PSEG Public Service Electric and Gas Company

80 Park Plaza, Newark, NJ 07101 / 201 430-8217 MAILING ADDRESS / P.O. Box 570, Newark, NJ 07101

Robert L. Mittl General Manager
Nuclear Assurance and Regulation

September 14, 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.

In addition, enclosed (see Attachment 5), is revised supplementary information to FSAR Section 13.4. This information supercedes the proposed HCGS Technical Specifications transmitted on September 13, 1984.

Also, enclosed (see Attachment 6), is Revision 1 to the response to IE Bulletin 81-03 (supercedes 8/12/84 submittal) as requested by the Auxiliary System Branch.

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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 open items, please contact us.

Very truly yours,

RL Mittl/RP Douglas

Attachments/Enclosure

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

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

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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 NRC requests for additional information 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 14th day of September 1984.

David K. Burd

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

MC 28 02

DATE: 9/14/84

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
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 T 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)

CPSS 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 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	Complete	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	8/3/84 (Rev 1)
147b	9.3.1	Compressed air systems	Complete	8/3/84 (Rev 1)
147c	9.3.1	Compressed air systems	Complete	8/3/84 (Rev 1)
147d	9.3.1	Compressed air systems	Complete	8/3/84 (Rev 1)
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 T 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	9/5/84 (Rev. 1)
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	/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 syst	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	9/13/84 (Rev. 1)
224	8.2.2.4	Common failure mode between onsite and offsite power circuits	Complete	8/1/84

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	8/1/84
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	9/13/84 (Rev. 1)
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	9/13/84 (Rev. 1)
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	8/1/84
238	8.2.2.7	Sequencing of loads on the offsite power system	Complete	8/1/84

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 after prolonged no load operation of the diesel generator	Complete	9/13/84 (Rev. 2)
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	9/13/84 (Rev. 2)
245	8.3.3.3.2	The use of 18 versus 36 inches of separation between raceways	Complete	8/15/84 (Rev. 1)
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	9/14/84 (Rev. 1)
252	8.3.3.5.6	The use of a single breaker to provide penetration protection	Complete	9/14/84 (Rev. 1)
253	8.3.3.1.4	Commitment to protect all Class 1E equipment from external hazards versus only class 1E equipment in one division	Complete	9/13/84 (Rev. 1)
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	9/13/84 (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 as an isolation device	Complete	8/1/84
260	8.3.3.3.5	Use of a single breaker tripped by a LOCA signal used as an isolation device	Complete	9/13/84 (Rev. 1)
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 T 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	

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ATTACHMENT 2 DATE: September 14, 1984

DRAFT SER SECTIONS AND DATES PROVIDED

SECTION	DATE
3.1	
3.2.1	
3.2.2	
5.1	
5.2.1	
6.5.1	See Notes 1&5
8.1	See Note 2
8.2.1	See Note 2
8.2.2	See Note 2
8.2.3	See Note 2
8.2.4	See Note 2
8.3.1	See Note 2
8.3.2	See Note 2
8.4.1	See Note 2
8.4.2	See Note 2
8.4.3	See Note 2
8.4.5	See Note 2
8.4.6	See Note 2
8.4.7	See Note 2
8.4.8	See Note 2
9.5.2	See Note 3
9.5.3	See Note 3
9.5.7	See Note 3
9.5.8	See Note 3
10.1	See Note 3
10.2	See Note 3
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

SECTION	DATE
11.4.1	See Notes 1&5
11.4.2	See Notes 1&5
11.5.1	See Notes 1&5
11.5.2	See Notes 1&5
13.1.1	See Note 4
13.1.2	See Note 4
13.2.1	See Note 4
13.2.2	See Note 4
13.3.1	See Note 4
13.3.2	See Note 4
13.3.3	See Note 4
13.3.4	See Note 4
13.4	See Note 4
13.5.1	See Note 4
15.2.3	
15.2.4	
15.2.5	
15.2.6	
15.2.7	
15.2.8	
15.7.3	See Notes 1&5
17.1	8/3/84
17.2	8/3/84
17.3	8/3/84
17.4	8/3/84

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)

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ATTACHMENT 3

OPEN ITEM	DSER SECTION	SUBJECT
222	8.2.2.2	Design provisions for re-establishment of an offsite power source
254	8.3.3.1.5	Protection of Class 1E power supplies from failure of unqualified Class 1E loads
251	8.3.3.5.5	Fault current analysis for all representative penetration circuits
252	8.3.3.5.6	The use of a single breaker to provide penetration protection

ATTACHMENT 4

DSER Open Item No. 222 (DSER Section 8.2.2.2)

DESIGN PROVISIONS FOR REESTABLISHMENT OF AN OFFSITE POWER SOURCE

GDC 17 requires, in part, that each of the offsite circuits be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. The description in the FSAR as to compliance with this part of GDC 17 is not sufficient to reach a conclusion of acceptability.

By Amendment 4 to the FSAR the applicant in response to a request for information, stated that in the event of relay operation, the relays can be reset and the equipment returned to service within one hour. This design provision description for reset of relays may be related to design provisions used for reestablishing an offsite circuit from the transmission system through the switchyard to the Class 1E system; however, the description by itself is not sufficient to reach a conclusion of acceptability nor is it responsive to the staff request for information.

RESPONSE

FSAR section 8.2.1.4 and Figure 8.2-2 have been revised to provide the requested information.

QUESTION 430.3 (SECTION 8.2)

GDC 17 requires, in part, that each of the offsite circuits be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. The description in the FSAR as to compliance with this part of GDC 17 is not sufficient to reach a conlcusion of acceptability. Describe design provisions for establishing an offsite circuit from the transmission system through the switchyard to the Class 1E system describing some event in the switchyard or protective relaying that has tripped all 500 kV switchyard breakers.

RESPONSE

Section 8.2.1.4 and Figure 8.2-2 has been revised to provide this response.

provide an auxiliary switch contact for input to generating station computer systems via a data input/output (I/O) cabinet for status indication. For safety reasons, the control switches are provided with a lock-in handle. The generating station control room operator must release keys in his possession to operate these switches.

8.2.1.4 Switchyard

The 500-kV switchyard, located to the east of the Hope Creek plant, is designed with tapered tubular steel structures and rigid aluminum bus work. This yard consists of two breaker-and-a-half bays containing five SF-6 circuit breakers connected to two 500-kV main buses, 10X and 20X, as shown on Figure 8.2-2. Bus 10X is protected by primary and backup differential relays. Breaker failure relaying detects a failure-to-trip or failure-to-interrupt condition at the line terminal and trips associated breakers necessary to isolate the line. A

Generating station auxiliary services are supplied via two 13.8-breaker bays by four 500/14.4 kV, 42/56/70-MVA, oil-immersed, self-cooled/forced-air-forced-oil-cooled (OA/FOA/FOA) three-phase transformers connected to the 500-kV busses 10X and 20X, as shown on Figure 8.2-2. Station power transformers T1 and T4 each supply two 13.8/4.16-kV and one 13.8/7.2-kV station service transformers. The remaining two transformers, T2 and T3, each supply one 13.8/4.16-kV station service transformer and one 14.4-kV/208V station light and power transformer. Each 13.8-kV breaker bay consists of three breakers in series. To prevent paralleling of the transformers, one of the breakers is normally open. This breaker is closed in case one of the transformers is out of service.

As shown on Figure 8.2-2, there are six 13.8-kV, 1500-MVA oil circuit breakers. Breaker failure protection detects the failure to trip or failure to interrupt conditions at the line terminals and electrically isolates faulty equipment. Primary and backup relay protection on the 500/14.4-kV station power transformers is provided by the use of harmonic restraint differential relays.

The 13.8-kV system is ungrounded and connected to the delta side of all station power and station service transformers. To detect a phase-to-ground fault in the system, a 13.8-kV/208-V grounded-wye grounding transformer is installed on the secondary side of each station transformer. The neutral of the grounding

DSER OPEN ITEM 222

INSERT 'A'

The 500 KV circuit breakers are pneumatically operated and each breaker has sufficient stored air for a minimum of three operations without compressor actuation. The compressor motor is fed from the breaker A.C. distribution panel, which is provided with two independent A.C. circuits from the switchyard control house.

The control room and the switchyard control house have independent and simultaneous control of the 500 KV circuit breakers. The electric system operation center, located in Newark, N.J., has limited control of the line breakers 51x, 60x, and 61x and the tie breaker 50X, and no control of the generator breaker, 52x.

Restoration of the 500 KV lines would generally consist of the following procedural steps:

The system load dispatcher would be contacted to verify availability of 500 KV circuits.

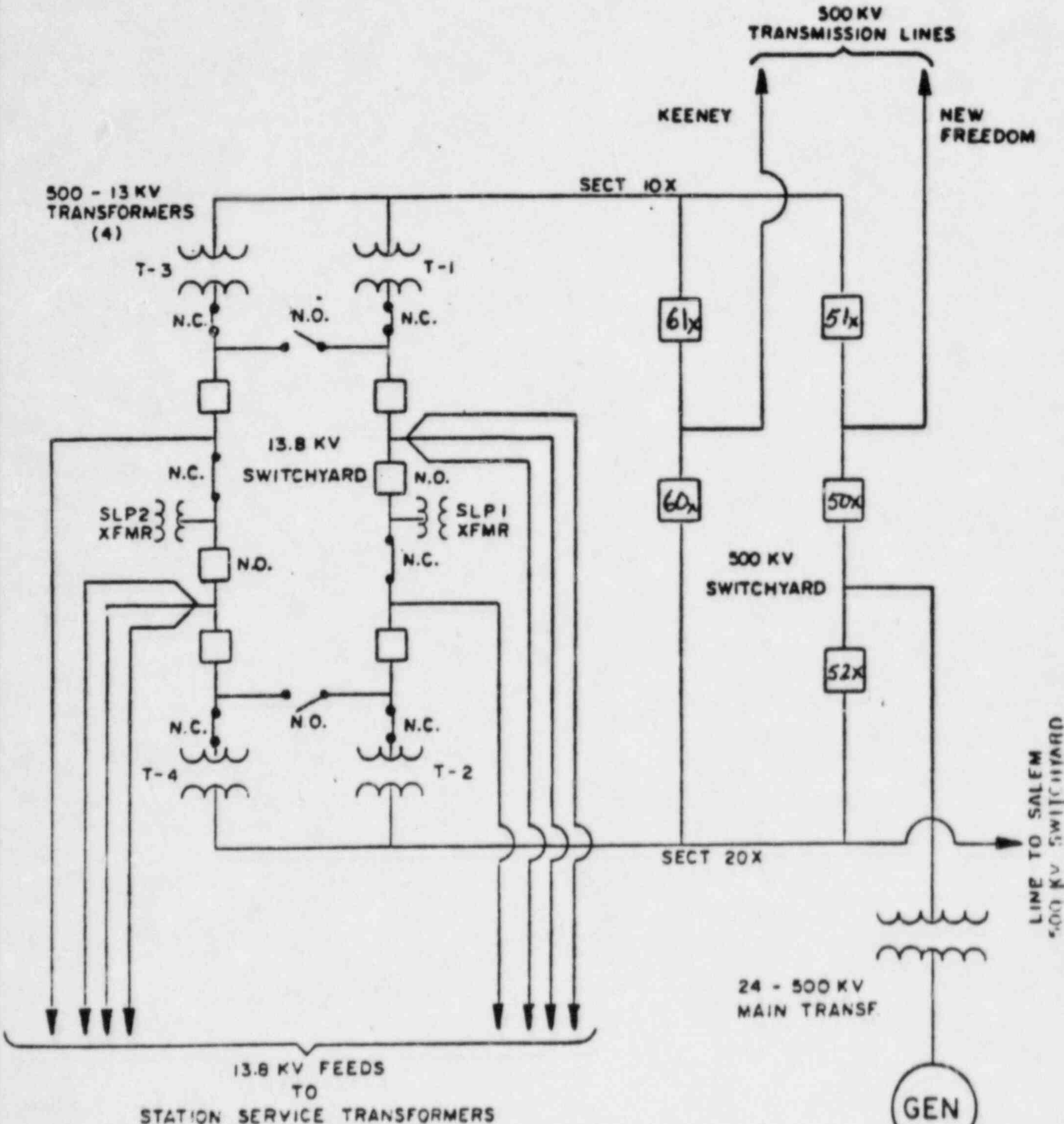
Verify 4 KV & 7.2 KV non-1E bus infeed breakers are opened.

Verification of 500/13KV transformer and 13 KV ring bus breaker positions aligned to restore offsite power.

The load dispatcher is contacted for final clearance to reclose 500 KV breakers.

Once 500 KV power is reestablished, 4 KV & 7.2 KV power is provided to the respectives non-IE Buses, loading of these non-IE buses can then comense.

Final tranfer of class IE loads from the stand-by to the perferred power source can be made when plant conditions are stable.



DSER OPEN ITEM 222

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

ONE-LINE DIAGRAM

FIGURE 8.2-2

DSER Open Item No. 254 (DSER Section 8.3.3.1.5)

PROTECTION OF CLASS 1E POWER SUPPLIES FROM FAILURE OF UNQUALIFIED CLASS 1E LOADS

In Section 8.1.4.6 of the FSAR, it is stated that Class 1E equipment is qualified to perform its function during applicable design basis accidents. The terminology "applicable design basis accidents" is of concern Sections 4.2 and 4.7 of IEEE standard 308-1974 requires that Class 1E equipment be designed and qualified to perform their function during any design basis event. If a Class 1E component is subject to the effects of a design basis event environment, that component must be designed and qualified to function in that environment irrespective of the fact that the component may not be directly required to mitigate the design basis event.

By Amendment 4 to the FSAR, the applicant indicated that safety-related equipment that is not qualified (because it does not have to perform a safety function to mitigate the design basis event condition to which it is being subjected) are identified in Table 3.11-6 of the FSAR.

In justification of this design, the applicant further indicated that this identified equipment is connected to its power supply by a Class 1E circuit breaker. The circuit breaker will operate to clear any fault caused by the failure of unqualified equipment. Thus, under the single failure criterion only one Class 1E circuit breaker is postulated to sail. The failure of this one circuit breaker can degrade only its associated power supply bus. The redundant power supply and load will be available to perform the safety load.

Further justification or assurance that Class 1E power supplies will not fail as a result of failure of unqualified equipment and results of analysis that provide a positive statement to the effect that the unqualified equipment failure position will not affect station shutdown capability will be pursued with the applicant.

RESPONSE

Additional justification has been provided in the revised response to Question 430.~~30~~.
.37

QUESTION 430.37 (SECTION 8.3.1 and 8.3.2)

In Sections 8.1.4.6 of the FSAR you state that Class 1E equipment is qualified to perform its function during applicable design basis accidents. The terminology "applicable design basis accidents" is not clear. Section 4.2 and 4.7 of IEEE standard 308-1974 requires that Class 1E equipment be designed and qualified to perform their function during any design basis event. If a Class 1E component is subject to the effects of a design basis event environment, that component must be designed and qualified to function in that environment irrespective of the fact that the component may not be directly required to mitigate the design basis event. For each design basis event defined in Table 1 of IEEE standard 308-1974:

a. Identify each Class 1E component that does not meet the design and qualification guidelines of Sections 4.2 and 4.7 of IEEE Standard 308-1974, and

b. Provide an analysis that demonstrates compliance with the single failure criterion assuming simultaneous failure of all components identified above with their associated power supplies.

RESPONSE

a. The term "applicable design basis accident" is used to more precisely describe the postulated DBE which the safety-related components needed to mitigate that DBE will be required to operate in, and thus describe the conditions the equipment must be qualified to. This is ian complaince with NUREG-0588, Part 2.1(3)(a) which states in part, "should be qualified by test to demonstrate its operability for the time required in the environmental conditions resulting from that accident." It is PSE&G's position to comply with this requirement for each piece of safety-related (Class 1E) equipment.

PSE&G agrees that safety-related components (Class 1E) should be designed and qualified to function for each DBA. However, function in the case of components not required to mitigate a DBA is the requirement not to fail in a manner detrimental to plant safety as specified in NUREG-0588, Part 2.1(3)(b). This is interpreted to mean that if the component is not required to operate during the DBE, the qualification requirement for such components is to demonstrate that they will not fail in a manner which would prevent safe shutdown under the postulated DBA environmental condition.

~~The specific design basis events considered for HCGS are discussed in Chapter 15. These events are comparable to the generic postulated events of Table I of IEEE-308-1974.~~

~~Safety-related equipment that does not have to be qualified as determined by functionality reviews and/or DBA conditions have been identified in Table 3.11-6 of the FSAR.~~

~~b. The equipment identified above is connected to its power supply bus by a Class 1E circuit breaker. The power supply bus and the circuit breaker are qualified for the DBE environment in which they are located. The Class 1E circuit breaker will operate to clear any fault caused by the failure of the equipment identified above. Under the single failure criteria application, only one Class 1E circuit breaker is postulated to fail. The failure of this circuit breaker can degrade only its associated power supply bus. In this event, a combination of the redundant power supply bus and load is available to perform the safety function.~~

See attached revised response

DSER OPEN ITEM 254

RESPONSE TO 430.37

There are no unqualified Class 1E components in use at Hope Creek Generating Station. Each Class 1E component is qualified in accordance with the requirements of 10CFR50.49 "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants", NUREG 0588, and IEEE 308-1974.

All Class 1E devices are powered from Class 1E power supplies and are separated from these Class 1E supplies by qualified Class 1E circuit breakers or interrupting devices.

DSER OPEN ITEM 254

DSER Open Item No. 251 (DSER Section 8.3.3.5.5)

FAULT CURRENT ANALYSIS FOR ALL REPRESENTATIVE PENETRATION CIRCUITS

By Amendment 4 to the FSAR, the applicant indicated that coordinated fault-current versus time curves for representative penetration conductors and their protective devices are included in Figures 420.46-1 of the FSAR. Based on a review of these figures, the staff concludes that representative curves for motor differential relay, current transformer, and instrumentation circuits were not included in Figure 430.46-1. Inclusion of these circuits as well as other circuits such that the coordinated fault-current versus time curves is representative of all penetration circuits will be pursued with the applicant.

RESPONSE

The response to Question 430.46 has been revised to provide additional fault current versus time curves.

larger than the circuit conductor. The remainder of the requested information is as follows:

a. HCGS complies with position 1 of Regulatory Guide 1.63 as stated in Section 8.1.4.12. In addition, the penetration assemblies are designed to withstand, without loss of mechanical integrity, the maximum short-circuit current vs. time conditions that could occur, given single random failures of circuit overload protection devices. Time current characteristic curves, based on tests, of the penetration conductors have been established by the penetration supplier; these curves show the maximum duration of symmetrical short circuit current. Based on these curves the primary and backup protective devices are selected to ensure that the mechanical integrity of the penetrations is maintained. This is further demonstrated in Part b, below. The testability of the primary and backup protective devices is addressed in the response to Question 430.48.

b. Coordinated fault-current versus time curves for representative penetration conductors and the protective devices are included in attached Figures 430.46-1.

c. The test report that substantiates the capability of the electrical penetration to withstand fault current without seal failure for worst case environmental conditions has been submitted under a separate cover.

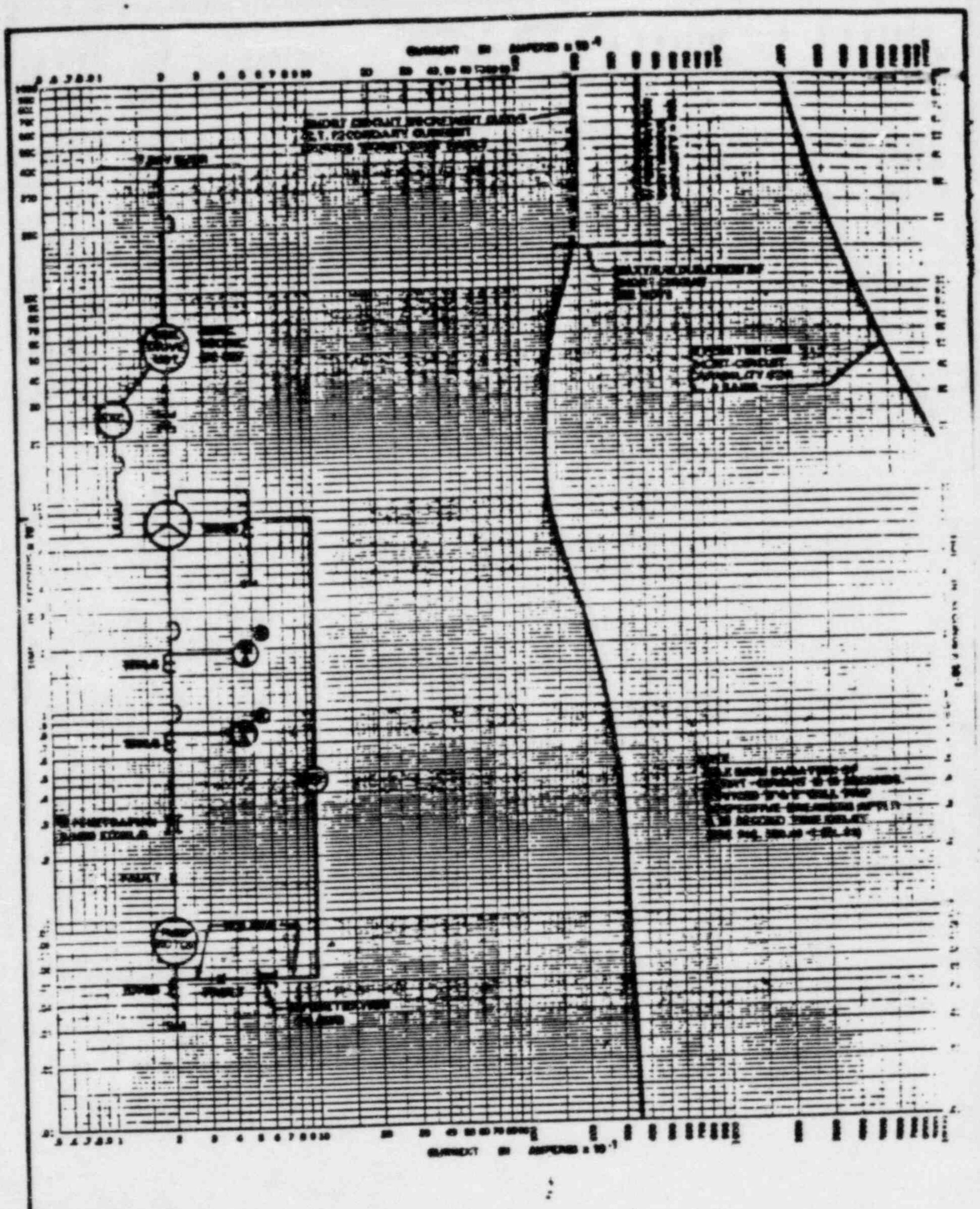
THE 1000 kcmil MEDIUM VOLTAGE PENETRATIONS ASSOCIATED WITH REACTOR RECIRCULATION PUMP MOTORS, ARE PROTECTED BY TWO CLASS 1E CIRCUIT BREAKERS IN SERIES AS SHOWN ON REVISED FSAR FIG. 8.3-4. SECTION 8.1.4.12 HAS BEEN REVISED TO INCLUDE THIS RESPONSE.

Add insert

DSER OPEN ITEM 251

Insert

The 1000 kc mil medium voltage penetrations associated with reactor recirculation pump motors, are protected by two Class 1E circuit breakers in series as shown on revised FSAR Figure 8.3-4. Section 8.1.4.12 has been revised to include this response. Figure 430.46-1 Sheet 21 is a typical coordination curve for a #16 AWG penetration for RTD and thermocouple circuits. The curves show that the instrument penetration is protected for the maximum short-circuit current.

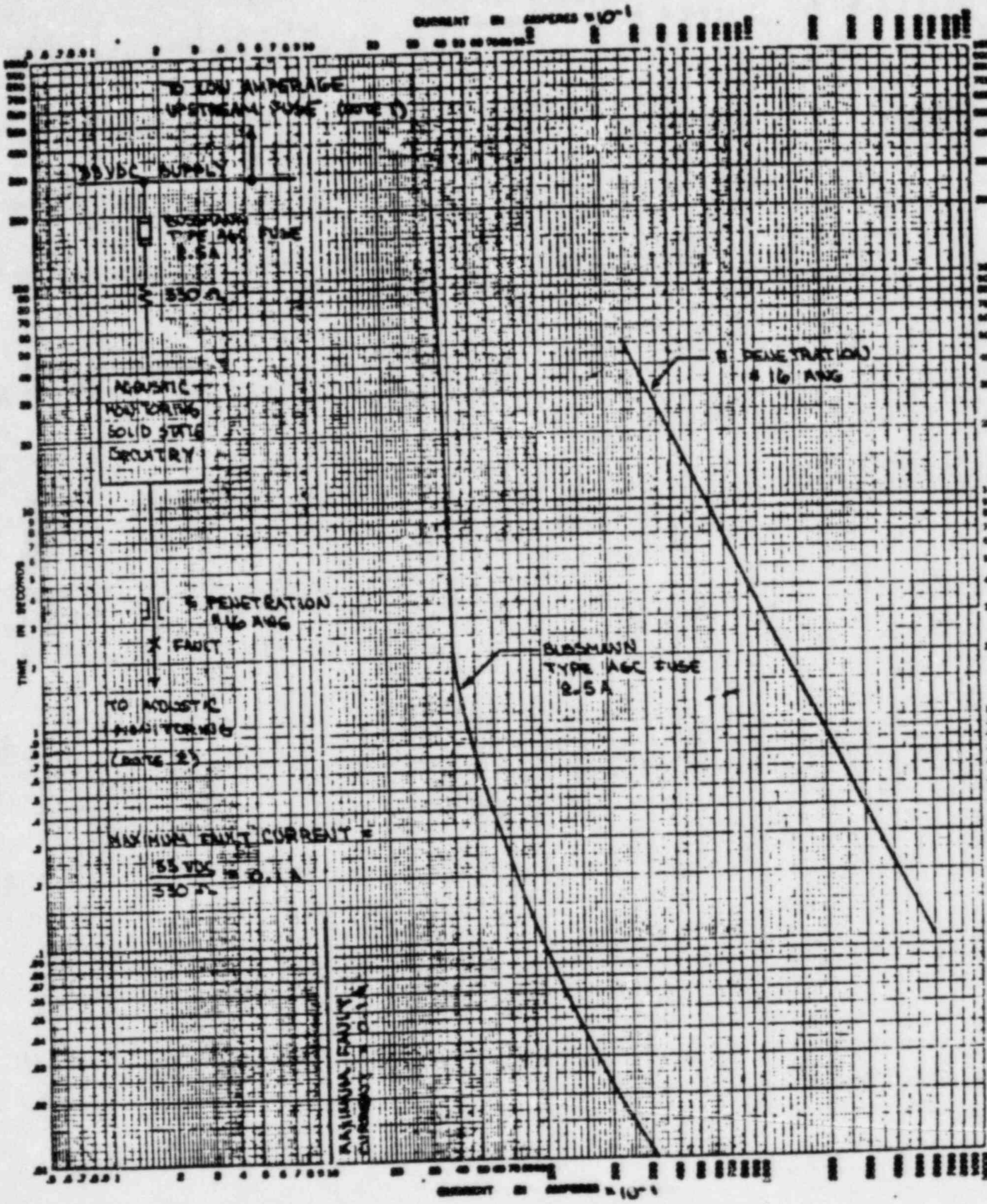


HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

PENETRATION CONDUCTOR OVERCURRENT PROTECTION

FIGURE 8.3
SHEET 13

DSER OPEN ITEM 251



NOTES

1) THE UPSTREAM FUSE PROVIDES OVERCURRENT PROTECTION OF THIS POWER SUPPLY AND OTHER POWER SUPPLIES OF THIS SYSTEM.
2) THIS CIRCUIT IS SIMILAR TO OTHER INSTRUMENTATION CIRCUITS THAT HAVE LOW ENERGY POWER SUPPLIES AND LOW SIGNAL LEVELS.

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f. 120-V ac lighting circuits

g. Motor differential relay current transformer circuits

h. Low voltage instrumentation circuits

i. Communication circuits.

The following system features are provided to ensure compliance with the Regulatory Guide position on single random failures of circuit overload protection devices:

a. Medium voltage penetration assemblies: The only medium voltage circuits routed through the penetration are the 3.92-kV circuits for the two reactor recirculation pump motors. Each motor is supplied from a variable frequency motor-generator set. The maximum fault current available for a fault inside the containment is limited by the generator contribution and the circuit resistance. PRIMARY AND BACKUP PROTECTION FOR THE 1000 kcmil PENETRATION IS PROVIDED BY TWO CLASS 1E CIRCUIT BREAKERS IN SERIES AS SHOWN IN FSAR FIG. 8.3-4. EACH CIRCUIT BREAKER IS PROVIDED WITH AN OVERCURRENT RELAY. THESE RELAYS ARE SET TO TRIP THEIR RESPECTIVE CIRCUIT-BREAKERS. FIG. 430.46 SHEET 11 SHOWS THAT THE TIME-CURRENT CAPABILITY OF THE 1000 kcmil PENETRATION IS GREATER THAN ANY MAXIMUM SHORT CIRCUIT CURRENT VS. TIME CONDITION THAT COULD OCCUR.

b. 480-V ac motor feeder circuits: The 480-V ac loads inside the containment consist of Class 1E and non-Class 1E motor-operated valves and non-Class 1E continuous-duty motors. All these loads are supplied from 480-V motor control centers (MCCs).

The magnetic-only circuit breaker used in the combination starter for the motor provides primary protection for penetration conductors. A thermal-

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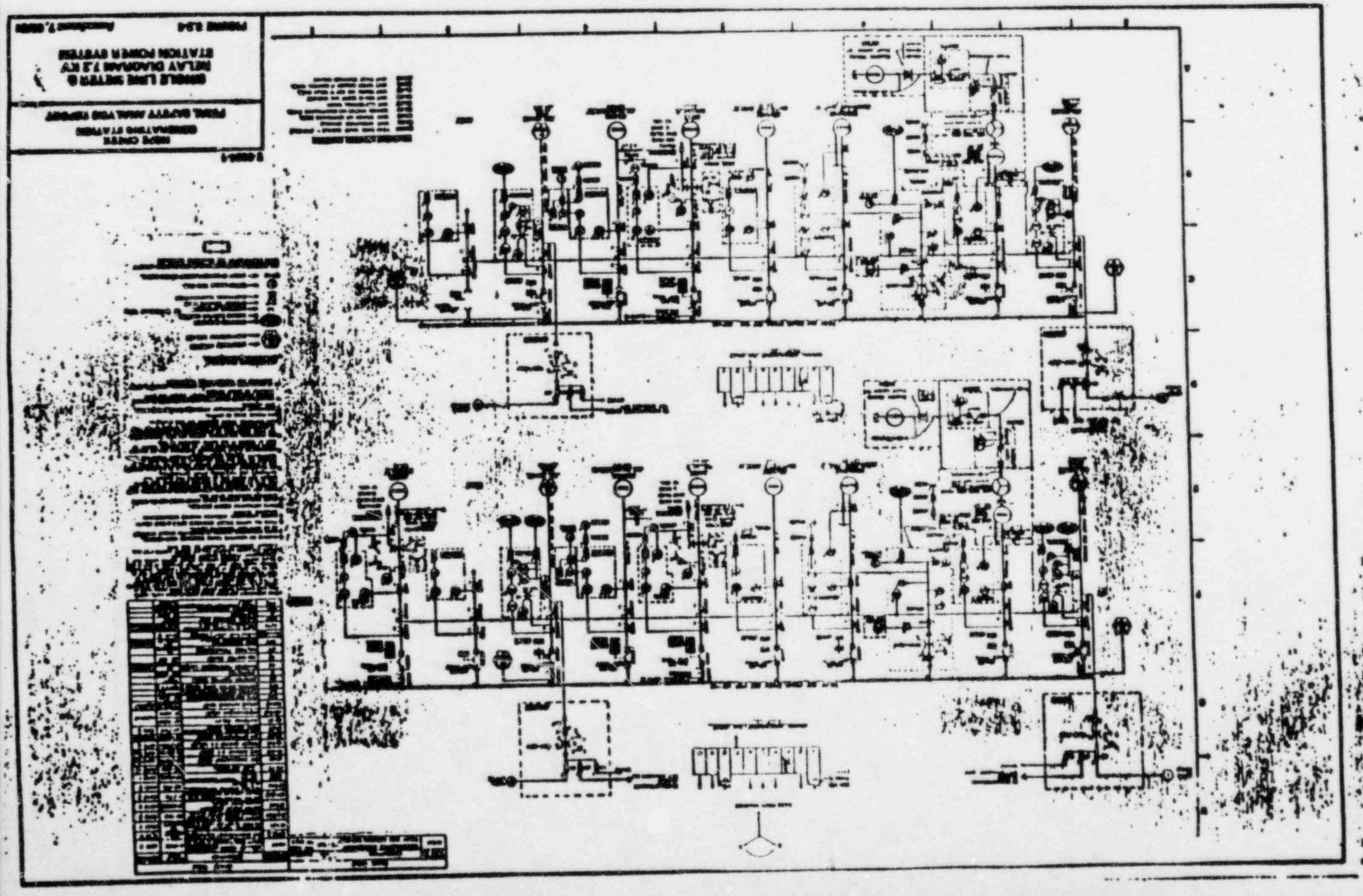


Rev 1

DSER Open Item No. 252 (DSER Section 8.3.3.5.6)

THE USE OF A SINGLE BREAKER TO PROVIDE PENETRATION PROTECTION

By Amendment 4 to the FSAR, the applicant has indicated that penetration protection for the two reactor recirculation pump motor circuits is provided by a single breaker that is tripped by primary and backup relaying. This design does not meet the requirements of position 1 of Regulatory Guide 1.63. Justification for noncompliane will be pursued with the applicant.

RESPONSE

Figure 430.46-1, Sheet 11, of Amendment 7, has been revised to show two breakers.

The only penetrations with instrument class circuits that are protected by a single circuit breaker or fuse are as follows:

1. Vibration Monitoring
    a. Circuit Breaker is 7 amps.
    b. Maximum short circuit current is 0.8 amps.
    c. Penetration is #16 wire with a continuous rating of 15 amps.
    d. These penetrations have a continuous rating in excess of 18 times the maximum short circuit current they may be expected to experience.

2. Neutron Monitoring System
    a. Circuit protected by a 1/4 amp fuse.
    b. Maximum short circuit current is 0.2 amps.
    c. Penetration is #16 wire with a continuous rating of 15 amps.
    d. These penetrations have a continuous rating in excess of 75 times the maximum short circuit current they may be expected to experience.

3. Acoustical Monitoring System
    a. Circuit protected by a 2.5 amp. fuse
    b. Maximum short circuit current <0.1 amp. (The 330kΩ resistor would limit the short circuit to 0.1 amp even if the rest of the circuit impedance was zero.)
    c. Penetration is #16 wire with a continuous rating of 15 amps.
    d. These penetrations have a continuous rating in excess of 150 times the maximum short circuit current they may be expected to experience.

REV 1

4. Thermocouple Circuits

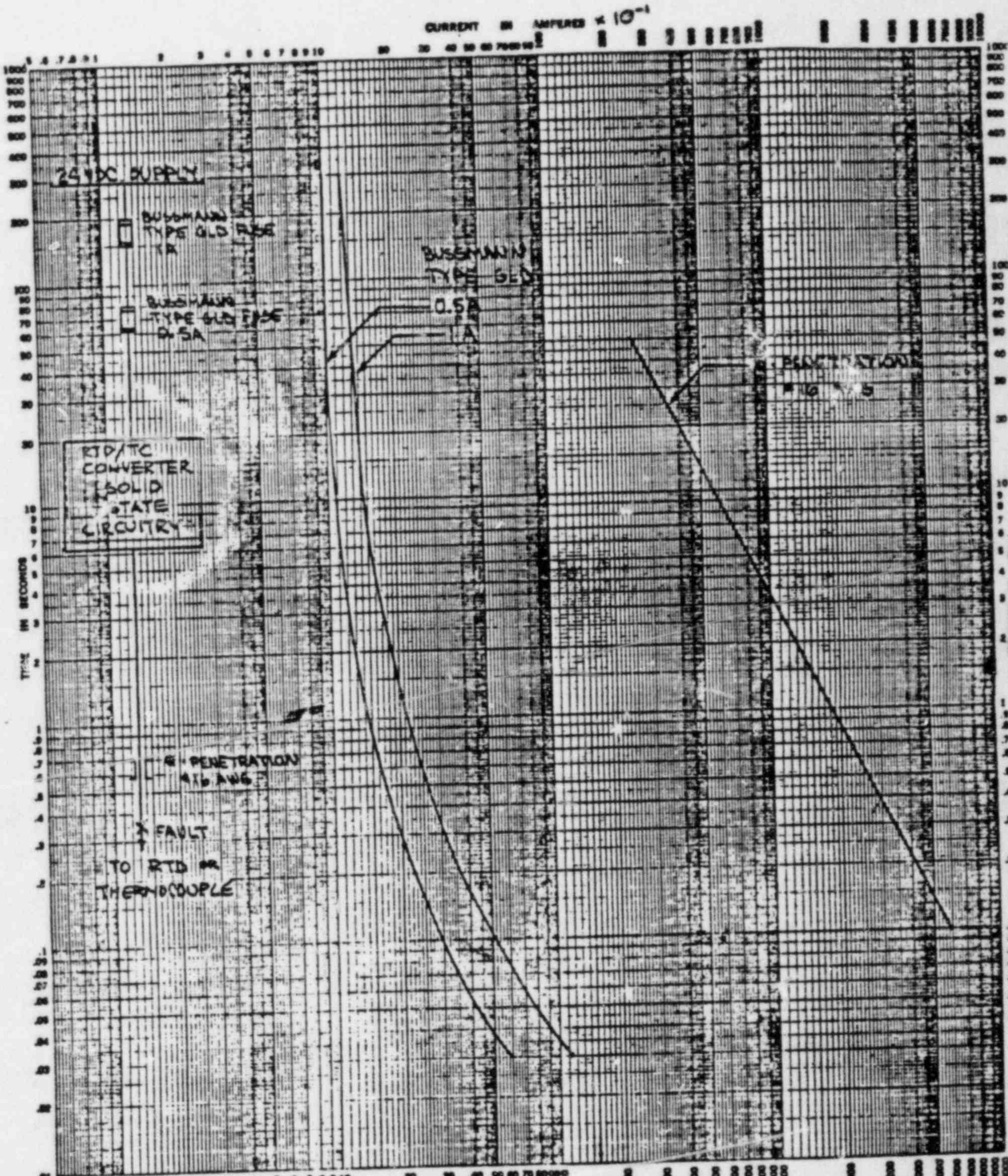
   a. Thermocouples cannot generate any conceivable short circuit challange to a penetration.

5. P.A. Voice Circuits

   a. These circuits carry millivolt signals only when they are actually transmitting a voice communication. The system cannot generate any conceivable short circuit challange to a penetration.

The above cases illustrate that the intent of Reg. Guide 1.63 is met. No single failure of a circuit over current protective device could cause a penetration failure. Refer to the representative curves of Figure 430.46-1.

REV 1



HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

PENETRATION CONDUCTOR OVERCURRENT PROTECTION

Amendment

ATTACHMENT 5

6.5 REVIEW AND AUDIT

6.5.1 STATION OPERATIONS REVIEW COMMITTEE (SORC)

FUNCTION

6.5.1.1 The Station Operations Review Committee shall function to advise the General Manager - Hope Creek Operations on operational matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Station Operations Review Committee (SORC) shall be composed of:

Chairman:	Assistant General Manager - Hope Creek Operations
Member and Vice Chairman:	Operations Manager
Member and Vice Chairman:	Technical Manager
Member and Vice Chairman:	Maintenance Manager
Member:	Operating Engineer
Member:	I & C Engineer
Member:	Senior Nuclear Shift Supervisor
Member:	Technical Engineer
Member:	Maintenance Engineer
Member:	Radiation Protection Engineer
Member:	Chemistry Engineer
Member:	Manager - On Site Safety Review Group or his designee.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the SORC Chairman.

a. Vice Chairmen shall be members of Station management.

b. No more than two alternates to members shall participate as voting members in SORC activities at any one meeting.

c. Alternate appointees will only represent their respective department.

d. Alternates for members will not make up part of the voting quorum when the member the alternate represents is also present.

MEETING FREQUENCY

6.5.1.4 The SORC shall meet at least once per calendar month and as convened by the SORC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the SORC necessary for the performance of the SORC responsibility and authority provisions of these technical specifications shall consist of the Chairman or his designated alternate and five members including alternates. No more than two alternates to members shall participate as voting members in SORC activities at any one meeting.

RESPONSIBILITIES

6.5.1.6 The Station Operations Review Committee shall be responsible for:

a. Review of: (1) Station Administrative Procedures and changes thereto and (2) Newly created procedures or changes to existing procedures that involve a significant safety issue as described in Section 6.5.3.2.d.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Review of the safety evaluations that have been completed under the provisions of 10CFR50.59.

f. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Vice President - Nuclear and to the General Manager - Nuclear Safety Review.

g. Review of all REPORTABLE EVENTS.

h. Review of facility operations to detect potential nuclear safety hazards.

i. Performance of special reviews, investigations or analyses and reports thereon as requested by the General Manager - Hope Creek Operations or General Manager - Nuclear Safety Review.

j. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the General Manager - Nuclear Safety Review.

k. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the General Manager - Nuclear Safety Review.

l. Review of the Fire Protection Program and implementing procedures and shall submit recommended changes to the General Manager - Nuclear Safety Review.

m. Review of all unplanned on-site releases of radioactivity to the environs including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President - Nuclear and to the General Manager - Nuclear Safety Review.

n. Review of changes to the PROCESS CONTROL MANUAL and the OFF-SITE DOSE CALCULATION MANUAL.

SORC REVIEW PROCESS

6.5.1.7 A technical review and control system utilizing qualified reviewers from within the station organization shall be established to perform the periodic or routine review of procedures and changes thereto. Only those items that have a safety significance will be reviewed by SORC. Details of this technical review process are provided in Section 6.5.3.

SORC reviews will concentrate on safe and reliable operation of the station. Independent reviews for determination or verification of USQ shall be performed by the Nuclear Safety Review Department (NSR) and the results of NSR reviews will be provided to SORC.

AUTHORITY

6.5.1.8 The Station Operations Review Committee shall:

a. Recommend to the General Manager - Hope Creek Operations written approval or disapproval of items considered under 6.5.1.6 (a) through (e) above.

b. Provide written notification within 24 hours to the Vice President - Nuclear and the General Manager - Nuclear Safety Review of disagreement between the SORC and the General Manager - Hope Creek Operations; however, the General Manager - Hope Creek Operations shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.9 The Station Operations Review Committee shall maintain written minutes of each meeting and copies shall be provided to the Vice President - Nuclear, the General Manager - Nuclear Safety Review and the Manager - Off-Site Review.

6.5.2 NUCLEAR SAFETY REVIEW

FUNCTION

6.5.2.1 The Nuclear Safety Review Department (NSR) shall function to provide the independent safety review program and audit of designated activities.

COMPOSITION

6.5.2.2 NSR shall consist of a General Manager, a Manager of the On-Site Safety Review Group (SRG) supported by at least four dedicated, full-time engineers located on-site, and a Manager of the Off-Site Review Group (OSR) supported by at least four dedicated, full time engineers located off-site.* The OSR staff shall possess experience and competence in the general areas listed in Section 6.5.2.4. The General Manager and Managers will determine when technical experts shall be used to assist in reviews of complex problems.

NSR shall establish a system of qualified reviewers from other technical organizations to augment its expertise in the disciplines of Section 6.5.2.4. Such qualified reviewers shall meet the same qualification requirements as the NSR staff, and will not have been involved with performance of the original work.

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*Since the Nuclear Department is located on Artificial Island site, the terms on-site and off-site are intended to convey the distinction between inside and outside of the station fence.

Establishment of the Manager - Off-Site Review and Staff is guided by the provisions for independent review of Section 4.3 of ANSI N18.7 (ANS-3.2), and the qualification requirements for the review staff will meet or exceed those described in Section 4.7 of ANS-3.1. The Manager - On Site Review and staff will meet or exceed the qualifications described in Section 4.4 of ANS 3.1.

CONSULTANTS

6.5.2.3 Consultants shall be utilized as determined by the NSR General Manager to provide expert advice to the NSR.

OFF-SITE REVIEW GROUP

6.5.2.4 The Off-Site Review Group (OSR) shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear Power Plant Operations

b. Nuclear Engineering

c. Chemistry and Radiochemistry

d. Metallurgy

e. Instrumentation and Control

f. Radiological Safety

g. Mechanical Engineering

h. Electrical Engineering

i. Quality Assurance

j. Nondestructive Testing

k. Emergency Preparedness

It shall also function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources which may indicate areas for improving plant safety.

REVIEW

6.5.2.4.1 The OSR shall review:

a. The Safety evaluations for

   1) Changes to procedures, equipment, or systems and

   2) Tests or experiments completed under the provision of Section 50.59, 10CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment, or systems that involve an unreviewed safety question as defined in Section 50.59, 10CFR.

c. Proposed tests or experiments that involve an unreviewed safety question as defined in Section 50.59, 10CFR.

d. Proposed changes to Technical Specifications or to the Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. All REPORTABLE EVENTS

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems or components.

i. Reports and meeting minutes of the Station Operations Review Committee.

AUDITS

6.5.2.4.2 Audits of facility activities that are required to be performed under the cognizance of OSR are listed below:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training, and qualifications of the entire facility staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the Criteria of Appendix "B", 10CFR50, at least once per 24 months.

e. The Facility Emergency Plan and implementing procedures at least once per 12 months.

f. The Facility Security Plan and implementing procedures at least once per 12 months.

g. Any other area of facility operation considered appropriate by the General Manager - Nuclear Safety Review or the Vice President - Nuclear.

h. The Facility Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified off-site licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

k. The radiological environmental monitoring program and the results thereof at least once per 12 months.

The above audits shall be conducted by the Quality Assurance Department or an independent consultant. Audit results and recommendations shall be reviewed by NSR.

ON-SITE SAFETY REVIEW GROUP

6.5.2.5 The On-Site Safety Review Group (SRG) shall function to provide: the review of plant design and operating experience for potential opportunities to improve plant safety; the evaluation of plant operations and maintenance activities; and advice to management on the overall quality and safety of plant operations.

The SRG will make recommendations for revised procedures, equipment modifications, or other means of improving plant safety to appropriate station/corporate management.

RESPONSIBILITIES

6.5.2.5.1 The SRG shall be responsible for:

a. Review of selected plant operating characteristics, NRC issuances, industry advisories, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety.

b. Review of selected facility features, equipment, and systems.

c. Review of selected procedures and plant activities including maintenance, modification, operational problems, and operational analysis.

d. Surveillance of selected plant operations and maintenance activities to provide independent verification* that they are performed correctly and that human errors are reduced to as low as reasonably achievable.

NSR AUTHORITY

6.5.2.6 NSR shall report to and advise the Vice President - Nuclear on those areas of responsibility specified in Sections 6.5.2.4 and 6.5.2.5.

RECORDS

6.5.2.7 Records of NSR activities shall be prepared and maintained. Reports of reviews and audits shall be distributed as follows:

a. Reports of reviews encompassed by Section 6.5.2.4.1 above, shall be prepared, approved and forwarded to the Vice President - Nuclear, within 14 days following completion of the review.

b. Audit reports encompassed by Section 6.5.2.4.2 above, shall be forwarded to the Vice President - Nuclear and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Programs required by Technical Specification 6.8 and other procedures which affect plant nuclear safety as

*Not responsible for sign-off function

determined by the General Manager - Hope Creek Operations, and changes thereto, other than editorial or typographical changes, shall receive an independent operability and technical review and be subjected to an independent USQ determination.

PROCEDURE RELATED DOCUMENTS

6.5.3.2 Procedures, Programs and changes thereto shall be reviewed as follows:

a. Each newly created procedure, program or change thereto shall be independently reviewed by an individual knowledgeable in the area affected other than the individual who prepared the procedure, program or procedure change, but who may be from the same organization as the individual/group which prepared the procedure or procedure change. Procedures other than Station Administrative procedures will be approved by the appropriate station Department Manager or by the Assistant General Manager - Hope Creek Operations. The General Manager - Hope Creek Operations shall approve Station Administrative Procedures, Security Plan implementing procedures, Emergency Plan implementing procedures, and Fire Protection Program implementing procedures.

b. On-the-spot changes to procedures which clearly do not change the intent of the approved procedures shall be approved by two members of the plant staff, at least one of whom holds a Senior Reactor Operator's License. For revisions to procedures which may involve a change in intent of the approved procedures, the person authorized above to approve the procedure, shall approve the revision.

c. Individuals responsible for reviews performed in accordance with item 6.5.3.2a above shall be members of the station staff previously approved by the SORC Chairman and designated as a Qualified Reviewer. A system of Qualified Reviewers shall be maintained by the SORC Chairman. Each review shall include a determination of whether or not additional cross-disciplinary review is necessary. If deemed necessary, such review shall be performed by the appropriate designated review personnel.

d. If the Department Manager determines that the documents involved contain significant safety issues, the documents shall be forwarded for SORC review and also to NSR for an independent review to determine whether or not an unreviewed safety question is involved. Pursuant to 10CFR50.59, NRC approval of items involving unreviewed safety questions or Technical Specification changes shall be obtained prior to implementation.

NON-PROCEDURE RELATED DOCUMENTS

6.5.3.3 Tests or experiments, changes to Technical Specifications, and changes to equipment or systems shall be reviewed in a manner similar to that described in items 6.5.3.2a, c, and d above with the exception that the recommendations for approval are made by SORC to the General Manager - Hope Creek Operations. Independent safety reviews for determination or verification of unreviewed safety questions will be performed by NSR and the results of NSR reviews will be provided to SORC. NSR reviews will be performed not only by using its own staff, but, when needed, also through the use of a system of qualified reviewers established throughout the corporate organization to support NSR. Pursuant to 10CFR50.59, NRC approval of items involving unreviewed safety questions or Technical Specification changes shall be obtained prior to implementation.

RECORDS

6.5.3.4 Written records of reviews performed in accordance with item 6.5.3.2a above, including recommendations for approval or disapproval, shall be maintained. Copies shall be provided to the General Manager - Hope Creek Operations, SORC, NSR, and/or NRC as necessary when their reviews are required.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Section 50.73 to 10CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the SORC and the resultant Licensee Event Report submitted to the NSR and the Vice President - Nuclear.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT STANDBY within one hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President - Nuclear and General manager - NSR shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the General Manager - Nuclear Safety Review and the Vice President - Nuclear within 14 days of the violation.

ATTACHMENT 6

Rev 1

ASB OPEN ITEM
IE BULLETIN 81-03

Hope Creek has been requested to address the applicability of IE Bulletin 81-03: Flow Blockage of Cooling Water to Safety Components by Corbicula sp. (Asiatic Clam) and Mytilus sp. (Mussel).

RESPONSE

Experience at the site has been shown that the referenced organisms are not indigenous to the local esturay. However, biofouling by similar species could potentially occur.

At Hope Creek, the only safety related heat exchangers which receive esturine water are the safety auxiliaries cooling system (SACS) heat exchangers. The balance of safety related heat exchangers are cooled with condensate quality water which is cooled on the shell side of the SACS heat exchangers.

Biofouling will be controlled by the continuous injection of sodium hypochlorite in front of the service water pumps. Should this control be temporarily disrupted, sodium hypochlorite can be injected at a higher rate to assure the cleanliness of the system.

Biofouling would be detected by monthly ~~measurement of~~ calculation of the heat transfer rate of differential pressure across the SACS heat exchangers. The heat exchangers will also be visually inspected during refueling outages. The SACS heat exchangers are tubed with 3/4 inch diameter titanium tubes. Titanium is not subject to erosion from contact or turbulent flow.

Since the service water system incorporates redundant equipment with piping cross ties, it would be possible to physically clean a SACS heat exchanger while operating.

Chlorine discharge for the service water system is not a concern since the service water system discharges to the closed loop circulating water systems. Blowdown from the circulating water system will be dechlorinated.

JES:vw

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