rev.3

2.2 FAULT TESTING

C0. METERING AND CONNECTIONS PER FIG 2, RECORDER IN MANUAL TRIGGER MODE. APPLY FAULT BY CLOSING "FAULT" CB AND AT THE SAME TIME (OR 0 TO 10 MILLISECONDS BEFORE) TRIGGER THE RECORDER IN "STORE" MODE. REMOVE THE FAULT AND RECORD THE MEMORY TO PAPER.
AFTER EACH FAULT APPLICATION CHECK THE UPS FOR DAMAGE. REPAIR THE UPS IF REQUIRED BEFORE PROCEEDING.

C1. INSTALL JUMPER "A" TO "FAULT" CB WITH "BACKUP SOURCE" CB OPEN WITH RECORDER AT 20KHZ TIME BASE APPLY FAULT PER CO.
C2. REPEAT C1 EXCEPT WITH 500HZ TIME BASE.
C3. OPEN "NORMAL SOURCE" CB AND CLOSE "BACKUP" WITH RECORDER 20KHZ TIME BASE APPLY FAULT PER CO.
C4. REPEAT C3 EXCEPT WITH 500HZ TIME BASE.
C5. REPEAT C1, C2, C3 & C4 WITH JUMPER "B" INSTEAD OF "A" CONNECTED TO "FAULT" CB.
C6. REPEAT C1, C2, C3, & C4 WITH JUMPER "C" INSTEAD OF "A" CONNECTED TO "FAULT" CB.
C7. REPEAT C1, C2, C3, & C4 WITH CONNECTIONS TO HOT FAULT SOURCE (UPS RUNNING AT NO LOAD).

2.3 COMPLETE TEST SUMMARY SHEET FOR EACH TEST OR TEST GROUP.



ISSUE	DESCRIPTION	BY	APPVD	DATE
B	REV & ADDED INFO			

CYBEREX INC.
7171 INDUSTRIAL PARK BLVD. MENTOR, OHIO 44060

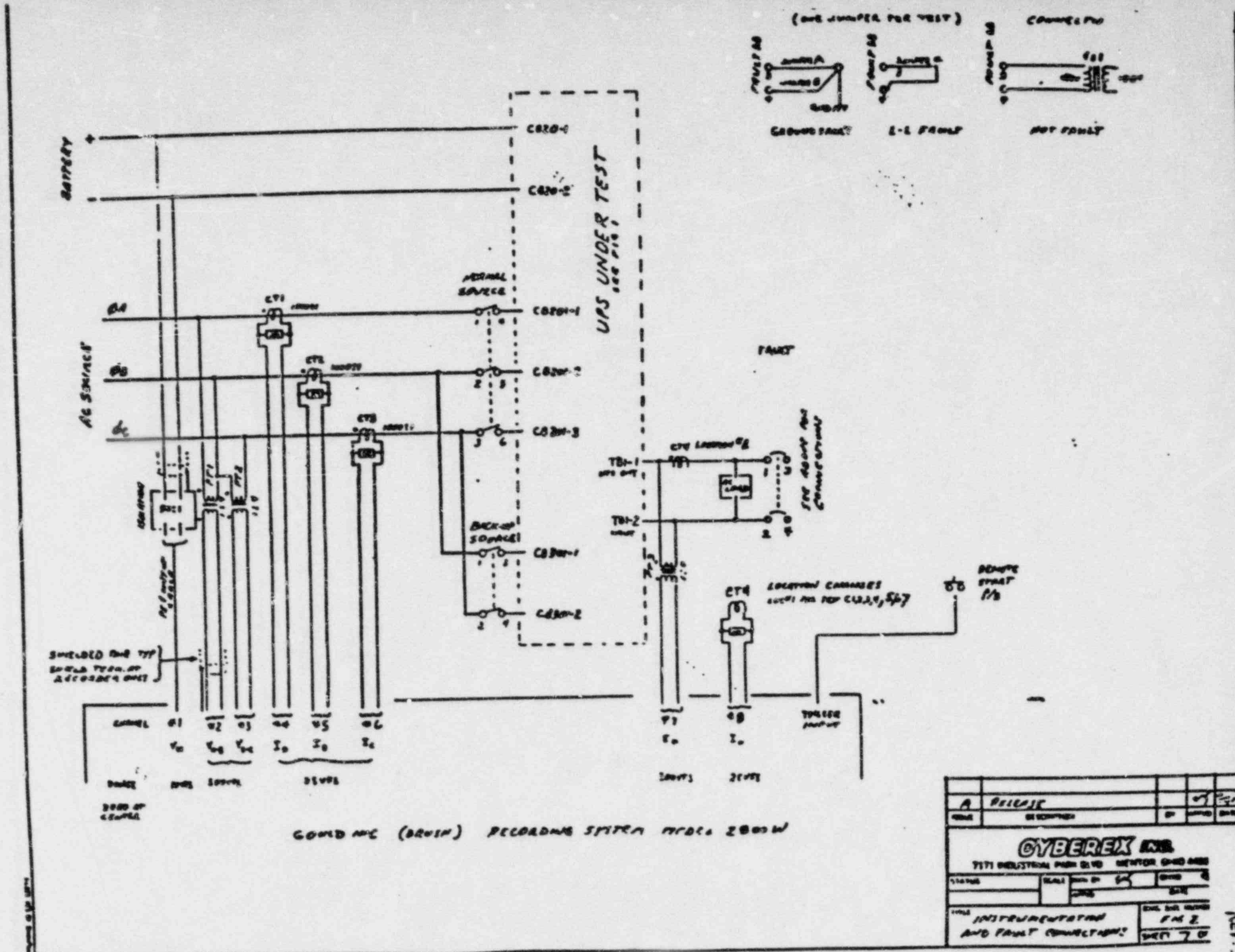
STATUS | SCALE | DWN BY | APPVD | CHKD. | DATE

TITLE: TEST SETUP ONE LINE DIAGRAM

DWG SIZE NUMBER: FIG. 1

SHEET 6 OF 8

REV. 3


UPS UNDER TEST
BATTERY
AC SOURCE
NORMAL SOURCE
BACK-UP SOURCE
CB20-1
CB20-2
CB201-1
CB201-2
CB201-3
CB301-1
CB301-2
CT1
CT2
CT3
CT4
TB1-1
TB1-2
FAULT
GROUND FAULT
L-L FAULT
HOT FAULT
REMOTE START
LOCATION CHANGES
TRIGGER INPUT
CHANNEL
SHIELDED PAIR TYP
GOULD INC (BRUSH) RECORDING SYSTEM MODEL 2800W
A RELEASE
CYBEREX INC.
INSTRUMENTATION AND FAULT CONNECTIONS

REV. 3

Rev. 5

TEST SUMMARY

TEST # ____________ CHART # ____________ CHART SPEED ____________

BY ________ DATE ________ APP'D BY ________ DATE ________

TEST DESCRIPTION:

CHAN #	CHART SCALE UNITS/mm	CHANGE DURING TEST		REMARKS
1				
2				
3				
4				
5				
6				
7				
8				

CHART TIME BASE ____________

DAMAGED PARTS:

UPS BREAKER TRIPPED DURING TEST: ____________

UPS FUSE CLEARED DURING TEST: ____________

REMARKS:

ISSUE	DESCRIPTION	BY	APP'D	DATE

CYBEREX INC.
7171 INDUSTRIAL PARK BLVD. MENTOR, OHIO 44060

TITLE

STATUS | SCALE | DWG BY | CHKD | APP'D | DATE

DWG. SIZE NUMBER

SHEET 8 OF 8

Rev. 3

FAULT LOCATION AND TYPE

FAULTS WILL BE APPLIED TO ISOLATING TRANSFORMER OUTPUT TERMINALS BY CLOSING A SWITCH AS REQUIRED.

FAULT TYPES:

1. PHASE (HOT) TO GROUND
2. NEUTRAL TO GROUND
3. PHASE TO NEUTRAL W/O GROUND
4. 480VAC APPLIED ACROSS UPS OUTPUT W/O GROUND (HOT SHORT)

THE CONDITION OF THE THREE CLASS IE SOURCES WILL BE MONITORED THROUGH SUITABLE SIGNAL CONDITIONERS, BY GOULD INC., 2000W SERIES HIGH FREQUENCY RECORDING SYSTEM.

TEST PROCEDURES

1.0 GENERAL NOTES

1.1 BEFORE STARTING TEST DETERMINE AND RECORD ALL SIGNAL CONDITIONER TRANSFER RATION (MULTIPLIER) VALUES.

1.2 NORMAL SYSTEM OPERATION DURING EACH TEST

A. CONNECTION PER FIG. 1.
B. THE LOAD ON THE UPS SHALL BE ADJUSTED FOR EACH OF THREE SEPARATE TESTS FOR EACH UPS INPUT SOURCE:
 (1) NO LOAD
 (2) OUTPUT LOAD AT .08 PF TO ACHIEVE 56 AMPERES INPUT CURRENT WHEN FED FROM 125 VOLT DC. LOAD SHOULD REMAIN THE SAME FOR AC INPUTS
 (3) OUTPUT LOAD AT .08 PF TO ACHIEVE 158 AMPERES INPUT CURRENT WHEN FED FROM 125 VOLT DC. LOAD SHOULD REMAIN THE SAME FOR AC INPUTS
C. UPS POWERED BY "ALTERNATE" DC SOURCE (BATTERY) AND ONE OR BOTH AC SOURCES, "NORMAL" & "BACK-UP"
D. STATIC SWITCH IN "PREFERRED" POSITION.
E. ALL BREAKERS & SWITCHES CLOSED, BOTH BYPASS SWITCHES IN "NORMAL" POSITION
 "TEST" SWITCH - CENTERED
 "RETURN MODE" SWITCH - IN "AUTO" POSITION
 "ISOLATION" TOGGLE SWITCHES - ON
 "SYNC" TOGGLE SWITCH - ON

1.3 TEST INSTRUMENTATION

A. GOULD INC., MODEL 2800W HIGH FREQUENCY RECORDING SYSTEM. EIGHT CHANNEL, INDEPENDENT SCALE SELECT ±.050 TO ±500 VOLTS FULL SCALE.
B. POTENTIAL TRANSFORMER 480V, 60HZ PRIMARY 120V SECONDARY (4:1 RATIO).
C. CURRENT TRANSFORMER 1000:1 RATIO WITH 10 OHM BURDEN RESISTOR. (.01V/A).
D. WIDEBAND DC ISOLATION AMPLIFIER, GOULD INC. MODEL 13-4615-10 OR EQUIVALENT.

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A. DC SUPPLY - C&D 4LCW-15 BATTERY (60 CELLS, 80KW FOR 30 MIN.) AND BATTERY CHARGER.
B. AC SUPPLY - 480V, 3 PHASE, 4W, 60 HZ, 1200A GROUNDED NEUTRAL.
C. AC LOAD BANK - 0-30KW OR 0-30KVA @ 0.8PF.
D. FAULT APPLICATION DEVICE - G.E. CIRCUIT BREAKER TJC 36400G 400A, 3P. MAGNETIC ONLY.
E. HOT FAULT SOURCE - TRANSFORMER, 1 PH 480:120V 30KVA OR LARGER.

2.0 TEST PROCEDURE

2.1 BASE LINE DATA

START UP THE UPS WITH ALL SOURCES AVAILABLE. SET UP "NORMAL OPERATION" PER 1.2 AND ALLOW SYSTEM TO WARM UP FOR AT LEAST 30 MINUTES.

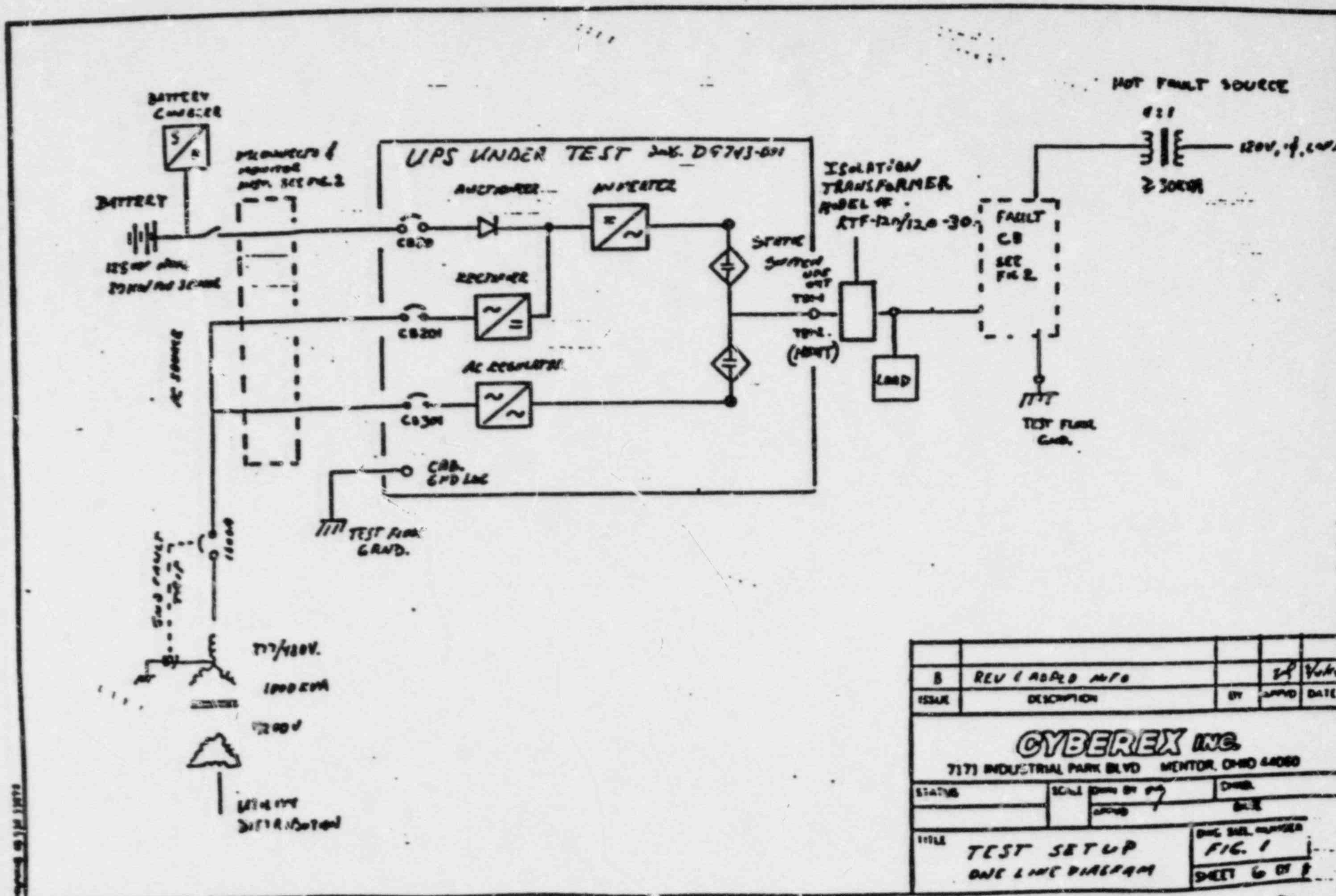
A1. METERING AND CONNECTIONS PER FIG. 2 AND "BACKUP SOURCE" BREAKER OPEN. RECORD IN "STORE" MODE AT 20KHZ TIME BASE. COPY MEMORY TO PAPER.
A2. REPEAT A1 EXCEPT USE 500HZ TIME BASE.
B1. WITH METERING AND CONNECTIONS PER FIG. 2 AND "NORMAL SOURCE" BREAKER OPEN. RECORD IN "STORE" MODE AT 20KHZ TIME BASE. COPY MEMORY TO PAPER.
B2. REPEAT B1 EXCEPT STATIC SWITCH TRANSFERRED TO BACKUP.
B3. REPEAT B1 EXCEPT USE 500HZ TIME BASE.
B4. REPEAT B2 EXCEPT USE 500HZ TIME BASE.

2.2 FAULT TESTING

C0. METERING AND CONNECTIONS PER FIG 2, RECORDER IN MANUAL TRIGGER MODE. APPLY FAULT BY CLOSING "FAULT" CB AND AT THE SAME TIME (OR 0 TO 10 MILLISECONDS BEFORE) TRIGGER THE RECORDER IN "STORE" MODE. REMOVE THE FAULT AND RECORD THE MEMORY TO PAPER.
AFTER EACH FAULT APPLICATION CHECK THE UPS FOR DAMAGE. REPAIR THE UPS IF REQUIRED BEFORE PROCEEDING.

C1. INSTALL JUMPER "A" TO "FAULT" CB WITH "BACKUP SOURCE" CB OPEN WITH RECORDER AT 20KHZ TIME BASE APPLY FAULT PER C0.
C2. REPEAT C1 EXCEPT WITH 500HZ TIME BASE.
C3. OPEN "NORMAL SOURCE" CB AND CLOSE "BACKUP" WITH RECORDER 20KHZ TIME BASE APPLY FAULT PER C0.
C4. REPEAT C3 EXCEPT WITH 500HZ TIME BASE.
C5. REPEAT C1, C2, C3 & C4 WITH JUMPER "B" INSTEAD OF "A" CONNECTED TO "FAULT" CB.
C6. REPEAT C1, C2, C3, & C4 WITH JUMPER "C" INSTEAD OF "A" CONNECTED TO "FAULT" CB.
C7. REPEAT C1, C2, C3, & C4 WITH CONNECTIONS TO HOT FAULT SOURCE (UPS RUNNING AT NO LOAD).

2.3 COMPLETE TEST SUMMARY SHEET FOR EACH TEST OR TEST GROUP.



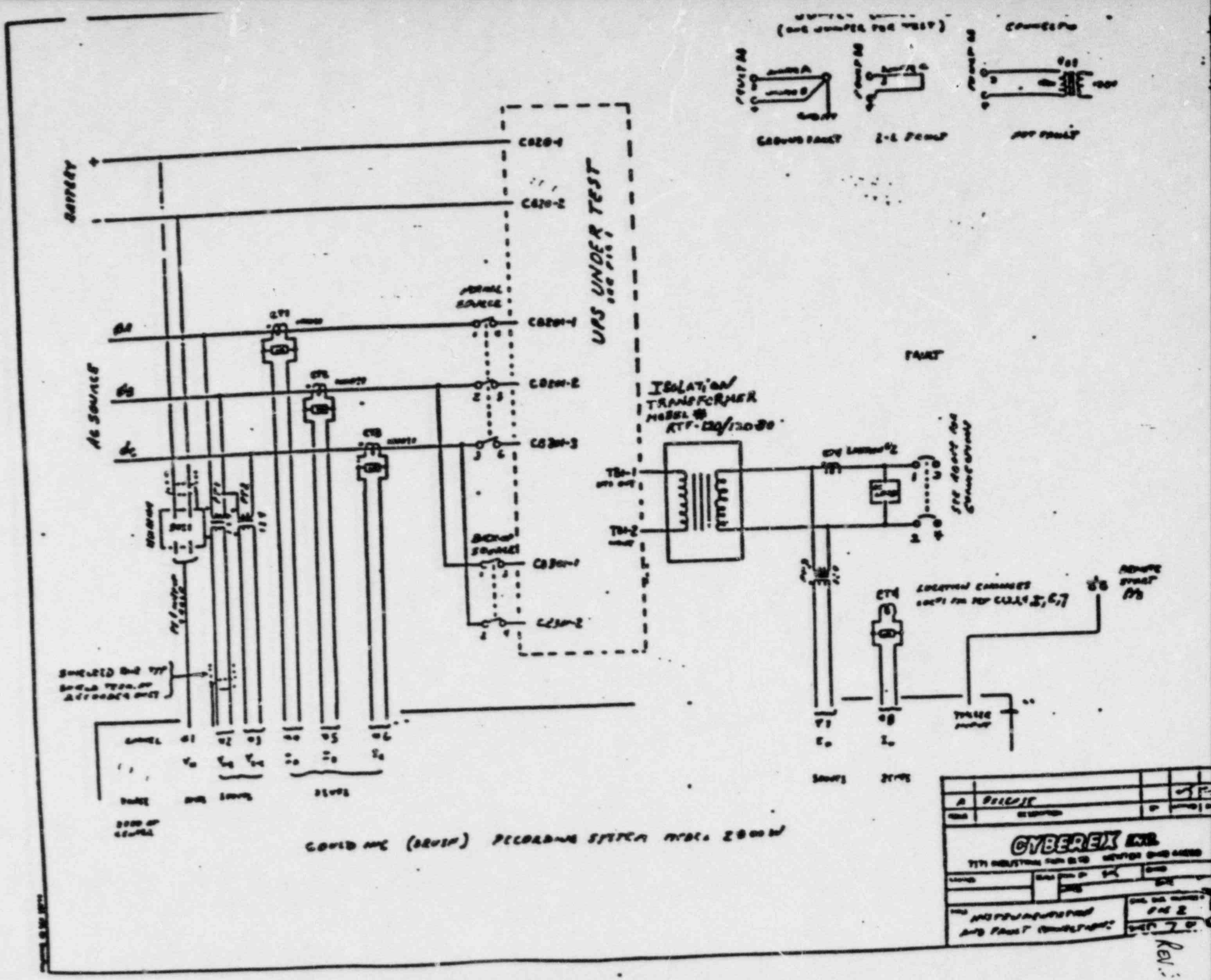
ISSUE	DESCRIPTION	BY	APPVD	DATE
B	REV & ADDED INFO			

CYBEREX INC.
7171 INDUSTRIAL PARK BLVD MENTOR, OHIO 44060

TITLE: TEST SETUP ONE LINE DIAGRAM

DWG. SIZE. NUMBER: FIG. 1

SHEET 6 OF 8



UPS UNDER TEST
BATTERY
AC SOURCE
ISOLATION TRANSFORMER
CB20-1
CB20-2
CB201-1
CB201-2
CB201-3
NORMAL SOURCE
BACKUP SOURCE
GROUND FAULT
L-L FAULT
FAULT
GOULD INC (BRUSH) RECORDING SYSTEM MODEL 2800W
CYBEREX INC.
INSTRUMENTATION AND FAULT CONNECTIONS
REV.

Rev. 3

[illegible]

TEST SUMMARY

TEST # ______________ CHART # ______________ CHART SPEED ______________

BY __________ DATE __________ APR'D BY __________ DATE __________

TEST DESCRIPTION:

CHAN #	CHART SCALE UNITS/mm	CHANGE DURING TEST		REMARKS
1				
2				
3				
4				
5				
6				
7				
8				

CHART TIME BASE ______________

DAMAGED PARTS:

UPS BREAKER TRIPPED DURING TEST: ______________

UPS FUSE CLEARED DURING TEST: ______________

REMARKS:

ISSUE	DESCRIPTION	BY	APPVD	DATE

CYBEREX INC.
7171 INDUSTRIAL PARK BLVD. MENTOR OHIO 44060

TITLE

STATUS | SCALE | DWN BY | CHKD | APPVD | DATE

DWG SIZE NUMBER

SHEET 8 OF 8

Attachment 5

Revised Text FSAR Sections:	13.2.1
	13.2.1.1
	13.2.1.1.1
	13.2.1.1.1.1
	13.2.1.1.1.2
	13.2.1.1.1.3
	13.2.1.1.1.4
	13.2.1.1.2
	13.2.2
	Appendix 13C
	Appendix 13F
	Appendix 13I
	Appendix 13J
	Appendix 13K (new)
	630.4
	630.7
	630.10

13.2 TRAINING

13.2.1 PLANT PERSONNEL TRAINING PROGRAM

The training program for Hope Creek Generating Station (HCGS) is formulated to develop and maintain an organization qualified to assume the responsibility for preoperational testing, operation, maintenance, and technical considerations for the facility.* To accomplish these objectives and to provide the necessary control of the overall plant, the following three general training programs will be implemented:

Insert 1

a. Initial Plant Staff Training Programs - These programs are designed to provide competent, trained personnel in all disciplines and at all levels of plant organization. The programs are designed to allow personnel to be placed at various points, according to their training, experience and intended position. The training procedures are detailed in the Nuclear Department Training Manual.

b. Requalification Training Program - A requalification program as required by 10 CFR 50.54 (i-1) will be developed to provide continuous training and upgrading of plant personnel and will meet the requirements of 10 CFR 55, Appendix A and NUREG 0737 Enclosure 1. Use will be made of the Hope Creek specific simulator scheduled to be delivered to the facility in the summer of 1984. Therefore, a specific requalification program will not be available until late 1984. Upon formal acceptance of the Hope Creek specific simulator and establishment of operator shift rotation, the licensed operator requalification program will be implemented to ensure that all cold license candidates maintain a high level of knowledge and operator confidence. The requalification program will run on an annual basis with all program requirements completed during the two year requalification cycle. The requalification program will consist of three areas; pre-planned lectures, on-the-job training and requalification examinations.

The pre-planned lectures will cover fundamental review and operational proficiency. Fundamental review training will be in those areas of heat transfer, fluid flow, thermodynamics, mitigation of accidents involving a degraded core and these subject areas delineated in 10CFR55, Appendix A. Operational proficiency training

Insert (1)

Achievement of this goal is based on a philosophy of providing training developed from a systematic analysis of job requirements ~~using~~ and using job and task analysis where available. This philosophy is consistant with both Nuclear Regulatory Commission requirements and Institute of Nuclear Power Operations (INPO) recommendations necessary for accreditation of training programs. The timetable for achieving accreditation shall be consistant with INPO recommendations.

CH 455

will involve lectures that will focus on essential plant operational guidelines and changes or experiences in the nuclear industry.

The on-the-job training will ensure that each licensed operator maintains an acceptable level of skills and familiarity associated with plant systems, controls and operational procedures. This will be accomplished through reactivity manipulations, plant evolutions and operational reviews.

Requalification examinations will be given to determine the licensed operator's knowledge of the material covered, areas where additional training may be required and operational proficiency. These examinations will consist of a segmented written examination and an oral examination.

Personnel demonstrating a significant deficiency in a given area of knowledge and proficiency may be placed into an accelerated training program. This program will be specifically structured to upgrade knowledge and skills identified as deficiencies. Successful completion of the accelerated training program will be evaluated by a written and/or oral examination. Procedures describing the content and conduct of the requalification program will be developed and will be maintained in the Nuclear Department Training Procedure Manual.

c. Replacement training - These programs are designed to provide qualified personnel for the station organization. The General Manager - Hope Creek Operations, or the designated representative, may waive portions of the training program for individuals based on their previous experience and/or qualifications. The training procedures are detailed in the Nuclear Department Training Manual.

The Manager - Nuclear Training is responsible for implementation of this program. Prior to implementation, each course, its scheduled starting date, and its duration shall be approved by the General Manager - Hope Creek Operations.

The Manager - Nuclear Training will ensure that all individuals providing instruction are technically qualified to present the material and that they have demonstrated a knowledge of instructional techniques as required by ANS/ANSI 3.1-1981, 4.4.7.2. Individuals providing instruction to license operator candidates will have received all appropriate training and hold or have held an SRO license or certification as required by the H.R. Denton letter of March 28, 1980, Enclosure 1, and ANS/ANSI 3.1-1981, 4.4.7.2. These individuals will take an active part in the license operator shift ~~cycle~~ training program. Upon completion of the cold license training program and establishment of the operator requalification program, individuals providing specific license training outlined in ANS/ANSI 3.1-1981, 4.4.7.2.c will participate in the requalification program as specified in ANS/ANSI 3.1-1981, 5.5.1.5.

Figure 13.2-1 shows the present schedule for the various initial plant training program. If significant differences or changes occur in those courses not yet conducted, the appropriate course outlines and descriptions will be revised by Amendment.

13.2.1.1 Operating Department Training Programs

These programs are designed for individuals who will assume the responsibility for both licensed and nonlicensed plant operating functions, as outlined in job specifications.

The program is divided into the following areas ~~basic segments~~:

a. Non-licensed Operator training ~~Nuclear Reactor Fundamentals~~

b. Reactor Operator training ~~Reactor Startup Experience~~

c. Senior Reactor Operator training

d. Shift Technical Advisor ~~(Advanced Technical)~~ Training

e. Licensed Operator Requalification training

~~d. Pre-Certification Systems Training~~

~~e. BWR Cold Certification Training~~

~~f. Shift Supervisor Nuclear Training~~

~~g. Hope Creek Systems Training~~

h. Equipment Operator Training

i. Cold License Operator In-Plant Training

j. Pre-NRC License Exam Testing & Training

To assure the experience criteria of ANS 3.1 (1981) is met, as well as the general guidelines of NUREG-0094, additional experience will be provided by a structured ~~PWR/BWR~~ observation program for all licensed operator candidates. A detailed description of this observation training is shown in appendix K.

13.2.1.1.1 Cold License Training Program

This program is designed for NRC reactor operator (RO) and senior reactor operator (SRO) cold license candidates of varying backgrounds and experience. Candidates will be factored into the program at various points, depending on their previous experience and training. Testing and screening will be an intimate part of the overall training program. All license candidates who are supervisors will attend the PSE&G Supervisory Skills Training Program and will meet the supervisory training requirements of ANSI/ANS-3.1-1981, Section 5.2.1.8 prior to core load.

~~It is the intended of this training program that all SRO candidates have at least thirty (30) semester hours of equivalent college level education.~~

13.2.1.1.1.1 Senior Reactor Operator Training Program

The senior reactor operator (SRO) candidates will attend a training program consisting of, but not limited to, the following areas of instruction:

a. Nuclear Reactor Fundamentals

b. Reactor Startup Experience

c. Advanced technical training

d. Pre-Certification system training

e. BWR Cold certification training

f. ~~Shift Supervisor Nuclear training~~
In-Plant Training

g. Hope Creek Systems training
h. Prelicense Examination testing and training

Detailed course descriptions and outlines are shown in Appendices 13A, 13B, 13C, 13D, 13E, 13F, ~~and~~ 13G, 13I and 13J.

insert (A) →

It is the intended that through ~~of~~ this training program ~~that~~ all SRO candidates will obtain ~~have~~ at least thirty (30) semester hours of equivalent college level education.

~~Manual~~.

insert (B) →

Following the Hope Creek systems training the SRO candidates will be assigned to a shift where they will participate in the cold license operator in-plant training program described in Appendix 13I.

13.2.1.1.1.2 Reactor Operator (RO) Training Program

The RO candidates will attend a training program consisting of, but not limited to, the following:

a. Nuclear Reactor Fundamentals

b. Reactor Startup Experience

c. Pre-Certification system training

d. BWR Cold Certification training

e. Hope Creek system training.
f. In-Plant training
g. Pre-license examination testing and training

Detailed course descriptions and outlines are shown in Appendices 13A, 13B, 13D, 13E, ~~and~~ 13G, 13I and 13J.

insert (B) →

(A)

The Advanced Technical training ~~program~~ will consist of two separate programs, Advanced Technical Training as outlined in appendix 13C and SS-N Training as outlined in appendix 13F. The advanced Technical training program ~~it is~~ was designed for those individuals who are to be senior shift supervisors or shift technical advisors. The SS-N training program was ~~is~~ designed for those individuals who are to be shift supervisors

(B) With the exception of Hope Creek Systems Training the General Manager - Hope Creek may waive any of these programs as recommended by the Manager - Nuclear Training for selected individuals based on previous experience, training and/or licensing.

Previously licensed PWR operators who do not attend a simulator certification program, shall attend a BWR operational review training program at an appropriate BWR simulator or the Hope Creek simulator when it becomes operational.

~~Procedures describing the conduct of these programs are under~~ entered ~~in the Nuclear Department Training Manual.~~

5 Following the Hope Creek systems training the RO candidates will be assigned to a shift where they will ~~complete~~ participate in
8 the cold license operator in-plant training program described in Appendix 13I.

13.2.1.1.1.3 Shift Technical Advisor Training

Shift technical advisor (STA) training will meet the requirements outlined in ANSI/ANS-3.1-1981. Training programs will consist of those areas where their prior education did not meet those requirements and will include plant specific thermodynamics, fluid flow, reactor physics, system engineering, transient and accident analysis, nuclear instrumentation, process computer, plant response, and duties and responsibilities.

The STA training program will consist of, but is not limited to, the following areas of instruction:

a. Nuclear reactor fundamentals

b. Reactor startup experience

c. Advanced technical training

d. Pre-certification system training

e. BWR cold certification training

f. Hope Creek systems training

g. In-Plant training

Detailed course descriptions and outlines are shown in Appendices 13A, 13B, 13C, 13D, 13E, ~~and~~ 13G, and 13I.

Insert (B) →

The reactor startup experience and BWR cold certification training may be waived for those individuals who are previously licensed. They will however attend a BWR operational review training program at an appropriate BWR simulator or the HCGS specific simulator when it becomes available.

All STA candidates will be assigned to HCGS staff where they will participate in the cold license operator in-plant training program as described in Appendix 13I. STA candidates will continue to attend training with the SRO candidates. It is not intended at this time to test in lieu of training as stated in ANS/ANS 3.1 1981, 5.2.1.7.

Procedures describing the conduct and grading criteria of this program are under development and will be entered into the nuclear department training procedure manual.

13.2.1.1.1.4 BWR Prelicense Refresher Training

Because of the long lead time required for cold license training, a Prelicensing Refresher Course will be conducted. This course will be approximately 13 weeks in duration and will be scheduled to end about 3 to 6 months prior to initial fuel loading. An NRC-type audit examination will be given at the end of the during the refresher training. Further training will be conducted in areas identified by the audit examination. Appendix 13J provides a detailed description of this program.

13.2.1.1.2 Nonlicensed Operator Training Program

This program is designed to make equipment operators knowledgeable of HCGS systems, operations, and procedures. The program will cover, but is not limited to, the following material:

a. Mathematics Refresher

b. Physics and Basic Heat Transfer and Fluid Flow (HTFF) Refresher

c. Basic Power Plant Equipment (valves, pumps, etc,), Lubrication, and Job Duties

d. NSSS

e. Electrical Systems

f. Auxiliary Systems

g. Health Physics

h. Firefighting

i. Heating Boiler

j. Procedures (as applicable)

k. Administrative Functions, Equipment Tagging, and Log Keeping

l. Technical Specifications (as applicable).

It is anticipated that the classroom program, Appendix 13H, will last 12 to 14 weeks and will be followed by a period of in-plant training where the equipment operators will complete required checklists. ~~Procedures describing the conduct of these programs are located in the Nuclear Department Training Manual.~~ w/fg

13.2.1.1.3 Maintenance Department Training Program |

Maintenance supervisors, electricians, machinists, and boiler repair personnel will generally be selected from other operating PSE&G facilities (fossil and nuclear) or be direct hire, journeyman level qualified. As such, they will already have received training appropriate for their particular skill area. Through their previous experience and selection/testing procedures these personnel will exhibit a high degree of manual dexterity and the capability to learn and apply basic job skills in performing maintenance activities.

Maintenance personnel will receive on-the-job training during the preoperational test program by performing maintenance activities. Selected personnel will receive specialized vendor training on specific equipment or skills. Personnel promoted to the journeyman or supervisory level will be required to satisfactorily complete the PSE&G Advanced or Supervisory Training Program associated with their particular skill area.

Additional training for experienced personnel will include a BWR Technology Course, appropriate quality assurance training, training on plant specific maintenance procedures, and radiation worker and general employee training, as well as other programs deemed necessary. Procedures for these training programs will be available in the Nuclear Department Training Manual.

Personnel below the supervisory and journeyman level, as a minimum, will complete the various required apprentice level training programs as their career progresses. These programs will also be detailed in the Nuclear Department Training Manual.

Training will be conducted by PSE&G and qualified vendor personnel.

13.2.1.1.4 Technical Department Training Program |

The objective of the Technical Department Training Program is to provide highly skilled personnel to effectively support the preoperational testing program and plant power operations.

Procedures for these training programs will be available in the Nuclear Department Training Manual.

13.2.1.1.4.1 Chemistry Section Training

Supervisor and technician level personnel will be selected only after meeting applicable experience requirements. As such, they will generally have completed the appropriate training program associated with their respective job position. Procedure for conducting these programs will be available in the Nuclear Department Training Manual. Experienced personnel who fit that description will, as a minimum, undergo training in the following general subject areas:

a. BWR Technology

b. Chemistry Practices and Procedures

c. Chemistry Equipment and Use

d. Applicable Administrative Procedures

e. Special Courses presented by the Nuclear Training Center and/or vendors, as appropriate.

f. QA Program

g. General Employee and Radiation Worker Training.

Personnel promoted to the supervisory or technician level will be required to complete the PSE&G Chemistry Technician Advanced Course or Nuclear Supervisor Course, as appropriate to the respective job position.

Personnel below the supervisory and technician level, as a minimum, will complete the various required apprentice level training programs as their career progresses.

Personnel below the supervisory and technician level, as a minimum, will complete the various required apprentice level training programs as their career progresses.

I&C personnel will receive on-the-job training during the preoperational testing program by performing their job associated tasks in support of that testing.

Training will be conducted by qualified PSE&G and vendor personnel.

13.2.1.1.4.3 Reactor Engineering Training Program

Prior to core load, selected reactor engineering personnel will have attended a vendor-offered course typically entitled "Station Nuclear Engineer". Typical subject matter will include reactor behavior, control rods, shutdown margins, technical specifications and Fuel Warranty Operation Provisions, core flow and thermal limit calculations, fuel failure and Preconditioning Interim Operating Management Recommendation and water chemistry.

13.2.1.1.5 Radiation Protection Department Training Program

Supervisory and technician level personnel will be selected only after meeting applicable experience requirements. As such, they will generally have completed the appropriate training program associated with their respective job position. Procedures for conducting these programs will be available in the Nuclear Department Training Manual. Experienced personnel who fit that description will, as necessary, undergo training in the following general subject areas:

a. BWR Technology

b. Radiation Protection Practices and Procedures

c. Radiation Protection Equipment and Use

d. Applicable Administrative Procedures

Chemistry personnel will receive on-the-job training during the preoperational testing program by performing their job associated tasks in support of that testing.

Training will be conducted by qualified PSE&G and vendor personnel.

13.2.1.1.4.2 Instrumentation and Controls Section Training

Supervisory and technician level personnel will be selected only after meeting applicable experience requirements. As such, they will generally have completed the appropriate training program associated with their respective job position. Procedure for conducting these programs will be available in the Nuclear Department Training Manual. Experienced personnel who fit that description will, as a minimum, undergo training in the following general subject areas:

a. BWR Technology

b. Instrumentation and Controls Practices and Procedures

c. Instrumentation and Controls Equipment

d. Applicable Administrative Procedures

e. Special Courses presented by the Nuclear Training Center and/or vendors, as appropriate.

f. QA Program

g. General Employee and Radiation Worker Training.

Personnel promoted to the supervisory or technician level will be required to complete the PSE&G Instrumentation and Controls (I&C) Technician Advanced Course or Nuclear Supervisor Course, as appropriate to the respective job position.

Personnel below the supervisory and technician level, as a minimum, will complete the various required apprentice level training programs as their career progresses.

I&C personnel will receive on-the-job training during the preoperational testing program by performing their job associated tasks in support of that testing.

Training will be conducted by qualified PSE&G and vendor personnel.

13.2.1.1.4.3 Reactor Engineering Training Program

Prior to core load, selected reactor engineering personnel will have attended a vendor-offered course typically entitled "Station Nuclear Engineer". Typical subject matter will include reactor behavior, control rods, shutdown margins, technical specifications and Fuel Warranty Operation Provisions, core flow and thermal limit calculations, fuel failure and Preconditioning Interim Operating Management Recommendation and water chemistry.

13.2.1.1.5 Radiation Protection Department Training Program

Supervisory and technician level personnel will be selected only after meeting applicable experience requirements. As such, they will generally have completed the appropriate training program associated with their respective job position. Procedures for conducting these programs will be available in the Nuclear Department Training Manual. Experienced personnel who fit that description will, as necessary, undergo training in the following general subject areas:

a. BWR Technology

b. Radiation Protection Practices and Procedures

c. Radiation Protection Equipment and Use

d. Applicable Administrative Procedures

e. Special Courses presented by the Nuclear Training Center and/or vendors, as appropriate

f. QA Program

g. General Employee and Radiation Worker Training.

Personnel promoted to the supervisory or technician level will be required to complete the PSE&G Radiation Protection Technician Course or Nuclear Supervisor Course, as appropriate to the respective job position.

Personnel below the supervisory and technician level, as a minimum, will complete the various, programs as their career progresses.

Radiation Protection personnel will receive on-the-job training during the pre-operational testing program by performing their job associated tasks in support of that testing.

Training will be conducted by qualified PSE&G and vendor personnel.

13.2.1.1.6 General Employee Indoctrination

All persons regularly employed at HCGS, including temporary maintenance and service personnel, who are permitted unescorted access shall be given General Employee Indoctrination. This training covers the following areas:

a. Site Description

b. Emergency Plan

c. Security System

d. Quality Assurance Program

e. Radiological Health.

Personnel will be tested in the above areas to determine the effectiveness of General Employee Indoctrination.

Personnel who will routinely work in radiation and/or contaminated areas will also complete a Radiation Worker Training Program of approximately 12 hours.

13.2.1.2 <u>Refresher Training for Nonlicensed Plant Personnel</u>

A retraining program will be provided for all personnel to ensure that they remain proficient in their particular jobs.

Retraining in specific areas is provided to the extent necessary for personnel to safely and efficiently carry out their assigned responsibilities in accordance with established policies and procedures. This includes operating experiences, design changes, revisions to procedures, and new procedure indoctrination.

Such training may consist of vendor presentations, technical training sessions, on-the-job work experience or programmed instruction. Personnel are evaluated on an annual basis where individual needs for retraining will be identified.

13.2.1.3 <u>General Employee Indoctrination Requalification</u>

All persons regularly employed at HCGS, including temporary maintenance and service personnel who are permitted unescorted access, shall requalify in General Employee Indoctrination annually. This is accomplished by attending the requalification class and obtaining a satisfactory score on an examination covering the areas mentioned in Section 13.2.1.1.6.

Personnel trained in the Radiation Worker Training Program will requalify annually by attending the Radiation Worker Review Program of approximately 4 hours. Satisfactory completion of that program also meets General Employee Indoctrination Requalification requirements.

13.2.1.4 Replacement Training for Nonlicensed Plant Personnel

Replacement training is designed to supply qualified personnel at all levels and job positions within the plant organization. Training is carried on at all job levels to qualify that particular individual to effectively perform the required job functions. Qualified personnel who are promoted to the next job level are placed, as rapidly as possible, into the appropriate training program. It is the general policy of PSE&G to promote from within. In this manner, as an individual progresses, he/she is immediately trained for the new position and capable of supporting and training personnel in the lower classifications.

Personnel who are directly hired into job positions above the entry level will meet or exceed the applicable requirements of that position. Training programs will be developed for these personnel to familiarize them with appropriate HCGS-specific material.

Training will be conducted by qualified PSE&G and vendor personnel. The training programs will be described in the Nuclear Department Training Manual.

13.2.1.5 Replacement Training for NRC Licensed Plant Personnel

Training for NRC licensed replacement personnel will, as a minimum, meet the existing NRC requirements as outlined in 10 CFR 55.21, .22, .23, appropriate NUREGs, and the H. Denton letter of March 28, 1980 and all applicable training requirements of ANS/ANSI 3.1-1981. These programs are described in the Nuclear Department Training Manual and are revised as regulations and job requirements change.

13.2.2 FIRE BRIGADE TRAINING PROGRAM

Insert 2 →

~~Fire protection training will be conducted in accordance with the guidelines of the SRP (NUREG 0800) Section 13.2.2.II.6, 10CFR50 Appendix R and Branch Technical Position CMEB 9.5.1 Section C.3.d. This training will include classroom instruction, hands-on fire extinguishing and plant drills.~~

The classroom instruction will include the following course material:

a. Firefighting Plan

1. Response to alarms

2. Responsibility of members

3. Reason for fire brigade

b. Identification of Fire Hazards

1. Concept of fire

2. Properties of flammable and combustible liquids

3. Hazardous chemical properties

4. Boiling liquid, expanding vapor explosion

c. Products of Combustion

1. Products of burning plastics

2. Products of smoke

3. Properties of carbon monoxide

4. Properties of contaminated smoke

5. Effects of heat

6. Ventilation

d. Firefighting Equipment

1. Fire detection

2. Fire suppression

e. Types of Fires

f. Auxiliary Equipment

g. Plant Modifications

Actual hands-on fire extinguishing will be conducted to provide brigade members with actual fire extinguishing and the use of emergency breathing apparatus under strenuous conditions. These practice sessions will be held at least once per year for each fire brigade member.

Plant drills will be held at specified intervals not to exceed 3 months for each shift to allow fire brigade members the opportunity to practice as a team and to ensure adequate procedures and readiness.

Each drill will be preplanned to establish training objectives and will be critiqued to determine how well the training objectives have been met. Performance deficiencies noted will be remedied by additional training.

Fire drills as a minimum will assess the fire alarms effectiveness, time to assemble the fire brigade, use of the firefighting equipment, firefighting strategies and the effectiveness of the brigade leader.

The Fire Brigade Training program is designed to ensure that the employees assigned to the fire brigade are capable of providing adequate manual firefighting strategies to control fires that might occur at the Hope Creek Generating Station. The program will cover, but is not limited to the following:

a. Indoctrination of the plant firefighting plan.

b. Identification of fire hazards.

c. The properties of the products of combustion.

d. Identification and use of all firefighting equipment.

e. The proper use of communication, lighting, ventilation, and emergency breathing equipment.

f. The proper method for fighting fires inside buildings and confined spaces.

g. The direction and coordination of the firefighting activities. (Fire Brigade leaders only).

h. Detailed review of firefighting strategies and procedures.

i. Review of the latest plant modifications and corresponding changes in firefighting plans.

Procedures describing course content, grading criteria and recordkeeping are under development. These procedures are scheduled to be completed by January 1985.

Insert 2

Replaces existing FSAR Section 13.2.2 in its entirety.

13.2.2 FIRE BRIGADE TRAINING PROGRAM

Fire protection training will be conducted in accordance with the guidelines of the SRP (NUREG 0800) Section 13.2.2.II.6, 10CFR50, Appendix R and Branch Technical Position CMEB 9.5.1, Section C.3.d. This training will include classroom instruction, hands-on fire extinguishing and plant drills.

The Fire Brigade Training Program is designed to ensure that the employees assigned to the fire brigade are capable of providing adequate manual fire fighting strategies to control fires that might occur at the Hope Creek Generating Station. The program will cover, but is not limited to the following:

a. Indoctrination of the plant fire fighting plan.

b. Identification of fire hazards.

c. The properties of the products of combustion.

d. Identification and use of all fire fighting equipment.

e. The proper use of communication, lighting, ventilation, and emergency breathing equipment.

f. Familiarization with the layout of the plant, including access and egress routes to each area.

g. Correct method of fighting fires, including fires in energized electrical equipment, fires in cable and cable trays, hydrogen fires, fires involving flammable and combustible liquids or hazardous process chemicals, fires resulting from construction or modifications (welding) and record file fires.

h. The direction and coordination of the fire fighting activities (fire brigade leaders only).

i. Detailed review of fire fighting strategies and procedures.

j. Review of the latest plant modifications and corresponding changes in fire fighting plans.

The classroom instruction will include the following course material:

a. Fire Fighting Plan

  1. Response to alarms
  2. Responsibility of members
  3. Reason for fire brigade

b. Identification of Fire Hazards

   1. Concept of fire
   2. Properties of flammable and combustible liquids
   3. Hazardous chemical properties
   4. Boiling liquid, expanding vapor explosion

c. Products of Combustion

   1. Products of burning plastics
   2. Products of smoke
   3. Properties of carbon monoxide
   4. Properties of contaminated smoke
   5. Effects of heat
   6. Ventilation

d. Fire Fighting Equipment

   1. Fire detection
   2. Fire suppression

e. Types of Fires

f. Auxiliary Equipment

g. Plant Modifications

Actual hands-on fire extinguishing will be conducted to provide brigade members with actual fire extinguishing and the use of emergency breathing apparatus under strenuous conditions. These practice sessions will be held at least once per year for each fire brigade member.

Plant drills will be held for each shift to allow fire brigade members the opportunity to practice as a team and to ensure adequate procedures and readiness.

Each fire brigade member must participate in at least two drills per year.

Each drill will include the simulated use of fire-fighting equipment required to cope with the situation and type of fire selected for the drill. The area and type of fire chosen for the drill will differ from those used in the previous drill so that brigade members are trained in fighting fires in various plant areas. The situation selected will simulate the size and arrangement of a fire that could reasonably occur in the area selected, allowing for fire development due to the time required to respond, to obtain equipment, and organize for the fire, assuming the loss of automatic suppression capability.

At least one drill per year will be performed on a back shift for each shift fire brigade.

At least one drill for each shift fire brigade per year will be unannounced to determine the fire fighting readiness of the plant fire brigade, brigade leader, and fire protection systems and equipment. Personnel planning and authorizing an unannounced drill will ensure that the responding shift fire brigade members are not aware that a drill is being planned until it is begun. Unannounced drills will not be scheduled closer than four weeks.

Unannounced drills will be planned and critiqued by members of the management staff responsible for plant safety and fire protection. Performance deficiencies of a fire brigade or individual fire brigade members will be remedied by scheduling additional training for the brigade or members. Unsatisfactory drill performance will be followed by a repeat drill within thirty days.

At three-year intervals, a randomly selected unannounced drill will be critiqued by qualified individuals independent of the licensee's staff. A copy of the written report from such individuals shall be available for NRC review.

Regularly planned meetings will be held every three months for all members to review changes to the program.

Periodic refresher training will repeat classroom instruction over a two-year period. These sessions may be concurrent with planned meetings.

Training of the plant fire brigade will be coordinated with the local fire department so that responsibilities and duties are delineated in advance. This coordination will be part of the training course and will be included in the training of the local fire department staff.

Local fire departments will be provided training in operational precautions when fighting fires on nuclear power plant sites and will be made aware of the need for radiological protection of personnel and the special hazards associated with a nuclear power plant site.

Instruction will be provided by qualified individuals who are knowledgeable, experienced and suitably trained in fighting types of fires that could occur in the plant and using types of equipment available in nuclear power plants.

Instruction will be provided for all employees once a year. It will be repeated on an annual basis. The instruction will be given on (1) the fire protection plant, (b) the evacuation routes, and (c) the procedure for reporting a fire.

Instruction will be provided for security personnel that addresses (a) entry procedures for outside fire departments, (b) crowd control for people exiting the station, and (c) procedures for reporting potential fire hazards observed when touring the facility.

Instruction will be provided to appropriate shift personnel that complements that given to members of the fire brigade.

Instruction will be provided to temporary employees so that they are familiar with (a) evacuation signals, (b) evacuation routes, and (c) the procedure for reporting fires.

Station personnel will participate in an annual accountability and evacuation drill.

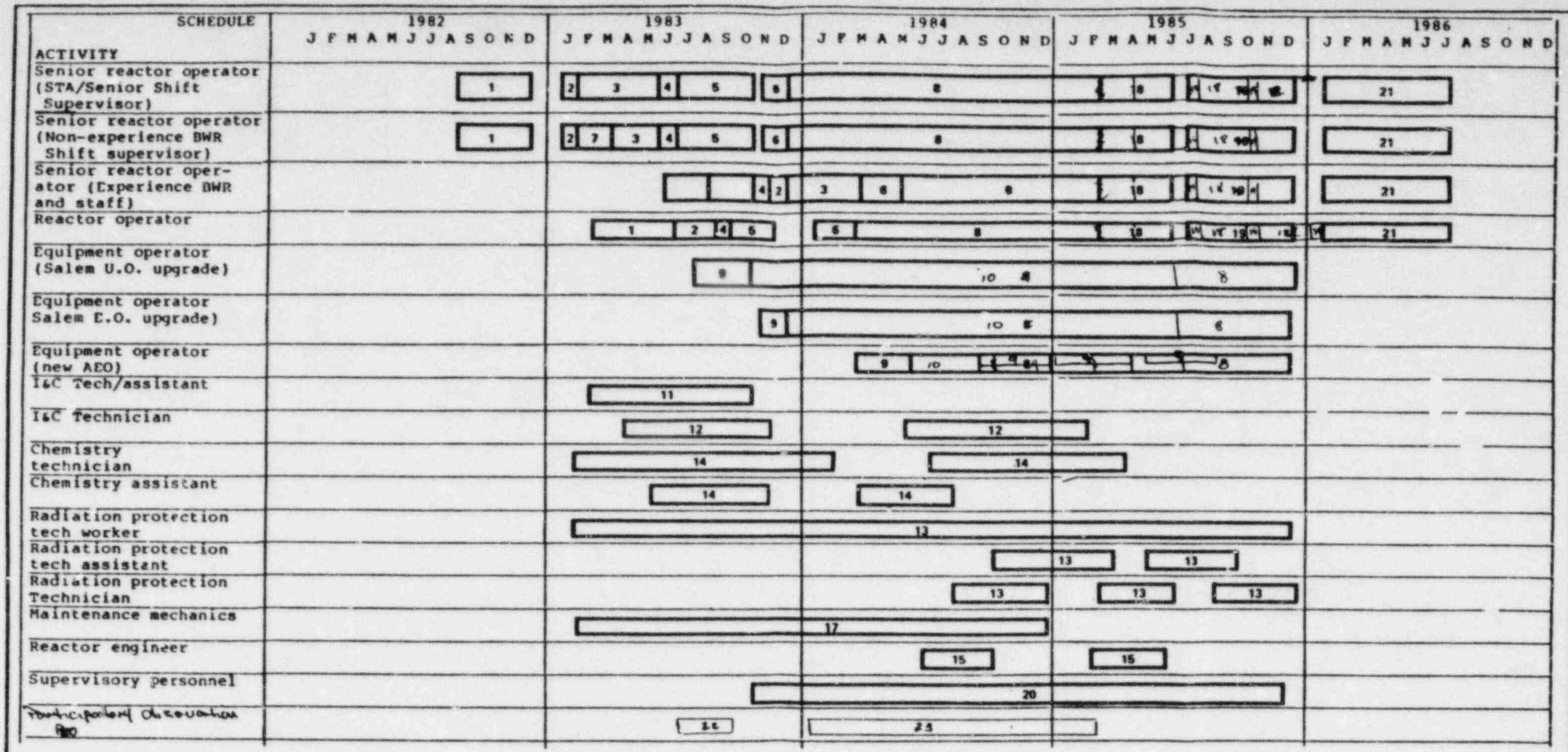
Fire Protection Staff

Training for the fire protection staff members shall include courses in:

1. Design and maintenance of fire detection, suppression and extinguishing systems.

2. Fire prevention techniques and procedures.

3. Training and manual fire-fighting techniques and procedures for plant personnel and the fire brigade.



1 – NUCLEAR FUNDAMENTALS
2 – REACTOR STARTUP TRAINING
3 – SHIFT TECHNICAL ADVISOR TRAINING – PERSONNEL
4 – BWR SYSTEMS TRAINING
5 – COLD CERTIFICATION TRAINING
6 – HOPE CREEK SYSTEM TRAINING
7 – GE NUCLEAR ENGINEERING COURSE – PERSONNEL
8 – CONTINUAL TRAINING HOPE CREEK SIMULATOR TRAINING STARTUP TESTING, PROCEDURE DEVELOPMENT
9 – EQUIPMENT OPERATOR TRAINING
10 – SYSTEM CHECKOUTS / ON-THE-JOB TRAINING
11 – INSTRUMENTATION AND CONTROLS FUNDAMENTALS, SYSTEMS AND PROCEDURE
12 – SYSTEMS AND PROCEDURES
13 – HEALTH PHYSICS FUNDAMENTALS, SYSTEMS AND PROCEDURES, INSTRUMENTS
14 – CHEMISTRY FUNDAMENTALS, INSTRUMENTS, ELECTRONICS, SYSTEMS AND PROCEDURES
15 – GE NUCLEAR ENGINEERING COURSE
16 – SIMULATOR TRAINING
17 – JOURNEYMAN MECHANICAL, ELECTRICAL AND WELDING TRAINING
18 – PRE LICENSE REVIEW, AUDIT EXAMINATION
19 – NRC COLD LICENSE EXAMINATIONS
20 – TECHNICAL SUPERVISORY SKILLS PROGRAM(TSSP 1, 2, 3, 4)
21 – HOT LICENSE (REPLACEMENT) TRAINING
22 – PWR
23 – BWR

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

OVERALL TRAINING SCHEDULE

FIGURE 13.2-1 Amendment 5, 4/84

APPENDIX 13B

REACTOR STARTUP EXPERIENCE

Presented by: Selected Contractor ~~Memphis State University Facility~~

Objective

To assign cold license applicants, with no previous nuclear experience, to a Research Reactor Training Course.~~conducted by Memphis State University~~. This ~~1-week~~ course gives the student actual hands-on experience with a research ~~an AGN-201~~ nuclear reactor and allows the cold license applicant to obtain at least the minimum of 10 reactor startups necessary to establish cold license eligibility requirements of ANS 3.1, 1981.

Reference:

ANS/ANSI 3.1-1981 Section 5.2.1.1

APPENDIX 13C

ADVANCED TECHNICAL TRAINING

Presented by: ~~Memphis State University~~ PSE&G or selected Contractor

Objective

To provide advanced technical training in Thermodynamics, Heat Transfer, Fluid Dynamics, Reactor Materials, Reactor Physics and Human Behavior to senior supervisors and STA candidates.

Course Description

The Advanced Technical Training Program consists of nine (9) courses. ~~which total 29 semester credit hours of academic instruction~~. A list of the courses and their content is outlined in the following pages.

References:

ANS/ANSI 3.1-1981	Section 5.2.1.6, 5.3.3	
10 CFR 55.22		
NUREG 0737	Appendix C Section 6.1.2	

The initial training program was taught by Memphis State University. Future programs will be conducted by PSE&G or selected contractors.

APPENDIX 13E

COLD LICENSE CERTIFICATION TRAINING

Presented By: PSE&G or selected Contractor at an approved simulator
~~General Physics, Corp. at the Susquehanna Simulator. The first 4 weeks of systems training will be at our facility.~~

Objective:

1. To ensure that non-experienced (nuclear) personnel meet the cold license eligibility requirements of NUREG-0094 and ANS 3.1 1981

References:

ANS/ANSI 3.1-1981	Section 5.2.1.3.2
NUREG 0737	enclosure 1
10 CFR 55.23	

The initial training program was conducted by General Physics Corp at the Susquehanna training simulator. Future programs will be conducted by PSE&G or selected contractor.

APPENDIX 13F

SS-N TRAINING PROGRAM

Presented By: ~~In part by General Electric and in part by Memphis State University~~
PSE & G or selected contractor

Objective:

To provide advanced instruction to Senior Operators and Supervisors on BWR Specific topics

insert (A) → next page

GENERAL ELECTRIC

1. BWR Chemistry - 1 wk.

2. Nuclear Engineering - 3 wks

3. Corrosion - Materials - 1 wk.

4. Radiological Emergencies - 1 wk.

5. Abnormal Event Analysis - 1 wk.

6. Degraded Core Damage - 1 wk.

*MSU

7 ~~1~~. Materials Study - 2 wks. ~~(3 credits)~~

8 ~~2~~. Human Behavior - 2 wks. ~~(3 credits)~~

NOTES:

a. College Credits for GE Course - 8 (awarded through the N.Y.S. Regents)

b. College Credits for MSU Course - 6

c. Total Course Length - 12 wks.

d. Description of Modules 7 and 8 are provided in Appendix 13C.

*The STA's and SS-N's will be integrated for these courses.

References:

ANS/ANSI 3.1-1981
10 CFR 55.21 and 55.22
NUREG-0737

(A) The initial SRO candidates will attend the SS-N Training program taught by General Electric Company and Memphis State University. ~~in order to obtain college credits.~~ Future programs will be taught by PSE&B or selected contractor personnel.

APPENDIX 13I

COLD LICENSE OPERATOR INPLANT TRAINING

new Paragraph

Presented by: PSE&G or a selected contractor personnel Objectives: To provide cold license candidates with a structured and documented program of plant ~~observation~~, preoperational testing and work assignment participation requirements. system checkouts,

Description: The cold license in-plant training program is designed to ~~give the operator the minimum requirements necessary to be completed during preoperational testing and~~ ensure that each candidate receives sufficient practical work experience necessary to gain a thorough knowledge of the plant. In addition, this program provides for a structural ~~observation~~ program where each candidate receives an oral examination and system check out on plant systems emphasizing system operation, local control and interactions. This in-plant training is documented in the form of individual In-plant Training Guidelines for the RO, SRO and STA candidates.* The completed in-plant training guideline will be maintained in the individuals training record. Insert (B)

Insert (C) →

References: ANS/ANSI 3.1-1981
Section 5.2.1.2.2, 5.2.1.3, 5.2.1.4.

Insert (B)

Because of the scope of this program, completion of these guidelines will be scheduled to coincide with the conclusion of plant hot functional testing.

Incert (C)

Cold License Operator In-Plant Training Guidelines are divided into sections designated by the following groups.

RO

I. System Knowledge Guide Questions

II. System Knowledge

III. Performance Items

IV. Technical Specifications

V. Reactivity Manipulations

SRO

I. Control Board Checkouts

II. Technical Specifications

III. Radiological Controls

IV. Plant Safety

V. Refueling

VI. Procedures

VII. Performance Requirements

STA

I. Control Board Checkouts

II. Plant Safety

III. Procedures

IV. Performance Requirements

APPENDIX 13J

PRE-LICENSE EXAMINATION TESTING AND TRAINING

Presented by: PSE&G or selected contractor personnel.

Objectives: To determine individual candidate's ability to operate the plant in a safe and competent manner and to identify areas of weakness that may be corrected prior to administration of the NRC license examinations.

Description: The pre-license examination testing and training period will consist of an intensive period of instruction and testing prior to the NRC license examinations. The instructional phase of this program will consist of the the following:

a. Classroom presentations on: (4 weeks)

1. Reactor theory review
2. Heat transfer review
3. Fluid mechanics review
4. Thermodynamics review
5. ~~Procedural and operating philosophies~~ Health Physics review
6. Technical Specification and Adminstrative Procedures review
7. Related industry events relevant to operation.

b. Simulator Operation/classroom preperation (≈50/50) (7 weeks)

1. During normal, abnormal and emergency operations to ensure understanding of procedural and operating phosophies

2. To demonstrate the proper use of the emergency operating procedures.

~~c. In-plant demonstrations on equipment operation from local operating panels and equipment locations.~~

The testing phase of this program will ~~consist of:~~ be conducted by other than the normally assigned instructional staff and will consist of: (1 wk)

a. A written examination to determine knowledge level of theory, operating procedures and philosophies, system construction and design and technical specification requirements.

b. An oral examination to determine knowledge level of plant operation from both simulator demonstration and in-plant walk through.

References: ANS/ANSI 3.1-1981, Section 5.2.1.5.

HCGS FSAR

APPENDIX K

Plant Observation / Experience Training

Objective: To provide each cold license candidate (RO/S with extensive operating experience of an operating nuclear facility.

Description: Demonstration of extensive operating experience of each cold license candidate is essential to ensure a safe and timely initial reactor and plant startup. This program is designed to augment the operator training described in appendixes 13A, 13B, 13C, 13D, 13E, 13F, 13G, 13I and 13J to ensure adequate operating experience of a comparable reactor facility. The following sections describe the observation/experience training requirements for each area of training, Reactor Operator and Senior Reactor Operator. Specific segments of each section may be waived by the General Manager Hope Creek Operations for select individuals based on previous training and experience.

I Reactor Operator (RO)

A. Complete simulator certification training program at either the Susquehana simulator or the Hope Creek simulator as described in appendix 13E.

- This program gives each operator hands on experience related to plant operating characteristics of a large (1100 MW_e) BWR under, normal, abnormal and emergency conditions.

B. Participate on shift for two weeks at the Salem Generating Station (1000 MW_e PWR).

1. This program will introduce the operator to PSE&G corporate policies ~~reg~~ ~~arding~~ regarding

the operation of the nuclear facility and administrative procedures covering shift conduct, safety tagging, emergency response and surviellences procedures. The format and bases for many of these procedures will be very similar to those used at the HCGS. This therefore provides early training for the operator on the conduct of operations at HCGS.

2. This program will introduce the operator to the Bailey controls system and their ~~Bailey man~~ operator interface requirements as these will be identical to those utilized in the HCGS control room.

3. This program will introduce each operator to the size and complexity of a commercial nuclear facility including the radiological precautions and health physics procedures.

C. Complete Operator in-plant training requirements described in appendix 13 I.

D. Complete the pre-license examination and testing program described in appendix 13 J including simulator training on the HCGS plant referenced simulator or a simulator of a similar type plant.

II Senior Reactor Operator (SRO)

- Non-Previously Licensed

A. Complete simulator certification training program at the Susquehanna training simulator as described in appendix 13 E.

B. Participate on shift for two (2) weeks at the Salem Generating Station (1000 MWe PWR)

C. Complete Operator in-plant training requirements described in appendix 13 I.

D. Participate on shift for a minimum of six (6) weeks at a large commercial operating BWR facility to meet the experience requirements of ANSI 3.1-1981 section 4.3.1.2 (b)

1. This participation allows for the involvement in the day-to-day operation of the facility as a member of the operating shift. This ~~involvement~~ participation includes review of procedures and technical specifications, observation of ~~observing~~ surviellence testing and control manipulations. This participation gives the supervisor first hand experience in the operation of a large commercial BWR facility.

E. Complete the pre-license examination and testing program described in appendix 13 J including simulator training on the HCGS plant referenced simulator or a simulator of a similar type plant.

- Previously Licensed - PWR

A. Complete a two (2) week simulator training program at the Susquehanna training simulator or similar type plant to familiarize the individual with the controls and response characteristics of a large BWR.

B. Participate on shift for two (2) weeks at the Salem Generating Station

C. Complete operator in-plant training requirements described in appendix 13I.

D. Participate on shift for a minimum of six (6) weeks at a large commercia[l] operating BWR facility to meet the experience requirements of ANSI 3.1-19[illegible] section 4.3.1.2(b).

E. Complete the pre license examination and testing program described in appendix 13J including simulator training on the HCGS plant referenced simulator or a simulator of a similar type plant.

- Previously licensed - BWR

A. Participate on shift for two (2) weeks at the Salem Generating Station

B. Complete Operator in-plant training requirements described in appendix 13I.

C. Complete the pre-license examination and testing program described in appendix 13J including simulator training on the HCGS plant referenced simulator or a simulator of a similar type plant.

In addition to those requirements stated in II above, those SRO license candidates who are scheduled to be senior shift supervisors will participate in a six (6) month program at the Susquehanna Steam Electric Station designed to meet the experience requirements set forth in Generic Letter 84-16 dated June 27, 1984. This program incorporates both participation as a member of the shift of an operating power reactor and during the initial fuel loading and power ~~ascention~~ ascension testing of a large BWR.

Summary: Through this program, each cold license candidate will obtain an extensive working knowledge of large commercial BWR nuclear facilities and, combined with previou training and ~~exper~~ operating experience make safe and reliable operators.

References: ANSI 3.1-1981 section 4.3.1.1
4.3.1.2
4.5.1.2

10 CFR 55.25(b)

Generic Letter 84-10 dated 4/26/84

Generic Letter 84-16 dated 6/27/84

QUESTION 630.4 (SECTION 13.2)

With regard to training in the use of plant systems to control or mitigate an accident in which the core is severely damaged, please provide the training programs and schedule for:

a. Licensed operators and senior operators

b. Other plant personnel
(Ref. H. R. Denton letter of March 28, 1980 and II.B.4 of NUREG-0737)

RESPONSE

Licensed Operators and Operations Personnel

NUREG 0737, Section II.B.4 requires that training of plant personnel be conducted to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged. Enclosure 3 to the H. R. Denton letter dated 3/28/80 identifies the topics that should be included in the training program. In addition this training will stress HCGS system information as it relates heat transfer, fluid flow and thermodynamics considerations to mitigation of core damage.

The HCGS operator training for mitigating core damage is under development. It will incorporate all areas identified in enclosure 3 of the 3/28/80 letter as they are applicable to a BWR:

A. Incore Instrumentation

1. Use of fixed or movable incore detectors to determine the extent of core damage and geometry changes.

2. Methods of determining peak temperatures, extended range readings and direct readings at terminal junction.

3. Methods of calling up incore data from plant process computer.

B. Vital Instrumentation

1. Instrumentation response in an accident environment; failure sequence & indication reliability.

2. Alternate methods for measuring flows, pressures, levels and temperature.

QUESTION 630.7 (SECTION 13.2)

Section 13.2 of the HCGS FSAR contains the training program segments for licensed and non-licensed operations personnel. The segment outlines are contained in the Appendices of 13.2. Please provide the details or information for the following:

a. Prerequisites for personnel assigned to each program.

b. For licensed training, which course(s) will contain the use of HCGS specific procedures including; Administrative, Individual Systems, Integrated Plant, Abnormal and Emergency, Radiological Emergency Response Plan, Technical specifications, Initial Fuel Loading, Low Power and Periodic Surveillance Testing?

c. Please provide the applicable references (Industry Standards, NUREGs, 10 CFR and Regulatory Guides for each of the segments outlined in the Appendices.

d. Identify those training segments which include the subject areas contained in 10 CFR Part 55 Section 21, 22 and 23.

e. The Appendices do not contain a course description of segments i-k of 13.2.1.1. Please provide the course description or a schedule for submittal of the course description.

f. The Appendices do not contain the details of the observation training referenced in 13.2.1.1. Please provide the course description or a schedule for submitting the observation program.

g. Concerning replacement training (hot licenses) for NRC candidates in Section 13.2.1.5, the FSAR must contain, as a minimum, those courses or segments identified in Section 13.2.1.1 or provide a schedule for submittal of this program prior to fuel loading. Ref. (NUREG-0800, 13.2.1.B)

h. Please provide information on the details of SS-N training contained in Appendix 13F. In addition, why are Senior Operators with previous experience excluded from this course as indicated in Figure 13.2-1? (sic) (Ref. NUREG-0800, 13.2.1B)

RESPONSE

a. Personnel assigned to the licensed and non-licensed operator training programs come with diverse backgrounds; however, each individual will meet the education and experience requirements of ANS/ANSI 3.1 - 1981 prior to initiate fuel loading.

In general, the personnel assigned to the licensed operator training come from one of the following areas:

1. Degreed engineer
2. Previously licensed (BWR/PWR)
3. Navy nuclear plant operator
4. Fossil plant operator
5. Salem EO upgrade

In general, personnel assigned to the non-licensed operator training will come from one of the following areas:

1. Qualified utility/equipment operator from Salem Generating Station
2. Navy nuclear plant operator
3. Fossil plant operator

These potential license candidates are required to achieve a satisfactory score on a screening examination as a prerequisite to assignment to the operator training program. At present, Power Operator Service Selection (POSS) is used. Exception to the requirement is made for individual who previously held a NRC license and for degreed personnel. All prospective employees must participate in a physiological screening process. The Minnesota Multi-Phasie Personnality Inventory (MMPI) is presently in use.

b. Training on the HCGS plant specific procedures and technical specifications will be conducted as the procedures become available. These procedures are under development and will become available at various intervals throughout the training period. To ensure that all licensed operator candidates are thoroughly familiar with the procedures and technical specifications, training on plant specific procedures and technical specifications will be incorporated into the training programs outlined in Appendices 13 G, 13 H & 13 I. In addition to this training an intense pre-license training program Appendix 13 J, will be implemented three (3) to six (6) months prior to the license examinations. This training will cover all the HCGS specific operating, abnormal and emergency procedures, administrative and emergency response procedures, technical specifications and low power and surveillance testing procedures. Training will be covered by classroom instruction, in-plant oral examinations, written examinations and performance testing on the Hope Creek specific simulator.

c. Applicable references for each of the segments outlined in the appendices are shown on the appropriate cover sheet of each appendix.

d. Training segments which include 10CFR Part 55 Section 21, 22 and 23 are identified in Appendix 13A, 13C, 13E, 13F and 13G.

e. The following segments of the training program are still under development:

Appendix I - Cold license operator in-plant training
Appendix J - Pre-license examination testing and training

f. A course description for segments i and j of the training program is contained in Appendices 13 I and 13 J, respectively. Appendix 13 K provides a description of on-shift operating experience training.

g. Hot license training for NRC candidates will be conducted to augment the shift staffing allotment, allow for promotion or fill vacancies due to reassignment. This training will utilize a major portion of the existing cold license training program; however, certain areas may be waived based on an individual's prior experience and educational background. Procedures describing the content and administrative requirements will be completed by June 1985.

h. Appendix 13F has been revised to incorporate this response.

QUESTION 630.10 (SECTION 13.2)

Please provide the training programs for all management personnel, technical support staff, and other personnel contained in Figure 13.1-9 through 13.1-13. We believe that Figure 13.2-1 may be modified to include the personnel and training programs. (Ref. NUREG-0800 Section 13.2.I)

RESPONSE

Figures 13.1-9 through 13.1-13 outline the organization structures of the HCGS operations department. The training for each department varies as does the training for the different levels of personnel within each department. This training is conducted as the need arises and the procedures describing the content of the programs is contained in the Nuclear Department Training Procedure Manual. Figure 13.2-1 reflects the initial training of plant staff personnel; however, it is our policy to provide additional training whenever personnel performance identifies as training need.

In addition to the technical training received by department personnel, the Technical Supervisory Skills Program (TSSP) offers technical and management skills training tailored to the identified needs of first line station supervisors and senior supervisors. Required elements of this program shall be completed ~~an individual's~~ by the second anniversary ~~with the Hope Creek staff,~~ of an individual with the Hope Creek staff.

Major areas covered are:

- BWR Technology
- Leadership
- Aberrant Behavior Identification
- Labor Relations
- Management Processes
- Technical Administration
- QA

Procedures describing the contents of these programs are contained in the Nuclear Department Training Procedure manual.

ATTACHMENT 6