PSE&G Public Service Electric and Gas Company

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Robert L. Mittl General Manager
Nuclear Assurance and Regulation

September 13, 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, and NRC questions listed in Attachment 3.

In addition, enclosed for your review is revised FSAR Section 1.10 Item I.C.6 as requested by the Licensee Qualification Branch (Note: This supersedes the 8/24/84, Mittl to Schwencer letter, submittal of Section 1.10 Item I.C.6) (see Attachment 5), PSE&G's response to SIL No. 402 as requested by D. Wagner (see Attachment 6), and the responses to those open items, listed in Attachment 7, discussed with the Containment System Branch at the August 31, 1984 meeting (see Attachment 8).

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Also, enclosed (see Attachment 9), is supplementary information to FSAR Section 13.4. This information consists of proposed Technical Specifications and a document titled, "Streamlining of SORC Review Process."

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,

R J Whitt

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, NRC Questions, and 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 13th 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: 9/13/84

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

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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

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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 TO 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 · 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 8 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

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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	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	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

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO 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	8/1/84
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	8/1/84
252	8.3.3.5.6	The use of a single breaker to provide penetration protection	Complete	8/1/84
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	8/1/84
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 TO 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: 9/13/84

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)

CT:db

MP 84 95/03 01

Date 9/13/84

ATTACHMENT 3

OPEN ITEM	DSER SECTION	SUBJECT
5a,b,d	2.4.5	Wave impact and runup on service water intake structure
143b	9.1.5	Overhead heavy load handling
166	12.3.4.2	Airborne radioactivity monitor positioning
223	8.2.2.3	Independence of offsite circuits between the switchyard and class 1E buses
233	8.3.3.4.1	Periodic system testing
235	8.3.1.5	Diesel generators load acceptance test
241	8.3.1.10	Load acceptance test after prolonged no load operation of the diesel generator
242	8.3.2.1	Compliance with position 1 of Regulatory Guide 1.128
243	8.3.3.1.3	Protection or qualification of class 1E equipment from the effects of fire supression systems
244	8.3.3.3.1	Analysis and test to demonstrate adequacy of less than specified separation
247	8.3.3.5.1	Capability of penetrations to withstand long duration short circuits at less than maximum or worst case short circuits
253	8.3.3.1.4	Commitment to protect all class 1E equipment from external hazards versus only class 1E equipment in one division
256	8.3.2.3	Automatic trip of loads to maintain sufficient battery capacity
257	8.3.2.5	Justification for a 0 to 13 second load cycle
260	8.3.3.3.5	Use of a single breaker tripped by a LOCA signal used as an isolation device
261	8.3.3.3.6	Automatic transfer of loads and interconnection between redundant divisions
268		Air filtration unit drains

Question No.	FSAR Section
430.141	9.5.8
430.143	9.5.8
640.11	14.2

ATTACHMENT 4

DSER Open Item No. 5 a, b and d (DSER Section 2.4.5)

WAVE IMPACT AND RUNUP ON SERVICE WATER INTAKE STRUCTURE

The applicant has analyzed the wind waves that would traverse plant grade coincident with the PMH surge hydrograph and runup on safety-related facilities. These calculations were based on the assumption that wind waves would be generated in the Delaware Estuary and progress to the site. As the surge level would begin to rise, resulting from the approaching eye of the postulated hurricane, the wind speed would progressively change direction from the southeast clockwise to the west. Waves encroaching on the southern end of the Island would be depth-limited (i.e., the waves would "feel" bottom and thus become shallow water waves) by plant grade elevation on both the Salem and Hope Creek sites. These depth-limited (shallow water) waves will impact and runup on the southern and western faces of the safety-related structures in the power block. The applicant has stated that the southern face of the Reactor Building and the Auxiliary Building are designed for a flood protection level of 38.0 ft msl or 3.2 ft above the maximum calculated wave runup height of 34.8 ft msl and the other exposures of safety-related structures have a flood protection level of 32.0 ft msl or 1 ft above the maximum calculated wave runup height of 31.0 ft msl.

The staff has requested the applicant to provide additional information on the waves that impact on the river face of service water intake structure. The waves impacting on this face of the structure are not reduced in height (depth-limited) as those that traverse plant grade.

As indicated in Section 2.4.1, the applicant states that all accesses to safety-related structures (doors and hatches) are provided with water-tight seals designed to withstand the head of water associated with the flood protection levels. But, the applicant has not indicated whether the water-tight doors are designed to withstand either the combined loading effects of both static water level and the dynamic wave impact or, as cited in Sections 3.4.1 and 3.5.1.4 of this report, the impact of a barge propelled by winds and waves associated with a hydrologic event that floods plant grade.

Based upon its analysis according to SRP 2.4.5, the staff concludes that the flood protection level of El. 38.0 ft msl for the southern face of the Reactor Buidling and Auxiliary Building and El. 32.0 ft msl for the remaining safety-related structures within the power block meets the requirements of Regulatory Guide 1.59. Until additional information and analysis are available, the staff cannot conclude that the flood protection level of El. 32.0 ft msl for the Service Water Intake Structure meets the requirements of Regulatory Guide 1.59. Based on its analysis, the staff cannot conclude that the plant meets the requirements of GDC 2 with respect to the hydrologic aspects of Probable Maximum Surges and Seiche Flooding.

RESPONSE

The requested information for the service water intake structure has been provided in the responses to the following NRC questions:

Information Provided	Question No.
Wave runup elevations	240.8
Wave impact loads	240.9
Flood protection	240.8 and 410.69

As a result of discussions with the NRC staff, the response to Question 410.69 has been revised and the following summary calculations have been submitted under separate cover:

1. Analysis of overtopping of service water intake structure.

2. Runup on the east face of the service water intake structure.

Information on the ability of the doors and hatches to withstand the combined loading effects of static water level and the dynamic wave impact is provided in the response to FSAR Question 240.14.

QUESTION 410.69 (Section 9.2.1)

Provide a figure(s) in the FSAR which shows the protection of the station service water system from the flood water (including wave effects) of the design basis flood.

RESPONSE

The general arrangement of the intake structure is provided in Figures 1.2-40 and 1.2-41. Section AA of Figure 1.2-41 is reproduced here as Figure 410.69-1 which identifies the watertight areas and the walls and slabs designed to accommodate flood loads. As described in Sections 2.4.2 and 2.4.5, the south and west exterior walls of the intake structure are subject to a maximum wave run-up elevation of 134.4 feet due to the probable maximum hurricane (PMH). Such waves could overtop the roof of the western portion of the structure at elevation 128 feet. However, a rigorous analysis has been performed to determine the depth of water in the low area (elevation 122.0 feet) after wave impact and to confirm that water does not enter the building through the air intake control dampers (bottom elevation 128.5 feet). Therefore, flood water will not enter into the dry area of the intake structure. On the north side of the intake structure, the maximum water level will be only slightly higher than the still water elevation (113.8 feet) during the PMH. According to Table 2.4.6, the maximum wave elevation for the north side of the intake structure is 26.3 feet MSL (elevation 115.3 feet) due to a postulated multiple dam break. Therefore, flood protection of the north exterior wall to elevation 121.0 feet is adequate.

On the east side of the intake structure, the maximum wave run-up elevation due to the PMH equals 122.3 feet. This elevation is due to a 1% wave traveling in the direction of Fetch "A". Fetch A, which is rotated about 15 degrees from Fetch 1 (as shown in Figures 410.69-2 and 410.69-3), is chosen to maximize the wave run-up elevation. Elevation 122.3 feet exceeds the elevation of the bottom of the HVAC exhaust openings at elevation 122.0 feet by 0.3 feet. Curbs will be added at the bottom of these openings to prevent water from entering into the building.

In addition the following assessments have been made to confirm the adequacy of the structure and interior components for the overtopping wave:

a. The exterior walls are designed to withstand the flood loads including the dynamic wave action effects.

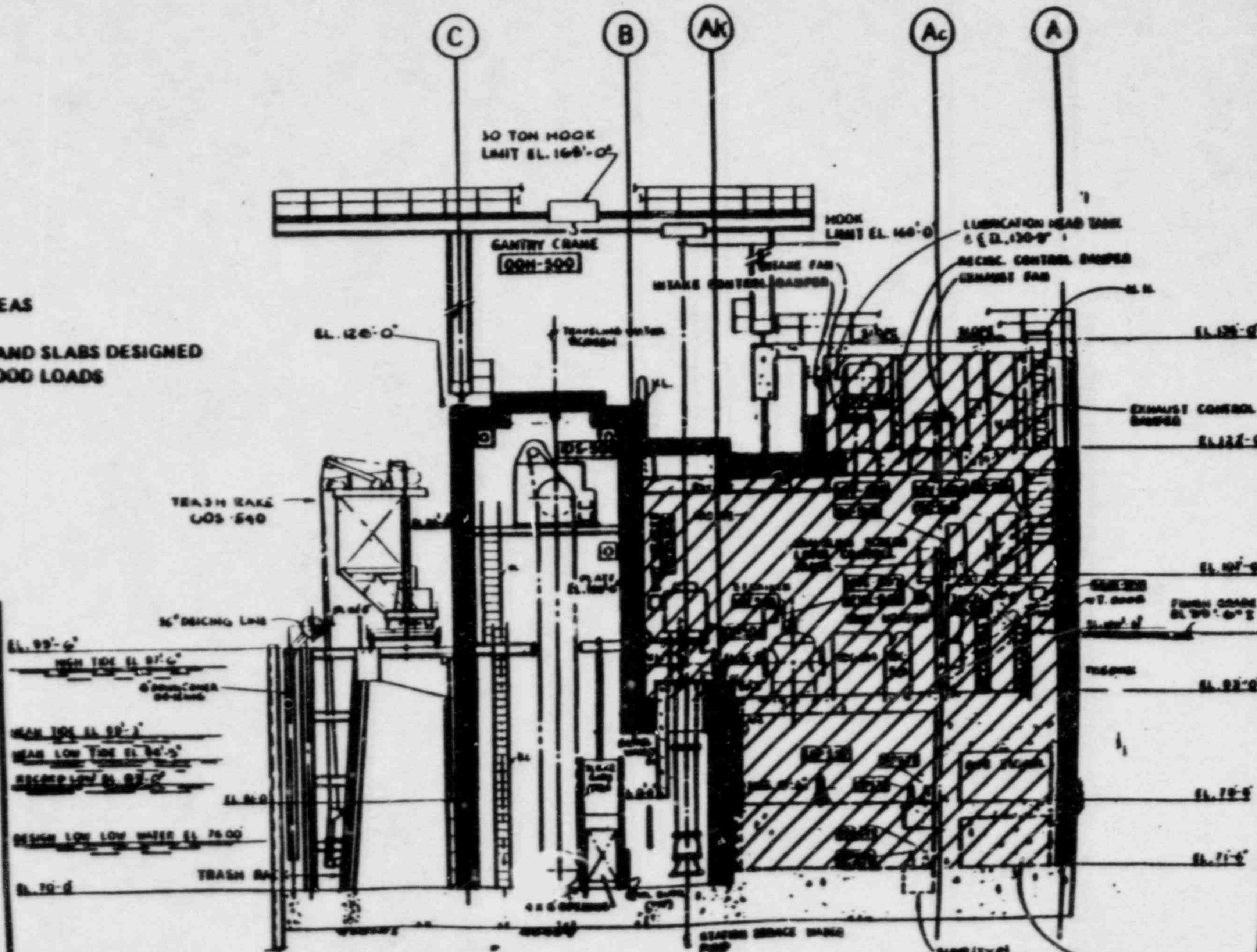
b. The roof hatches at both elevations 122.0 and 128.0 feet have been sealed (caulking, gaskets, etc.) to prevent any intrusion of water. The hatch covers are keyed into the openings to prevent any adverse slippage due to wave induced loadings.

c. All Seismic Category I components except for the traveling water screens are located within the dry areas of the structure.

d. The traveling water-screens, located in the "wet" area between column lines B and C have electric motors which are fully protected against the flood water level.

e. A condition was postulated where suspended moisture enters the dry areas of the structure through the air intake control dampers. It has been assessed that all of the Seismic Category I components subjected to this environment will continue to function as required.

Section 3.4.1 and Table 3.4-1 have been revised for clarification.



LEGEND:

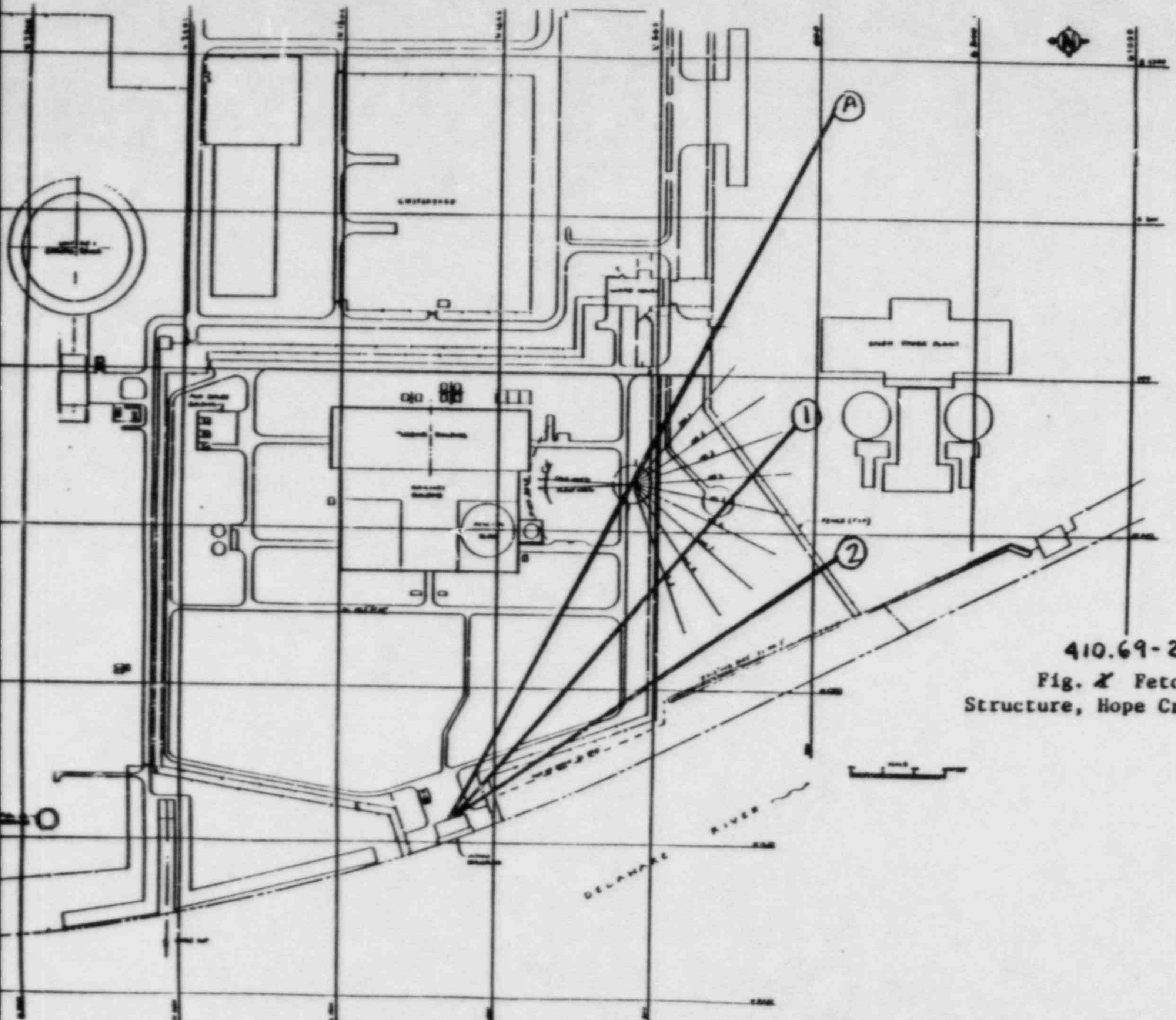
DRY AREAS

WALLS AND SLABS DESIGNED FOR FLOOD LOADS

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

SERVICE WATER INTAKE STRUCTURE — FLOOD PROTECTION

FIGURE 410.80-1



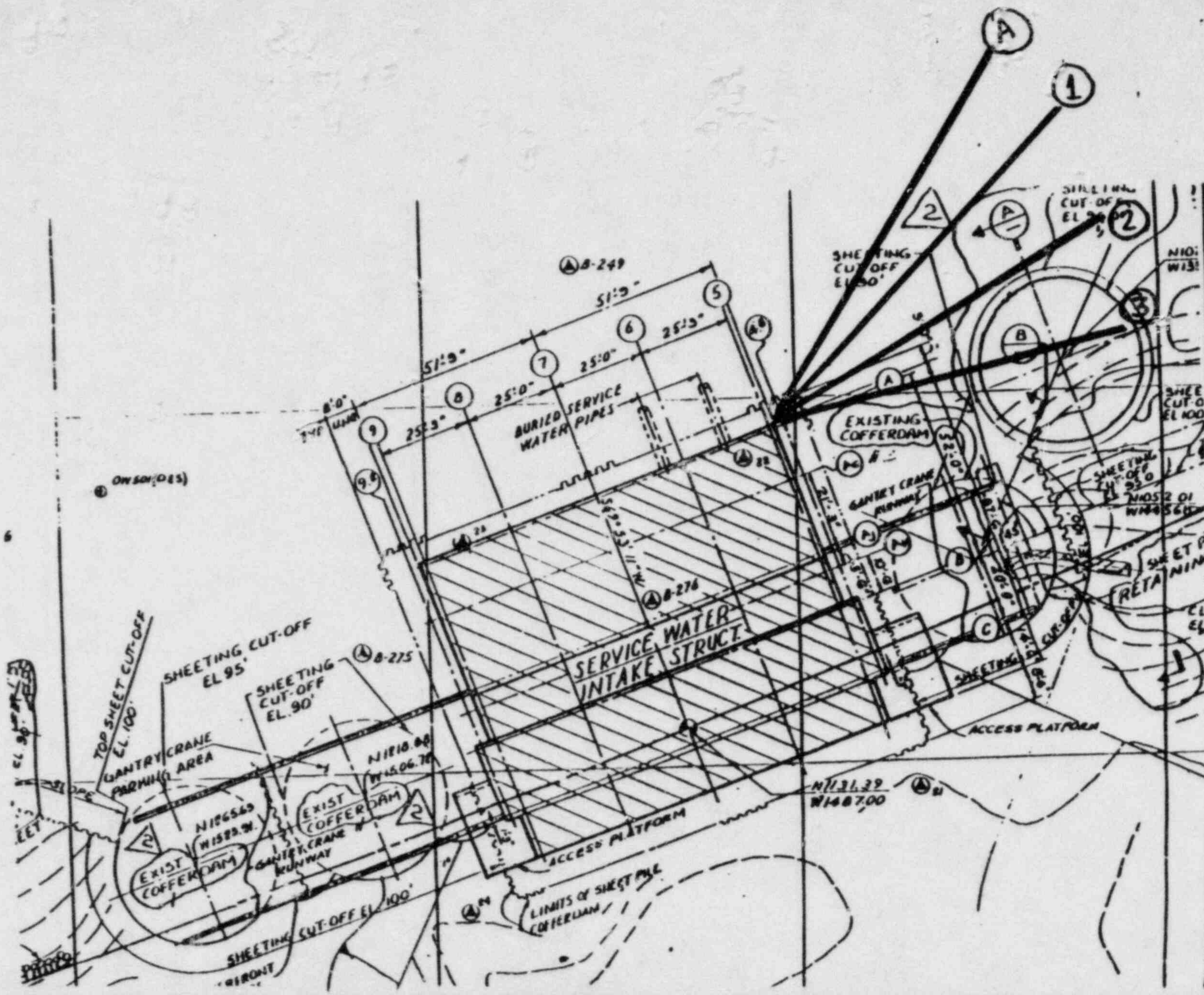
410.69-2

Fig. X Fetches for Service Water Intake Structure, Hope Creek Generating Station.

SEP 12 '84 0270976

410.69-3

Fig. 2. Fetches for Service Water Intake Structure, Hope Creek Generating Station



SEP 12 8402270976

August 6, 1984

Hope Creek Generating Station
Analysis of Overtopping of Service Water Intake Structure

I. **Wave Calculations**

- **Wave heights and periods as well as still-water levels and runup elevations are as given in Table 2.4-10a of FSAR (Amendment 5, April 1984).**

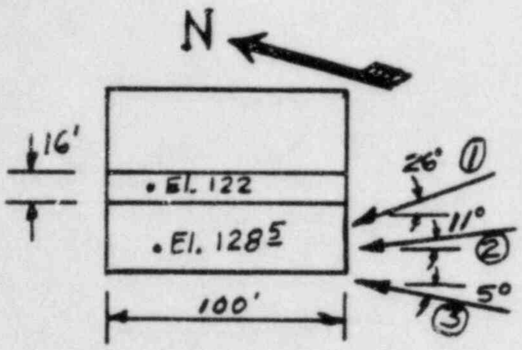
II. Overtopping Calculations

- Overtopping rates were calculated for west face and south face where top of wall elevations are 128.5 and 122.0, respectively.

- Equations from Weggel (1976) were used for the overtopping calculations.

$$Q = (g\, Q_0^* H_0'^3)\exp\left(-\frac{0.217}{2\alpha}\log_e\left(\frac{R+h-d_s}{R-h+d_s}\right)\right)$$

$$Q_0^* = \frac{\left(\frac{\epsilon}{2\pi}\right)^2\left(\frac{H}{H_0'}\right)^2\tanh\left(\frac{2\pi d_s}{L}\right)}{\frac{H_0'}{gT^2}}$$



- where ϵ was taken as $1/2\pi$ in order to maximize the value of Q_0^* (see Figure 6 of Weggel's paper)

- α was taken as 0.06 in order to maximize Q (see Equation 4 of Weggel's paper).

- Conservative assumptions in calculating overtopping rates were:

  - It was assumed that waves attacked normal to the wall of the structure.

  - It was assumed that the train of waves was made up of all 1% waves.

  - It was assumed that wave height was constant along the crest.

- Calculated overtopping rate was increased to allow for wind speed using Equation (7-11) of the 1977 edition of the U. S. Army Corps of Engineers Shore Protection Manual.

$$K' = 1.0 + W_f\left(\frac{h-d_s}{R} + 0.1\right)\sin\theta$$

- In making the wind adjustment the factor W_f was assumed to be 2.0 for onshore winds greater than 60 mph. The angle 0 was 90°.

o After adjustment for wind the overtopping rates were adjusted for angle of attack by multiplying the overtopping rate by the sin of the angle between the fetch vector and the wall.

III. Maximum water surface elevations were calculated by backwater calculation starting from the north end of the roof.

o The separate overtopping rates were added and the total was assumed to flow off the top of the structure at the north end.

o Critical depth was assumed to occur at the downstream end of the channel and was calculated as:

$$y_c = \left[\frac{(Q_{TOT}/16)^2}{32.2}\right]^{1/3}$$

where Q is the rate of flow from the west side in cfs/ft.

o The backwater calculation assumes a gradually varied steady flow.

$$y_{x+\Delta x} = \sqrt{\frac{2\Delta Q \cdot \Delta x \cdot Q_x}{16 \cdot 32.2 \cdot y_x} + y_x^2}$$

o Calculations were performed moving upstream starting with the depth at the north end.

o The calculations showed that fetch 3 was the critical case. The total flow rate for fetch 3 was 0.5 cfs/ft from the west and 14.7 cfs/ft from the south end.

o The maximum water surface elevation reached was 126.9 for the fetch 3 condition which is well below the critical 128.5 elevation at which flow could enter the air intakes.

IV. A separate calculation was made considering a surge generated by flow coming over the south end of the building. The depth of flow and velocity of flow ahead of the surge resulting from the previous surge had to be assumed. Velocity ahead of the surge was assumed to be zero, since that condition maximizes the surge height. Depth ahead of the surge was assumed to be 1.0' and does not have a really significant affect on the height of the following surge. The resulting elevation of the crest of the generated surge was 126.9 which is below the 128.5 elevation at which water can flow into the air intake.

V. A check was made to see if flow could surge into the air intakes as a result of plunging from the roof at elevation 128.5.

- Loss coefficients of 0.5 at the entrance to the air intake opening and 0.5 at the band (see attached sketch were assumed).

- Velocity at the edge of the 128.5 elevation roof section was calculated assuming critical depth there and was increased by 50% for reasons of conservancy.

- The velocity approaching the entrance to the air intake chamber was calculated using the energy equation and neglecting losses.

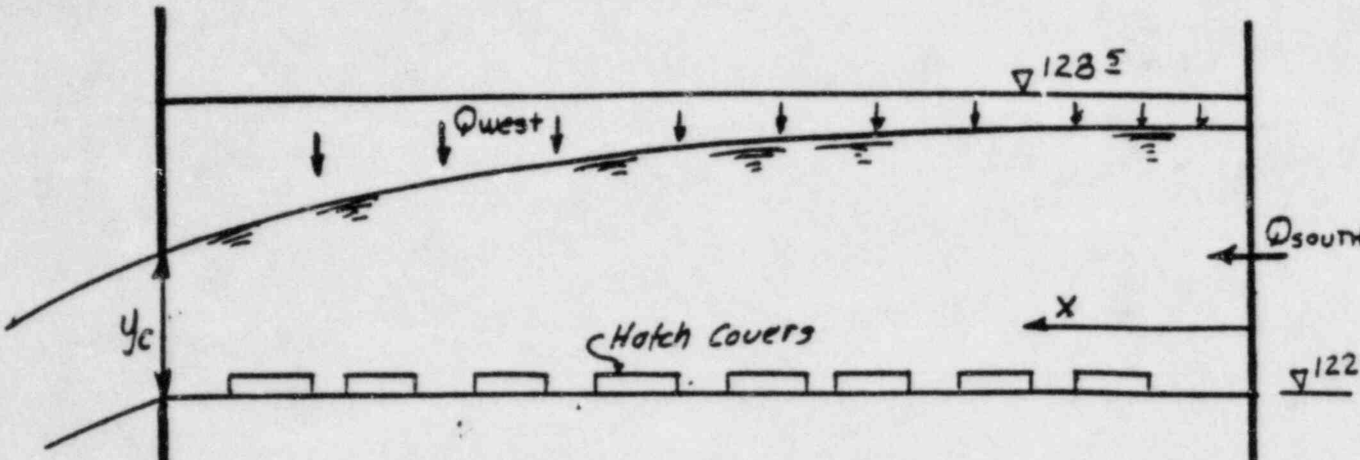
- Losses incurred by turbulence and impact of the jet entering water ponded on top of elevation 122.0 were neglected.

- Headloss through the screens was neglected.

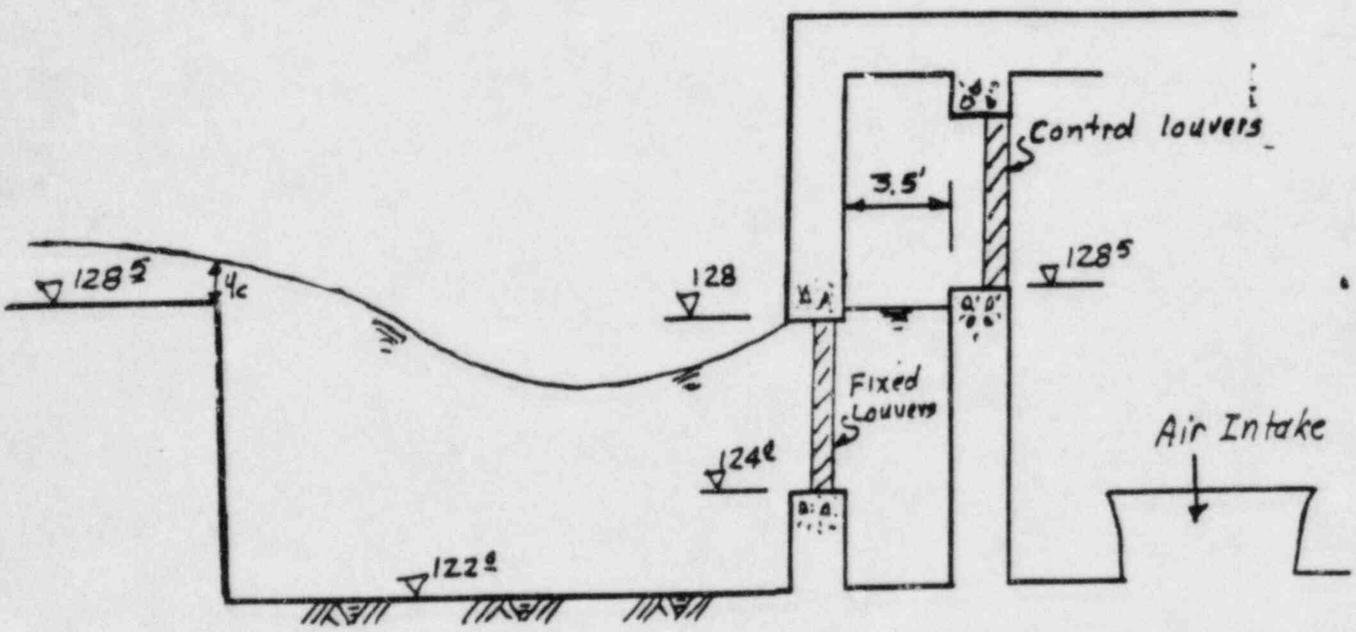
- The maximum elevation achieved was calculated to be 126.3 or well below the 128.5 elevation at which water could flow into the building.

- A separate analysis was made using a one-dimensional momentum approach. The presence of the louver on top of the outer wall was neglected. A velocity of 26 feet per second was assumed to occur over the top of the lower outer wall whose top elevation is at 124.0. This velocity was calculated assuming that the total potential energy in a wave runup to 134.4 would be converted to kinetic energy at elevation 124 without energy loss. The one-dimensional energy analysis, assuming a flow rate of 5.75 cfs/foot indicates that the water surface within the intake could rise to elevation 127.0 which is below the 128.5 elevation at which water could flow into the service-water intake structure. The assumption of a flow rate of 5.75 cfs/foot is very conservative since that is the total overtopping rate from the west side of the structure for the critical fetch conditions assuming the wave strikes normal to the structure wall.

- The total pressure of the air intake fans equals 4.5 inches of water. The maximum elevations of 126.3 feet and 127.0 feet given above result in margins of 2.2 and 1.5 feet respectively with respect to the 128.5 feet elevation at which water could flow into the building. Therefore, there is sufficient margin to accommodate a rise in water level due to fan suction pressure.



Flow along center of building



Sketch of flow conditions at entrance to air intakes

References

1. Weggel, J. R., "Wave Overtopping Equation" Proceedings of the 1976 Coastal Engineering Conference.

2. Jackowski, R. A. (Editor) Shore Protection Manual, U. S. Army Corps of Engineers, Coastal Engineering Research Center, 1977.

Calculation Summary
Runup on the East Face of the
Service Water Intake Structure
Hope Creek Generating Station

The attached Figure 1 shows the fetches considered for wave runup on the service water intake structure. Fetch A which has a direction of N58°W, is 4800 feet long and passes between the Salem Plant and the Hope Creek Generating Plant.

Waves approaching the Service Water Intake Structure would be tripped by passage over the dike at the edge of the island. The top of this dike is at elevation 108 (PSE&G Datum).

Wave heights, still water levels, and wave lengths are assumed as given in Table 2.4-10A of the FSAR. For Fetch A conditions we have assumed that the incident wave characteristics, still water level, and wind speed are the same as for Fetch 1. Thus, the incident wave (maximum wave) has a height of 15.8 ft., period of 6.4 seconds, and a length of 180 feet. The corresponding wind speed is 108.6 mph and the still water level is 112.1 feet (PSE&G Datum).

In accordance with the results presented in Reference 1, the dike will trip all large waves and it is reasonable to assume that a significant wave height of 0.4 d will be transmitted over the dike and over Fetch A.

The bottom elevation is 101 ft. (PSE&G Datum) which makes the depth equal to 11.1 ft. (112.1 - 101.0). Thus, the initial significant wave height to be propagated along Fetch A is 0.4 * 11.1 = 4.4 ft.

Energy will be added by wind shear along Fetch A. The energy addition was computed in accordance with Figures 3-24 and 3-25 of Reference 2 which give a dH/dx equal to 0.00014. For an additional fetch of 4800 ft. a total gain in wave height would be 0.65 ft. (0.00014*4800) due to wind shear.

Energy dissipation was estimated on the basis of Figure 3-34 of Reference 2. A friction coefficient of 0.04 was assumed for the conditions along Fetch A which will exist once the plant goes into operation. This assumption gives a K_f value of 0.54. Thus the propagated significant wave height would be:

$H_s = 4.44*0.54 + 0.65 = 3.05$ ft.
Converting to a 1% wave gives:
$H_{1.0} = 1.67*3.05 = 5.1$ ft.

To determine the runup of this wave on the east face of the service water intake structure a runup coefficient of 2.0 was chosen in accordance with the results presented in Reference 3 and shown in Figure 2.

Thus, the runup would be R = 2*5.1 = 10.2 ft. and the runup elevation would be 112.1 + 10.2 = 122.3 ft.

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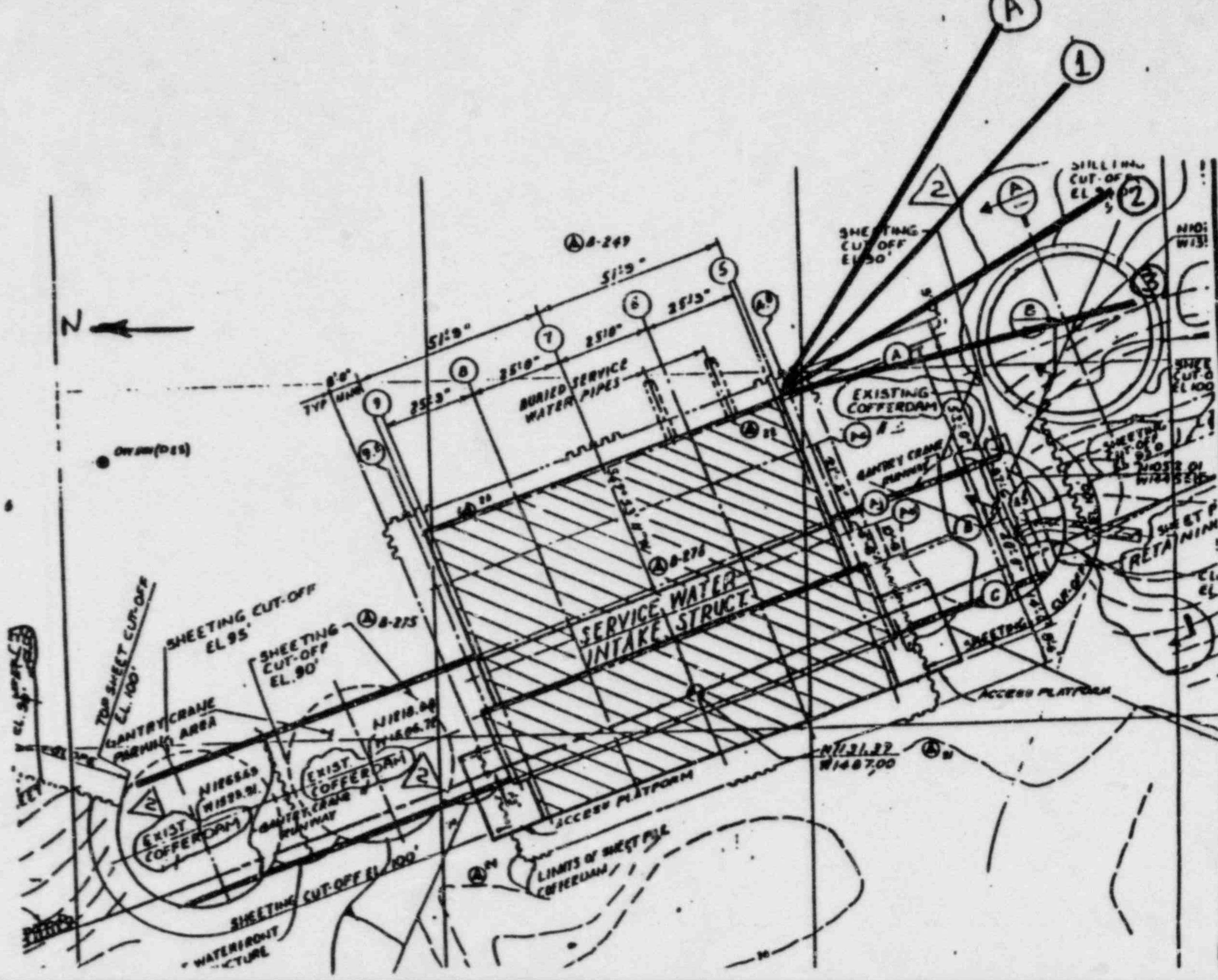
REFERENCES

1. Dally, W. R., A Numerical Model for Beach Profile Evolution, M. S. Thesis, University of Delaware, May 1980.

2. U. S. Army Corps of Engineers, Shore Protection Manual, Coastal Engineering Research Center, Fort Belvoir, Virginia, 3rd Edition, 1977.

3. Losada, M. A., and L. A. Gimenez - Curto, "Mound Breakwaters Under Wave Attack", Proceedings of the International Seminar on Criteria For Design and Construction of Breakwaters and Coastal Structures, Department of the Oceanographical and Ports Engineering of the University of Santander, Spain, 1980, p. 127-238.

Fig. 1   Fetches for Service Water Intake Structure, Hope Creek Generating Station



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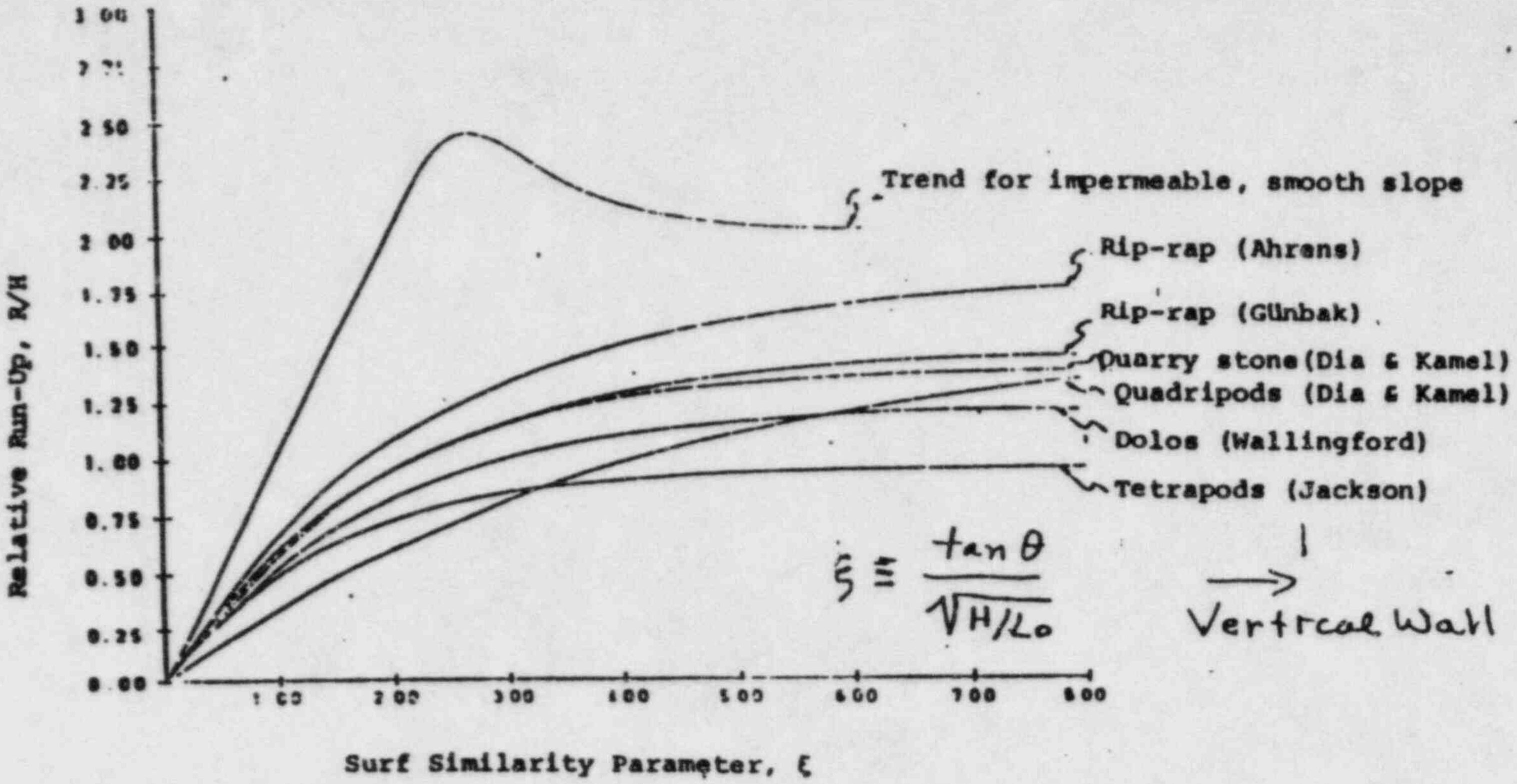

Fig. 2 Relative Run-Up Versus ξ For Various Breakwater Armor Units. (From Losada and Gimenez-Curto, 1980).

DSER OPEN ITEM No. 143b(DSER Section 9.1.5)

OVERHEAD HEAVY LOAD HANDLING

We cannot conclude that the overhead heavy load handling systems are in compliance with the Phase I and Phase II criteria contained in NUREG-0612 until the applicant provides an acceptable response to the guidelines. The overhead heavy load handling systems do not meet the acceptance criteria of SRP Section 9.1.5. We will report resolution of this item in a supplement to this SER.

Guideline 2.3.1-Reactor Building [NUREG-0612, Article 5.1.4]

(Reference: DSER Appendix B, Section 2.3.1)

Recommendation: Provide for review the analyses for lifting the heavy loads on the refueling floor by the lifting devices that are not single-failure-proof.

RESPONSE

FSAR Section 9.1.5 has been revised to provide the analyses for lifting the heavy loads on the refueling floor by the non-single-failure-proof lifting devices. As discussed in the telecon of May 30, 1984, between the applicant and the NRC, the analyses provided for the non-single-failure-proof lifting devices explicitly address the four evaluation criteria of NUREG-0612, Section 5.1.

The following revisions to FSAR Section 9.1.5 have been made in response to this recommendation:

1. The following lifting devices and lift points will be upgraded to satisfy the single-failure-proof guidelines of NUREG-0612, Section 5.1.6:

   - RPV head strongback
   - Dryer and separator sling
   - RPV service platform sling
   - RPV service platform lift points
   - RPV head lift points
   - Moisture separator lift points
   - Steam dryer lift points

   The design upgrade will be completed prior to fuel load. The text and tables of FSAR Section 9.1.5 have been revised to reflect this upgrade.

2. Table 9.1-19 has been added to provide a listing of the single-failure-proof heavy load lifting devices and the associated heavy load lift points. The applicable criteria of Section 5.1.6 of NUREG-0612 are referenced in Table 9.1-19.

DSER Open Item No. 143 (Cont'd)

Guideline 2.3.2 - Other Areas [NUREG-0612, Article 5.1.5]

(Reference: DSER Appendix B, Section 2.3.2)

Recommendation a. Provide equipment layout drawings with safety-related equipment and load-target areas marked on the drawings.

RESPONSE

The safe load path drawings (revised Figures 9.1-32 through 9.1-37), together with the information in revised Table 9.1-12, identify the safety-related equipment beneath the load path of each heavy load handled by the non-exempt cranes and hoists listed in revised Table 9.1-10. The load-target areas correspond to the cross-hatched load path areas shown in Figures 9.1-32 through 9.1-37. Furthermore, Table 9.1-10 provides the FSAR figure number of the plant equipment location drawing and the area on that drawing, defined by building column lines, below each crane or hoist to supplement the load path drawing/Table 9.1-10 information. Therefore, because the load-target areas are already in the FSAR, no new equipment location drawings are provided.

The "precise identification of each safety-related equipment in Table 2.2" requested in item C.1 on page 18 of DSER Appendix B is provided in revised Table 9.1-12. As agreed in the May 30, 1984 conference call between the applicant and the NRC, Table 9.1-12 has been revised to provide more precise equipment identification by listing each equipment item beneath each load path on a separate line rather than in series on the same line.

Additional information to "define the load impact area for each postulated load drop" as requested in item C.1 on page 18 of DSER Appendix B is not provided based on the May 30, 1984 NRC telephone clarification referenced above. The necessary impact area information is already provided in Table 9.1-10 and Figures 9.1-32 through 9.1-37.

Recommendation b. Provide evaluation for crane Nos. 15, 16, and 17.

RESPONSE

As described in the response to Guideline 1.b on Page 22 of DSER Appendix A, hoists number 15 (CRD Service Hoist), 16 (SACS Pumps

Hoist), and 17 (SACS Heat Exchanger Hoists) are classified as non-exempt hoists. They are shown in revised Table 9.1-12. The safety evaluation for these three hoists is provided in revised Sections 9.1.5.3.3.h, 9.1.5.3.3.11, 9.1.5.3.3.mm and Table 9.1-12.

Recommendation c. If an alternative to the NUREG-0612 criterion is used, provide details to demonstrate that the alternative criterion is consistent with the intent of the NUREG-0612 requirements.

RESPONSE

As noted above in the response to Guideline Recommendation 2.3.1 for the polar crane, hazard elimination criterion "f" has been deleted from Table 9.1-12. Because all of the other hazard elimination criteria used in the table are obtained from Enclosure 3 to Reference 4 of DSER Appendix B, no alternatives to the published NRC criteria are used in revised Table 9.1-12.

Guideline 2.3.3 - Single-Failure-Proof Handling Systems
[NUREG - 0612, Article 5.1.6]

(Reference: DSER Appendix B, Section 2.3.3)

Recommendation a. Provide additional information to demonstrate that every guideline of NUREG-0554 related to the polar crane is satisfied.

RESPONSE

The additional information is provided in revised Table 9.1-13.

Recommendation: b. List all the single-failure proof lifting devices and the associated lift points. Provide additional details to substantiate the single-failure-proof status of the items that are not adequately addressed.

RESPONSE

As discussed above in the response to Section 5.1.4, Table 9.1-19 has been added to provide a listing of the single-failure-proof heavy load lifting devices and the associated heavy load lift points. The applicable criteria of Section 5.1.6 of NUREG-0612 are referenced in the table. When calculating the design factors of safety for determining compliance with NUREG-0612, the combined maximum dynamic and static loads were used. As shown in Table 9.1-19 several lifting devices and lift points will be upgraded to be single-failure-proof. The design details are not yet known, but the lifting devices and lift points will satisfy the single-failure-proof guidelines of NUREG-0612, Section 5.1.6.

The design of the HCGS RPV head strongback and the HCGS dryer separator sling is the same as the design of the corresponding WPPSS - Nuclear Plant No. 2 and Limerick Generating Station special lifting devices. The planned upgrade of these two HCGS devices includes as a minimum the modifications of these lifting devices that were agreed to by the NRC for the WPPSS - Nuclear Plant No. 2.

CHAPTER 9

TABLES

valve operator from one of the control valves in the feed lines to the offgas recombiners, carries it to the hatch in the valve cell, and lowers it to a maintenance cart in the the access corridor at elevation 54 feet. Each valve operator weighs 943 pounds.

9.1.5.3 Safety Evaluation

All of the OHLHS cranes are evaluated in Table 9.1-10 with respect to whether they carry heavy loads over safety-related equipment located under the load path or on the next lower elevation. Table 9.1-10 excludes from further evaluation those OHLHS cranes that have no safety-related equipment below their load paths or only handle loads lighter than 1200 pounds although their design capacity is greater.

Those OHLHS cranes not excluded in Table 9.1-10 are listed in Table 9.1-12 along with the loads they carry, the lifting device, if any, for each load, and the safety-related equipment beneath the load path. Hazard elimination criteria are applied to each load handling situation identified in Table 9.1-12 to determine if it can be excluded from further evaluation. All equipment hatch load handling situations are dealt with in compliance with the guidelines of NUREG-0612.

Application of the NUREG-0612 guidelines, the exclusion criteria in Table 9.1-10, and the hazard elimination criteria in Table 9.1-12 show that there are no remaining OHLHS for which heavy load drops might prevent safe shutdown or decay heat removal, cause unacceptable radioactivity release, or expose spent fuel. The safe load paths for the OHLHS load situations in Table 9.1-12 are presented on Figures 9.1-32 through 9.1-3~~8~~ 7.

INSERT 1

9.1.5.3.1 Reactor Building Polar Crane

Figure 9.1-32 shows the load paths for this crane. The reactor building polar crane is the only one of the OHLHS cranes, that is physically capable of carrying heavy loads over irradiated fuel. Both the main and auxiliary hoists are single-failure proof. Trolley and bridge travel limit switches, plus a set of bridge stops on the rail and main trolley stops near the middle of the bridge, together ensure that the main hoist cannot travel over the fuel pool. Figure 9.1-31 shows the main hook exclusion area. The cask loading pit is outside the exclusion area and separate from the spent fuel pool. The spent fuel cask, therefore, can

(Ref. 9.1.5.3) INSERT 1

Additional drawings showing plan and elevation views of the reactor building and other areas are provided in the general arrangement drawings and equipment location drawings provided in FSAR Section 1.2.

(Ref. 9.1.5.3.1) INSERT 2

Table 9.1-15 presents a failure mode and effects analysis for the reactor building polar crane.

not accidentally drop into the spent fuel pool. The cask is moved directly between the hatch, the cask washdown area, and the cask loading pit on the refueling floor, as shown on the load path drawing, Figure 9.1-32.

Some safety features of the polar crane design are discussed in Section 9.1.5.2.1. In addition, the crane is designed to Seismic Category I criteria so that either hoist will retain its load during and after a SSE. Manually engaged anti-derail devices on both trolleys secure the trolleys when not in use and prevent rolling during an earthquake. Flat plate earthquake restraints welded onto the bottom of the girder end ties transfer the seismic loads to the reactor building wall through the crane rail.

The single-failure proof aspects of the polar crane design include complete redundancy for the sheaves, ropes, reeving, reducing gears, holding brakes, and other load path components of both the main and auxiliary hoists.

Figure 9.1-30 illustrates the single-failure proof auxiliary hoist design. The load is supported by the hook and two shackles, one on either side of the hook. The two separate load paths from the hook and shackles extend through the four side plates up to two separate sheave pins. Each of the two plates on either side of the load block is designed to support the design load. The trunnion applies the hook load to all 4 plates. Each shackle applies the hook load to the two side plates on its side. The side plates transmit the load to the two sheave pins. Each pin holds a sheave that is reeved independently. The block housing includes two through-bars that are designed to catch the wire ropes and/or sheaves if a sheave or sheave pin fails. Each sheave is independently reeved to the hoist drum, where the ropes are dead-ended to the drum.

Table 9.1-13 presents a point-by-point comparison of the reactor building polar crane design with the criteria of NUREG-0554, Single-Failure Proof Cranes for Nuclear Power Plants.

INSERT 2

9.1.5.3.2 Reactor Building Polar Crane Lifting Devices

Lifting devices used by the polar crane are listed in Table 9.1-12. The special ~~lifting~~ devices, as defined by NUREG-0612, are listed in Table 9.1-14 along with the ~~status of compliance with ANSI N14.6-1978 and the~~ design safety factors

DELETE SECTION 9.1.5.3.2
AND REPLACE WITH
INSERT 3

A single-failure proof spent fuel shipping cask lifting device and cask lift point design in accordance with the requirements of NUREG-0612 will be selected for HCGS.

A single-failure proof conventional sling selected in accordance with NUREG-0612, Section 5.1.6(1) is used to lift the fuel pool gates. The fuel pool gates are the only heavy loads which must routinely be carried over the fuel pool. There are two lift points on each fuel pool gate. They are designed with a minimum static factor of safety of 20 with respect to material ultimate strength. This satisfies the NUREG-0612, Section 5.1.6 requirement for a safety factor of 5.

The fuel pool slot plug sling is a single-failure proof ~~special lifting device designed~~ conventional sling selected to meet the requirements of NUREG-0612, Section 5.1.6. Each fuel pool slot plug has a single lifting point designed with a minimum static factor of safety of 20 with respect to material ultimate strength. This satisfies the NUREG-0612, Section 5.1.6 requirement for a safety factor of 10.

Although the special lifting device for the dryer-separator pool plugs is single-failure proof, the lift points are not. The dryer-separator pool plugs each have four lift points designed with a minimum static factor of safety of 10 with respect to material yield strength. Although not in strict compliance with NUREG-0612, Paragraph 5.1.6(3)(a), which requires redundant points, each having a design safety factor with respect to ultimate strength of five times the maximum combined concurrent static and dynamic load, the design is conservative and satisfies the intent of NUREG-0612.

The special lifting device for the reactor well shield plugs is single-failure proof in accordance with NUREG-0612, Section 5.1.6, but the lift points are not. Each shield plug has four lift points to prevent uncontrolled lowering of the load, assuming a single lift point failure. Each lift point has a static design safety factor of 5 with respect to yield strength. Although not in strict compliance with NUREG-0612, Paragraph 5.1.6(3)(a), the design is conservative and statisfies the intent of NUREG-0612.

The dryer-separator pool plugs and reactor well shield plugs discussed above are not carried over the fuel pool, but are carried over the reactor vessel. They are only carried over the reactor vessel when both the drywell head and the RPV head are in place. A shield plug drop will not damage fuel or cause

INSERT 4

The fuel rack lifting fixture will be used for several non-routine heavy load lifts over the fuel pool. It is used for installing the spent fuel rack modules. As described in Section 9.1.2.2.2.2, a base capacity of 1078 spent fuel cells plus 30 multipurpose cavities will be installed for initial plant operation. The remaining capacity of 17 rack modules, providing an additional 2976 cells, will be installed during plant operation. The lifting fixture design factors of safety versus yield and ultimate strengths are provided in Table 9.1-14. These factors meet the criteria of paragraph 5.1.6(1)(a) of NUREG-0612 for a single-failure-proof single load path special lifting device.

The lifting eye of the fixture is connected to the crane hook by a sling arrangement. The slings are selected to meet the single-failure-proof criteria of Section 5.1.6(1)(b) of NUREG-0612. The four legs of the fixture each have a J-shaped plate at the bottom. The fixture legs are lowered through four of the empty cells of the rack module being lifted, moved horizontally a short distance, and raised to hook to the module base. The four J-shaped plates contact the underside of the module base when it is being lifted. This design eliminates the need for lifting eyes on the module. The weight of the module, together with the shape of the lifting fixture plates, provides assurance that the fixture is securely attached to the module during lifting.

Thus, because there are no lift points on the modules, and both the crane and lifting fixture are single-failure-proof, the modules will be installed with a single-failure-proof handling system.

The modules will be lifted with the main hoist of the polar crane. Limit switches and travel stops, described in Section 9.1.5.2.1.5, will be removed as necessary to permit the main hoist to travel into the main hook exclusion area shown on Figure 9.1-31 when the modules are installed.

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unacceptable water leakage from the reactor. This conclusion is based on the assumption that a plug drop could damage the drywell head and seal plate, but would have a less severe impact than a drywell or RPV head drop. In the highly unlikely event of a plug drop, the consequences would satisfy the four evaluation criteria of NUREG-0612, Section 5.1.

The drywell head is lifted by the RPV head strongback. It is carried over the reactor vessel while the RPV head is in place. A drywell head drop will not damage fuel or cause unacceptable water leakage from the reactor. This conclusion is based on the assumption that a drywell head drop would be less severe than a RPV head drop. Depending on orientation, a drywell head drop could damage the insulation support structure, rupture the RPV vent and head spray piping, damage the seal plate, and hit the RPV itself. But because the drywell head weighs about 2/3 as much as the RPV head, and because some of its kinetic energy would be absorbed by the insulation support structure and head piping before it strikes the RPV head, which is still in place, a drywell head drop would not cause fuel damage or unacceptable water leakage. In the highly unlikely event of a drywell head drop, the consequences would satisfy the four evaluation criteria of NUREG-0612, Section 5.1.

The RPV head strongback lifts the RPV head. The strongback design satisfies the guidelines of ANSI N14.6-1978, Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kilograms) or More for Nuclear Materials, in general. However, it does not explicitly comply as recommended by NUREG-0612, Section 5.1.1(4). Further, the design satisfies the minimum design safety factor of 5 with respect to the material ultimate strength requirement of Section 5.1.1(4), but not the single-failure proof criterion of Section 5.1.6(1)(a) for a design safety factor of 10.

Because the strongback is not single-failure proof, an RPV head drop onto the open reactor vessel has been analyzed. Results show that vessel and core integrity would be maintained within the guidelines criteria of NUREG-0612, Section 5.1. The effects would be less severe than those due to the fuel handling accident analyzed in Chapter 15. Damage to the vessel would not be severe enough to cause water leakage that uncovers the fuel.

The dryer-separator sling lifts the steam dryer and the moisture separator. The sling design satisfies the guidelines of ANSI N14.6-1978 in general, but does not explicitly comply as recommended by NUREG-0612, Section 5.1.1(4). The design ~~also~~ factors of safety versus yield and ultimate strengths are provided in Table 9.1-14. They are less than the values of 3 versus yield and 5 versus ultimate required by Section 5.1.1(4).

~~satisfies the safety factor of 5 requirement of Section 5.1.1(4), but not the single-failure proof requirement of 5.1.6(1)(a) for a safety factor of 10.~~

Because the sling is not single-failure proof, both a dryer drop and a separator drop have been analyzed. Results show that vessel and core integrity would be maintained within the guideline criteria of NUREG-0612, Section 5.1. Damage to the reactor vessel would not be severe enough to cause water leakage that uncovers the fuel.

The service platform sling lifts the RPV service platform. The sling design satisfies the guidelines of ANSI N14.6-1978 in general, but does not explicitly comply as recommended by NUREG-0612, Section 5.1.1(4). ~~Also, the design satisfies the safety factor of requirement of Section 5.1.1(4), but not the single-failure proof requirement of Section 5.1.6(1)(a) for a safety factor of 10.~~

Because the service platform sling is not single-failure proof, a service platform drop has been analyzed. Results show that vessel and core integrity would be maintained within the guideline criteria of NUREG-0612, Section 5.1.

The fuel pool jib cranes are carried over the reactor vessel when the RPV head is off, but only when the RPV service platform is in place on the RPV flange. A jib crane drop could damage fuel if it managed to cause structural failure of the service platform. A conventional sling, selected in accordance with NUREG-0612, Paragraph 5.1.6(1)(b)(ii), is used to lift the jib crane. The load used to select the sling is two times the sum of the maximum static plus dynamic load. The dynamic load is assumed to be 0.25W, where W equals the weight of the jib crane. The load used is, therefore, 2(W+0.25W). The jib crane design has a single lift point with a design safety factor of 10 times the maximum combined concurrent static and dynamic load with respect to material ultimate strength as required by NUREG-0612, Paragraph 5.1.6(3)(b). The jib crane handling system, therefore, meets the single-failure proof criteria of NUREG-0612, Section 5.1.6.

No other heavy loads will be carried over the open reactor vessel.

(Ref. p.9.1-90) INSERT 5

~~The design factors of safety versus yield and ultimate strengths are provided in Table 9.1-14. The factor versus yield is greater than the value of 3, and the factor versus ultimate is less than the value of 5 required by Section 5.1.1(4) of NUREG-0612.~~

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The RPV Head insulation and its support structure is carried over the RPV when the head is on. It is lifted by ~~slings selected to meet the single failure proof criteria of NUREG-0612, Section 5.1.6(1)~~ the RPV head strongback. The support structure is lifted in two pieces. The lift points on each piece are designed to meet the single-failure proof criteria of NUREG-0612, Section 5.1.6(3)(a).

The other heavy loads carried over the RPV while the head is on are the RPV stud tensioner and the RPV head stud rack. They will not cause fuel damage or unacceptable leakage because the drop would be less severe than a drywell or RPV head drop.

All heavy loads that need not be carried over the reactor well are restricted from this area during refueling. Administrative procedures help to control safe movement of all heavy loads.

In summary, a load drop into the reactor well could not affect safe shutdown capability since the well is only open when the reactor is shut down. Decay heat removal capability could be threatened only by a load large enough to damage the seal plate. Failure of the seal plate would not allow the large, heavy loads to fall into the drywell because their size is greater than the space between the RPV and the drywell. The reactor well and the drywell are lined with steel plate which will retain any concrete which is fragmented by swinging or falling loads. It is doubtful that other debris large enough to damage shutdown cooling piping could fall through the labyrinth of intervening piping and structural steel, including the massive primary containment radial box beams. The RHR shutdown cooling subsystem described in Section 5.4.7 includes a single suction line from reactor reciculation loop B. Therefore, a load drop into the reactor well could disable the shutdown cooling function of the RHR system. The design basis for this event is that any debris that managed to fall and disable RHR shutdown cooling would not have enough residual energy when it reached the components of this subsystem to do sufficient damage to prevent manual restoration of the cooling function. Damage such as a severed or crimped pipe, or complete loss of function of a suction line valve operator is not considered credible. Shutdown cooling would be manually restored as described in Section 5.4.7.1.5. If manual restoration cannot be achieved, an alternate flow path as described in Section 15.2.9 could be used. Similarly, if debris from the load drop were able to cause leakage from exposed reactor vessel piping, makeup water could be supplied by any of a number of RHR and core spray injection lines until the leak could be repaired. Therefore, the drop of a heavy load into the reactor well would not affect decay heat removal capability.

The flux monitor shipping crate is carried over the refueling floor by slings selected to meet the single-failure proof guidelines of NUREG-0612, Paragraph 5.1.6(1)(b).

Heavy loads carried over the refueling floor that employ lifting devices or lift points that are not single-failure proof weigh up to 107.5 tons.

These loads include the items listed below and are also tabulated, with their weights, in Table 9.1-12.

a. RPV head

b. Drywell head

c. Reactor well plugs-curved, 4

d. Reactor well plugs-straight, 2

e. Dryer separator pool plug-curved

f. Dryer separator pool plugs-straight, 3

g. RPV service platform

h. RPV stud tensioner

i. RPV head stud rack

The RPV and drywell heads each have four lift points. The drywell head lift points meet the single-failure proof guidelines of NUREG-0612, Section 5.1.6. The heads are handled as close to the refueling floor as is practical. Both heads are lifted by the RPV head strongback. As described above for loads handled over the reactor, the head strongback is not single-failure proof. However, the design is conservative and the potential for a load drop is very small.

The reactor well and dryer separator pool plugs are handled as close to the refueling floor as is practical. As described above for loads handled over the reactor, the four lift points of each plug are not single-failure proof. However, the design is conservative and the potential for a load drop is very small.

The RPV service platform has three lift points. The platform is handled as close to the refueling floor as is practical. It is lifted by the service platform sling. As described above for loads handled over the reactor, the sling is not single-failure proof. However, the design is conservative and the potential for a load drop is very small.

INSERT 16

The RPV stud tensioner has four lift points. The tensioner is handled as close to the refueling floor as is practical. The stud tensioner lifting device consists of four slings supplied with the tensioner. ~~The design is conservative and the potential for a load drop is very small.~~

The RPV head stud rack has a single lifting point. The stud rack is handled as close to the refueling floor as is practical. The stud rack is lifted by a sling selected to meet the single-failure proof criteria of NUREG-0612, Section 5.1.6(1).

Because the polar crane main hoist is prevented from traveling over the fuel pool, as described in Section 9.1.5.3.1, a load drop would not damage the fuel pool, spent fuel racks, or spent fuel. The RPV service platform, stud tensioner, and head stud rack are light enough to be handled by the polar crane auxiliary hoist. The loads paths are administratively controlled to keep these loads out of the main hoist exclusion area, i.e., from over the fuel pool.

In summary, a load drop on the refueling floor of any of the loads normally carried over the floor by a nonsingle-failure proof overhead handling system would satisfy the four evaluation criteria of NUREG-0612, Section 5.1.

Table 9.1-15 presents a failure modes and effects analysis for the reactor building polar crane.

(Ref. p.9.1-93)

INSERT 6

The tensioner sling design factors of safety versus yield and ultimate strengths are provided in Table 9.1-14. The factors calculated for the maximum combined static and dynamic load, assuming the entire load is carried by only two of the four wire ropes, are greater than the values of 6 versus yield and 10 versus ultimate required by paragraph 5.1.6(1)(a) of NUREG-0612 for a single-failure-proof single load path special lifting device.

INSERT 5 (1)

9.1.5.3.2 Reactor Building Polar Crane Lifting Devices

Heavy loads lifted by the polar crane, and the lifting device used, are listed in Table 9.1-12. The load paths for each heavy load are shown in Figure 9.1-32. Table 9.1-19 compares the polar crane lifting devices and associated lift points with the NUREG-0612 criteria for special lifting devices and single-failure-proof systems. The special lifting devices along with the design safety factors are listed in Table 9.1-14.

Because the polar crane main hoist is prevented from traveling over the fuel pool, as described in Section 9.1.5.3.1, a load drop would not damage the fuel pool, spent fuel racks, or spent fuel. The RPV service platform, stud tensioner, and head stud rack are light enough to be handled by the polar crane auxiliary hoist. The load paths are administratively controlled to keep these loads out of the main hoist exclusion area, i.e., from over the fuel pool.

All heavy loads that do not need to [redacted] be carried over the reactor well are restricted from this area during refueling. Administrative procedures help to control safe movement of all heavy loads.

A load drop into the reactor well could not affect safe shutdown capability since the well is only open when the reactor is shut down. Decay heat removal capability could be threatened only by a load large enough to damage the seal plate. Failure of the seal plate would not allow the large, heavy loads to fall into the drywell because their size is greater than the space between the RPV and the drywell. The reactor well and the drywell are lined with steel plate which will retain any concrete which is fragmented by swinging or falling loads. It is doubtful that other debris large enough to damage shutdown cooling piping could fall through the labyrinth of intervening piping and structural steel, including the massive primary containment radial box beams.

The RHR shutdown cooling subsystem described in Section 5.4.7 includes a single suction line from reactor recirculation loop B. Therefore, a load drop into the reactor well could potentially disable the shutdown cooling function of the RHR system. As discussed above, a load drop damaging and bypassing the seal plate is highly unlikely. In addition, any debris that managed to fall and disable RHR shutdown cooling would not have enough residual energy when it reached the components of this subsystem to do sufficient damage to prevent manual restoration of the cooling function. Damage such as a severed or crimped pipe, or complete loss of function of a suction line valve operator is not considered credible. Shutdown cooling would be manually restored as described in Section 5.4.7.1.5. If manual restoration cannot be achieved, an alternate flow path as described in Section 15.2.9 could be used. Similarly, if debris from the load drop were able to cause leakage from exposed reactor vessel piping, makeup water could be supplied by any of a number of RHR and core spray injection lines until the leak could be repaired. Therefore, the drop of a heavy load into the reactor well would not affect decay heat removal capability.

Heavy loads carried over the refueling floor that employ lifting devices or lift points that are not single-failure proof weigh up to 10 tons. These loads are listed below and are also tabulated, with their weights, in Table 9.1-12.

a. Refueling bellows guard ring

b. RPV stud tensioner

c. Flux monitor shipping crate

d. RPV head stud rack

In summary, a load drop on the refueling floor of any of the loads normally carried over the floor by a non-single-failure proof overhead handling system would satisfy the four evaluation criteria of NUREG-0612, Section 5.1. Table 9.1-20 presents an analysis of a postulated heavy load drop against the four evaluation criteria of NUREG-0612. The following paragraphs provide additional details for each polar crane lifting device.

9.1.5.3.2.1 Fuel Cask Yoke

A single-failure proof spent fuel shipping cask lifting device (yoke) and cask lift point design in accordance with the requirements of NUREG-0612 will be selected for HCGS.

9.1.5.3.2.2 RPV Head Strongback

The RPV head strongback is used as a lifting device for the following loads:

- drywell head
- RPV head
- RPV head insulation and frame

The RPV head strongback is a special lifting device as defined by NUREG-0612, Section 5.1.1.4. The design factors of safety versus yield and ultimate strengths are provided in Table 9.1-14. The RPV head strongback design will be upgraded to meet the single-failure-proof guidelines of NUREG-0612, Section 5.1.6.

The RPV and drywell heads each have four lift points. The drywell head lift points meet the single-failure-proof guidelines of NUREG-0612, Section 5.1.6(3)(a). The RPV head lift points will be upgraded to also satisfy the single-failure-proof guidelines of NUREG-0612, Section 5.1.6(3).

The RPV head insulation and its support structure is carried over the RPV when the head is on. The support structure is lifted in two pieces. The lift points on each piece are designed to meet the single-failure-proof guidelines of NUREG-0612, Section 5.1.6(3)(a).

In summary, the RPV head strongback and the associated heavy load lift points will satisfy the single-failure-proof guidelines of NUREG-0612, Section 5.1.6. A postulated heavy load drop is not considered credible due to the single-failure-proof design.

9.1.5.3.2.3 Shield Plug Sling

The special lifting device for the reactor well shield plugs is single-failure proof in accordance with NUREG-0612, Section 5.1.6(1)(a). The design factors of safety versus yield and ultimate strength are provided in Table 9.1-14. Each shield plug has four lift points to prevent uncontrolled lowering of the load, assuming a single lift point failure. Each lift point has a maximum combined static plus dynamic design safety factor of greater than 5 with respect to material ultimage strength. The design is conservative and satisfies the single-failure-proof guidelines of NUREG-0612, Section 5.1.6(3)(a). A postulated heavy load drop is not considered credible due to the single-failure-proof design.

9.1.5.3.2.4 Dryer-Separator Sling

The dryer-separator sling lifts the steam dryer and the moisture separator. The sling design satisfies the guidelines of ANSI N14.6-1978 in general, but does not explicitly comply as recommended by NUREG-0612, Section 5.1.1(4). The design factors of safety versus yield and ultimate strengths are provided in Table 9.1-14. They are less than the values of 3 versus yield and 5 versus ultimate required by Section 5.1.1(4).

The dryer-separator sling and the lift points on the moisture separator and steam dryer will be upgraded to satisfy the single-failure-proof guidelines of NUREG-0612, Section 5.1.6. A postulated heavy load drop is not considered credible due to the single-failure-proof design.

9.1.5.3.2.5 Pool Plug Grapple

The special lifting device for the dryer-separator pool plugs is single-failure proof in accordance with NUREG-0612, Section 5.1.6(1)(a). The design factors of safety versus yield and ultimate strength are provided in Table 9.1-14. The dryer-separator pool plugs each have four lift points designed with a maximum combined static plus dynamic factor of safety greater than 5 with respect to material ultimate strength. The design is conservative and satisfies the single-failure-proof guidelines of NUREG-0621, Section 5.1.6(3)(a). A postulated heavy load drop is not considered credible due to the single-failure-proof design.

9.1.5.3.2.6 Service Platform Sling

The service platform sling lifts the RPV service platform. The sling design satisfies the guidelines of ANSI N14.6-1978 in general, but does not explicitly comply as recommended by NUREG-0612, Section 5.1.1(4). The design factors of safety versus yield and ultimate strengths are provided in Table 9.1-14. The factor versus yield is greater than the value of 3, and the factor versus ultimate is less than the value of 5 required by Section 5.1.1(4) of NUREG-0612. The RPV service platform has three lift points. The platform is handled as close to the refueling floor as is practical.

The service platform sling and the lift points on the service platform will be upgraded to meet the single-failure-proof guidelines of NUREG-0612, Section 5.1.6. No load drop analysis is required due to the single-failure-proof design.

9.1.5.3.2.7 Fuel Rack Lifting Fixture

The fuel rack lifting fixture will be used for several non-routine heavy load lifts over the fuel pool. It is used for installing the spent fuel rack modules. As described in Section 9.1.2.2.2.2, a base capacity of 1078 spent fuel cells plus 30 multipurpose cavities will be installed for initial plant operation. The remaining capacity of 17 rack modules, providing an additional 2976 cells, will be installed during plant operation. The lifting fixture design factors of safety versus yield and ultimate strengths are provided in Table 9.1-14. These factors meet the criteria of paragraph 5.1.6(1)(a) of NUREG-0612 for a single-failure-proof single load path special lifting device.

The lifting eye of the fixture is connected to the crane hook by a sling arrangement. The slings are selected to meet the single-failure-proof criteria of Section 5.1.6(1)(b) of NUREG-0612. The four legs of the fixture each have a J-shaped plate at the bottom. The fixture legs are lowered through four of the empty cells of the rack module being lifted, moved horizontally a short distance, and raised to hook to the module base. The four J-shaped plates contact the underside of the module base when it is being lifted. This design eliminates the need for lifting eyes on the module. The weight of the module, together with the shape of the lifting fixture plates, provides assurance that the fixture is securely attached to the module during lifting.

Thus, because there are no lift points on the modules, and both the crane and lifting fixture are single-failure-proof, the modules will be installed with a single-failure-proof handling system. A postulated heavy load drop is not considered credible due to the single-failure-proof design.

The modules will be lifted with the main hoist of the polar crane. Limit switches and travel stops, described in Section 9.1.5.2.1.5, will be temporarily bypassed as necessary to permit the main hook to travel into the main hook exclusion area shown on Figure 9.1-31 when the modules are installed. The temporary bypassing of limit switches and travel stops will be done under strict administrative control.

9.1.5.3.2.8 RPV Stud Tensioner Sling

The RPV stud tensioner has four lift points. The tensioner is handled as close to the refueling floor as is practical. The stud tensioner lifting device consists of four slings supplied with the tensioner. The tensioner sling design factors of safety versus yield and ultimate strengths are provided in Table 9.1-14. The factors calculated for the maximum combined static and dynamic load, assuming the entire load is carried by only two of the four wire ropes, are greater than the values of 6 versus yield and 10 versus ultimate required by paragraph 5.1.6(1)(a) of NUREG-0612 for a single-failure-proof single load path special lifting device.

The RPV stud tensioner is carried over the RPV while the head is on. A potential drop of the RPV stud tensioner would not cause fuel damage or unacceptable leakage because the drop would be less severe than a drywell or RPV head drop. An analysis of a postulated load drop against the four evaluation criteria of NUREG-0612, Section 5.1, is provided in Table 9.1-20.

9.1.5.3.2.9 Miscellaneous Single Failure Proof Slings

Single-failure proof slings selected in accordance with NUREG-0612, Section 5.1.6(1)(b), are used to lift the following loads:

- spent fuel pool slot plugs
- spent fuel pool and cask pool gates
- head stud rack
- flux monitor shipping crate
- 4'x4'-6" hatch cover
- 10'x10' hatch cover
- refueling bellows guard ring
- jib crane

The fuel pool slot plug sling is a single-failure proof conventional sling selected to meet the requirements of NUREG-0612, Section 5.1.6(1)(b), as clarified by FSAR Section 9.1.5.1.n. Each fuel pool slot plug has a single lifting point designed with a maximum combined static plus dynamic factor of safety greater than 15 with respect to material ultimate strength. This satisfies the NUREG-0612, Section 5.1.6(3)(b) requirement for a safety factor of 10. A postulated heavy load drop is not considered credible due to the single-failure-proof design.

A single-failure proof conventional sling selected in accordance with NUREG-0612, Section 5.1.6(1)(b), as clarified by FSAR Section 9.1.5.1.n is used to lift the fuel pool and cask pool gates. The pool gates are the only heavy loads which must routinely be carried over the fuel pool. There are two lift points on each fuel pool gate. A single lift point failure will not result in an uncontrolled lowering of the gate. The lift points are designed with a maximum combined static plus dynamic factor of safety greater than 15 with respect to material ultimate strength. This satisfies the NUREG-0612, Section 5.1.6(3)(a) requirement for a safety factor of 5. A postulated heavy load drop is not considered credible due to the single-failure-proof design.

The RPV head stud rack has a single lifting point. The stud rack is lifted by a sling selected to meet the single-failure-proof guidelines of NUREG-0612, Section 5.1.6(1)(b), as clarified by FSAR Section 9.1.5.1.n. The stud rack is handled as close to the refueling floor as is practical, and is only carried over the RPV while the head is on. The RPV head stud rack is not carried over the spent fuel pool. An analysis of a postulated drop against the four evaluation criteria of NUREG-0612, Section 5.1, is provided in Table 9.1-20. A potential drop would not cause fuel damage or unacceptable leakage because the drop would be less than a drywell or RPV head drop.

9.1.5.3.2.9 (cont'd)

The flux monitor shipping crate is carried over the refueling floor by slings selected to meet the single-failure proof guidelines of NUREG-0612, Paragraph 5.1.6(1)(b), as clarified by FSAR Section 9.1.5.1.n. The shipping crate is not carried over the RPV or spent fuel pool. An analysis of a postulated drop against the four evaluation criteria of NUREG-0612, Section 5.1, is provided in Table 9.1-20.

The 4'x4'-6" hatch cover and the 10'x10' hatch cover are carried over the refueling floor by slings selected to meet the single-failure-proof guidelines of NUREG-0612, Section 5.1.6(1)(b), as clarified by FSAR Section 9.1.5.1.n. The hatch covers are not carried over the RPV or the spent fuel pool. The lift points on the hatch covers satisfy the singlefailure-proof guidelines of NUREG-0612, Section 5.1.6(3)(a). A postulated heavy load drop is not considered credible due to the single-failure-proof design.

The refueling bellows guard ring is carried over the refueling floor by a single-failure proof sling selected to meet the guidelines of NUREG-0612, Section 5.1.6(1)(b), as clarified by FSAR Section 9.1.5.1.n. The guard ring is not carried over the spent fuel pool and is only carried over the RPV when the RPV head is on. An analysis of a postulated drop against the four evaluation criteria of NUREG-0612, Section 5.1, is provided in Table 9.1-20.

The fuel pool jib cranes are carried over the reactor vessel when the RPV head is off, but only when the RPV service platform is in place on the RPV flange. A conventional sling, selected in accordance with NUREG-0612, Paragraph 5.1.6(1)(b)(ii), as clarified by FSAR Section 9.1.5.1.n. is used to lift the jib crane. The load used to select the sling is two times the sum of the maximum static plus dynamic load. The dynamic load is assumed to be 0.25W, where W equals the weight of the jib crane. The load used is, therefore, 2(W+0.25W). The jib crane design has a single lift point with a design safety factor of 10 times the maximum combined concurrent static and dynamic load with respect to material ultimate strength as required by NUREG-0612, paragraph 5.1.6(3)(b). The jib crane handling system, therefore, meets the single-failure proof guidelines of NUREG-0612, Section 5.1.6. A postulated heavy load drop is not considered credible due to the single-failure-proof design.

9.1.5.3.2.10 Channel Handling Boom Crane

The channel handling boom crane is lifted by the auxiliary hook. No lifting device is necessary as the boom crane connects directly to the auxiliary hook of the polar crane.

The channel handling boom crane is not carried over the RPV or spent fuel pool. Table 9.1-20 provides an analysis of a postulated drop against the four evaluation criteria of NUREG-0612, Section 5.1

RHR suppression pool cooling, ADS relief valve blowdown, and core spray return flow to the reactor, could then be used. Therefore, this hoist satisfies guideline 5.1.5(1)(c) of NUREG-0612.

i. Vacuum breaker valve removal hoist (10H207)

This hoist does not handle heavy loads.

j. Main steam relief valve removal hoist (10H202)

This hoist does not handle heavy loads.

k. Turbine building bridge crane (10H102)

There is no safe shutdown or decay heat removal equipment beneath the load path of this crane or on the next lower elevation, but there are safety-related instruments, cables, or conduits of the reactor protection system (RPS) on both elevations. The safety-related function of the RPS is to initiate reactor scram after certain abnormal operational transients. The stator lift beam is intended to only be used during construction. If an unforeseen problem requires that the stator be lifted after plant startup, the lift will only be made when the reactor is shut down. The RPS is not required to function then. Therefore, a stator lift beam load drop will not compromise the safety function of the RPS.

The main and auxiliary hoists are used mainly during reactor shutdown, but they are also used during reactor operation. Because the RPS is a fail safe system, main or auxiliary hoist load drop could cause a reactor scram but would not affect safe shutdown or decay heat removal capability.

l. Feedwater heater removal hoist (1AH103, 1BH103)

There is no safe shutdown or decay heat removal equipment beneath the load path of these hoists or on the next lower elevation.

that the impact could cause water loss. However, water loss would not prevent decay heat removal.

jj. Solid radwaste monorail (00H316)

The hoist is remotely controlled with the aid of closed-circuit television from the drum-handling control panel located in the radwaste control room. If the hoist becomes inoperable, a mechanical retrieval device permits removal and/or repair as necessary, while keeping operator exposure as low as reasonably achievable.

There is no safe shutdown or decay heat removal equipment in the load path or on the next lower floor elevation. The drop of a drum could require implementation of isolation and decontamination procedures, but could not affect safe shutdown of the plant.

kk. Solid radwaste bridge crane (00H317)

The hoist is remotely controlled with the aid of closed-circuit television from the drum-handling control panel located in the radwaste control room. Independent motors control low and high speed crane movement. Eyelets on the bridge provide attachment points for a winch-type retrieval hoist in the event of a loss of crane electrical power.

There is no safe shutdown or decay heat removal equipment in the load path or on the next lower floor elevation. The drop of a drum could require implementation of isolation and decontamination procedures, but could not affect safe shutdown of the plant.

ll. SACS pumps ~~rigging beam~~ hoist ~~(future)~~

Figure 9.1-35 shows the safe load paths.

One monorail ~~rigging beam~~ serves the two pumps associated with safety auxiliaries cooling system (SACS) loop A, and the other serves the two pumps associated with loop B. A pump motor (6160 pounds) is only removed when the SACS coooling

The heaviest anticipated maintenance load is the upper half of the pump casing (825 pounds), which is not a heavy load.

loop associated with (physically separated) that pump is shutdown and completely isolated from the other (redundant) loop. This is not a normal maintenance lift. It would be done infrequently, if at all and would be subject to administrative control procedures. ~~The rigging beam monorail restricts the load path so that a load drop could only disable a pump or other equipment associated with the down loop.~~

Add Insert A here

~~A dropped motor would not punch through the elevation 102 feet floor because the deformation of the motor shroud, the intermediate pipe restraint steel, and the floor strength would absorb the kinetic energy of the dropped load. A SACS pump motor weighs 1155 pounds.~~

mm. SACS heat exchanger ~~rigging beam~~ hoist ~~(future)~~

Figure 9.1-35 shows the safe load paths.

Two hoists, one mounted on each ~~rigging beam~~ monorail, work in tandem to remove a SACS heat exchanger return end cover. The configuration includes a separate sling and lifting point for each hoist. Each of the two hoist, sling, and lift point combinations is capable of independently supporting the cover. ~~The OHLHS is thus single-failure proof in the sense that a single failure would not cause uncontrolled lowering of the load.~~

Add Insert B here

nn. Recombiner system hoists (00H318, 10H318)

This hoist does not handle heavy loads.

9.1.5.4 Inspection and Testing

9.1.5.4.1 Reactor Building Polar Crane

Final assembly and initial power operation of the bridge, both trolleys, and both hoists is done on site rather than in Paceco's shop. All crane parts subject to hoisting or seismic loads are nondestructively examined as described in Section 9.1.5.4.1.1.

INSERT A

\# Above elevation 102 ft. a dropped SACS Pump A or C motor (SACS loop A) would not affect safe shutdown capability because loop A will be down when the lift is made. If no credit is taken for the elevation 102 ft. floor, a dropped Pump A or C motor could possibly disable one of the following SACS loop B pipes above the next lower floor elevation (elevation 77 ft.):

- 30" - TACS Supply
- 20" - SACS Loop B Disch. to Diesels
- 20" - SACS Loop B Return from Diesels
- 30" - TACS Return

Loss of any one of these pipes could cause loss of the decay heat removal function of SACS loop B, which in turn could cause loss of safe shutdown capability. To preclude the possibility that a dropped loop A motor could punch through the elevation 102 feet floor, the motor lift height will be mechanically restricted [redacted] INSERT A

to the minimum necessary distance above the floor, and energy absorbing material will be placed beneath the load path.

The pump A motor is lifted vertically until it is approximately 8 feet above the floor so it will clear spring can pipe support EG-123-H01 when it is moved horizontally south ~~approximately~~ 10 feet past the pump A discharge line, EG-123-HBC-20", before it is lowered between building column line 21R and 20R. The load handling procedure for the pump A motor requires that a sling long enough to permit the motor to be lifted only as high as is necessary to clear the pipe support be used. The procedure also requires that energy absorbing material, ~~be verified in place~~ sufficient to prevent the dropped motor from punching through the floor or causing spalling, be verified in place by the hoist operator before the lift is made.

The pump C motor is lifted vertically until it is approximately 4 feet above the floor so it will clear the lip of the pump baseplate when it is moved horizontally north approximately 6 feet before it is lowered

INSERT A

between column lines 21R and 20R. The load handling procedure for the pump C motor requires that a sling long enough to permit the motor to be lifted only as high as is necessary to clear the baseplate lip be used. The procedure also requires that energy absorbing material sufficient to prevent the dropped motor from punching through the floor or causing spalling, be verified in place by the hoist operator before the lift is made.

d Above elevation 102 ft. a dropped SACS Pump B and D motor (SACS loop B) would not affect safe shutdown capability because loop B will be down when the lift is made. If no credit is taken for the elevation 102 ft. floor, a dropped Pump B or D motor could possibly disable the 6" RHR Post-LOCA containment flooding crosstie from the station service water system that runs above elevation 77 ft. This

INSERT A

line is not required for safe shutdown or decay heat removal. However, it is ~~necessary~~ used for long term decay heat removal (containment flooding) after a LOCA. Therefore, the load handling procedure for the pumps B and D motors requires that a sling long enough to permit the motor to be lifted only as high as is necessary to clear the baseplate lip be used. It also requires that energy absorbing material sufficient to prevent the dropped motor from punching through the floor or causing spalling, be verified in place by the hoist operator before the lift is made.

Therefore, these hoists satisfy guideline 5.1.5(1)(a) of NUREG-0612.

(Ref. Section 9.1.5.3.3.mm) INSERT B

\# There is no safe shutdown or decay heat removal equipment beneath the load paths on elevation 102 ft. or on the next lower elevation (77 ft.). But the 18-inch RHR heat exchanger A inlet line, 3 Channel A Class 1E cable trays, and some Channel A Class 1E conduits are located ~~beneath a portion of the SACS heat exchanger A hoist load~~ in the northwest corner of the RHR heat exchanger A compartment below elevation 77 ft and beneath a portion of the SACS heat exchanger A hoist load path. The 18-inch RHR heat exchanger B inlet line ~~is located above elevation 77~~ and two Channel B Class 1E cable tray are located ~~above elevation 77 ft.~~ beneath these load path on elevation 77 ft. Three additional Channel B Class 1E cable trays and some Channel B Class 1E conduits are located in the southwest corner of the RHR heat exchanger B compartment below elevation 77 ft and beneath a portion of the

INSERT B

SACS heat exchanger B hoist load path.

To preclude the possibility that a dropped SACS heat exchanger end cover could penetrate the elevation 102 ft. floor, the cover lift height will be mechanically restricted to the minimum necessary distance above the floor, and energy absorbing material will be placed beneath the load path, or another load handling system that satisfies the four evaluation criteria of Section 5.1.1 of NUREG 0612 will be used.

Therefore, these hoists satisfy guideline 5.1.5(1)(c) of NUREG-0612.

TABLE 9.1-12

OHLHS LOADS OVER SAFETY-RELATED EQUIPMENT

				First Elevation			Second Elevation		
Heavy Load	Load Weight	Lifting Device	Safe Load Path Fig. (4)	Feet	Safety Related, Safe Shutdown, or Decay Heat Removal Equipment	Hazard Elimination Criterion(1)	Feet	Safety Related, Safe Shutdown, or Decay Heat Removal Equipment	Hazard Elimination Criterion
Crane/Hoist: Reactor Building Polar Crane (Item 1, Table 9.1-10)									
a. Reactor well shield plugs	107-1/2 tons	Shield plug sling	9.1-32 Sht. 1	201	RPV	d	178	FRVS Recirc. Unit	d
b. Drywell head	65 tons	RPV head strongback	9.1-32 Sht. 2	201	RPV	d	178	FRVS Recirc. Unit	d
c. Reactor vessel head	97 tons	RPV head strongback	9.1-32 Sht. 2	201	RPV	d	162 " "	Standby Liquid Control H_2 Recombiner H_2O_2 Analyzer	d d d
d. Moisture Separator	73-1/4 tons	Dryer/ separator sling	9.1-32 Sht. 3	201	RPV	d	162	A & B H_2 Recombiner	d
e. Steam dryer	45 tons	Dryer/ separator sling	9.1-32 Sht. 3	201	RPV	d	162	A & B H_2 Recombiner	d
f. Dryer/separator pool plugs	38 tons	Pool plug grapple	9.1-32 Sht. 4	201	None	NA	178	FRVS Recirc. Unit	d
g. Spent fuel shipping cask	110 tons	Fuel cask yoke	9.1-32 Sht. 5	201	None	NA	162	Fuel Pool Cooling System	d
h. Auxiliary hoist load block	1 ton	(None required)	9.1-32 Sht. 10	201 " "	A & B SACS Expansion Tank RPV Spent Fuel Pool	d d d	178 162 " "	FRVS Recirc. Unit SLC H_2O_2 Analyzer H_2 Recombiners	d d d d
i. Main hoist load block	10 tons	(None required)	9.1-32 Sht. 9	201 "	A & B SACS Expansion Tank RPV	d d	178 162 " "	FRVS Recirc. Unit SLC H_2O_2 Analyzer H_2 Recombiners	d d d d
j. Spent fuel pool slot plugs	25 tons	Single-failure-proof sling	9.1-32 Sht. 3	201 "	Spent Fuel Pool RPV	d d	178	FRVS Recirc. Unit	d
k. Spent fuel pool gates & cask pool gates	3.4 tons	Single-failure-proof sling	9.1-32 Sht. 2	201	Spent Fuel Pool	d	162	Spent Fuel Pool	d

OHLHS LOADS OVER SAFETY-RELATED EQUIPMENT

	Heavy Load	Load Weight	Lifting Device	Safe Load Path Fig. (4)	First Elevation Feet	First Elevation Safety Related, Safe Shutdown, or Decay Heat Removal Equipment	First Elevation Hazard Elimination Criterion(1)	Second Elevation Feet	Second Elevation Safety Related, Safe Shutdown, or Decay Heat Removal Equipment	Second Elevation Hazard Elimination Criterion
l.	RPV service platform	5 tons	Service platform sling	9.1-32 Sht. 5	201	RPV	d	~~177~~ 178	FRVS Recirc. Unit	d
m.	Head stud rack	2.1 tons	Single-failure-proof sling	9.1-32 Sht. 8	201	RPV	e	178	FRVS Recirc. Unit	e Leave blank line here
								162	H_2 Recombiners	e
								"	H_2O_2 Analyzers	e
								"	SLC	e
n.	Vessel head insulation and frame	5 tons	RPV head strongback	9.1-32 Sht. 1	201	RPV	d	178	FRVS Recirc. Unit	d
o.	Flux monitor shipping crate	2.5 tons	Single-failure-proof sling	9.1-32 Sht. 2	201	None	NA	162	'A' # H_2 Recombiner	e
								"	SLC	e
								"	H_2O_2 Analyzer	e
p.	Stud tensioner frame	5.3 tons	RPV stud tensioner sling	9.1-32 Sht. 3 & 11	201	RPV	e	178	FRVS Recirc. Unit	e
								162	SLC	e
								"	H_2O_2 Analyzer	e
q.	Head strongback	4.4 tons	(None required)	9.1-32 Sht. 1&2	201	RPV	d	178	FRVS Recirc. Unit	d
								162	SLC	d
								"	H_2O_2 Recombiner	d
								"	H_2O_2 Analyzer	d
r.	Spent fuel cask yoke	6 tons	(None required)	9.1-32 Sht. 11	201	None	NA	162	H_2O_2 Analyzer	d
								"	H_2 Recombiner	d
								"	SLC	d
s.	Hatch cover 4' x 4'-6"	2.4 tons	Single-failure-proof sling	9.1-32 Sht. 1	201	None	NA	162	None	NA
t.	Hatch cover 10' x 10'	7.5 tons	Single-failure-proof sling	9.1-32 Sht. 1	201	None	NA	162	FRVS Recirc. Unit	d
u.	Refueling bellows guard ring	10 tons	Single-failure-proof sling	9.1-32 Sht. 6	201	RPV	e	162	H_2 Recombiner	e
								"	H_2O_2 Analyzer	e
								"	SLC	e
v.	Jib crane	1.6 tons	Single-failure-proof sling	9.1-32 Sht. 5	201	RPV	d	178	FRVS Recirc. Unit	d
								162	None	NA

OHLHS LOADS OVER SAFETY-RELATED EQUIPMENT

	Heavy Load	Load Weight	Lifting Device	Safe Load Path Fig. (4)	First Elevation Feet	First Elevation Safety Related, Safe Shutdown, or Decay Heat Removal Equipment	First Elevation Hazard Elimination Criterion(1)	Second Elevation Feet	Second Elevation Safety Related, Safe Shutdown, or Decay Heat Removal Equipment	Second Elevation Hazard Elimination Criterion
w.	Channel handling boom crane	0.8 ton	(None required)	9.1-32 Sht. 7	201	None	NA	178	FRVS Recirc. Unit	e
								162	SLC	e
								"	A&B H_2O_2 Analyzer	e
								"	A&B H_2O_2 Recombiner	e
x.	Dryer-Separator sling	2 tons	(None required)	9.1-32 sht. 12	201	RPV	d	178	FRVS Recirc. Unit	d
								162	SLC	d
								"	A&B H_2 Recombiner	d
								"	A&B H_2O_2 Analyzer	d
y.	Spent fuel rack modules	10 tons	Fuel rack lifting fixture	9.1-32 Sht. 4	201	RPV Spent Fuel Pool	d d	162	None	NA
z.	Fuel rack lifting fixture	1.1 tons	Single-failure-proof sling	9.1-32 Sht. 4	201	RPV Spent fuel pool	d d	162	None	NA
aa.	Reactor well shield plug sling	4.5 tons	(None required)	9.1-32 Sht. 9	201	RPV	d	178	FRVS Recirc. Unit	d
								162	SLC	d
								"	A&B H_2 Recombiner	d
								"	A&B H_2O_2 Analyzer	d
bb.	Dryer/separator pool plug grapple	6 tons	(None required)	9.1-32 Sht. 9	201	RPV	d	178	FRVS Recirc. Unit	d
								162	SLC	d
								"	A&B H_2 Recombiner	d
								"	A&B H_2O_2 Analyzer	d

Crane/Hoist: Personnel Air Lock Hoist (Item 2, Table 9.1-10)

	Heavy Load	Load Weight	Lifting Device	Safe Load Path Fig. (4)	First Elevation Feet	First Elevation Equipment	First Elevation Hazard Elimination Criterion(1)	Second Elevation Feet	Second Elevation Equipment	Second Elevation Hazard Elimination Criterion
a.	Air lock	30 tons	Air lock strongback	9.1-33	102	None	NA	77	-Torus & core spray	b, c
								"	-HPCI	b, c
								"	-SRV Discharge piping	b, c
b.	Upper shield block	21 tons	(None required)	9.1-33	102	None	NA	77	-Torus & core spray	b, c
								"	-HPCI	b, c
								"	-SRV Discharge piping	b, c
c.	Lower shield blocks (8)	17 tons	(None required)	9.1-33	102	None	NA	77	-Torus & core spray	b, c
								"	-HPCI	b, c
								"	-SRV Discharge piping	b, c

OHLHS LOADS OVER SAFETY-RELATED EQUIPMENT

Heavy Load	Load Weight	Lifting Device	Safe Load Path Fig. (4)	First Elevation: Feet	First Elevation: Safety Related, Safe Shutdown, or Decay Heat Removal Equipment	First Elevation: Hazard Elimination Criterion(1)	Second Elevation: Feet	Second Elevation: Safety Related, Safe Shutdown, or Decay Heat Removal Equipment	Second Elevation: Hazard Elimination Criterion
Crane/Hoist: Recirculation Pump Motor Hoist (Item 3, Table 9.1-10)									
Recirculation pump motor	24 tons	(None required)	9.1-33	102 (Inside drywell)	Recirculation (1AP201, 1BP201) and associated piping and conduit	b, c	87 (Bottom of drywell)	None	NA
Crane/Hoist: HPCI Pump and Turbine Hoist (Item 5, Table 9.1-10)									
HPCI pump and turbine parts (turbine case)	3.75 tons	Conventional slings	9.1-34	54	pumps (10P204, 10P217) turbine (10S211) & HPCI piping	b, c	(No lower elevation)		NA
Crane/Hoist: Main Steam Tunnel Underhung Crane (Item 1, Table 9.1-10)									
Valve Operators:									
Main steam ~~Isolation~~ valve	1.8 tons	Conventional slings	9.1-35	102	-MSIVs (HV F028A-D)	c	54	-Torus & Core spray	b, c
Main steam stop valve	0.9 tons				-main steam piping	c		-Containment instrument gas	b, c
M.O. feedwater check valve	0.9 tons				-feedwater ~~piping~~ piping	c		-RCIC	b, c
								-HPCI	b, c
								-Nuclear boiler system instrumentation	b, c
Crane/Hoist: Inboard MSIV Hoist (Item 8, Table 9.1-10)									
Main steam isolation valve operators	1.8 tons	Conventional slings	9.1-35	102	-MSIVs (HV F022 A-D)	c	87 (bottom of drywell)	-Main steam	c, e
					-Main steam piping	c		-Containment instrument gas	c, e
								-Breathing air piping	c, e

OHLHS LOADS OVER SAFETY-RELATED EQUIPMENT

Heavy Load	Load Weight	Lifting Device	Safe Load Path Fig. (4)	First Elevation: Feet	First Elevation: Safety Related, Safe Shutdown, or Decay Heat Removal Equipment	First Elevation: Hazard Elimination Criterion(1)	Second Elevation: Feet	Second Elevation: Safety Related, Safe Shutdown, or Decay Heat Removal Equipment	Second Elevation: Hazard Elimination Criterion
Crane/Hoist: Diesel Generator Underhung Crane (Item 35, Table 9.1-10)									
Diesel Generator parts, e.g., combustion air cooling water heat exchanger tube bundle	3540 lb.	Conventional slings	9.1-36	102	Diesel generators (1AG400-1DG400) and assoc cooling piping	b, c	77	Associated cooling piping	b, c
Crane/Hoist: Intake Structure Gantry Crane (Item 36, Table 9.1-10)									
Travelling screen, S.W. pump, and misc. equipment	19 tons	Conventional slings	9.1-37	123	Screens (S501) & heaters (VE507) & S.W. pumps (P502)	b, c	93	Strainers (F509)	b, c
Crane/Hoists: Reactor Building Personnel Lock Shield Removal Hoist (Item 37, Table 9.1-10)									
T-shaped upper shield block	21 tons	(None required)	9.1-33	102	None	NA	54 " "	-Torus & Core spray -HPCI -SRV discharge piping	b, c b, c b, c
Crane/Hoist: CRD Service Hoist (Item 39, Table 9.1-10)									
CRD maintenance equipment	1 ton (maximum)	Conventional slings	9.1-35	102	None	NA	77	-RHR Pump A discharge piping -RHR shutdown cooling suction ~~Hater+e~~ -HPCI pump discharge line -HPCI turbine steam supply -RPV instrument lines	e e e e e
Crane/Hoist: SACS Pumps A and C Hoist (Item 40, Table 9.1-10)									
Motor	3.1 tons	Conventional sling	9.1-35	102	SACS loop A pumps, remaining motor, associated piping	b	77	SACS Loop B piping (TACS & diesel supply & return)	~~Hater+e~~ e

TABLE 9.1-12 (Cont'd)

OHLHS LOADS OVER SAFETY-RELATED EQUIPMENT

				First Elevation			Second Elevation		
Heavy Load	Load Weight #	Lifting Device	Safe Load Path Fig. (4)	Feet	Safety Related, Safe Shutdown, or Decay Heat Removal Equipment	Hazard Elimination Criterion(1)	Feet	Safety Related, Safe Shutdown, or Decay Heat Removal Equipment	Hazard Elimination Criterion
Crane/Hoist: SACS Pumps B and D Hoist (Item 40, Table 9.1-10)									
Motor	3.1 tons	Conventional sling	9.1-35	102	SACS Loop B pumps, remaining motor, associated piping	b	77	~~None~~ RHR Post-LOCA containment flooding line	~~NA~~ e
Crane/Hoist: SACS Heat Exchanger A Hoists (Item 41, Table 9.1-10)									
Return end cover	9.2 tons	Conventional sling	9.1-35	102	None	NA	77	- RHR loop A piping - Channel A Class 1E cable trays - Channel A Class 1E conduit	e e e
Crane/Hoist: SACS Heat Exchanger B Hoists (Item 41, Table 9.1-10)									
Return end cover	9.2 tons	Conventional sling	9.1-35	102	None	NA	77	- RHR loop B piping - Channel B Class 1E cable trays - Channel B Class 1E conduit	e e e

(1) Hazard elimination criteria:

a. Crane travel for this area/load combination is prohibited by electrical interlocks or mechanical stops.

b. System redundancy and separation precludes the loss of the capability of the system to perform its safety-related function following this load drop in this area.

c. Site-specific considerations, such as maintenance sequencing, eliminate the need to consider this load/equipment combination.

d. The likelihood of a handling system failure for this load is extremely small; i.e., Section 5.1.6 of NUREG-0612 is satisfied, the OHS is single-failure-proof.

e. Analysis demonstrates that crane failure and load drop will not prevent safe shutdown or decay heat removal, or cause unacceptable radiation release.

f. Deleted

(~~2~~ 4) Supplementary drawings showing plan and elevation views of equipment location are provided in FSAR Section 1.2.

K54/5

(2) Deleted

(3) Deleted

REACTOR BUILDING POLAR CRANE DESIGN COMPARISON WITH NUREG 0554, SINGLE FAILURE PROOF CRANES FOR NUCLEAR POWER PLANTS (MAY 1979)

NUREG Section	Complies	Does Not Comply	Notes
1. INTRODUCTION	X		
2. SPECIFICATION AND DESIGN CRITERIA			
2.1 Construction and Operating Periods	X		(1)
2.2 Maximum Critical Load	X		(2)
2.3 Operating Environment	X		(3)
2.4 Material Properties	X		(4)
2.5 Seismic Design	X		(5)
2.6 Lamellar Tearing	X		(6)
2.7 Structural Fatigue	X		(7)
2.8 Welding Proce~~s~~dures	X		(8)
3. SAFETY FEATURES			
3.1 General	X		(9)
3.2 Auxiliary Systems	X		(10)
3.3 Electric Control System	X		(~~8~~) 11
3.4 Emergency Repairs	X		(12)
4. HOISTING			
4.1 Reeving System	X		(~~9~~) 13
4.2 Drum Support	X		(~~10~~) 14
4.3 Head and Load Blocks	X		(15)
4.4 Hoisting Speed	X		(16)
4.5 Design Against Two-Blocking	X		(~~11~~; 17

TABLE 9.1-13 (cont)

NUREG Section	Complies	Does Not Comply	Notes
4.6 Lifting Devices	X		(18)
4.7 Wire Rope Protection	X		(19)
4.8 Machinery Alignment	X		(20)
4.9 Hoist Braking System	X		(21)
5. BRIDGE AND TROLLEY			
5.1 Braking Capacity	X		(22)
5.2 Safety Stops	X		(23)
6. DRIVERS AND CONTROLS			
6.1 Driver Selection	X		(~~12~~) 24
6.2 Driver Control Systems	X		(~~13~~) 25
6.3 Malfunction Protection	X		(26)
6.4 Slow Speed Drives	X		(~~14~~) 27
6.5 Safety Devices	X		(28)
6.6 Control Stations	X		(~~15~~) 29
7. INSTALLATION INSTRUCTIONS			
7.1 General	X		(30)
7.2 Construction and Operating Periods	X		(31)
8. TESTING AND PREVENTIVE MAINTENANCE			
8.1 General	X		(32)
8.2 Static and Dynamic Load Tests	X		(33)
8.3 Two-Block Test	X		(~~16~~) 34
8.4 Operational Tests	X		(35)
8.5 Maintenance	X		(36)

TABLE 9.1-13 (cont)

NUREG Section	Complies	Does Not Comply	Notes
9. OPERATING MANUAL	X		(37)
10. QUALITY ASSURANCE	X		(~~37~~)38

Notes:

(1) Section 2.1 - The load lifts during construction were not greater than those for plant operation; therefore, no separate specifications were prepared.

(2) Section 2.2 - The reactor building polar crane main hoist is designed to handle a maximum critical load (MCL) of 130 tons. The MCL rating will be clearly marked on the main hoist. The design rated load (DRL) of 150 tons provides an overall increase of 15% in the crane's load handling ability above its MCL capacity to compensate for wear and exposure.

The reactor building polar crane auxiliary hoist is designed to handle a MCL of 8.5 tons. The MCL rating will be clearly marked on the auxiliary hoist. The design rated load (DRL) of 10 tons provides an overall increase of more than 15% in the crane's load handling ability above its MCL capacity to compensate for wear and exposure.

(3) Section 2.3 - All identified parameters, except maximum rate of pressure increase and emergency corrosive conditions, were specified. A maximum rate of pressure increase was not specified because it was judged not significant to safe design of the crane. Because it is in the reactor building, outside the drywell, the crane would not be subjected to the high accident pressure (62 psig) possible inside the drywell. The maximum pressure increase specified for crane design is -.25 in. wg minimum to +7 in. wg maximum. Emergency corrosive conditions were not specified because none were identified that would prevent safe crane operation.

(4) Section 2.4 - The minimum specified operating temperature is 60°F. Materials for structural members essential to structural integrity are impact-tested unless exempted by the provisions of Paragraph AM-218 of the ASME Code, Section VIII, Division 2. All structural members, except the main hoist drums, are exempt under Paragraph AM-218.2, which withdraws the impact test requirement if stress intensity is less than 6000 psi. The main hoist drums are Charpy-tested

per ASTM A 370. The crane was not subjected to coldproof testing because low alloy steel, such as ASTM A 514, is not used. Cast iron is not used for any crane parts.

(5) Section 2.5 - The SSE design vertical acceleration is less than 1g. Therefore the bridge and trolley wheels will not jump up off their tracks during a seismic event. The bridge and trolley designs include horizontal seismic restraints that would prevent the wheels from leaving the tracks.

(6) Section 2.6 - Nondestructive examination (NDE) was done on all welds whose failure could cause a drop of a critical load. Section 9.1.5.4.1.1 describes the NDE in more detail. Lamellar tearing of these welds is not expected to occur.

(7) Section 2.7 - A structural fatigue analysis was not part of the design requirements for the reactor building polar crane. The crane is classified as a low-use crane according to the guidelines of CMAA Specification 70. Structural fatigue analysis is not considered necessary in view of the low number of load cycles expected.

Add insert 1a & 1b

11(~~8~~) Section 3.3 - Cab controls are deadman-type with spring return. A deadman foot switch in the cab must be held down during crane operation. Release of the switch will stop the crane and set the brakes. Overspeed switches on the hoist drives stop the motors and set the brakes at 120% of no load speed. Pendant controls are momentary contact pushbuttons that return to off when released. Pendant control includes an emergency stop pushbutton that stops power to all drivers.

Add Insert 2

13(~~9~~) Section 4.1 - The maximum fleet angle from drum to lead sheave in the load block or between individual sheaves does not exceed 3-1/2 degrees at any one point during hoisting. Reverse bends are not used in the reeving system. Each main hoist rope is reeved through block and upper sheave assemblies so that its eight parts provide two parts in each quadrant of the load block about the vertical axis of the hook. With both ropes effective, the load is supported by sixteen parts at an effective static factor of safety of 10. If one rope loses its effectiveness, the load is supported by the eight parts of the remaining rope at a static factor of safety of ~~60~~ 6. The extra improved plow steel main hoist wire ropes, with independent wire rope center are 1-1/2 inches in diameter with an ultimate breaking strength of 228,000 pounds each. With both auxiliary hoist ropes effective, the load is supported by four parts at an effective static factor of safety of 15. If one rope loses its effectiveness, the load is supported by two parts of the remaining rope at a static factor of safety of 5. The stainless steel auxiliary hoist wire ropes, with independent wire rope center, are 1 inch in diameter with an ultimate breaking strength of 77,200 pounds each.

14 (~~20~~) Section 4.2 - The main hoist and auxiliary hoist drum assemblies, each with its shafts and bearings, are designed at factors of safety not less than 10. Safety lugs are provided inside each trolley truck to sustain the drum assembly hubs in the event of drum shaft failure at either end. Upper sheave shafts and block swivel assemblies are provided with safety retainers and block housings capable of sustaining the load in case of shaft or swivel failure. Drum movement in this event is mechanically limited so that the gears and holding brakes remain engaged.

Add Insert 3a & 3b

17 (~~21~~) Section 4.5 - Dual upper limit switches of diverse design in series, and an overload cutoff switch on each hoist stop the hoist motor and set the brakes. Motor overtemperature switches activate warning lights in the cab and on the pendant. Each limit switch allows the hoist motor to be operated in reverse after it has opened.

Add Insert 4a, 4b & 4c

24 (~~22~~) Section 6.1 - An emergency breaker switch located at the refueling floor level cuts power to the crane independently of the crane controls.

25 (~~23~~) Section 6.2 - The crane ~~is~~ does not lift spent fuel assemblies. ~~assemblies.~~

Add Insert 5

27 (~~24~~) Section 6.4 - Jogging and plugging are considered in the crane controls design. Drift point is not provided for bridge or trolley movement.

Add Insert 6

29 (~~25~~) Section 6.6 - Manual controls for hoisting and trolley movement are not provided on the trolley. Manual controls for the bridge are not located on the bridge.

Add Insert 7

34 (~~26~~) Section 8.3 - The crane design does not include an energy controlling device between the load and head blocks. Therefore, the two-block test is not done. Instead, the two-block test consists of verification that the two uptravel limit switches on each hoist function as designed.

Add Insert 8

38 (~~27~~) The crane is procured under a QA program that complies with the applicable provisions of ANSI N45.2-1971. Field installation, testing, operator qualification, and crane operation comply with ANSI B30.2.

HCGS FSAR

Table 9.1-14

HOPE CREEK SPECIAL LIFTING DEVICE FACTORS OF SAFETY

Item	Special Lifting Device	Rated Capacity tons	Maximum Load Weight tons	Lifting Device Weight tons	Maximum Static Load tons	Dynamic Load Factor(4)	Maximum Combined Static and Dynamic Load tons	Stress Design Factor vs. Yield(5)	Stress Design Factor vs. Ultimate(5)	NUREG-0612 Section 5.1.6(4) Factor of Safety Compliance
1.	RPV head strongback	100	97	4.4	101.4	0.15	116.6	3.5	57	Yes (6)
2.	Dryer separator sling	73.5	73.3	2	75.3	0.15	86.6	2.6	4.3	No (6)
3.	RPV service platform sling	7.2	5.9	0.1	6	0.175	7.1	3.0	3.1	NO (6)
4.	Fuel cask yoke	110	110	(2)	(2)	0.15	(2)	(2)	(2)	(2)
5.	~~Reactor~~ well shield plug sling	107.5	107.5	4.5	112	0.15	128.8	3	5	Yes (7)
6.	~~Dryer separator~~ pool plug grapple	38	38	6	44	0.15	50.6	6	10	Yes
7.	DELETED									
8.	RPV stud tensioner sling	5.3	5.3	0.2	5.5	0.175	6.5	14.2	14.7	Yes
9.	Personnel air lock strongback	30	30	0.7	30.7	0.15	35.3	6	10	Yes
10.	Fuel rack lifting fixture	10	10	1.1	11.1	0.175	13.0	6	10	Yes

Notes:

(1) Deleted

no change (2) The spent fuel shipping cask and yoke are not yet known for HCGS. The cask weight and dimensions used for the HCGS design are based on the projected 125-ton NLI 20/32 rail cask. A 110-ton load is used here to acknowledge assumption of the NLI cask design responsibility by Nuclear Assurance Corporation (NAC) and subsequent development of the NAC 12/32 cask to replace the projected NLI cask.

(3) Deleted

new (4) Dynamic load factor = 0.15 ≤ 0.005 (HOIST SPEED, feet per minute) ≤ 0.50

new (5) NUREG-0612, Section 5.1.6(4) requires a factor of safety of 3 versus yield and 5 versus ultimate strength for the combined static and dynamic load. The static load includes the weight of the load plus the weight of the special lifting device.

NOTES (cont'd):

(6) The lifting device will be upgraded to meet the single-failure-proof guidelines of NUREG-0612, Section 5.1.6(1)(a)

(7) The reactor well shield plug sling has dual load paths and therefore satisfies the single-failure-proof requirements of NUREG-0612, Section 5.1.6(1)(a)

Insert 1a

(8) Section 2.8 - Crane fabrication is in accordance with AWS D1.1, Structural Welding Code. The weld procedures that were used are qualified in accordance with AWS D1.1.

Insert 1b

(9) Section 3.1 - The crane specification included provisions that addressed the design, fabrication and testing of the load bearing components, equipment, and subsystems. In addition, the provisions of withdrawn Regulatory Guide 1.104, Overhead Crane Handling Systems for Nuclear Power Plants, that pertain to crane design, fabrication and testing were invoked in an appendix of the crane specification.

(10) Section 3.2 - As stated in Design Basis Section 9.1.5.1.c, the design basis for the auxiliary hoist is that it be single failure proof. It is described in Section 9.1.5.3.1.

Insert 2

(12) Section 3.4 - The crane design basis is to safely hold the load in the event of a control or component failure. The design permits the load to be manually lowered.

Insert 3a

(15) Section 4.3 - As described in Section 9.1.5.2.1.2, both the main and auxiliary hoists are provided with dual reeving systems, and each load block assembly is provided with dual load attachment points. The parts of the vertical hoisting system, including the head block, reeving system, load block, and hook for both the main and auxiliary hoists are designed to support a static load of 200 percent of the design rated load, (DRL) instead of the maximum critical load (MCL) as required by NUREG-0554. For the main hoist, the DRL is 150 tons and the MCL is 130 tons. For the auxiliary hoist, the DRL is 10 tons and the MCL is 8.5 tons. Each load path of each dual path hook was given a 200 percent static load test. Geometric configuration measurements of the hook were made before and after each test, and were followed by both volumetric and surface non-destructive examination. The examination results are documented and recorded.

Insert 3b

(16) Section 4.4 - As given in Table 9.1-11, the maximum main hoist speed is 4.5 ft/min and the maximum auxiliary hoist speed is 35 ft/min. The "slow" column of Figure 70-6 of CMAA-70 suggests speeds of 5 and 20 ft/min for the main and auxiliary hoists, respectively. The static stepless magnetorque control provides smooth hoist acceleration and deceleration and precise spotting of the load. The auxiliary hoist speed is only 17 percent above the slow speed (30 ft/min.) recommended for cab operated cranes in Table 2 of the Whiting Crane Handbook, 4th Edition, and is well below the recommended medium speed of 60 ft/min. X

Insert 4a

(18) Section 4.6 - As described in Section 9.1.5.2.1.2, the main hoist sister hook and lifting eye bolt are independently supported by their respective crosshead and bearings that are in turn supported by the load block. The auxiliary hoist hook and shackles are independently supported by the load block.

Insert 4b

(19) Section 4.7 - Side loads are not planned. The main and auxiliary hoist reeving systems do not include wire rope guards.

(20) Section 4.8 - The main and auxiliary hoists employ redundant holding brakes. Each brake is coupled to the drum via a separate gear train.

(21) Section 4.9 - As described in Section 9.1.5.2.1.2, the mechanical holding brakes are automatically activated on loss of electric power. The torque rating of each brake is 150 percent of rated full load hoist motor torque. Each hoist also includes one dc-actuated eddy current, power control type load brake.

(22) Section 5.1 - As described in Section 9.1.5.2.1.2, the trolley and bridge brakes are automatically applied on loss of power. They are rated at 125 percent of drive motor full load torque. Drag brakes are not used. The reversing static stepless controls provided for the main and auxiliary trolleys and the bridge permit minimum incremental movements of 1/8 inch for the main trolley, 3/8 inch for the auxiliary trolley, and 1/4 inch for the bridge. The maximum speeds of the bridge (40 ft/min), main (10 ft/min), and auxiliary (50 ft/min) trolleys are within the limits of 50, 30, and 125 ft/min, respectively, recommended in the "slow" column of Figure 70-6 of CMAA 70.

Insert 4c

(23) Section 5.2 - Section 9.1.5.2.1.5 describes the bridge and trolley limit switches, bridge rail stops, and trolley bumpers.

Insert 5

(26) Section 6.3 - Malfunction protection that includes sensing and response to excessive current, motor temperature, speed, load, and travel is provided for the hoists, trolleys, and bridge.

Insert 6

(28) Section 6.5 - The crane safety devices are separate from the control devices.

Insert 7

(30) Section 7.1 - The manufacturer provided installation instructions.

(31) Section 7.2 - Separate construction specifications were not prepared because the construction duty was expected to be enveloped by the specified design requirements. After construction use the crane will be thoroughly inspected and preoperationally tested as described in Section 9.1.5.4.1.2.

(32) Section 8.1 - Mechanical and electrical system checks were done after initial installation. The shop testing records are available at the jobsite.

(33) Section 8.2 - Static and dynamic preoperational load tests were performed as described in Section 9.1.5.4.1.2.

Insert 8

(35) Section 8.4 - The operational tests are performed in accordance with Chapter 2-2 of ANSI B30.2-1976.

(36) Section 8.5 - See Note (2).

(37) Section 9 - An operating and maintenance manual was provided by the manufacturer. It includes operating requirements for all travel movements.

TABLE 9.1 - 19

SINGLE-FAILURE-PROOF LIFTING DEVICES AND ASSOCIATED HEAVY LOAD LIFT POINTS

Lifting Device/Heavy Load Lift Point	Special Lifting Device (1)	Single Failure Proof	NUREG-0612 Applicable Criteria
1. Fuel Cask Yoke	Yes	Yes	Note 2
- Spent Fuel Shipping Cask	NA	Yes	Note 2
2. RPV Head Strongback	Yes	Yes	Note 3
- Drywell head	NA	Yes	5.1.6(3)(a)
- RPV head	NA	Yes	Note 3
- RPV head insulation & frame	NA	Yes	5.1.6(3)(a)
3. Shield Plug Sling	Yes	Yes	5.1.6(1)(a)
- Reactor well shield plugs	NA	Yes	5.1.6(3)(a)
4. Dryer/Separator Sling	Yes	Yes	Note 3
- Steam dryer	NA	Yes	Note 3
- Moisture separator	NA	Yes	Note 3
5. Pool Plug Grapple	Yes	Yes	5.1.6(1)(a)
- Dryer/Separator pool plugs	NA	Yes	5.1.6(3)(a)
6. Service Platform Sling	Yes	Yes	Note 3
- Service platform	NA	Yes	Note 3
7. Fuel Rack Lifting Fixture	Yes	Yes	5.1.6(1)(a)
- Spent fuel rack module	NA	Yes	Note 4
8. RPV Stud Tensioner Sling	Yes	Yes	5.1.6(1)(a)
- RPV stud tensioner	NA	No	NA
9. Miscellaneous Slings (Note 6)	No	Yes	5.1.6(1)(b)
- Spent fuel pool slot plugs	NA	Yes	5.1.6(3)(b)
- Spent fuel pool & cask pool gates	NA	Yes	5.1.6(3)(a)
- Head stud rack	NA	No	NA
- Flux monitor shipping crate	NA	No	NA
- 4'x4'-6" Hatch cover	NA	Yes	5.1.6(3)(a)
- 10'x10' Hatch cover	NA	Yes	5.1.6(3)(a)
- Refueling bellows guard ring	NA	No	NA
- Jib crane	NA	Yes	5.1.6(3)(b)
10. Polar Crane Main and Auxiliary Hoists	Note 5	Yes	5.1.6(2)

TABLE 9.1 - 19

SINGLE-FAILURE-PROOF LIFTING DEVICES AND ASSOCIATED HEAVY LOAD LIFT POINTS

Notes:

(1) Special lifting device factors of safety are given in Table 9.1-14.

(2) The spent fuel shipping cask and yoke are not yet known for HCGS. A single-failure-proof shipping cask lifting device (yoke) and cask lift point design in accordance with NUREG-0612 Sections 5.1.6(1)(a) and 5.1.6(3) will be selected.

(3) The lifting device and/or lift points of this heavy load will be upgraded to meet the single-failure-proof guidelines of NUREG-0612, Section 5.1.6.

(4) The spent fuel rack modules have no lift points. The design of the fuel rack lifting fixture eliminates the need for lift points on the module.

(5) The polar crane main and auxiliary hoists are integral parts of the polar crane and are not considered special lifting devices.

(6) Miscellaneous slings that are not special lifting devices are selected as discussed in FSAR Section 9.1.5.1.n.

TABLE 9.1-20

POLAR CRANE LOAD DROP ANALYSIS COMPARISON AGAINST NUREG-0612 EVALUATION CRITERIA

Heavy Load	NUREG-0612 EVALUATION CRITERIA				FSAR Section for Safety Evaluation
	I Doses Less than 25% of 10CFR100	II Keff Less than 0.95	III No Fuel Uncovery	IV No Loss of Safe Shutdown Function	
a. Reactor Well Shield Plugs	(1)	(1)	(1)	(1)	9.1.5.3.2.3
b. Drywell Head	(5)	(5)	(5)	(5)	9.1.5.3.2.2
c. Reactor Vessel Head	(4)	(4)	(4)	(4)	9.1.5.3.2.2
d. Moisture Separator	(4)	(4)	(4)	(4)	9.1.5.3.2.4
e. Steam Dryer	(4)	(4)	(4)	(4)	9.1.5.3.2.4
f. Dryer/Separator Pool Plugs	(1)	(1)	(1)	(1)	9.1.5.3.2.5
g. Spent Fuel Shipping Cask	(2)	(2)	(2)	(2)	9.1.5.3.2.1
h. Auxiliary Hoist Load Block	(3)	(3)	(3)	(3)	9.1.5.3.1
i. Main Hoist Load Block	(3)	(3)	(3)	(3)	9.1.5.3.1
j. Spent Fuel Pool Slot Plugs	(1)	(1)	(1)	(1)	9.1.5.3.2.9
k. Spent Fuel Pool Gates and Cask Pool Gates	(1)	(1)	(1)	(1)	9.1.5.3.2.9
l. RPV Service Platform	(4)	(4)	(4)	(4)	9.1.5.3.2.6
m. Head Stud Rack	(6)	(6)	(6)	(6)	9.1.5.3.2.9
n. Vessel Head Insulation and Frame	(5)	(5)	(5)	(5)	9.1.5.3.2.2

POLAR CRANE LOAD DROP ANALYSIS COMPARISON AGAINST NUREG-0612 EVALUATION CRITERIA

Heavy Load	NUREG-0612 EVALUATION CRITERIA I Doses Less than 25% of 10CFR100	II Keff Less than 0.95	III No Fuel Uncovery	IV No Loss of Safe Shutdown Function	FSAR Section for Safety Evaluation
o. Flux Monitor Shipping Crate	(9)	(9)	(9)	(9)	9.1.5.3.2.9
p. Stud Tensioner Frame	(7)	(7)	(7)	(7)	9.1.5.3.2.8
q. Head Strongback	(4)	(4)	(4)	(4)	9.1.5.3.2.2
r. Spent Fuel Cask Yoke	(2)	(2)	(2)	(2)	9.1.5.3.2.1
s. Hatch Cover 4'x4'-6"	(1)	(1)	(1)	(1)	9.1.5.3.2.9
t. Hatch Cover 10'x10'	(1)	(1)	(1)	(1)	9.1.5.3.2.9
u. Refueling Bellows Guard Ring	(8)	(8)	(8)	(8)	9.1.5.3.2.9
v. Jib Crane	(1)	(1)	(1)	(1)	9.1.5.3.2.9
w. Channel Handling Boom Crane	(10)	(10)	(10)	(10)	9.1.5.3.2.10
x. Dryer-Separator Sling	(4)	(4)	(4)	(4)	9.1.5.3.2.4
y. Spent Fuel Rack Modules	(1)	(1)	(1)	(1)	9.1.5.3.2.7
z. Fuel Rack Lifting Fixture	(1)	(1)	(1)	(1)	9.1.5.3.2.7
aa. Reactor Well Shield Plug Sling	(1)	(1)	(1)	(1)	9.1.5.3.2.3
bb. Dryer-Separator Pool Plug Grapple	(1)	(1)	(1)	(1)	9.1.5.3.2.5

HCGS

TABLE 9.1-20

NOTES:

(1) The crane, lifting device, and lift points of the heavy load satisfy the single-failure-proof guidelines of NUREG-0612, Section 5.1.6. No load drop analysis is required.

(2) A single-failure-proof fuel cask lifting device (yoke) and cask lift point design in accordance with NUREG-0612 will be selected for HCGS. No load drop analysis is required.

(3) The polar crane and its main hoist load block and auxiliary hoist load block satisfy the single-failure-proof guidelines of NUREG-0612. No load drop analysis is required.

(4) The heavy load lift points and associated lifting device will be upgraded to satisfy the single-failure-proof guidelines of NUREG-0612, Section 5.1.6. No load drop analysis is required.

(5) The lift points of the heavy load currently satisfy the single-failure-proof guidelines of NUREG-0612. The lifting device will be upgraded to satisfy single-failure-proof guidelines. No load drop analysis is required.

(6) The head stud rack is lifted by a single-failure-proof sling selected in accordance with NUREG-0612, Section 5.1.6(1)(b). The head stud rack has a single lifting point. The head stud rack is not carried over the spent fuel pool, and therefore cannot impact irradiated fuel. Administrative controls will be used to ensure the head stud rack lift height above the refueling floor is minimized. No safety related, safe shutdown, or decay heat removal equipment is located at the refueling floor elevation within the load path for the head stud rack. A postulated drop of the head stud rack (2.1 tons) is not expected to penetrate the massive refueling floor which is designed to support the heavier shield plugs, pool plugs, RPV head, etc.

Concrete spalling is not expected as the heavy load being considered is relatively light, and the bottom of the refueling floor is steel decking which would contain any concrete spalling. If local concrete spalling of the refueling floor were postulated along with impact and damage to equipment and piping on the elevation below, safe shutdown functions would not be affected as the equipment and piping are not required for safe shutdown, or redundant systems not affected by the postulated load drop are available.

HCGS

TABLE 9.1-20

The head stud rack is only carried over the RPV when the RPV head is in place. The probability of a postulated drop of the head stud rack (2.1 tons) is considered small, and would not affect the more massive RPV head (97 tons) and reactor pressure vessel. Also, a drop of the head stud rack would be enveloped by a postulated RPV head drop for which a General Electric analysis has shown to be acceptable and meets the evaluation criteria of NUREG-0612, Section 5.1.

(7) Same as Note (6) except that the RPV stud tensioner has four lift points. A postulated drop of the RPV stud tensioner, while not considered likely, satisfies the four evaluation criteria of NUREG-0612, Section 5.1 for the same reasoning as discussed in Note (6).

(8) The refueling bellows guard ring is lifted by a single-failure-proof sling selected in accordance with NUREG-0612, Section 5.1.6(1)(b). The 10-ton guard ring, if assumed to drop, would dissipate much of the energy through deformation of the circular guard ring. A postulated drop of the refueling bellows guard ring, while not considered likely, satisfies the four evaluation criteria of NUREG-0612, Section 5.1 for the same reasoning as discussed in Note (6).

(9) The flux monitoring shipping crate is lifted by a single-failure-proof sling selected in accordance with NUREG-0612, Section 5.1.6(1)(b). Administrative controls will be used to ensure the lift height above the refueling floor is minimized. No safety-related, safe shutdown, or decay heat removal equipment is located at the refueling floor elevation within the load path for the shipping crate. The flux monitor shipping crate is not carried over the spent fuel pool or RPV.

A postulated drop of the shipping crate (2.5 tons) is not expected to penetrate the massive refueling floor which is designed to support the heavier shield plugs, pool plugs, RPV head, etc. Concrete spalling is not expected as the heavy load being considered is relatively light, and the bottom of the refueling floor is steel decking which would contain any concrete spalling. If local concrete spalling of the refueling floor were postulated along with impact and damage to equipment and piping on the elevation below, safe shutdown functions would not be affected as the equipment and piping are not required for safe shutdown, or redundant systems not affected by the postulated load drop are available.

In summary, an analysis of a postulated drop of the flux monitor shipping crate demonstrates that the four evaluation criteria of NUREG-0612, Section 5.1 are satisfied.

HCGS

TABLE 9.1-20

(10) The channel handling boom crane is lifted by the auxiliary hook of the polar crane. No lifting device is necessary as the handling boom crane connects directly to the auxiliary hook of the polar crane. A postulated drop of the channel handling boom crane satisfies the four evaluation criteria of NUREG-0612 for the same reasoning as discussed in Note (9).

DSER Open Item No. 166 (Section 12.3)

AIRBORNE RADIOACTIVITY MONITOR POSITIONING

The applicant should clarify how he intends to use the ventilation monitors to accurately monitor plant iodine levels when the air being monitored by these monitors has been filtered through the plant HEPA and charcoal filter banks.

RESPONSE

FSAR Section 12.3.4.2.2 has been revised to address how HCGS intends to accurately monitor particulates and iodine from any compartment which has a possibility of containing airborne radioactivity and which normally may be occupied by personnel, taking into account dilution in the ventilation system.

taps are located in the ducts next to the detectors so that grab samples can be taken.

Additional mobile samplers with monitoring detectors that are displayed, controlled, and recorded by the CRP are provided for use if needed.

More details about airborne radioactive material sampling and monitoring are included in Section 11.5.

The above described airborne radioactive material monitoring equipment and procedures are used to meet the applicable parts of Regulatory Guides 1.21, 1.97, 8.2, 8.8, 8.12, and ANSI N13.1-1969.

Acceptance Criteria II.B.17 of standard review plan 12.3 - 12.4 provides criteria for the establishment of locations for fixed continuous area gamma radiation monitors. The specific document referenced is ANSI/ANS-HPSSC-6.8.1-1981. The locations and numbers of monitors used at HCGS are not in full compliance with this standard. The location of these monitors are in the vicinity of personnel access areas only. These locations are based on the dose assessment and operating experiences from other nuclear power plants. In addition, these locations were finalized prior to the issuance of this standard and provide an acceptable method of monitoring area radiation levels.

Insert

delete

~~Acceptance Criterion II.4.b.3 requires ventilation monitors to be~~ placed upstream of the HEPA filters. HCGS design places the ventilation monitors downstream of the HEPA filter in order to assess the plant's effluents. This is achieved best at this location as:

a. It is more efficient to have a single monitoring point rather than multiple points.

b. The instrument is sufficiently sensitive to ensure compliance with technical specification release limits.

c. The ventilation effluent monitors referred to above and the HVAC in line monitors (see P&IDs in Section 9.4) are scintillation detectors. These monitors are used ~~to detect gross activity and as such will indicate~~

DSER OPEN ITEM 166 (REV.2)

delete

~~increases in airborn radioactivity concentrations. Maintenance of iodine concentration within 10 MPC-hours will be assured by the use of several methods including these monitors, in-plant surveys, and portable particulate and iodine sampling monitors. Grab samples may be obtained from the duct systems or the room air by using the portable samplers. These samples are then analyzed in the laboratory by multichannel analyzer (MCA). (See Section 12.5 for further information about MCA). Therefore, particulate and iodine sampling monitors are not provided upstream of the HEPA filters.~~

12.3.5 REFERENCES

12.3-1 J.J. Martin and P.H. Blichert-Toft, "Radioactive Atoms, Auger Electrons, and X-Ray Data," Nuclear Data Tables, Academic Press, October 1970.

12.3-2 J.J. Martin, Radioactive Atoms Supplement 1, ORNL 4923, Oak Ridge National Laboratory, August 1973.

12.3-3 W.W. Bowman and K.W. MacMurdo, "Radioactive Decays Ordered by Energy and Nuclide," Atomic Data and Nuclear Data Tables, Academic Press, February 1970.

12.3-4 M.E. Meek and R.S. Gilbert, Summary of X-Ray and Gamma- Ray Energy and Intensity Data, NEDO-12037, General Electric, January 1970.

13.3-5 C.M. Lederer, et al, Table of Isotopes, 6th edition, John Wiley, New York, 1967 (1st corrected printing March 1968).

12.3-6 D.S. Duncan and A.B. Spear, "Grace 1 - An IBM 704-709 Program Design for Computing Gamma Ray Attenuation and Heating in Reactor Shields," Atomics International, NAA-SR-3719, June 1959.

12.3-7 D.S. Duncan and A.B. Spear, "Grace 2 - An IBM 709 Program for Computing Gamma Ray Attenuation and

DSER OPEN ITEM 166 (REV. 2)

Insert

Acceptance Criterion II.4.b.3 requires ventilation monitors to be placed upstream of HEPA filters. The HCGS design places scintillation detectors in ducts that are tributary to the release vent in order to provide warning of increased releases within the plant. These instruments detect increases in the gross noble gas concentrations of the effluent. Hence, placement of the detectors relative to HEPA and/or charcoal filters does not significantly affect their response. Since releases of iodines and particulates will be accompanied by much larger releases of noble gases, the changes in ventilation monitor readings provide indication of a change in airborne activity concentration in one or more of the plant's areas. If an increase is detected, its source and magnitude will be determined using portable samplers.

Normally occupied non-radiation areas in the plant do not have potential for significant airborne concentrations of particulates and iodine during plant operation because:

a. The ventilation systems are designed to prevent the spread of airborne radioactivity into normally occupied areas.

b. Highy radioactive piping/components are not located in normally occupied areas.

Certain activities, such as refueling, solid waste handling, or turbine teardown may increase the possibility of encountering significant airborne activities in some normally occupied areas. Continuous local airborne monitoring will be provided during these activities, as needed.

Exposure of personnel to high concentrations of airborne activity in radiation areas will be prevented through in-plant surveys and these portable particulate and iodine sampling monitors prior to personnel entrance. Continuous monitoring will be provided as required by area conditions and the nature of the entry. Administrative control will prevent inadvertent entry of personnel into normally unoccupied areas (Zone III and above). The provisions discussed above ensure that personnel will not be inadvertently exposed to significant concentrations of airborne activity. INSERT A

DELETE

~~Therefore~~, continuous ventilation radioactivity monitors capable of ~~detecting~~ 10 MPC-hrs ~~of particulate and iodine~~ from any ~~normally occupied compartments are not~~ provided as ~~permanently~~ installed equipment.

INSERT A

The location of portable monitors, capable of detecting 10 MPC-hours of particulates and iodines, which will be positioned within the station to provide supplemental inplant monitoring of particulates and iodine levels will be provided by July 1, 1985. The positioning of supplemental continuous air monitors is part of the Radiation Protection Program and a July 1, 1985 date is consistent with finalizing other details of the program (i.e., instrument and equipment calibration). The location, quantity, and monitor type will be provided at that time.

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Rev 1

DSER Open Item No. 223 (DSER Section 8.2.2.3) **Question 430.4**

INDEPENDENCE OF OFFSITE CIRCUITS BETWEEN THE SWITCHYARD AND THE CLASS 1E BUSSES

The Hope Creek design provides two immediate access offsite circuits between the switchyard and the 4.16 kv Class 1E busses. It is the staff position that these two circuits be physically separate and independent such that no single event can simultaneously affect both circuits in such a way that neither can be returned to service in time to prevent fuel design limits or design conditions of the reactor coolant pressure boundary from being exceeded. The physical separation and independence of these two circuits from and including station service transformers 1AX501 and 1BX501 to the 4.16 kv Class 1E busses has not been described or analyzed in the FSAR.

By Amendment 4 to the FSAR, the applicant implied, in response to a request for information, that the offsite circuits are non-Class 1E and thus do not have to be physically separated in accordance with the requirements of Criterion 17 of Appendix A to 10CFR50. The staff finds this interpretation to be unacceptable.

RESPONSE

The response to Question 430.4 has been revised to provide a drawing showing the physical routing of the two offsite circuits between the transformers and the Class 1E busses.

<INSERT A>

430.4

Insert A

Station service transformers 1AX501 and 1BX501 are provided with individual water spray systems and are separated from each other by a 1-hour fire barrier. Each transformer has a collection dike and drainage outlet for collecting transformer oil spills and fire suppression system water and draining it to the oily waste drainage system. The drainage outlet for each transformer is designed to drain the entire volume of oil from the transformer plus the maximum flow of water from the automatic water spray system.

The station service transformers water spray sprinkler system will be modified to provide sprinkler coverage on the crossover of the two non-segregated buses.

The non-segregated bus ducts are designed and constructed for outdoor adverse weather conditions (rain, ice, etc.). The bus ducts are designed as per ANSI standard C37.20-1969/C37.20C-1974 Section 8.2.2.4 Watertight tests, and, therefore water from the sprinkler system of one transformer will not endanger the operation of the non-segregated bus of the other transformer.

These design features ensure that a station service transformer fire can not damage the bus duct from the other transformer and cause a loss of both offsite sources of power.

QUESTION 430.4 (SECTION 8.2)

The Hope Creek design provides two immediate access offsite circuits between the switchyard and the 4.16 kV Class 1E buses. It is the staff position that these two circuits be physically separate and independent such that no single event can simultaneously affect both circuits in such a way that neither can be returned to service in time to prevent fuel design limits or design conditions of the reactor coolant pressure boundary from being exceeded. The physical separation and independence of transformers 1AX501 and 1BX501 to the 4.16 kV Class 1E buses has not been described or analysed in the FSAR. Provide the description and analysis and justify areas of noncompliance with the above staff position. The analysis should include separation and independence of control and protective relaying circuits as well as the power circuits.

RESPONSE

Power to the station service transformers comes from separate and opposite sides of the 13.8 kV ring bus. These are run in separate duct bank manholes to each respective transformer. In each duct bank the lines are enclosed in PVC conduit and encased in concrete.

Figure 8.3-5 shows that each of the four 4.16 kV Class 1E switchgear buses is supplied from two offsite (preferred) power sources and one onsite standby diesel generator (SDG). The offsite power to these buses is supplied from station service transformers 1AX501 and 1BX501 by non-segregated phase buses that are insulated and enclosed in metallic ducts. The non-segregated phase buses from the station service transformers to 4.16 kV Class 1E switchgear are designated as non-Class 1E.* The onsite power to the 4.16 kV class 1E bus is supplied from its associated SDG. The cables and the raceways associated with it are designated as Class 1E.

The non-Class 1E, non-segregated phase buses carrying the offsite power to 4.16 kV Class 1E switchgear buses are separated from Class 1E raceways of the onsite SDG power supply in accordance with the requirements of Regulatory Guide 1.75.

INSERT 'A' (NEXT PG) ~~Station service transformers 1AX501 and 1BX501 are separated and protected by concrete walls.~~

Analysis of circuitry independence and common mode failures are discussed in the response to Question 430.5.

INSERT * THE TWO NON-SEGREGATED BUSES ARE SEPARATED FROM EACH OTHER AS SHOWN ON FIGURE 430.4-1

INSERT 'A'

STATION SERVICE TRANSFORMERS 1AX501 AND 1BX501 ARE PROVIDED WITH INDIVIDUAL WATER SPRAY SYSTEMS AND ARE SEPARATED FROM EACH OTHER BY A 1-HOUR FIRE BARRIER EACH TRANSFORMER HAS A COLLECTION DIKE AND DRAINAGE OUTLET FOR COLLECTING TRANSFORMER OIL SPILLS AND FIRE SUPPRESSION SYSTEM WATER AND DRAINING IT TO THE OILY WASTE DRAINAGE SYSTEM

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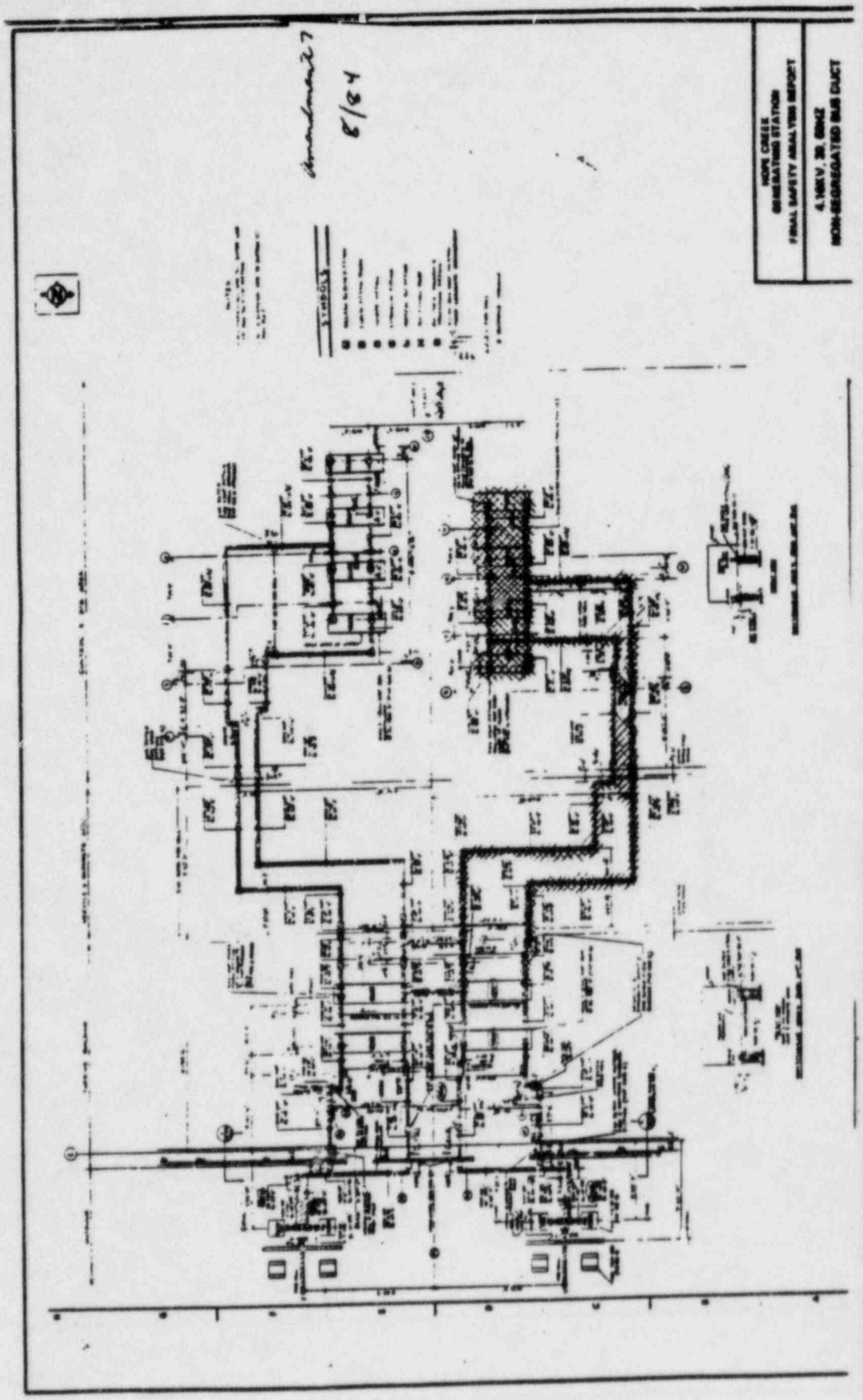

DSER OPEN ITEM 223

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DSER Open Item No. 233 (DSER Section 8.3.3.4.1) QUESTION 430,13

PERIODIC SYSTEM TESTING

Description of compliance to Section 6.4, Periodic System Tests, of IEEE Standard 308-1974, had not been included in Section 8.1.4.6 of the FSAR. By Amendment 4 to the FSAR, the applicant provided the following description of compliance: "Periodic system tests shall be performed using written procedures which will be designed to demonstrate system performance. The frequency of testing shall be governed by the frequencies specified in the Technical Specifications."

The following periodic system tests are required by Section 6.4 of IEEE Standard 308-1974 in order to demonstrate:

(1) The Class 1E loads can operate on the preferred power supply.

(2) The loss of the preferred power supply can be detected.

(3) The standby power supply can be started and can accept design load within the design basis time.

(4) The standby power supply is independent of the preferred power supply.

Pending inclusion of each of these tests in the Hope Creek Technical Specifications, the staff concludes that periodic system testing will comply with the guidelines of Section 6.4 of IEEE Standard 308-1974. This testing meets the requirements of GDC 17 and 18 and is acceptable.

RESPONSE

This item is considered as closed since the Standard Technical Specification Section 4.8.1.1.2e includes the above tests as part of the diesel generator testing every 18 months during shutdown. (The Hope Creek Technical Specifications, when issued, are based on the Standard Technical Specifications.)

Specifically, the above test requirements are included in the Standard Technical Specification (STS) as described below:

(1) STS Section 4.8.1.1.2e.11 requires, in part, verifying transfer of the diesel generator's emergency loads to the offsite power source (preferred power supply). STS Section 4.8.1.1.2e.12 requires verification that emergency loads are automatically energized from the offsite power source.

(2) STS Section 4.8.1.1.2e.4 and .7 require, in part, verifying deenergization of the emergency busses upon simulating a loss of offsite power. The deenergization of the bus will initiate control room alarms that monitor the incoming breaker position. STS Section 4.3.3.1 (Table 4.3.3.1-1,

Item 5) requires testing and calibration of emergency bus undervoltage devices.

(3) STS Section 4.8.1.1.2e.7 requires, in part, verifying that the diesel generator starts and energizes the shutdown loads.

(4) STS Section 4.8.1.1.2e.4 and .7 require, in part, verifying deenergization of the emergency busses upon simulating a loss of offsite power and that the busses are energized by the diesel generator. Additionally, the STS Section 4.8.1.1.2e will be modified for the HCGS Technical Specifications to verify the independence of the standby and preferred power sources by functional testing of the associated feeder breaker interlocks.

Rev 1

DSER Open Item No. 235 (DSER Section 8.3.1.5)

DIESEL GENERATOR LOAD ACCEPTANCE TEST

Position C.2.a(2), of Regulatory Guide 1.108, requires that the preoperational and periodic tests demonstrate proper operation of the diesel generator for design accident loading sequence to design load requirements. Section 1.8.1.9 of the FSAR states that for preoperational testing actual loads are started but may not duplicate their design basis condition. This statement implies exception to the above position. Justification for non-compliance with the guidelines of Regulatory Guide 1.108 will be pursued with the applicant, and the results of the stuff review will be reported in a supplement to this report.

RESPONSE

The response to Question 430.15 has been revised to clarify compliance with Regulatory Guide 1.108.

QUESTION 430.15 (SECTION 8.3.1)

In Sections 1.8.1.9 and 8.1.4.2 of the FSAR. You state (1) that preoperational testing at Hope Creek does not meet the guidelines of position C3 of Regulatory Guide 1.9 (revision 1), (2) predicted loads are verified by testing; however, loads that cannot be tested are verified by analysis or by comparison, and (3) for preoperational testing, actual desgin loads are started but may not duplicate their design basis condition. The above statement imply (1) that the diesel generators at Hope Creek will not be preoperationally or periodically tested to demonstrate their capacity and capability to operate properly when subject to design load, (2) that the guidelines of position C.2.a(2) of Regulatory Guide 1.108 (revision 1) will not be followed, and (3) that the design does not meet the requirements of criterion 17 of Appendix A to 10 CFR 50. In Section 8.1.4.20 of the FSAR provide justification for noncompliance

RESPONSE

~~Section 1.8.1.9 has been revised to delete the clarification to position C.3 of Regulatory Guide 1.9, Revision 1. The preoperational test program at HCGS for diesel generator testing will follow the guideline of Regulatory Guides 1.9 and 1.108, as shown in Sections 14.2.12.1.30 and 14.2.12.1.47. One exception to Regulatory Guide 1.108 has been taken as stated in Section 1.8.1.108 and discussed in response to Question 640.10~~

~~Periodic testing of the SDGs, at the required 18 month intervals, will be performed using written procedures in accordance with the requirements of the Hope Creek Technical Specifications. Section 8.1.4.20 has been revised to reflect this response.~~

See attached response

DSER No. 235

Question 430.15

Response

Section 1.8.1.9 has been revised to delete the clarification to position C.3 of Regulatory Guide 1.9, Revision 1. The preoperational test program at HCGS for diesel generator testing will follow the guidelines of Regulatory Guides 1.9 and 1.108, as shown in Sections 14.2.12.1.30 and 14.2.12.1.47.

Periodic testing of the SDGs, at the required 18 month intervals, will be performed using written procedures in accordance with the requirements of the Hope Creek Technical Specifications. Sections 1.8.1.108 and 8.1.4.20 have been revised to reflect this response.

Position C.2.a(2) of Regulatory Guide 1.108 is met with the exception discussed and justified in 1. below:

1. During the preoperational test phase, the proper design accident loading sequence will be demonstrated by the test described in Section 14.2.12.1.47. This test will verify the ability of the SDG to start and accept the sequenced design loads as specified in Table 8.3-1 while maintaining voltage and frequency within specified limits. Because this test will not provide ECCS flows to the reactor vessel, the ECCS pumps of RHR and core spray will not be delivering their design flows during this test condition. Though the testing does not duplicate the exact functional loads of an actual LOCA condition, the diesel generator will be loaded to the same kW output as in an actual LOCA. This is due to the higher density of the relatively cold water that the RHR pump will be pumping at rated flow in its recirculation mode. The real difference between the test functional loads and the actual LOCA functional loads is that in an actual event there is a transitory load from motor operated valves. This transitory load is offset by the higher RHR pump load of the test.

2. For periodic testing required by the Hope Creek Technical Specifications, the test per this regulatory position will be performed during shutdown. This test will simulate, separately, a loss of offsite power, and a loss of offsite power plus a LOCA condition, to verify the SDGs' ability to start and accept the sequenced design loads.

See Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

1.8.1.107 Conformance to Regulatory Guide 1.107, Revision 1, February 1977: Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures

Regulatory Guide 1.107 is not applicable to HCGS.

1.8.1.108 Conformance to Regulatory Guide 1.108, Revision 1, August 1977: Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants

Although Regulatory Guide 1.108 is not applicable to HCGS, per its implementation section, HCGS complies with it, with the following exception:

~~Position C.2.a(5) requires that the accident loading sequence to design load requirements be performed directly after the 24-hour run. This does not test the sequencing controls under a more severe condition than if sequentially loaded at an earlier or later period. A restart simulating loss of ac power can be performed directly after the 24-hour run. Sequencing, however, will be performed when the loads can be lined up for operation and all four diesels are available.~~

1.8.1.109 Conformance to Regulatory Guide 1.109, Revision 1, October 1977: Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I

HCGS complies with Regulatory Guide 1.109.

For further discussion, see Chapter 15.

INSERT A
ATTACHED
REF. Q 430.15
DSER

INSERT A

Position C.2.a(2) of Regulatory Guide 1.108 is met with the exception discussed and justified in 1. below:

1. During the preoperational test phase, the proper design accident loading sequence will be demonstrated by the test described in Section 14.2.12.1.47. This test will verify the ability of the SDG to start and accept the sequenced design loads as specified in Table 8.3-1 while maintaining voltage and frequency within specified limits. Because this test will not provide ECCS flows to the reactor vessel, the ECCS pumps of RHR and core spray will not be delivering their design flows during this test condition. Though the testing does not duplicate the exact functional loads of an actual LOCA condition, the diesel generator will be loaded to the same kW output as in an actual LOCA. This is due to the higher density of the relatively cold water that the RHR pump will be pumping at rated flow in its recirculation mode. The real difference between the test functional loads and the actual LOCA functional loads is that in an actual event there is a transitory load from motor operated valves. This transitory load is offset by the higher RHR pump load of the test.

2. For periodic testing required by the Hope Creek Technical Specifications, the test per this regulatory position will be performed during shutdown. This test will simulate, separately, a loss of offsite power, and a loss of offsite power plus a LOCA condition, to verify the SDGs' ability to start and accept the sequenced design loads.

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DSZR Open Item No. 241 (DSER Section 8.3.1.10)

LOAD ACCEPTANCE TEST AFTER PROLONGED NO LOAD OPERATION OF THE DIESEL GENERATOR

Section 6.4.2 of IEEE Standard 387-1977 requires, in part, that the load acceptance test consider the potential effects on load acceptance after prolonged no load or light load operation of the diesel generator. This capability should be demonstrated over the full range of ambient air temperatures that may exist at the diesel engine air intake.

By Amendment 4 to the FSAR, the applicant indicated that this diesel generator capability is being reviewed by the diesel engine manufacturer and that additional information with respect to the diesel generators capability will be provided at a later time. This item will continue to be pursued with the applicant.

RESPONSE

The response to Question 430.22 has been revised to indicate that the requested information is furnished in the response to Question 430.145.

QUESTION 430.22 (SECTION 8.3.1)

Section 6.4.2 of IEEE Standard 387-1977 requires, in part, that the load acceptance test consider the potential effects on load acceptance after prolonged no load or light load operation of the diesel generator. Provide the results of load acceptance tests or analysis that demonstrates the capability of the diesel generator to accept the design accident load sequence after prolonged no load operation. This capability should be demonstrated over the full range of ambient air temperatures that may exist at the diesel engine air intake. If this capability cannot be demonstrated for minimum ambient air temperature conditions, describe design provision that will assure an acceptable engine air intake temperature during no load operation.

RESPONSE

See the responses to Question 430.111 and 430.145 for the information requested above.

The Hope Creek diesel generators can accommodate a full load acceptance test per IEEE 387-1977 after a no load operation of the diesel generator.

A full load acceptance test per IEEE 387-1977 will be performed after an uninterrupted no load operation of four hours on the diesel generator. The four hours of unloaded operation is considered to be a realistic time based on expected operation of the diesel generators.

Station Operating Procedures will be provided to assure that after a cumulative four hours of operation at light load, i.e., less than 20% of rated, on any diesel, that diesel will be operated for one hour at a minimum of 50% rated load as per the diesel manufacturer's recommendations.

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DSER Open Item No. 242 (DSER Section 8.3.2.1)

COMPLIANCE WITH POSITION 1 OF REGULATORY GUIDE 1.128

Sections 1.8.28 and 8.1.4.22 of the FSAR indicates that the battery room ventilation system has the capability to limit hydrogen concentrations to less than 2 percent by volume within the battery room area but does not have the capability to limit hydrogen concentration to less than 2 percent by volume at any location within the battery area in accordance with the guidelines of position C1 of Regulatory Guide 1.128.

By Amendment 4 to the FSAR, the applicant, in response to a request for information, indicated that even though the ventilation exhaust duct is located just below the ceiling, there is sufficient air mixing within the battery area to limit hydrogen accumulation. Clarification of this item will be pursued with the applicant.

RESPONSE

See Attached response.

DSER Open Item 242 (DSER Section 8.3.2.1) QUESTION 430.23

RESPONSE

FSAR Section 9.4.5 has been revised to provide the requested information.

The mixing capability of the battery room ventilation system to assure that localized concentrations of hydrogen will not exceed the 2% level will be tested. This test will sample the battery room's air at three levels within the room. These levels shall be in the range of 1) the floor to 1/3 the height of the room, 2) 1/3 the room height to 2/3 the room height, and 3) the ceiling of the room. Samples in the areas specified above shall be taken at each end and center of the room.

Each sample shall be tested to verify the uniformity of the hydrogen concentration over the three areas. These areas will be tested at the completion of the battery discharge test - recharge cycle with the control area battery exhaust system operative.

Should the above test indicate no detectable hydrogen concentrations, the test shall be repeated as above except that a suitable gas will be released in the vicinity of the battery racks as a measurement media. The amount of gas released as a measurement media will be approximately 120 ft^3 per hour which is equivalent to the greatest calculated hydrogen generation rate for any of the Class 1E battery rooms. Air samples shall be tested for uniformity of gas concentration to verify the proper operation of the battery room ventilation system. If the concentrations measured vary by greater than 2%, the ventilation duct will be modified to correct the mixing capability.

 DSER 242

This test will be performed prior to fuel load. Any modifications, if required, will be made prior to power testing.

b. Meet the specified cooling and ventilation requirements during normal, shutdown, and accident conditions without loss of function

c. Provide redundancy for active and passive components to meet the single failure criteria

d. Operate the redundant active components from separate Class 1E power sources

e. Provide missile protection for the equipment, ducts, and accessories

f. Provide tornado protection for redundant and separate fresh air intake ducts that penetrate to the outdoors

g. Meet Seismic Category I requirements.

9.4.1.1.5 Control Area Battery Exhaust System

The CABE system exhausts air from the battery rooms to ensure that hydrogen concentrations remain within acceptable limits.

The CABE system is safety-related and is designed to accomplish the following objectives during normal plant operation, as well as during abnormal conditions:

2% INSERT

a. Maintain hydrogen concentrations for all battery rooms below a ~~safe~~ 1% level. This is done in conjunction with the CERS system

b. Provide redundancy for active components to meet the single failure criteria

c. Operate the redundant active components from separate Class 1E power sources

d. Meet Seismic Category I requirements

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DSER Open Item No. 243 (DSER Section 8.3.3.1.3)

PROTECTION OR QUALIFICATION OF CLASS 1E EQUIPMENT FROM THE EFFECTS OF FIRE SUPPRESSION SYSTEM

For the design basis event "Fire protection system operation," it is the staff position that Class 1E systems and components located in areas with fire suppression systems should be capable and qualified to perform their function when subject to the effects of the subject design basis event (Sections 4.2 and 4.7 of IEEE Standard 308-1974).

By Amendment 4 to the FSAR, the applicant implied that the only Class 1E equipment located in a zone of influence for which automatic water sprinkling systems are installed in the lower cable spreading area. The only electrical equipment installed in this room are electrical cables that are qualified for water submergence. When the effluent from the fire suppression system is water, the staff concludes, based on the above information, that Class 1E systems are adequately protected or qualified for the subject design basis event. Protection or qualification of Class 1E equipment from effluents other than water will be pursued with the applicant.

RESPONSE

The response to Question 430.59 has been revised to provide discussion on effects of CO_2 effluent on electrical equipment.

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HCGS FSAR

QUESTION 430.59 (SECTION 8.3.1 and 8.3.2)

For the design basis event "Fire protection system operation", it is the staff position that Class 1E systems and components located in areas with fire suppression systems should be capable and qualified to perform their function when subject to the effects of the subject design basis event (Sections 4.2 and 4.7 of IEEE Standard 308-1974). Either provide a positive statement of compliance to this position in the FSAR or justify non-compliance.

RESPONSE

Permanent fire protection systems installed throughout the station have been analyzed to determine the effects of their operation on Class 1E equipment.

The Class 1E equipment has been protected or is qualified for the inadvertent actuation of the permanent fire suppression systems. Protection is provided by spray shields, drip covers and other features. In addition, protection is provided by fire protection system design. Such design features include designing the system to require manual initiation, e.g., by opening a normally closed block valve. Partial protection is also afforded by the seismic design of certain CO_2 systems to prevent seismically induced spurious actuation.

Qualification is provided by tests or analysis of water impingement from overhead sprinklers or by sealing equipment covers and conduit openings in accordance with the suggested guidance of IE Information Notice 83-14.

Cable Concentration Protection

A number of areas, of high cable concentration, in the plant are equipped with automatic preactuation systems in response to discussions with the NRC Chemical Engineering Branch and 10CFR50, Appendix R, requirements. In general, Class 1E equipment is not impinged by water sprays. Where the potential exists for spray impingement the equipment will be protected or qualified.

The Class 1E 4.17 kv switchgear, unit substations, motor control centers, uninterruptible power supplies, distribution panels, etc. are not located in the zones of influence of automatic water sprinkling systems.

Cable Spread Areas

A preaction water sprinkler system is installed in the lower cable spreading room and a manual water spray backup to the CO_2 is being added in the upper cable spreading room. The only electrical components installed in this room are electrical cables that are qualified to operate for any design basis event parameters for HCGS, including water submergence.

A carbon dioxide system protects the upper cable spreading room. In the upper cable spreading room (control equipment room mezzanine), electrical cables will be subjected to the CO_2 system effluent if there is a fire or inadvertent actuation; however, cable performance is not affected by the effluent.

Diesel Generators And Auxiliaries

A CO_2 system is installed in the diesel generator rooms. Inadvertent CO_2 discharge caused by a seismic event is prevented in that the system has been analyzed or tested for seismic events.

An inadvertent discharge in one of the diesel generator rooms will not affect any of the other diesel generators. In the room containing the CO_2 discharge the diesel engine and generator components are not expected to be adversely affected (although no specific qualification of all components for CO_2 discharge has been performed). If the engine is running at the time of an inadvertent discharge, the engine will likely continue to operate properly except that the actuation will also close the fire dampers in the recirculation ventilation system. Continued operation with the fire dampers shut will lead eventually to an elevated room temperature. Although the engine would likely be shutdown in a non-emergency, the operator could reset the fire dampers and restore ventilation to the unit and quickly restore the generator to service. The three remaining diesel generators are capable of safely shutting down the plant. Shutdown of the plant requires only two diesel generator units in the same mechanical division.

Diesel Fuel Storage Room

The diesel generator fuel oil storage rooms are protected by CO_2 total flooding systems. The CO_2 flooding is not expected to cause any detrimental effects to any Class 1E equipment, however, qualification for CO_2 environment has not been performed. However, the CO_2 systems for each room are seismically qualified to ensure spurious actuations will not be caused by a seismic event. The room is also protected by a manual deluge system. Since the system is manually initiated, spurious actuations are not postulated. Loss of one diesel generator fuel oil tank room would jeopardize the capability to refuel the diesel day tanks, however, the capability would still exist to fill the day tanks from the other fuel oil storage tanks.

Rev 1

DSER Open Item No. 244 (DSER Section 8.3.3.3.1)

ANALYSIS AND TEST TO DEMONSTRATE ADEQUACY OF LESS THAN SPECIFIED SEPARATION

The applicant, by Amendment 4 to the FSAR, provided a description of physical separation between redundant enclosed raceways (covered trays and open top raceways, and between non-Class 1E trays and Class 1E conduit, as follows:

1. In the cable spreading rooms, the main control room, relay room, and control equipment room, the separation is twelve inches (12") horizontal, and eighteen inches (18") vertical.

2. In all other plant areas, the separation is three feet horizontal and five feet vertical.

The applicant further stated that where the separation distances specified above can not be maintained, cable trays shall either be covered with metal tray covers or an analysis, based on test results, will be performed.

The staff concludes that the above separation meets the guidelines of Regulatory Guide 1.75 and is acceptable except for the following:

(1) The use of 18 versus 36 inches of separation between raceways is evaluated in Section 8.3.3.3.2 of this report, and

(2) The use of an analysis to justify less than specified separation will be pursued with the applicant.

RESPONSE

The response to Question 430.52 has been revised to provide the requested analysis. One copy of each of the following reports were attached for your use: On August 30, 1984

1) Wyle Laboraties, Test Report No. 56719, Dated November 20, 1980, prepared for Susquehanna Steam Electric Station for electrical wire and cable isolation barrier ~~test~~ materials test.

2) Franklin Institute Research Laboratories, ~~for~~ Dated March 30, 1977, prepared for Toledo Edison Company for Conduit Separation test Program.

F70(8)

QUESTION 430.52 (SECTION 8.3.1 and 8.3.2)

RESPONSE

Physical separation between raceways for the configurations requested in the above question are described below:

1. Redundant enclosed raceways (covered trays and flexible metallic conduits or rigid conduits) are separated from open top raceways by:

   a. Twelve inches (12") horizontally and eighteen inches (18") vertically in the cable spreading room, control equipment room, and the control equipment room mezzanine.

   b. Three feet (3') horizontally and five feet (5') vertically in all other plant areas.

In cases where the separation distances specified above can not be maintained, cable trays shall either be covered with metal tray covers or an analysis, based on test results, shall be performed to demonstrate compliance with the intent of Regulatory Guide 1.75. There are only three generic cases where analysis is used to justify lesser separation distances. These are identified and analyzed as follows:

- Conduit-to-conduit less than one (1) inch apart.

  Because the minimum of space limitations in some areas of the plant, the minimum separation distance of one inch between rigid steel conduits can not be maintained. The use of the conduits is limited to instrumentation to instrumentation control to control, and instrumentation to power feeder with maximum 120 Vac or 125 Vdc cables only. Wyle Test Report No. 56719, prepared for Susquehanna Steam Electric Station, showed that rigid steel conduits in contact with each other are acceptable barriers. The testing demonstrated that shorting of conductors in one conduit until failure did not affect the performance of the conductors in the other conduit or damage the conduit. In addition, Franklin Institute Research Laboratories (FIRL) performed similar testing for the Toledo Edison Company in 1977 with successful results. The test configuration and cables used conservatively bound the HCGS conditions; therefore, the limited cases where the HCGS separation has not been met in the installation are justified.

- Non-Class 1E conduit separation from Class 1E tray.

  In safety-related areas of the plant there are non-Class 1E rigid steel conduits within one inch of Class 1E tray. The non-Class 1E conduit contains only control, instrumentation or 120 Vac/125 Vdc power cables. The testing performed for the above projects demonstrated that the rigid steel conduit is an effective barrier for protection of any cabling. Therefore, the HCGS cases where the non-Class 1E conduit is not installed as required is justified by the previous testing.

° Metal-clad cable separation from Class 1E raceways.

Metal-clad cables, type MC, are used in non-Class 1E circuits only. The minimum separation between the metal-clad cable and Class 1E raceways (open top trays or conduits) is one (1) inch. The type MC cable is a factory assembly of one or more conductors, each individually insulated, covered with an overall insulating jacket and all enclosed in a metallic sheath of interlocking galvanized steel. The cable has passed the vertical flame test of IEEE 383-1974.

The above analysis identifies the cases on a generic level. The installation and inspection of raceways are ongoing and the specific cases where the analysis applies are documented on nonconformance reports that are part of the QA/QC program.

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DSER Open Item No. 247 (DSER Section 8.3.3.5.1)

CAPABILITY OF PENETRATIONS TO WITHSTAND LONG DURATION SHORT CIRCUITS AT LESS THAN MAXIMUM OR WORST-CASE SHORT CIRCUITS

Section 8.1.4.12 (Item a) of the FSAR indicates that the time-current capability of the 1000 Kcmil conductor penetration is greater than the maximum short circuit current and its duration. The maximum short circuit current and its duration does not equate to Regulatory Guide 1.63 requirement for maximum short-circuit current versus time condition that could occur.

A positive statement in the FSAR to the effect that the time-current capability of the subject penetration is greater than any versus only maximum short circuit current and results of tests that demonstrates this time-current capability will be pursued with the applicant.

RESPONSE

The response to Question 430.44 has been revised to provide the requested information.

DH/em
F70(8)

QUESTION 430.44 (SECTION 8.3.1 and 8.3.2)

Section 8.1.4.12 (Item a) of the FSAR indicates that the time-current capability of the 1000 Kc mil conduction penetration is greater than the calculated worst-case short circuit current and its duration. The worst-case short circuit current and its duration does not equate to Regulatory Guide 1.63 requirement for maximum short-circuit current vs. time condition that could occur.

Provide a positive statement in the FSAR to the effect that the time-current capability of the subject penetration is greater than any versus only-worst case - short circuit current and its duration. Provide results of tests that demonstrates the time-current capability of each penetrative to maintain containment integrity when subject to any short circuit current for its duration worst case design basis event environment.

RESPONSE

THE TIME-CURRENT CAPABILITY OF THE 1000 Kcmil PENETRATION IS GREATER THAN ANY SHORT CIRCUIT CURRENT VS TIME CONDITION THAT COULD OCCUR. SECTION 8.1.4.12 HAS BEEN REVISED TO INCLUDE THE ABOVE RESPONSE. EXTRACTS OF WESTINGHOUSE TEST REPORT PEN-TR-79-05, TECHNICAL REPORTS AND QUALIFICATION DATA FOR MEDIUM VOLTAGE MODULAR ELECTRICAL PENETRATION, IS FORWARDED UNDER A SEPARATE COVER.

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 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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DSER Open Item No. 253 (DSER Section 8.3.3.1.4)

COMMITMENT TO PROTECT ALL CLASS 1E EQUIPMENT FROM EXTERNAL HAZARDS VERSUS ONLY CLASS 1E EQUIPMENT IN ONE DIVISION

In Section 8.1.14.3.3 of the FSAR, it is stated that where neither compartmentalization nor the construction of barriers is possible (to protect Class 1E circuits or equipment from hazards such as pipe break, flooding, missiles, and fires) an analysis is performed to demonstrate that none of the hazards disables redundant equipment, conduits, or trays. Based on this statement, the staff concludes that at least one of the redundant Class 1E systems and components at Hope Creek need not be protected from external hazards. The design, thus, does not meet the protection requirement of Criteria 2 and 4, nor the single failure requirement of Criterion 17 of Appendix A to 10 CFR 50. Justification for non-compliance with Criteria 2, 4, and 17 will be pursued with the applicant.

RESPONSE

The response to Question 430.38 has been revised to delete the cited statement from Section 8.1.14.3.3 and to provide discussion of protection against hazards.

QUESTION 430.38 (SECTION 8.3.1 and 8.3.2)

In Sections 8.1.4.14.3.3 of the FSAR you state that where neither compartmentalization nor the construction of barriers is possible (to protect Class 1E circuits or equipment from hazards such as pipe break, flooding, missiles, and fires) an analysis is performed to demonstrate that none of the hazards disables redundant equipment, conduits, or trays. Based on this statement it appears that at least one of the redundant Class 1E systems and components at Hope Creek may not be protected from external hazards. The design, thus, does not meet the protection requirement of Criteria 2 and 4 nor the single failure requirement of Criterion 17 of Appendix A to 10CFR50. Justify non-compliance with Criteria 2, 4, and 17.

RESPONSE

Section 3.5 indicates that Class 1E equipment is protected from postulated missiles by use of plant arrangement or suitable physical barriers such that a single missile cannot simultaneously damage a critical system component and its backup system. This is accomplished by locating redundant systems in different areas of the plant or separation by missile-proof walls. There are no Class 1E electrical equipment and components that can be damaged by missiles generated externally to the plant.

Section 3.6.1.1 indicates that, as part of the design basis for protection against dynamic effects associated with the postulated rupture of piping, a single active component failure is assumed to occur in systems used to mitigate the consequences of the postulated piping rupture and to shut down the reactor. A thorough review of the plant using the design bases provided in Section 3.6.1.1 was conducted and no cases were found where the piping failure would prevent safe shutdown (Reference: Question/Response 410.23).

Section 8.1.4.14.3.3 has been revised to replace the statement on compartments and barriers with one that references Sections 3.5 and 3.6.

The HCGS separation review (hazard analysis) confirms that no external hazard originating in a non-safety related system or component can prevent safe shutdown of the plant, even when the loss of offsite power and the worst single active failure of any safety related system or component is assumed.

monitoring cables, boxes also shall not be considered in determing the required separation.

b. In case of open ventilated trays, redundant trays are separated by 3 feet horizontally and 5 feet vertically, respectively. If the redundant trays cannot be separated by the distances specified above, solid covers for trays are provided as designated in Section 6.1.4 of IEEE 384-1981.

Separation requirements between Class 1E and non-Class 1E circuits are the same as those required between redundant circuits.

8.1.4.14.3.3 Hazardous Areas

These are areas where one or more of hazards such as pipe break, flooding, missile, and fire can be postulated.

Routing of redundant Class 1E circuits or the locating of redundant Class 1E equipment in hazardous areas is avoided. The preferred separation between redundant Class 1E circuits or equipment in these areas is by a wall, floor, or barrier that is structurally adequate to shield redundant raceways from potential hazards in the area.

~~Where neither compartmentalization nor the construction of barriers is possible, an analysis is performed to demonstrate that no missile, fire, jet stream impingement, or pipe whip hazard disables redundant equipment, conduits, or trays. In no case, regardless of the distance of physical separation, are redundant equipment cable trays located in the direct line of sight of the same potential missile source.~~

The plant design for fire protection separation of electrical cables and equipment is reviewed against 10 CFR 50, Appendix R, which is discussed in Section 9.5.1.

Sections 3.5 and 3.6 describe the methods of protection against missiles and pipe ruptures.

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DSER Open Item No. 256 (DSER Section 8.3.2.3)

AUTOMATIC TRIP OF LOADS TO MAINTAIN SUFFICIENT BATTERY CAPACITY

Section 8.3.2.1.2.2 of the FSAR states that the Class 1E dc system has sufficient capacity to supply the required loads except Class 1E instrument and balance of plant computer ac power supply inverter loads for 4 hours without support from battery chargers. By Amendment 4 to the FSAR, the applicant indicated that the Class 1E instrument and balance of plant computer ac power supply inverter loads will be automatically disconnected after 40 and 60 minutes respectively. In addition, the applicant indicated that the automatic trip circuit is testable during normal plant operation.

The staff concludes that a design that automatically disconnects loads to assure sufficient battery capacity meets the capacity requirements of GDC 17 and is acceptable except for the following concerns.

1. Periodic and preoperational testing of the trip circuit.

2. Safety classification of automatic trip circuit.

3. Results of analysis which demonstrates that the auto disconnected loads have no safety function after the 40 and 60 minute time periods.

These concerns will be pursued with the applicant.

RESPONSE

For the information requested above see the responses to questions 430.28 and 430.29

QUESTION 430.28 (SECTION 8.3.2)

In Section 8.3.2.1.2.2 and on Figure 8.3-8 of the FSAR you state that the Class 1E instrument load and the non-Class 1E BOP computer load are disconnected after 40 and 60 minutes respectively from the time that battery chargers are lost. Provide description with electrical schematic drawings of the circuitry for disconnecting these loads. Describe the capability to test this circuitry during normal power operation.

RESPONSE

See the revised response to Question 430.29

Figure 430.28-1

DELETED

QUESTION 430.29 (SECTION 8.3.2)

In section 8.3.2.1.2.2 of the FSAR you state that the Class 1E dc system does not have sufficient capacity to supply the Class 1E instrument loads for more than 40 minutes. Provide reference to Section 7 of the FSAR where this 40 minute time for Class 1E instruments is described and analyzed.

RESPONSE:

All of the connected loads will be connected to the Class 1E battery systems for the complete 4 hour battery duty cycle. There will be no tripping of Class 1E loads that are powered from the Class 1E batteries.

The following change to Section 8.3.2.1.2.2 of the FSAR will be made. This change removes the statement that initiated Questions 430.28 and 430.29 and, consequently, these questions should be closed out.

4. MCC

(a) Bus

(1) Main horizontal bus: 600 A continuous rating, 10,000 A short-circuit bracing

(2) Vertical bus: 300 A continuous rating, 10,000 A short-circuit bracing

(b) Breakers

Molded-case breakers: 100 A frame size, 10,000 A interrupting capacity

8.3.2.1.2.2 Class 1E Batteries

A 125-V battery consists of a set of 60 shock-absorbing, clear-plastic cells of the lead-calcium type. Four of the six batteries are rated at 1800 ampere hour and the remaining two at 560 ampere hour at an 8-hour discharge rate based on the end terminal voltage of 1.75 V per cell at 77±5°F.

Each Class 1E battery bank has sufficient capacity to independently supply the required loads ~~except Class 1E ac instrument power supply and balance of plant (BOP) computer power supply~~, for 4 hours without support from battery chargers. ~~Class 1E ac instrument power supply and BOP computer can be supplied for 40 and 60 minutes, respectively from the time that battery chargers are lost~~. Th~~e~~s~~e~~ time interval~~s~~ ~~are~~ is sufficient to ensure that the Class 1E instrument ac power supply is uninterrupted during a loss of offsite power, because the battery chargers will be reenergized from Class 1E 480 V motor control centers once the standby diesel generators are started.

The initial battery capacity is 25% greater than required. This margin is consistent with the battery replacement criterion of 80% rated capacity given in IEEE 450-1975.

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DSER Open Item No. 257 (DSER Section 8.3.2.5)

JUSTIFICATION FOR A 0 TO 13 SECOND LOAD CYCLE

Table 8.3-7 of the FSAR indicates that each of the station battery duty cycle consist, in part, of a 0 to 13 seconds and a 13 to 120 second load periods. The basis and justification for separating loads into these two time periods for all modes of plant operation will be pursued with the applicant.

RESPONSE

Tables 8.3-7a,b,c, and d, have been revised to change the 13 seconds to 1 minute.

DSER OPEN ITEM 257

TABLE 8.3-7a

CHANNEL A 125-V DC LOADS
BATTERY 1AD411
DC SWITCHGEAR 10D410

Item	Load Description	Equipment Number	Equipment Rating: Full Load (Amps)	Equipment Rating: Load Inrush (Amps)	Load Cycle (Amperes): 0 to 1 min (1)	1 to 2 min	2 min to 10 min	10 min to 60 min	60 min to 240 min	~~10 min to 60 min~~	~~61 min to 240 min~~
1.	Plant PA system ac power supply inverter	10D496	180	—	56	56	56	56	61		
2.	Class 1E instrument ac power supply inverter	1AD481	180	—	94	94	94	94	105		
3.	Class 1E instrument ac power supply inverter	1AD482	180	—	73	73	73	73	80		
4.	125-V dc distribution panel	1AD417	-	-	-	-	-	-	-	~~-~~	~~-~~
a.	4.16-kV Class 1E swgr control power	10A401	1.2	120 ~~[illegible]~~	121.2	61.2	1.2	1.2	1.2	~~1.2~~	~~1.2~~
b.	Standby diesel generator field flashing power	1AC420, 421, 422, & 423	-	50	50	-	-	-	-	~~-~~	~~-~~
c.	480-V Class 1E unit substation control power	10B410 10B450	2.4	54 ~~[illegible]~~	56.4	2.4	2.4	2.4 ~~[illegible]~~	2.4	~~2.4~~	~~2.4~~
d.	Div I RHR & core spray vertical board	10C617	9.92	9.92	9.92	9.92	9.92	9.92	9.92	~~9.92~~	~~9.92~~
e.	Main steam inboard isolation valve relay vertical board	10C622	.68	.68	.68	.68	.68	.68	.68	~~[illegible]~~	~~[illegible]~~
f.	RPS trip system vertical board	10C609	0.4 ~~[illegible]~~	0.4 ~~[illegible]~~	0.4 ~~[illegible]~~	0.4 ~~[illegible]~~	0.4 ~~[illegible]~~	0.4 ~~[illegible]~~	0.4 ~~[illegible]~~	~~[illegible]~~	~~[illegible]~~
g.	250-V dc HPCI system MCC	10D251	.34 ~~[illegible]~~	~~[illegible]~~	15.2 ~~[illegible]~~	8.5 ~~[illegible]~~	11.1 ~~[illegible]~~	8.5 ~~[illegible]~~	13.2 ~~[illegible]~~	~~[illegible]~~	~~[illegible]~~
h.	Reactor recirc pump breaker trip	1AN205 1BN205	.2	10	10.2	.2	.2	.2	.2	~~.2~~	~~.2~~
i.	HPCI relay vertical board	10C620	5.92 ~~[illegible]~~	5.92 ~~[illegible]~~	5.92 ~~[illegible]~~	5.92 ~~[illegible]~~	5.92 ~~[illegible]~~	5.92 ~~[illegible]~~	5.92 ~~[illegible]~~	~~[illegible]~~	~~[illegible]~~
j.	Standby diesel generator control power	1AC420, 421, 422, & 423	5	5	5	5	5	5	5	~~5~~	~~5~~

DSER OPEN ITEM 257

TABLE 8.3-7a (cont)

Item	Load Description	Equipment Number	Equipment Rating: Full Load (Amps)	Equipment Rating: Load Inrush (Amps)	Load Cycle (Amperes): 0 to 1 MIN (1)	1 to 2 min	2 min to 10 min	10 min to 60 min	60 min to 240 min	[illegible] min to [illegible] min	[illegible] min to 2[illegible] min
h.	Reactor bldg exhaust room and pipe chase isolation dampers terminal box	1AC281	~~[illegible]~~ 4.5	~~[illegible]~~ 4.5	~~[illegible]~~ 4.5	~~[illegible]~~ 4.5	~~[illegible]~~ 4.5	~~[illegible]~~ 4.5	~~[illegible]~~ 4.5	[illegible]	[illegible]
L.	125-V dc swgr control power	1[illegible]	.7	~~[illegible]~~ .7	.7	.7	.7	.7	.7	2.6	.7
Total (amperes)					451.1	330.4	273	270.4	298.1	242.72 ~~[illegible]~~	240.72 ~~[illegible]~~
N.	REDUNDANT REACTIVITY CONTROL SYSTEM	10CL02	~~4~~ 8	~~8~~ 16	8	8	8	8	8	8	8

(1) BATTERY IS SIZED FOR THE TOTAL MAXIMUM CONCURRENT LOAD AT ANY INSTANT DURING THE FIRST MINUTE.

DSEB OPEN ITEM 257

TABLE 8.3-7b

CHANNEL B 125-V DC LOADS
BATTERY 1BD411
DC SWITCHGEAR 1BD410

Item	Load Description	Equipment Number	Equipment Rating: Full Load (Amps)	Equipment Rating: Load Inrush (Amps)	Load Cycle (Amperes): 0 to 1 min(1)	1 to 2 min	2 min to 10 min	10 min to 40 min	40 min to 60 min	60 min to 240 min	[illegible] min to 90 min	[illegible] min to [illegible] min	[illegible] min to 240 min
~~1.~~	~~BOP computer ac power supply inverter~~	~~1BD492~~	~~200~~		~~100~~	~~100~~	~~100~~	~~100~~	~~100~~	~~200~~	200	-	-
2.	Class 1E instrument ac power supply inverter	1BD481	180 ~~200~~		128 ~~100~~	128 ~~100~~	128 ~~100~~	128 ~~100~~	128 ~~100~~	142	-	-	-
3.	Class 1E instrument ac power supply inverter	1BD483	180 ~~200~~		73 ~~100~~	73 ~~100~~	73 ~~100~~	73 ~~100~~	73 ~~100~~	80	-	-	-
4.	125-V dc Class 1E distribution panel	1BD417	-	-	-	-	-	-	-	-	-	-	-
	a. 4.16-kV Class 1E swgr control power	1BB402	1.2	120 ~~120~~	121.2	61.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2
	b. Standby diesel generator field flashing power	1BG420, 421, 422 & 423	-	50	50	-	-	-	-	-	-	-	-
	c. 480-V Class 1E unit substations control power	1BB420, 1BB440	2.4	54 ~~105~~	56.4	2.4	2.4	2.4 ~~26.4~~	2.4	2.4	2.4	2.4	2.4
	d. Reactor recirc pump breakers trip	1CN205, 1DN205	.2	10	10.2	.2	.2	.2	.2	.2	.2	.2	.2
	e. Division II RHR & core spray vertical board	1BC610	8.12	8.12	8.12	8.12	8.12	8.12	8.12	8.12	0	0	0
	f. 250-V dc RCIC system MCC	1BD251	.34	-	8.96	6.2	6.89	7.58	6.89	8.27	2.3	2.3	2.3

DSER OPEN ITEM 257

TABLE 8.3-7b (cont)

Item	Load Description	Equipment Number	KVA	Equipment Rating: Full Load (Amps)	Equipment Rating: Load Inrush (Amps)	Load Cycle (Amperes): 0 to (1) 1 MIN	1 to 2 min	2 min to 10 min	10 min to 40 min	40 min to 60 min	60 min to 240 min	11 min to 60 min	60 min to 61 min	61 min to 240 min
g.	RPS trip system vertical board	1DC611	-	0.4	0.4	0.4	0.4	0.4	0.4	0.4	0.4	0.5	0.5	0.5
h.	Remote shutdown panel	1DC399	-	.42	0.42	0.42	0.42	0.42	0.42	0.42	0.42	8	8	8
i.	RCIC relay vertical board	1DC621	-	3.9	24	24	24	24	24	24	24	0.5	0.5	0.5
j.	Auto depress relay vertical board	1DC628	-	2.8	2.8	2.8	2.8	2.8	2.8	2.8	2.8	3	3	3
k.	Reactor bldg exhaust room isolation dampers terminal box	1DC701	.	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	0.08	0.08	0.08
l.	Standby diesel generator control power	1DC020, 021, 022 & 023	-	5	5	5	5	5	5	5	5	5	5	5
m.	125-V dc swgr control power	1DB0411	-	.7	.7	.7	.7	.7	.7	.7	.7	.7	2.6	.7
Total						443.3	326.5	267.2	267.9	267.2	289.6			
n.	Redundant reactivity control system	1DC002	-	8	16	8	8	8	8	8	8	8	8	8
o.	RCIC min flow bypass to suppr pool vlv	1RD-50-F019		1.6	-	1.6	1.6	1.6	1.6	1.6	1.6	1.5	1.5	1.5

(1) BATTERY IS SIZED FOR THE TOTAL MAXIMUM CONCURRENT LOAD AT ANY INSTANT DURING THE FIRST MINUTE

DSER OPEN ITEM 257

TABLE 8.3-7c

CHANNEL C 125-V DC LOADS
BATTERY 1CD411
DC SWITCHGEAR 1DD430

Item	Load Description	Equipment Number	Equipment Rating: Full Load (Amps)	Equipment Rating: Load Inrush (Amps)	Load Cycle (Amperes): 0 to 1 min (1)	1 to 2 min	2 min to 60 min	60 min to 240 min	12 min to 30 min	30 min to 61 min	61 min to 90 min	90 min to 91 min	91 min to 240 min
1.	Plant security system ac power supply inverter	0AD895	180 ~~200~~	-	158 ~~180~~	158 ~~180~~	158 ~~100~~	175 ~~200~~	180	200	200	200	200
2.	Class 1E instrument ac power supply inverter	1CD801	180 ~~200~~	-	95 ~~100~~	95 ~~180~~	95 ~~200~~	105 ~~200~~	100	-	-	-	-
3.	Deleted ~~RPS converter ac power supply inverters~~		~~200~~	~~-~~	~~180~~	~~180~~	~~180~~	~~180~~	~~180~~	~~200~~	~~200~~	~~-~~	~~-~~
4.	125-V dc distribution panel	1CD817	-	-	-	-	-	-	-	-	-	-	-
a.	4.16-kV Class 1E swgr control power	1DA403	1.2	120 ~~110~~	121.2	61.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2
b.	Standby diesel generator field flashing power	1CC420, 421, 422 & 423	-	50	50	-	-	-	-	-	-	-	-
c.	480-V unit substation control power	1DB410 1DB470	2.8	27 ~~105~~	29.4	29.4	2.8	2.4 ~~37.8~~	2.8	2.8	2.8	2.8	2.8
d.	125-V swgr control power	1DD430	.7	.7 ~~1.4~~	.7	.7	.7	.7	.7	2.6	.7	2.6	.7
e.	Division III RHR and core spray vertical board	1DC641	5.36 ~~6~~	5.36 ~~6~~	5.36 ~~6~~	5.36 ~~6~~	5.36 ~~6~~	5.36 ~~6~~	6	6	6	6	6
f.	Standby diesel generator control power	1CC420, 421, 422 & 423	5	5	5	5	5	5	5	5	5	5	5

DSER OPEN ITEM 257

RCGS FSAR

TABLE 8.3-7c (cont)

			Equipment Rating			Load Cycle (Amperes)								
Item	Load Description	Equipment Number	KVA	Full Load (Amps)	Load Inrush (Amps)	0 to 1 MIN (1)	1 to 2 min	2 min to 60 min	60 min to 240 min	12 min to 30 min	40 min to 41 min	41 min to 60 min	60 min to 61 min	61 min to 240 min
9.	Reactor bldg exhaust room and pipe chase isolation damper term box	1CC281	-	4.5 ~~[illegible]~~	4.5 ~~[illegible]~~	4.5 ~~[illegible]~~	4.5 ~~[illegible]~~	4.5 ~~[illegible]~~	4.5 ~~[illegible]~~	~~8.08~~	~~8.88~~	~~8.08~~	~~8.88~~	~~8.88~~
Total (amperes)						419.2	359.2	272.2	299.2	379.78 ~~[illegible]~~	221.68 ~~[illegible]~~	219.78 ~~[illegible]~~	~~[illegible]~~	~~[illegible]~~

(1) BATTERY IS SIZED FOR THE TOTAL MAXIMUM CONCURRENT LOAD AT ANY INSTANT DURING THE FIRST MINUTE.

USER OPEN ITEM 257

TABLE 8.3-7d

CHANNEL D 125-V DC LOADS
BATTERY 10D411
DC SWITCHGEAR 10D440

Item	Load Description	Equipment Number	Equipment Rating: Full Load (Amps)	Equipment Rating: Load Inrush (Amps)	Load Cycle (Amperes): 0 to 1 min (1)	1 to 2 min	2 min to 60 min	60 min to 240 min	12 min to 30 min	30 min to 31 min	31 min to 60 min	30 min to 61 min	31 min to 240 min
~~1.~~	~~200 computer ac power supply inverter~~	~~10D??3~~	~~200~~	~~-~~	~~180~~	~~180~~	~~180~~	~~180~~	180	200	200	-	-
2.	Class 1E instrument ac power supply inverter	10D481	180 ~~200~~	-	100 ~~120~~	100 ~~100~~	100 ~~100~~	110 ~~100~~	180	-	-	-	-
3.	RMSS computer ac power supply inverter	10D485	180 ~~200~~	-	145 ~~200~~	145 ~~200~~	145 ~~200~~	160 ~~200~~	180	200	200	200	200
4.	125-V dc distribution panel	10D417	-	-	-	-	-	-	-	-	-	-	-
a.	4.16-kV Class 1E swgr control power	10A404	1.2	120 ~~110~~	121.2	61.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2
b.	Standby diesel generator field flashing power	10C420, 421, 422 & 423	-	50	50	-	-	-	-	-	-	-	-
c.	480-V unit substation control power	10B440 10B480	2.4	27 ~~105~~	29.4	29.4	2.4	2.4 ~~37.4~~	2.4	2.4	2.4	2.4	2.4
d.	Remote shutdown panel	10C399	3.05 ~~8~~	3.05 ~~8~~	3.05 ~~8~~	3.05 ~~8~~	3.05 ~~8~~	3.05 ~~8~~	8	8	8	8	8
e.	Div IV RHR & core spray vertical board	10C640	5.13 ~~8~~	5.13 ~~8~~	5.13 ~~8~~	5.13 ~~8~~	5.13 ~~8~~	5.13 ~~8~~	8	8	8	8	8
f.	Main steam outboard isolation valve relay vertical board	10C621	.68 ~~8~~	.68 ~~8~~	.68 ~~8~~	.68 ~~8~~	.68 ~~8~~	.68 ~~8~~	8	8	8	8	8

TABLE 8.3-7d (cont)

Item	Load Description	Equipment Number	KVA	Equipment Rating: Full Load (Amps)	Equipment Rating: Load Inrush (Amps)	Load Cycle (Amperes): 0 to 1 MIN (1)	1 to 2 min	2 min to 60 min	60 min to 240 min	12 min to 40 min	40 min to 41 min	41 min to 60 min	60 min to 61 min	61 min to 240 min
g.	Auto depress. relay vertical board	10C631	-	1.6 ~~3~~	1.6 ~~3~~	1.6 ~~3~~	1.6 ~~3~~	1.6 ~~3~~	1.6 ~~3~~	3	3	3	3	3
h.	125-V dc swgr control power	10D440	-	.7	0.7 ~~3.5~~	.7	.7	.7	.7	.7	2.6	.7	2.6	.7
i.	Standby diesel generator control power	10DC420, 421, 422 & 423	-	5	5	5	5	5	5	5	5	5	5	5
j.	Reactor bldg exhaust room isolation dampers terminal box	10C281	-	4.5 ~~8.00~~	4.5 ~~8.00~~	4.5 ~~8.00~~	4.5 ~~8.00~~	4.5 ~~8.00~~	4.5 ~~8.00~~	8.00	8.00	8.00	8.00	8.00
Total (amperes)						416.3	356.3	264.3 ~~320.78~~	294.3 ~~620.78~~	576.78	838.68	836.78	338.68	236.78

(1) BATTERY IS SIZED FOR THE TOTAL MAXIMUM CONCURRENT LOAD AT ANY INSTANT DURING THE FIRST MINUTE.

TABLE 8.3-~~X~~ 10a

CHANNEL C 125-V DC LOADS
BATTERY 1CD447
DC SWITCHGEAR 10D436

Item	Load Description	Equipment Number	Equipment Rating: Full Load (Amps)	Equipment Rating: Load Inrush (Amps)	Load Cycle (Amperes): 0 to 60 MIN	60 TO 240	~~2 min to 10 min~~	~~10 min to 12 min~~	~~12 min to 60 min~~	~~40 min to 41 min~~	~~41 min to 60 min~~	~~60 min to 61 min~~	~~61 min to 240 min~~
1.	Class IE instrument ac power supply inverter	1CD482	~~210~~ 180		68	77							

TABLE 8.3-~~[illegible]~~ 10b

CHANNEL D 125-V DC LOADS
BATTERY 1DD447
DC SWITCHGEAR 1DD446

			Equipment Rating		Load Cycle (Amperes)								
Item	Load Description	Equipment Number	Full Load (Amps)	Load Inrush (Amps)	0 to, 60 min	60 to 240 min	~~2 min to 10 min~~	~~10 min to 12 min~~	~~12 min to 40 min~~	~~40 min to 41 min~~	~~41 min to 60 min~~	~~60 min to 61 min~~	~~61 min to 240 min~~
1.	Class 1E instrument ac power supply inverter	1DD482	~~200~~ 180		73	80							

TABLE 8.3-8

CHANNEL A 250-V DC LOADS
BATTERY 10D421
MOTOR CONTROL CENTER 10D251

		Equipment Rating				Load Cycle				
				Amperes						
Item	Load Description	Equipment Number	HP	Full Load	Inrush	0 to 1 min	1 to 8 min	8 to 9 min	9 to 50 min	50 to 51 min
1.	HPCI gland seal cond vac pump	10P216	1.5	5.8	17.5	~~5.8~~ 17.5	5.8	5.8	5.8	5.8
2.	HPCI turb aux oil pump	10P213	7.5	25.8	78	78	25.8	25.8	25.8	25.8
3.	HPCI vacuum tank cond pump	10P215	3	11	33	33	11	11	11	11
4.	HPCI test bypass to cond str tank	BJ-HV-F008	5.8	20	~~36~~ 131	-	-	-	-	-
5.	HPCI test bypass to cond str tank	AP-HV-F011	1.8	7.4	41	-	-	-	-	-
6.	HPCI min flow bypass to supp pool	BJ-HV-F012	1.8	7.4	41	41	-	-	-	41
7.	HPCI barom cond clg water sply valve	BJ-HV-F059	.33	2	11.3	11.3	-	-	-	-
8.	HPCI turb exhaust to supp pool	FB-HV-F071	1.8	7.4	41	-	-	-	-	-
9.	HPCI steam supply line to turbine	FD-HV-F001	4.3	17	87	87	-	-	-	-
10.	HPCI pump suction from cond str tank	BJ-HV-F004	.75	3	19.5	-	-	-	-	-
11.	HPCI pump suction from supp pool	BJ-HV-F042	.75	3	19.5	-	-	-	-	-
12.	HPCI pump discharge to RPV	BJ-HV-F007	10.8	40	~~250~~ 225	~~250~~(3) 225	-	-	-	-
13.	HPCI pump discharge to RPV	BJ-HV-F006	10.8	40	~~250~~ 225	~~250~~ 225	-	~~250~~ 225	-	~~250~~ 225
	TOTAL					~~767.8~~ 758.1	42.6	~~292.6~~ 307.9	42.6	~~379.6~~ 348.9
14.	HPCI PUMP DISCHARGE TO FEEDWATER LINE	BJ-HV-8278	1.6	7	40.3	40.3		40.3		40.3

(1) The MOV can be jogged at any time.
(2) Suction to HPCI pump is assumed to change over from condensate storage tank to suppression pool at the end of 2 1/2 hours.
(3) Valve under test.

TABLE 8.3-8 (cont)

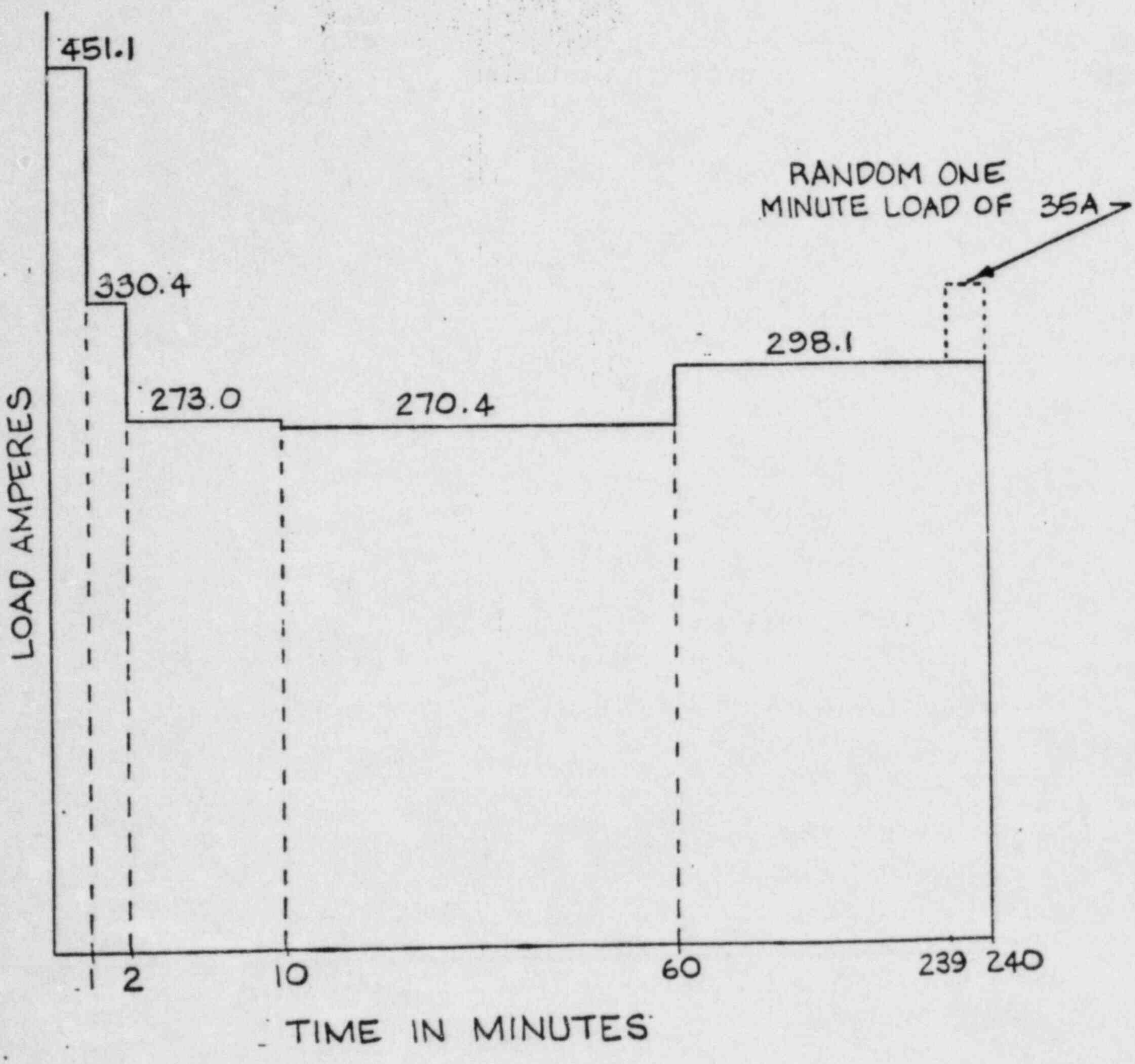
Item	Load Description	Equipment Rating: Equipment Number	Amperes / Load Cycle: 51 to 58 min	58 to 59 min	59 to 100 min	100 to 101 min	101 to 108 min	108 to 109 min	109 to 150 min	150 to 151 min	151 to 200 min
1.	HPCI gland seal cond vac pump	10P216	5.8	5.8	5.8	5.8	5.8	5.8	5.8	5.8	5.8
2.	HPCI turb aux oil pump	10P213	25.8	25.8	25.8	25.8	25.8	25.8	25.8	25.8	25.8
3.	HPCI vacuum tank cond pump	10P215	11	11	11	11	11	11	11	11	11
4.	HPCI test bypass to cond str tank	BJ-HV-F008	-	-	-	-	-	-	-	-	-
5.	HPCI test bypass to cond str tank	AP-HV-F011	-	-	-	-	-	-	-	41(1)	41(1)
6.	HPCI min flow bypass to supp pool	BJ-HV-F012	-	-	-	41	-	-	-	-	-
7.	HPCI barom cond clg water sply valve	BJ-HV-F059	-	-	-	-	-	-	-	-	-
8.	HPCI turb exhaust to supp pool	FB-HV-F071	-	-	-	-	-	-	-	-	-
9.	HPCI steam supply line to turbine	FD-HV-F001	-	-	-	-	-	-	-	19.5(2)	-
10.	HPCI pump section from cond str tank	BJ-HV-F004	-	-	-	-	-	-	-	19.5(2)	-
11.	HPCI pump suction from supp pool	BJ-HV-F042	-	-	-	-	-	-	-	-	-
12.	HPCI pump discharge to RPV	BJ-HV-F007	-	-	-	-	-	-	-	-	-
13.	HPCI pump discharge to RPV	BJ-HV-F006	-	~~250~~ 225	-	~~250~~ 225	-	~~250~~ 225	-	~~250~~ 225	-
14.	HPCI PUMP DISCHARGE TO FEEDWATER LINE	BJ-HV-F278	-	40.3	-	40.3	-	40.3	-	40.3	-
	Total (amperes)		42.6 ✓	~~292.6~~ 307.9	42.6 ✓	~~335.6~~ 348.9	42.6 ✓	~~292.6~~ 307.9	42.6	~~372.6~~ 387.9	83.6 ✓

TABLE 8.3-9

CHANNEL B 250-V DC LOADS
BATTERY 10D431
MOTOR CONTROL CENTER 10D261

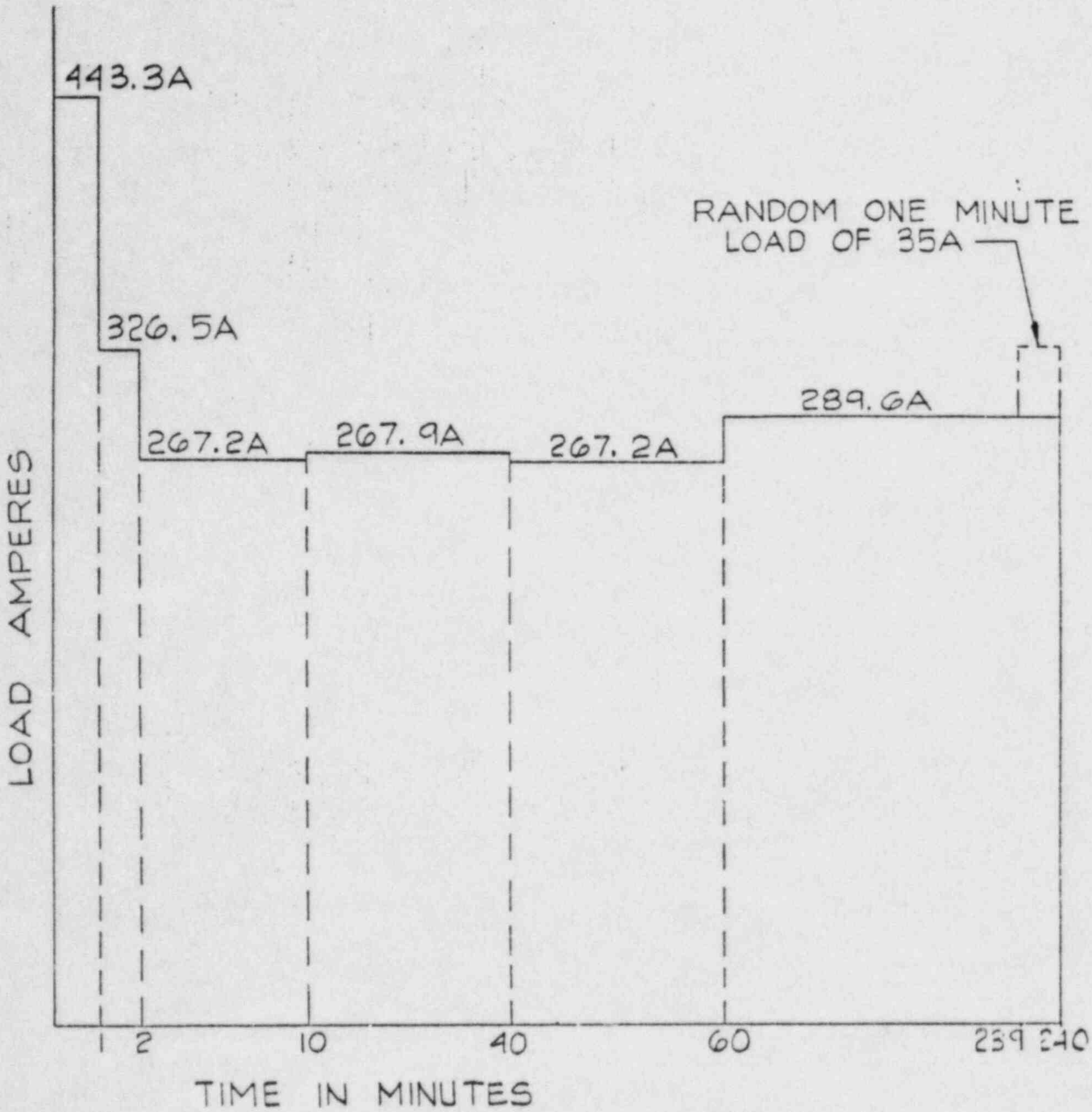
			Equipment Rating			Load Cycle								
				Amperes										
Item	Load Description	Equipment Number	HP	Full Load	Inrush	0 to 1 min	1 to 6 min	6 to 7 min	7 to 24 min	24 to 25 min	25 to 150 min	150 to 210 min	210 to 211 min	211 to 240 min
1.	RCIC gland seal vac pump	10P219	3	14.3	42.5	42.5	14.3	14.3	14.3	14.3	14.3	14.3	14.3	14.3
2.	RCIC vac tank cond pump	10P220	3	11	33	33	11	11	11	11	11	11	11	11
3.	RCIC turb exhaust to supp pool	FC-HV-F059	0.33	~~4~~ 1.9	11.3	-	-	-	-	-	-	-	-	-
4.	RCIC test line to cond storage tank	BD-HV-F022	0.75	3	19.5	-	-	-	-	-	-	-	-	-
~~5.~~	~~RCIC min flow bypass to suppression pool~~	~~BD-HV-F019~~	~~0.34~~	~~2~~	~~10.3~~	~~10.3~~	~~-~~	~~-~~	~~-~~	~~10.3~~	~~-~~	~~10.3 (1)~~	~~10.3 (1)~~	~~10.3 (1)~~
5 ~~6.~~	RCIC stm supply to turbine	FC-HV-F045	1.00	4	30	30	-	-	-	-	-	-	-	-
6 ~~7.~~	RCIC pump inlet from cond storage tank	BD-HV-F010	0.33	~~4~~ 1.9	11.3	-	-	-	-	-	-	-	11.3 (2)	-
7 ~~8.~~	RCIC pump inlet from supp pool	BD-HV-F031	0.33	2	11.3	-	-	-	-	-	-	-	11.3 (2)	-
8 ~~9.~~	RCIC pump outlet to RPV	BD-HV-F012	1.8	7.4	41	41 (1)	-	-	-	-	-	-	-	
9 ~~10.~~	RCIC pump outlet to RPV	BD-HV-F013	1.8	7.4	41	41	-	41	-	41	-	-	-	
10 ~~11.~~	RCIC vac pump disch to suppression pool	FC-HV-F0~~[illegible]~~ 60	0.14	~~1.4~~ .76	14	-	-	-	-	-	-	-	-	
11 ~~12.~~	RCIC barom cond cooling water supply	BD-HC-F046	0.33	~~1.3~~ 2.17	10.~~8~~ 9	10.~~8~~ 9	-	-	-	-	-	-	-	-
12 ~~13.~~	RCIC turb trip & throttle valve	FV-HV-4282	0.09	0.7	8.4	-	-	-	-	8.4 (1)	8.4 (1)	8.4 (1)	8.4 (1)	8.4 (1)
	Total (amperes)					~~208.1~~ 197.9	25.3 ✓	66.30 ✓	~~[illegible]~~ 25.3	~~85~~ 74.7	33.70 ✓	~~[illegible]~~ 33.70	~~[illegible]~~ 56.30	~~[illegible]~~ 33.7

(1) MOV can be jogged any time.
(2) Suction for the RCIC pump is assumed to change from condensate storage tank to suppression pool at the end of 3-1/2 hours.



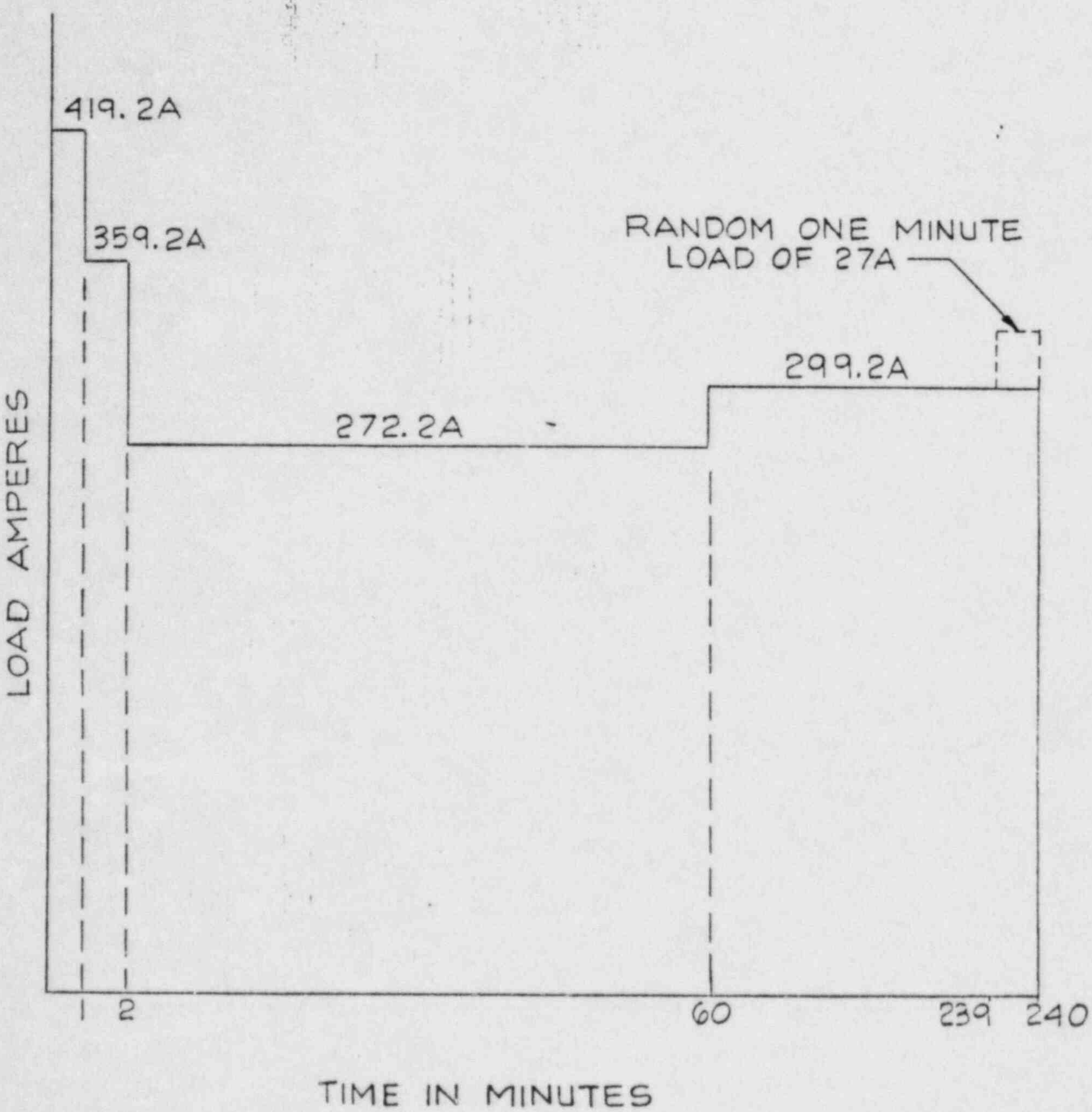
HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

CLASS IE 125 VDC
BATTERY 1A D411
LOAD PROFILE
FIGURE 8-3-16
SHEET 1 OF 8



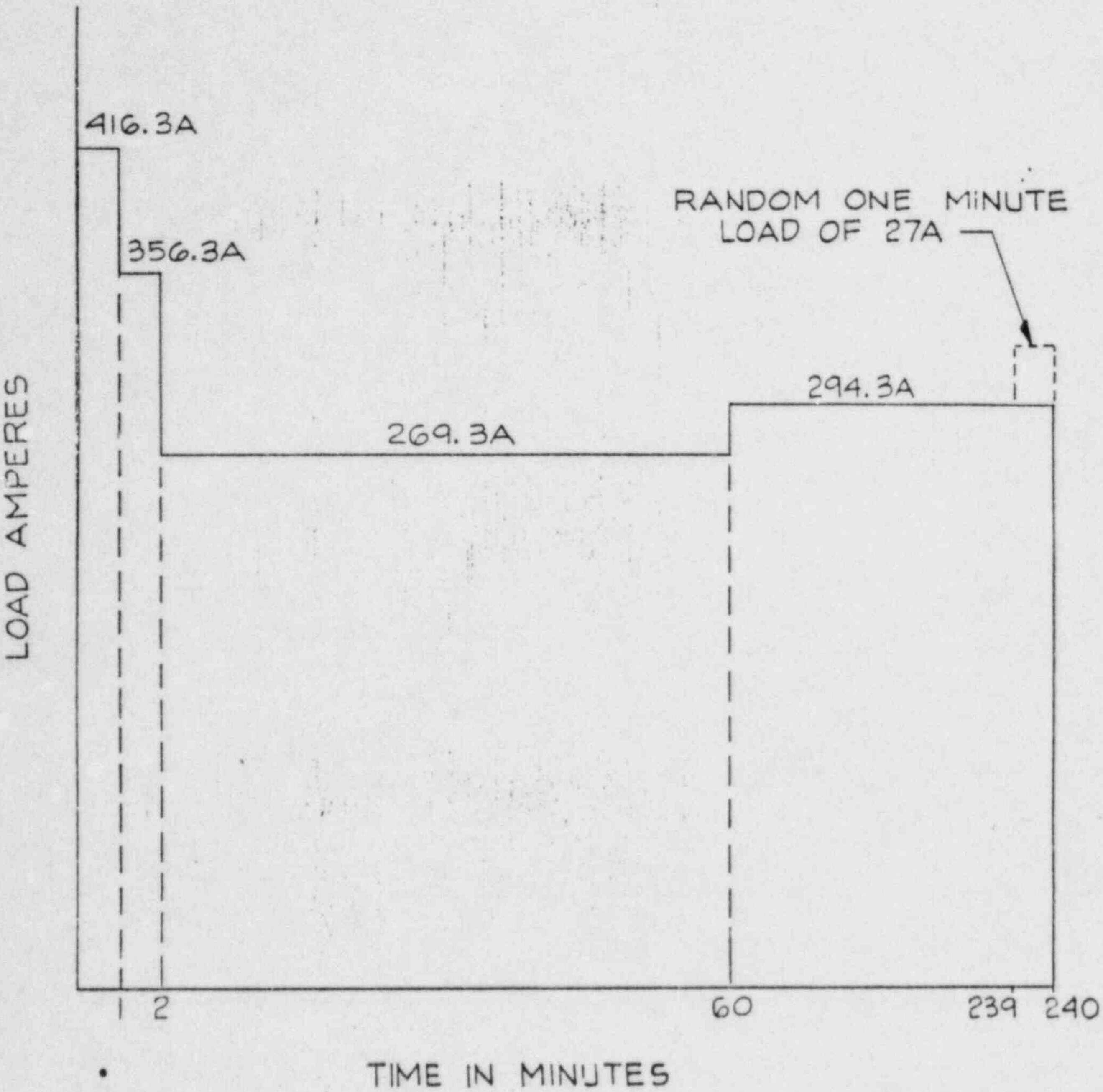
HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

CLASS 1E 125 VDC
BATTERY 1B D411
LOAD PROFILE
FIGURE 8-3-16
SHEET 2 OF 8



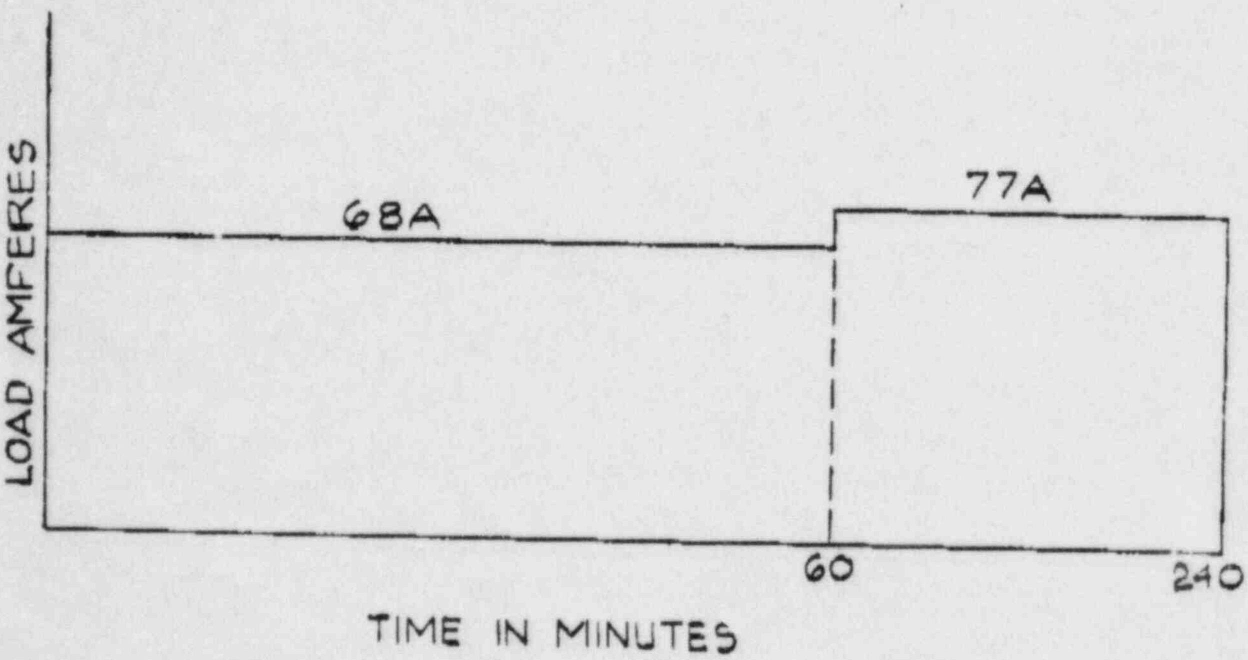
HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

CLASS 1E 125 VDC
BATTERY 1C D411
LOAD PROFILE
FIGURE 8-3-16
SHEET 3 OF 8



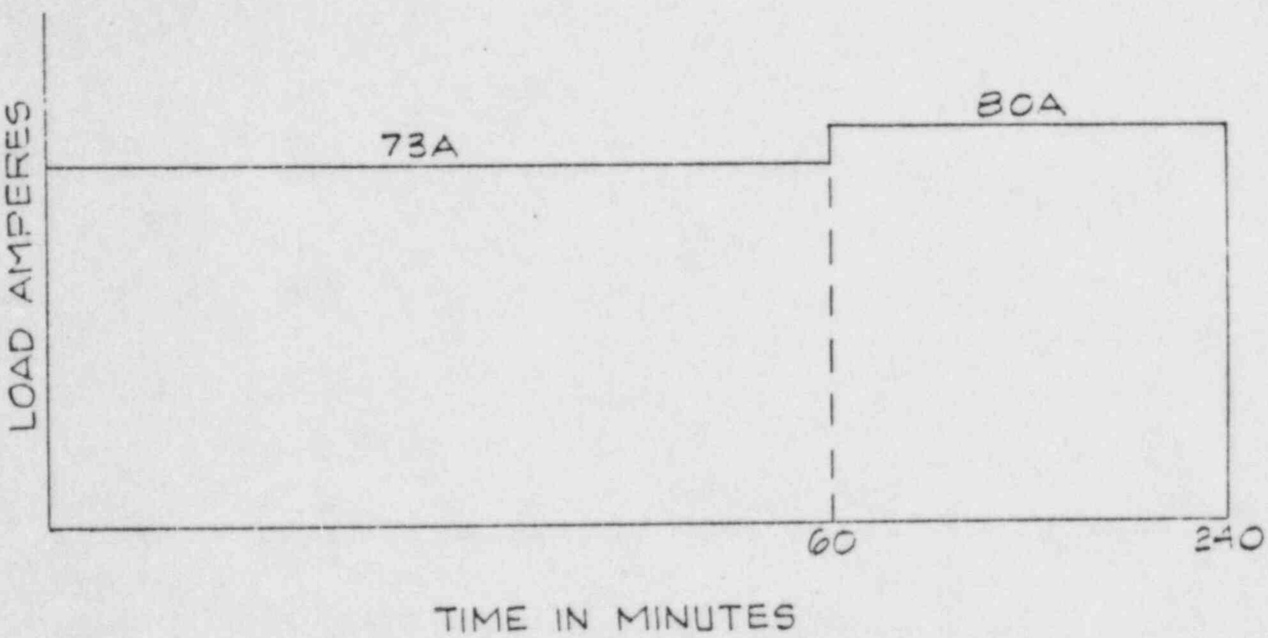
HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

CLASS 1E 125 VDC
BATTERY 1D D411
LOAD PROFILE
FIGURE 8-3-16
SHEET 4 OF 8



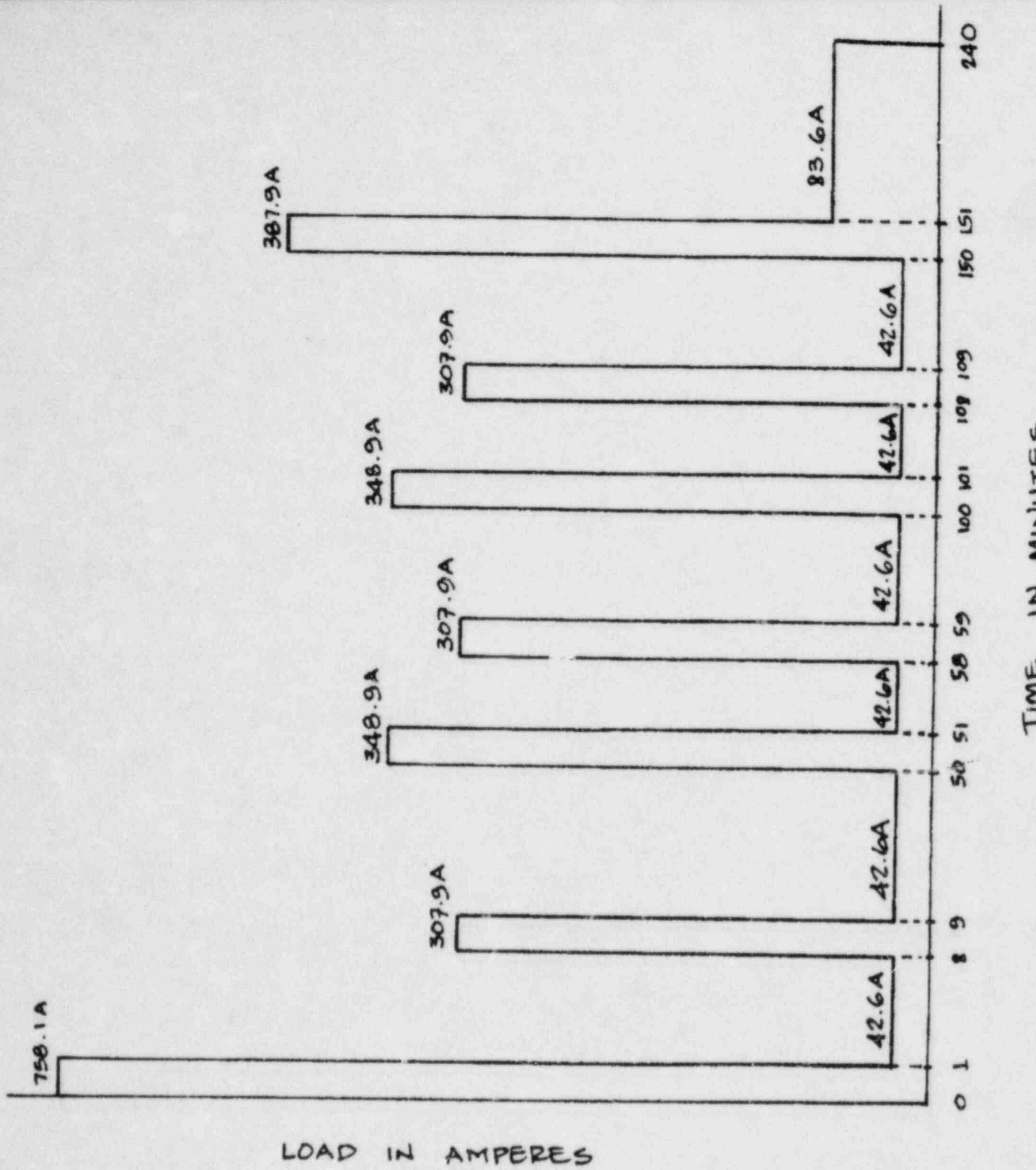
HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

CLASS IE 125 VDC
BATTERY 1CD447
LOAD PROFILE
FIGURE 8-3-18
SHEET 5 OF 8



HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

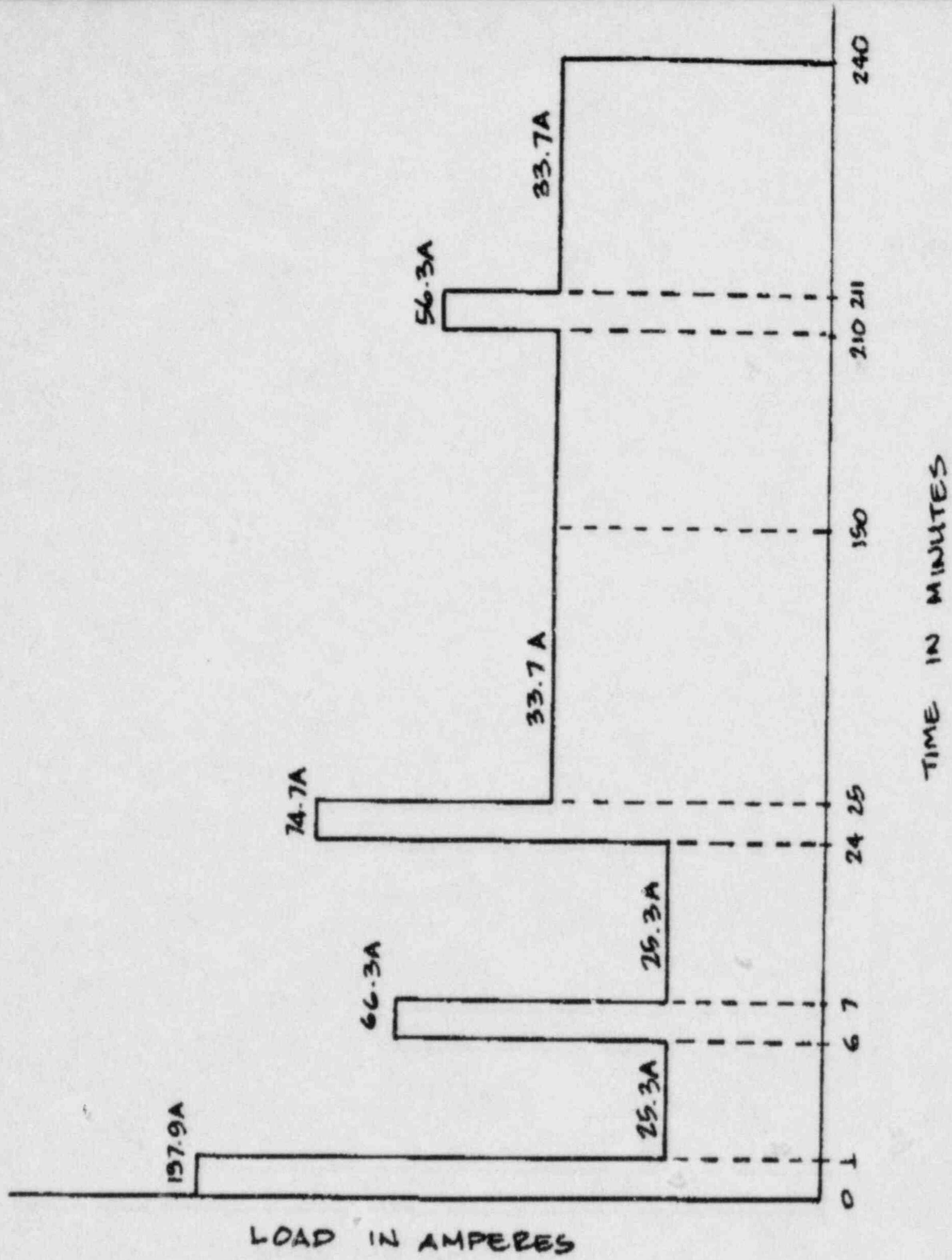
CLASS 1E 125 VDC
BATTERY 1DD447
LOAD PROFILE
FIGURE 8-3-16
SHEET 6 OF 8



HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

CLASS IE 250 VDC
BATTERY 1OD421
LOAD PROFILE

FIGURE 8.3-16
SHEET 7 OF 8



HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

CLASS 1E 250 VDC
BATTERY 10D431
LOAD PROFILE

FIGURE 8.3-16
SHEET 8 OF 8

Rev 1

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 base 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.

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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

QUESTION 430.33

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

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 directing acting trip devices and do not require external control power supply for tripping for electrical fault conditions. Therefore, the failure of the dc 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 PAGE 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

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

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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

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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~~	~~52-441043~~
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

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	150-140 VDC 0-364 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.

TEST PROCEDURES

1.0 GENERAL NOTES

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

1.2 NORMAL SYSTEM OPERATION DURING EACH TEST

A. CONNECTION PER FIG. 1.
B. OUTPUT LOAD 10KVA @ .08PF (66.7 AMP RESISTIVE AND 50 AMP INDUCTIVE) @ 120VAC NOMINAL.
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.

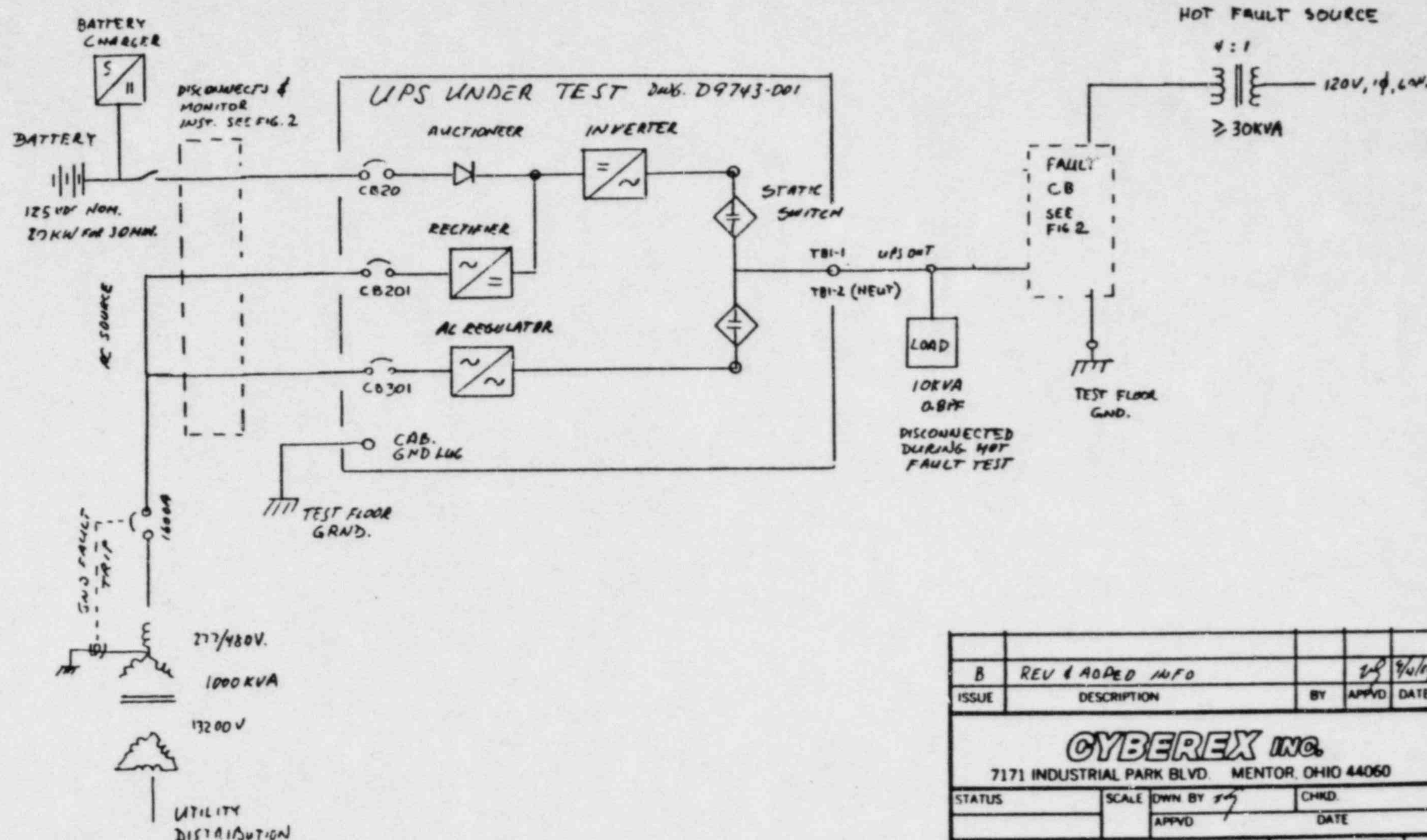
A1. METERING AND CONNECTIONS PER FIG. 2 AND "BACKUP SOURCE" BREAKER OPEN. RECORD IN "STORE" MODE AT 20KHZ TIME BASE. COPY MEMORY TO PAPER.
A2. REPEAT A1 EXCEPT USE 500HZ TIME BASE.
B1. WITH METERING AND CONNECTIONS PER FIG. 2 AND "NORMAL SOURCE" BREAKER OPEN. RECORD IN "STORE" MODE AT 20KHZ TIME BASE.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

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.



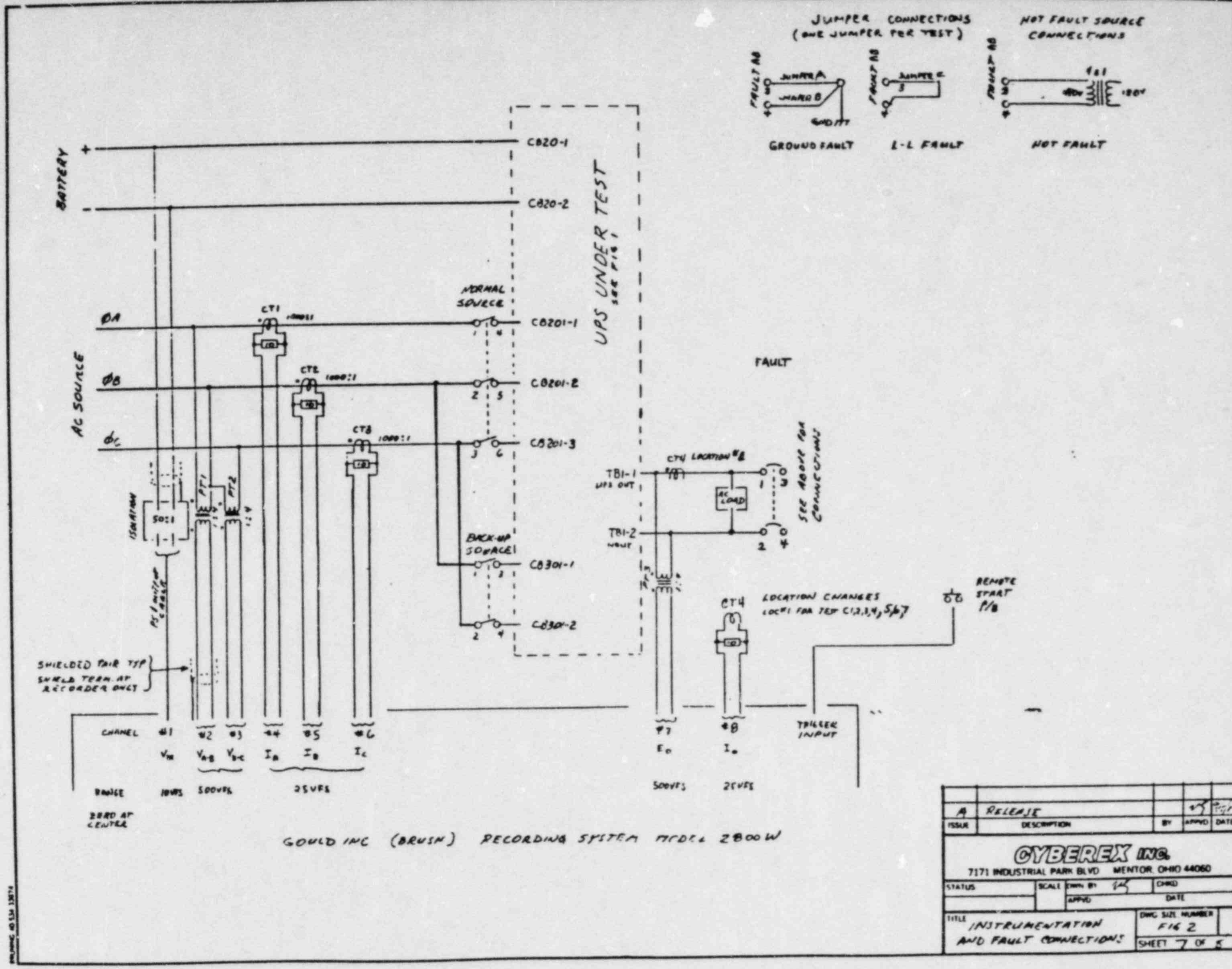
ISSUE	DESCRIPTION	BY	APPVD	DATE
B	REV & ADDED INFO		[initials]	9/6/84

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

STATUS	SCALE	DWN BY [initials]	CHKD.
		APPVD	DATE

TITLE: TEST SETUP
ONE LINE DIAGRAM

DWG SIZE, NUMBER: FIG. 1
SHEET 6 OF 8


JUMPER CONNECTIONS
(ONE JUMPER PER TEST)
HOT FAULT SOURCE CONNECTIONS
FAULT AB
JUMPER A
JUMPER B
JUMPER C
GROUND FAULT
L-L FAULT
HOT FAULT
4:1
120V
BATTERY
CB20-1
CB20-2
UPS UNDER TEST
SEE FIG 1
AC SOURCE
ØA
ØB
ØC
NORMAL SOURCE
CB201-1
CB201-2
CB201-3
BACK-UP SOURCE
CB301-1
CB301-2
CT1
CT2
CT3
1000:1
PT1
PT2
ISOLATION
50:1
FAULT
TB1-1
UPS OUT
TB1-2
NEUT
CT4 LOCATION #2
AC LOAD
SEE ABOVE FOR CONNECTIONS
CT4
LOCATION CHANGES
LOC#1 FOR TEST C1,2,3,4,5,6,7
REMOTE START P/B
TRIGGER INPUT
SHIELDED PAIR TYP
SHIELD TERM. AT RECORDER ONLY
CHANEL
#1
#2
#3
#4
#5
#6
#7
#8
RANGE
ZERO AT CENTER
500VFS
25VFS
GOULD INC (BRUSH) RECORDING SYSTEM MODEL 2800W
A
RELEASE
ISSUE
DESCRIPTION
BY
APPVD
DATE
CYBEREX INC.
7171 INDUSTRIAL PARK BLVD MENTOR, OHIO 44060
STATUS
SCALE
DWN BY
CHKD
APPVD
DATE
TITLE INSTRUMENTATION AND FAULT CONNECTIONS
DWG SIZE NUMBER
FIG 2
SHEET 7 OF 8

BRUNING 40-534 33674

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 :

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

ISSUE | DESCRIPTION | BY | APPVD | DATE

TITLE

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

DWG SIZE, NUMBER

SHEET 8 OF 8

TEST PROCEDURE, ISOLATION VERIFICATION

S/N 9743 1E 20KVA UPS (INSTRUMENTATION AC POWER SUPPLY) IN SERIES WITH A POWER CONVERSION PRODUCTS ISOLATING TRANSFORMER MODEL # RTF-120/120-30

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

OBJECTIVE:

TESTING TO ESTABLISH THE ISOLATING TRANSFORMER IN SERIES WITH A 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	150-140 VDC 0-364 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 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 RATIO (MULTIPLIER) VALUES.

1.2 NORMAL SYSTEM OPERATION DURING EACH TEST

A. CONNECTION PER FIG. 1.
B. OUTPUT LOAD 10KVA @ .08PF (66.7 AMP RESISTIVE AND 50 AMP INDUCTIVE) @ 120VAC NOMINAL.
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.

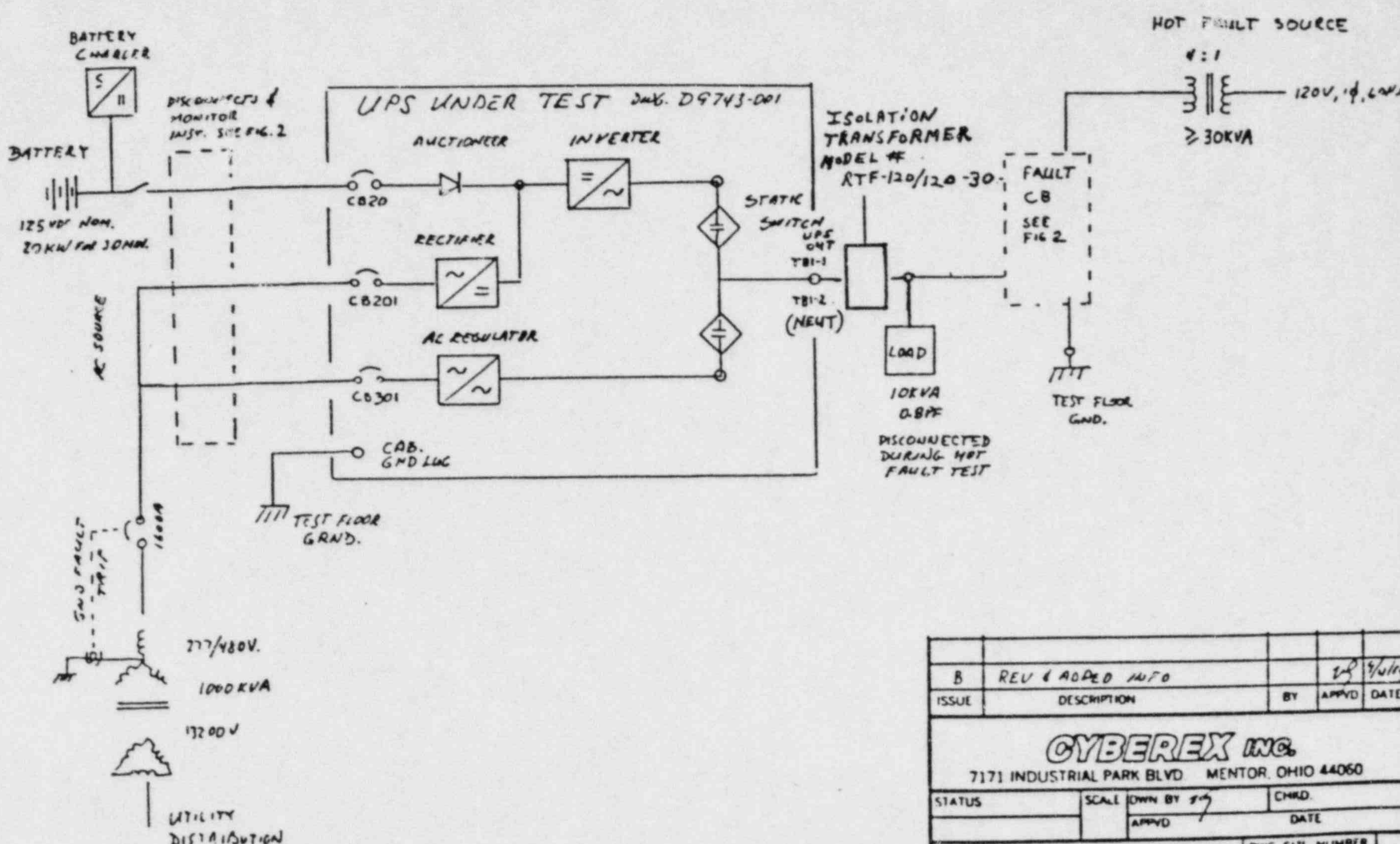
A1. METERING AND CONNECTIONS PER FIG. 2 AND "BACKUP SOURCE" BREAKER OPEN. RECORD IN "STORE" MODE AT 20KHZ TIME BASE. COPY MEMORY TO PAPER.
A2. REPEAT A1 EXCEPT USE 500HZ TIME BASE.
B1. WITH METERING AND CONNECTIONS PER FIG. 2 AND "NORMAL SOURCE" BREAKER OPEN. RECORD IN "STORE" MODE AT 20KHZ TIME BASE.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

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).

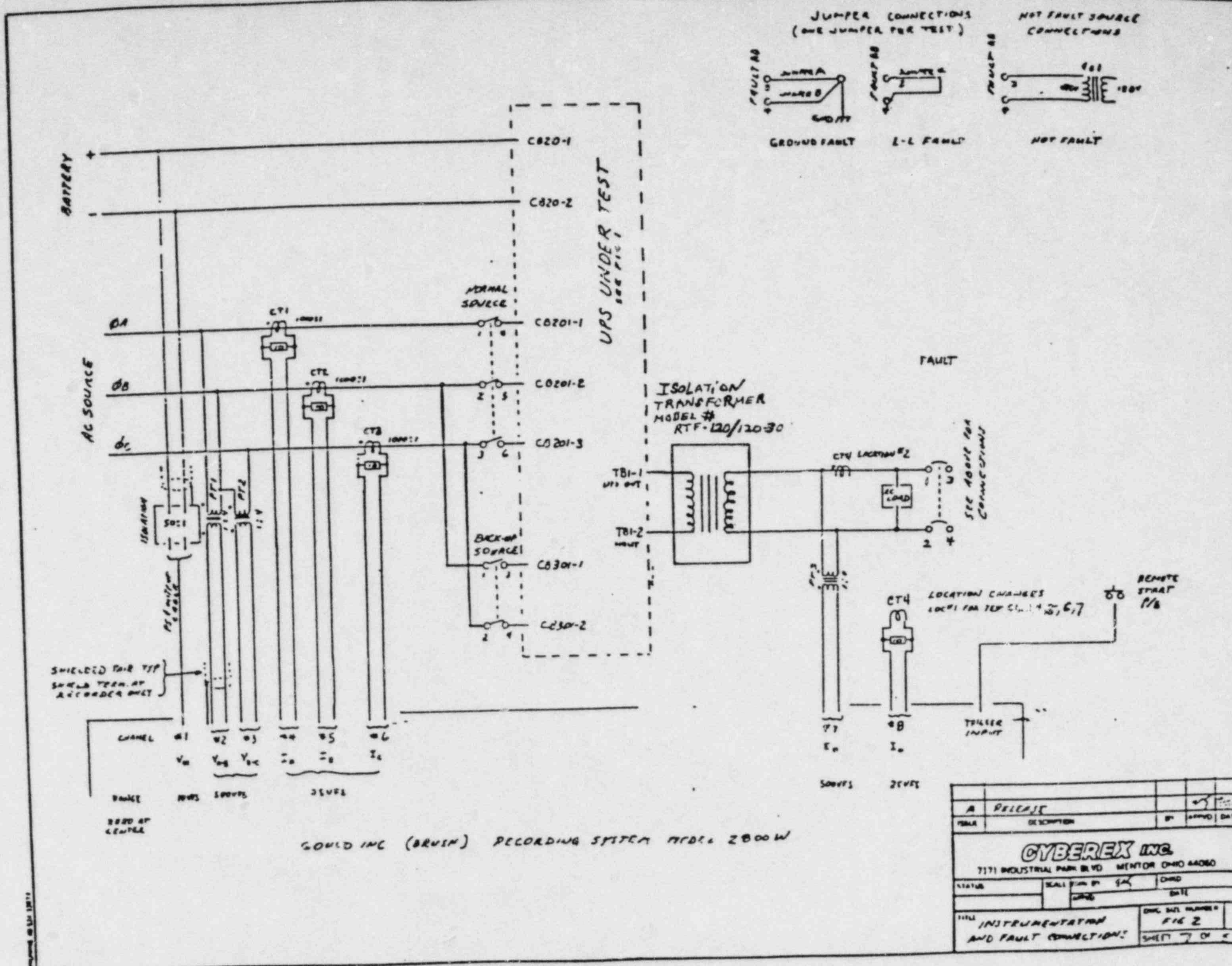
2.3 COMPLETE TEST SUMMARY SHEET FOR EACH TEST OR TEST GROUP.



B	REV & ADDED INFO			
ISSUE	DESCRIPTION	BY	APP'VD	DATE

CYBEREX INC.
7171 INDUSTRIAL PARK BLVD MENTOR, OHIO 44060

STATUS	SCALE	DWN BY	CHK'D.
		APP'VD	DATE
TITLE TEST SETUP ONE LINE DIAGRAM		DWG. SIZE. NUMBER FIG. 1	SHEET 6 OF 8


JUMPER CONNECTIONS
(ONE JUMPER PER TEST)
HOT FAULT SOURCE CONNECTIONS
GROUND FAULT
L-L FAULT
HOT FAULT
BATTERY
C820-1
C820-2
UPS UNDER TEST
NORMAL SOURCE
C0201-1
C0201-2
C0201-3
BACK-UP SOURCE
CB301-1
CB301-2
AC SOURCE
ØA
ØB
ØC
CT1
CT2
CT3
CT4
PT1
PT2
FAULT
ISOLATION TRANSFORMER
MODEL #
RTF-120/120-30
TB1-1
TB1-2
SEE ABOVE FOR CONNECTIONS
LOCATION CHANGES
REMOTE START
SHIELDED PAIR TSP
TRIGGER INPUT
CHANEL
#1
#2
#3
#4
#5
#6
#8
GOULD INC (BRUSH) RECORDING SYSTEM MODEL 2800W
A
RELEASE
DESCRIPTION
CYBEREX INC.
7171 INDUSTRIAL PARK BLVD MENTOR OHIO 44060
INSTRUMENTATION AND FAULT CONNECTIONS
FIG 2

PRUNING AO 534 33674

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	APP'D	DATE

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

STATUS | SCALE | DWN BY | CHKD. | APP'D. | DATE

TITLE

DWG. SIZE, NUMBER

SHEET 8 OF 8

REV 1

DSER Open Item No. 261 (DSER Section 8.3.3.3.6)

AUTOMATIC TRANSFER OF LOADS AND INTERCONNECTION BETWEEN REDUNDANT DIVISIONS

In Sections 8.1.4.1, 8.3.1.1.2.4, and 8.3.2.2 of the FSAR, it is stated that no provision exists for either automatic or manual transfer of loads between redundant load groups. The design depicted by this statement, meets the requirements of criterion 17 of Appendix A to 10 CFR 50, the guidelines of Regulatory Guide 1.6 and is therefore, acceptable. However, based on staff review of single line diagrams presented in Section 8.3 of the FSAR, provision for both automatic and manual transfer of loads have been identified. Sheet 2 of Figure 8.3-11 (E-0012-1) of the FSAR shows the non Class 1E BOP computer load normally connected to Class 1E division "D" with provision for automatic transfer to divisions "B" or "C". In addition, Sheet 5 of Figure 8.3-8 (E-0009-1) of the FSAR shows the non Class 1E loads on 125 v DC bus 10D486 having provision for manual transfer between Class 1E divisions "A" and "B" and the capability for simultaneous connection of this same load to both division "A" and "B." Similar provisions for load transfer also exists between division "C" and "D".

By Amendment 4 to the FSAR, the applicant indicated that a BOP computer load powered from Class 1E, Channel B is automatically transferred to Class 1E Channel C on loss of its Class 1E power supply. The applicant further indicated that this automatic transfer design does not violate the requirements of GDC 17 nor does it fall under guidelines of Regulatory Guide 1.6 because the BOP computer load is not safety-related. The staff disagrees. The automatic transfer does not meet position 4c of Regulatory Guide 1.6. In addition, this automatic transfer or interconnection between redundant divisions does not meet the independence requirements of GDC 17. The applicant has been requested to provide this results of an analysis that identifies and justifies use of all physical and electrical interconnections between redundant ac and dc divisions and between redundant associated divisions. This items will be pursued with the applicant.

RESPONSE

The automatic/manual transfer feature has been deleted as indicated in the revised response to Question 430.34.

DSER No. 261

QUESTION 430.34 (SECTION 8.3.2)

In Sections 8.1.4.1, 8.3.1.1.2.4 and 8.3.2.2 of the FSAR you state that no provision exists for either automatic or manual transfer of loads between redundant load groups. The design depicted by this statement meets the requirements of Criterion of Appendix A to 10 CFR 50, the guidelines of Regulatory Guide 1.6 and is, therefore, acceptable. However, based on sta review of single line diagrams presented in Secition 8.3 of the FSAR, provision for both automatic and manual transfer of loads have been identified. Sheet 2 of Figure 8.3-11 (E-0012-1) of th FSAR shows the non Class 1E BOP computer load normally connecte to Class 1E division "D" with provision for automatic transfer divisions "B" or "C". In addition, Sheet 5 of Figure 8.3-8 (E-0009-1) of the FSAR shows the non Class 1E loads on 125 v DC bus 10D486 having provision for manual transfer between Class 1 divisions "A" and "B" and the capability for simultaneous connection of this same load to both division "A" and "B". Similiar provisions for load transfer also exists between division "C" and "D".

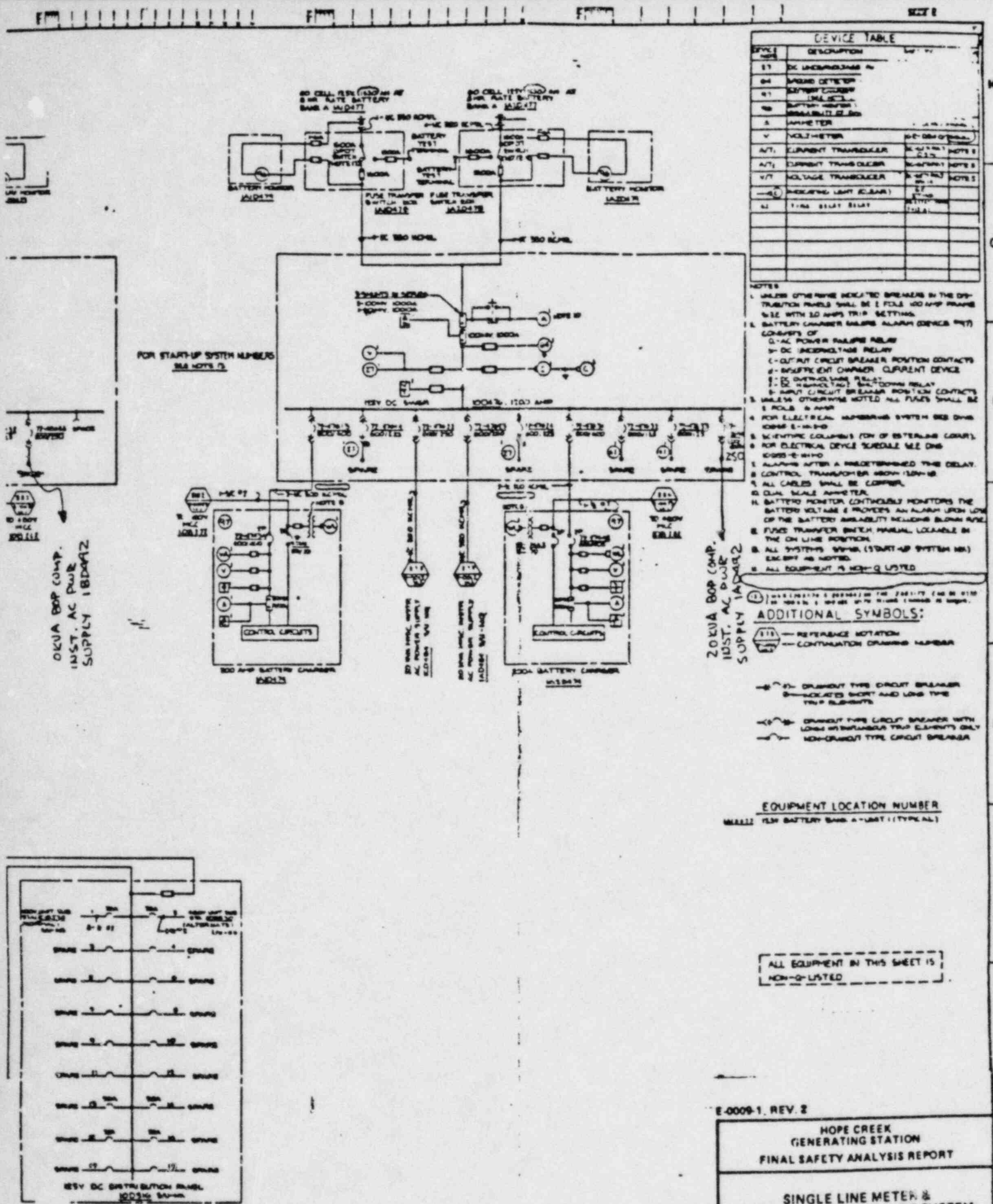
a. Correct the above identified inconsistency so that th Hope Creek design is consistent with design commitmen contained in the FSAR. Describe how one can conclude in the future with reasonable assurance that the actu Hope Creek electrical desing meets design commitments documented in the FSAR.

b. Provide the results of an analysis that demonstrates that the physical and electrical independence of the four independent electrical divisions have not been compromized by the connection of Non Class 1E loads c the Class 1E AC and DC system. The results of the analysis should include but not be limited to (1) identification and justification of all electrica interconnections, (2) description with electrical schematic diagrams for each non Class 1E load, (3) description of the physical routing of circuits associated with each Class 1E load group with respect to other non-Class 1E loads connected to redundant Class 1E load groups, and (4) where separation betwe redundant associated circuits or between associated circuits and non Class 1E circuits is less than the separation required by IEEE standard 384, justificat should be provided.

RESPONSE

FSAR FIG 8.3-11 SHEET 2 HAS BEEN REVISED TO SHOW THAT THE BOP COMPUTER

POWER SUPPLIES 1AD492 AND 1BD492, ARE POWERED FROM NON-CLASS 1E ELECTRIC POWER SOURCES. THERE ARE NO MANUAL OR AUTOMATIC PROVISIONS FOR TRANSFERRING LOADS BETWEEN REDUNDANT AC OR DC ELECTRIC POWER SUPPLY CHANNELS.



E-0009-1, REV. 2

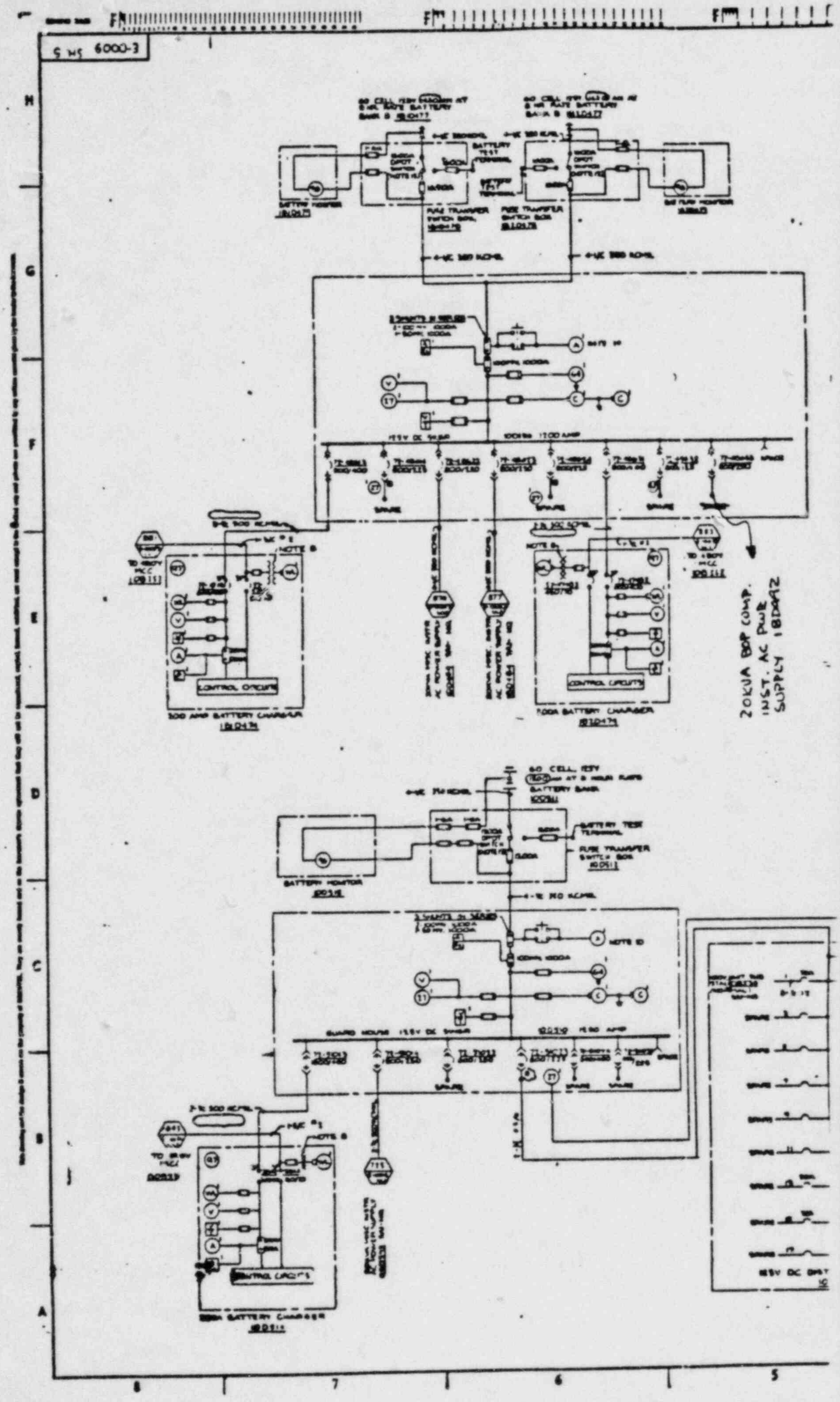
HOPE CREEK GENERATING STATION

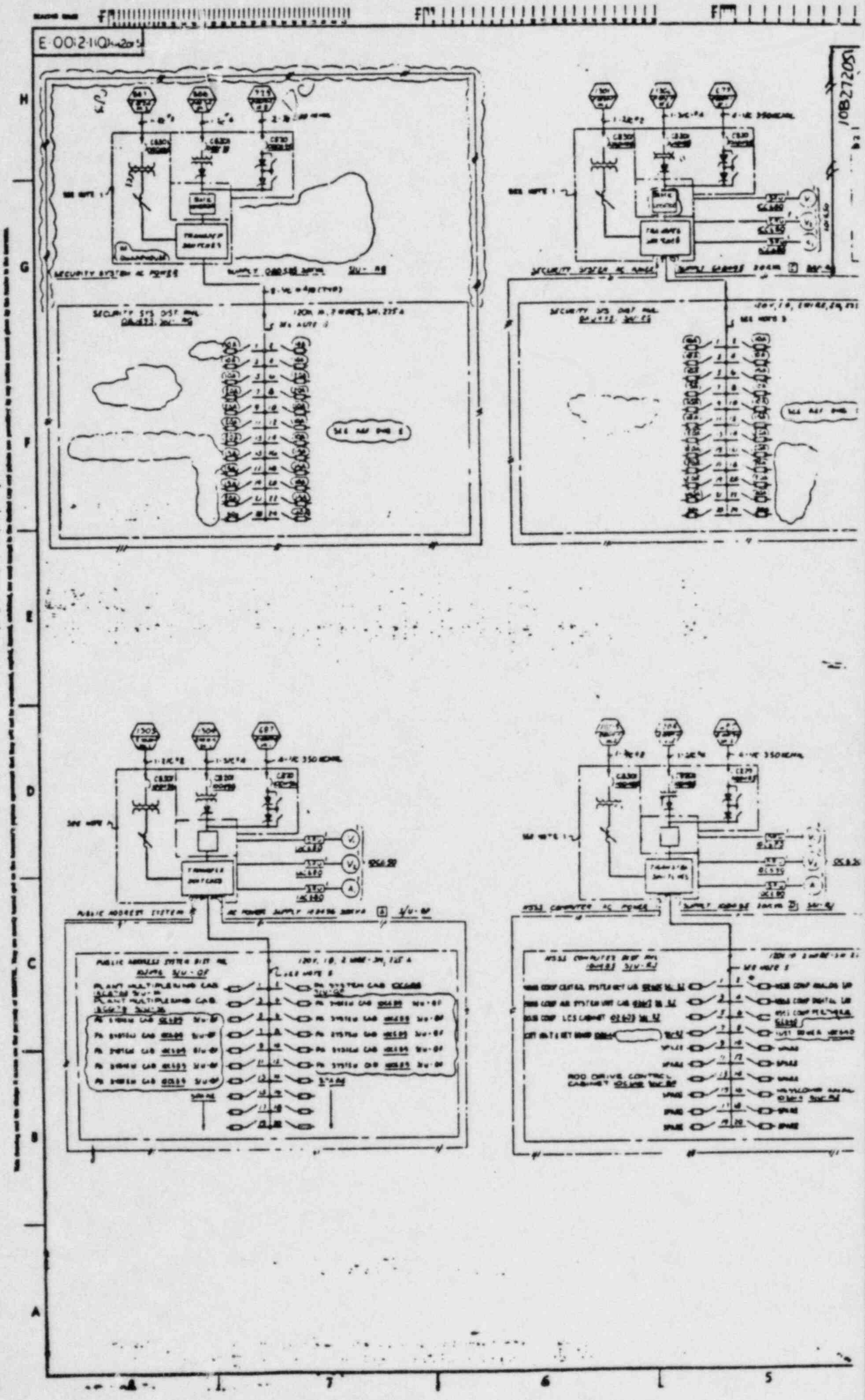
FINAL SAFETY ANALYSIS REPORT

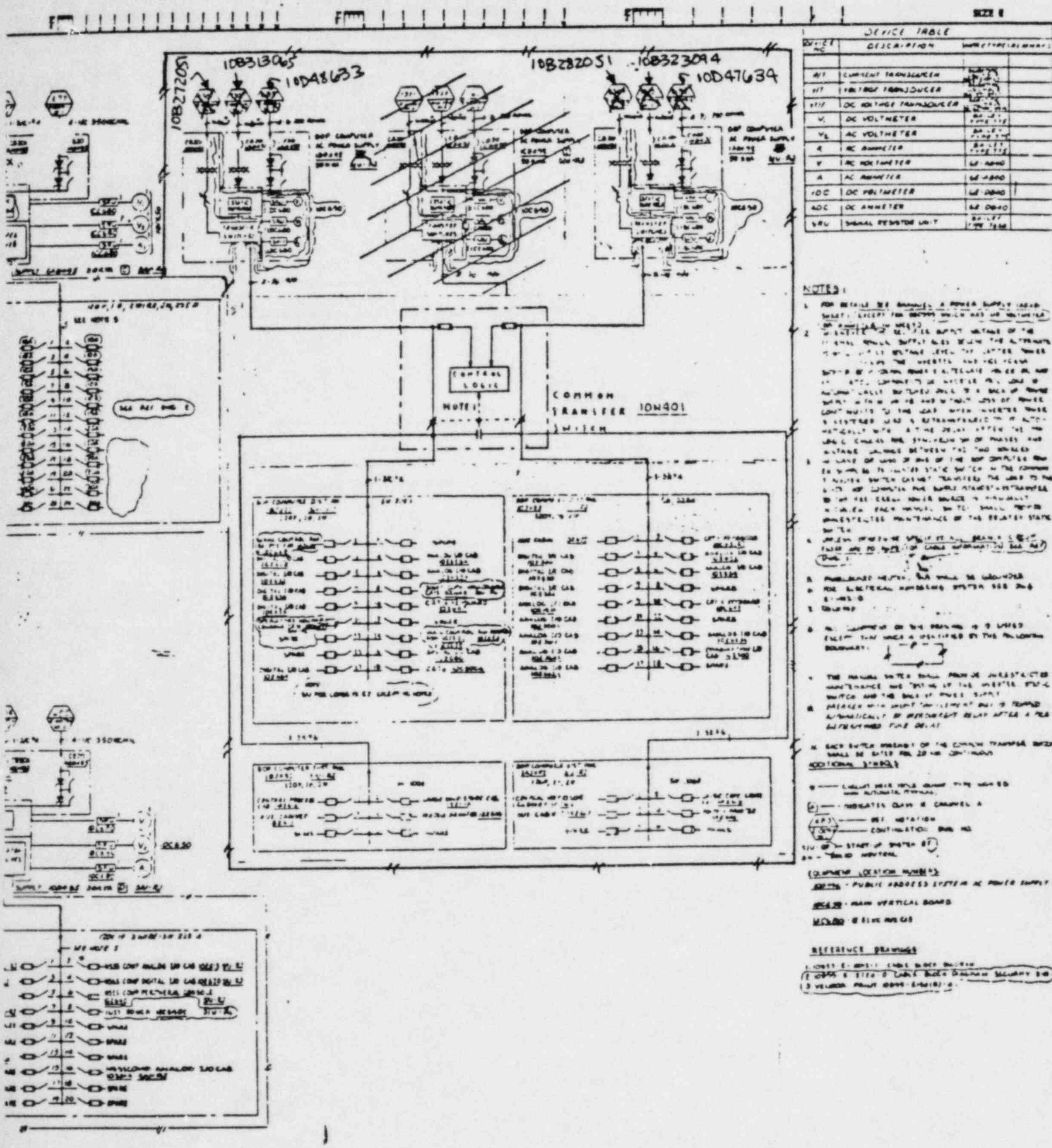
SINGLE LINE METER & RELAY DIAGRAM 125 V DC SYSTEM

FIGURE 8.3-8
SHEET 5 OF 5

Amendment 6, 06/84


S HS 6000-3
20KVA BOP COMP.
INST. AC PWR
SUPPLY 1BD0A2
CONTROL CIRCUITS
BATTERY MONITOR
SPARE
125V DC DIST
NOTE 8
NOTE 10


E-0021101
SECURITY SYSTEM AC POWER
SECURITY SYS DIST PNL
TRANSFER SWITCHES
PUBLIC ADDRESS SYSTEM
AC POWER SUPPLY
ROD DRIVE CONTROL CABINET
SPARE
H
G
F
E
D
C
B
A
7
6
5



E-0012-1 REV. 5

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

SINGLE LINE METER &
RELAY DIAGRAM 120 V AC
INSTRUMENTATION & MISC SYSTEMS

FIGURE 8.3-11
SHEET 2 OF 5

DSER Open Item No. 268 (DSER Section 6.8.1.2)

ESF AND NON-ESF AIR FILTRATION UNIT DRAINS

Regarding the ESF and non-ESF air filtration unit drains, what keeps the air traps in the water drains filled with water? Is there an automatic fill system?

RESPONSE

Not all filtration units have water drain traps. Of the ESF air filtration systems, only the filtration, recirculation, and ventilation system (FRVS) recirculation system and the FRVS vent system units are provided with drum traps. A regular inspection of the water level in the drums will be implemented.

The control room emergency filter (ESF) and the technical support center emergency filter (non-ESF) units are provided with ball float type drainers. The discharge port remains closed when the water level is low. Thus sealing integrity is maintained.

The radwaste tank vent filter (non-ESF) units are provided with check valves in the upstream and downstream drain lines of the charcoal compartment preventing backflow of air and water. Thus, ~~maintaining~~ sealing integrity of drain lines is maintained.

For additional information, see the attached discussion on the Filtration System Drain Trap.

→ INSERT A

INSERT A TO OI 268 RESPONSE:

FILTRATION SYSTEMS DRAIN TRAP DISCUSSION

Water filled drum traps are provided on the drain line from the ESF FRVS recirculation units and the FRVS vent units. They are also provided on the non-ESF CPCS system. Refer to FSAR figures 9.4-4 and 9.4-5. These drum traps are periodically filled with water to its outlet inorder to maintain a 20" unit drain line submergence. Refer to the attached drum trap detail.

CONCERN: What is the evaporation rate of the water in the trap? How long between maintenance intervals would be required to keep the traps filled.

RESPONSE: The maximum evaporation rate of the water in the trap was evaluated under the most conservative conditions. These conditions are:

1. Reactor building ambient temperature of 104°F. (Maximum normal design temp.)

2. Water temperature of 104°F.

3. Reactor building relative humidity of 20% (Very conservatively dry for a 104°F. temp.)

The maximum evaporation is in the drum traps for the FRVS recirculation units. Under the above conditions the rate is 0.038 lb/hr which results in a drop of drum level of 1 inch in 5.35 days.

The rate for the FRVS vent drum traps is 0.016 lb/hr or a drop of 1 inch in 12.9 days.

The rate for the CPCS unit drum trap is 0.0076 lb/hr or a drop of 1 inch in 26.6 days.

The maximum static pressure that any drum trap would see during normal system operation is 15 inch WG. Thus 5 inches could evaporate before a loss of water seal would occur. This would take longer than 26.75 days (5.35 days/in. x 5 in.) Since the 104°F conditions described above would not last this long. At 80°F air and water temp. and 20% RH the evaporation rate is only 0.00424 lb/hr. or 11% of the maximum rate for the FRVS recirc. units.

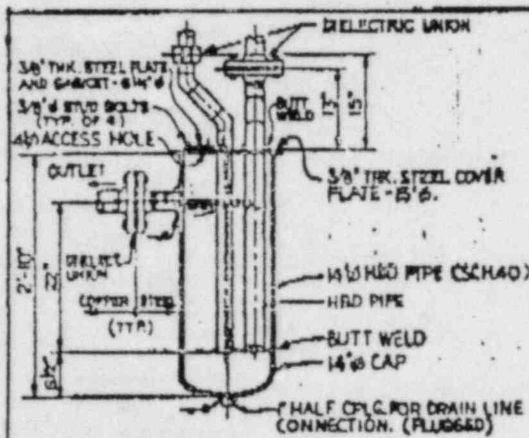
With the removal of the access hole cover, which will be implemented in the design, the evaporation rates will be as shown on the attached table.

Add insert

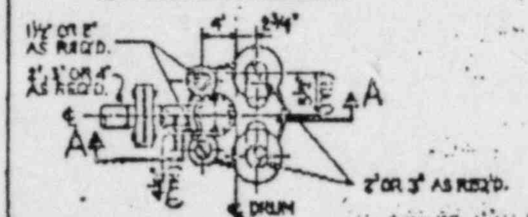
Even with the covers ~~reombed~~ removed, it is safe to conclude that a maintenance interval of 14 days will keep the water seals intact.

INSERT

The station surveillance procedures will include a requirement to inspect and refill to the level of the outlet drain connection as required for each of the FRVS and CPCS drain traps on a 14 day interval.



SECTION A–A



PLAN (SEE TABLE)

NOTE:

DIELECTRIC UNIONS "EPCO" SALES, INC. MANUFACTURING CO.
MODEL FX - IRON PIPE THREAD TO COPPER SOLDER JOINT. (FOR 1 1/2" AND 2" PIPES)
MODEL BX - IRON PIPE THREAD TO COPPER SOLDER JOINT. (FOR 3" AND 4" PIPES.)

FOR UNITS	DRAIN LINE	OUTLET
1AVH 213 1BVH 213 1CVH 213 1DVH 213 1EVH 213 1FVH 213	TWO 2" & TWO 3" ([illegible])	4"
1AVH [illegible] 1BVH [illegible]	TWO [illegible] ([illegible])	3"
1[illegible]H 200	[illegible] ([illegible] TRAP)	2"

51 DRUM TRAP

TABLES OF DRUM TRAP WATER EVAPORATION RATE

MAXIMUM EVAPORATION RATES, DRAIN TRAP ACCESS HOLE COVERED (NOTE 1)

UNIT DRUM TRAP	EVAPORATION RATE lb/hr.	TIME FOR WATER TO DROP (DAYS) 1 INCH	5 INCHES	20 INCHES
FRVS RECIRC.	0.033	5.35	26.8	107.1
FRVS VENT	0.016	12.98	64.9	259.6
CPCS	0.008	24.6	123.2	492.8

MAXIMUM EVAPORATION RATES, DRAIN TRAP ACCESS HOLE VENTED (NOTE 1)

FRVS RECIRC	0.059	3.45	17.25	69.0
FRVS VENT	0.034	6.02	30.1	120.5
CPCS	0.025	8.02	40.1	160.3

EXPECTED EVAPORATION RATES (NOTE 2) DRAIN TRAP ACCESS HOLE COVERED

FRVS RECIRC	0.029	7.10	35.5	142.0
FRVS VENT	0.012	17.3	86.3	345
CPCS	0.006	32.8	164.2	656.7

NOTES:

1. MAXIMUM EVAPORATION RATES BASED ON VERY CONSERVATIVE 104°F BUILDING AMBIENT TEMPERATURE, 104°F TRAP WATER TEMPERATURE, AND 20% RELATIVE HUMIDITY

2. EXPECTED EVAPORATION RATES BASED ON 99°F BUILDING AMBIENT TEMPERATURE, AND 30% RELATIVE HUMIDITY.

QUESTION 430.141 (SECTION 9.5.8)

Provide the results of an analysis that demonstrates that the function of your diesel engine air intake and exhaust system design will not be degraded to an extent which prevents developing full engine rated power or cause engine shutdown as a consequence of any meteorological or accident condition. Include in your discussion the potential and effect of fire extinguishing (gaseous) medium, recirculation of diesel combustion products, or other gases that may intentionally or accidentally be released on site, on the performance of the diesel generator. (SRP 9.5.8, Parts II & III)

INSERT A

RESPONSE

~~Due to the stragegic location of the SDG air intake in relation to the exhaust gas stack, recirculation of the exhaust gas to the air intake is minimized and therefore will not pose a hazard to the performance of the SDG. This is discussed in Section 9.5.8.3.~~

~~As discussed in Section 9.5.1 and indicated in Table 9A-1, a water hose is provided in the SDG combustion air intake areas, and portable fire extinguishers (CO_2 or dry chemicals) are also available for limited use. This possibility of limited use of CO_2 or dry chemical fire extinguishers does not pose a potential threat to the diesel engine since the area is vented to the outside via air inlet louvers, as shown on Figures 1.2-11 and 1.2-36.~~

~~A potential fire in or near the SDG area is discussed in response to Question 430.143.~~

~~Other gases that may intentionally or accidentally be released on site are either located remote to the diesel generator enclosures or are small enough in volume to not pose a hazard to the performance of the diesel generators. Refer also to Section 9.5.1.1.11.~~

~~A safety evaluation of the air intake and exhaust system which discusses meteorological and accident conditions is provided in Section 9.5.8.3, with further discussion in Section 3.3, 3.4, 3.5, 3.6, and 3.11. Additionally, onsite wind direction frequency distributions, Tables 2.3-5 and 2.3-6, indicate that the normal or prevailing winds disburse diesel exhaust gases and any other onsite gaseous releases away from the SDG air intake louvers. The equipment is designed to remain operable for the range of design conditions given in Section 3.3.2.1.a and b.~~

~~From the above, no circumstances as a consequence of meteorological or accident conditions could be postulated that~~

would degrade system design to an extent which prevents developing full rated engine power or causes engine shutdown.

INSERT (A)

[redacted]

Section 9.5.1.1.18 states that diesel generator combustion air intakes are located remotely from exhaust openings and smoke vents of other fire areas. The relative location of the diesel generator combustion air openings (points J, K, L, and M on Figure 430.141-1) are located along column line 24.3 of the diesel generator building at elevation 130 ft. No ventilation openings are located directly in front of these openings or nearby on the same wall.

The nearest openings are found at a higher elevation and set back compared to the diesel intakes. These openings are shown in Figure 430.141-1 and include diesel area air intakes (D), switchgear room intakes (E, F, G, H), and the control room intakes (I). These openings are not of concern to diesel engine intakes because the openings are much higher and hot, rising, combustion products will not flow down to the diesel intakes. In addition, the rooms served by these ventilation systems in many cases contain fire dampers that would prevent smoke escaping and, further, being intakes, are not used to exhaust smoke.

Other openings on the diesel and control building located on the roof include exhausts. These openings are higher and more remote than those previously discussed.

The release of toxic gases has been analysed for their ingestion into the control room ventilation which is in the vicinity of the diesel generator intakes. As discussed in Section 6.4.4, the analysis has shown that, using conservative dispersion models, no problem exists for air intakes from this area. It should be noted also in considering the dispersion of CO_2 from a postulated release of the CO_2 storage tank adjacent to the building, that the engine manufacturer has stated that the diesels can produce full rated power with CO_2 concentrations up to 15% by volume.

A BATTERY ROOM EXHAUST EL. 198'-0" & CONTROL AREA SMOKE EXHAUST
B. DIESEL AREA EXHAUST " "
C WING AREA EXHAUST " "
D. DIESEL AREA OUTSIDE AIR INTAKE EL. 178'-3"
E SWITCHGEAR RM. O.A. INTAKE EL. 178'-3"
F " " " " "
G " " " " EL. 163'-9"
H " " " " EL. 163'-9"
I CONTROL ROOM AIR INTAKE
J INTAKE O.A. D.G.
K " " "
L " " "
M " " "

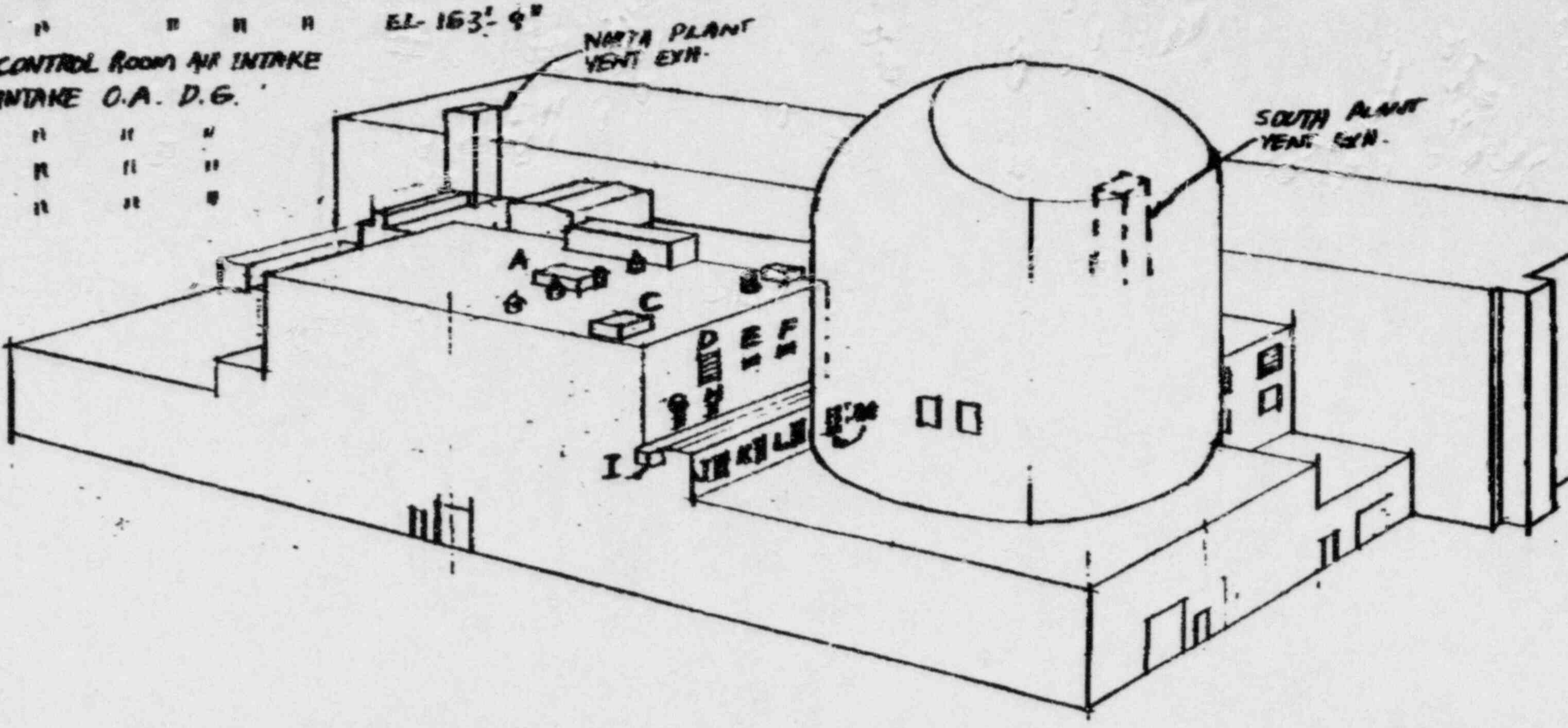


FIGURE 430.141-1

ReJ 1

QUESTION 430.143 (SECTION 9.5.8)

Show by analysis that a potential fire in the diesel generator building or any of the other surrounding buildings (reactor building, control building, etc.) together with a single failure of the fire protection system for that area will not degrade the quality of the diesel combustion air so that the remaining diesels will be able to provide full rated power. (SRP 9.5.8, Parts II & III)

RESPONSE

A 3-hour-fire-barrier has been added to separate the diesel combustion air intakes by safe shutdown division. Since the divisionalized intakes are in separate rooms, a fire in one zone, and an automatic closure of the fire door will not affect the remaining diesels' combustion air. Therefore, the remaining two diesels will be able to provide full rated power. This analysis was performed as part of the Appendix R fire hazard analysis (see revised Appendix 9A).

The Appendix R analysis shows that a fire in any one fire area of the control, diesel or reactor buildings will affect no more than one division of the diesel generator intakes. Ths Appendix R analysis assumes a failure of any automatic fire protection system for that area.

The SDG HVAC systems exhaust from missile protected areas located at elevation 198'-0". The possibility of significant quantities of smoke or other combustion by-products bypassing dampers or failed dampers from any of the areas and exiting at the 198 ft elevation and consequently being drawn down to other diesel generator intakes at the 130 ft elevation is not credible.

→ INSERT 1

430.143 - Insert 1

With a postulated failure of the automatic fire suppression system in one diesel area, the fire damper would close to contain the fire. Failure of the dampers is not considered credible, since it is a UL listed device and uses only the physical properties of the fusible link to operate upon high temperature of the link. In addition to the hot gases, the damper link is heated by an electro-thermal signal upon CO_2 system actuation. However, even if such a failure is postulated, the consequences of the smoke release are not of concern. The failure could release smoke into the large volume common corridor, but the HVAC system design would prevent any smoke from affecting more than one diesel. Section 9.4.6 describes how the system consists of 100% recirculating fan coil units with only a minimum of air exchange from the common corridor during diesel generator operation. Thus, cooling of the diesels would not be significantly affected, considering the small influx of warmer air from the common corridor. The manufacturer has stated that the diesel generator itself is insensitive to smoke in the compartment.

During normal plant operations, i.e., no diesels operating, the diesel area ventilation will exhaust air from each diesel compartment and out of the roof vent. Smoke from one compartment would have to exit to the large volume common corridor through the fire damper. It could then enter the other diesel generator compartments through each respective compartment's fire damper. Should the temperature rise the recirculation coil units would automatically start (9.4.6.2g) to maintain acceptable room temperatures. It should also be noted that such an event, a fire in a diesel generator compartment, would not be expected to cause a loss of offsite power requiring immediate response from the diesel generators.

Question 640.11 Part 2

Testing of dc loads necessary for safe shutdown should be conducted at minimum dc system voltage or the voltage drop at load to these components should be measured to verify that the dc loads are supplied with appropriate voltage under minimum battery voltage conditions.

Response

The equipment and components are designed to operate within the minimum and maximum battery voltage range as addressed in the response to Question 430.32.

A review program has been performed to verify that under minimum battery voltage conditions the final terminal voltage for all safe-shutdown equipment meets the minimum voltage requirement for the device to perform it's proper function. An additional study was performed to address the designed cable lengths and associated voltage drops for each safe shutdown load.

In those cases where the study has indicated that the terminal voltage at the load is approaching the equipment's design minimal voltage requirement, these specific loads will be checked during the functional test as identified in FSAR Subsections 14.2.12.1.35 (PJ-250 Vdc Class 1E Power) and 14.2.12.1.36 (PB-125-Vdc Class 1E Power). In addition a test will be performed to measure the actual terminal voltage for two cases; (1) the largest load and (2) the longest cable run, to verify the voltage is in the allowable range. This program will verify that the dc loads are supplied with appropriate voltage under minimum battery voltage conditions.

ATTACHMENT 5

Rev 2

I.C.6 VERIFY CORRECT PERFORMANCE OF OPERATING ACTIVITIES

Position

It is required (from NUREG-0660) that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations, and maintenance activities independent of the people performing the activity (see NUREG-0585, Recommendation 5), or both.

Response

Verification of operating activities to provide a means of reducing human errors and to improve the quality of normal operations shall be assured by the following procedures:

INSERT A

a) ~~OP-AP.ZZ-108(Q) Removal and Return of Equipment to Service shall be used to track equipment out of service, determine if the equipment is safety-related, determine if a Limiting Condition for Operation exists, determine if independent verification is required, and determine the pre and post testing requirements.~~

b) OP-AP.ZZ-109(Q) Equipment Operational Control shall contain the requirements to prevent unauthorized operation of equipment by establishing panel and valve lock and tagging control.

INSERT B

c) ~~OP-AP.ZZ-002(Q) Conduct of Operations will be revised to include independent verification requirements for safety related system line-ups.~~

d) SA-AP.ZZ-12(Q) Surveillance Program shall contain the requirements for independent verification of safety related system line-up and temporary modification for testing. In addition this procedure will require, prior to start of testing, permission from designated operations personnel holding an SRO license.

LQB

Insert A

a) OP-AP ZZ-108(Q) "Removal and Return of Equipment to Service" shall

1) Describe a program to track a system's status, i.e., operability.

2) Identify which system's or subsystems are consiiered to perform a safety-related function.

3) Determine if a system's change in status results in the entering or clearing of a limiting condition for operation.

4) Describe a program to ensure that technical specification required operability of redundant safety-related equipment is verified.

When like equipment is removed from service this program shall also ensure the appropriate retest of equipment following preventive or corrective maintenance and prior to the equipment's return to an operable status.

5) Prescribe independent verification of any activity which affects the mechanical or electrical line-ups of safety-related systems. This shall include the removal from and return to service.

Individuals performing the independent verification associated with mechanical and electrical line-ups shall, as a minimum, meet the requirements of the nonlicensed operator training program for equipment operators. These training requirements are outlined in Section 13.2.1.1.2 and Appendix 13H. Equipment operators performing the verifications will be those operators assigned to the nuclear shift supervisor on-duty.

In some cases the independent verifications may be performed by a nuclear control operator or shift technical advisor assigned to the on duty shift.

The training program for nuclear control operators and shift technical advisors are outlined in 13.2.1.1.1.2 and 13.2.1.1.1.3, respectively.

LQB

Insert B

OP-APZZ-002(Q) conduct of operations will describe the independent verification program. This procedure will prescribe the method and technique for performing the independent verification as well as what plant systems will require the verification.

ATTACHMENT 6

HOPE CREEK RESPONSE TO G.E. INFORMATION LETTER, SIL No. 402, DATED FEBRUARY 14, 1984

Recommendation 1

Evaluate Inerting System Design

Evaluate the design of the nitrogen inerting system. Investigate the potential for introducing cold (less than 40 degrees F) nitrogen and the orientation of the nitrogen port relative to the vent header, downcomers, or other equipment in the wetwell and drywell which may be in the path of the injected nitrogen. Assure that the temperature monitoring devices, the low temperature shutoff valve, and overall system design are adequate to prevent the injection of cold nitrogen into the containment.

RESPONSE

An evaluation was performed on the HCGS inerting system design to review the potential for injecting nitrogen less than 40°F into the containment, similar to the occurrence at Hatch Unit 2 which resulted in cracking of the vent header caused by brittle fracture. In the HCGS design, the nitrogen line from the vaporizer connects to the drywell and torus purge lines. Similar to Hatch, the torus purge line is also located above the vent header but offset from the centerline of the header by about three feet. Both the vent header, the containment penetration, and the purge piping connected to the containment are protected from being exposed to temperatures below their specified minimum service temperature due tc malfunctions of the inerting system allowing the injection of cold nitrogen to the containment by features discussed below. There are significant differences in the types of nitrogen vaporizers used at HCGS and Hatch Unit 2. Hatch has a direct cycle vaporizer with heating steam condensing directly on the nitrogen tubes in a heat exchanger. The failure of the steam supply may result in a rapid temperature drop in the nitrogen.

At HCGS the nitrogen vaporizer is a steam heated water bath type. The thermal inertia of the water bath will preclude step changes in the nitrogen temperature. A self-operated temperature regulator with its sensing bulb in the water bath is provided to control the steam inlet. The temperature range of the water bath is 115° to 180°F. The normal nitrogen outlet temperature is 70°F. The HCGS vaporizer includes controls to stop the nitrogen flow if the temperature drops below 40°F. These two control loops are independent of each other. Therefore a single failure of a sensor, fuse, power supply, etc. would not lead to a nitrogen injection temperature below 40°F.

The HCGS design discussed above provides sufficient assurance that this cracking problem will not occur in the vent header on the purge line piping.

Recommendation 2

Evaluate Inerting System Operation

Review the operating experience of the inerting system to assure that the vaporizer, the low temperature shutoff valve and the temperature indicators have functioned properly. Evaluate the plant calibration, maintenance and operating procedures for the inerting system. Assure that cold nitrogen injection would be detected and prevented.

RESPONSE

Hope Creek currently does not have an operating license and therefore has no operating experience related to the inerting system. Plant calibration, maintenance, and operating procedures will reflect the detection and prevention of cold nitrogen injection.

Hope Creek Operations will prepare a system operating procedure, OP-SOGS-001(Q) requiring an operator to be stationed at the nitrogen vaporizer to monitor and control the N_2 temperature to assure it does not drop below +40°F during the operation of the drywell and torus N_2 inerting system.

Recommendation 3

Test for Drywell/Wetwell Bypass Leakage

Perform a bypass leakage test as soon as convenient to confirm the integrity of the vent system. This test should be conducted during plant operation following normal plant procedures. If no procedures exist, the followig is a general guide for preparing your procedure: pressurize the drywell to approximately 0.75 psi above the wetwell pressure, maintain this drywell pressure and measure the pressure buildup in the wetwell. Any bypass leak area can then be calculated (and is limited by Technical Specifications on many plants) from the wetwell pressure and the drywell/wetwell pressure difference. This will provide an indication that the vent system integrity is intact and that no gross failure exists.

RESPONSE

Not applicable to HCGS.

Recommendation 4

Inspect Nitrogen Injection Line

Conduct an ultrasonic test (UT) as soon as convenient of all accessible welds in the nitrogen injection line from the last isolation valve to the wetwell and drywell penetrations. Also, UT the containment penetrations and the containment shell within 6 inches of the penetration. UT is recommended because cracks would be most likely to initiate on the inside of the pipe or on the side of the metal in contact with cold nitrogen.

RESPONSE

Not applicable to HCGS.

Recommendation 5

Inspect Containment

During the next planned outage, perform a visual inspection of the vent header, downcomers and other equipment in the containment which might be expected to be affected by the injection of cold nitrogen. The vent header should be inspected on the outside and the inside. Also inspect the containment shell or steel liner for at least six inches around the nitrogen penetration.

RESPONSE

Not applicable to HCGS.

Date 9/13/84

Attachment 7

SUBJECT	REVISED FSAR PAGES
DSER Open Item 130- Potential bypass leakage paths- Includes description of a single failure proof feedwater line fill system to prevent containment bypass leakage in feedwater lines following a LOCA	6.2-36, 6.2-62b,T6.2-16 page 1, T6.2-24 page 1, F6.2-xx, and F5.1-3 page 1.
Deletion of steam condensing mode	T1.11-1 page 13, T3.2-1 pages 3&4, 5.4-21, 5.4-25, 5.4-29, 5.4-41, 5.4-43, 5.4-45, 5.4-46, 5.4-49, 5.4-50, 5.4-53, 5.4-55, 6.2-57, 6.2-58, 6.2-64, 6.2-65, T6.2-16 page 7, T6.2-24 pages 4&5, T6.2-26 page 1, F6.2-28 sheets 23&27, F5.4-13, F6.2-47, 440.18-1, 440.18-2, 480.25-1, SRAI (1)-13,

ATTACHMENT 8

DSER Open Item No. 130 (Section 6.2.3)

POTENTIAL BYPASS LEAKAGE PATHS

Although the primary containment is enclosed by the secondary containment, there are systems that penetrate both the primary and secondary containment boundaries, creating potential paths through which radioactive material in the primary containment could bypass the filtration, recirculation, ventilation system. The criteria by which potential bypass leakage paths are determined are the BTP CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants." These criteria include specific requirements for barriers - such as water sealing systems, leakage control systems, and closed systems employed to process or preclude bypass leakage. Utilizing these criteria the applicant has identified in FSAR Table 6.2-15 those lines penetrating the primary containment that are potential reactor building bypass leakage paths, and the bypass leakage barrier(s) that will prevent bypass leakage. Since the applicant has not fully responded to our concerns regarding the Containment Isolation System (Section 6.2.4), we are unable to complete our review of the potential bypass leakage paths. We will report on this matter in a supplement to this SER.

RESPONSE

For the information requested above see the response to DSER Open Item No. 132.

Section 6.2.3 has been revised to include a description of the single failure proof feedwater line fill system to prevent containment bypass leakage in the feedwater lines following a LOCA.

b. Closed Seismic Category I piping system inside or outside primary containment

c. A water seal maintained for at least 30 days following a LOCA

d. The line terminates outside the reactor building in a filtered area

e. Positive in-line air seal

f. A temporary spool piece in the line that is removed during normal operation and replaced by blind flanges so that any leakage through the flange is into the reactor building.

Type a. leakage barriers are considered to limit but not eliminate bypass leakage. Types b. through f. are considered to effectively eliminate any bypass leakage.

The design criterion for bypass leakage is to minimize allowable leakage because of the effect any allowed activity release would have on the accident dose analysis. No bypass leakage paths have been identified. Therefore, no bypass leakage is postulated to reach the environment. The quality group and seismic qualification of the closed systems that are relied upon to eliminate bypass leakage are identified in Table 3.2-1.

The containment leakage is monitored during periodic tests as discussed in Section 6.2.6. Those penetrations for which credit is taken for water seals as a means of eliminating bypass leakage, as outlined in Table 6.2.-15, are preoperationally leak-tested with air or water. For these water seals, either a loop seal is present, or the water for the seal is replenished from a large reservoir. These seals are in:

a. INSERT A ~~Deleted.~~

INSERT A

Feedwater line - The feedwater line fill network is normally used to maintain a water seal in the feedwater lines between the inboard and outboard containment isolation valves following a LOCA. The fill network consists of the HPCI and the RCIC jockey pump loops, as shown on Figures 6.3-1, 6.3-2, 5.4-8 and 5.4-9, and utilizes the HPCI and RCIC injection lines to the feedwater piping to provide makeup water to the piping between the isolation valves. In order to permit the fill network to perform its intended safety function following a single active failure, a piping crosstie is provided between the feedwater lines immediately upstream of the outboard containment isolation valves as illustrated in Figure 5.1-3. This crosstie includes a normally-closed key-locked motor operated valve. This valve, and its respective controls, is provided with Class 1E channelized power such that no single active failure could disable both the crosstie valve and either of the HPCI or RCIC injection valves or associated jockey pumps. The crosstie piping and valve is safety-related and designed to Seismic Category I criteria.

Following a LOCA, the feedwater line fill network is manually aligned from the main control room by opening the HPCI and RCIC injection valves to provide sealing water to the feedwater lines. In the unlikely event that either the HPCI or the RCIC injection line cannot be used as a flow path to the feedwater piping, the motor operated valve in the crosstie would be manually opened from the main control room. Manual operator action to align the fill network is not required sooner than 20 minutes following detection of a LOCA. This is due to the fact that during the time period required to refill the feedwater lines, no radioactive contaminants would be expected to leak through the feedwater isolation valves out to the environment. ADD INSERT # 1

An analysis will be performed to demonstrate that during the initial portion of a LOCA event, water in the feedwater system piping downstream of the No. 3 feedwater heaters will flash to steam and continue to flow toward the RPV until the feedwater line pressure decreases to the containment pressure, at which time the isolation valves will close. The feedwater lines inside containment will contain essentially non-radioactive steam during the depressurization. Based upon the volume of the steam in the feedwater piping from the RPV to the outermost isolation valve, no substantial concentration of radioactive contaminants is expected to buildup through diffusion and mixing at the isolation valves before the water seal is reestablished. Also, the steam that is trapped in the feedwater lines between the outermost isolation valve and the feedwater pump discharge check valves, which consist of approximately 435 feet of pipe for the shortest path to feedwater heater 6C as illustrated in Figure 6.2-XX, will remain pressurized since the feedwater piping is insulated and retains sufficient sensible heat to prevent the steam from condensing. The intent of the analysis will be to verify that pressure in this portion of the feedwater piping will be sufficient to prevent the outward leakage of radioactive contaminants through the isolation valves during the approximate one hour period after the accident until the water seal is reestablished between the isolation valves via the fill system. Thus, no bypass leakage is expected to occur.

INSERT 1

The abnormal operating procedures will include the actions to be taken by the operator to mitigate the unlikely event of the HPCI or RCIC injection line being unavailable as a flow-path to the feedwater piping.

Insert A (Cont'd)

In the event of a feedwater line break inside containment the extent of radioactive contaminants generated would be much less substantial than the recirculation or steam line break and the containment pressure transient is much less severe. Again the feedwater piping as discussed above will remain steam filled and pressurized in the short term during the transient, thus preventing bypass leakage.

As further positive containment of any isolation valve leakage in the short term, the residual unflashed water which is retained in the feedwater system piping upstream of the No. 3 feedwater heaters will form a water seal, thus preventing a direct pathway to the environment.

The analysis will be completed by November 30, 1984.

1. Globe valves - Test pressure in the reverse direction will tend to unseat the valve

2. Butterfly valves - All applicable valves have seat constructions which are designed for sealing against pressure on either side

3. Gate valves - Some valves are tested by pressurizing between the seats. Pressurizing in the normal direction tends to seat one of the discs whereas pressurizing between the discs has applies pressure equally to each seat.

The above noted testing methods satisfy the requirement of ASME Section XI - Division I Article IWV-3423.

e. Requirement: Section III.C.2 states, "Valves unless pressurized with fluid (e.q.water,nitrogen) from a seal system, shall be pressurized with air or nitrogen at a pressure of Pa."

f. Exception: NUREG-0800, SRP 6.2.6 states that hydrostatic testing of containment isolation valves is permissible if the line is not a potential containment atmospheric leak path. The suppression pool, although not a water seal system, provides a water seal for all the valves, except for the feedwater lines, identified in Table 6.2-24 as being tested with water. These valves will be tested at Pa and the limits for liquid leakage are specified in Chapter 16.

Deleted.

g. ~~Exception: Water is maintained in the feedwater line piping by a loop whose elevation difference between the containment inboard isolation valve and the feedwater nozzle is approximately 38 feet, and between the outboard isolation valve and the horizontal run is approximately 16 feet. There is sufficient water in the feedwater piping after vessel blowdown from a LOCA to maintain a water seal for at least 30 days. The ECCS and RCIC jockey pumps can be used to maintain pressure and to provide makeup or to fill up the feedwater system piping in the event that it is necessary.~~

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all pages

TABLE 6.2-16

CONTAINMENT PENETRATIONS

Containment Penetration Number	Line Isolated	Fluid	Line Size, in.	NRC General Design Criterion	ESF System(1)	Valve Number and/or Orifice Plate	Valve Type(1)	Valve Location	Valve Arrangement(2) FIGURE	Type C Test	Length of Pipe from Cont. to Outside Valves, ft	Primary Mode of Actuation(3)	Secondary Mode of Actuation(4)	Normal Valve Position(6)	Shutdown Valve Position(8)	Post-Accident Position(9)	Power Failure Valve Position	Containment Isolation Signal(5)	Valve Closure Time(10)	Test(16)	Remarks(7)
PROCESS LINE PENETRATIONS																					
P-1A	Main Steam	Steam/ Water	26 26 2 2	56	No	[illegible]	[illegible]	Inside Outside Outside Outside	1/A	[illegible]	[illegible]	Instr. gas Comp. air AC motor AC motor	Spring Manual Spring Manual	[illegible]	[illegible]	C C C C	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
P-1B	Main Steam	Steam/ Water	26 26 2 2	56	No	[illegible]	[illegible]	Inside Outside Outside Outside	1/A	[illegible]	[illegible]	Instr. gas Comp. air AC motor AC motor	Spring Manual Spring Manual	[illegible]	[illegible]	C C C C	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
P-1C	Main Steam	Steam/ Water	26 26 2 2	56	No	[illegible]	[illegible]	Inside Outside Outside Outside	1/A	[illegible]	[illegible]	Instr. gas Comp. air AC motor AC motor	Spring Manual Spring Manual Manual	[illegible]	[illegible]	C C C C	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
P-1D	Main Steam	Steam/ Water	26 26 2 2	56	No	[illegible]	[illegible]	Inside Outside Outside Outside	1/A	[illegible]	[illegible]	Instr. gas Comp. air AC motor AC motor	Spring Manual Spring Manual	[illegible]	NO	C C C C	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
P-2A	Feedwater	Water	24 24 4 4 24	55	No	[illegible]	[illegible]	Inside Outside Outside Outside Outside	2/A	[illegible]	[illegible]	Flow Flow Flow DC motor Flow	Comp. air (17) AC motor (18) Manual AC motor (13)	[illegible]	[illegible]	C C C C C	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
P-2B	Feedwater	Water	24 24 4 4 24	55	No	[illegible]	[illegible]	Inside Outside Outside Outside Outside	2/B	[illegible]	[illegible]	Flow Flow Flow DC motor Flow	Comp. air (17) AC motor (18) Manual AC motor (13)	[illegible]	[illegible]	C C C C C	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
P-3	RHR Shutdown Cooling Suction	Water	20 20	55	Yes	[illegible]	GT GT	Inside Outside	3/C	Yes Yes	[illegible]	AC motor AC motor	Manual Manual	C C	[illegible]	C	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
P-4A	RHR Shutdown Cooling Return	Water	12	55	Yes	[illegible]	GT	Outside	4/C	Yes	[illegible]	AC motor	Manual	C	[illegible]	C	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
P-4B	RHR Shutdown Cooling Return	Water	12	55	Yes	[illegible]	GB	Outside	4/D	Yes	[illegible]	AC motor	Manual	C	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
P-5A	Core Spray to Reactor	Water	12	55	Yes	[illegible]	GT	Outside	5/E	Yes	[illegible]	AC motor	Manual	C	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
P-5B	Core Spray to Reactor	Water	12 14	55	Yes	[illegible]	GT GT	Outside Outside	5/E	Yes Yes	[illegible]	AC motor AC motor	Manual Manual	C C	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

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[illegible]

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TABLE 6.2-24

CONTAINMENT PENETRATIONS/ISOLATION VALVE COMPLIANCE WITH 10 CFR 50, APPENDIX J

Penet Number	P&ID Number	System Description	Test Type	Inboard Isolation Barrier Description/ Valve Number	Notes	Inboard Isolation Barrier Description/ Valve Number	Notes
P 1A	M-41	Main steam line A	-	AB V028	6	AB-V032, AB-V059, KP-V010	6
P 1B	M-41	Main steam line B	-	AB V029	6	AB-V033, AB-V060 KP-V009	6
P 1C	M-41	Main steam line C	-	AB V030	6	AB-V034, AB-V061, KP-V008	6
P 1D	M-41	Main steam line D	-	AB-V031	6	AB-V035, AB-V062 KP-V007	6
P 2A	M-41	Feedwater	~~-~~ C	AE-V003	~~15~~ ~~-~~	AE-V002, AE-V001, AE-V021, BD-V005	~~15~~ ~~-~~ ~~15~~ ~~-~~
P 2B	M-41	Feedwater	~~-~~ C	AE-V007	~~15~~ ~~-~~	AE-V006, AE-V005 AE-V021, BJ-V059	~~15~~ ~~-~~ ~~15~~ ~~-~~
P 3	M-51	RHR shutdown cooling suction	C A,C	BC-V071 BC-PSV-4425	- 7,17	BC-V164	~~14~~ -
P 4A	M-51	RHR shutdown cooling return	C C	BC-V016 BC-V118	- -	BC-V013	-
P 4B	M-51	RHR shutdown cooling return	C C	BC-V111 BC-V117	- -	BC-V110	-
P 5A	M-52	Core spray to reactor	C C	BE-V002 BE-V072	- -	BC-V003	-
P 5B	M-52	Core spray to reactor	C C	BE-V006 BE-V071	- -	BE-V007 BJ-V001	- -
P 6A	M-51	LPCI	C	BC-V005, BC-V119	-	BC-V004	-
P 6B		LPCI	C	BC-V017, BC-V120	-	BC-V016	-
P 6C		LPCI	C	BC-V114, BC-V121	-	BC-V111	-
P 6D		LPCI	C	BC-V102, BC-V122	-	BC-V101	-
P 7	M-55	HPCI turbine	C	FD-V001	8	FD-V002	8

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check, but no Type C test is performed or required. The line does not isolate during a LOCA and can leak only if the line or instrument should rupture. Line and instrument leak tightness is verified during the ILRT Type A test.

2. Penetration is sealed by a blind flange or door with double O-ring seals. See Table 6.2-30.

3. Inboard valve tested in the reverse direction. Reverse pressure testing gives equivalent results to normal direction testing and therefore complies with Appendix J.

4. Manual containment isolation valve.

5. Valve is containment isolation valve for more than one penetration.

6. The main steam containment isolation valves and the seal system boundary valves are leak tested in accordance with the ISI program (ASME Section II, Article IWV, Category A valves.) See also Question 410.35 and FSAR Section 6.2.5.7, 6.7.1.3, 6.2.4.4 and 6.7.2.3.

7. Exception to Appendix J required. For further discussion and justification, see Section 6.2.4.4.

8. Gate valve with two-piece construction is tested by pressurizing between the seats and is a conservative seat leakage test.

9. The isolation barrier remains water filled post-LOCA and will be tested with water. Isolation valve leakage is not included in 0.60 La total for type B and C tests.

10. Explosive actuated valve. Not Type "C" tested. Explosive charge tested as category "D" valve per ASME, Section XI, Article IWV. See FSAR Section 6.2.4.4.

11. The valve does not receive an isolation signal but remains open to measure containment conditions post-LOCA. Leak tightness of the penetration is verified during the Type A test.

12. All isolation barriers are located outside containment.

13. The control rod drive (CRD) insert and withdraw lines can be isolated by solenoid valves outside containment. The CRD insert lines each have a ball check valve inside containment.

14. The isolation provisions for this penetration consist of at least one isolation valve and a closed system outside containment. A single active failure can be accommodated. The system is missile-protected and Seismic Category I and becomes an extension of containment post-LOCA. System leak tightness is verified by the testing requirements of Section 1.10 Paragraph III.D.1.V.

15. Deleted. ~~The feedwater containment isolation valves and the seal system boundary valves are leak tested with water in accordance with the ISI program (ASME, Section XI, Article IWV, Category A; see FSAR Section 6.2.4.4).~~

16. This penetration is a boundary between the drywell and the suppression chamber. It is not a path from the primary containment to the environment.

17. Pressure safety valves (PSVs) are type "C" tested when attached to a type "C" test boundary and as category "C" (relief) valves per ASME Section XI, Article IWV.

18. Deleted.

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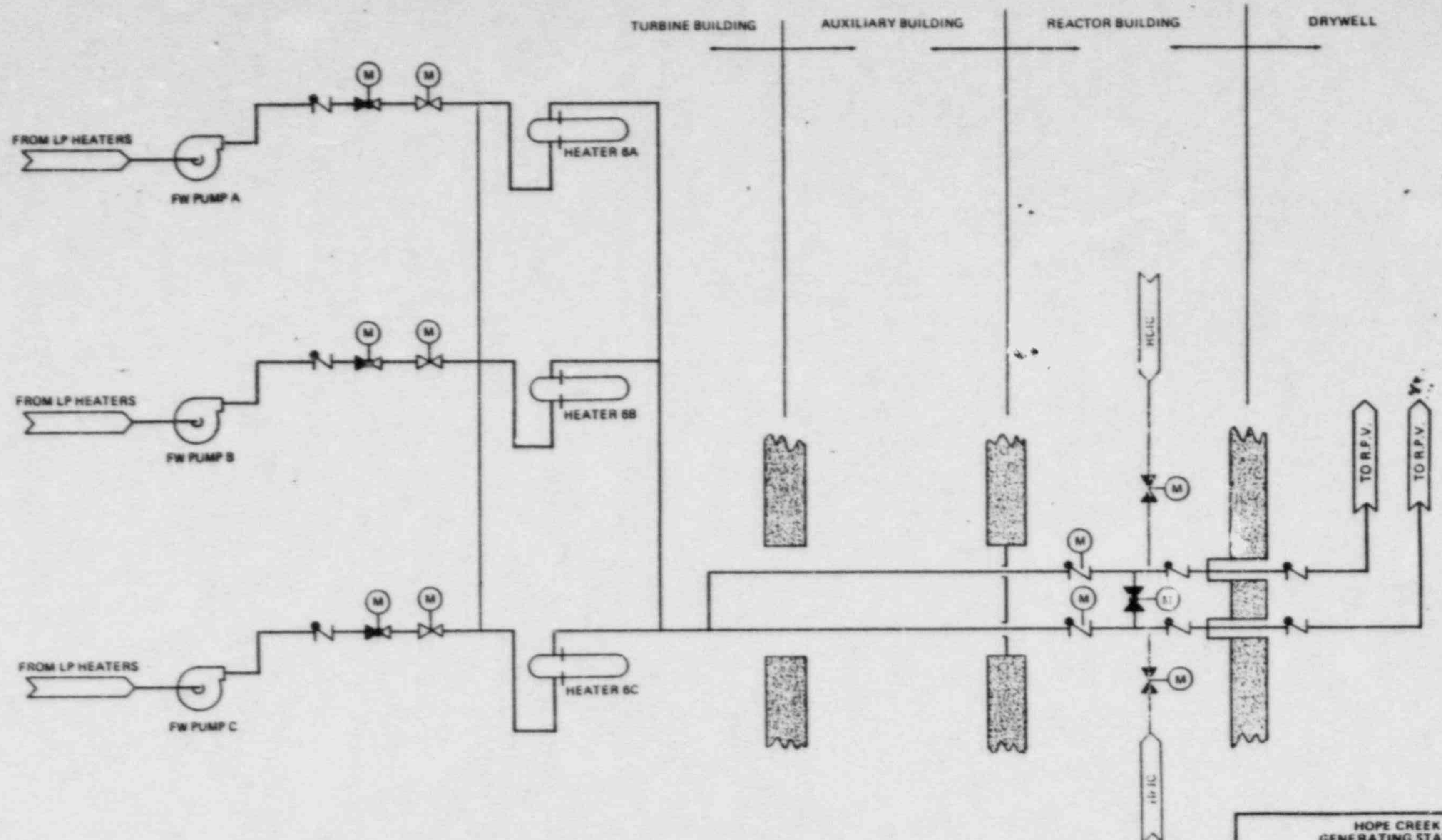

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

FEEDWATER SYSTEM
FEEDWATER PUMPS TO DRYWELL

FIGURE 6.2-XX   AMENDMENT

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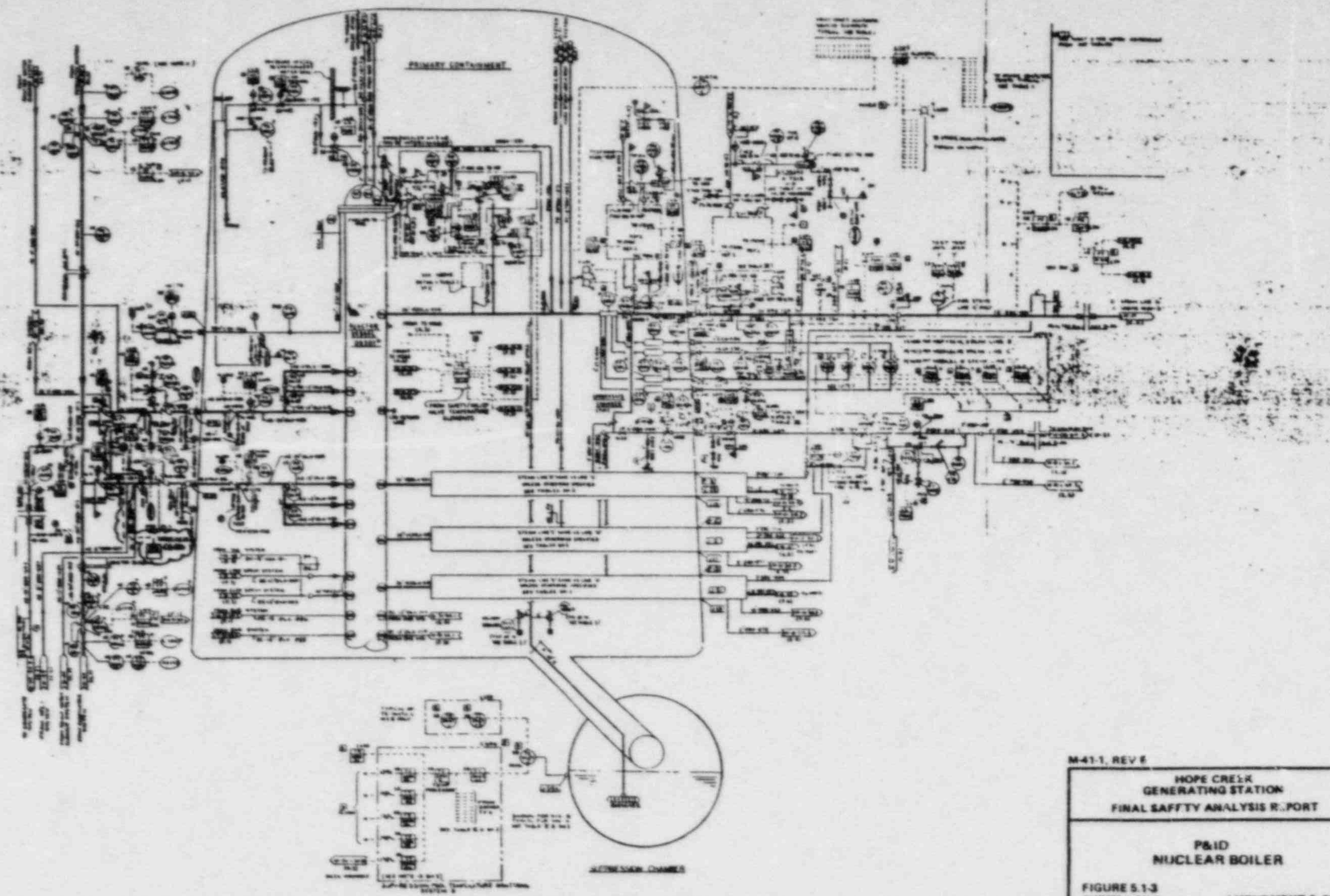

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HOPE CREEK
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FINAL SAFETY ANALYSIS REPORT

P&ID
NUCLEAR BOILER

FIGURE 5.1-3
SHEET 1 OF 2

~~AMENDMENT 5, 06/84~~

DELETION OF STEAM

CONDENSING MODE

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Differences	FSAR Section(s) Where Discussed
6.2.3 (Rev 2)	II.3.e The external design pressure of the secondary containment structure should provide an adequate margin above the maximum expected external pressure.	The secondary containment for tornado depressurization is not designed with any margin above the maximum expected external pressure as stated in Regulatory Guide 1.76.	6.2.3.6
6.2.4 (Rev 2)	~~II.6.g~~ ~~Relief valves used as isolation valves should have a relief setpoint greater than 1.5 times the containment design pressure.~~	~~Relief valve setpoint is not greater than 1.5 times the containment design pressure.~~	~~6.2.4.1~~
	II.6.d Valve nearest the containment and piping between the containment and the first valve, when both valves are located outside primary containment, should be enclosed in a leak-tight or controlled leakage housing.	An enclosure or leak-tight housing has not been designed.	
6.2.5 (Rev 2)	II.4 Following a LOCA, repressurization of the containment should be limited to less than 50% of containment design pressure.	Pressure increase due to main steam isolation valve (MSIV) inleakage after a LOCA will result in repressurization of more than 50% of the containment design pressure.	6.2.5.7
6.5.1 (Rev 2)	II Design of instrumentation for ESF atmosphere cleanup systems to the guidelines of Regularoty Guide 1.52 and to the recommendations of ANSI N509 as summaried in SRP Table 6.5.1-1.	Compliance with the minimum instrumentation requirements for the CREF system are discussed in Table 6.5-4 and for the FRVS systems in Table 6.8-5	6.5.1.2

Principal Components	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
IV. CRD Hydraulic System	4.6.1							
a. Piping and valves, reactor building penetration		P	C	C	III-3	I	Y	
b. Valves, scram discharge volume lines		P/GE	C	B	III-2	I	Y	(10)
c. Valves, insert and withdraw lines		P/GE	A,C	B	III-2	?	Y	(8)(48)
d. Valves, other		P/GE	C	D	B31.1.0	NA	N	
e. Pipe cap, water return line		GE	A	A	III-1	I	Y	
f. Piping, scram discharge volume lines		P	C	B	III-2	I	Y	
g. Piping, insert and withdraw lines		P	A,C	B	III-2	I	Y	
h. Piping, other		P	C	D	B31.1.0	NA	N	(13)
i. Hydraulic control unit including scram accumulator		GE	C	Special	(44)	I	Y	(12)
j. Electrical modules with safety function (27)		GE	C	NA	IEEE-279/323	I	Y	
k. Cable with safety function		P	C	NA	IEEE-279/323	NA	Y	(10)
l. Pumps		GE	C	D	None	NA	N	
m. Pump motors		GE	C	NA	None	NA	N	
V. Engineered Safety Features								
a. RHR system	6.3/5.4.7							
1. Heat exchangers, primary side (shutdown cooling, suppression pool cooling, ~~steam condensing~~)		GE	C	B	III-C & TEMA C(9)	I	Y	
2. Heat exchangers, secondary side		GE	C	C	VIII-1 TEMA C(9)	I	Y	
3. Piping, within outermost containment isolation valves (LPCI, shutdown cooling, head spray)		P	C,A	A	III-1	I	Y	(10)

Principal Components	FSAR Section	Source of Supply (1)	Location (2)	Quality Group Classification (3)	Principal Construction Codes and Standards (5)	Seismic Category (6)	QA Requirements (7)	Comments
4. Piping, beyond outermost containment isolation valves (LPCI, shutdown cooling, suppression pool cooling, head spary, containment a spray, ~~steam condensing~~)		P	C	B	III-2	I	Y	(10)
5. Piping and spray nozzles, containment spray lines within outermost isolation valves		P	A	B	III-2	I	Y	
6. Deleted								
7. Pumps (LPCI, shutdown cooling, suppression pool cooling, head spray, containment spray)		GE	C	B	P&V-II(9)	I	Y	
8. Pump motors		GE	C	NA	NEMA MG-1	I	Y	
9. Valves, inboard isolation, LPCI line & shutdown return line		GE	A	A	III-1	I	Y	(40)
10. Valves, isolation and within (shutdown suction, head spray)		P	C,A	A	III-1	I	Y	(10)(40)
11. Valves, beyond isolation valves (LPCI, shutdown cooling, suppression pool cooling, head spray, containment a spray, ~~steam condensing~~)		P	C	B	III-2	I	Y	(10)(40)
12. Mechanical modules with safety function (27)		GE	C	NA	None	I	Y	
13. Electrical modules with safety function (27)		GE	C	NA	IEEE-279/323	I	Y	
14. Cable with safety function		P	C	NA	IEEE-279/323	NA	Y	(18)
15. ECCS jockey pumps		P	C	B	III-2	I	Y	
16. Piping and valves, reactor building penetration and isolation		P	C	C	III-3	I	Y	
17. ECCS jockey pump motors		P	C	NA	IEEE-323/344	I	Y	
b. Core spray system:	6.3							
1. Piping, within outermost isolation valves		P	A,C	A	III-1	I	Y	(10)
2. Piping, beyond outermost		P	C	B	III-2	I	Y	(10)

demineralized water, but is demineralized water when added. The turbine is driven by a portion of the decay heat steam from the reactor vessel, and exhausts to the suppression pool.

During RCIC operation, the suppression pool serves as the heat sink for steam generated by reactor decay heat. This results in a rise in pool water temperature. The RHR heat exchangers are used to maintain the suppression pool temperature within acceptable limits by cooling the pool water. ~~directly or by condensing the generated steam. The condensate discharge from the RHR heat exchangers can be used as a RCIC pump suction supply or it can be directed to the suppression pool.~~

5.4.6.1.1 Residual Heat and Isolation

5.4.6.1.1.1 Residual Heat

The RCIC system is designed to initiate and, within 30 seconds, discharge a specified constant flow into the reactor vessel over a specified pressure range. The temperature of the RCIC water discharged into the reactor varies from 40°F up to and including 140°F. The mixture of the cool RCIC water and the hot steam results in the following:

a. Quenches the steam

b. Removes reactor residual heat

c. Replenishes the reactor vessel inventory.

The high pressure coolant injection (HPCI) system can perform these same RCIC functions, thereby providing single failure protection. Both systems use different electrical power sources of high reliability that permit operation with either onsite or offsite power. In addition, the RHR system performs its residual heat removal function.

The RCIC system design includes interfaces with redundant leak detection devices. The steam supply to the RCIC steam turbine is automatically isolated upon the receipt of any one of the following leak detection signals:

~~After the RHR system is placed in the steam condensing mode, the operator can select the condensate discharge from the RHR steam condensing heat exchangers as the RCIC pump suction supply. The steam condensing mode of the RHR system is manually placed in operation. Once steam condensing has been established, the water level in the RHR heat exchangers is maintained automatically by means of a regulating valve in the condensate discharge line. Initially, the condensate discharge is directed to the suppression pool. After proper water quality is obtained, the condensate discharge can be directed to the RCIC pump suction. The level control for the RHR heat exchangers is independent from the RCIC control system. The operator selects the flow setpoint of the RCIC system to match the condensate flow rate from the RHR heat exchangers. See Sections 5.4.6.2.5.1, 5.4.6.2.5.2, and 5.4.6.2.5.3 for additional information.~~

5.4.6.1.3 Loss of Offsite Power

The RCIC system electrical power is obtained from a highly reliable source that is maintained by either onsite or offsite power. Refer to Sections 5.4.6.1.1 and 5.4.6.2.4. For further details, see Sections 8.2 and 8.3.

5.4.6.1.4 Physical Damage

The system is designed to meet the requirements of Table 3.2-1 commensurate with the safety importance of the system and its equipment. Moreover, the RCIC is located in a physically different area of the reactor building, a Seismic Category I structure, and uses different divisional power and separate electrical routings from its redundant system, HPCI, as discussed in Sections 5.4.6.1.1 and 5.4.6.2.4. Further discussion can be found in the sections listed below:

a. Protection from wind and tornado effects - Section 3.3

b. Flood design - Section 3.4

c. Missile protection - Section 3.5

2. Turbine exhaust to the suppression pool

3. Makeup supply from the CST to the pump suction

4. Makeup supply from the suppression pool to the pump suction

5. Deleted ~~Makeup supply from the RHR steam condensing heat exchangers to the pump suction~~

6. Pump discharge to the feedwater line, feedwater spray nozzle, including a test line to the CST; a minimum flow bypass line to the suppression pool, and a coolant water supply to accessory equipment.

d. One line-fill jockey pump, and associated piping, valves, and instrumentation.

5.4.6.2.2.2 Design Parameters

Design parameters for the RCIC system components are listed below. See Figures 5.4-8 and 5.4-9 for cross-reference of component numbers listed below:

a. RCIC pump operation (E51-C001)

Flow rate	
Injection flow	600 gpm
Cooling water flow	16 gpm
Total pump discharge (includes no margin for pump wear)	616 gpm
Water temperature range	40 to 140°F
Net positive suction head (NPSH) required	20 feet minimum at 4500 rpm

Deleted

5.4.6.2.5.3 ~~Steam Condensing (Hot Standby) Operation~~

This mode of operation is manually initiated by the operator as follows:

a. Complete the verification made in steps a. through j. of Section 5.4.6.2.5.1.

b. When the reactor is going to be maintained in the hot standby mode and the level starts to drop, the RCIC system can be started by manually pushing the RCIC "manual initiation" pushbutton. See step k. in Section 5.4.6.2.5.1 for RCIC subsequent starts. Concurrently, the RHR system water quality should be readied for vessel injection, as discussed in Section 5.4.6.1.2.2.

c. Adjust the controller so it may be switched to manual mode and maintain the same flow at pressure condition established by step b. in this section. Then switch to manual mode.

d. Adjust the flow controller setpoint as required to maintain the desired reactor water level.

e. When RHR water is ready for vessel injection, open the RHR suction valve to the RCIC system pump. During steam condensing operation, if the RHR produces more condensate than required to maintain reactor level, the excess can be dumped to the suppression pool via the RHR system. Also, if more flow is required than is supplied from the RHR heat exchangers, it comes from the CST.

f. When steam condensing is completed and the RCIC system is no longer required, close the RHR suction valve, manually trip the RCIC system, and turn the flow controller back to automatic.

g. Follow steps n. through s. of Section 5.4.6.2.5.1.

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

5.4.7.1 Design Bases

The residual heat removal (RHR) system consists of four independent loops. Each loop contains a motor-driven pump, piping, valves, instrumentation, and controls. Each loop takes suction from the suppression pool and is capable of discharging water to the reactor vessel via separate low pressure coolant injection (LPCI) nozzles, or back to the suppression pool via a full flow test line. In addition, two loops have heat exchangers that are each cooled by an independent loop of the safety auxiliaries cooling system (SACS). These two RHR heat exchanger loops can also take suction from the reactor recirculation system suction or the fuel pool and can discharge into the reactor recirculation pump discharge, fuel pool cooling discharge, or to the suppression pool and drywell spray spargers. ~~The two heat exchanger loops also have connections to reactor steam via the high pressure coolant injection (HPCI) steam line and can discharge reactor steam condensate to the reactor core isolation cooling (RCIC) pump suction or the suppression pool.~~ For a comparison of the HCGS RHR system with other plants of similar RHR design, see Section 1.3.

5.4.7.1.1 Functional Design Basis

The RHR system has ~~five~~ four subsystems or modes of operation, each of which has its own functional requirements. Each subsystem is discussed separately to provide clarity.

5.4.7.1.1.1 Residual Heat Removal Mode (Shutdown Cooling Mode)

a. The functional design basis of the shutdown cooling mode is to have the capability to remove decay and sensible heat from the reactor primary system so that the reactor outlet temperature is reduced to 125°F, 20 hours after the control rods have been inserted, to permit refueling when the maximum SACS water temperature is 95°F, the core is "mature", and the tubes have reached maximum design fouling. See Section 5.4.7.2.2 for exchanger design details. The capacity of the heat exchangers is such that the time to reduce the vessel outlet water temperature to 212°F corresponds to a cooldown rate in excess of 100°F per hour with both loops in service. However, the flushing drywell and suppression pool vapor space to reduce internal pressure to below design limits.

5.4.7.1.1.5 Deleted ~~Reactor Steam Condensing Mode~~

~~The functional design basis for the reactor steam condensing mode is that, at 1-1/2 hours after a reactor scram, the heat exchanger in one loop of the RHR system, in conjunction with the RCIC turbine, is able to condense all of the steam being generated.~~

5.4.7.1.2 Design Basis for Isolation of RHR System from the Reactor Coolant System

The low pressure portions of the RHR system are isolated from full reactor pressure whenever the primary system pressure is above the RHR system design pressure. See Section 5.4.7.1.3 for further details. In addition, automatic isolation may occur for reasons of vessel water inventory retention, which is unrelated to line pressure rating. See Section 5.2.5 for an explanation of the leak detection system and the isolation signals.

The RHR pumps are protected against damage from a closed discharge valve by means of automatic minimum flow valves, which open upon low main line flow and close upon high main line flow.

5.4.7.1.3 Design Basis for Pressure Relief Capacity

The relief valves in the RHR system are sized on ~~one of three bases:~~ the basis of either thermal relief protection or valve bypass leakage capacity (i.e., excessive leakage past the isolation valves).

~~a. Thermal relief~~

~~b. Valve bypass leakage~~

~~c. Control valve failure and the subsequent uncontrolled flow.~~

~~Items a. and c. result from transients. Item b. results from excessive leak past the isolation valves. Relief valve E11-PSV-F055 is sized to maintain upstream piping at 450 psig and~~

~~10% accumulation with E11-PV-F051 fully open and a reactor pressure equal to the lowest nuclear boiler safety/relief valve spring setpoint. Valve E11-PSV-F097 is sized to maintain upstream pressure at 68 psig and 10% accumulation with both PCV E11-LV-F053 A and B failed open.~~ Valves E11-PSV-F025, -F029, -F030, are set at the design pressure specified in the process data drawing plus 10% accumulation. Valve E11-PSV-4425 is set at the maximum design pressure of the shutdown suction line.

Redundant interlocks prevent opening valves to the low pressure suction piping when the reactor pressure is above the shutdown range. These same interlocks initiate valve closure on increasing reactor pressure.

In addition, a high pressure check valve in each discharge line to the vessel closes to prevent reverse flow from the reactor if the reactor pressure increases above the RHR system pressure. Relief valves in the discharge piping are sized to account for leakage past the check valve.

5.4.7.1.4 Design Basis with Respect to General Design Criterion 5

The RHR system for this unit does not share equipment or structures with any other nuclear unit.

5.4.7.1.5 Design Basis for Reliability and Operability

The design basis for the shutdown cooling mode of the RHR system is that this mode is controlled by the operator from the control room. The only operation performed outside of the control room for a normal shutdown is manual operation of local flushing water admission and discharge valves, which is the means of providing clean water to the shutdown portions of the RHR system.

Two separate shutdown cooling loops are provided. Although both loops are used for shutdown under normal circumstances, the reactor coolant can be brought to 212°F in less than 20 hours with only one loop in operation. With the exception of the shutdown suction, shutdown return, head spray, and steam supply and condensate discharge lines, the entire RHR system is part of the emergency core cooling system (ECCS) and the containment cooling function, and is therefore required to be designed with the redundancy, flooding protection, pipe whip protection, and power separation required of such systems. See Section 6.3 for maximum shutdown cut-in pressures and temperature, minimum ambient temperature, and maximum shutoff head. The pump pressure vessel is carbon steel; the shaft and impellers are stainless steel. A comparison between the available and the required net positive suction head (NPSH) can be obtained from the pump characteristic curves provided on Figures 5.4-14 and 6.3-12. Available NPSH is calculated according to Regulatory Guide 1.1. Additional information can be found in Section 6.3.

b. Heat exchangers - The RHR system heat exchangers are sized on the basis of the duty for the shutdown cooling mode, i.e., mode E of the process data. All other uses of these exchangers~~, including steam condensing,~~ require less cooling surface.

Flow rates are 10,000 gpm (rated) on the shell side and 9000 gpm (rated) on the tube side, which is the SACS water side. Rated inlet temperatures are 125°F shell side and 85°F tube side. The overall heat transfer coefficient is 375 Btu/h-ft²-°F. The exchangers contain 3550 square feet of effective surface. The design temperature range of both the shell and tube sides is 40 to 470°F. Design pressure is 450 psig on both sides. Fouling factors are 0.0005 shell side and 0.0005 tube side. The construction materials are carbon steel for the pressure vessel with 304L stainless steel tubes and stainless steel clad tube sheet.

c. Valves - All of the directional valves in the system are gate, globe, and check valves designed for nuclear service. The injection valves, reactor coolant isolation valves, and pump minimum flow valves are high speed valves, as operation for LPCI injection or vessel isolation requires. Valve pressure ratings, as necessary, provide the control or isolation function, i.e., all vessel isolation valves are rated as ASME B&PV Code, Section III, Class 1 nuclear valves rated at the same pressure as the primary system.

~~Steam pressure reducing valves are designed to regulate steam flow into the heat exchangers from full reactor pressure to maintain downstream pressure at 200 psig.~~

d. ECCS and containment cooling portions of the RHR system:

1. The ECCS portions of the RHR system include those sections described through mode A-1 of Figure 6.3-12. The route includes suppression pool suction strainers, suction piping, RHR pumps, discharge piping, injection valves, and drywell piping into the vessel nozzles and core region of the reactor vessel.

2. Suppression pool cooling components include pool suction strainers, suction piping, pumps, heat exchangers, and pool return lines.

3. Containment spray components are the same as pool cooling except that the spray headers replace the pool return lines.

e. Deleted ~~Steam condensing components - The steam condensing components include steam supply piping and valves, heat exchangers, and condensate piping.~~

f. RHR suction strainers - Each of the four 24-inch RHR pump suction nozzles penetrates the torus wall at a point on the circumference 30 degrees up from the bottom of the pool. The suction nozzles extend 6 inches beyond the torus interior surface, and the strainers are mounted on top of the nozzle penetration end. Each pump suction line is equipped with a nozzle and strainer. Each strainer is designed to have no more than 1-foot head loss at a flow of 10,750 gpm with 50% of the total strainer area plugged. See the paragraph below for the effect on the NPSH. The strainer mesh is sized to screen out all particles greater than 0.125 inches in diameter. Particles equal to or smaller than 0.125 inches in diameter do not impair RHR pump, heat exchanger, drywell spray, and suppression pool spray performance.

The minimum height of the suppression pool water level above the centerline of the strainer base is 11 feet 6 1/2 inches. The system NPSH calculations include head losses for strainer plugging and are based on a reference level 2 feet above the RHR pump mounting in progress. Cooldown rate is subsequently controlled via valves E11-HV-F015, total flow, and E11-HV-F048, heat exchanger bypass flow. All operations are performed from the control room except for opening and closing of local flush water valves.

In the event that the main control room becomes uninhabitable, the RHR shutdown cooling mode can also be initiated from the remote shutdown panel (RSP) on RHR loop B (see Section 7.4.1.4). Operation from the RSP is totally operator controlled and all RHR loop B automatic initiation signals are disabled when the Channel B RSP transfer switch is placed in the "Emergency" position.

The RHR shutdown cooling mode can be manually initiated locally on RHR loop A as a backup to operation of RHR loop B from the RSP. The RHR loop A local pump and valve controls are identified on Table 7.4-3.

The manual actions required for the most limiting failure are discussed in Section 5.4.7.1.5.

Deleted

b. ~~Steam condensing - The operator closes the RHR heat exchanger inlet and outlet valves, starts the SACS water pumps, opens the SACS water valve, opens the RHR heat exchanger vent, and actuates the drain valve logic, which opens the drain valve to the suppression pool. The RHR heat exchanger water level drains to a preset value and the level controller shuts the outlet valve. The operator admits steam slowly to the RHR heat exchangers by slowly increasing the pressure setting. The automatic pressure regulator controls steam flow to maintain steam pressure in the exchanger. The operator regulates the opening of noncondensable vent valves to prevent a buildup of noncondensables in the exchanger. When condensate quality attains the appropriate level, the operator switches condensate from the pool to RCIC pump suction. All operations are performed from the control room.~~

For detailed discussion of the design and operation of the SACS for shutdown cooling and steam condensing, see FSAR Section 9.2.5.

c. A non-NSSS intertie between the station service water system (SSWS) and the RHR system piping allows an

5.4.7.4 Preoperational Testing

The preoperational test program and startup test program as discussed in Chapter 14 are used to generate data to verify the operational capabilities of each piece of equipment in the system: each instrument, each setpoint, each logic element, each pump, each heat exchanger, each valve, and each limit switch. In addition, these programs verify the capabilities of the system to provide the flows, pressures, ~~condensing rates,~~ cooldown rates, and reaction times required to perform all system functions as specified for the system or component in the system data sheets and process data. Logic elements are tested electrically. Valves, pumps, controllers, and relief valves are tested mechanically. Limit switches are tested for correct adjustment and operation. Finally, the system is tested for total system performance against the design requirements as specified above using both the offsite power and standby emergency power. Preliminary heat exchanger performance can be evaluated by operating in the pool cooling mode, but a vessel shutdown is required for the final check due to the small temperature differences available with pool cooling. X

5.4.8 REACTOR WATER CLEANUP SYSTEM

The reactor water cleanup (RWCU) system is classified as a primary power generation system (not an engineered safety feature), a small part of which is part of the reactor coolant pressure boundary (RCPB). Those portions of the system are not part of the RCPB and are isolable from the reactor. The RWCU system may be operated at any time during planned reactor operations, or it may be shut down if reactor coolant quality is within the technical specification limits.

5.4.8.1 Design Bases

5.4.8.1.1 Safety Design Bases

The RWCU system meets the requirements of Regulatory Guides 1.26 and 1.29 (See FSAR Section 3.2) in order to:

a. Prevent excessive loss of reactor coolant because the system is designed as a closed system outside primary containment.

6.2.4.3.2.11 High Pressure Coolant Injection and Reactor Core Isolation Cooling Turbine Exhaust Line Vacuum Breaker Valve Network

The HPCI and RCIC turbine exhaust line vacuum breaker valve network runs between the suppression pool air space to the turbine exhaust lines on the HPCI and RCIC systems and the RHR heat exchanger relief valve discharge lines. The network is designed as a closed system outside primary containment. Each one of the two branching lines to the HPCI and RCIC system is isolated by a single normally open motor-operated gate valve. The branching line to the RHR system is isolated by a normally open motor-operated globe valve.

The system does not receive a containment isolation signal so that a supply of cooling water can be initiated to the reactor. However, should a break be detected in the steam supply line in either the HPCI or RCIC system, the respective portion of the network will automatically isolate.

6.2.4.3.2.12 Suppression Chamber Spray Header Lines

The RHR suppression chamber spray lines have a normally closed, motor-operated isolation valve located outside the primary containment. This valve receives a containment isolation signal. Use of a single valve is justified on the basis that the system is designed as a closed system outside containment.

6.2.4.3.2.13 Residual Heat Removal Heat Exchanger Relief Valve Discharge Lines

Each of ~~T~~the RHR heat exchanger relief valve discharge line~~s~~ to the suppression pool from the RHR heat exchangers is isolated by ~~two~~ a relief valve~~s~~. ~~that discharge through the common header. Also connected to the header is a vent line from the RHR heat exchanger. This line is isolated by a normally closed motor-operated globed valve that does not receive a containment isolation signal. In addition, the RHR vacuum breaker network connects to the header. Isolating the vacuum breaker network is a normally open motor operated globe valve that does not receive a containment isolation signal.~~

~~The discharge line from RHR heat exchanger B is isolated in the same way, except that there are three relief valves that connect to the discharge line.~~

~~These lines are all designed as part of a closed system outside primary containment.~~

6.2.4.3.2.14 Suppression Chamber To Containment Prepurge Cleanup Lines

The suppression chamber to containment prepurge cleanup lines are isolated by two redundant valves outside the primary containment. The valves are normally closed. To limit the possibility of an uncontrolled release of radioactivity, the valves will be sealed closed during reactor operation and will be verified closed. In addition, there are connections to the containment hydrogen recombiners between the first containment isolation valves and the primary containment. These lines are isolated by two motor-operated gate valves. All isolation valves receive a containment isolation signal.

6.2.4.3.2.15 Suppression Pool Cleanup Lines

The suppression pool cleanup lines are isolated by redundant containment isolation valves that close upon a containment isolation signal.

6.2.4.3.2.16 Post-Accident Sampling System Lines

The post-accident sampling system penetrates the primary containment in seven locations. One line is for gathering liquid samples and it forms part of the RCPB. Two lines are sampling return lines to the suppression chamber. The other four lines sample the primary containment atmosphere at different locations within the drywell and suppression chamber. Isolation for these lines consists of two solenoid-operated valves in series, located outside of primary containment. The valves are normally closed, and the penetrations are designed to be a sealed closed system. Administrative procedures prevent the valves from being inadvertently opened by ensuring that power is not supplied to the normally deenergized solenoids until the system is required to operate.

See Chapter 14 for a discussion of the test program.

6.2.4.5 SRP Rule Review

6.2.4.5.1 Acceptance Criterion II.6.d

Acceptance Criterion II.6.d requires that when it is not practical to provide one isolation valve inside and one outside containment, and both valves are located outside the primary containment, that the valve nearest the containment and the piping between the containment and the first valve, be enclosed in a leak-tight or controlled leakage housing. The valve and/or piping compartment must be capable of detecting leakage from the valve shaft and/or bonnet seals and must terminate the leakage.

HCGS does not have a dedicated system for detecting leakage from individual containment isolation valves or from individual lines that penetrate primary containment. Nevertheless, the design is acceptable since reactor building sumps level alarms and flooding alarms in ECCS pump rooms alert the main control room operators of excess leakage. Furthermore, all leakage is collected within the reactor building before its controlled release to the environment.

6.2.4.5.2 Deleted ~~Acceptance Criterion II.6.g~~

~~Acceptance Criterion II.6.g of SRP Section 6.2.4 states that relief valves may be used as isolation valves provided the relief~~

~~setpoint is greater than 1.5 times the containment design pressure.~~

~~For relief valve PSV-F097, shown on Figure 5.4-13, the relief setpoint is less than 1.5 times the containment design pressure. Nevertheless, this is acceptable since valve F097 discharges into the suppression pool. Any increase in valve backpressure due to an increase in suppression chamber pressure resulting from an accident will tend to better seat the valve, thus enhancing its containment isolation capabilities.~~

6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

Following a postulated loss-of-coolant accident (LOCA), hydrogen gas may be generated within the primary containment as a result of the following processes:

a. Metal-water reaction involving the Zircaloy fuel cladding and the reactor coolant

b. Radiolytic decomposition of water in the reactor vessel and the suppression pool (oxygen also evolves in this process)

c. Corrosion of metals and paints in the primary containment.

To preclude the possibility of a combustible mixture of hydrogen and oxygen accumulating in the primary containment, the containment atmosphere is inerted with nitrogen gas before power operation of the reactor.

To ensure that the hydrogen and oxygen concentration in the primary containment is maintained below the lower flammability limit given in Regulatory Guide 1.7, the following features are provided:

a. A containment hydrogen recombiner system

b. A hydrogen/oxygen analyzer system (HOAS)

TABLE 6.2-16

CONTAINMENT PENETRATIONS

Containment Penetration Number	Line Isolated	Fluid	Line Size, in.	NRC General Design Criterion	ESF System(11)	Valve Number and/or Orifice Plate	Valve Type(1)	Valve Location	Valve Arrangement(2) Fig(8)	Type C Test	Length of Pipe from Cont. to Outside Valves, ft.	Primary Method of Actuation(3)	Secondary Method of Actuation	Normal Valve Position(4)	Shutdown Valve Position(10)	Post-Accident Position(9)	Power Failure Valve Position	Containment Isolation Signal(5)	Valve Closure Time,s	Power Source(6)	Remarks(7)
P-212B	BHR Suppression Pool Cooling & System Test	Water	1	56	Yes	BC-PSV-F025C	SRV	Outside	26/B	No	210.5	Pressure	None	C	C	C	NA	-	-	NA	[illegible]
			1			BC-PSV-F025A	SRV	Outside		No	170.4	Pressure	None	C	C	C	NA	-	-	NA	[illegible]
			18			BC-V124	GB	Outside		Yes	49.8	AC motor	Manual	C	C	C	AS IS	B,A	120	A	[illegible]
			18			BC-V125	GB	Outside		Yes	63.5	AC motor	Manual	C	C	C	AS IS	B,A	120	C	[illegible]
			4			BC-V126	GT	Outside		Yes	100.1	AC motor	Manual	C	C	C	AS IS	B,A	37	A	[illegible]
			4			BC-V128	GT	Outside		Yes	52.0	AC motor	Manual	O	C	C	AS IS	None	37	B	[illegible]
			4			BC-V131	GT	Outside		Yes	40.8	AC motor	Manual	O	C	C	AS IS	None	37	C	[illegible]
			1			BC-V206	CK	Outside		Yes	66.7	Flow	None	O	C	C	NA	-	-	NA	[illegible]
P-213A	BHR Relief to Suppression Pool	Steam/Water	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	Pressure	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]		[illegible]	[illegible]	Pressure	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	-	-	[illegible]	P,5
			1	56	Yes	BC-PSV-4431B	SRV	Outside	27/D	No	95.5	Pressure	None	C	C	C	[illegible]	-	-	NA	P,5
			[illegible]			[illegible]	[illegible]	[illegible]		[illegible]	[illegible]	AC motor	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	None	[illegible]	[illegible]	[illegible]
			[illegible]			[illegible]	[illegible]	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	None	[illegible]	[illegible]	[illegible]
			[illegible]			[illegible]	[illegible]	[illegible]		[illegible]	[illegible]	AC motor	Manual	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
P-213B	BHR Relief to Suppression Pool	Steam/Water	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	Pressure	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	-	-	[illegible]	[illegible]
			1	56	Yes	BC-PSV-4431A	SRV	Outside	27/D	No	71.3	Pressure	None	C	C	C	[illegible]	-	-	NA	P,5
			[illegible]			[illegible]	[illegible]	[illegible]		[illegible]	[illegible]	AC motor	Manual	[illegible]	[illegible]	[illegible]	[illegible]	None	[illegible]	[illegible]	[illegible]
			[illegible]			[illegible]	[illegible]	[illegible]		[illegible]	[illegible]	AC motor	Manual	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
			[illegible]			[illegible]	[illegible]	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
P-214A	BHR to Suppression Chamber Spray Header	Water	6	56	Yes	BC-V015	GB	Outside	20/B	Yes	(21)	AC motor	Manual	C	O	C	AS IS	B,A	60	B	5,4
P-214B	BHR to Suppression Chamber Spray Header	Water	6	56	Yes	BC-V112	GB	Outside	20/B	Yes	19.0	AC motor	Manual	C	O	C	AS IS	B,A	60	A	5,4
P-215A	Spare																				8
P-215B	Spare																				8
P-216A	Core Spray Pump Suction	Water	16	56	Yes	BE-V019	GT	Outside	29/B	Yes	31.0	AC motor	Manual	O	O	O	AS IS	None	90	B	5
P-216B	Core Spray Pump Suction	Water	16	56	Yes	BE-V020	GT	Outside	29/B	Yes	32.3	AC motor	Manual	O	O	O	AS IS	None	90	B	5
P-216C	Core Spray Pump Suction	Water	16	56	Yes	BE-V118	GT	Outside	29/B	Yes	32.3	AC motor	Manual	O	O	O	AS IS	None	90	C	5
P-216D	Core Spray Pump Suction	Water	16	56	Yes	BE-V017	GT	Outside	29/B	Yes	32.3	AC motor	Manual	O	O	O	AS IS	None	90	A	5

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TABLE 6.2-24 (cont)

Penet Number	P&ID Number	System Description	Test Type	Inboard Isolation Barrier Description/ Valve Number	Notes	Inboard Isolation Barrier Description/ Valve Number	Notes
P 213A	M-51	RHR relief to torus line	A	BC-PSV-4431B	9,7,12,17	-	-
P 213B	M-51	RHR relief to torus line	~~C(W)~~	~~BC-V155~~	~~3,9,12~~	~~BC-V156~~	
			~~C~~	~~BC-V233~~	~~3,9,12~~	~~BC-256, BC-255~~	~~5~~
			~~A~~	~~BC-PSV-F055A~~	~~9,7,12,17~~		
			A	BC-PSV-4431A	9,7,12,17	-	-
P 214A	M-51	RHR to torus spray header	C	BC-V015	7,12,14	-	-
P 214B	M-51	RHR to torus spray header	C	BC-V112	7,12,14	-	-
P 216A	M-52	Core spray pump suction	C(W)	BE-V019	7,8,9,12,14	-	-
P 216B	M-52	Core spray pump suction	C(W)	BE-V020	7,8,9,12,14	-	-
P 216C	M-52	Core spray pump suction	C(W)	BE-V018	7,8,9,12,14	-	-
P 216D	M-52	Core spray pump suction	C(W)	BE-V017	7,8,9,12,14	-	-
P 217A	M-52	Core spray test and min flow to torus	A	BE-PSV-F012B	7,12,17	-	-
			C(W)	BE-V026	9,12,14	-	-
			C(W)	BE-V036	9,12,14	-	-
P 217B	M-52	Core spray test & min flow to torus	A	BE-PSV-F012A	7,12,17	-	-
			C(W)	BE-V025	9,12,14	-	-
			C(W)	BE-V035	9,12,14	-	-
P 219	M-57	Torus purge outlet & torus vacuum relief	C	GS-V080	3,12	GS-PSV-5030	-
			C	GS-V028	3,12	GS-V076, GS-V027	-
			C	GS-V007	8,12	GS-V006	-
P 220	M-57	Torus purge outlet & torus Vacuum relief	C	GS-V022	3,5	GS-V020, GS-V021,	5
						GS-V023, GS-V009	5
					8	GS-V008	-
			C	GS-V010	3	GS-PSV-5032	-
			C	GS-V038			
P 221A-D		Construction hatch	A	-	19	-	-
P 222	M-53	Torus water cleanup return	C(W)	EE-V002	8,9,12	EE-V001	-
P 223	M-53	Torus water cleanup supply	C(W)	EE-V003	8,9,12	EE-V004	-
P 224		Spare	A	-	-	-	-

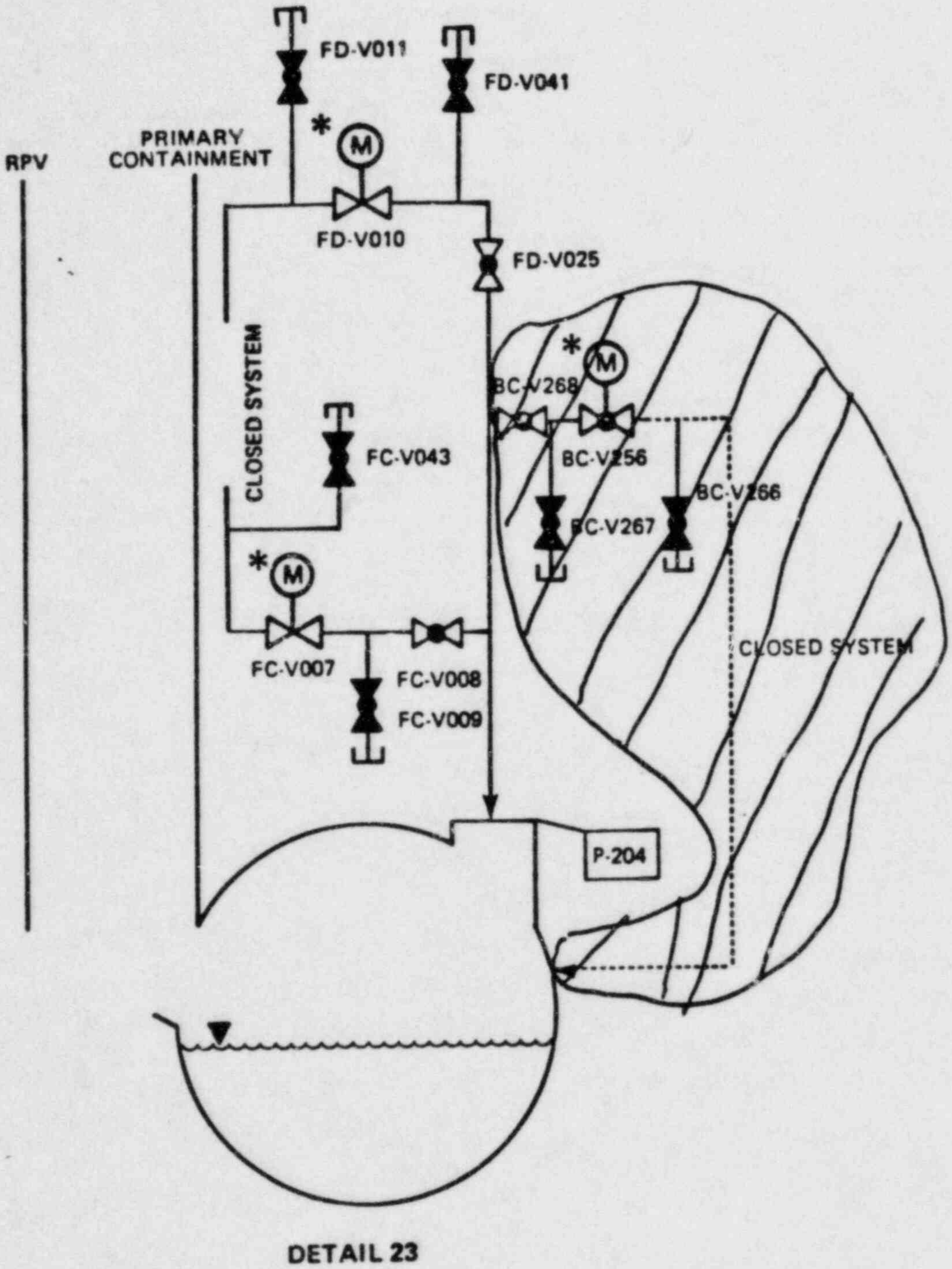
(W) Tested with water.

Penet Number	P&ID Number	System Description	Test Type	Inboard Isolation Barrier Description/ Valve Number	Notes	Inboard Isolation Barrier Description/ Valve Number	Notes
P 201	M-55	HPCI turbine exhaust	C(W)	FD-V006	8, 12	FD-V004	7
				FD-V007	8, 12		
P 202	M-55	HPCI pump suction	C(W)	BJ-V009	8, 9, 12, 14	-	-
P 203	M-55	HPCI minimum return	C(W)	BJ-V016	8, 9, 12, 14	-	-
P 204	M-55	HPCI & RCIC vacuum network	C	FC-V007, FD-V010,	12		-
			C	BC-V256			
P 207	M-49	RCIC turbine exhaust	C(W)	FC-V005	8, 12	FC-V003	7
			C	FC-V006	8, 12		
P 208	M-49	RCIC pump suction	C(W)	BD-V003	8, 9, 12, 14,	-	-
P 209	M-49	RCIC min return	C(W)	V007	9, 12, 14	-	-
P 210	M-49	Non-condensable gas from RCIC vacuum pump	C(W)	FC-[illegible]	7, 9, 12, 20	FC-V010	7
P 211A	M-51	RHR pump suction	C(W)	BC-V001	7, 9, 12, 14, 8	-	-
P 211B	M-51	RHR pump suction	C(W)	BC-V006	7, 9, 12, 14, 8	-	-
P 211C	M-51	RHR pump suction	C(W)	BC-V103	7, 9, 12, 14, 8	-	-
P 211D	M-51	RHR pump suction	C(W)	BC-V098	7, 9, 12, 14, 8	-	-
P 212A	M-51	RHR torus water cooling & system test	A	BC-PSV-F025 D	7, 12, 17	-	-
			A	BC-PSV-F025 B	7, 12, 17	-	-
			C(W)	BC-V028, BC-V027	9, 12, 14	-	-
			C(W)	BC-V026, BC-V034	9, 12, 14	-	-
			C(W)	BC-V031, BC-V260	9, 12, 14	-	-
P 212B	M-51	RHR torus water cooling & system test	A	BC-PSV-F025 A	7, 12, 17	-	-
			A	BC-PSV-F025 C	7, 12, 17	-	-
			C(W)	BC-V124, BC-V125	9, 12, 14	-	-
			C(W)	BC-V126, BC-V128	9, 12, 14	-	-
			C(W)	BC-V131, BC-V206	9, 12, 14	-	
~~P 213A~~	~~M-51~~	~~RHR relief to torus line~~	~~C(W)~~	~~BC-V037~~	~~7, 9, 12~~	~~BC-V036~~	
			~~C~~	~~BC-V253~~	~~9, 8, 12~~	~~BC-V256, BC-V252~~	~~5~~
			~~A~~	~~BC-PSV-F097~~	~~9, 7, 12, 17~~		
			~~A~~	~~BC-PSV-F055B~~	~~9, 7, 12, 17~~		

(W) Tested with water.

SYSTEM ISOLATION VALVES WITH PRIMARY CONTAINMENT ISOLATION(1)

Line Isolated	Valve(4) Number	Operator Number	Essential/ Non-Essential	Isolation(2) signals	(3) Comments
RHR to Radwaste	BC-V042 BC-V041	HV-F049 HV-F040	Non-Essential Non-Essential	B,D B,D	A
RHR to Process Sampling	— —	BC-SV-F079A BC-SV-F080A	Non-Essential Non-Essential	B,D B,D	A
RHR To Process Sampling	— —	BC-SV-F079B BC-SV-F080A	Non-Essential Non-Essential	B,D B,D	A
RHR to Post-Accid. Sampling	— —	RC-SV-F0645A RC-SV-F0645B	Non-Essential Non-Essential	None None	A,B,C
RHR to Post-Accid. Sampling	— —	RC-SV-F0646A RC-SV-F0646B	Non-Essential Non-Essential	None None	A,B,C
RHR to Contain. Hydrogen Recomb.	GS-V520 GS-V150	HV-5055A HV-5057A	Non-Essential Non-Essential	A,B,C A,B,C	A
RHR to Contain. Hydrogen Recomb.	GS-V521 GS-V151	HV-5055B HV-5057B	Non-Essential Non-Essential	A,B,C A,B,C	A
RCIC to CST	BD-V012	HV-F022	Non-Essential	A	D
RCIC from CST	BD-V001	HV-F010	Essential	None	
RCIC to Lube Oil Cooler	BD-V022	HV-F046	Essential	None	
HPCI to CST	BJ-V010	HV-F008	Non-Essential	A,B	
HPCI from CST	BJ-V005	HV-F004	Essential	None	
HPCI to Lube Oil Cooler	BJ-V028	HV-F059	Essential	None	
Steam Condensing	BC-V161	HV-F052A	Non-Essential	None ~~A,B~~	E
Steam Condensing	BC-V022	HV-F052B	Non-Essential	None ~~A,B~~	E
Steam Condensing Warmup	BC-V374	HV-4428	Non-Essential	None ~~A,B~~	E



HOPE CREEK GENERATING STATION

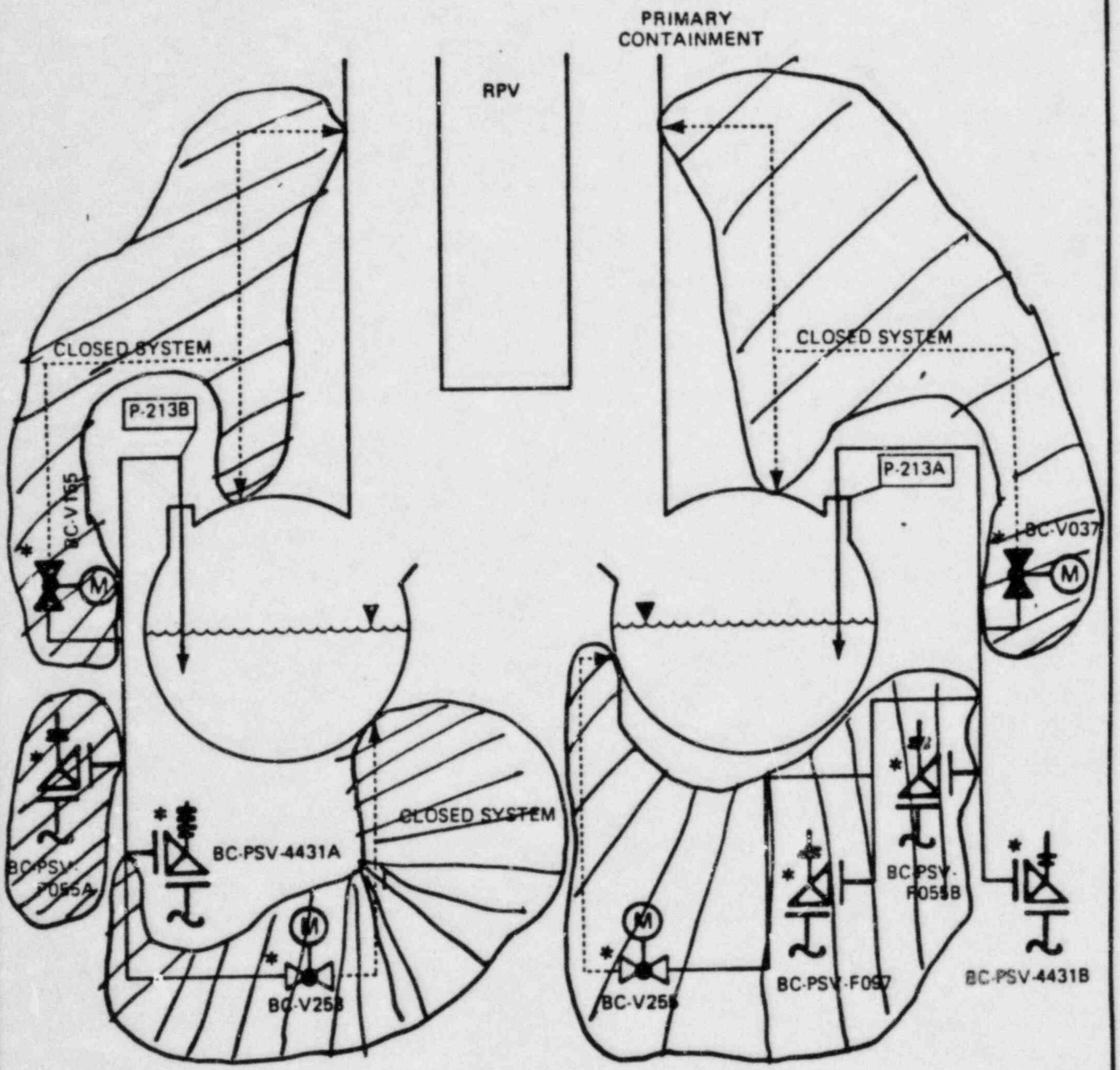
FINAL SAFETY ANALYSIS REPORT

HPCI AND RCIC VACUUM BREAKER NETWORK LINE

FIGURE 6.2-28
SHEET 23 OF 48

AMENDMENT 6. 06/84

*(SEE LEGEND)



DETAIL 27

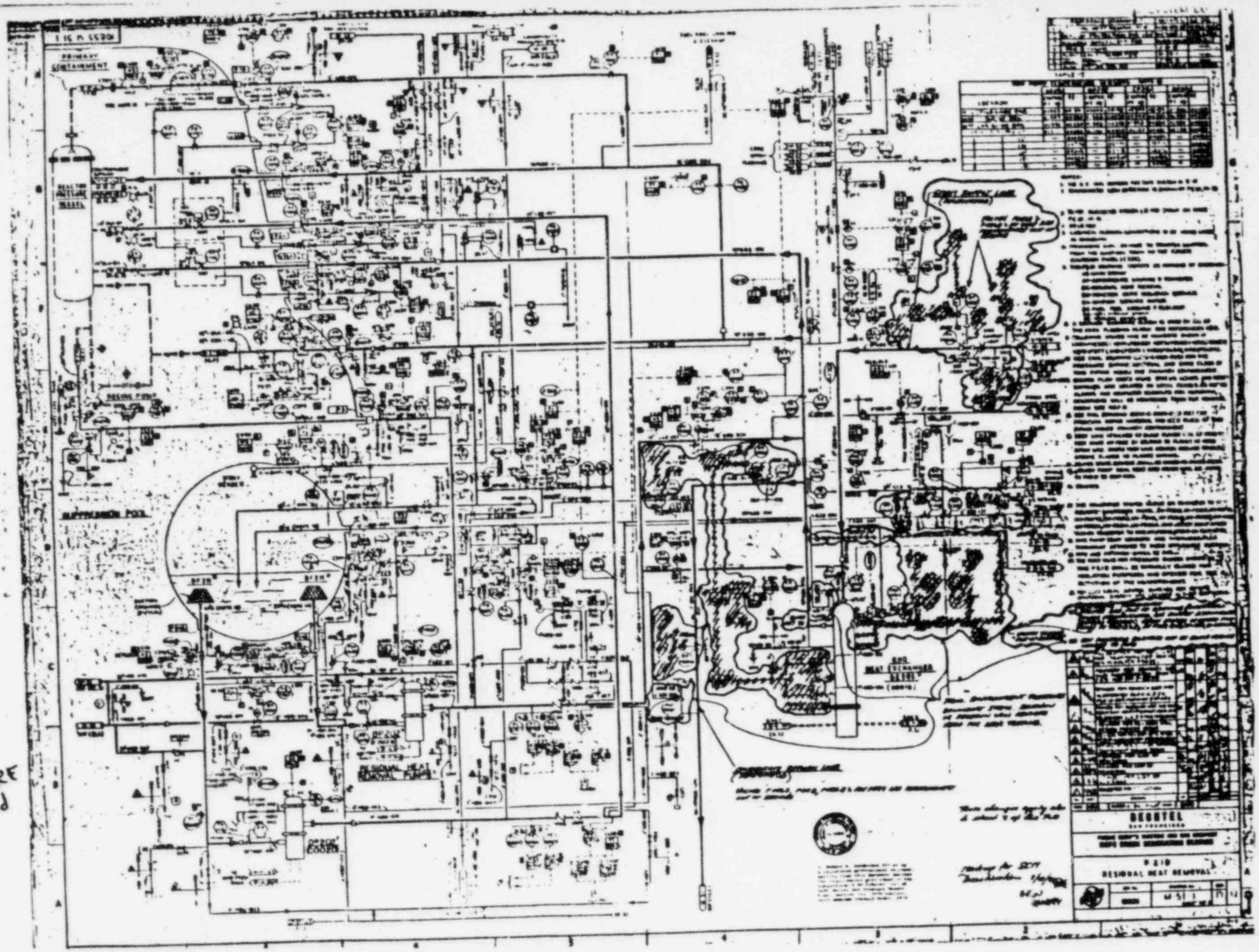
*(SEE LEGEND)

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

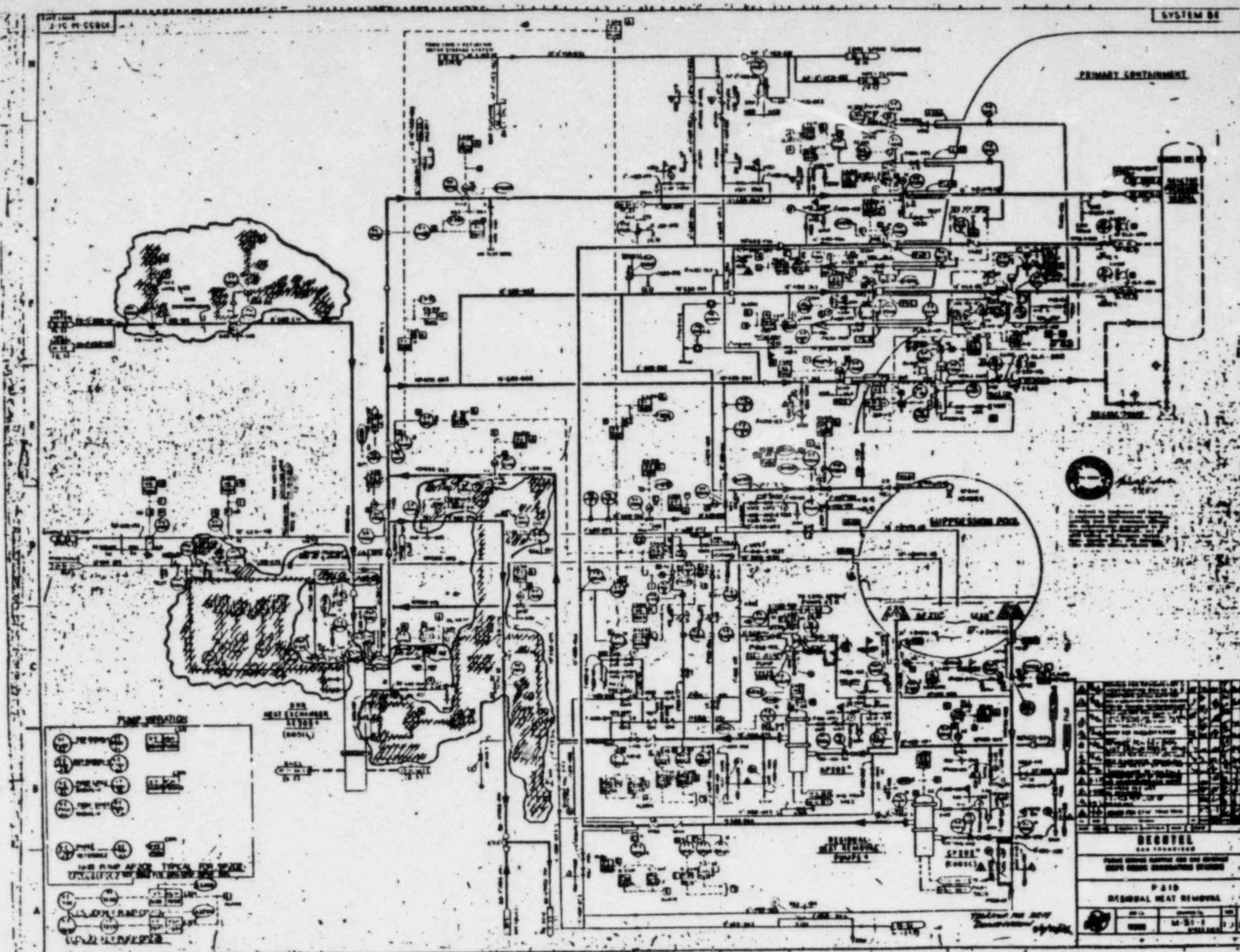
RHR RELIEF TO
SUPPRESSION CHAMBER LINES

FIGURE 6.2-28
SHEET 27 OF 48

AMENDMENT 6, 06/84



FSAR FIGURE
5.4-13 and
6.2-47



FSAR FIGURE
5.4-13 and
6.2-47

QUESTION 440.18 (SECTION 5.4.7)

Operation of the RHR system in the steam condensing mode involves partial draining of one or both RHR heat exchangers and introduction of reactor steam into initially cold lines and heat exchangers. Describe the methods (e.g., valve operation, air introduction, etc.) and provisions to be used to prevent occurrence of water hammer during the initiation of operation in this mode, and the change to the pool cooling mode. When the RHR is used in the steam condensing mode with one or both heat exchangers, can the jockey pump system fill the lines to the injection valve in the core spray and RHR lines? If not, what procedures would be used to prevent water hammer following startup of the core spray or RHR pump.

Pressure relief valves and lines designed to prevent overpressurization of the RHR system are routed outside containment before being returned to suppression pool. Discuss design provisions made to mitigate possible water hammer in these lines.

RESPONSE

Refer to Figure 5.4-13 for valve numbers. The methods used to prevent the occurrence of water hammer during steam condensing initiation are:

a. lowering the heat exchanger water level while at the same time admitting air and then using low pressure steam (approximately 10 psig) by cracking open the steam pressure control valves F051 and F052;

b. initially admitting steam at a low pressure into the air-blanketed heat exchanger and then slowly increasing steam pressure to 200 psig to avoid high pressure surges; and

c. opening all valves slowly to avoid sudden flow surges.

The methods used to prevent the occurrence of water hammer following the termination of steam condensing and the change to the pool cooling mode are:

a. closing the heat exchanger condensate discharge, closing the steam supply valves, and allowing air to enter the heat exchanger through open vent valves E11-F104 and F103;

b. cracking open the valve (F003) connecting the heat exchanger to the main pump loop; and

c. opening the high point vent and filling the heat exchanger shell and connecting piping using the condensate supply valve.

The RHR injection lines remain water-filled during steam condensing operation as described in revised Section 6.3.2.2.6. The core spray system is not affected by steam condensing, and the core spray injection lines will remain full.

Design provisions to mitigate possible water hammer in the RHR pressure relief valve lines are discussed in Section 6.3.2.6.

The RHR steam condensing mode has been deleted from the HCGS design.

QUESTION 480.25 (SECTION 6.2.4)

Table 6.2-16 indicates that the RHR relief valve (PSV-F097) to the suppression pool setpoint is less than 1.5 times the containment design pressure. Provide justification for the lesser setpoint.

RESPONSE

~~The justification for the setpoint for the RHR relief valve E11-PSV-F097 being less than 1.5 times the containment design pressure is described in Section 6.2.4.5.2, SRP Rule Review.~~

RHR relief valve E11-PSV-F097 has been abandoned in place due to the deletion of the RHR steam condensing mode. It is no longer functionally operable in the HCGS design.

of revised Table 3.2-1). However, it is designed to accommodate design flood and seismic event.

e) The roof drainage system is not Q-listed and is not a "structure system or component" that should be included in Table 3.2-1. Roof drainage cannot adversely impact safety-related equipment because of flood protection measures discussed in Section 3.4.1.1.

Site grading should not be included in Table 3.2-1 as discussed in the response to item a.20 of SRAI(1).

f) The purge (containment inerting) system is described under the containment atmosphere control system (Item V.d.3), not the reactor building ventilation system (Item VIII.c).

g) Containment isolation valves used at HCGS meet the requirements outlined in GDCs 54-56 of 10 CFR 50 Appendix A as outlined in Table 6.2-16.

h) Table 3.2-1, Item V.a has been revised to clearly identify piping, valves and other equipment used for suppression pool cooling, ~~steam condensing~~ and suction lines for the shutdown cooling modes of the RHR system. The RHR steam condensing mode has been deleted from the HCGS design.

i) There are no nuclear codes and standards applicable to the design and manufacture of the HPCI and RCIC turbines. Approximately 50 to 75 components of the turbines' lubricating oil systems contribute to the electrohydraulic control of the governing valves. Footnotes (11) and (48) provide the applicable quality assurance, documentation, maintenance, and material fabrication information.

j) Process and effluent radiation monitoring systems are listed in Item X.d of Table 3.2-1. See Sections 7.6 and 11.5 for the differences between the process radiation monitoring systems and the process and effluent radiation monitoring systems.

k) Table 3.2-1 will be revised to incorporate the Emergency Response Facilities Data Acquisition System (ERFDAS). This system is non-Q, non-class 1E and non-seismic, except for the Class 1E isolation devices supplied with the ERFDAS.

l) The MSIV sealing system consists of valves, valve operators, and piping only; the sealing system is supplied by the instrument gas system (see Item XVIII.b).

m) The unit vent stacks are Q-listed as shown in revised Table 3.2-1, Item XIX.g.

ATTACHMENT 9

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, and to advise the General Manager - Nuclear Safety Review on operational considerations for all 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. Recommend to the General Manager - Nuclear Safety Review written approval or disapproval of items considered under 6.1.5.6 (b) through (e) above.

c. 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.

*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 OSR or the General Manager - Nuclear Safety Review.

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. In addition, an annual effectiveness audit of the Q.A. program shall be conducted under the cognizance of 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.

STREAMLINING OF SORC REVIEW PROCESS

1.0 INTRODUCTION

The purpose of the recommendations on the Station Operations Review Committee (SORC) streamlining is to allow the SORC to focus its efforts on its area of primary responsibility, assuring the safe, efficient operation of the station. The paper flow problems of the present SORC process identified earlier by other review groups and as part of the Action Plan 2.2.1 effort, have been well documented, and are generally understood by top management throughout the Nuclear Department. Therefore, the intent of the streamlining process is to remove the bulk of the routine paper processing and review function and transfer this to qualified individuals within the station organization who will provide the detailed review in an environment outside of the committee process. A simultaneous reduction in paper flow to the SORC and improvement in the quality of the review is expected to result. The SORC would then be expected to have more time to consider items in-depth and to act as a senior review and evaluation committee.

In accomplishing this, a system of Qualified Reviewers (QR) will be established. The designation of Qualified Reviewer will be based on specified credential requirements, and the Qualified Reviewer would function to provide documented evidence of review and findings. In addition, on most of the important documents being reviewed, the Qualifed Reviewer would be expected to provide a verbal report to the full SORC just as the document sponsor would be expected to provide a verbal report to the full SORC.

The Unreviewed Safety Question determination is recommended to be part of the responsibility of the newly created Nuclear Safety Review (NSR) Department. The intent of this is to allow application of the experience and credentials of more analytically oriented individuals into the review process in establishing the USQ rather than that of the operations oriented individual. It is felt that the critical operability review provided by the SORC will be enhanced by the deeper technical review provided by the NSR. Moving the USQ determination to the NSR Department also removes the burden of that responsibility and allows the SORC more time for more appropriate issues that are important for station operation.

Notwithstanding, it is expected that concerns of operations oriented individuals will also be considered within the overall safety review context, and that a dialogue between the On-Site Safety Review Group (SRG) and SORC members as well as the system of Qualified Reviewers will help assure communications. Concerns of operationally and technically oriented staff will be considered by the full SORC during the final review process.

In general, streamlining the SORC is expected to improve its efficiency and productivity in areas beneficial to the station operation. Establishing the Nuclear Safety Review Department is expected to enhance the technical and analytical aspects of the safety review process.

The following sections of the report present descriptions of the proposed flows of documents in the new review process. Starting from the present SORC procedure related review process, the recommended procedure related and non-procedure related document review processes are discussed. The interaction of the SORC with the Nuclear Safety Review Department are included in the discussion to note the reassignment of responsibilities and to define points of interaction. It is expected that the recommended streamlining of the SORC review process willenhance the quality of safety review and simultaneously inprove organizational efficiency.

2.0 ADMINISTRATIVE AND OPERATING PROCEDURES

The flow of paper work and review responsibilities of the Station Operations Review Committee as it is functioning today is shown in Figure 1. The recommended flow which denotes signficant changes is shown in Figure 2. Using as a reference the proposed Technical Specification change concerning procedure reviews for Salem, a very general statement is that the intent of the recommendation presented within this document is consistent with the proposed Tech Spec change except with regards to the Unreviewed Safety Question determination. The general thrust of the recommendation is to remove from SORC the burdensome review of thousands of procedures and changes to procedures on a periodic basis with which the SORC is now encumbered. As mentioned in previous studies of the SORC and the reports created as part of Action Plan 2.2.1, these procedures can be more effectively reviewed outside of the SORC by using a system of Qualified Reviewers.

Referring now to Figure 2, consider the review process for the administrative and operating procedures. For discussion, the review process will encompass procedures newly created for the system, changes to existing procedures and safety evaluations which would be provided by the procedure originator. A procedure originator will be defined as the individual with responsibility for creation of a new procedure or a modification to an existing one. The procedure originator will also have responsibility for submitting a safety evaluation with the document that addresses the 10 CFR 50.59 criteria. A standard form will be created to assist in the process and will also serve as an effective record keeping device.

After the document has been created and the safety evaluation performed, the complete package is sent to a Qualified Reviewer (QR). The SORC Chairman is not only responsible for the normal operation of the SORC but also responsible for establishing and maintaining the system of Qualified Reviewers. For review of procedures and changes thereto as well as the safety evaluations of same, this list of Qualified

Reviewers will include independent individuals or groups knowledgeable in the areas of concern. The Qualified Reviewer will be a different individual than the individual or group who prepared the document under consideration. This list of Qualified Reviewers will be comprised of staff located primarily at the plant as part of the operations group, or as part of the service groups identified in the new organization. Qualifications criteria will be established for the Qualified Reviewers according to ANS/ANSI 3.1 and proper records of same will be maintained by the SORC Chairman. The records will demonstrate compliance with established criteria, include documentation of special training, and provide a suitable audit trail.

The Procedure Originator is also responsible for assuring that Quality Assurance is notified so that proper compliance of 10CFR50 Appendix B requirements is established. Quality Assurance will then be responsible for establishing hold points and other actions that are normally required.

Once the assignment of the appropriate Qualified Reviewer (an individual or group) has been made, the initial action on the part of the Qualfied Reviewer will be to establish whether or not a cross-disciplinary review is required. If such a review is deemed appropriate, the Qualified Reviewer will be responsible for assuring that the appropriate reviews are performed. The Qualified Reviewer at the station will be specifically responsible for the operational review of the document and a general review of the safety evaluation. The intent of this recommendation is to utilize the expertise of plant staff where it would be most beneficial, that is, in the area of operational considerations.

After the Qualified Reviewer has finished his responsibilities for the complete package (document and safety evaluation) review, the package is transmitted to the Manager who will be responsible for two actions. The initial action is that the Manager will determine if any significant safety issue is involved. If there is none, the package will be approved by the manager and implemented without SORC consideration. If the manager decides that the package contains safety issues that should be considered by the SORC, the package is forwarded to the SORC Chairman and to the Nuclear Safety Review Department where the Unreviewed Safety Question determination will be made.

The responsibilities of the QR in the recommended scheme is different than the responsibilities considered prior to the Action Plan 2.2.1 recommendations. Earlier recommendations were based on the SORC still having the responsibility for the USQ determination rather than having the USQ determination made by the Nuclear Safety Review Department as proposed herein. The actual review for USQ determination by NSR can be handled as part of the Off-Site Review (OSR) Group or the On-Site Safety Review (SRG) Group depending on the nature and general character of the material in the document.

After the document has been forwarded to the Station Operations Review Committee Chairman, the Chairman then brings it forward on to the schedule of SORC meetings for evaluation by the full SORC. On major items, evaluation might include a verbal report on the document by the originator, a verbal report by the Qualified Reviewer, as well as a report from the Nuclear Safety Review Department regarding the USQ determination and safety review. The SORC will therefore concentrate its efforts on important or critical issues and not expend time on the more mundane issues since only major items will reach the full SORC for consideration.

If an Unreviewed Safety Question exists or a Technical Specification change is involved, the document would be forwarded through the normal corporate licensing channel to be submitted to the NRC for approval prior to approval and implementation by the General Manager - Salem Operations. If no Unreviewed Safety Question exists, documents sent to the Station General Manager for approval would include the administrative procedures, all the security procedures and security plan, and documents associated with the emergency plan and fire protection plan. Documents sent to the Department Managers or Assistant Manager for approval would include the non-administrative procedures.

As a result of the streamlining process, the non-administrative procedures and changes thereto would have been initiated and prepared by appropriate Departmental staff, reviewed by the Qualified Reviewers, and provided to the Departmental Manager for his signature and approval. In the streamlined SORC, the Departmental Manager approval is all that will be required for the procedures to be implemented when no significant safety issues are involved and no USQ is determined to exist.

Verification that the process is being performed to procedures can be accomplished by normal Quality Assurance audits of the Department records. Written records of the reviews performed as a regular part of the SORC will be the responsibility of the SORC Chairman who will maintain up-to-date records of the disposition of documents.

3.0 TESTS AND EXPERIMENTS, CHANGES TO TECH SPECS, AND CHANGES OR MODS TO THE PLANT OR EQUIPMENT

The streamlining of the SORC process with regard to reviews of tests and experiments, changes to the Tech Specs, and changes-to or modifications-of the plant systems or equipment, follows a pattern similar to that being recommended for the procedure review described in the previous **Section 2.0**. There are some variations to that process however in that the final acceptance and approval recommendation comes from the Station Operations Review Committee proceedings for all documents.

Referring now to Figure 3, the documents subjected to the review process will be transmitted to the SORC and to NSR. The SORC Chairman receives the document from the originator with verification that a copy has been transmitted the Nuclear Safety Review Department for the Unreviewed Safety Question determination and for the safety review. The USQ determination being provided by Nuclear Safety Review is consistent with the previous recommended flow of responsibilities. NSR is expected to perform the USQ determination using not only its own staff of technically qualified specialists, but also through the use of a system of Qualified Reviewers established throughout the corporate organization for explicit support of NSR.

While the QR system established as part of the streamlined SORC in part removes the burden of procedure review and in part supports the SORC with expert opinions, the QR system established for NSR augments the technical review function exclusively.

Upon receiving the document for review, the SORC Chairman makes the decision whether to use a Qualifed Reviewer or whether to proceed with the review by the SORC. Changes of a relatively unimportant nature would be expected to go directly to the SORC for review and recommendation. Documents relating to significant tests or experiments, or major changes in parameters specified in the Tech Specs would be first routed to a Qualified Reviewer for assessment. The Qualified Reviewer initially would also make a determination (as before) as to whether or not cross-disciplinary review would be required. Generally speaking, for documents of significance, a cross-disciplinary review would be required.

The function of the Qualified Reviewer and the SORC is to assure that an operability review of the document has been performed. During the course of this operability review, it would be a normal procedure for the staff to consider the safety aspects; however, the SORC review responsibilities will also include consideration of the USQ determination as received from NSR. The intent is to move the requirement of the more technical and analytical review from SORC responsibility and move those considerations back within the technical review responsibility of the Nuclear Safety Review Department.

The SORC Chairman receives documents for review either directly or through the QR system. The full Station Operations Review Committee has the responsibility to consider the document and to provide a recommendation for approval. Typically, the SORC would request discussion of the document by its sponsor and by the Qualfied Reviewer if used so that the SORC may act in a senior review perspective without having to invest significant amounts of committee time discussing minutiae. As part of the document review by the full SORC, the SORC would also receive the USQ determination and results of the safety review from the Nuclear Safety Review Department. Thus, the full SORC would perform a comprehensive evaluation, reporting, and recommendation function for the General Manager - Salem Operations. In the event that an Unreviewed Safety Question is determined to exist or a change to the Technical Specifications is involved, the recommendation from SORC will be to process the document through normal licensing channels to the NRC. If no USQ exists, then the recommendation will be forwarded to the General Manager - Salem Operations for his approval.

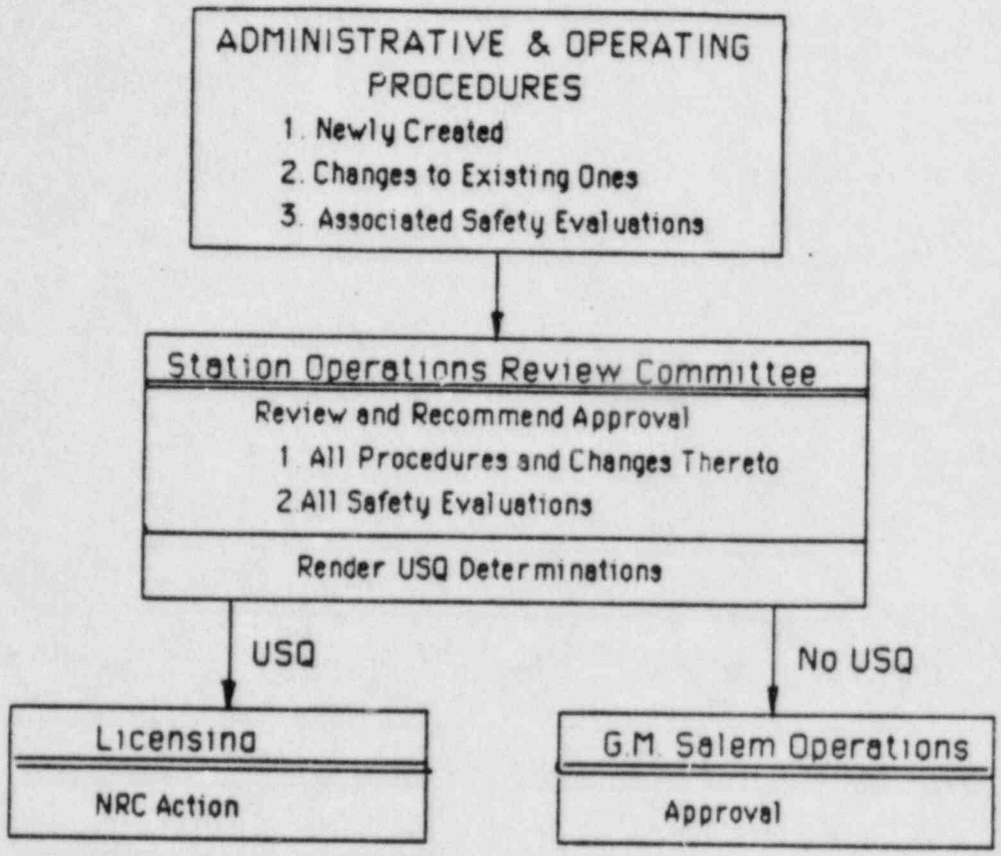


Figure 1. Present Document Flow to SORC, Procedure Related

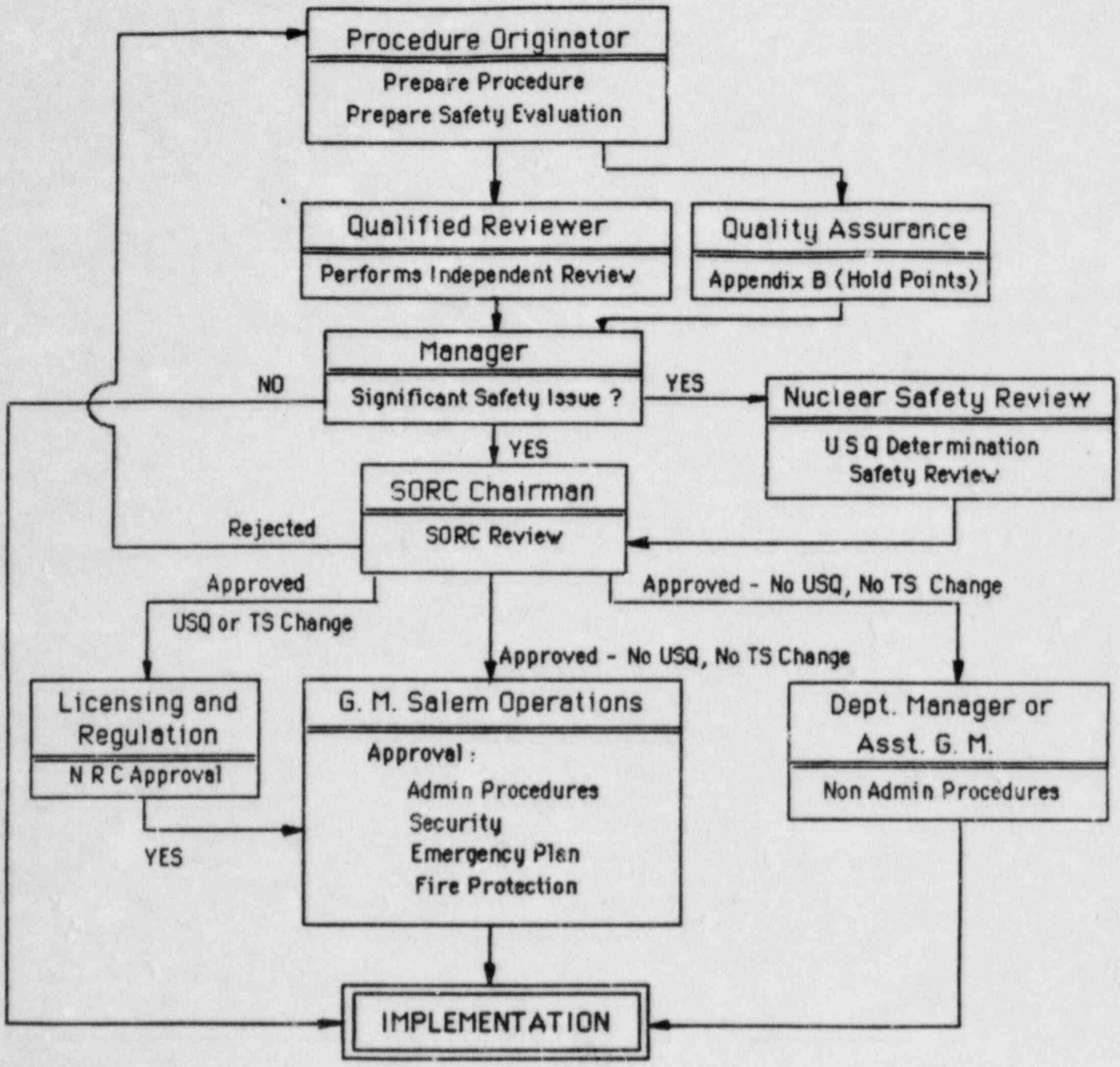


Figure 2. Recommended Document Flow, Procedure Related

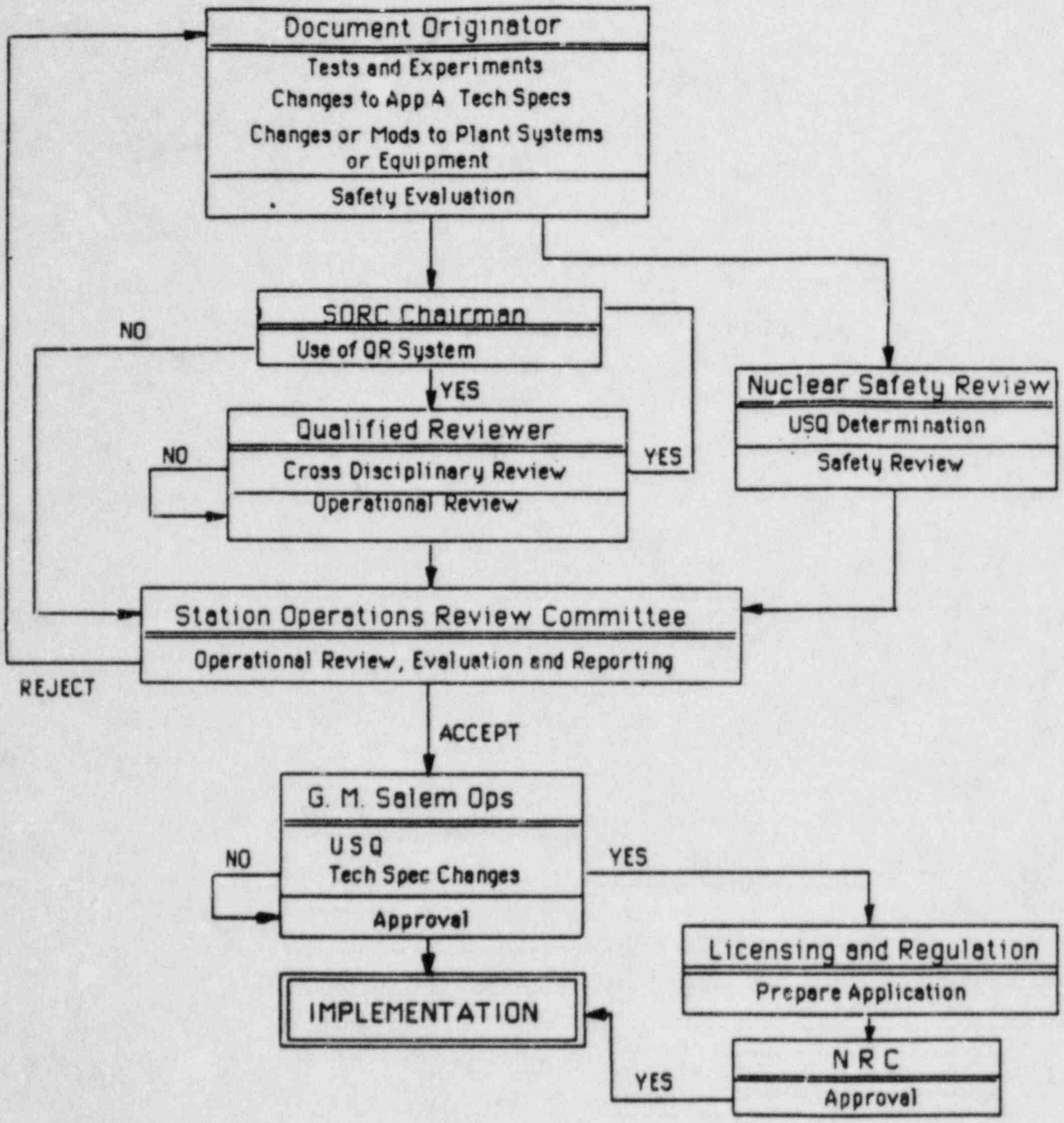


Figure 3. Recommended Document Flow, Non-Procedure Related.