PSE&G Public Service Electric and Gas Company

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

Robert L. Mittl General Manager
Nuclear Assurance and Regulation

August 24, 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.

In addition, enclosed for your review and approval (see Attachment 4) are the resolutions to the Draft SER open items listed in Attachment 3, and per your request in the July 30, 1984 meeting with the Geosciences Branch, Attachment 5 contains a copy of our comments (telecopied to

B001
1/1 Limited Dist

D. Wagner on August 10, 1984), on the Lawrence Livermore National Laboratory draft report entitled "Site Spectra for the Hope Creek Site." 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,

Attachments/Enclosure

C D. H. Wagner
USNRC Licensing Project Manager

W. H. Bateman
USNRC Senior Resident Inspector

FM05 1/2

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 Hope Creek Generating Station Draft Safety Evaluation Report open item responses and comments on the Lawrence Livermore National Laboratory draft report entitled "Site Spectra for the Hope Creek Site."

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 24th day of August 1984.

David K. Burd

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

GJ02

DATE: 8/24/84

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	8/20/84 (Rev. 1)
5b	2.4.5	Wave impact and runup on service water intake structure	Complete	8/20/84 (Rev. 1)
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	8/20/84 (Rev. 1)
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

M P84 80/12 1-gs

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
95	3.9.3.2	Fatigue evaluation on SRV piping and LOCA downcomers	Complete	6/15/84
96	3.9.3.3	IE Information Notice 83-80	Complete	8/20/84 (Rev. 1)
97	3.9.3.3	Buckling criteria used for component supports	Complete	6/29/84
98	3.9.3.3	Design of bolts	Complete	6/15/84
99a	3.9.5	Stress categories and limits for core support structures	Complete	6/15/84
99b	3.9.5	Stress categories and limits for core support structures	Complete	6/15/84
100a	3.9.6	10CFR50.55a paragraph (g)	Complete	6/29/84
100b	3.9.6	10CFR50.55a paragraph (g)	Complete	8/20/84
101	3.9.6	PSI and ISI programs for pumps and valves	Complete	8/20/84
102	3.9.6	Leak testing of pressure isolation valves	Complete	6/29/84
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	7/27/84
110b	4.6	Functional design of reactivity control systems	Complete	7/27/84
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	7/27/84
112b	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	7/27/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
112c	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	7/27/84
112d	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	7/27/84
112e	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	7/27/84
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	7/18/84
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	6/29/84
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

M P84 80/12 12- gs

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	8/15/84 (Rev. 1)
140b	9.1.2	Spent fuel pool storage	Complete	8/15/84 (Rev. 1)
140c	9.1.2	Spent fuel pool storage	Complete	8/15/84 (Rev. 1)
140d	9.1.2	Spent fuel pool storage	Complete	8/15/84 (Rev. 1)
141a	9.1.3	Spent fuel cooling and cleanup system	Complete	8/1/84
141b	9.1.3	Spent fuel cooling and cleanup system	Complete	8/1/84
141c	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/1/84

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	8/1/84
141e	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/1/84
141f	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/1/84
141g	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/1/84
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	Open	
143b	9.1.5	Overhead heavy load handling	Open	
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	8/20/84
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	7/27/84
151b	9.4.1	Control structure ventilation system	Complete	7/27/84
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	8/1/84 (Rev 1)
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	7/18/84
167	12.3.4.2	Portable continuous air monitors	Complete	7/18/84
168	12.5.2	Equipment, training, and procedures for inplant iodine instrumentation	Complete	6/29/84
169	12.5.3	Guidance of Division B Regulatory Guides	Complete	7/18/84
170	13.5.2	Procedures generation package submittal	Complete	6/29/84
171	13.5.2	TMI Item I.C.1	Complete	6/29/84
172	13.5.2	PGP Commitment	Complete	6/29/84
173	13.5.2	Procedures covering abnormal releases of radioactivity	Complete	6/29/84

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL 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	8/1/84
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	8/1/84
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	8/1/84
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	8/20/84 (Rev. 1)
242	8.3.2.1	Compliance with position 1 of Regulatory Guide 1.128	Complete	8/1/84
243	8.3.3.1.3	Protection or qualification of Class 1E equipment from the effects of fire suppression systems	Complete	8/1/84
244	8.3.3.3.1	Analysis and test to demonstrate adequacy of less than specified separation	Complete	8/1/84
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	8/1/84
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	8/1/84
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	8/20/84
257	8.3.2.5	Justification for a 0 to 13 second load cycle	Complete	8/1/84
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	8/1/84
261	8.3.3.3.6	Automatic transfer of loads and interconnection between redundant divisions	Complete	8/1/84
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	8/20/84
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: 8/24/84

DRAFT SER SECTIONS AND DATES PROVIDED

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

Notes:

1. Open items provided in letter dated July 24, 1984 (Schwencer to Mittl)

2. Open items provided in June 6, 1984 meeting

3. Open items provided in April 17-18, 1984 meeting

4. Open items provided in May 2. 1984 meting

5. Draft SER Section provided in letter dated August 7, 1984 (Schwencer to Mittl)

CT:db

MP 84 95/03 01

DATE: 8/24/84

ATTACHMENT 3

OPEN ITEM	DSER SECTION	SUBJECT
176d	14.2	Initial plant test program
199	7.4.2.1	IE Bulletin 79-27-Loss of non-class IE instrumentation and control power system bus during operation
207	7.7.2.1	HELBs and consequential control system failures
208	7.7.2.2	Multiple control system failures

ATTACHMENT 4

Rev 2

DSER Open Item 176d (Section 14.2)

INITIAL PLANT TEST PROGRAM

The response does not address the concerns of I&E Information Notice Number 83-17, March 31, 1983. The concern is that if a time delay prevents fuel from being supplied to the diesel generator following a shutdown signal, the air supply may be exhausted before the fuel supply is reinstated. The response to this item should be modified to address these concerns.

RESPONSE

The response to Q640.10 has been revised to provide the information requested above and to address item 2 per discussions with the NRC.

QUESTION 640.10 (SECTION 14.2.12)

Modify your FSAR submittal to address the following concerns regarding emergency diesel generator testing:

1. FSAR Subsections 1.8.1.108 and 14.2.13.5 state that Regulatory Guide 1.108 (Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants) is not applicable to Hope Creek. It is the staff's position that this guide is applicable to your facility. Therefore, either delete or provide justification for this statement.

2. FSAR Subsections 1.8.1.108 and 14.2.13.5 take exception to Position C.2.a(5) of Regulatory Guide 1.108. These subsections state that testing of the sequencing controls after the 24 hour test run does not subject the controls to more severe conditions than testing accomplished under other circumstances. Provide technical justification for your position or perform this test in accordance with this guide. Additionally, modify FSAR Subsection 14.2.12.1.30 (KJ-Emergency Diesel Generators) to perform a restart simulating loss of ac directly after the 24-hour run in accordance with your statement in the aforementioned FSAR subsections.

3. Modify FSAR Subsections 14.2.12.1.30 (KJ-Emergency Diesel Generators), 14.2.12.3.30 (Loss of Turbine-Generator and Offsite Power), or other test abstracts as appropriate, to:

   a. Perform the simultaneous, redundant diesel starts specified in Position C.2.b of Regulatory Guide 1.108.

   b. Include prerequisite testing to ensure the satisfactory operability of all check valves in the flow path of cooling water for the diesel generators from the intake to the discharge (see I&E Bulletin No. 83-03: Check Valve Failures in Raw Water Cooling Systems of Diesel Generators).

   c. Provide assurance that any time delays in the diesel generator's restart circuitry will not cause the supply of compressed air used to initially rotate the engine to be consumed in the presence of a safety injection signal (see I&E Information Notice Number 83-17, March 31, 1983).

RESPONSE FSAR Sections 1.8.1.108 and 14.2.13.5 will be revised as requested above

~~NRC Regulatory Guide 1.108 is not applicable to HCGS. This is justified as stated in Implementation Section D of Regulatory~~

~~Guide 1.108 which provides that the guide is to be used in the evaluation of submittals for construction permits.~~

Section 14.2.12.1.30.c.6 has been revised to state that a restart simulating loss of ac power will be performed following the 24-hour run.

Upon restart, a sequencing check will not be performed since the 24-hour run test has no effect on the sequencing circuit. The sequencing circuits are located in the emergency load sequencer panels remote from the diesel generator room. The circuits will not have left their standby state since the 24-hour run is accomplished without a loss-of-power or loss-of-coolant accident condition, and is synchronized to the grid. However, the sequencing will be checked during the ECCS integrated initiation during loss-of-offsite power test described in Section 14.2.12.1.47. However immediately following the 24 hour run, the simulated loss of ac power will be followed by an immediate manual loading to design load condition. Simultaneous redundant diesel starts are accomplished as described in Section 14.2.12.1.47.c.2.

Section 14.2.12.1.30 has been revised to include prerequisite component testing on all diesel generator cooling water check valves.

The diesel generator control design has a time delay relay which holds the fuel racks closed to allow the unit to come to a complete stop. However, in the event of an emergency start signal due to ECCS requirements during the count down of the time delay relay, this relay is functionally overridden and the fuel racks open to allow the diesel to continue to run or restart through the normal starting air sequence described in Section 9.5.6.

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:
Regulatory Guide 1.108

immediately

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~~ will 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.

due to the emergency load sequencer panels are locate remote from the diesel generator room.

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

followed by an immediate manual loading to design load conditions.

HCGS complies with Regulatory Guide 1.109.

For further discussion, see Chapter 15.

1.8.1.110 Conformance to Regulatory Guide 1.110, Revision 0, March 1976: Cost-Benefit Analysis for Radwaste Systems For Light-Water-Cooled Nuclear Power Reactors

HCGS complies with Regulatory Guide 1.110.

4. Demonstrate that manual and automatic operation of the diesel generators is satisfactory, and that they start automatically upon simulated loss of ac voltage and attain the required frequency and voltage.

5. Verify that proper response and operation of the design basis accident loading sequence to design basis load requirements, and verify that voltage and frequency are maintained within specified limits. This test may be accomplished in the preoperational test described in Section 14.2.12.1.47, ECCS integrated initiation during loss of offsite power.

6. Demonstrate full load carrying capability of the diesel generators for a period of not less than 24 hours, of which 22 hours are not less than the equivalent DBA full load for the respective bus, and 2 hours are at the 2-hour 110% load rating. Following the 24 hour run, an automatic restart due to simulated loss of ac power, ~~will be~~ and manual ~~performed~~. loading to design-load requirements will immediately be performed

7. Verify that the diesel generators can be synchronized to an offsite power source while maintaining the Class 1E loads.

8. Verify that the standby diesel generator (SDG) system is capable of transferring the Class 1E load from the generator to the offsite power source, and of isolating the generator from the bus and returning it to standby status.

9. Verify that the rate of fuel consumption at design basis load for each diesel generator is such that the requirements for 7-day storage inventory are met.

10. During surveillance testing, verify the capability of the diesel generators to respond to an emergency signal and supply power to the Class 1E bus, while monitoring time, frequency, and voltage.

6. In response to DBA simulation the loading sequence is as specified in Table 8.3-1 and voltage and frequency are maintained within the values specified in Section 8.3.1.1.3.

7. and loading
   The diesel generator shall operate for 24 hours under load as specified in Section 8.3.1.1.3. The automatic restart after the 24 hour run shows that the diesel generator attains rated speed and voltage as specified in Section 8.3.1.1.3.

8. The diesel generator will synchronize to offsite power while maintaining the Class 1E loads, transfer the load to offsite power, and resume standby status following an operational mode.

9. The diesel generator fuel oil storage tanks have a demonstrated capacity as specified in Section 9.5.4.2.1, based upon engine fuel consumption.

10. With the diesel generator operating in the surveillance mode, it will respond to an emergency signal to supply power to Class 1E bus loads.

11. Load rejection does not result in exceeding speeds or voltages which cause diesel generator tripping or mechanical damage.

12. The standby diesels start the number of times specified in Section 9.5.6.3 without the air receiver recharging compressor available.

14.2.12.1.31 KP-Main Steam Isolation Valve Sealing

a. Objective

The test objective is to verify flow paths, controls operation, interlocks, and alarms associated with the main steam isolation valve (MSIV) leakage control system.

calibration completed prior to performing the preoperational test.

14.2.13.5 <u>SRP II.e, Regulatory Guide 1.108, Revision 1, August 1977: Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants</u>

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

a. 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 ~~.~~ due to the emergency load sequencer panels are located remote from the diesel generator room. A restart simulating loss of ac ~~can~~ will be performed immediately ~~directly~~ after the 24-hour run ~~.~~ followed by an immediate manual loading to design load conditions. Sequencing, however, will be performed when the loads can be lined up for operation and all four diesels are available.

Rev 1

DSER Open Item No. 199 (DSER Section 7.4.2.1)

IE BULLETIN 79-27 - LOSS OF NON-CLASS 1E INSTRUMENTATION AND CONTROL POWER SYSTEM BUS DURING OPERATION.

We will require the applicant to document the results of the analysis, providing recommendation of hardware or procedural changes as appropriate in response to IEB 79-27. This is presently scheduled for submittal during the fourth quarter of 1984.

RESPONSE

The response to Question 421.42 has been revised to provide the information requested above. A copy of the cold shutdown/Power Bus Failure analysis report ~~is~~ dated august, 1984 is attached for your use systems used in attaining the cold shutdown condition. Identify busses that could affect the ability to achieve cold shutdown. Use plant operating procedures and procedures developed for certain power bus failures to ensure the identification of all critical power busses.

2. Identify the instrumentation and control devices connected to each identified power bus. Evaluate the effects of a loss of power to each load, including the limiting effects on the ability to achieve cold shutdown.

3. Create bus trees denoting the bus hierarchy and the cascading bus configuration of all busses that power instrumentation and controls the operator would manipulate in going to cold shutdown.

4. Determine the annunciators and alarms that would alert the operator to a failure of any of the identified busses.

5. Determine the effects of any single power bus loss on the ability to continue in each particular shutdown path being used at the time the bus loss occurs. Include the cascading effects of any bus loss, and consider alternate indications and controls powered by unaffected busses that may aid the operator in the event of a bus loss. Identify alternative shutdown paths available and existing procedures for restoration of the affected bus.

6. Document the results of the analysis, providing recommendations of hardware or procedural changes as appropriate.

The programs described in the responses to this question and to Questions 421.51 and 421.52 will be conducted as a combined effort that will be completed by December, 1984.

The Analysis ~~of the report information~~ shows ed there is no situation where a single bus power failure would prevent plant personnel from achieving a safe shutdown condition. The results establish ed that no single bus supplies power to all existing shutdown paths. The assignment of the instrument loads identified in this analysis is such that the loss of one bus would not prevent the minimum safety function from being performed.

The failure of each of the buses ~~identified in Table 2 of Appendix A~~ are annunciated and are displayed by the computer in the control room, thereby giving the operator the knowledge of which power bus is lost. The analysis showed ~~review~~ that control room personnel will have knowledge of individual bus and/or circuit failures, and that the operator has alternate ive instruments and shutdown paths available to achieve a cold shutdown condition.

REFERENCE

1. "Cold Shutdown/Power Bus Failure Analysis Report," Hope Creek Generating Station, Public Service Electric and Gas Company, August 1984

QUESTION 421.42 (SECTION 7.5)

If reactor controls and vital instruments derive power form common electrical distribution systems, the failure of such electrical distribution systems may result in an event requiring operator action concurrent with failure of important instrumentation upon which these operator actions should be based. IE Bulletin 79-27 addresses several concerns related to the above subject. You are requested to provide information and a discussion based on each IE Bulletin 79-27 concern. Also, you are to:

1) Confirm that all a.c. and d.c. instrument buses that could affect the ability to achieve a cold shutdown condition were reviewed. Identify these buses.

2). Confirm that all instrumentation and controls required by emergency shutdown procedures were considered in review. Identify these instruments and controls at the system level of detail.

3) Confirm that clear, simple unambiguous annunciation of loss of power is provided in the control room for each bus addressed in item 1 above. Identify any exceptions.

4) Confirm that the effect of loss of power to each load on each bus identified in item 1 above including ability to reach cold shutdown, was considered in the review.

5) Confirm that the re-review of IE Circular No. 79-02 which is required by Action Item 3 of Bulletin 79-27 was extended to include both Class 1E and non-Class 1E inverter supplied instrument or control buses. Identify these buses or confirm that they are included in the listing required by Item 1 above.

RESPONSE

An analysis (see Reference 1.) was ~~will be~~ conducted based on the Limerick Generating Station (LGS-1) approach ~~General Electric methodology~~ for answering the concerns raised in IE Bulletin 79-27. This methodology has been reviewed and approved by the NRC via a report written for the LGS-1 ~~WNP-2~~ project. ~~The methodology provides for a systematic and comprehensive analysis to ensure that, in the event of a single power bus failure, sufficient control room indicators, instruments, and controls exist to achieve a cold shutdown.~~

~~An outline of the methodology follows:~~

~~1. Review the Class 1E and non-Class 1E busses including inverters supplying power to instrumentation and controls in~~

Rev 1

DSER Open Item No. 208 (DSER Section 7.7.2.2)

MULTIPLE CONTROL SYSTEM FAILURES

The applicant is required to submit the analysis and its conclusions concerning multiple control system failures to the NRC for staff review. This is scheduled for submittal during the fourth quarter of 1984.

RESPONSE

The response to FSAR Question 421.51 has been revised to provided the information requested above. A copy of the following reports are attached to this response for your use:

1) Common Power/Control Systems Failures Evaluation Report; Dated: August, 1984

2) Common Sensor Failure Evaluation Report; Dated: August, 1984.

QUESTION 421.51 (SECTION 7.7)

The transient and accident analyses included in the FSAR are intended to demonstrate the adequacy of safety systems in mitigating anticipated operational occurrences and accidents.

Based on the conservative assumptions made in defining these "design bases" events and the detailed review of the analyses by the staff, it is likely that they adequately bound the consequences of single control system failures. To provide assurance that the design basis event analysis for Hope Creek adequately bounds other more fundamental credible failures, provide the following:

(1) Identify those control systems whose failure or malfunction could seriously impact plant safety.

(2) Indicate which, if any, of the control systems identified in (1) receive power from common power sources. The power sources considered should include all power sources whose failure or malfunction could lead to failure or malfunction of more than one control system and should extend to the effects of cascading power losses due to the failure of higher level distribution panels and load centers.

(3) Indicate which, if any, of the control system identified in (1) receive input signals from common sensors. The sensors considered should include common taps, hydraulic headers and impulse lines feeding pressure, temperature, level or other signals to two or more control systems.

(4) Provide justification that any malfunctions of the control systems identified in (2) and (3) resulting from failures or malfunctions of the applicable common power source or sensor including hydraulic components are bounded by the analyses in Chapter 15 and would not require action or response beyond the capability of operators or safety systems.

RESPONSE

Two ~~An~~ analyses ~~will be~~ (see References 1 and 2) were conducted based on the General Electric methodology for answering NRC concerns for common power source failures and common sensor or sensing line failures. This methodology, which received NRC concurrance via reports for the Grand Gulf, Shoreham, and WNP-2 projects, was ~~will be~~ used for the Hope Creek project. ~~The methodology is systematic and comprehensive and examines control systems interactions to establish the limiting-case events. The consequences of single power-source or sensing-line failures will be evaluated with respect to control-grade systems and will ensure the limiting-case events are bounded by the events analyzed in Chapter 15.~~

A. Common Power Source Failure

An outline of the methodology for the common power source failure analysis follows:

1. Identify all nonsafety-grade control systems that have the potential of affecting the critical reactor parameters of water level, pressure, or power.

2. Review these control systems at the component level; identifying the effects of the loss of power to each system component and the sebsequent interactions with other components and systems.

3. Generate bus trees denoting the bus hierarchy and cascading configuration of all power busses that supply components of control systems under study.

4. Perform a combined effects analysis. Evaluate the failure of each power bus (load center, motor control center, etc.) starting with the lowest-level source common to multiple control systems and working up each bus tree to the highest common power level. At each level examine the effects of the single bus failure and the consequences of cascading bus failures on all control systems' components.

5. Postulate the limiting transient events as a result of the combined effects analysis and compare these events to those analyzed in Chapter 15.

6. Perform any additional transient calculations or analyses necessary to ensure the postulated limiting events are bounded by those analyzed in Chapter 15.

7. Document the results of the analyses of common power source failure, providing recommendations as appropriate.

B. Common Sensor or Sensing Line Failure

An outline of the methodology for the common sensor or sensing line failure analysis follows:

1. Identify the nonsafety-grade control systems that have the potential of affecting the critical reactor parameters of water level, pressure, or power.

2. Identify all instrument sensing lines and sensors utilized by two or more of these control systems.

3. Analyze the effects of failure of a common sensor of a complete plug or a guillotine break in each of these common instrument lines. Examine the effects of erroneous signals on each instrument and on each function (scrams, trips, permissive signals, etc.) that could be actuated or rendered inoperative.

4. Examine the interactive effects among all systems affected by the common sensing line or sensor failure and the consequential combined effects on the critical reactor parameters.

5. Compare the consequences of these postulated events with those analyzed in Chapter 15 to ensure the consequences of the postulated events are bounded by the results of the Chapter 15 events and to ensure the postulated events would not require actions or responses beyond the capabilities of the operators or the safety systems. Perform any additional transient calculations or analyses necessary to ensure the postulated limiting events are bounded by those analyzed in Chapter 15.

6. Document the results of the analyses of common sensing line or sensor failures and provide recommendations as appropriate.

The programs described in the responses to this question and to questions 421.42 and 421.52 will be conducted as a combined effort that will be completed by December, 1984.

The conclusion of th~~is~~ese analyses evaluation ~~is~~ was that the limits of minimum critical power ratio (MCPR), peak vessel and main steamline pressures, and peak fuel cladding temperature for the expected operational occurrence category of events would not be exceeded as a result of common power source ~~failures.~~ or common sensor failures for control systems. Although transient category events ~~have been~~ were postulated as a result of th~~is~~ese stud~~y~~ies, the net effects ~~have been~~ were positively determined to be less severe than those of the original, conservative, Chapter 15 events. It should be noted that th~~is~~ese stud~~y~~ies use~~s~~d the event-consequence logic of the Chapter 15 analys~~i~~es, but start~~s~~ed the logic chain from a specific source (e.g., a single bus or sensor failure) rather than a system condition (e.g., feedwater runout). By approaching the stud~~y~~ies in this manner, a great deal of confidence can be placed in the stud~~y~~ies conclusions. [redacted] The soundness of the total plant design ~~is~~ was demonstrated by its being tolerant of these interactions.

REFERENCES

1. "Common Power/Control Systems Failures Evaluation," Hope Creek Generating Station, Public Service Electric and Gas Company, August 1984.

2. "Common Sensor Failure Evaluation Report," Hope Creek Generating Station, Public Service Electric and Gas Company, August 1984

ATTACHMENT 5

AUG 10 '84 0269366

COMMENTS ON LAWRENCE LIVERMORE NATIONAL LABORATORY REPORT

Reference: "Draft Report: Site Spectra for the Hope Creek Site", prepared by the Lawrence Livermore National Laboratory (LLNL) for the NRC, July 16, 1984.

Comments

1. Earthquake magnitudes 6.2, 6.3 and 6.5 were used by the LLNL in preparing site specific spectra. These magnitudes are unnecessarily high.

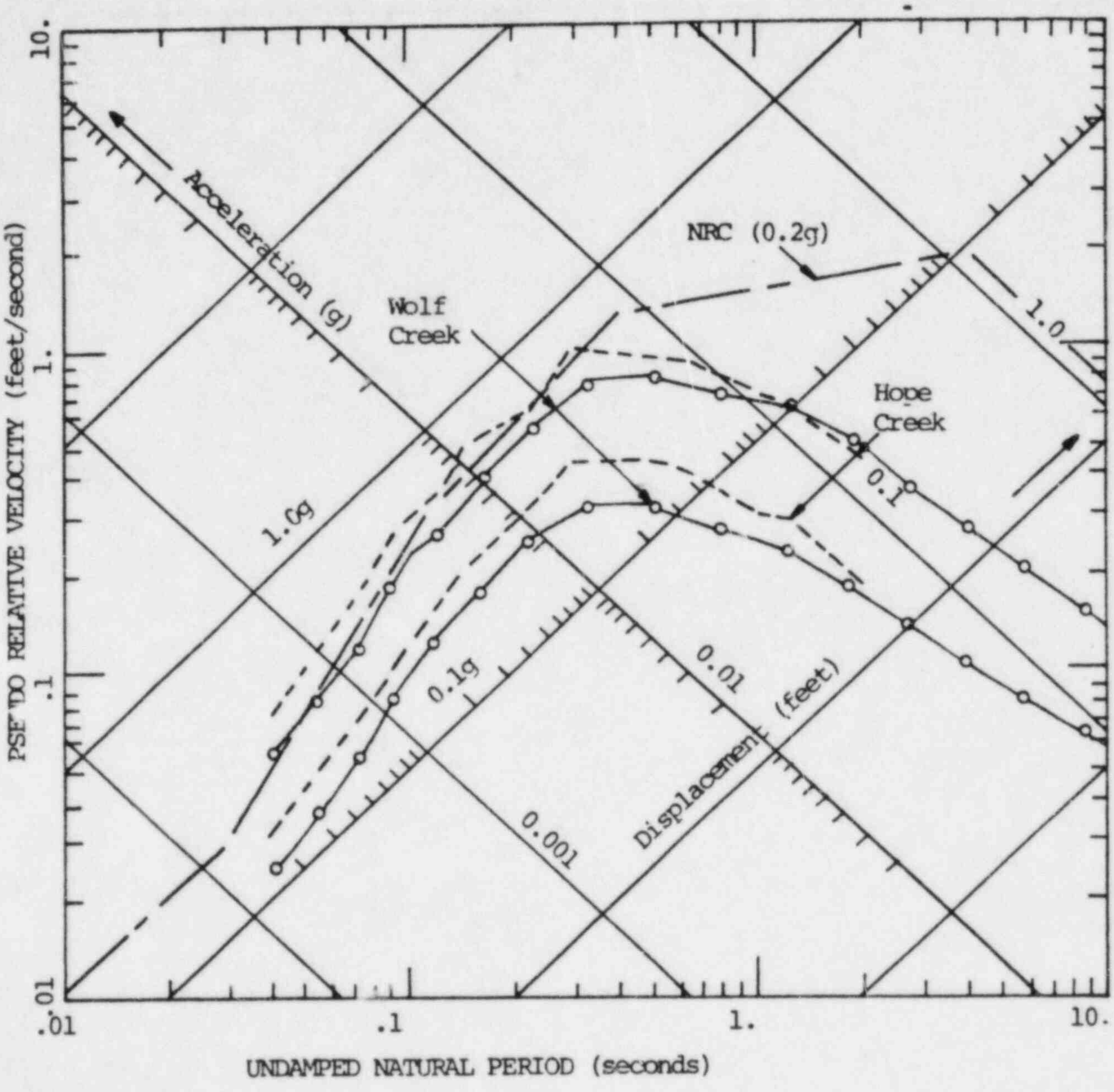
2. The distance weighting scheme has a data set with a magnitude bias. Mean magnitudes for the distance subsets are as follows:

0 - 10	5.07
10 - 15	5.04
15 - 20	5.43

   This difference is probably severe enough to invalidate the weighting process.

3. Bernreuter prepared a report for NRC on the Wolf Creek Site (March 5, 1982). In this report he performed a study similar to Hope Creek. Using 30 soil site records (mean Magnitude, ML = 5.2±0.2, Table 6 of that report) he produced median and 84th percentile spectra. These are plotted together with the median and 84th percentile base case spectra from the 27 sites used for Hope Creek in the July 16, 1984 report to the NRC on the attached figure (mean Magnitude ML = 5.2±0.3). The spectra on the attached figure differ from each other significantly. The choice of an almost entirely new data set for the recent report and rejection of records used for the Wolf Creek site study should be explained. This is especially important because of the implications being made for the higher recent spectral values.

4. Hope Creek design spectra is applied at elevation 40 ft which is approximately 60 feet below grade. The site specific spectra proposed by the LLNL correspond to a control point at or near the ground surface. See Appendix A for reconciliation.



COMPARISON OF SITE SPECIFIC SPECTRA FOR WOLF & HOPE CREEK SITES
Acceleration = 0.2g, Damping = 5.0%

APPENDIX A

EVALUATION OF HOPE CREEK SITE SPECTRA

Introduction

A meeting was held between the NRC (Geosciences Branch) and PSE&G on July 30, 1984 to discuss the "Draft Report: Site Spectra for the Hope Creek Site" prepared by Lawrence Livermore National Laboratory (LLNL) under contract from the NRC. The LLNL developed the preliminary site specific spectra to help assess the conservatism of the Hope Creek seismic input spectra which are based on Regulatory Guide 1.60 spectra. The purpose of this discussion is to evaluate the impact of the LLNL study on the Hope Creek design spectra.

Lawrence Livermore National Laboratory Proposed Hope Creek Site Spectra

The following three criteria were used by the LLNL in selecting earthquake records for possible inclusion in the development of Hope Creek Spectra:

a. The magnitude of earthquake corresponds to the ranges of m_{bLg} = 5.25 $\pm$0.5 and m_{bLg} = 5.75 $\pm$0.5

b. The distance between the epicenter and the recording station is approximately less than 20 km, and

c. The recording stations are located at a "deep soil" site (soil depth greater than 200 ft).

The recommended preliminary median and one sigma spectra in both the horizontal and vertical directions (5% damping) for magnitude 5.25 are shown in Figures 5 and 12C of Reference 1. A direct comparison of the LLNL spectra (m_{bLg} = 5.25) with the Hope Creek design spectra without considering the design earthquake control point elevation reveals the following:

- Horizontal Spectra: LLNL spectral velocities are higher than those of the Hope Creek design spectra at periods below 0.35 seconds.

- Vertical Spectra: Hope Creek design spectra generally envelop the LLNL spectra except at periods below 0.06 seconds.

A comparison of the spectra of m_{bLg} = 5.75 at about 30km distance with the Hope Creek design spectra reveals no exceedance.

AUG 10 '84 0269366

Evaluation of Hope Creek Design Spectra

The Hope Creek site specific spectra prepared by LLNL were developed from an ensemble of earthquake records obtained at or near the ground surface of sites whose top soil layers have an average shear wave velocity of approximately 1,500 ft/sec. and an average compressional wave velocity of 4,800 ft/sec.

As stated in the Hope Creek FSAR Section 3.7.1, the control point of the design earthquake input at the Hope Creek site is not at the ground surface but at the level of the foundation structure in the free field which is approximately 60.0 ft below ground surface. Therefore, a direct comparison of the LLNL spectra with the Hope Creek spectra is not appropriate. To present a more direct comparison, equivalent spectra are developed from the Hope Creek design spectra in accordance with the following procedures:

A. Horizontal Earthquake

i. A free-field soil column as shown in Figure 1 is used in deconvolution analysis. The top of the soil column is truncated at elevation 65.0 ft since the soil below this elevation (Kirkwood and Vincentown formation) has shear wave velocity of approximately 1,850 ft/sec which are comparable to the site condition of the LLNL Study. FLUSH computer code is used for the soil column analysis (Reference 2).

ii. The Hope Creek design spectra are applied at elevation 40.0 ft where the power block foundation is located and the response spectra (5% damping) at the top of the soil column are generated. Figure 2A shows the comparison plots between the regenerated spectra at the top of the soil column and the input spectra delineating amplification effect. Figure 2B shows comparison between the LLNL spectra (m_{bLg} = 5.25) and the regenerated design spectra.

iii. In all cases, the equivalent regenerated Hope Creek design spectra at elevation 65.0 ft envelop the proposed LLNL spectra. Therefore, it is concluded that the Hope Creek design basis spectra are adequate.

B. Vertical Earthquake

i. The soil column as shown in Figure 1 is used for the evaluation of the Hope Creek vertical site spectra. Due to the presence of water table near the ground surface, the compression wave velocity of soil is appropriately adjusted for the vertical analysis.

ii. The Hope Creek design response spectra are applied at elevation 40.0 ft of the soil column and the corresponding response spectra are regenerated at elevation 65.0 ft.

iii. Figure 3A shows the comparison plots between the velocity response spectra at the top of the soil column and the input spectra delineating amplification effect. Figure 3B provides the comparison between LLNL spectra and the equivalent regenerated Hope Creek spectra. The Hope Creek spectra envelop the LLNL spectra. Therefore, it is concluded that the Hope Creek input spectra are adequate.

Conclusion

Based on the above, it is concluded that the Hope Creek seismic input criteria meet or exceed the criteria proposed by LLNL.

References:

1. "Draft Report: Site Spectra for the Hope Creek Site", prepared by the Lawrence Livermore National Laboratory for the Nuclear Regulatory Commission, July 16, 1984.

2. Lysmer, J., Udaka, T., Tsai, C.F., and Seed, H.B., "FLUSH - A Computer Program for Approximate 3-D Analysis of Soil-Structure Interaction Problems", Report No. EERC 75-30, College of Engineering, University of California, Berkeley, California, November, 1975.

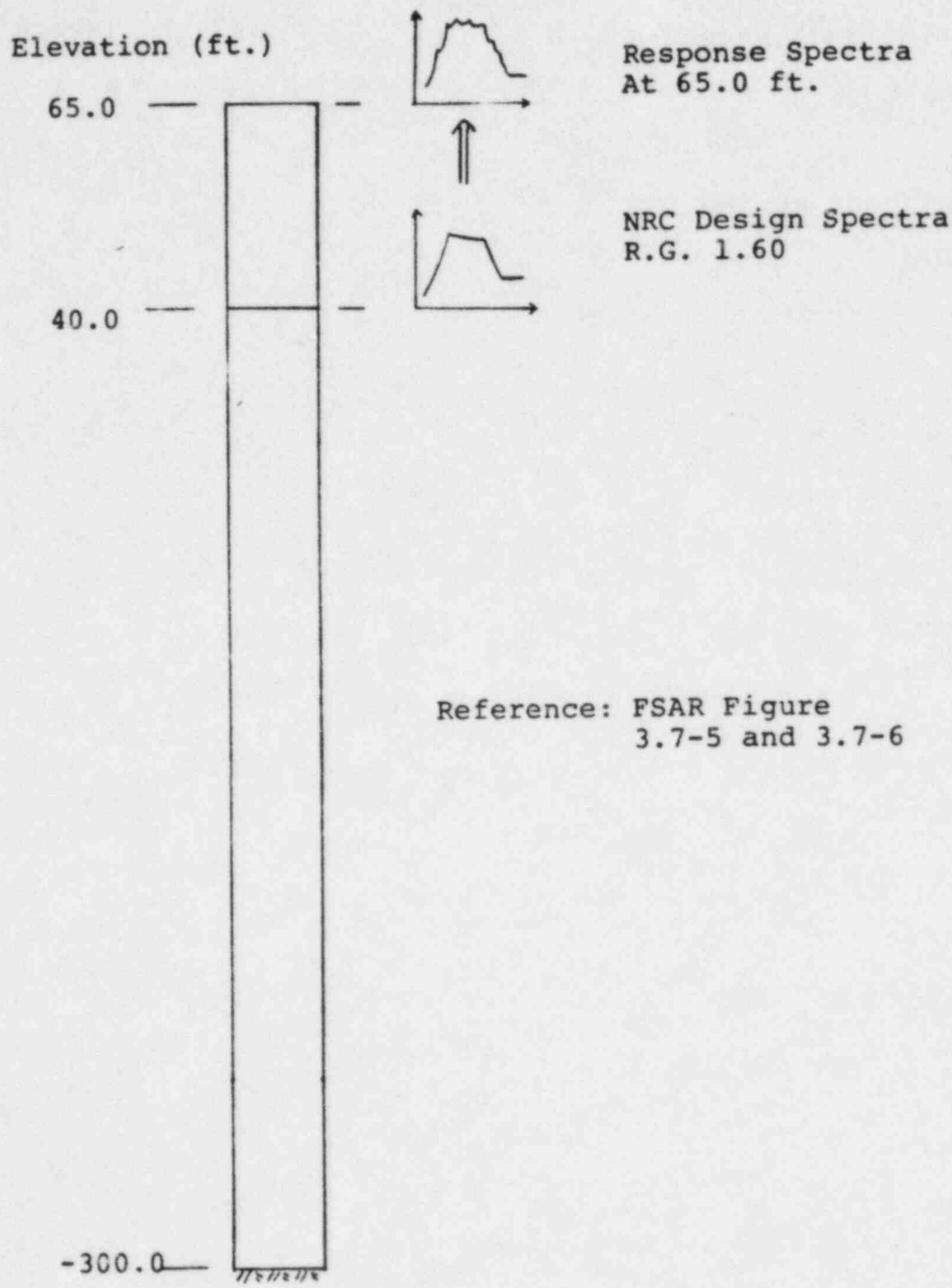


Figure 1 Soil Column for Evaluation of Hope Creek Site Spectra

AUG 10 '84 0269366

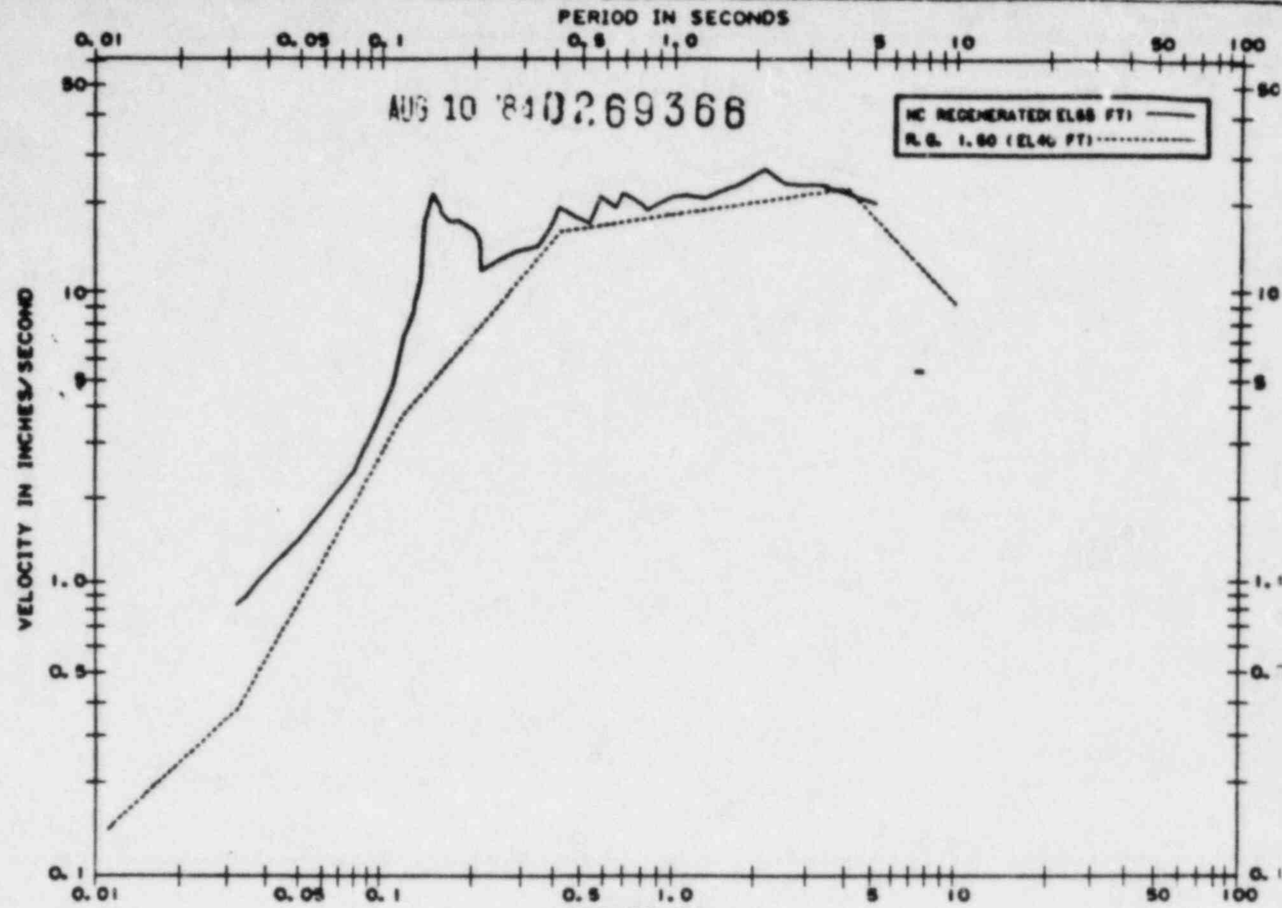


FIGURE 2A

RESPONSE SPECTRA COMPARISON
AT TOP OF FREE FIELD
HORIZONTAL, SSE, 5% DAMPING

HOPE CREEK UNIT 1
CAS.L1OG.REV 0
08/07/84

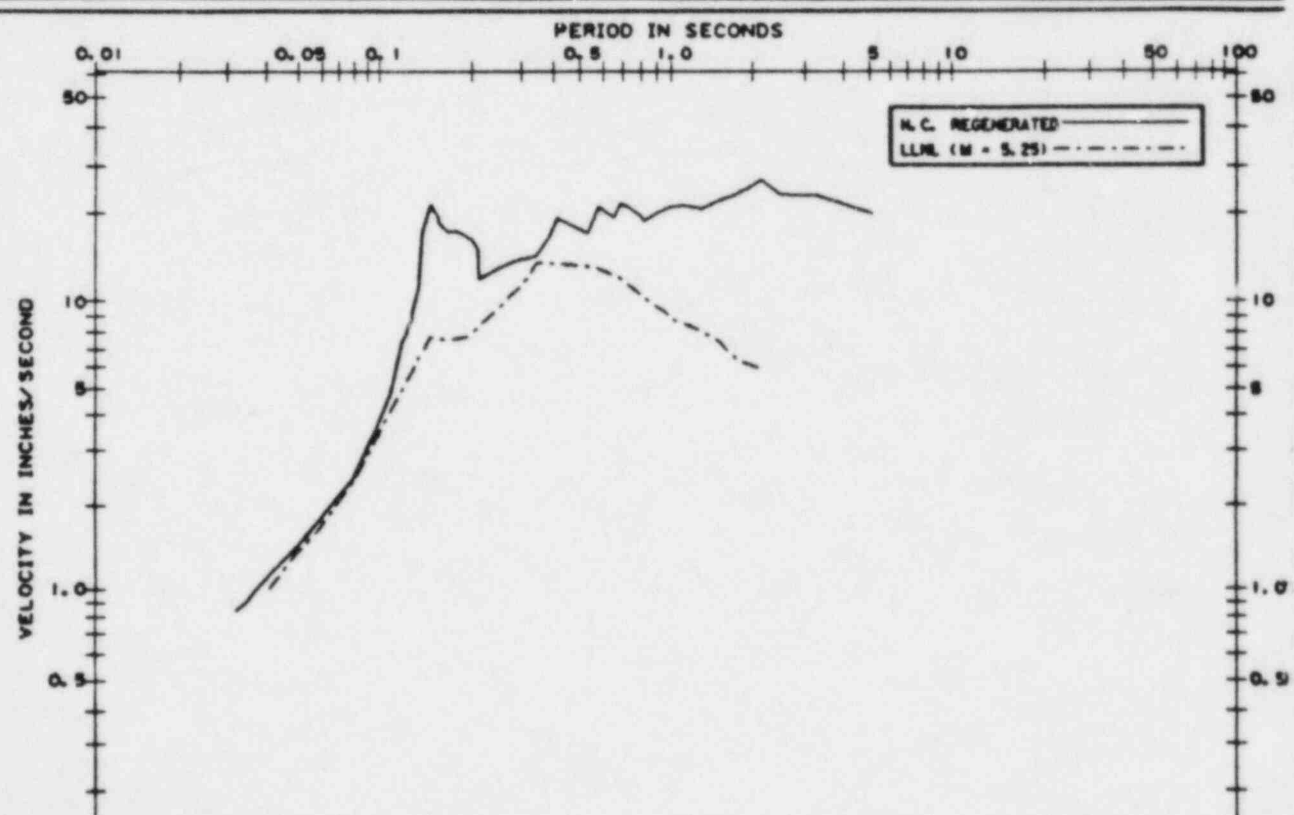


FIGURE 2B

RESPONSE SPECTRA COMPARISON
AT TOP OF FREE FIELD
HORIZONTAL, SSE, 5% DAMPING

HOPE CREEK UNIT 1
CAS.L1OG.REV 0
08/07/84

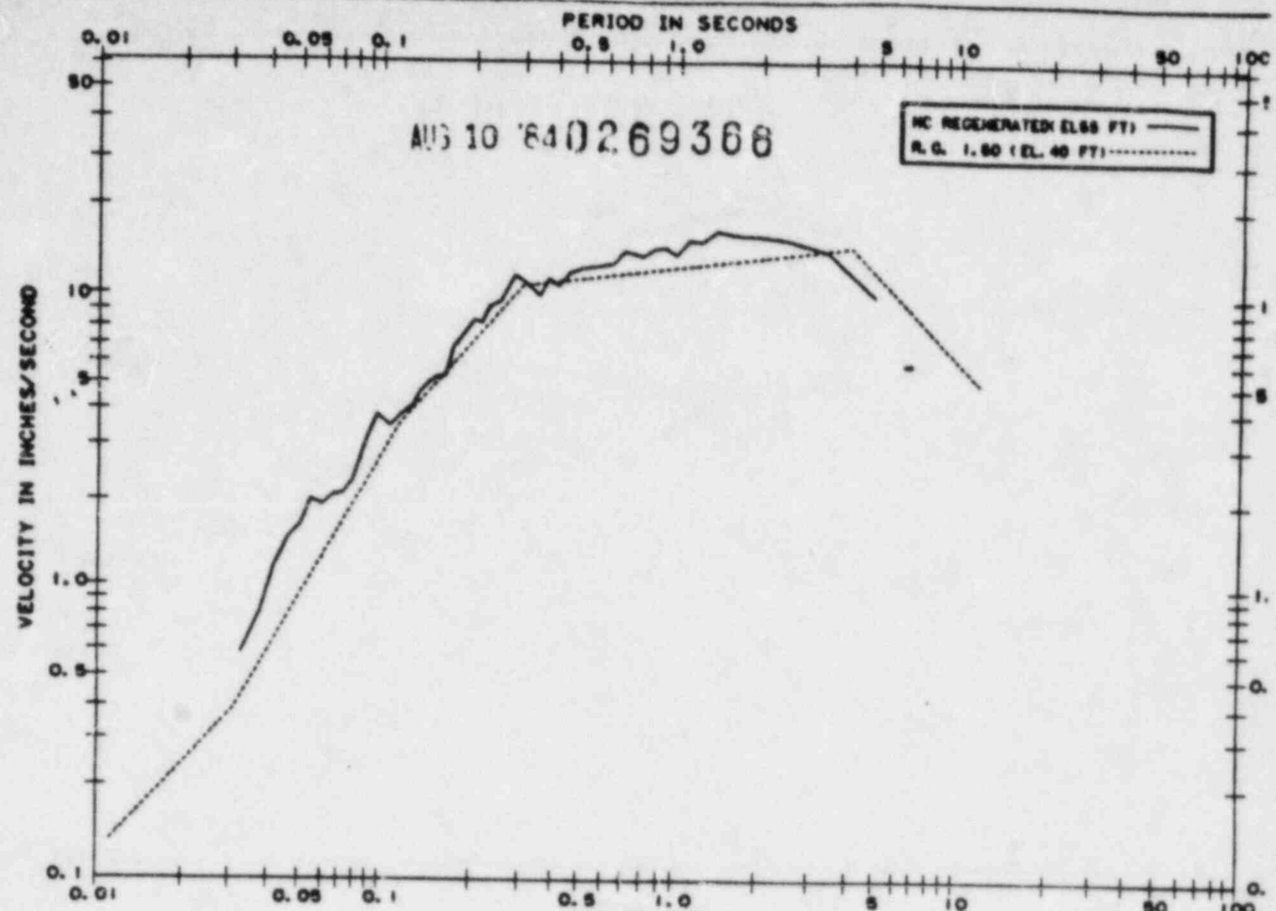


FIGURE 3A
RESPONSE SPECTRA COMPARISON
AT TOP OF FREE FIELD
VERTICAL, SSE, 5% DAMPING

HOPE CREEK UNIT 1

FIGURE 3B
RESPONSE SPECTRA COMPARISON
AT TOP OF FREE FIELD
VERTICAL, SSE, 5% DAMPING

HOPE CREEK UNIT 1

Rev 1

DSER Open Item No. 207 **(DSER Section 7.7.2.1)**

HELBs AND CONSEQUENTIAL CONTROL SYSTEM FAILURES

The applicant is required to submit the analysis and its conclusions concerning HELBs and consequential control system failures to the NRC for staff review. This is scheduleu for submittal during the fourth quarter of 1984.

RESPONSE

The response to FSAR Question 421.52 has been revised to provide the information requested above. The following report is attached to this response for your use:

1) High Energy Line Break/Control Systems Failure Analysis; Dated: August, 1984

QUESTION 421.52 (SECTION 7.7)

If control systems are exposed to the environmental resulting from the rupture of reactor coolant lines, steam lines, or feedwater lines, the control systems may malfunction in a manner which would cause consequences to be more severe than assumed in safety analyses. I&E Information Notice 79-22 discusses certain non-safety grade control equipment, which if subjected to the adverse environment of a high energy line break, could impact the safety analyses and the adequacy of the protection functions performed by the safety-related systems.

The staff is concerned that a similar potential may exist at light water facilities now under construction. You are, therefore, requested to perform a review per the I&E Information Notice 79-22 concern to determine what, if any, design changes or operator actions would be necessary to assure that high energy line breaks will not cause control system failures to complicate the event beyond the FSAR analyses. Provide the results of your review including all identified problems and the manner in which you have resolved them.

The specific "scenarios" discussed in the above referenced Information Notice are to be considered as examples of the kinds of interactions which might occur. Your review should consider analogous interactions as relevant to the BWR design.

RESPONSE

An analysis (see Reference 1) was ~~will be~~ conducted based on the General Electric methodology for answering the concerns raised in IE Information Notice 79-22. The NRC has concurred with this methodology via its review prepared for the Shoreham and Grand Gulf projects. The methodology assures a systematic, comprehensive analysis of high-energy line breaks and the consequential control systems failures. An outline of this methodology follows:

1. Identify all nonsafety control-grade systems and components within these systems whose failure could affect the critical reactor parameters of water level, pressure, and power.

2. Establish assumptions and criteria for determining high-energy lines and pipe break locations and for evaluating the consequences of pipe breaks. Pipe whip, jet impingement, and environmental parameters such as high temperature, high pressure, and high humidity will be considered in the analysis.

3. Identify from appropriate plant drawings those plant locations in which high-energy lines with postulated break

locations coexist with nonsafety components of control-grade systems.

4. Conduct a plant walkdown to verify the locations of control system components and to determine their proximity to high-energy line break locations.

5. Examine, one at a time, high-energy line breaks and establish the worst-case combined effects of each break and the consequential control-system failures.

6. Ensure that the consequences of those pipe-break events are bounded by those of the events analyzed in Chapter 15.

7. Choose two or more of the worst-case scenarios and postulate for each a worst-case additional failure in a safety-related, mitigating system. Ensure that the consequences of these new events do not fall outside the bounds of the capabilities of safety systems or the consequences of the events analyzed in Chapter 15.

8. Document the results of the analysis of the interactions between high-energy line breaks and control systems and recommend actions to be taken as appropriate.

The programs described in the responses to this question and to Questions 421.42 and 421.51 will be conducted as a combined effort that will be completed by December 1984.

The Analysis ~~Summary (Appendix A)~~ described each of the postulated HELB events and their limiting effects on the reactor parameters. In most cases, the effects of the postulated HELB/control systems failures events ~~are~~ were shown to be less severe than the Unacceptable Results for Incidents of Moderate Frequency - Anticipated Operational Transients presented in ~~FSAR~~ Chapter 15. In all cases, the effects of the postulated events ~~are~~ were shown to be bounded by the Unacceptable Results for Limiting Faults - Design Basis (Postulated) Accidents presented in ~~FSAR~~ Chapter 15. It ~~is~~ was concluded that safe reactor shutdown is assured for all ~~events~~ postulated events ~~herein~~, and the consequences of these postulated events would not result in any significant risk to the health and safety of the public.

REFERENCE

1. "High Energy Line Break/Control Systems Failures Analysis," Hope Creek Generating Station, Public Service Electric and Gas, August 1984