PSEG Public Service Electric and Gas Company

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Robert L. Mittl General Manager
Nuclear Assurance and Regulation

August 20, 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, and FSAR question responses listed in Attachment 3. 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 L Mittl

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Drawings To: D. Wagner

Attachments

C D. H. Wagner
USNRC Licensing Project Manager

W. H. Bateman
USNRC Senior Resident Inspector

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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 Hope Creek Generating Station Draft Safety Evaluation Report open item responses and FSAR Question responses.

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 20th day of August 1984.

David K. Burd

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

GJ02

DATE: 8/20/84

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)	Open	
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

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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
23	2.5.4	Clarification of FSAR Tables 2.5.13 and 2.5.14	Complete	6/1/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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.
48	3.8.6	Rocking time histories	Complete	8/20/84 (Rev 1)
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)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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
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
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
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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)
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
86	3.9.1	Computer code validation	complete	8/20/8
87	3.9.1	Information on transients	complete	8/20/8
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
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

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
103b6	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	complete	8/20/84
103cl	3.10	Seismic and dynamic qualification of m~chanical 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	
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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
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

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ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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
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
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)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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	
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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	8/13/84 Rev.
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
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.

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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)
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	8/1/84
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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	8/1/84
208	7.7.2.2	Multiple control system failures	Complete	8/1/84
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
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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
225	8.2.3.1	Testability of automatic transfer of power from the normal to preferred power source	Complete	8/1/34
226	8.2.2.5	Grid stability	Complete	8/13/84 Rev.
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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
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

M P84 80/12 21- gs

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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
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	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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	
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: 8/20/84

DRAFT SER SECTIONS AND DATES PROVIDED

SECTION	DATE	SECTION	DATE
3.1			
3.2.1		11.4.1	See Note 1
3.2.2		11.4.2	See Note 1
5.1		11.5.1	See Note 1
5.2.1		11.5.2	See Note 1
6.5.1	See Note 1	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 Note 1
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 Note 3		
10.4.3	See Note 3		
10.4.4	See Note 3		
11.1.1	See Note 1		
11.1.2	See Note 1		
11.2.1	See Note 1		
11.2.2	See Note 1		
11.3.1	See Note 1		
11.3.2	See Note 1		

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

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DATE: 8/20/84

ATTACHMENT 3

ITEM No.	SECTION No.	Subject
5a,b&d	2.4.5	Wave impact and runup on service water intake structure.
6a,b	2.4.10	stability of erosion protection structures
26	2.5.4.4	Intake structure sliding stability
37	3.6.2	Feedwater isolation check valve operability
38	3.6.2	Design of pipe rupture restraints
42	3.8.2	Steel containment ultimate capacity analysis
45	3.8.4	ACI 349 deviations for Category I structures
46	3.8.5	ACI 349 deviations for foundations
48	3.8.6	Rocking time histories
49	3.8.6	Gross concrete section
50	3.8.6	Vertical floor flexibility response spectra
51	3.8.6	Comparison of Bechtel independent verification results with the design-basis results
53	3.8.6	Design of seismic Category I tanks
56	3.8.6	Drywell stick model development
59	3.8.6	Overturning moment of reactor building foundation mat
60	3.8.6	BSAP element size limitations
64	3.8.6	SSI analysis 12 Hz cutoff frequency

77	3.8.6	Factor of safety against overturning of intake structure
79	3.8.6	Post-modification seismic loads for the torus
81	3.8.6	Seismic displacement of torus
82	3.8.6	Review of seismic Category I tank design
84	3.8.6	Ultimate capacity of containment
86	3.9.1	Computer code vaiidation
87	3.9.1	Information on transients
96	3.9.3.3	IE information Notice 83-80
100b	3.9.6	10CFR 50.55-a Paragraph (g)
101	3.9.6	PSI and ISI programs for pumps and values
102	3.9.6	Leak testing of pressure isolation values.
103	3.10	Seismic and dynamic qualification of mechanical and electrical equipment
105	4.2	Plant-specific mechanical fracturing analysis
106	4.2	Applicability of seismic and LOCA loading evaluation
109	4.4.7	TMI-2 ITem II.F.2
117	5.3.4	Compliance with NB-2332 of Winter 1972 Addenda of the ASME Code

118	5.3.4	Lead facters and neutron fluence for surveillance capsules
120	6.2	TMI item II.E.4.2
124	6.2.1.5.1	RPV shield annulus analysis
126	6.2.1.6	Redundant position indicators for vacuum breakers
127	6.2.1.6	Operability testing of vacuum breakers
133	6.2.4.1	Containment purge system
135	6.3.3	LPCS and LPCI injection valve interlocks
136	6.3.5	Plant-specific LOCA
137	6.4	Control room habitability
148	9.3.2	Post Accident Sampling System
157	9.5.1.4.e	Cable tray protection
176g	14.2	Initial plant test program
177	15.1.1	Partial feedwater heating
198	7.3.2.6	TMI Item II.K.3.18 - ADS actuation
241	8.3.1.10	Load acceptance test after prolonged no load operation of the DG
256	8.3.2.3	Automatic trip of loads to maintain sufficient battery capacity
262	11.4.2.d	Solid waste control program
264	6.2.5	Sources of oxygen
268	6.8.1.2	Air filtration unit drains
269	5.2.2	Code cases N-242 and N-242-1
270	5.22	Code case N-252

Attachment 3 (cont'd)

QUESTION NUMBER	FSAR SECTION
210.56	3.9.6
251.7	5.3.1, 5.3.2, 5.3.3
281.9	10.4.6
281.11	9.1.3
430.75	9.5.3
471.4	12.3.2

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8/20/84

ATTACHMENT 4

JUL 31 '84 0268742

Rev 1

DSER Open Item No. 5 a,b,&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 apporaching 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

Rev 1

Item No. 5 a, b + d (Cont'd)

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

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.

As a result of discussions with the NRC staff, the response to Question 410.69 has been revised and summary calculations for wave overtopping are attached.

K50/5-12

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.

REPLACE WITH INSERT A

Referring to Figure 410.69-1, the flood protection measures for the service water system are provided as follows:

a. The areas between column lines B and A, above elevation 93 feet and between column lines AK and A, below elevation 93 feet are designed to be watertight up to elevation 121 feet for the north and east exterior walls and up to elevation 128.5 feet for the south and west exterior walls. 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, worst case water levels will not exceed the top of the wall at the air intake screens at elevation 128.5 feet. Therefore, flood water will not enter into the dry area of the intake structure. On the north and east sides of the intake structure, the maximum water surface will be only slightly higher than the still water level (elevation 113.8 feet) during the PMH. According to Table 2.4-6, the maximum wave flood wave elevation for the north and east sides of the intake structure is 26.3 MSL (elevation 115.3 feet) due to a postulated multiple dam break. Therefore, flood protection of the north and east exterior walls to elevation 121 feet is adequate. As described in Section 3.4.1.1, the exterior walls and the foundation mat in these areas are protected with a waterproofing system. Furthermore, the exterior walls are designed to accommodate the flood loads including wave action effects. All Seismic Category I components, except for the traveling screens, are located in these watertight area.

b. The traveling screens are located in the "wet" area between column lines B and C. The exterior walls in this area are designed to accommodate flood loads including wave action effects. The motors for the

~~traveling screens are protected against the flood water level.~~

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

REPLACE WITH "INSERT A"

INSERT A

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.

Based on a question presented by the NRC Hydrology Branch in the August 1, 1984 meeting, the east exterior wall of the intake structure is being further evaluated to confirm that the maximum wave elevation is below the minimum height of flood protection. Results of this evaluation will be available by September 1, 1984.

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.

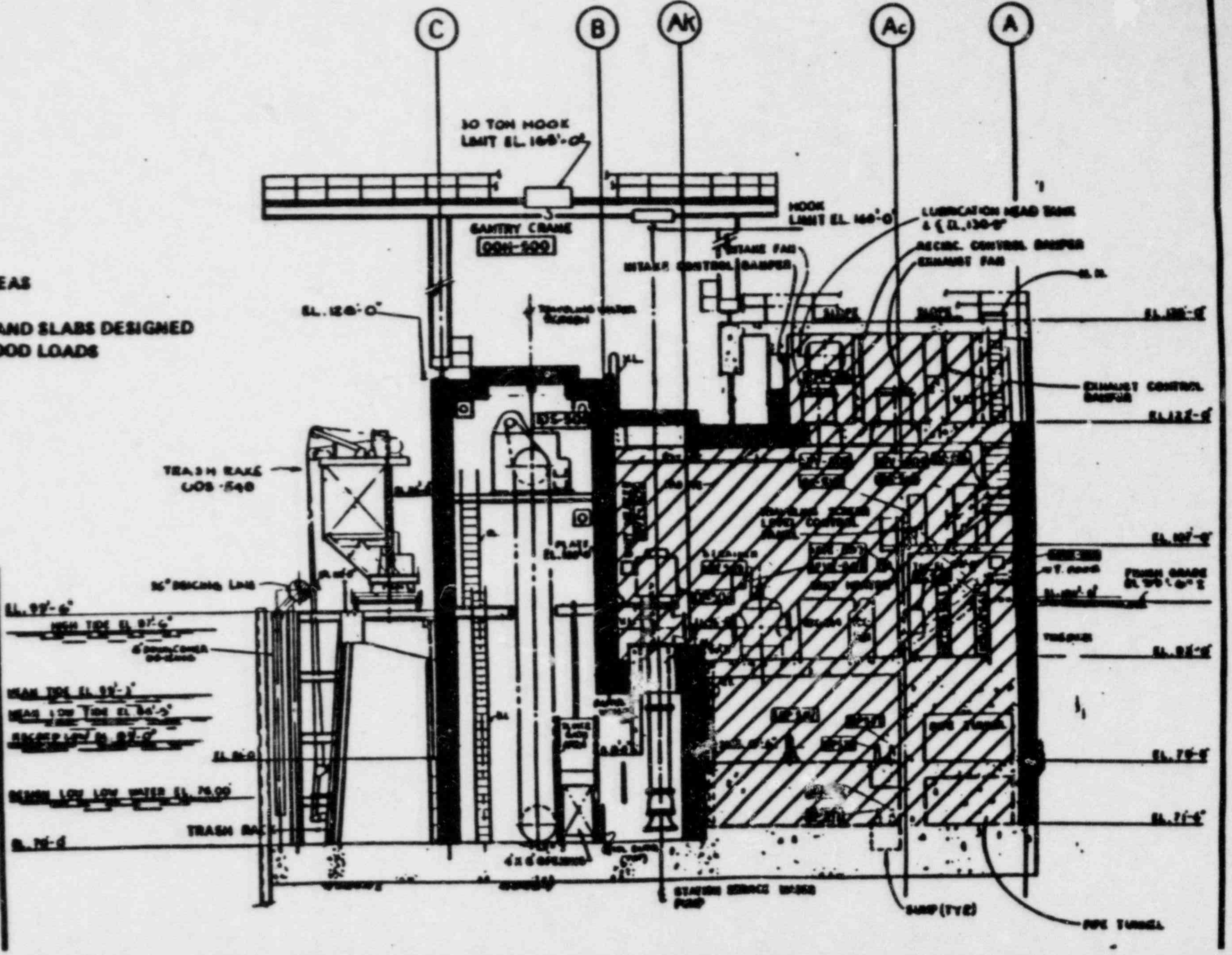
d. The traveling water-screens, located in the "wet" area between column lines B and C have electric motors which are fully protected against the flood water level.

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.

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 410.69-1

August 6, 1984

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

I. Wave Calculations

- o 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).

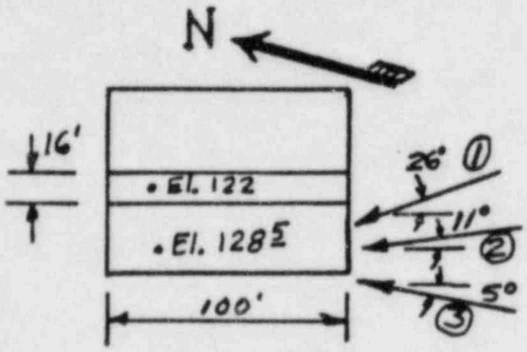
II. Overtopping Calculations

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

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

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

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



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

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

- o 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.

- o 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 bend (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.

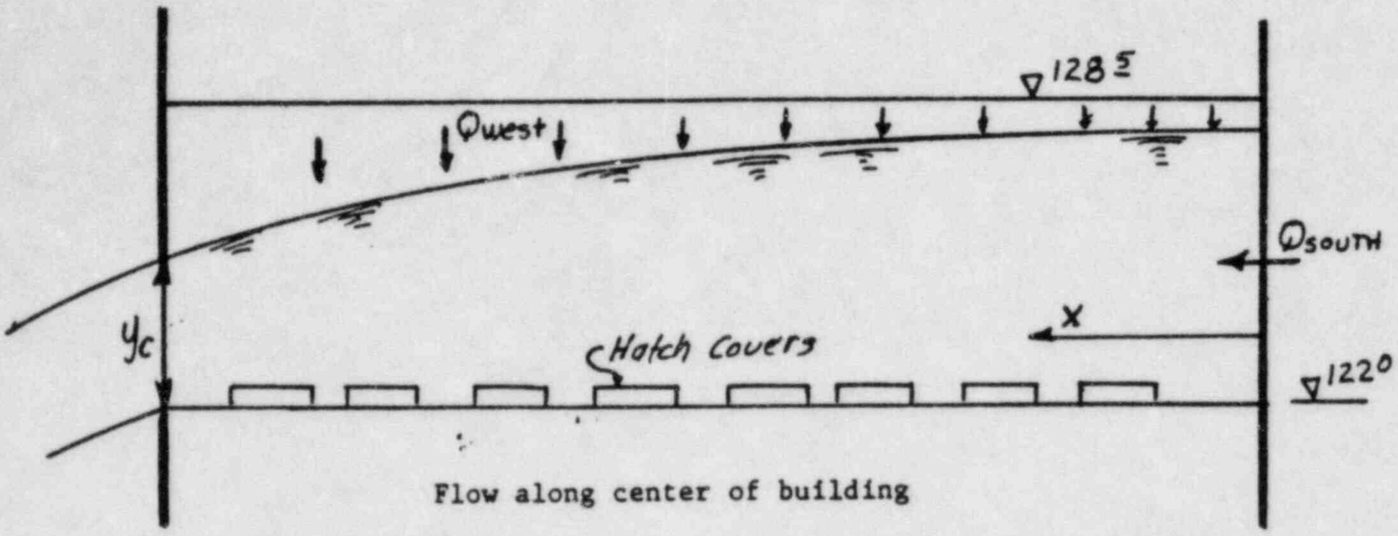
- Losses incurred by turbulence and impact of the jet entering water ponded on top of elevation 122.0 were neglected.

- Headloss through the screens was neglected.

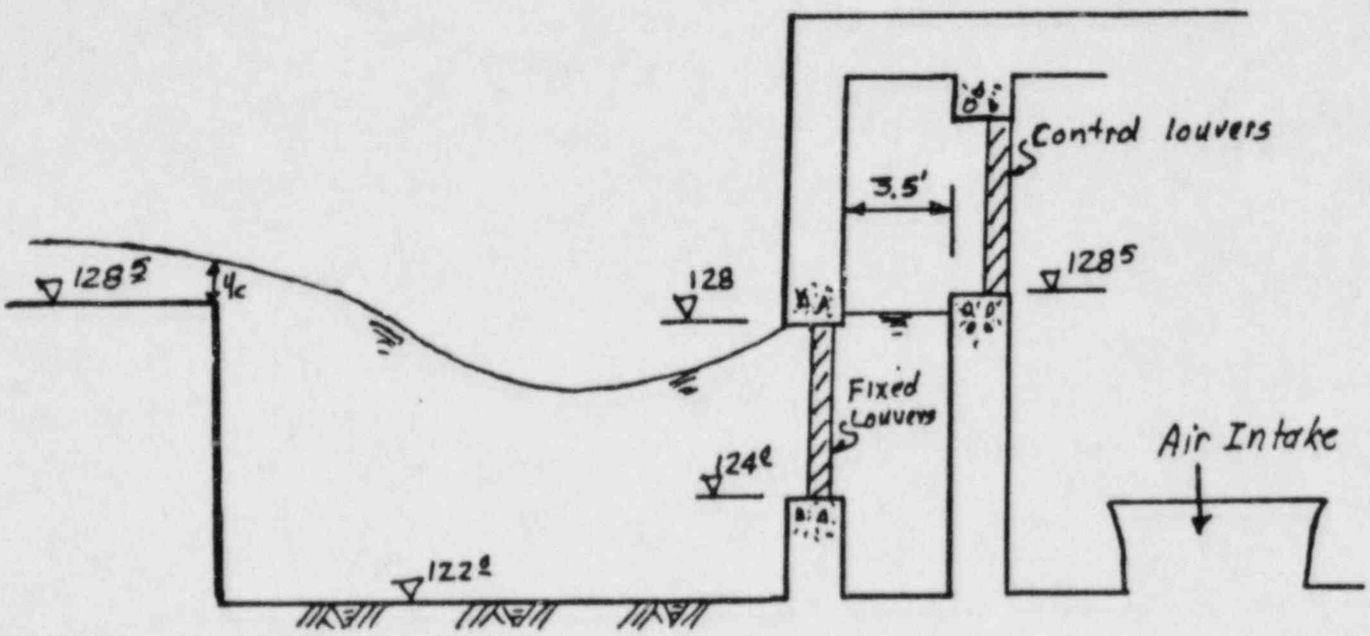
- The maximum elevation achieved was calculated to be 126.3 or well below the 128.5 elevation at which water could flow into the building.

- 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. Wiegel, R. L., "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.

<u>QUESTION 240.12</u> (Section 2.4.10)

Provide detailed discussion on the ability of each water-tight door and hatch to withstand the combined static and dynamic loadings (still water level and wave impact loadings) associated with the design basis flood event (PMH).

<u>RESPONSE</u>

ATTACHED

RESPONSE

All exterior doors in ~~Seismic~~ Category I structures are designed to withstand the combined static and dynamic loading associated with the design basis flood event (PMH). The design of each door is based on the more severe of two conditions:

a. The static loading due to the maximum stillwater level ~~height~~ plus the associated wave generated water rise ~~height~~ at the location of the door (Tables 2.4-10 ~~(Isam)~~ and 2.4-10a). The combined design elevation for doors on the south wall of the reactor building and a portion of the south wall of the auxiliary building radwaste and service area is elevation 127 feet (PSE&G datum). Other exterior doors are designed for the static load associated with a combined design water level to elevation 121 feet (PSE&G datum).

b. The static loading due to the combined still water ~~level~~ and wave ~~height~~ plus the dynamic effects associated with a breaking wave, utilizing the fetch that creates the most adverse loading on the door. The breaking wave height used is 78% of the still water depth based on ESAR Reference 2.4-24. Minikin's method is used to calculate the wave forces. The dynamic loads used take into account the wave celerity, period and the angle of incidence. Generally, the fetch selected for any particular door is such that the resultant force due to wave impact is near the top of the door. Parametric studies were made with various still water heights to establish the most severe loading for this condition.

Due to the angle of incidence and shadowing effects of the Salem ~~Power Plant~~ Generating Stat, loading condition "a" controls the design for all exterior doors except those along the south wall of the power block. For doors on the south wall of the power block the dynamic wave effects associated with Fetchs 4 or 5 control the design.

Sections 2.4.10 and 3.4.1.1 have been revised to respond to the question.

a. Static and dynamic flood resistance incorporated into the exterior wall and base slab designs of the Seismic Category I structures

b. Double water stop application in all seismic joints to above maximum water level

c. Full waterstop application at construction joints to above maximum wave run-up level

d. ~~Pressure tight doors~~ Static and dynamic flood resistance incorporated into the exterior door designs of the Seismic Category I structures.

e. Waterproof penetrations

f. Flood alarm system to warn of rising water at limit levels in watertight compartments

g. Floor drainage systems to sumps

h. Level-actuated sump pumps discharging to holding tanks.

Shore protection is required in the vicinity of the service water intake structure to assure that no blockage to the water intake will occur and that erosion will not impede the operation of the service water pipes. A study by Dames & Moore, completed in June 1977 (Reference 2.4-15A), addressed shore protection and makes recommendations to assure the safe operation of the plant. In summary, shore protection will extend 100 feet north and south of the intake structure, which will be stable under the design seismic event and design flood (PMH). Sheet piling and surface protection are being provided for the service water piping. ~~The shore protection will consist of sheet pile cellular cofferdams.~~

2.4.11 LOW FLOW CONSIDERATIONS

2.4.11.1 Low Flow in Streams

The HCGS is located within the tidally-affected portion of the Delaware Estuary System. Historical extremes in water-level occurred as a result of wind-related tide level variations, and not as a result of fluvial discharges; see Section 2.4.1.1. The are designed to prevent roof spillage or heavy rain from seeping inside the building.

~~Doors and~~ penetrations in exterior walls and slabs of the SSWS intake structure are protected against water inflow up to elevation 121 feet for the north and east exterior walls and up to elevation 128.5 feet for other exterior walls and slabs. As described in Section 2.4.2, the SSWS intake structure may be subjected to hurricane produced waves which could overtop the roof of the western portion of the structure at elevation 128 feet. However, worst case water levels will not exceed the top of the wall at the air intake screens at elevation 128.5 feet. Therefore, flood water will not enter into the dry area of the SSWS intake structure.

INSERT 1

In the event the capacity of the yard drainage system is exceeded as a result of an unusually severe rainstorm, the excess water accumulates in puddles in the vicinity of catch basins and runs off as the storm subsides. No significant barriers exist such as road crowns, dikes or mounds that could appreciably increase the ponding levels adjacent to Seismic Category I structures. Water does not enter any safety-related structure, since the structures are watertight up to elevations 121 feet or 127 feet. Therefore, no safety-related equipment is adversely affected as the result of severe rainstorms. Additional details on yard drainage and grading are provided in Figure 3.4-3.

In addition, buildings in the power block complex have a multi-ply waterproofing system between the leveling mat and the concrete topping below the foundation mat, as shown on Figure 3.4-2. The system extends upward to grade on outside walls.

The concrete topping mat and base mat subgrade exterior construction joints are also treated with an additional waterproofing system.

Vertical and horizontal construction joints are provided with waterstops to elevation 121 feet. Waterstops are provided to elevation 124 feet and 127 feet for a portion of the south exterior walls of the auxiliary building and reactor building, respectively.

Seismic joints have additional backup waterstops from within the base mat up to grade level.

The waterstop materials are selected and designed to resist possible deterioration due to potential environmental effects. The waterstop material is styrene-butadiene synthetic rubber.

INSERT #1 FROM PAGE 3.4-3

¶ All exterior doors in Seismic Category I structures are designed to withstand the static and dynamic effects from postulated floods and the associated wave action.

· The design of each door is based on the more severe of the following two conditions:

a. The static head ~~loading~~ due to the maximum still water level ~~height~~ plus the water rise from the wave effect ~~associated wave runup~~ at the location of the door (Tables 2.4-10). and 2.4-10a

b. The static head ~~loading~~ due to the combined still water ~~depth~~ ~~[illegible]~~ ~~[illegible]~~ and wave [illegible] ~~height~~ plus the dynamic effects associated with a breaking wave, utilizing the fetch that maximizes the total loading on the door.

DSER Open Item NO 6ab (DSER Section 2.4.10)

Flood Protection Requirements

As cited in Section 2.4.5, the applicant has not indicated whether the watertight doors are designed to withstand the combined loading effects of both static water level and the dynamic wave impact associated with hydrologic events that flood plant grade. The applicant has been requested to provide documentation that the doors and hatches can withstand the static and dynamic loadings associated with hurricane surge and flooding levels and associated waves up to and including the PMH. The applicant has also been requested to provide documentation to verify the structural stability of both the steel sheet pile caissons and the quarrystone revetment for hydrologic events up to and including the design basis PMH flood.

As indicated in Section 2.4.2.2, the applicant has been requested to provide detailed documentation to demonstrate that the ponding level on roofs of safety-related structures as a result of the PMP event do not exceed roof design loads and that ponding levels will not result in internal flooding through roof hatches.

RESPONSE

The requested information concerning the watertight doors has been provided in the response to FSAR Question 240.14. (See DSER Open item No. 5a,b+d)

An evaluation has been made of the cellular cofferdams adjacent to the Service Water Intake Structure for hydrologic events up to and including the design basis PMH flood. The resulting wave force horizontal loading of 14.2 kips per ft. and vertical loading of 15.5 kips per ft. is less than that imposed by postulated seismic events with loadings of 54.8 kips per ft. and 21.9 kips per ft., respectively. Thus, the loading due to wave-generated forces do not control the design of the cofferdams.

As discussed in the June 28, 1984 meeting with the NRC Hydrology Branch, shore protection is provided by cellular cofferdams without the addition of a quarrystone revetment. FSAR Section 2.4.10 and Figure 3.4-3 have been revised to indicate that the shore protection is provided by cellular cofferdams.

a. Static and dynamic flood resistance incorporated into the exterior wall and base slab designs of the Seismic Category I structures

b. Double water stop application in all seismic joints to above maximum water level

c. Full waterstop application at construction joints to above maximum wave run-up level

d. Pressure tight doors

e. Waterproof penetrations

f. Flood alarm system to warn of rising water at limit levels in watertight compartments

g. Floor drainage systems to sumps

h. Level-actuated sump pumps discharging to holding tanks.

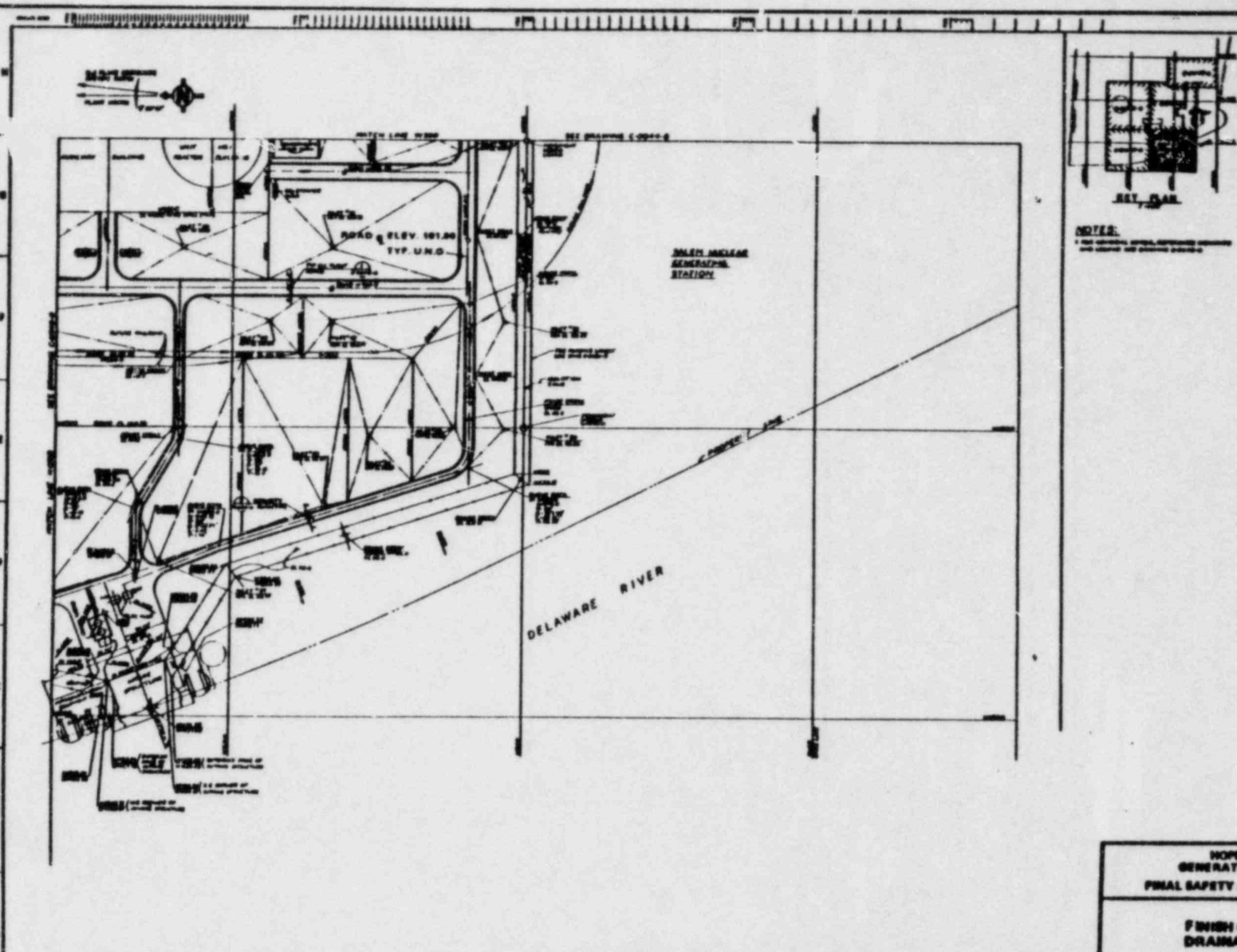
Shore protection is required in the vicinity of the service water intake structure to assure that no blockage to the water intake will occur and that erosion will not impede the operation of the service water pipes. A study by Dames & Moore, completed in June 1977 (Reference 2.4-15A), addressed shore protection and makes recommendations to assure the safe operation of the plant. In summary, shore protection will extend 100 feet north and south of the intake structure, which will be stable under the design seismic event and design flood (PMH). Sheet piling and surface protection are being provided for the service water piping.

Shore protection consists of sheet pile cellular cofferdams

2.4.11 LOW FLOW CONSIDERATIONS

2.4.11.1 Low Flow in Streams

The HCGS is located within the tidally-affected portion of the Delaware Estuary System. Historical extremes in water-level occurred as a result of wind-related tide level variations, and not as a result of fluvial discharges; see Section 2.4.1.1. The

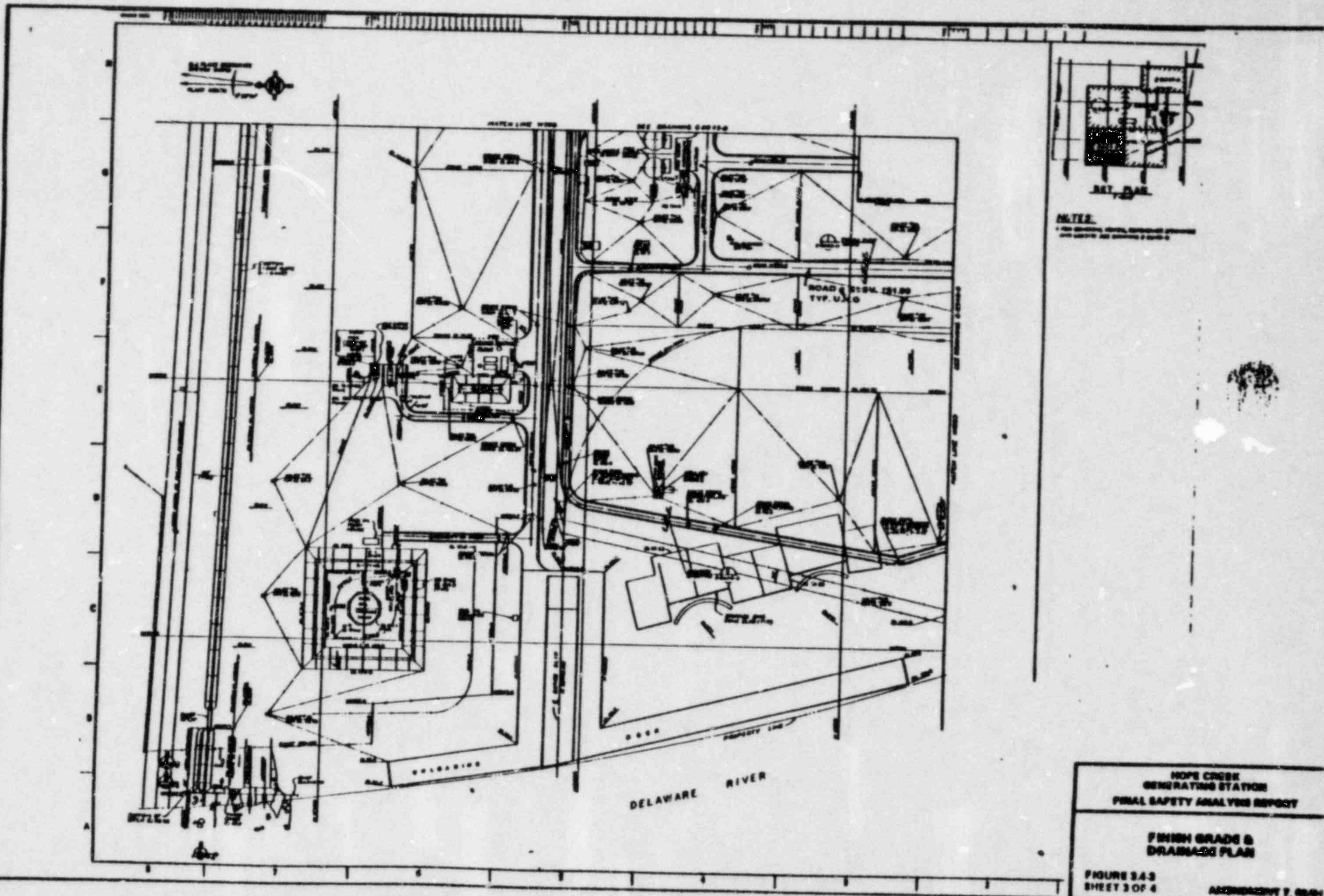


HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

FINISH GRADE &
DRAINAGE PLAN

FIGURE 2.4-3
SHEET 4 of 4

AMENDMENT 7



HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

FINISH GRADE &
DRAINAGE PLAN

FIGURE 2.4-3
SHEET 3 OF 4

AMENDMENT 7, 08/84

DSER Open Item No. 26 (DSER Section 2.5.4)

Intake Structure sliding stability.

The applicant has not provided sufficient information about the sliding stability of the intake structure.

RESPONSE

The requested information has been provided in the responses to NRC Question 241.23 and 241.24.

HCGS

DSER Open Item No. 37 (DSER Section 3.6.2)

FEEDWATER ISOLATION CHECK VALVE OPERABILITY

Assurance is required that the feedwater isolation check valves can perform their function following a postulated break of the feedwater lines outside of containment.

RESPONSE

For the information requested above, see the response to Question 210.20.

QUESTION 210.20 (SECTION 3.6.2)

Provide the basis for assuring that the feedwater isolation check valves can perform their function following a postulated pipe break of the feedwater line outside containment.

RESPONSE

HCGS is in the process of performing a dynamic analysis of the feedwater check valve response to a postulated break in the feedwater line outside containment. A description of the method of analysis, the break location, and the acceptance criteria will be provided in July, 1984. The results of the analysis will be provided by November, 1984.

delete

The feedwater line break outside the containment will be simulated using the RELAP5 computer code. A check valve model which has been developed specifically for calculations of this nature will be used for obtaining the valve dynamics. From this thermal hydraulic analysis, the peak pressures upstream and downstream of the valve disc as well as the maximum disc angular speed will be obtained. As part of this analysis, a sensitivity analysis will be performed to determine the break location and feedwater check valve selection that yields the most conservative stress results.

The stress analysis for the feedwater swing check valve will be performed for fluid transient loads induced by the pipe break event. Elastic and/or inelastic analysis will be performed to determine the primary stress intensities and/or strains at critical locations. For the pressure retaining boundaries such as valve body and valve disk, the calculated primary stress intensities shall not exceed Level D, Service Limit (3.0 Sm). For other valve parts such as seat ring, hinge, and actuator shaft, structural integrity will be evaluated based on strain criteria determined by material properties. The results of this analysis will be provided in November, 1984.

DSER OPEN ITEM NO. 38 (Section 3.6.2)

DESIGN OF PIPE RUPTURE RESTRAINTS

Additional information is required on the design of pipe rupture restraints.

RESPONSE

For the information requested above, see the responses to Questions 210.22 and 210.23.

JES:dh

8/20/84

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DSER Open Item No. 42 (DSER Section 3.8.2)

STEEL CONTAINMENT ULTIMATE CAPACITY ANALYSIS

The applicant has been requested to submit information regarding the ultimate capacity analysis of the containment and steel containment buckling analysis. The staff has not received all the required information on these two items. The applicant has committed to provide the required information to the staff for review by February 15, 1984. The staff will review and report its resolution of these two items in the Final SER.

RESPONSE

A description of the ultimate capacity analysis of the containment has been provided in FSAR Appendix 3I in response to Question 220.22. Additional information was requested in Item B.2 from the NRC Structural/Geotechnical meeting of January 12, 1984. A response to this item was submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer. As a result of discussions with the NRC staff, a revised response to this item has been submitted to the NRC by a letter dated August 3, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 45 (DSER Section 3.8.4)

ACI 349 DEVIATIONS FOR CATEGORY I STRUCTURES

Category I structures other than the containment and its interior structures are all of structural steel and concrete. The structural components consist of slabs, walls, beams, and columns. The major code used in the design of concrete Category I structures is ACI Std 381-71. For steel Category I structures, AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings" is used. The applicant had been requested to provide an assessment and justifications of all deviations from the applicable requirements of ACI 349 as augmented by RG 1.142. The applicant provided the information on January 23, 1984. However, the staff has not completed its review. It will report on the resolution of this item in the Final SER. Additionally, some of the 39 action items discussed in Section 3.8.6 of this SER pertain to this section, and the items remain to be resolved to the satisfaciton of the staff.

RESPONSE

The requested information is included in the revised response to NRC Question 220.26.

QUESTION 220.26 (SECTION 3.8.4)

The SRP specifies that Category I structures shall be designed in accordance with ACI 349 code as augmented by Regulatory Guide 1.142. The Hope Creek Category I structures are designed in accordance with the ACI 318-71 code. The applicant should identify and justify his Category I structural design from the requirements of ACI 349 code as amended by Regulatory Guide 1.142. Furthermore, Hope Creek Category I structural design does not fully comply with the provisions of Regulatory Guides 1.10, 1.55, and 1.94. Since these three Guides have been withdrawn and their regulatory positions are now incorporated into the ACI 359, ACI 349 and ANSI N45.2.5 codes, the applicant should identify and justify its design deviations with respect to the requirements of these codes.

RESPONSE

A. Load Combinations

The HCGS Seismic Category I concrete structures, other than the containment internal concrete structures, are designed for the load combinations listed in Table 3.8-8 which comply with SRP 3.8.4.II.3.b.

B. Structural Acceptance Criteria

The structural acceptance criteria for the HCGS Seismic Caregory I structures, other than the containment internal concrete structures, are in accordance with ACI 318-71 as stated in Section 3.8.4.5. Table 220.26-1 includes a comparison between ACI 318-71 and ACI 349-76 as amended by Regulatory Guide 1.142. As shown in Table 220.26-1, none of the differences have an impact on HCGS. Therefore, the structural acceptance criteria for the HCGS Seismic Category I concrete structures, other than the containment internal concrete structures, complies with the requirements of ACI 349-76 as amended by Regulatory Guide 1.142.

C. Compliance with portions of ACI 359, ACI 349 and ANSI N45.2.5 applicable to subjects previously addressed by Regulatory Guides 1.10, 1.55 and 1.94

Mechanical splicing of reinforcing bars for HCGS Seismic Category I concrete structures complies in general with Regulatory Guide 1.10, Revision 1 as well as ANSI N45.2.5 as described in Section 1.8.1.10.

Concrete placement for HCGS Seismic Category I concrete structures complies with Regulatory Guide 1.55, Revision 0 to the extent described in Section 1.8.1.55. Concrete placement for HCGS Seismic Category I structures also complies with Chapter 5 of ACI 349-76.

Quality assurance requirements for installation, inspection and testing of structural concrete and structural steel during the construction phase of the HCGS Seismic Category I structures complies with ANSI N45.2.5 to the extent described in Section 1.8.1.94.

D. Conclusion

Based on the information given above and in Table 220.26-1, it is concluded that the design of the HCGS Seismic Category I concrete structures, other than the containment internal structures, complies with the requirements of ACI 349-76 as amended by Regulatory Guide 1.142.

The above response was discussed and mutually agreed upon during the NRC structural audit on January 10-12,1984.

ACI 349-76 and Regulatory Guide 1.142	ACI 318-71	Impact of Difference on HCGS
11.16.7 Punching shear stress v_c = weighted average of v_{ch} and v_{cv} where v_{ch} and v_{cv} consider membrane stresses in wall.	11.16.6 refers to 11.10 where; $v = 4\sqrt{f_c'}$ DELETE & REPLACE WITH "INSERT A"	~~None. An evaluation of the HCGS Seismic Category I structures for the effects of the Unit 2 cancellation is in progress. Completion of this evaluation is forecasted by April, 1984. This evaluation will include the investigation of the effects of allowable concrete punching shear stress in walls per ACI 349-76 as amended by Regulatory Guide 1.142. Proper measures will be taken should the ACI 349-76 criteria as amended by Regulatory Guide 1.142 result in an impact to the HCGS design.~~
12.10 Development of welded wire fabric Additional requirements for development length as compared with ACI 318-71.	12.10 Development of welded wire fabric	None. Welded wire fabric does not serve a safety related function for HCGS Seismic Category I structures.
13.3.1.5 Direct design method-limitations "All loads shall be due to gravity only and uniformly distributed over the entire panel. The live load shall not exceed three times the dead load."	13.3.1.5 Direct design method-limitations "The live load shall not exceed three times the dead load."	None. A review of the design of slabs for the HCGS Seismic Category I structures indicates that the direct design method is not used.
15.10(b) Combined footings and mats "The Direct Design Method of Section 13.3 shall not be used to design combined footings and mats."	15.10(b) Combined footings and mats Use of Section 13.3 is not excluded.	None. A review of the design of the foundation mats for the HCGS Seismic Category I structures indicates that the direct design method is not used.

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Insert A

None. A review of the design of HCGS Seismic Category I structures indicates that ACI 349-76, Section 11.16.7 criteria is met.

HOPE CREEK GENERATING STATION
REVIEW FORM FOR RESPONSES TO NRC QUESTIONS

NRC QUESTION NO. 220.26

REFERENCED FSAR SECTIONS 3.8.4

RESPONSE PREPARED BY Bechtel

Return review form to R. S. CAMPANELLA · 22A by 8/17/84

- [x] Site Engineering - Civil, Andy S. Kao, 118, TB-1
- [ ] Controls & Electrical, Irene Felcon, 21D
- [ ] Construction Div./Startup, Jim T. Cox, H.C., 118
- [ ] Environmental Lic., Jim D. Moran, 22A
- [ ] Fuel Supply Dept., Larry A. Murray, 14B
- [ ] Site Engineering, R. Drewnowski, 118, TB-1
- [ ] Licensing & Analysis, Jim F. Scott, 22A
- [ ] Mechanical Div., Pete J. Carr, 19E
- [ ] Nuclear Dept., Maitri Banerjee, 150
- [ ] QA-E&C, Alex Sternberg, 118
- [ ] Other ____________

COMMENTS (Attach additional sheets if necessary)____________

- [ ] This response is not acceptable for the reasons given in comments.
- [ ] This response is acceptable with the modifications specified in the given comments.
- [x] This response is acceptable.

Review Sponsor A. S. Kao Date

Group Head AWC/bd Date

Project Licensing Manager Hope Creek BAP/bd Date

DSER Open Item No. 46 (DSER Section 3.8.5)

ACI 349 DEVIATIONS FOR FOUNDATIONS

The design and analysis procedures that were used for these Category I foundations are the same as those approved on previously licensed applications and, in general, are in accordance with procedures delineated in the ACI 318-71. The various Category I foundations were designed and proportioned to remain within limits established by the staff under the various load combinations. These limits are, in general, based on ACI 318-71 and on the AISC specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme. The applicant had been requested to provide an assessment and justifications of all deviations of this design from the applicable requirements of ACI 349 as augmented by RG 1.142. The applicant provided the information on January 23, 1984. However, the staff has not completed its review. It will report its resolution of this issue in the Final SER. In the meantime, this item remains open. Furthermore, some of the action items discussed in Section 3.8.6 of this SER, as they pertain to the foundation design and analysis, should be considered open items and remain to be resolved.

RESPONSE

The requested information is included in the revised response to NRC Question 220.26.

QUESTION 220.26 (SECTION 3.8.4)

The SRP specifies that Category I structures shall be designed in accordance with ACI 349 code as augmented by Regulatory Guide 1.142. The Hope Creek Category I structures are designed in accordance with the ACI 318-71 code. The applicant should identify and justify his Category I structural design from the requirements of ACI 349 code as amended by Regulatory Guide 1.142. Furthermore, Hope Creek Category I structural design does not fully comply with the provisions of Regulatory Guides 1.10, 1.55, and 1.94. Since these three Guides have been withdrawn and their regulatory positions are now incorporated into the ACI 359, ACI 349 and ANSI N45.2.5 codes, the applicant should identify and justify its design deviations with respect to the requirements of these codes.

RESPONSE

A. Load Combinations

The HCGS Seismic Category I concrete structures, other than the containment internal concrete structures, are designed for the load combinations listed in Table 3.8-8 which comply with SRP 3.8.4.II.3.b.

B. Structural Acceptance Criteria

The structural acceptance criteria for the HCGS Seismic Caregory I structures, other than the containment internal concrete structures, are in accordance with ACI 318-71 as stated in Section 3.8.4.5. Table 220.26-1 includes a comparison between ACI 318-71 and ACI 349-76 as amended by Regulatory Guide 1.142. As shown in Table 220.26-1, none of the differences have an impact on HCGS. Therefore, the structural acceptance criteria for the HCGS Seismic Category I concrete structures, other than the containment internal concrete structures, complies with the requirements of ACI 349-76 as amended by Regulatory Guide 1.142.

C. Compliance with portions of ACI 359, ACI 349 and ANSI N45.2.5 applicable to subjects previously addressed by Regulatory Guides 1.10, 1.55 and 1.94

Mechanical splicing of reinforcing bars for HCGS Seismic Category I concrete structures complies in general with Regulatory Guide 1.10, Revision 1 as well as ANSI N45.2.5 as described in Section 1.8.1.10.

Concrete placement for HCGS Seismic Category I concrete structures complies with Regulatory Guide 1.55, Revision 0 to the extent described in Section 1.8.1.55. Concrete placement for HCGS Seismic Category I structures also complies with Chapter 5 of ACI 349-76.

Quality assurance requirements for installation, inspection and testing of structural concrete and structural steel during the construction phase of the HCGS Seismic Category I structures complies with ANSI N45.2.5 to the extent described in Section 1.8.1.94.

D. Conclusion

Based on the information given above and in Table 220.26-1, it is concluded that the design of the HCGS Seismic Category I concrete structures, other than the containment internal structures, complies with the requirements of ACI 349-76 as amended by Regulatory Guide 1.142.

The above response was discussed and mutually agreed upon during the NRC structural audit on January 10-12,1984.

ACI 349-76 and Regulatory Guide 1.142	ACI 318-71	Impact of Difference on HCGS
11.16.7 Punching shear stress v_c = weighted average of v_{ch} and v_{cv} where v_{ch} and v_{cv} consider membrane stresses in wall.	11.16.6 refers to 11.10 where; $v = 4\sqrt{f_c'}$	~~None. An evaluation of the HCGS Seismic Category I structures for the effects of the Unit 2 cancellation is in progress. Completion of this evaluation is forecasted by April, 1984. This evaluation will include the investigation of the effects of allowable concrete punching shear stress in walls per ACI 349-76 as amended by Regulatory Guide 1.142. Proper measures will be taken should the ACI 349-76 criteria as amended by Regulatory Guide 1.142 result in an impact to the HCGS design.~~ DELETE & REPLACE WITH "INSERT A"
12.10 Development of welded wire fabric Additional requirements for development length as compared with ACI 318-71.	12.10 Development of welded wire fabric	None. Welded wire fabric does not serve a safety related function for HCGS Seismic Category I structures.
13.3.1.5 Direct design method-limitations "All loads shall be due to gravity only and uniformly distributed over the entire panel. The live load shall not exceed three times the dead load."	13.3.1.5 Direct design method-limitations "The live load shall not exceed three times the dead load."	None. A review of the design of slabs for the HCGS Seismic Category I structures indicates that the direct design method is not used.
15.10(b) Combined footings and mats "The Direct Design Method of Section 13.3 shall not be used to design combined footings and mats."	15.10(b) Combined footings and mats Use of Section 13.3 is not excluded.	None. A review of the design of the foundation mats for the HCGS Seismic Category I structures indicates that the direct design method is not used.

Insert A

None. A review of the design of HCGS Seismic Category I structures indicates that ACI 349-76, Section 11.16.7 criteria is met.

DSER Open Item No. 48 (DSER Section 3.8.6)

ROCKING TIME HISTORIES

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.4 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer. As a result of discussions with the NRC staff, a revised response to this item has been submitted to the NRC by a letter dated August 3, 1984 from R. L. Mittl to A. Schwencer.

HCGS

DSER Open Item No. 49 (DSER Section 3.8.6)

GROSS CONCRETE SECTION

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.11 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Shwencer. As a result of discussions with the NRC staff, a revised response to this item has been submitted to the NRC by a letter dated August 3, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 50 (DSER Section 3.8.6)

VERTICAL FLOOR FLEXIBILITY RESPONSE SPECTRA

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.12 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer. As a result of discussions with the NRC staff, a revised response to this item has been submitted to the NRC by a letter dated *August 3, 1984* from R. L. Mittl to A. Schwencer.

DSER Open Item No. 51 (DSER Section 3.8.6)

COMPARISON OF BECHTEL INDEPENDENT VERIFICATION RESULTS WITH THE DESIGN_BASIS RESULTS

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.13 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated August 3, 1984, from R. L. Mittl to A. Schwencer. As a result of discussion with the NRC staff, a revised response to this item is attached. In addition the response to Question 220.21 has been revised to reflect this information.

Revised Response
Revision 2
August 3, 1984

Meeting Date: January 10, 1984

Question No: A-13

Question: Provide comparison of Bechtel Independent Verification Results with the Design Basis Results.

Response:

As described in Amendment 1 of the FSAR (Section 3.7.2.4), three independent seismic soil-structure interaction analyses are performed for the major plant structures. The design basis analyses are performed using the finite element method by EDS Nuclear, Inc. (presently known as Impell Corporation). Independent finite element soil-structure interaction analyses are subsequently performed by Bechtel to verify the design basis analyses. In addition, in accordance with the requirements of the Standard Review Plan, Section 3.7.2 (NUREG 0800), impedance approach (the half-space) soil-structure interaction analyses are performed by Bechtel. The analytical method utilized for the impedance approach seismic soil-structure interaction analyses of power block structures and service water intake structure is given in FSAR Section 3.7.2.1. Figure A-13-1 summarizes the division of responsibilities for the seismic analyses. The structural models and soil properties used in the analysis are given in Appendix A.

Figures A-13-2 to A-13-37 show the comparison of the response spectra (2% damping) obtained from the above three seismic soil-structure interaction analyses. Discussions of these comparisons are as follows:

Power Block Structures

I. Comparison of Design basis and Independent Finite Element Verification Response Spectra

Bechtel's independent soil-structure interaction analyses are performed using the computer code FLUSH. The results of independent finite element analyses are in reasonable agreement with those of the design basis analyses. As can be seen from Figures A-13-2 through A-13-37, the horizontal response spectra obtained from the independent finite element analyses are generally enveloped by those obtained from the design basis analyses except for the frequency range lower than 2 Hz. The vertical response spectra showed some exceedances at the frequency range of 18 Hz. These exceedances are listed in Table A-13-1.

The effects of these exceedances are evaluated for the combined responses in three directions using the SRSS approach and compared with the design basis results. Table A-13-2 provides these comparisons. In all cases, these variations are judged to be minor and can be accommodated within the design margin. In areas where multimodal analysis is performed, the effects of these variations will be further reduced. It has been concluded that the variations between these two analyses are within the accuracy of analyses and can be accommodated within the design margin.

II. Comparison of Design Basis and Impedance Approach Response Spectra

The peak spectral accelerations obtained from the impedance approach analyses are generally lower than those obtained from the design basis analyses. However, these response spectra are not completely enveloped by those obtained from the design basis analysis, especially in the frequency range between 1.0 and 3.5 Hz. Also, there are some local exceedances in the higher frequency range, as shown in Figures A-13-2 through A-13-37.

As discussed during the NRC Structural Design Audit, dated January 10, 1984, sampling studies have been performed to confirm the adequacy of the plant design. Table A-13-3 describes the criteria used in selection of the samples for this study.

The results of sampling studies are as follows:

1. Structures

All major reinforced concrete shear walls at the base of the reactor building have been evaluated for seismic forces and moments obtained from the impedance approach analyses. These walls represent approximately 40 percent of the total number of shear walls in the reactor building. The actual shear stresses resulting from the impedance approach analyses were evaluated and found to be lower than the design basis stresses. Table A-13-4 provides the comparision of shear stresses at El. 54'-0. Tables A-13-5a and A-13-5b show the comparision of impedance approach and design basis moments for OBE and SSE cases respectively. The impedance approach moments exceed the design basis moments at a few wall locations as identified on Tables A-13-5a and A-13-5b. These walls were reevaluated and the resulting moments were found to be less than the allowables.

The auxiliary building seismic forces and moments obtained from the impedance approach analysis are less than the design basis shears and moments. Therefore, no further evaluation of the auxiliary building structure is necessary.

Based on the above, it is concluded that the as-built power block structures can accommodate the loads obtained from the impedance approach analysis.

2. Equipment

The effects of the impedance approach response spectra was evaluated on 26 types of equipment. The selected items are located in the areas where the impedance approach spectra were found to have higher spectral accelerations than those of the design basis response spectra. Each equipment was evaluated in accordance with the procedure described in Table A-13-3, and the results of the evaluation are summarized in Table A-13-6. In all cases, the as-built equipment designs were found acceptable.

3. Cable Tray and HVAC Supports

a. Cable Tray Support

Approximately 200 supports were evaluated. In all cases, the existing designs were determined to be acceptable.

b. HVAC Supports

Over 200 supports were evaluated. In all cases, it was found that the design basis spectral accelerations exceeded the impedance approach spectral accelerations for the support frequencies. Therefore, the HVAC supports were considered acceptable.

4. Piping and Pipe Supports

A total of 10 representative piping system calculations were selected out of 64 calculations affected by the impedance approach analysis results. The selection of these calculations was based on the criteria given in Table A-13-3.

The objective of performing detailed dynamic seismic analysis of the sample calculation was to demonstrate that although the design basis curve did not envelop the impedance curves in the low frequency range, such deviation do not have any affect on the adequacy of existing piping analysis and support design. In other words, the stresses and loads generated using the impedance response spectra curve as input are still within the ASME Section III code allowable for pipe and pipe support design.

The methodology used for evaluation was to subject the selected existing mathematical models of piping systems to the impedance approach response spectra and to compare the resulting pipe stresses with the ASME Section III code allowables for pipe and pipe support design. The reactions at equipment nozzles were compared with vendor's design allowables. All pipe supports were evaluated for adequacy under the revised loads.

In all cases, the pipe stresses were found to be within the code allowables as shown in Table A-13-7. Also, as illustrated in Table A-13-7, the equipment nozzle allowables were also met. The existing pipe support designs were also found adequate for the new loads and met the ASME Section III code Subsection NF allowables. This is illustrated in Table A-13-8.

Intake Structure

See responses to questions A-14 and A-16, meeting date January 11, 1984.

APPENDIX A

Impedance Approach Structural Models and Soil Properties

In the soil-structure interaction analysis, using the impedance approach, the effect of dynamic stiffness of the foundation medium is represented by the foundation impedances, which are functions of the base mat dimensions, embedment depth, elastic properties of the foundation medium, and forcing frequencies. With the foundation impedance known, the structure-foundation system is modeled by coupling the fixed-base structure model with the foundation impedances through the basemat (Figure A-13-38). For this study the effects of embedment which increase both damping and stiffness of the soil-structure systems are considered. However, the wave scattering effect is conservatively neglected in the present impedance approach analysis. This is consistent with the requirement specified in SRP Section 3.7.2.

The impedance approach seismic soil-structure interaction analysis of the reactor building (Figure A-13-39) and the auxiliary building (Figure A-13-40) is performed for both the SSE and OBE cases. The foundation soil is assumed to be a uniform visco-elastic half space. The weighted average of the final iterated shear moduli of 3,522 ksf (shear wave velocity of 989 ft./sec.) and 5,235 ksf (shear wave velocity of 1,205 ft./sec.) respectively, are used in calculating the horizontal SSE and OBE impedance functions. Since the groundwater table is located at elevation 98.0 ft., a compressional wave velocity, Vp, of 4,800 ft./sec. is used for the vertical analysis. The computed OBE and SSE translational and rocking impedances for the embedded reactor building and auxiliary building foundations are given in Tables A-13-9 to A-13-12.

Table A-13-1

Comparison of Design basis and Independent Finite Element Verification Response Spectra

Building	Key Elevation	Design Earthquake	Earthquake Direction	Locations of Variations (Note 1)	Figure No.	Item No.	Spectral Acceleration (Note 2)	
							Design Basis (g)	Bechtel FLUSH (g)
REACTOR	102	SSE	N-S	1.8 Hz	A-13-3	1	0.62	0.75
	201	SSE	N-S	1.8 Hz	A-13-4	2	1.00	1.22
	54	SSE	Vertical	18.5 Hz	A-13-8	3	1.50	1.75
	102	SSE	Vertical	22.0 Hz	A-13-9	4	1.35	1.68
	201	SSE	Vertical	18.0 Hz	A-13-10	5	2.15	2.45
AUXILIARY	54	SSE	N-S	3.6 Hz	A-13-11	6	1.34	1.56
	54	SSE	E-W	3.0 Hz	A-13-14	7	0.88	1.44
	102	SSE	E-W	3.0 Hz	A-13-15	8	1.10	1.68
	178	SSE	E-W	3.2 Hz	A-13-16	9	1.40	1.92
	102	SSE	Vertical	14.0 Hz	A-13-18	10	1.83	1.95
	178	SSE	Vertical	22.0 Hz	A-13-19	11	1.53	1.85

Table A-13-1 (Cont'd)

Comparison of Design Basis and Independent
Finite Element Verification Response Spectra

Building	Key Elevation	Design Earthquake	Earthquake Direction	Locations of Variations (Note 1)	Figure No.	Item No.	Spectral Acceleration (Note 2)	
							Design Basis (g)	Bechtel FLUSH (g)
REACTOR	102	OBE	N-S	1.7 Hz	A-13-21	12	0.34	0.42
	54	OBE	E-W	4.3 Hz	A-13-23	13	0.50	0.67
	201	OBE	E-W	1.8 Hz	A-13-25	14	0.38	0.55
	102	OBE	Vertical	22.0 Hz	A-13-27	15	1.20	1.42
	201	OBE	Vertical	18.0 Hz	A-13-28	16	1.68	1.85
AUXILIARY	54	OBE	N-S	4.9 Hz	A-13-29	17	1.15	1.40
	54	OBE	E-W	4.4 Hz	A-13-32	18	0.75	0.85
	54	OBE	Vertical	22.0 Hz	A-13-35	19	1.17	1.26
	102	OBE	Vertical	18.0 Hz	A-13-37	20	1.47	1.54
	178	OBE	Vertical	18.0 Hz	A-13-37	21	1.80	1.95

NOTES:
1. This column identifies those locations where the results of the independent analysis exceed those of the design basis analysis.
2. For vertical earthquake direction, spectral acceleration includes the effect of gravity load (1.0 g).

Table A-13-2

SRSS Spectral Acceleration Comparison between Design Basis and Finite Element Verification Analysis

Item No.	SRSS Spectral Acceleration Comparison(g) (Note 1)		
	(A) Design Basis	(B) Bechtel-FLUSH	(B-A)/A Difference (%)
1	1.97	1.75	-11
2	2.24	2.20	-2
3	1.53	1.78	16
4	1.39	1.72	24
5	2.23	2.49	12
6	2.86	2.68	-6
7	2.34	2.32	-1
8	2.56	2.48	-3
9	4.27	3.44	-19
10	1.87	1.93	4
11	1.73	1.93	11
12	1.41	1.38	-2
13	2.02	1.66	-18
14	1.52	1.50	-1
15	1.21	1.43	18
16	1.71	1.86	9
17	2.24	2.07	-8
18	2.23	1.94	-13
19	1.19	1.27	7
20	1.86	1.99	7
21	1.51	1.56	3

NOTE: 1. The SRSS spectral acceleration values include the effect of gravity loads (1.0 g)

TABLE A-13-3

PROCEDURES FOR EVALUATION OF STRUCTURES, EQUIPMENT & COMPONENTS USING IMPEDANCE ANALYSIS RESULTS

INTRODUCTION

The results of the impedance analysis are used to assess the existing design of the HCGS structures, equipment and components. A sampling approach is used. The procedure for this evaluation is as follows:

A. STRUCTURES:

Since the maximum shear and axial forces and the maximum overturning moments occur at the base of the structures, and the design margins for the upper elevations are greater than those of the base, the effects of these loads at the base of each structure are evaluated.

B. EQUIPMENT:

The impedance analysis spectra in general are not completely enveloped by the design basis spectra in the following areas,

i) 1.0 to 3.5 Hz range throughout the reactor and auxiliary buildings

ii.) 6 to 15 Hz range in the reactor building at elevation 102 ft and below.

iii.) 6 to 15 Hz in the auxiliary building at elevation 54 ft.

Since typical equipment frequencies are not found in the range of 1.0 to 3.5 Hz, the item (i) above does not need any further evaluation. Items (ii) and (iii) are reconciled as follows:

- Review the significant frequencies of approximately 30% of all equipment selected at random and located in the areas where spectral variations were noted.
- If the significant equipment frequencies fall in the range where the difference in the spectra exist, additional evaluation is necessary. No further evaluation is necessary if the significant frequencies are outside the frequency range in question.
- The evaluation is performed either by comparing the test response spectra of the equipment with the impedance spectra (if the equipment is qualified by testing) or comparing the actual-to-allowable stress ratios with the spectrum exceedance ratios.
- If the above evaluation shows the equipment may not be qualified for the impedance spectra, detailed evaluation consisting of analysis and/or testing is performed.

- As a result of evaluation, if equipment requires modifications, the sample size for this evaluation is expanded as required.

C. CABLE TRAY AND HVAC SUPPORTS

Cable tray and HVAC supports do not have frequencies in the range of 1.0 to 3.5 Hz. Therefore any differences between the two spectra in this frequency range do not require any evaluation.

The effects of the spectrum exceedances at frequency range between 6 and 15 Hz are evaluated for approximately 200 cable tray and HVAC supports. These supports are selected at random but are located at the lower elevation (Reactor Building El. 54 to 102 ft., Auxiliary Building El. 54 ft.) where the spectrum differences exist. If the results of evaluation indicate need for modifications to any support, the sample size for this evaluation is expanded as required.

D. PIPING AND PIPE SUPPORTS

In general, impedance curves resulted in significant reductions in response spectrum peak accelerations as compared to those of the design basis curves. However, frequency shifts were observed in some curves, particularly in the low frequency ranges. To evaluate the effects of the frequency shift, a "biased" sample of affected piping systems is reanalyzed and reevaluated. The sample is selected as follows:

Individual impedance curves for various elevations and structures are superimposed on their corresponding design basis curves to identify those impedance curves which are not enveloped by design basis curves. Those impedance curves are then superimposed on the design basis "enveloped" response spectra used for various piping system design calculations. If the design basis enveloped response spectra curves affecting a calculation did not totally envelop all the corresponding impedance curves, that particular calculation is then identified as "affected" and a candidate for sampling.

A "biased" sample of the "affected" calculations was selected which emphasized the following important piping parameters:

1. Stress levels in the existing pipe stress calculations. Samples included systems with high stress levels.

2. Difference in "g" level (Δg) between impedance and design basis curves in the affected frequency zones. Sample selected to include curves showing significant differences.

3. High equipment nozzle loads in existing calculation.

4. Relative location of piping system in the plant in an attempt to include response of all structures in the sample selected.

The number of calculations included in the sample is:

Building	Total No. of Q-Calcs	No. of Calcs Reviewed	No. of Calcs affected	No. of Calcs in the sample
Drywell	32	32	23	3
Reactor	213	213	34	5
Auxiliary	124	124	7	2

Results of the analysis including support loads are compared against the design basis values for acceptability.

TABLE A-13-4

REACTOR BUILDING SHEAR STRESSES AT EL. 54'-0"

Wall Location	Design Basis Psi	Impedance Approach Psi	Allowable Psi
North Wall	323	207	630
South Wall	333	224	630
East Wall	298	261	630
West Wall	303	268	630
Cylindrical Shell	257	251	630
Pedestal	27	91	126

SOUTH RADWASTE SHEAR STRESSES AT EL. 54'-0"

Wall Location	Design Basis Psi	Impedance Approach Psi	Allowable Psi
North Wall	183	207	630
South Wall	216	224	630
East Wall	208	276	630
West Wall	458	257	630

Notes: 1. Concrete f'c = 4000 Psi

2. See FSAR Figures 1.2-2 for wall location.

TABLE A-13-5a

REACTOR/RADWASTE BUILDING - OBE SEISMIC MOMENTS AT EL. 54'0"

Wall Location	Design Basis Method (Kip-Ft)	Impedance Approach Method (Kip-Ft)
North-Reactor North-Radwaste	359,200	414,500
South-Reactor South-Radwaste	517,400	847,700
East-Radwaste	461,000	421,900
West-Radwaste	329,000	290,700
East-Reactor	434,500	276,900
West-Reactor	588,600	482,900
Cylindrical Shell	2,772,000 (N-S) 1,723,000 (E-W)	1,847,000 (N-S) 1,639,000 (E-W)

Note: See FSAR Figure 1.2-2 for wall location.

TABLE A-13-5b

REACTOR/RADWASTE BUILDING - SSE SEISMIC MOMENTS AT EL. 54'0"

Wall Location	Design Basis Method (Kip-Ft)	Impedance Approach Method (Kip-Ft)
North-Reactor North-Radwaste	912,100	699,100
South-Reactor South-Radwaste	1,344,000	1,429,000
East-Radwaste	675,000	732,300
West-Radwaste	654,000	504,500
East-Reactor	909,000	480,200
West-Reactor	1,320,000	837,400
Cylindrical Shell	4,471,000 (N-S) 3,054,000 (E-W)	3,092,000 (N-S) 2,668,000 (E-W)

Note: See FSAR Figure 1.2-2 for wall location.

TABLE A-13-6

POWER BLOCK SEISMIC CATEGORY I EQUIPMENT

Equipment or Component	Tag No.	Location Blgd./El.	Equipment Frequencies (Hz)	Method of Seismic Qualification	Applicable Note
HPCI Turbine	E41-C002	Reactor Bldg. El. 54	Horizontal- 10, 12 Vertical - 23	Testing	1
Residual Heat Removal Pump/ Motor	E11-C002	Reactor Bldg. El. 54	Horizontal- 8.7, 9.7 Vertical - >33	Analysis	3
Control Room Panels	H11-P617 H11-P618 H11-P640 H11-P641	Aux. Bldg. El. 102	Horizontal- 11.5, 16 Vertical - >33	Testing	1
Control Room Panels	H11-P620 through H11-P623 H11-P628 H22-P631	Aux. Bldg. El. 102	Horizontal- 21, 29 Vertical - >33	Testing	1
Control Room Panels	H11-P635 H11-P636	Aux. Bldg. El. 137	Horizontal- 19, 37 Vertical - >33	Testing	1
Control Room Panels	H11-608	Aux. Bldg. El. 137	Horizontal- 7, 12 Vertical - >33	Testing	1
Control Room Panels	H11-609 H11-611	Aux. Bldg. El. 137	Horizontal- 22, 37 Vertical - >33	Testing	1
RCIC Turbine	E51-C002	Reactor Bldg. El. 54	Horizontal- 16 Vertical - 18	Analysis & Testing	1, 2
LPCS Pump/ Motor	E21-C001	Reactor Bldg. El. 54	Horizontal- 11.5, 12.7 Vertical - >33	Analysis	2

TABLE A-13-6 (Cont'd)

POWER BLOCK SEISMIC CATEGORY I EQUIPMENT

Equipment or Component	Tag No.	Location Blgd./El.	Equipment Frequencies (Hz)	Method of Seismic Qualification	Applicable Note
Chiller Water Tank	IAT, BT 410, 413	D. G. * El. 178	Horizontal - >33 Vertical - >33	Analysis	2
ECCS Jockey Pump	IAP, BP, CP, DP 228	Reactor Bldg. El. 54	Horizontal - >33 Vertical - >33	Analysis	2
SACS Expansion Tank	IAT, BT 205	Reactor Bldg. El. 201	Horizontal - 12.5 Vertical - >33	Analysis	2
5.0 Kv Switch-gear	IAN, BN, CN, DN 205	Reactor Bldg. El. 102	Horizontal - 8, 14 Vertical - 30	Testing	1
DC Switchgear & Control Center	IOD 251, 261	Reactor Bldg. EL. 54	Horizontal - 8, 35 Vertical - 20	Testing	1
Batteries Racks	IOD 421, 431	Aux. Bldg. El. 54	Horizontal - 14, 16 Vertical - 28	Testing	1
Inst.AC Power Panel	IYF 401-407 IYF 209	Aux. Bldg. El. 102	Horizontal - 17, 21 Vertical - 6	Testing	1
Control Panel	IAC, BC 201	Reactor Bldg. El. 102	Horizontal - 8, 17 Horizontal - >33	Analysis	2

Note: *D.G. - Diesel generator area of the auxiliary building.

TABLE A-13-6 (Cont'd)

POWER BLOCK SEISMIC CATEGORY I EQUIPMENT

Equipment or Component	Tag No.	Location Blgd./El.	Equipment Frequencies (Hz)	Method of Seismic Qualification	Applicable Note
Standby Diesel Generator Set	1(A-D)G 400	D. G. El. 102	Horizontal - >15 Vertical - >15	Analysis	2
SACS Heat Exchanger	1AIE, 1A2E201 1BIE, 1B2E201	Reactor Bldg. El. 54	Horizontal - 8, 10.4 Vertical - 21	Analysis	2
SACS Pumps	1(A-D)P210	Reactor Bldg. El. 201	Horizontal - >33 Vertical - >33	Analysis	2
Control Panel	ICC, DC201	Reactor Bldg. El. 102	Horizontal- 12.7, 17.6 Vertical - 29	Analysis	2
Accumulator Tank	1AT, BT412	D. G. EL. 54	Horizontal - 31, 33 Vertical - 35	Analysis	2
Air Handling Units A/C Units	1AVH407 1BVH407	D. G. El. 178	Horizontal - 16.6, 18 Vertical - 19	Analysis	2
Unit Cooler	1AVH208 1AVH209 1BVH208 1BVH209	Reactor Bldg. El. 102	Horizontal - 9.4, 21 Vertical - 26.4	Analysis	2
HVAC Control Panels	1AC, CC285 1AC, CC281 1AC, DC483	D. G. El. 178	Horizontal - 12.7, 16.4 Vertical - 16.9	Analysis	2
Centrifugal Water Chiller	1AK, BK403	D. G. El. 178	Horizontal - >30 Vertical - >30	Analysis	2

Notes: 1. TRS envelopes impedance approach spectra.

2. Impedance approach spectral acceleration is lower than that of the design-basis response spectra in the major equipment frequencies.

3. Although impedance approach spectral acceleration exceeds that of design basis response spectra in the equipment frequency range, a more detailed calculation showed that the equipment stresses are within the code allowables.

TABLE A-13-7

POWER BLOCK PIPE STRESS SUMMARY

Building	Calc. No.	Max. Seismic Stress Ratios: Max. Impedance Stress / Max. Design Basis Stress		ASME Code Equation Evaluation		Vendor Equip. Nozzle Allowables Met
				Eq. 9B* / Code Allowable	Eq. 9D* - / Code Allowable	
		OBE	SSE	Upset	Faulted	
Auxiliary	C1549	0.51	0.76	0.29	0.66	YES
	C1581	0.64	0.86	0.40	0.28	YES
Drywell	C118	0.75	0.83	0.44	0.34	YES
	C1842	0.65	0.83	0.63	0.85	YES
	C120	0.30	0.52	0.49	0.39	YES
Reactor	C988	0.88	0.75	0.54	0.35	YES
	C911	0.88	0.94	0.84	0.63	YES
	C963	1.10	1.18	0.71	0.47	YES
	C918	0.29	0.39	0.33	0.21	YES
	C937	0.90	1.15	0.70	0.38	YES

*ASME Section III NC, ND-3652

TABLE A-13-8

POWER BLOCK PIPE SUPPORT LOAD SUMMARY

Building	Calc. No.	Total No. of Supports	No. of Supports with Load Increase	Average Percentage increase in Load		Support Design Adequate
				Upset	Faulted	
Auxiliary	C1549	5	0	N/A	N/A	YES
	C1581	16	6	11%	NONE	YES
Drywell	C118	8	1	2%	1%	YES
	C1842	34	0	N/A	N/A	YES
	C120	18	2	7%	NONE	YES
Reactor	C988	11	3	NONE	14%	YES
	C911	34	6	20%	17%	YES
	C963	7	4	27%	28%	YES
	C918	10	0	N/A	N/A	YES
	C937	17	5	17%	21%	YES

TABLE A-13-9

VALUES OF SOIL STIFFNESS AND DAMPING COEFFICIENTS
OF REACTOR BUILDING (OBE CASE)

DIRECTION	STIFFNESS COEFFICIENTS	DAMPING COEFFICIENTS
VERTICAL TRANSLATION	1.53×10^{7} k/ft	9.19×10^{5} k-sec/ft
NORTH-SOUTH TRANSLATION	7.26×10^{6} k/ft	6.42×10^{5} k-sec/ft
EAST-WEST TRANSLATION	5.90×10^{6} k/ft	5.74×10^{5} k-sec/ft
ROCKING ABOUT NORTH-SOUTH AXIS	1.26×10^{11} k/ft/rad	9.50×10^{9} k-ft-sec/rad
ROCKING ABOUT EAST-WEST AXIS	7.56×10^{10} k/ft/rad	3.31×10^{9} k-ft-sec/rad

K56(2)

TABLE A-13-10

VALUES OF SOIL STIFFNESS AND DAMPING COEFFICIENTS OF 3-D REACTOR BUILDING (SSE CASE)

DIRECTION	STIFFNESS COEFFICIENTS	DAMPING COEFFICIENTS
VERTICAL TRANSLATION	1.53×10^{7} k/ft	9.19×10^{5} k-sec/ft
NORTH-SOUTH TRANSLATION	4.74×10^{6} k/ft	5.17×10^{5} k-sec/ft
EAST-WEST TRANSLATION	4.03×10^{6} k/ft	4.79×10^{5} k-sec/ft
ROCKING ABOUT NORTH-SOUTH AXIS	8.17×10^{10} k-ft/rad	8.47×10^{9} k-ft-sec/rad
ROCKING ABOUT EAST-WEST AXIS	5.14×10^{10} k-ft/rad	2.70×10^{9} k-ft-sec/rad

K56(2)

TABLE A-13-11

VALUES OF SOIL STIFFNESS AND DAMPING COEFFICIENTS
OF AUXILILARY BUILDING (OBE CASE)

DIRECTION	STIFFNESS COEFFICIENTS	DAMPING COEFFICIENTS
VERTICAL TRANSLATION	1.40×10^{7} k/ft	7.87×10^{5} k-sec/ft
NORTH-SOUTH TRANSLATION	7.15×10^{6} k/ft	5.71×10^{5} k-sec/ft
EAST-WEST TRANSLATION	5.58×10^{6} k/ft	5.21×10^{5} k-sec/ft
ROCKING ABOUT NORTH-SOUTH AXIS	1.15×10^{11} k-ft/rad	9.32×10^{9} k-ft-sec/rad
ROCKING ABOUT EAST-WEST AXIS	5.56×10^{10} k-ft/rad	1.76×10^{9} k-ft-sec/rad

K56(2)

TABLE A-13-12

VALUES OF SOIL STIFFNESS AND DAMPING COEFFICIENTS
OF AUXILIARY BUILDING (SSE CASE)

DIRECTION	STIFFNESS COEFFICIENTS	DAMPING COEFFICIENTS
VERTICAL TRANSLATION	1.40×10^7 k/ft	7.87×10^5 k-sec/ft
NORTH-SOUTH TRANSLATION	4.89×10^6 k/ft	4.26×10^5 k-sec/ft
EAST-WEST TRANSLATION	3.76×10^6 k/ft	4.28×10^5 k-sec/ft
ROCKING ABOUT NORTH-SOUTH AXIS	7.33×10^{10} k-ft/rad	7.72×10^9 k-ft-sec/rad
ROCKING ABOUT EAST-WEST AXIS	3.61×10^{10} k-ft/rad	1.62×10^9 k-ft-sec/rad

K56(2)

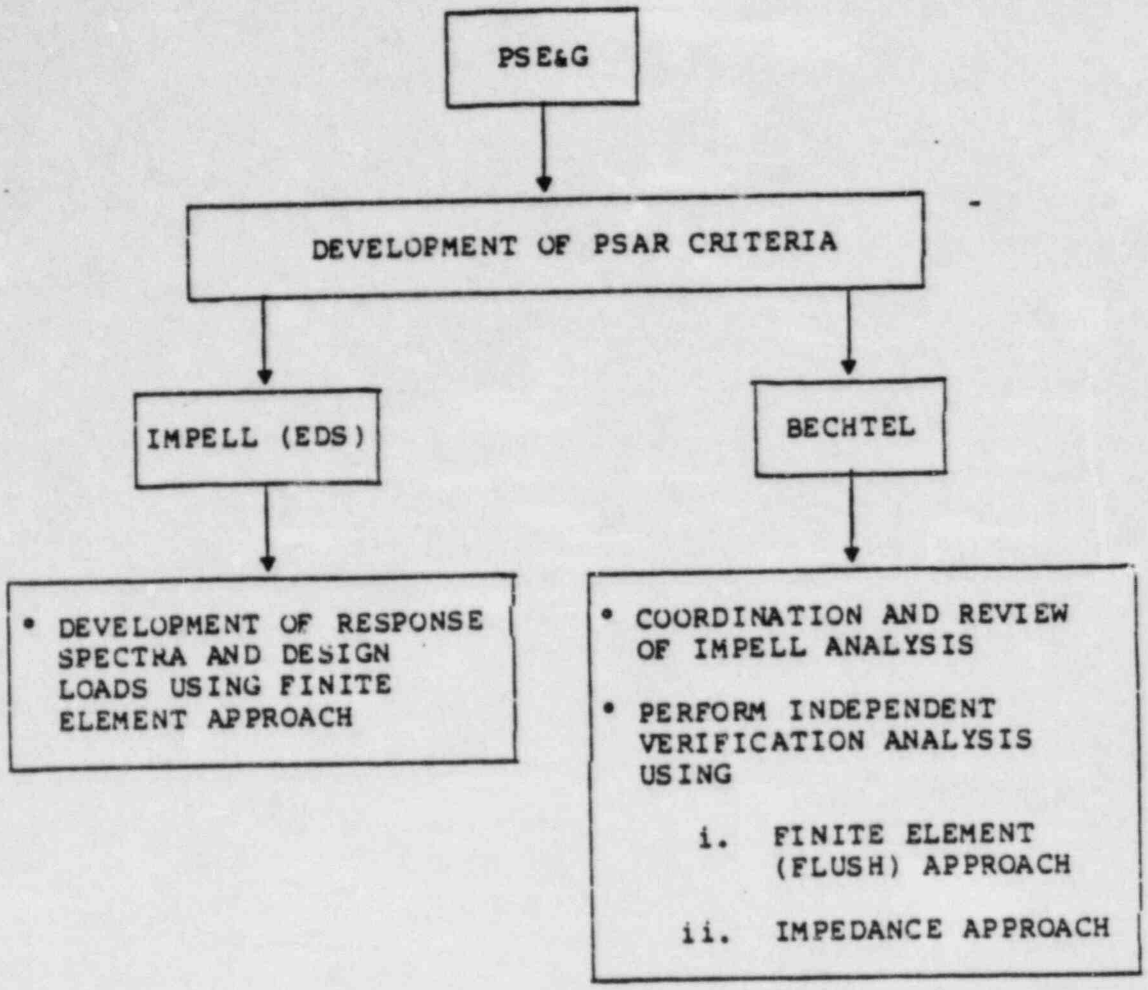


Figure A-13-1

Division of Responsibility

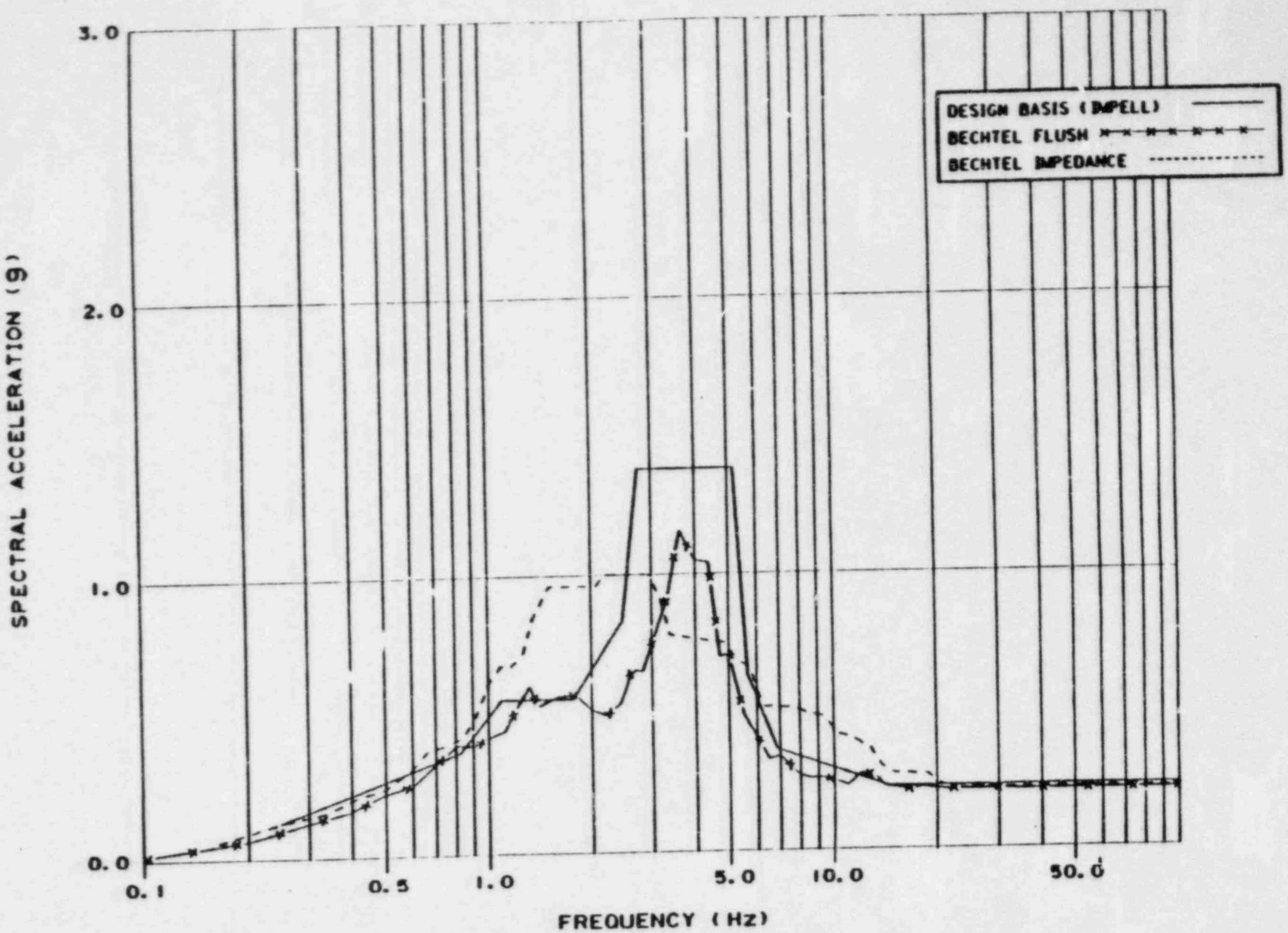


FIGURE A-13-2

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 54'-0",
N-S, SSE, 2% DAMPING

HOPE CREEK UNIT I
CAD, LIA, REV I
04/19/84

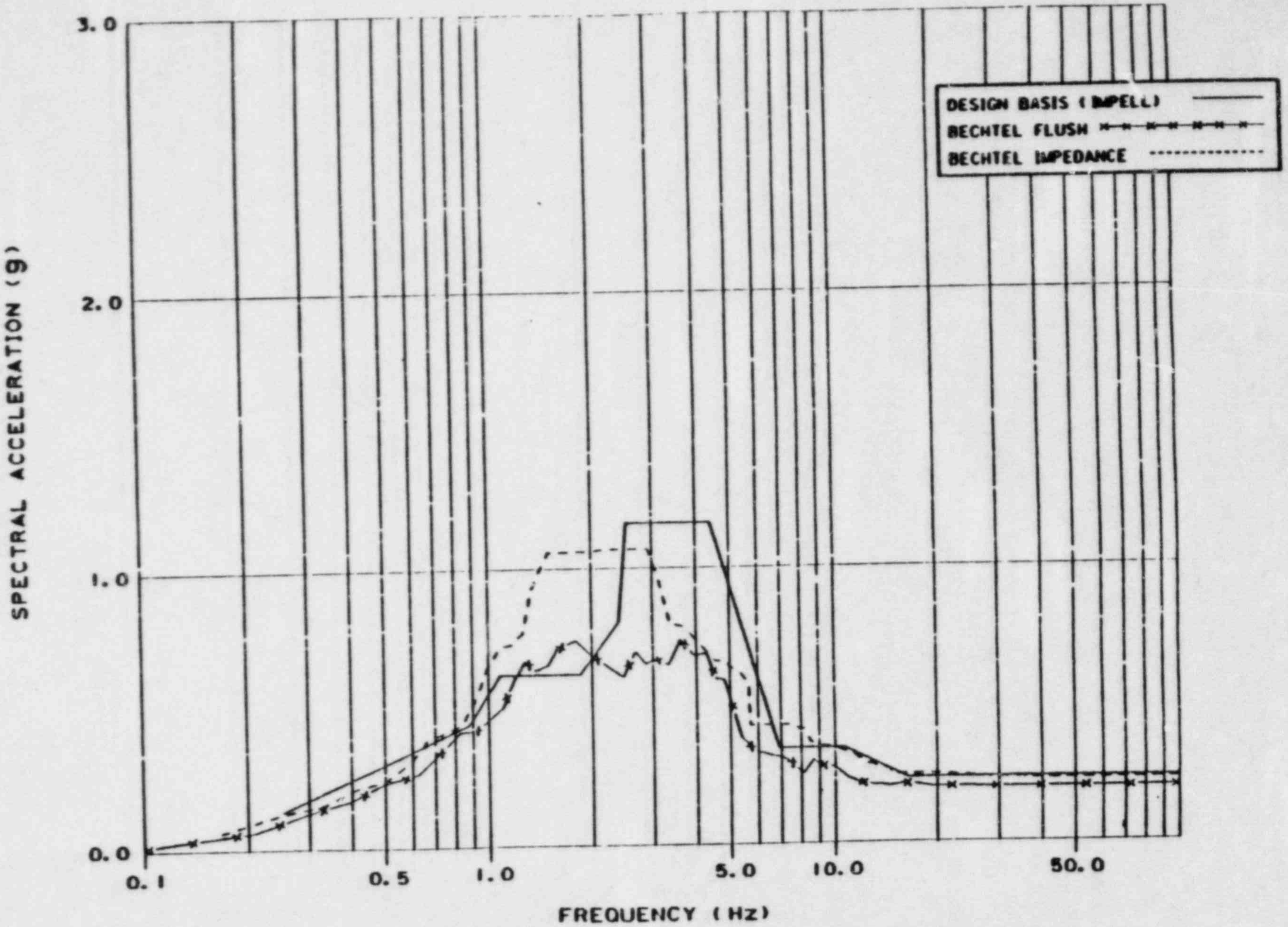


FIGURE A-13-3

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 102' -0",
N-S, SSE, 2% DAMPING

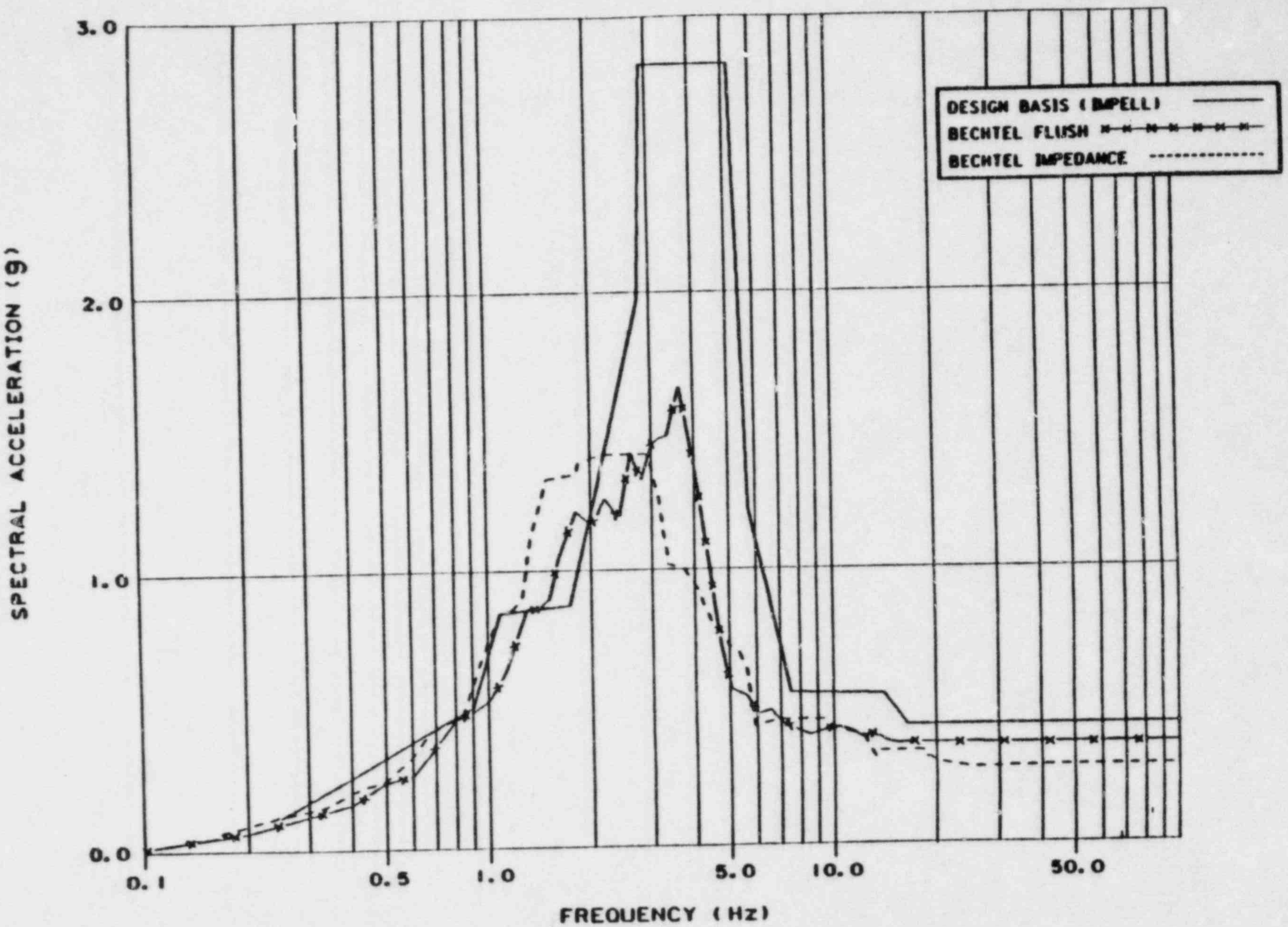


FIGURE A-13-4

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 201'-0",
N-S, SSE, 2% DAMPING

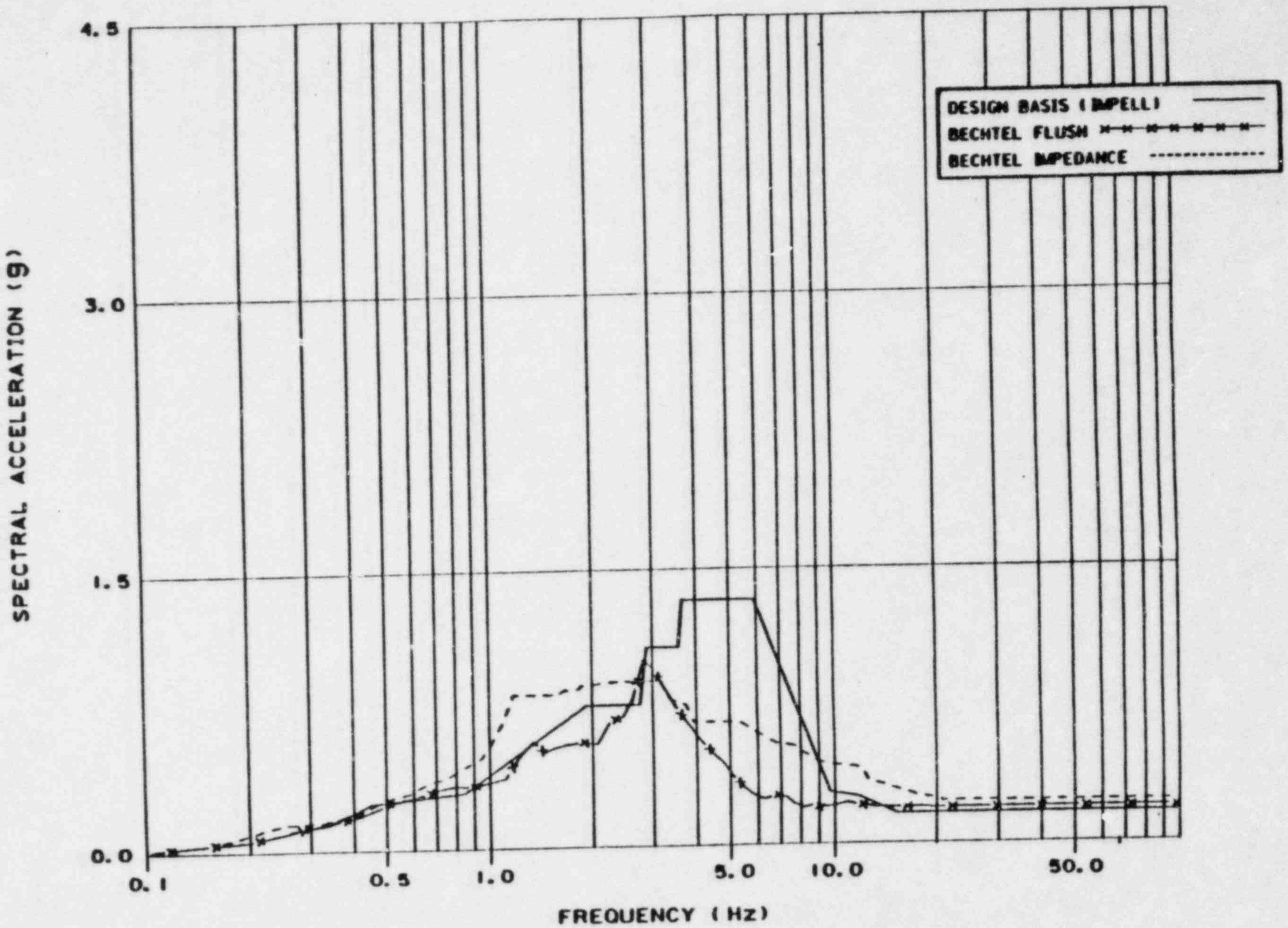


FIGURE A-13-5

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 54'-0",
E-W, SSE, 2% DAMPING

HOPE CREEK UNIT I
CAD, LTA, REV I

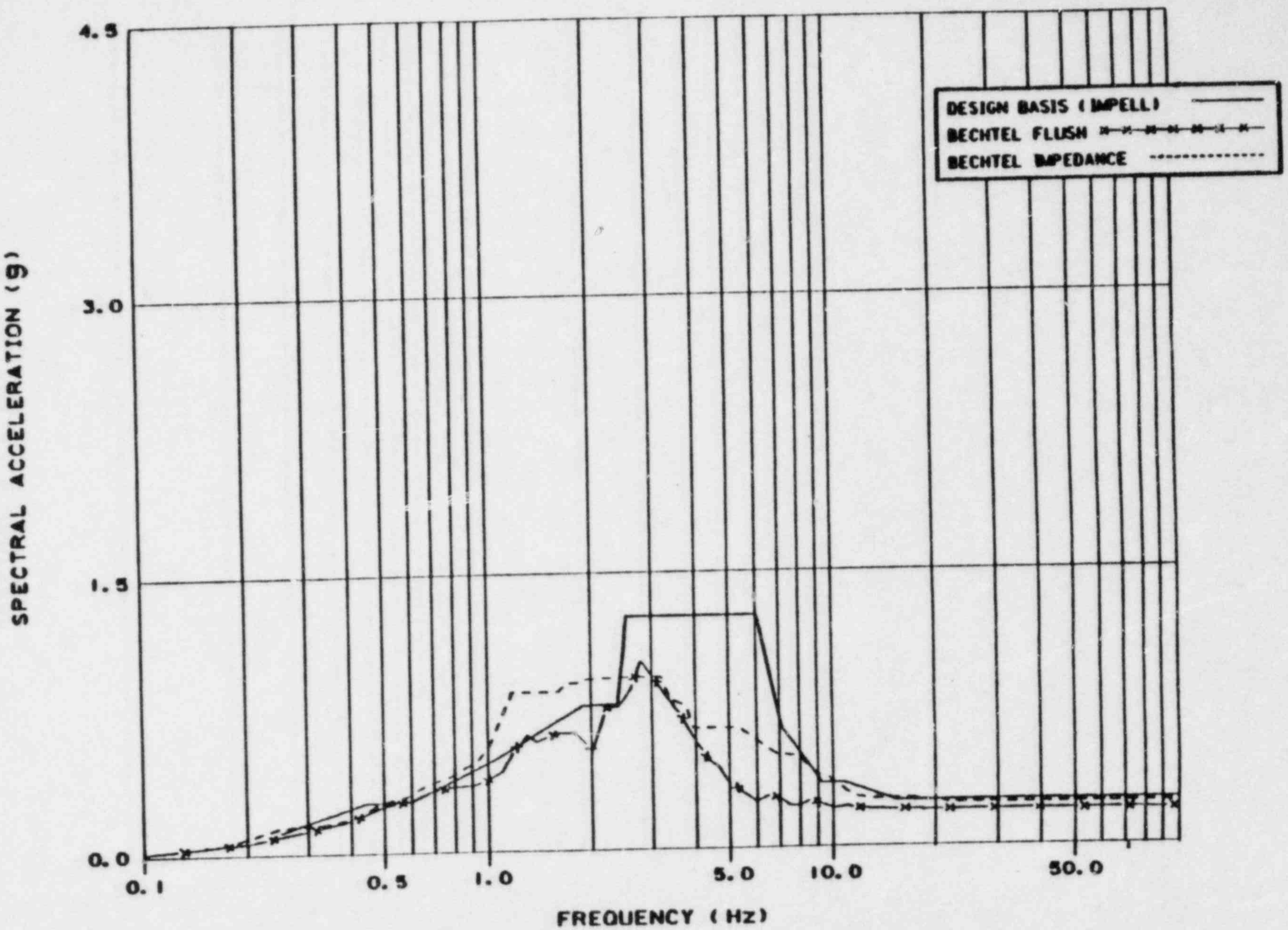


FIGURE A-13-6

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 102' -0",
E-W, SSE, 2% DAMPING

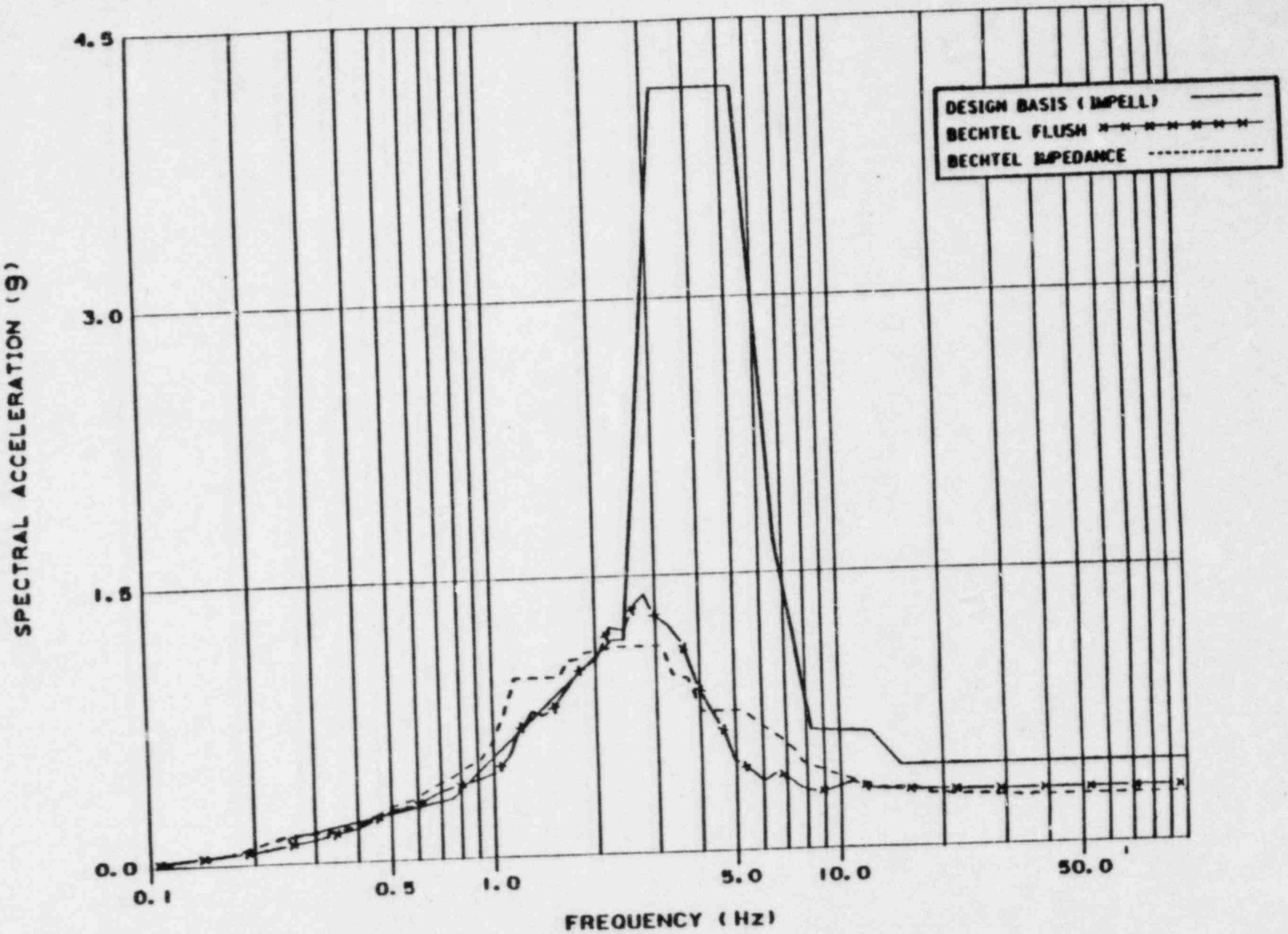


FIGURE A-13-7

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 201'-0",
E-W, SSE, 2% DAMPING

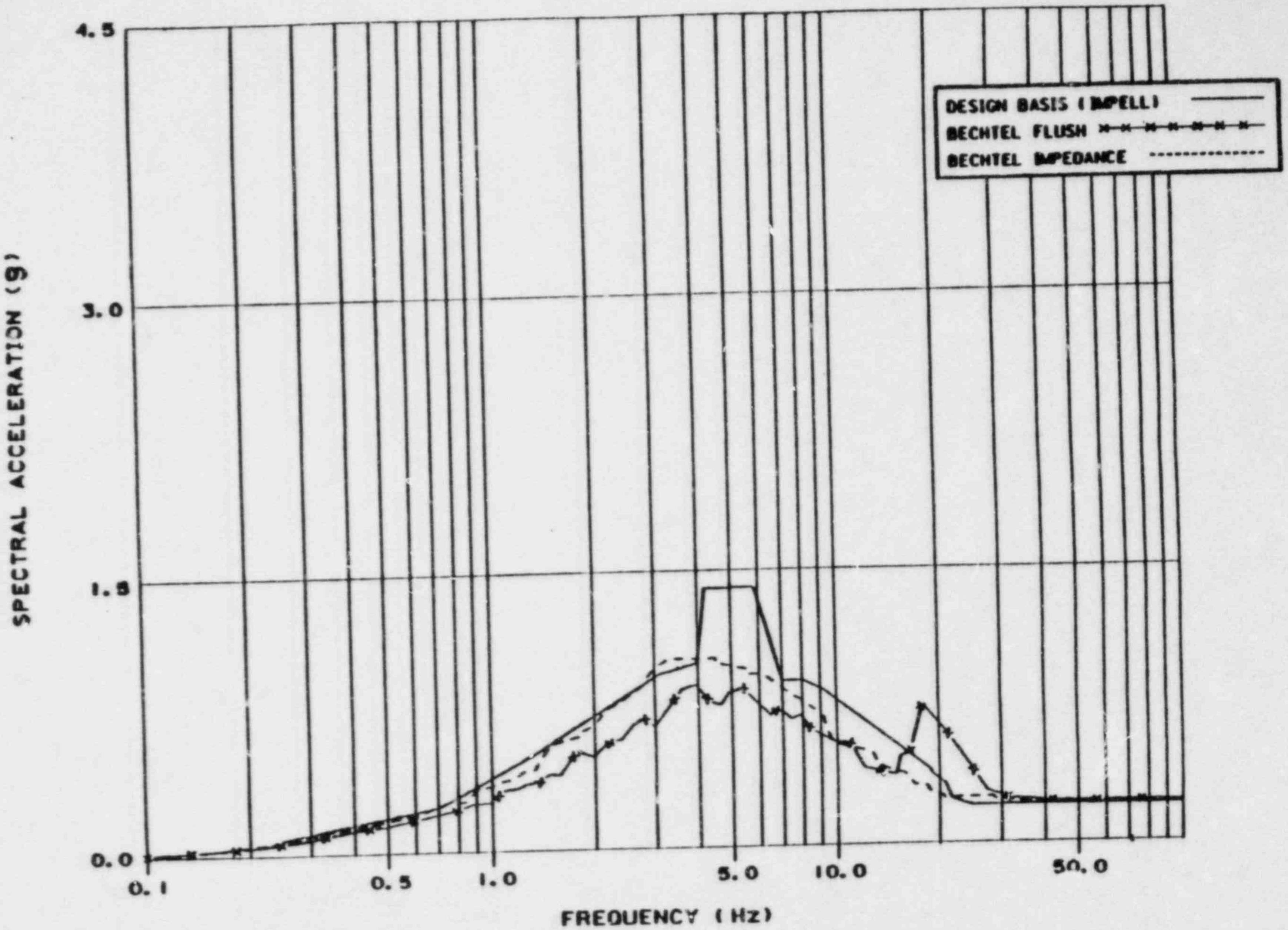


FIGURE A-13-8

RESPONSE SPECTRA COMPARISON,
UNIT 1 REACTOR BUILDING AT ELEV. 54'-0",
VERTICAL, SSE, 2% DAMPING

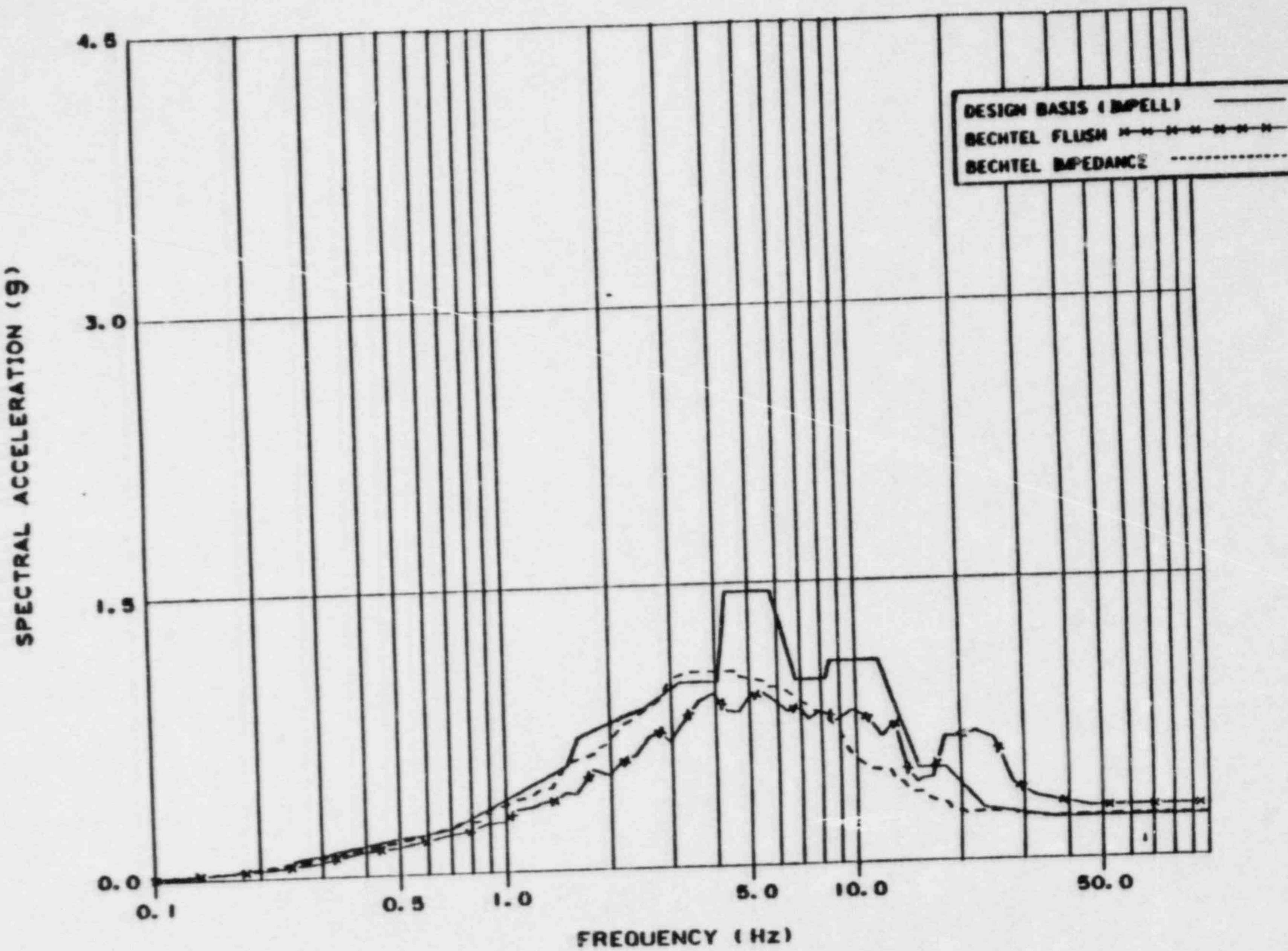


FIGURE A-13-9

RESPONSE SPECTRA COMPARISON,
UNIT 1 REACTOR BUILDING AT ELEV. 102'-0",
VERTICAL SSE, 2% DAMPING

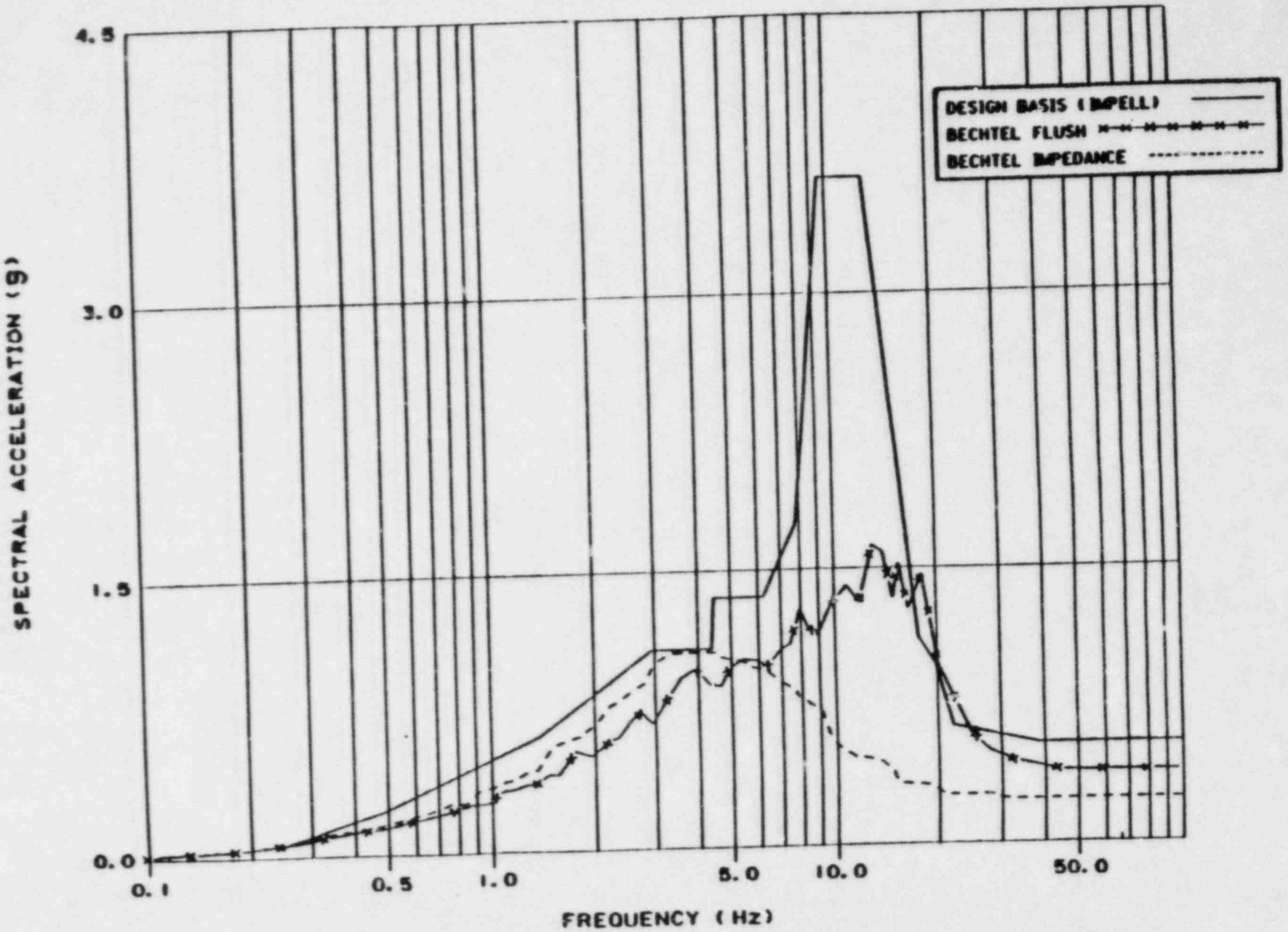


FIGURE A-13-10

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 201'-0",
VERTICAL, SSE, 2% DAMPING

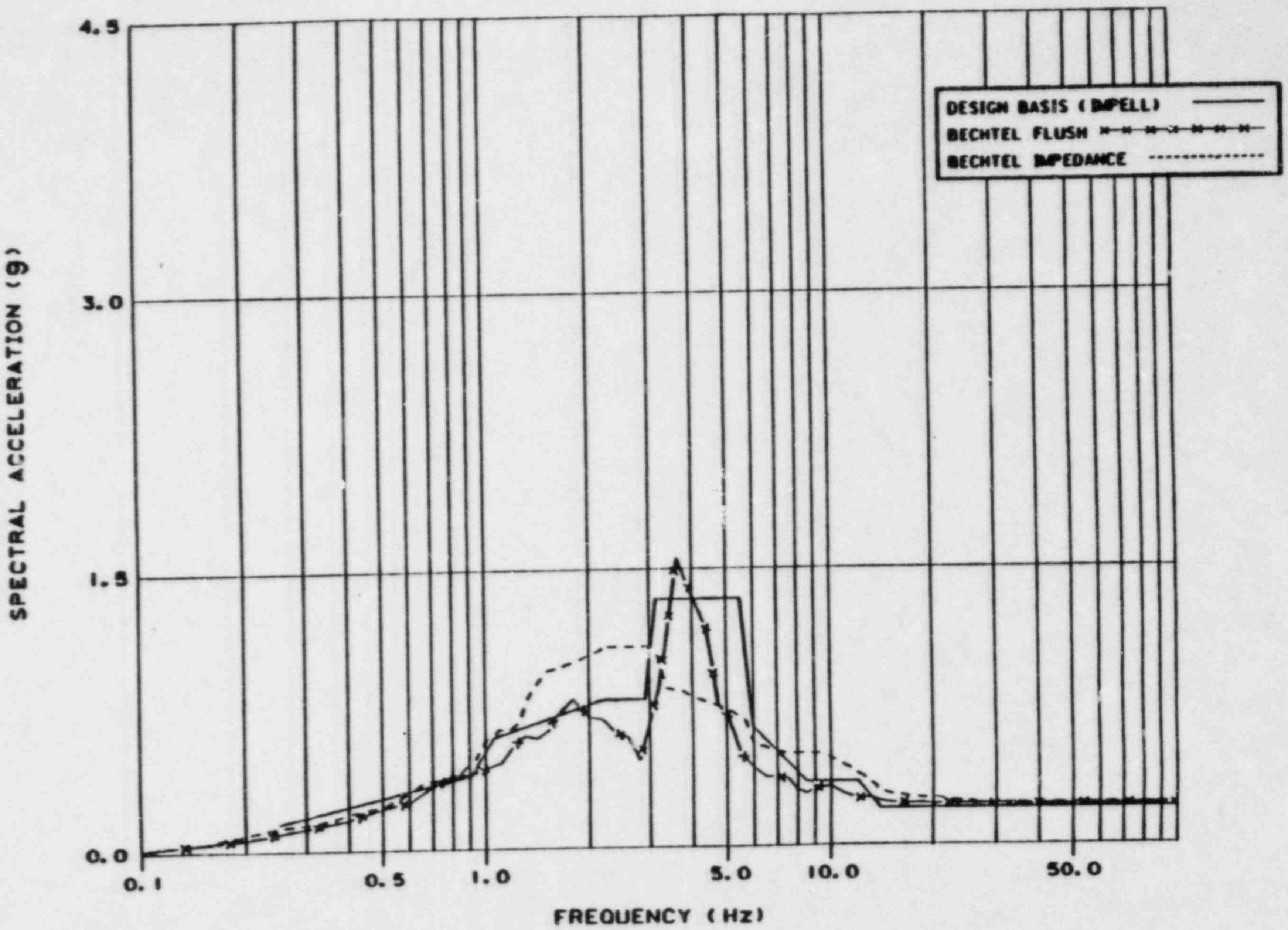


FIGURE A-13-11

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 54'-0",
N-S, SSE, 2% DAMPING

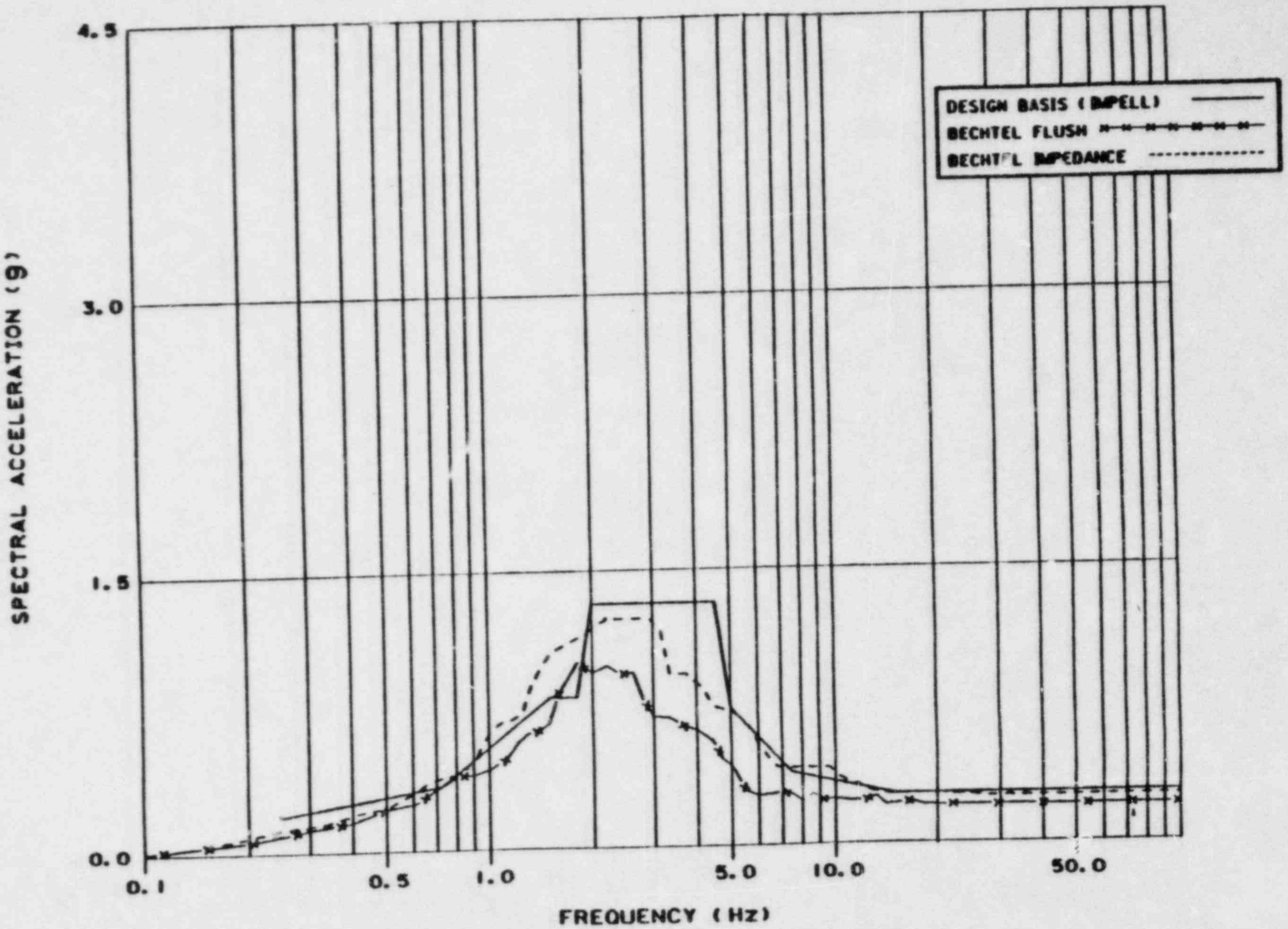


FIGURE A-13-12

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 102'-0",

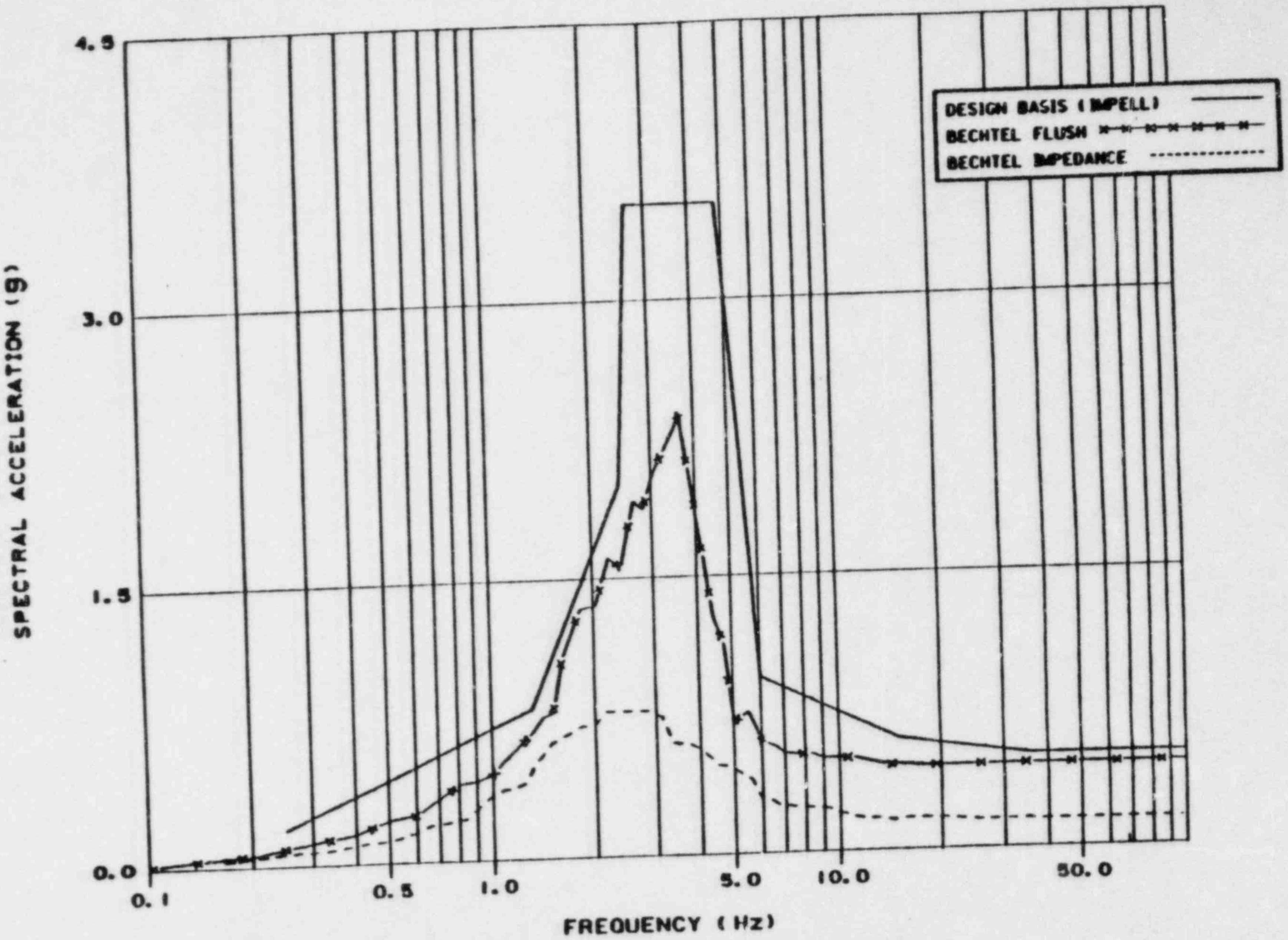


FIGURE A-13-13

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 178'-0",
N-S, SSE, 2% DAMPING

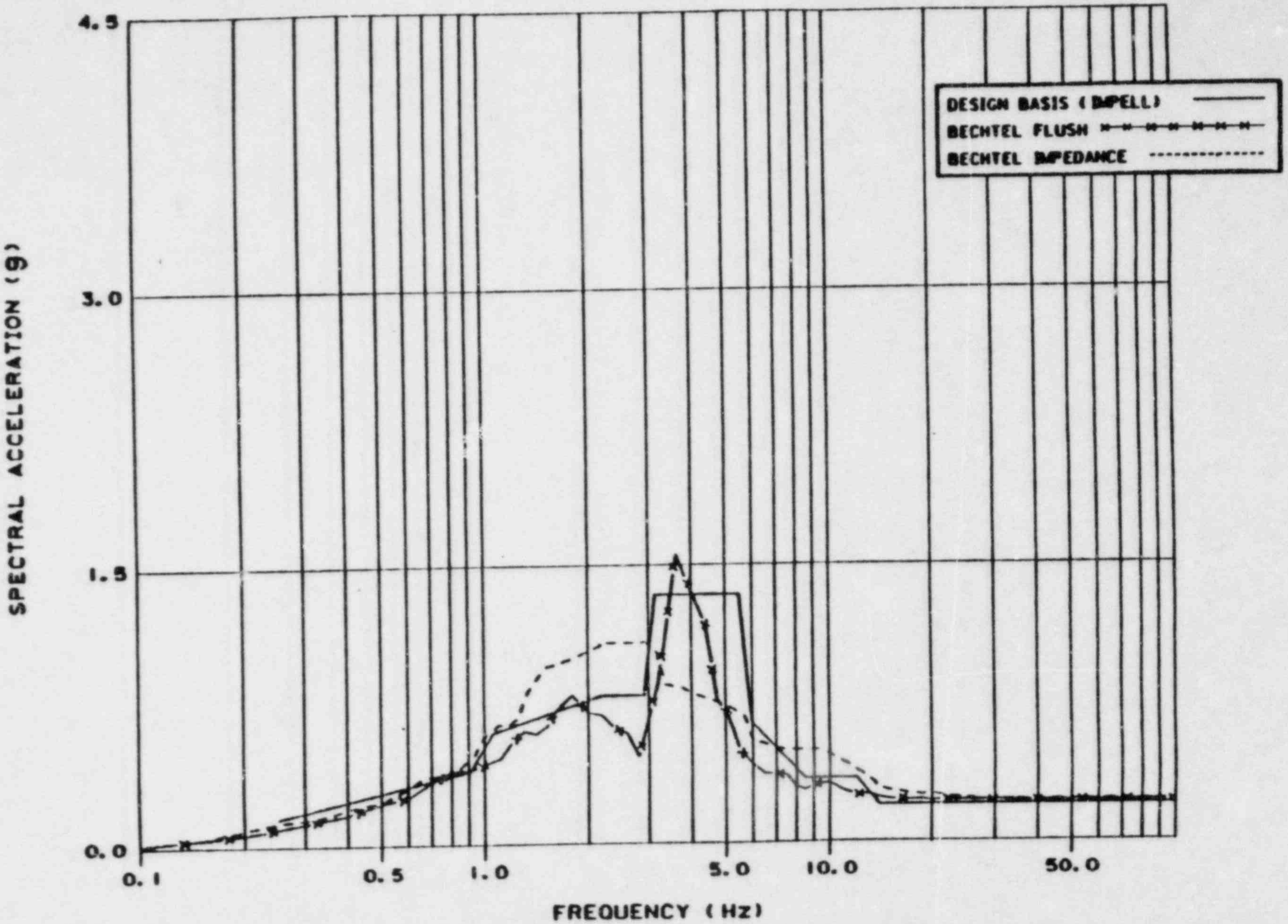


FIGURE A-13-11

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 54'-0",
N-S. SSE. 2% DAMPING

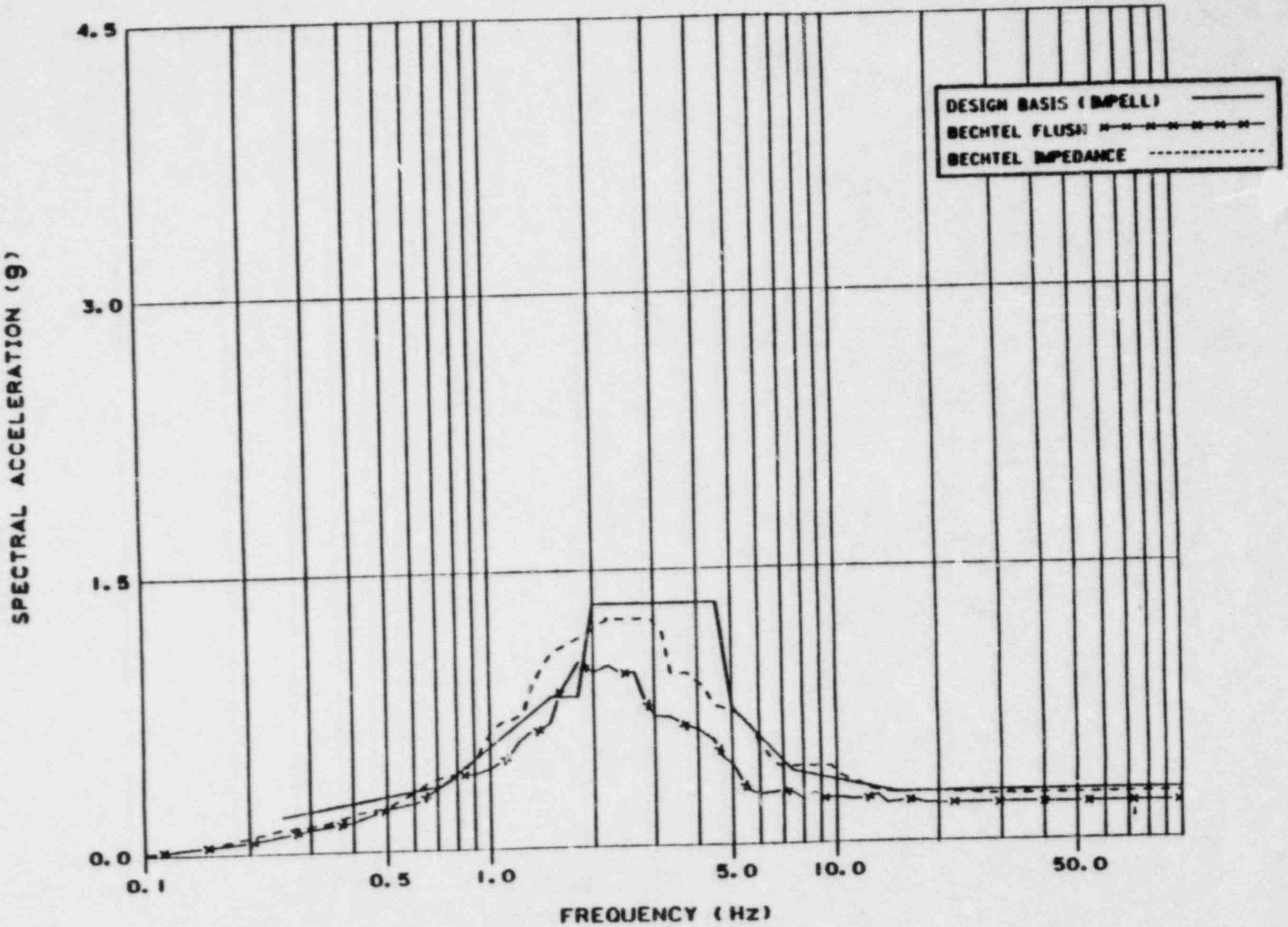


FIGURE A-13-12

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 102'-0",

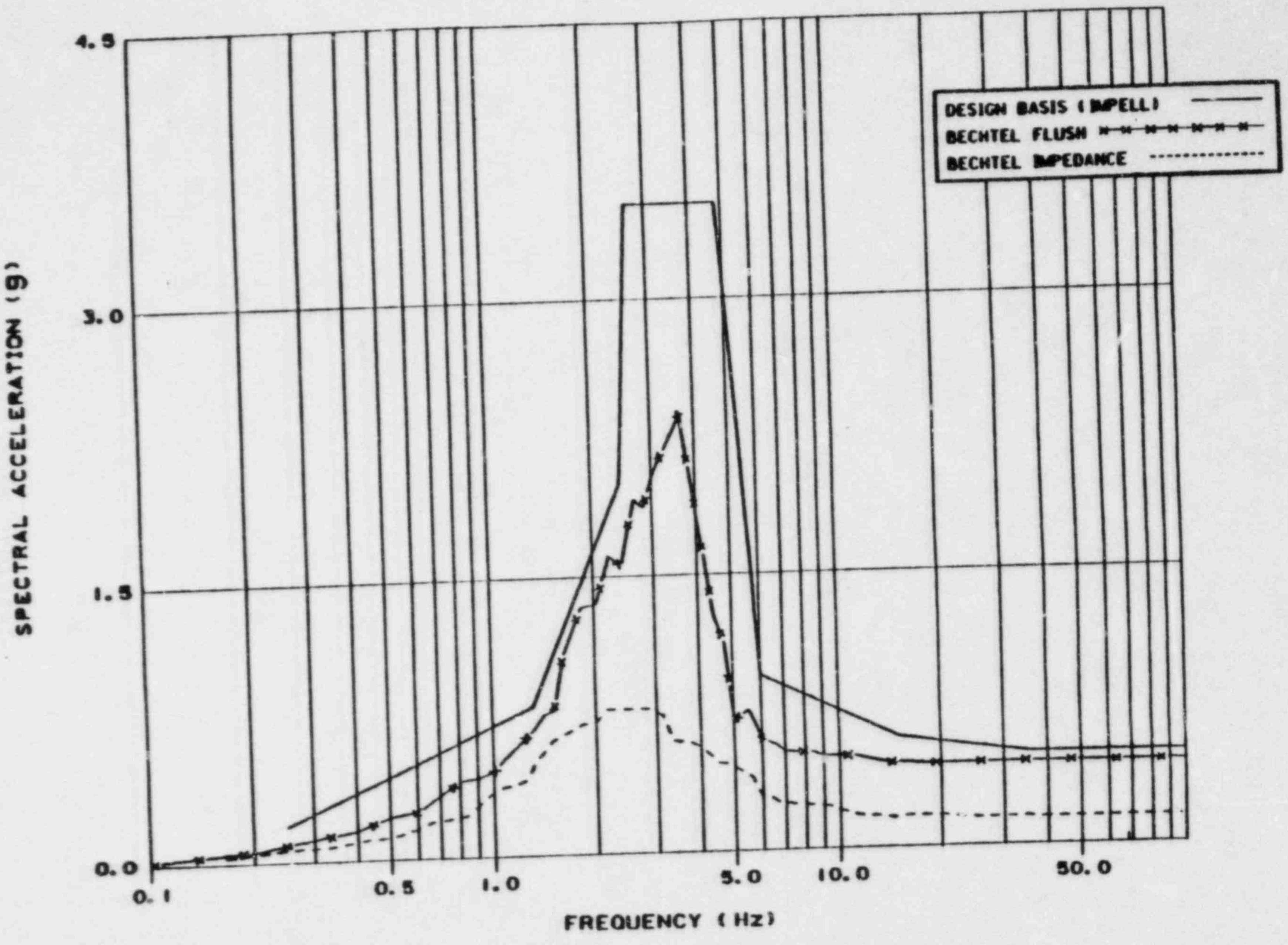


FIGURE A-13-13

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 178'-0",
N-S, SSE, 2% DAMPING

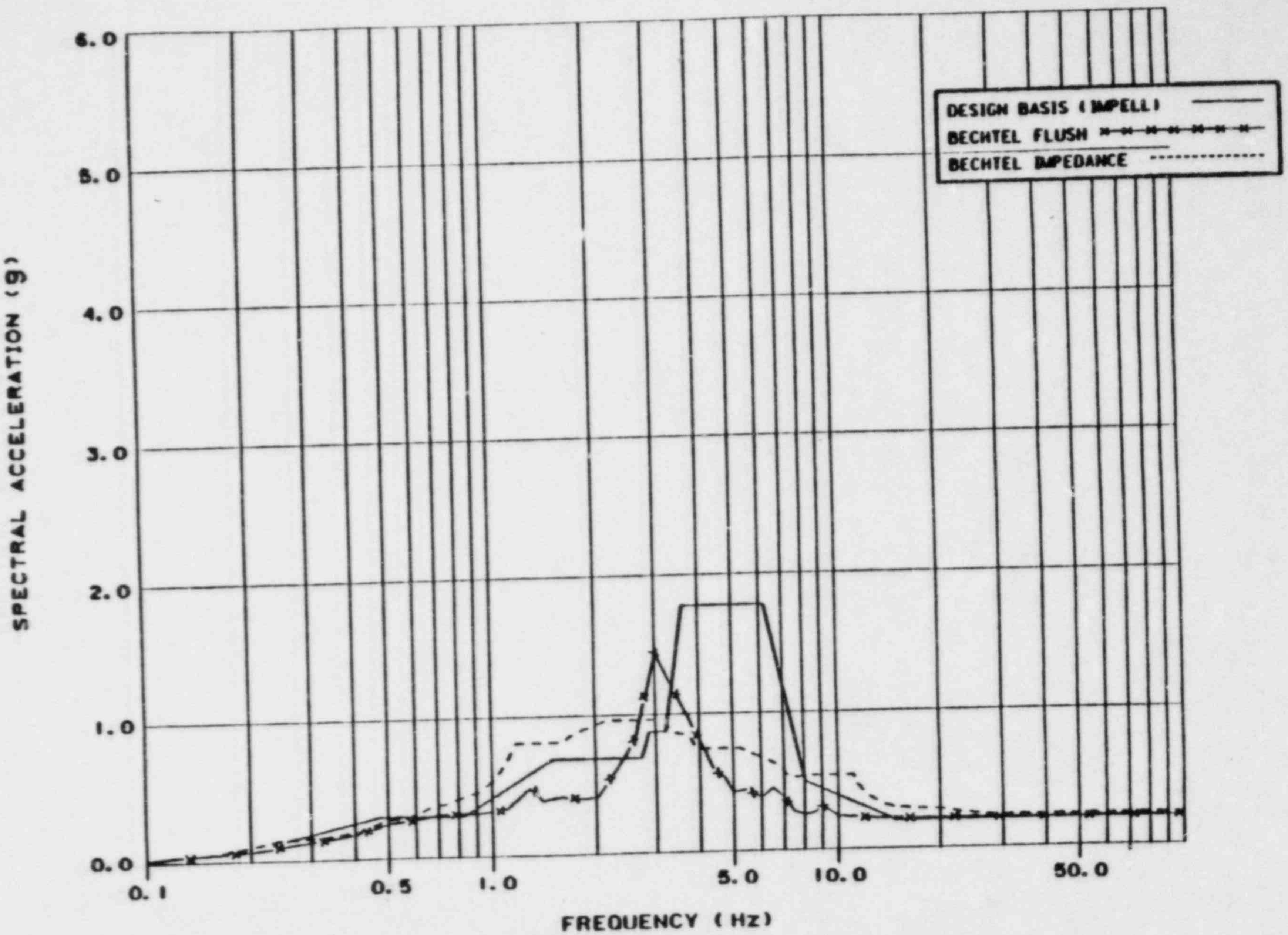


FIGURE A-13-14

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 54' -0",
E-W, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD. L19A, REV 1

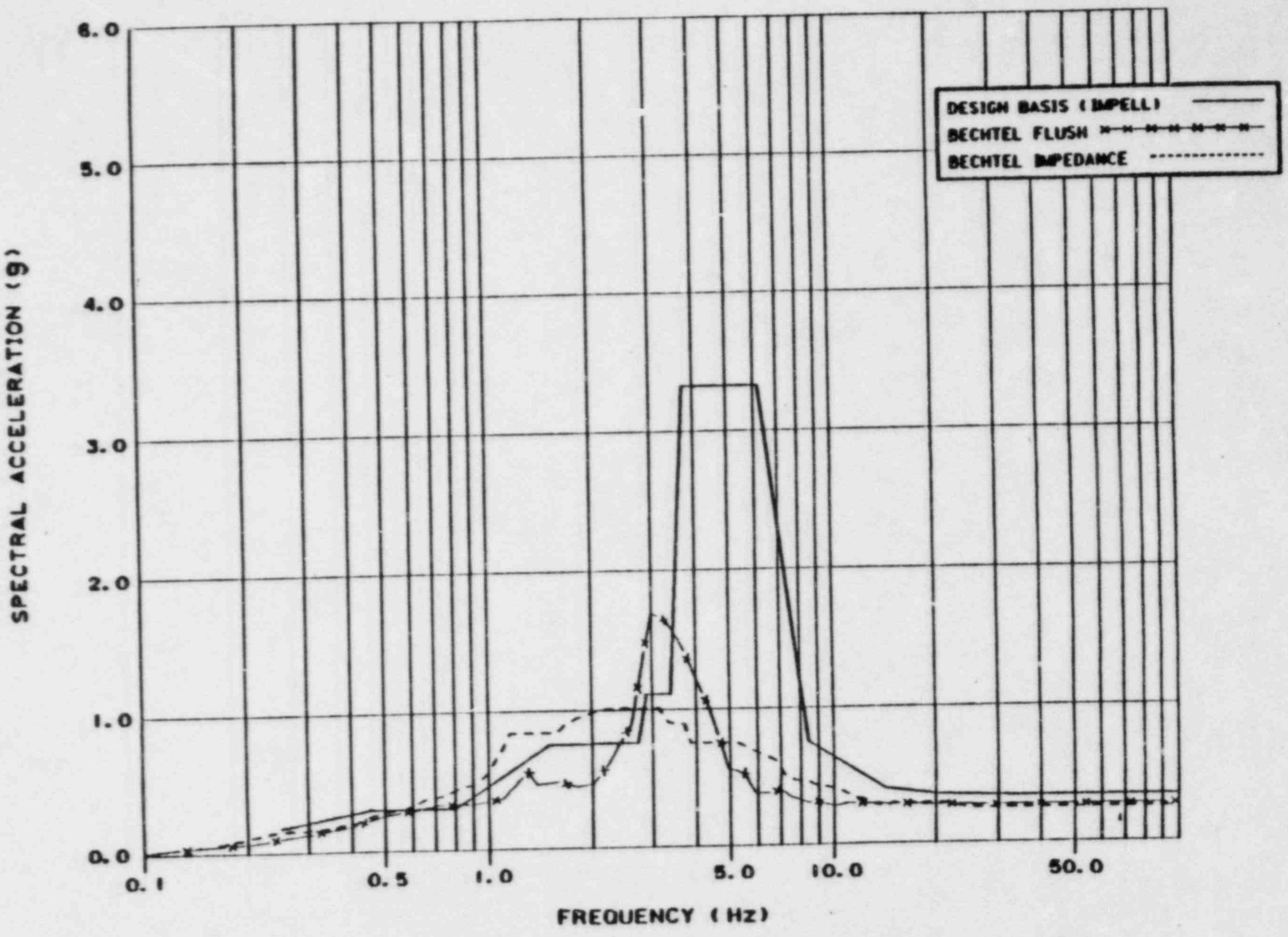


FIGURE A-13-15

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 102' -0",

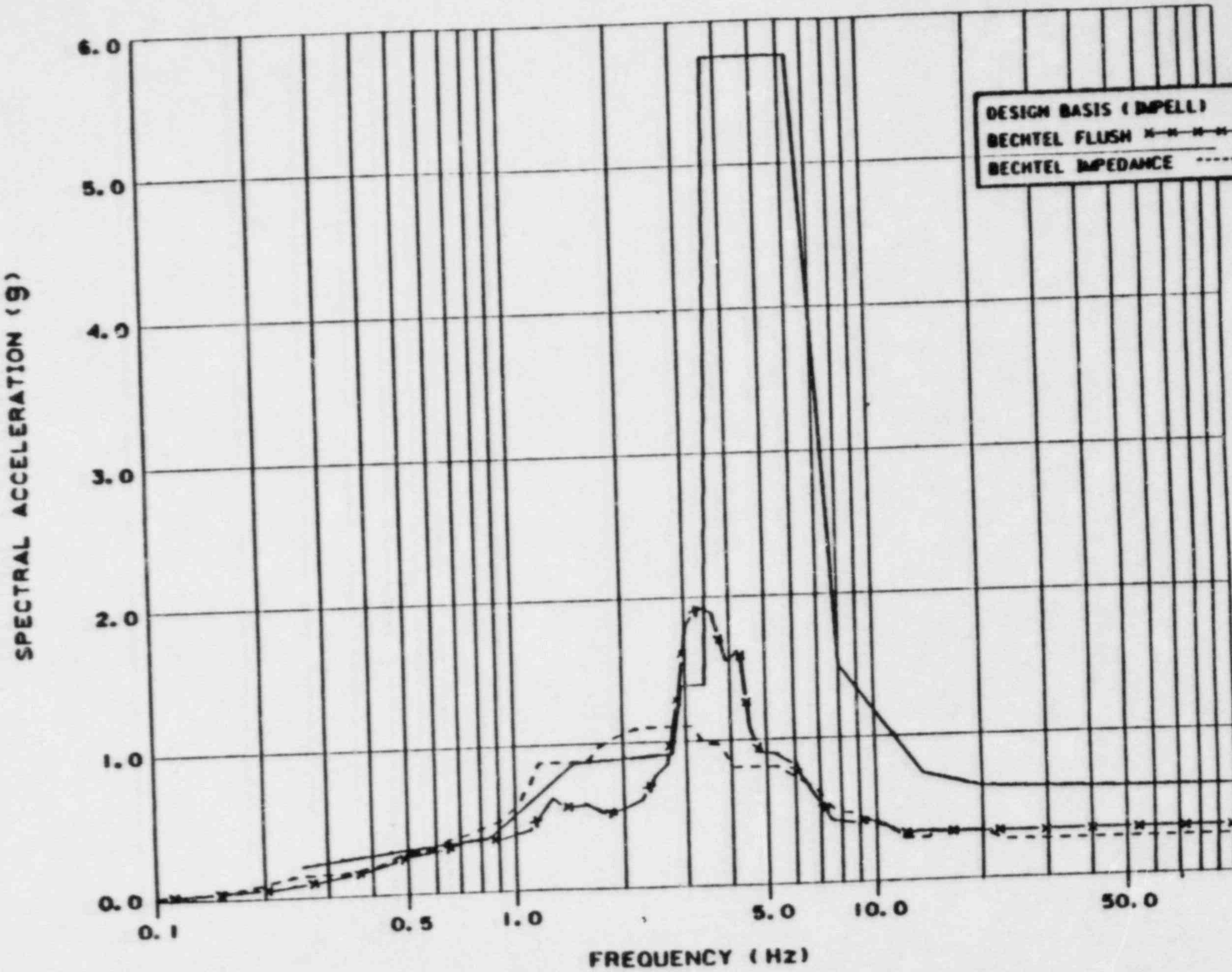


FIGURE A-13-16

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 178'-0",
E-W, SSE, 2% DAMPING

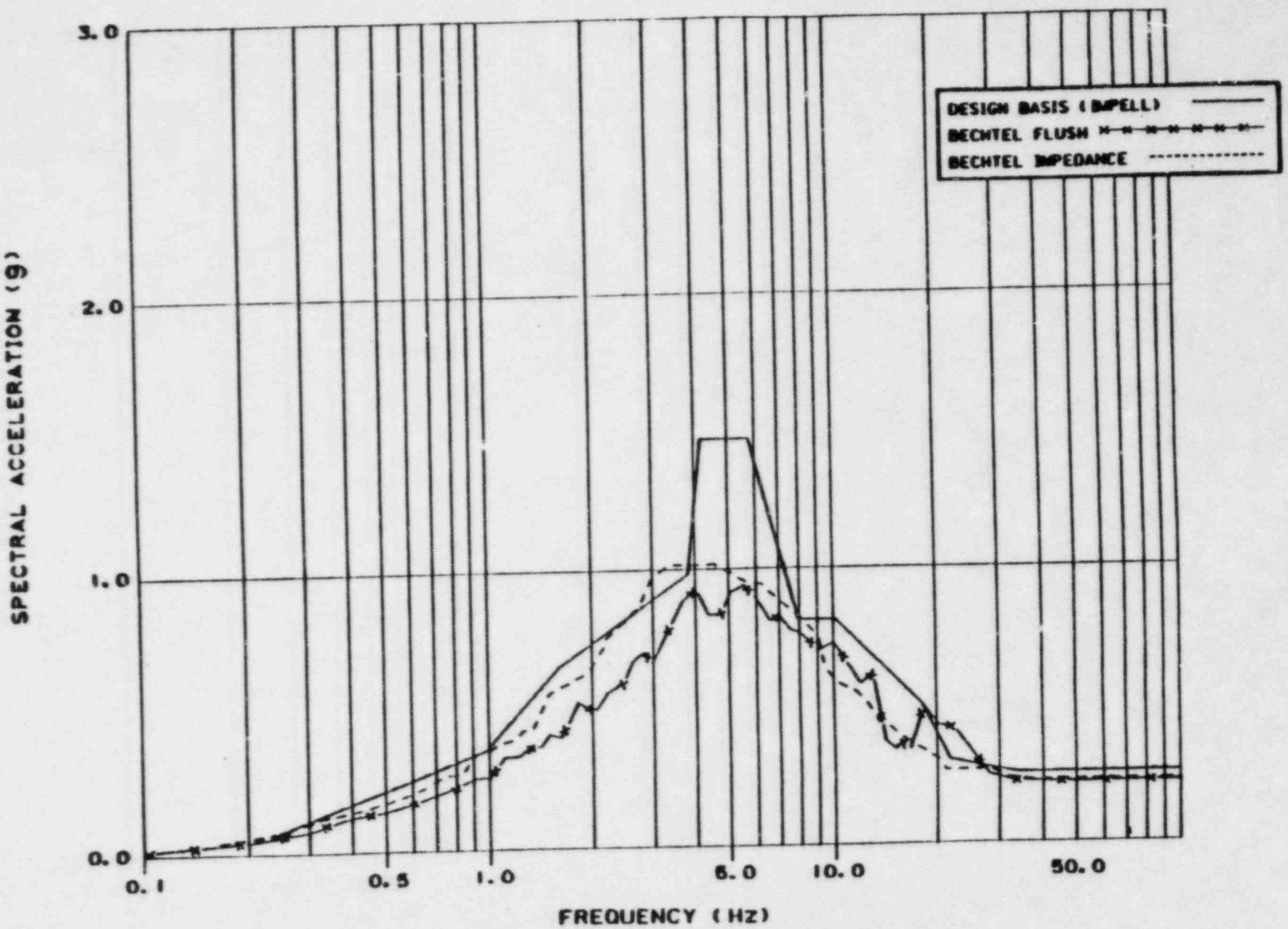


FIGURE A-13-17

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 54' -0",
VERTICAL, SSE, 2% DAMPING

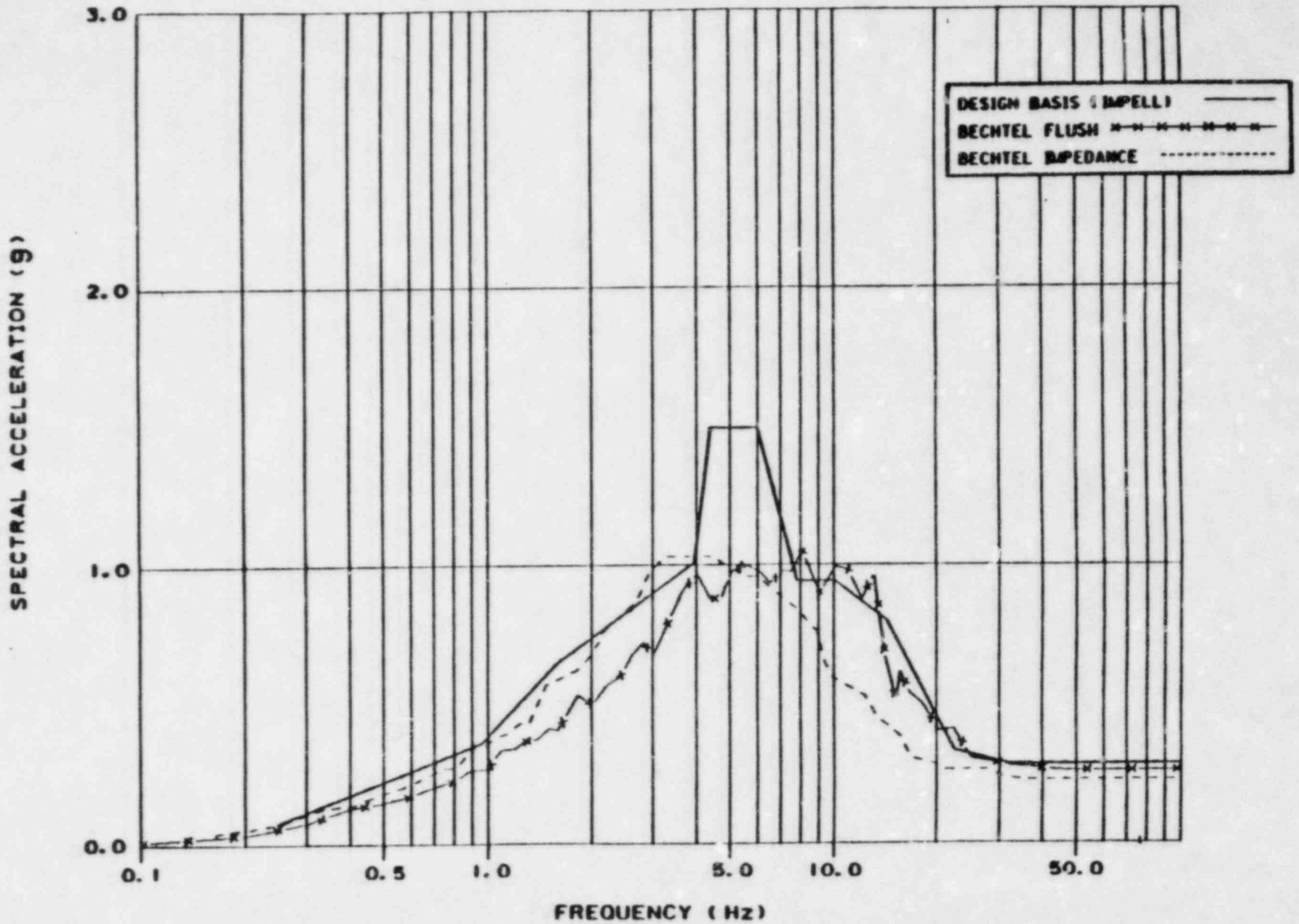


FIGURE A-13-18

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 102'-0",
VERTICAL, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD. L32A, REV 1
04/20/84

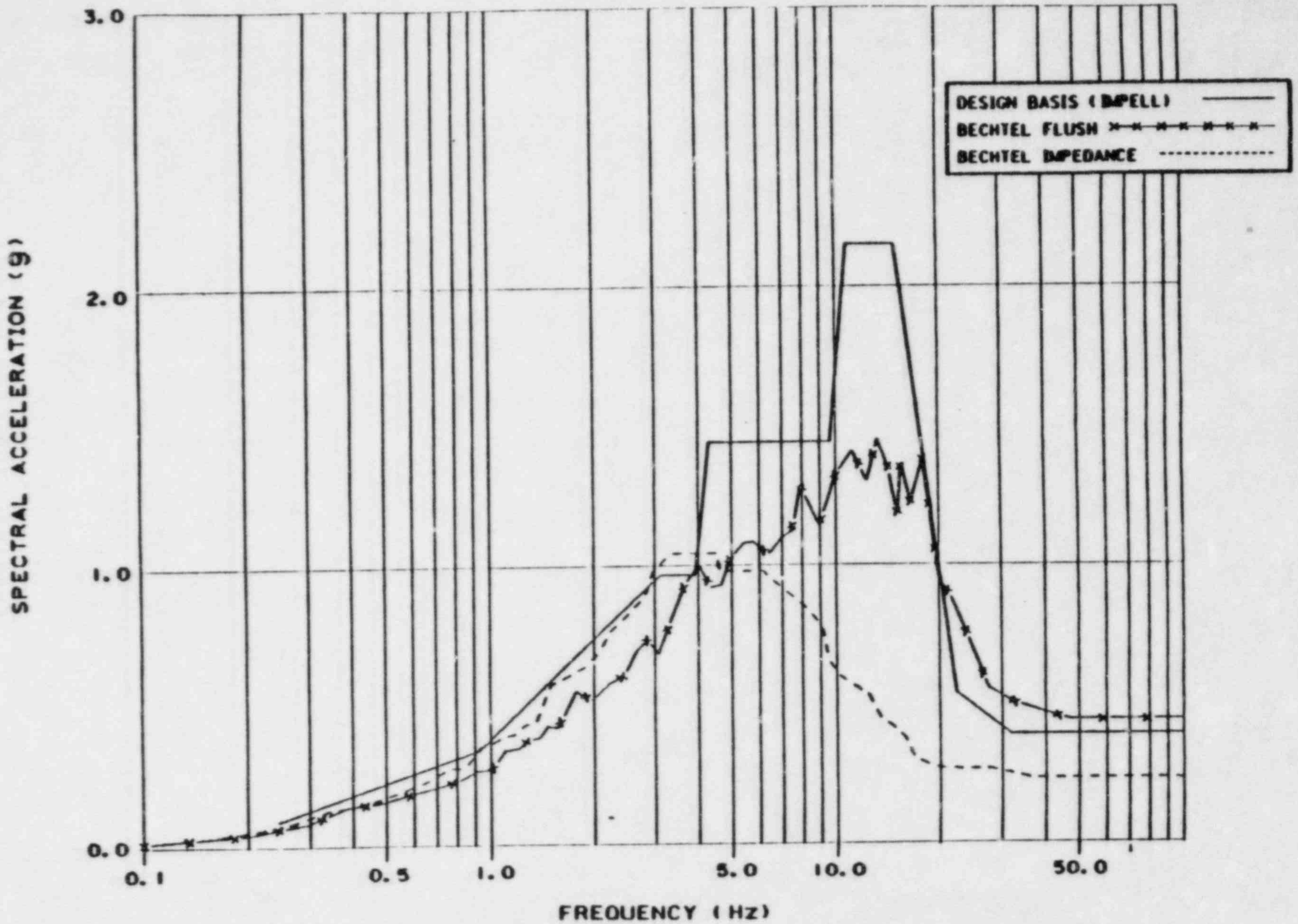


FIGURE A-13-19

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 178'-0",
VERTICAL, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L33A, REV 1
04/20/84

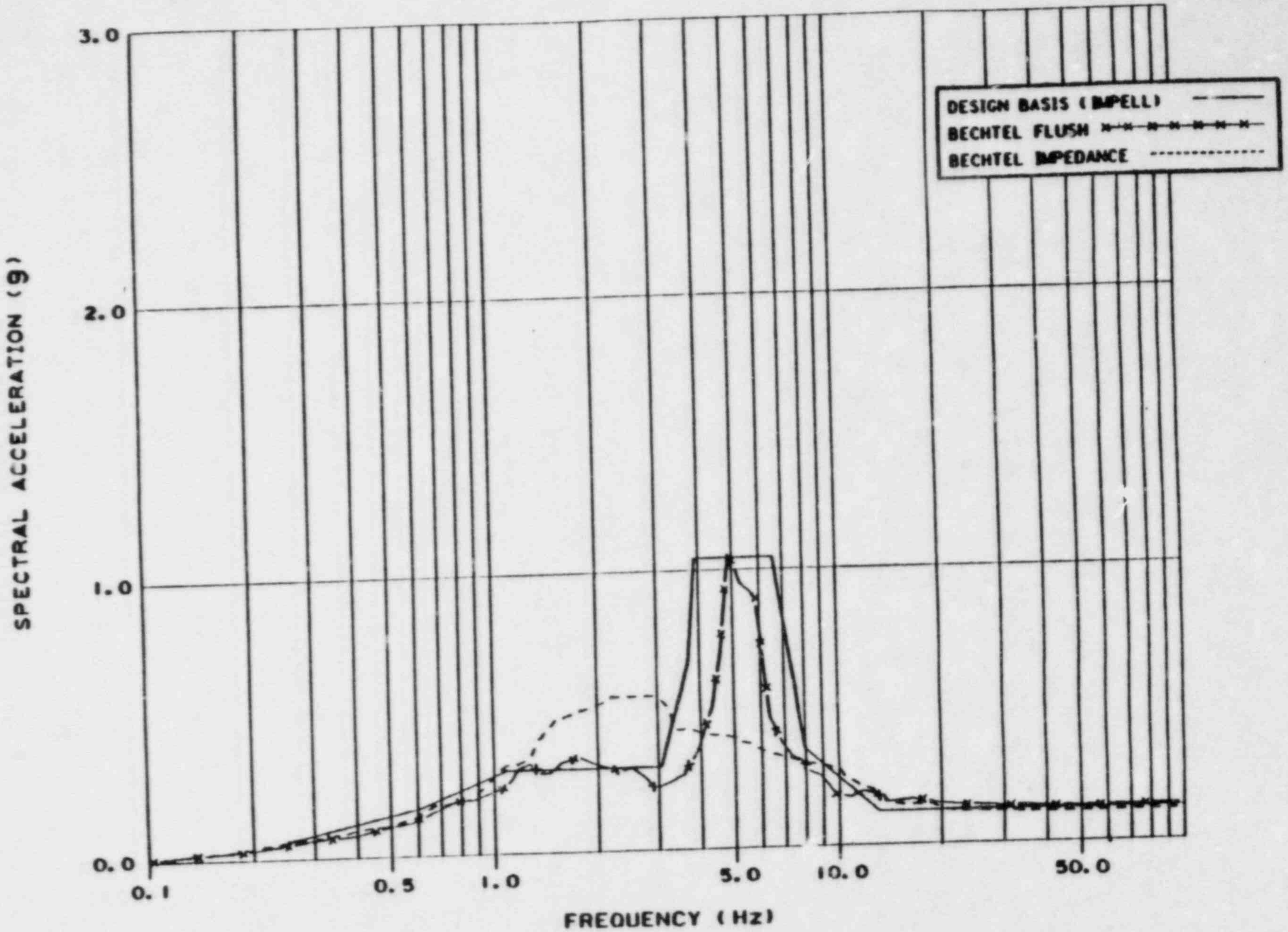


FIGURE A-13-20

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 54'-0"
N-S, OBE, 2% DAMPING.

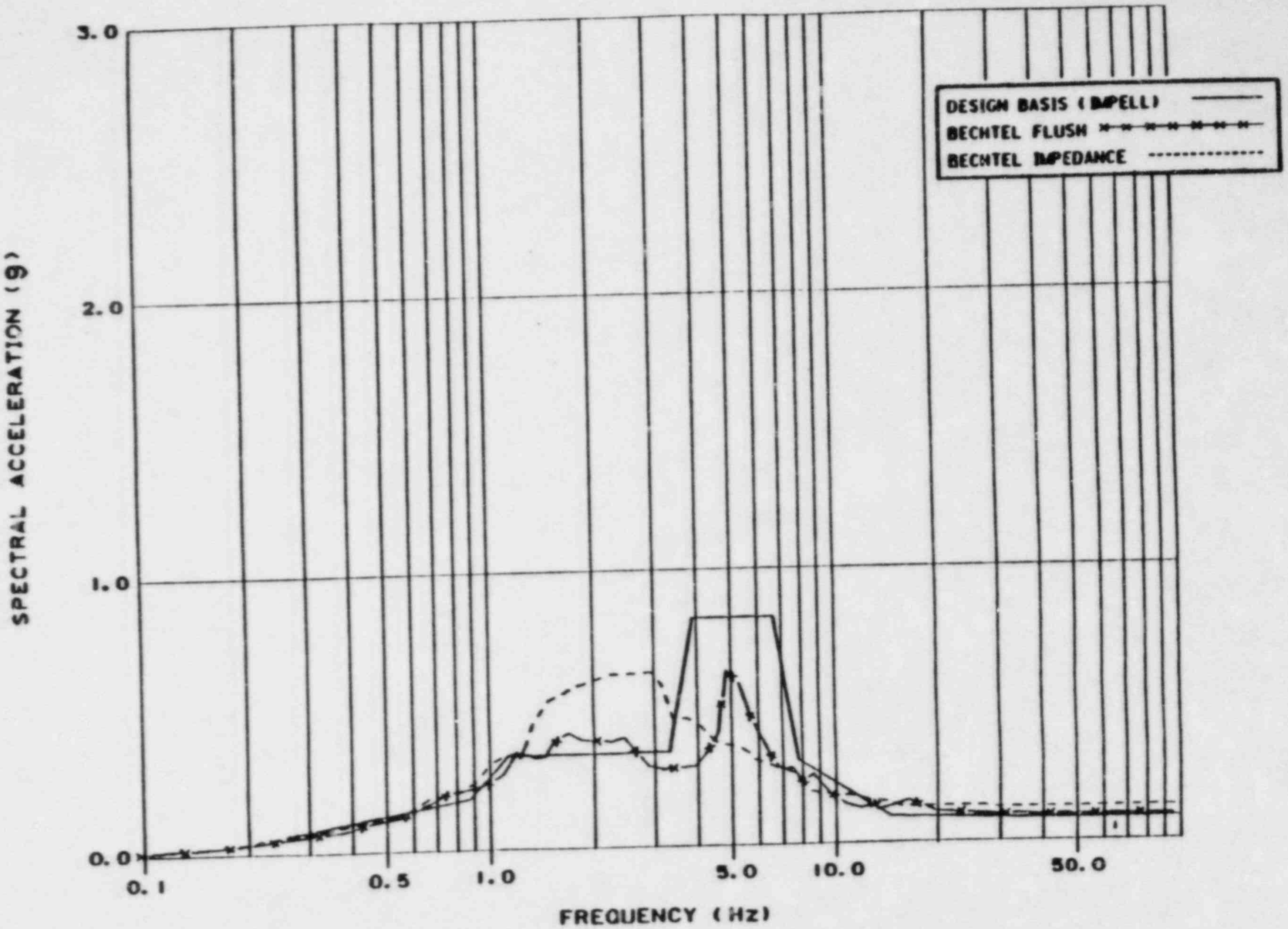


FIGURE A-13-21

RESPONSE SPECTRA COMPARISON.
UNIT I REACTOR BUILDING AT ELEV. 102'-0",
N-S. OBE. 2% DAMPING.

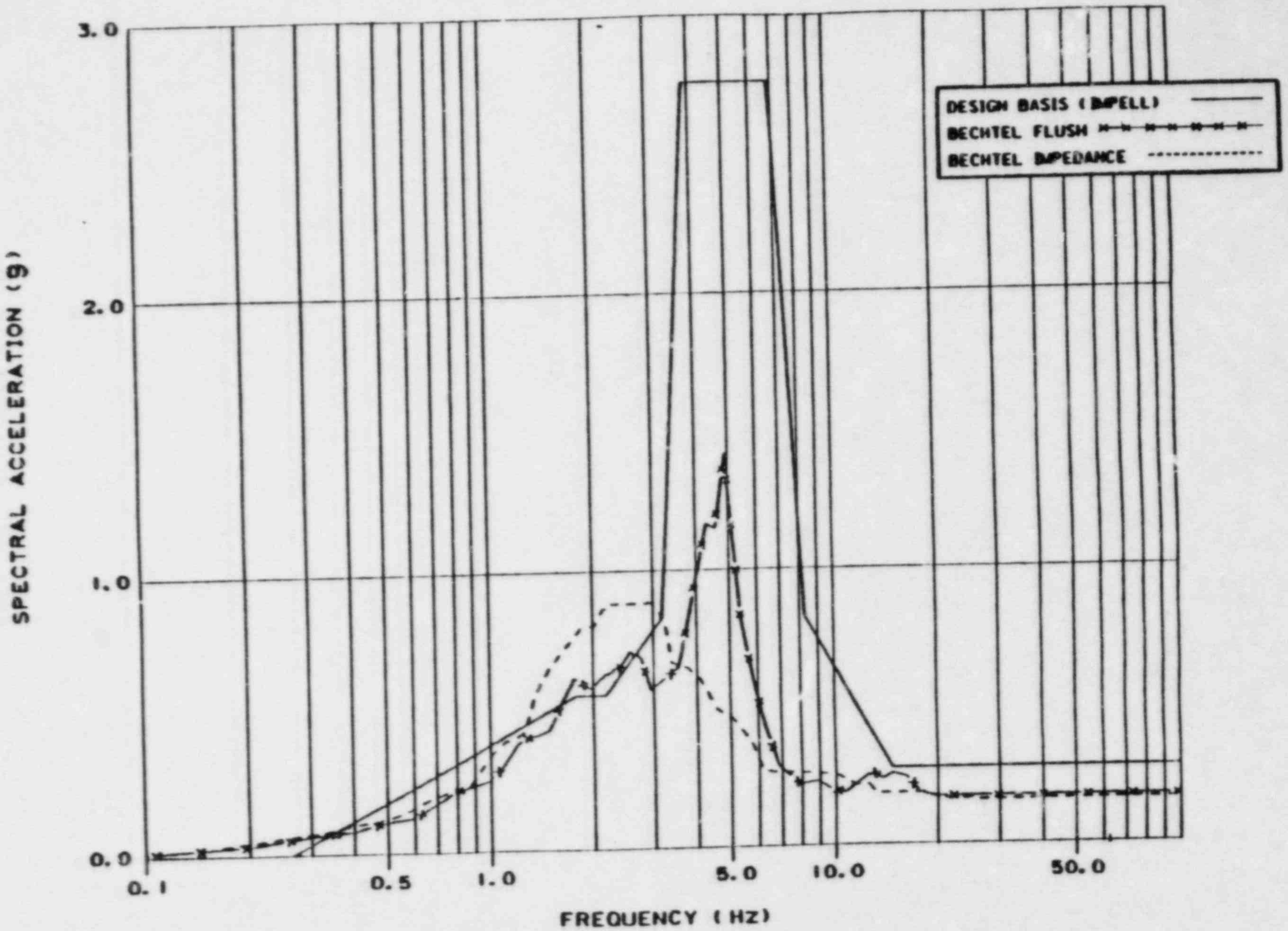


FIGURE A-13-22

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 201'-0",
N-S, OBE, 2% DAMPING

HOPE CREEK UNIT I
CAD. L6A, REV I

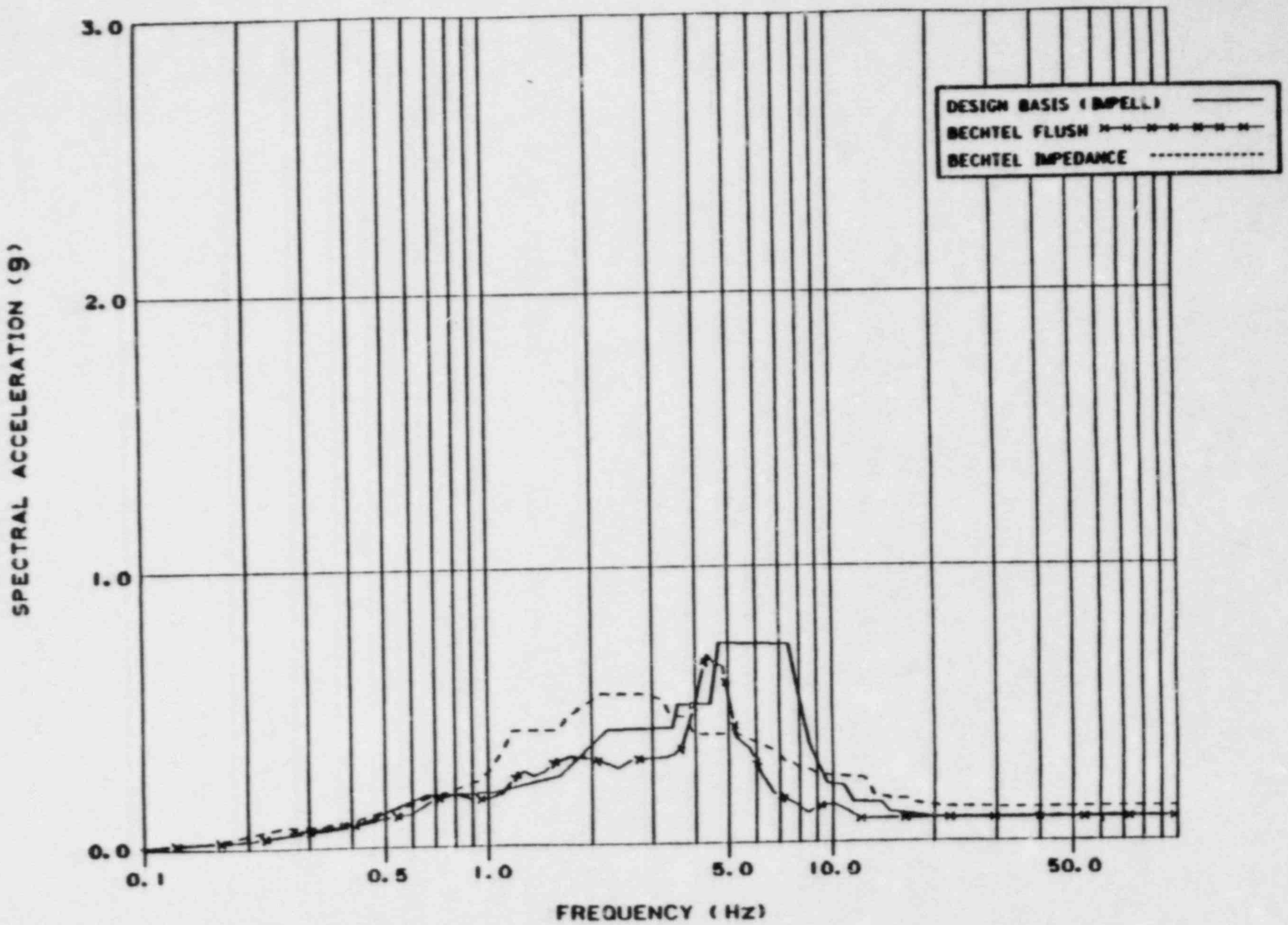


FIGURE A-13-23

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 54' -0",
E-W, OBE, 2% DAMPING

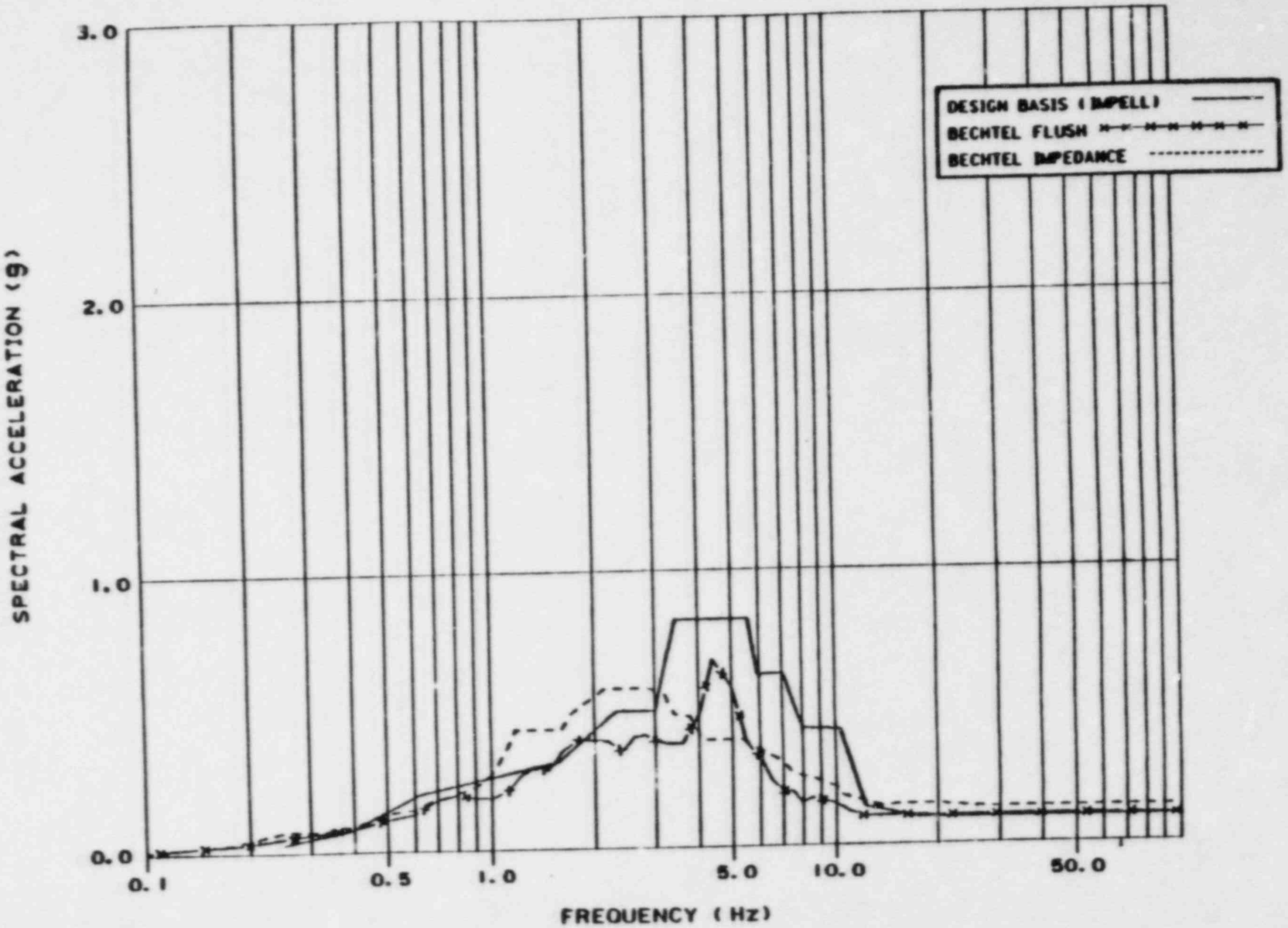


FIGURE A-13-24

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 102'-0",
E-W, OBE, 2% DAMPING

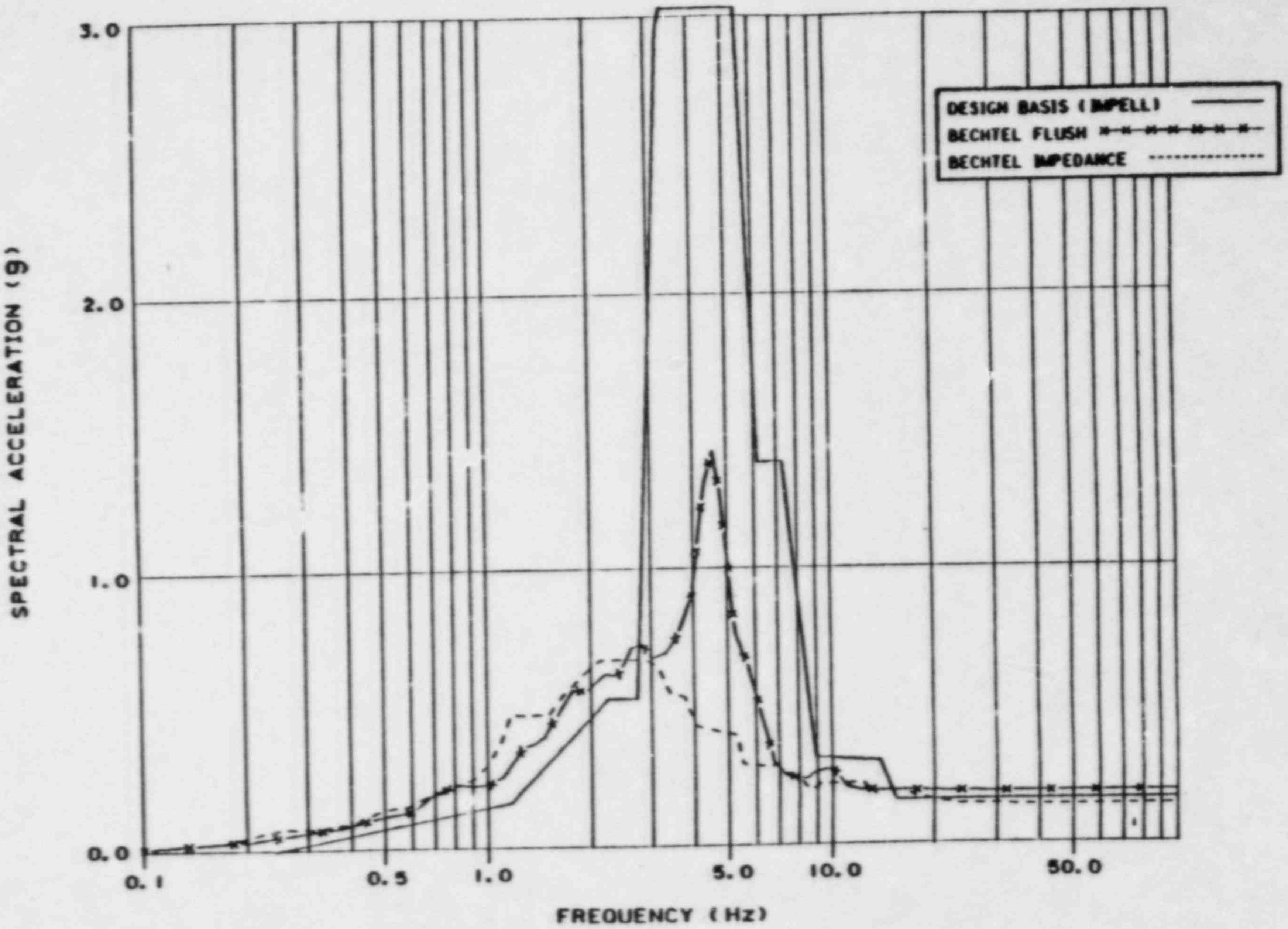


FIGURE A-13-25

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 201'-0",
E-W, OBE, 2% DAMPING

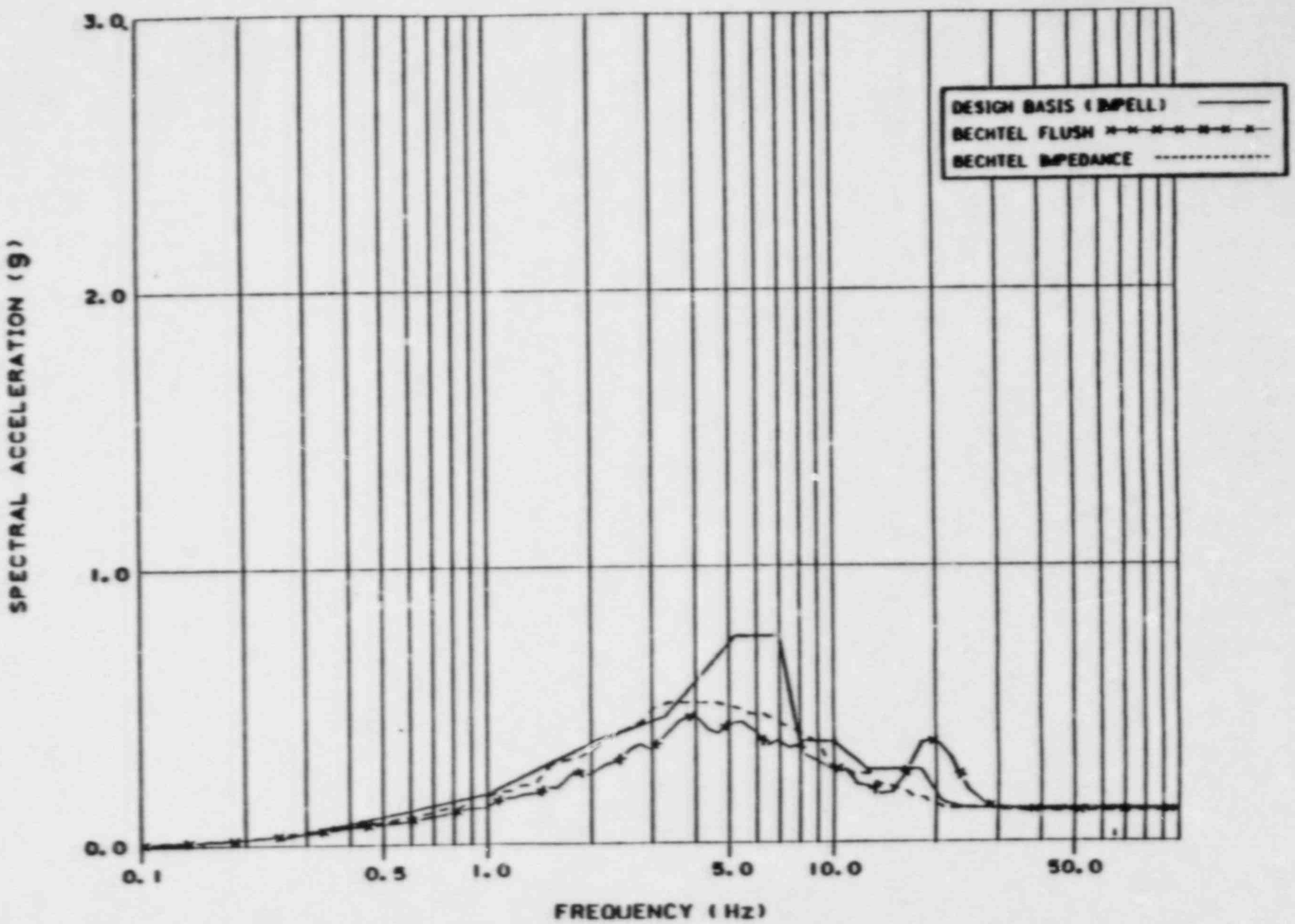


FIGURE A-13-26

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 54' -0",
VERTICAL, OBE, 2% DAMPING

HOPE CREEK UNIT I
CAD, L28A, REV I

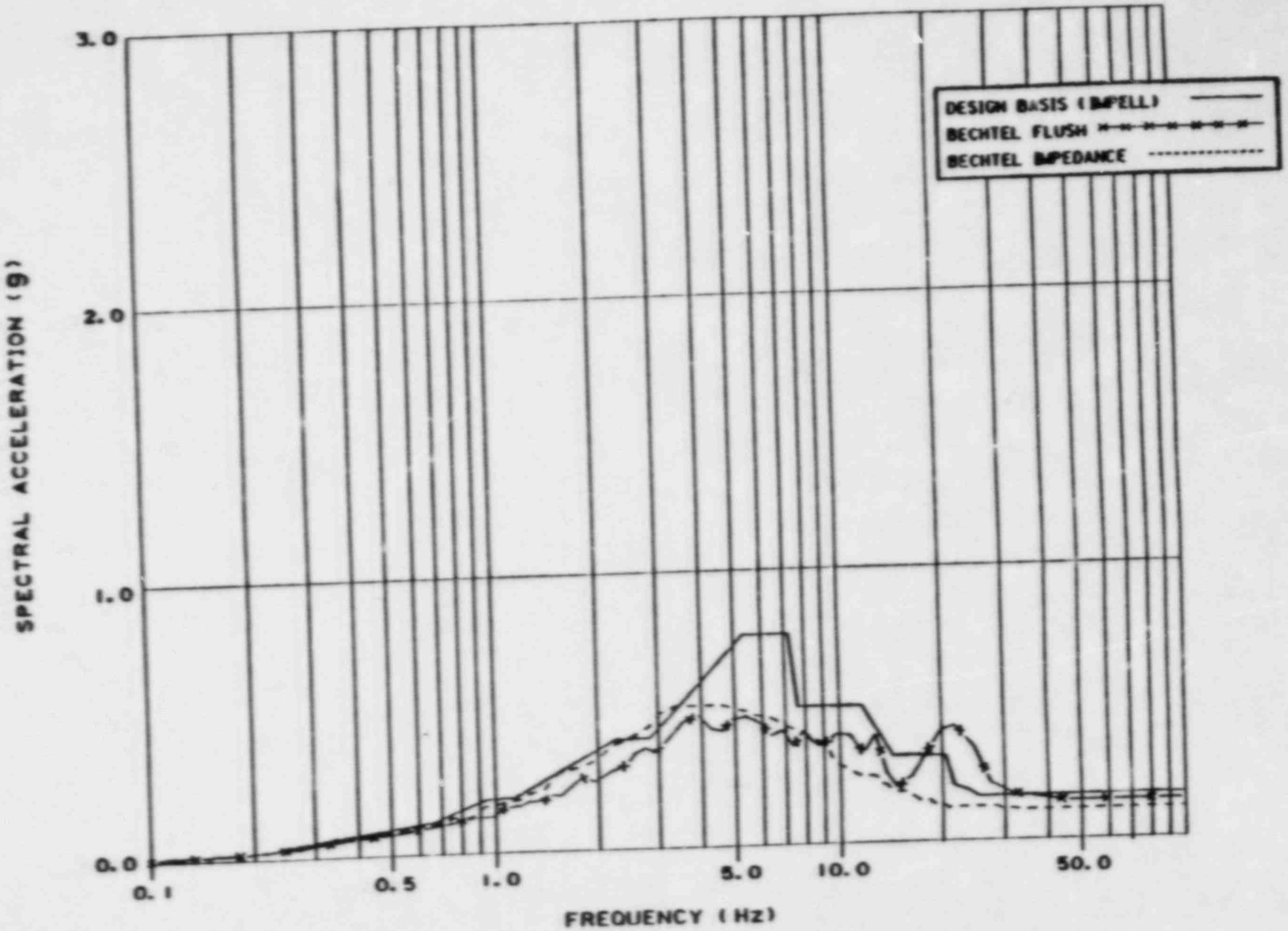


FIGURE A-13-27

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 102' -0",
VERTICAL, OBE, 2% DAMPING

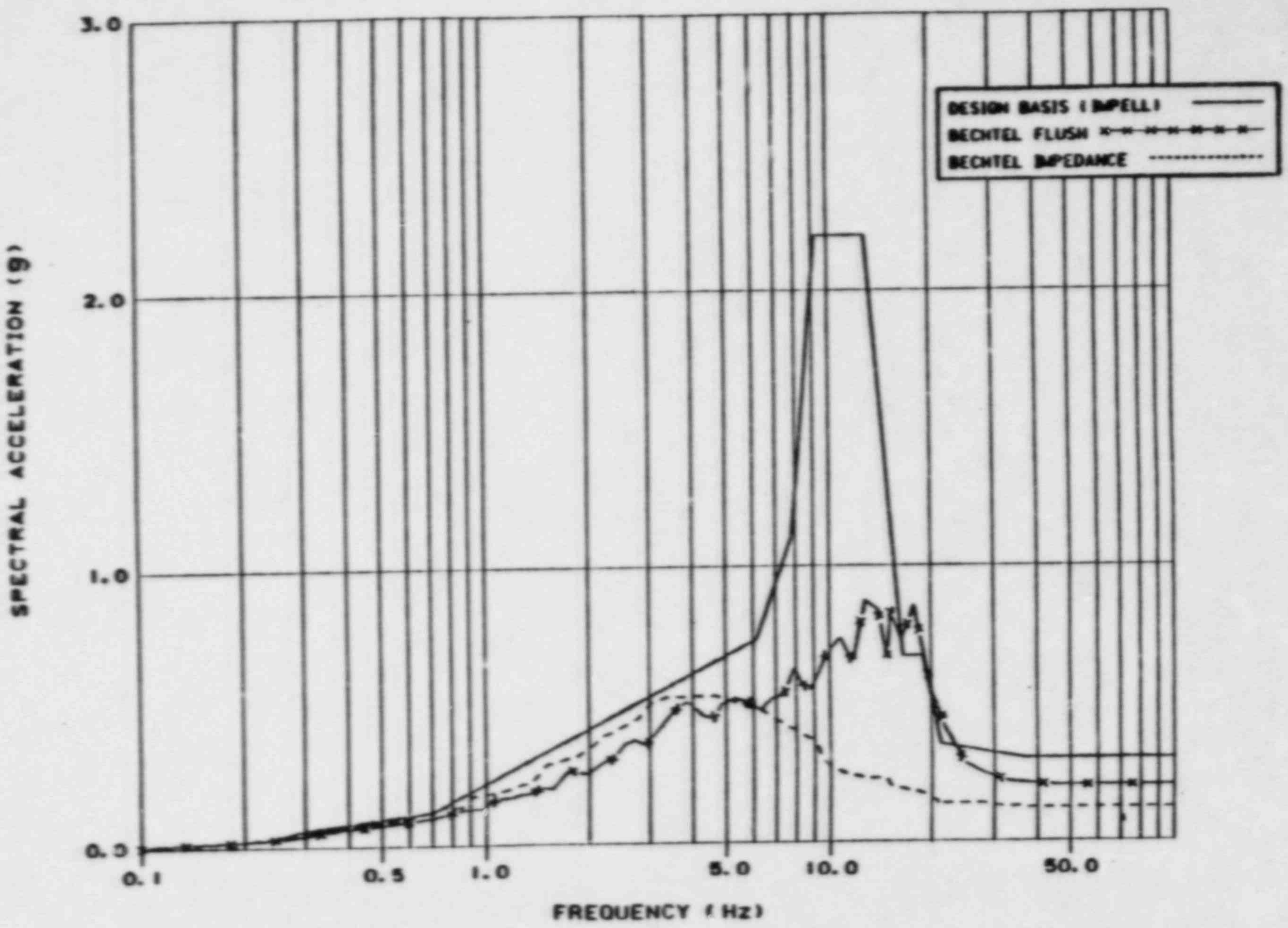


FIGURE A-13-28

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 201'-0",
VERTICAL, OBE, 2% DAMPING

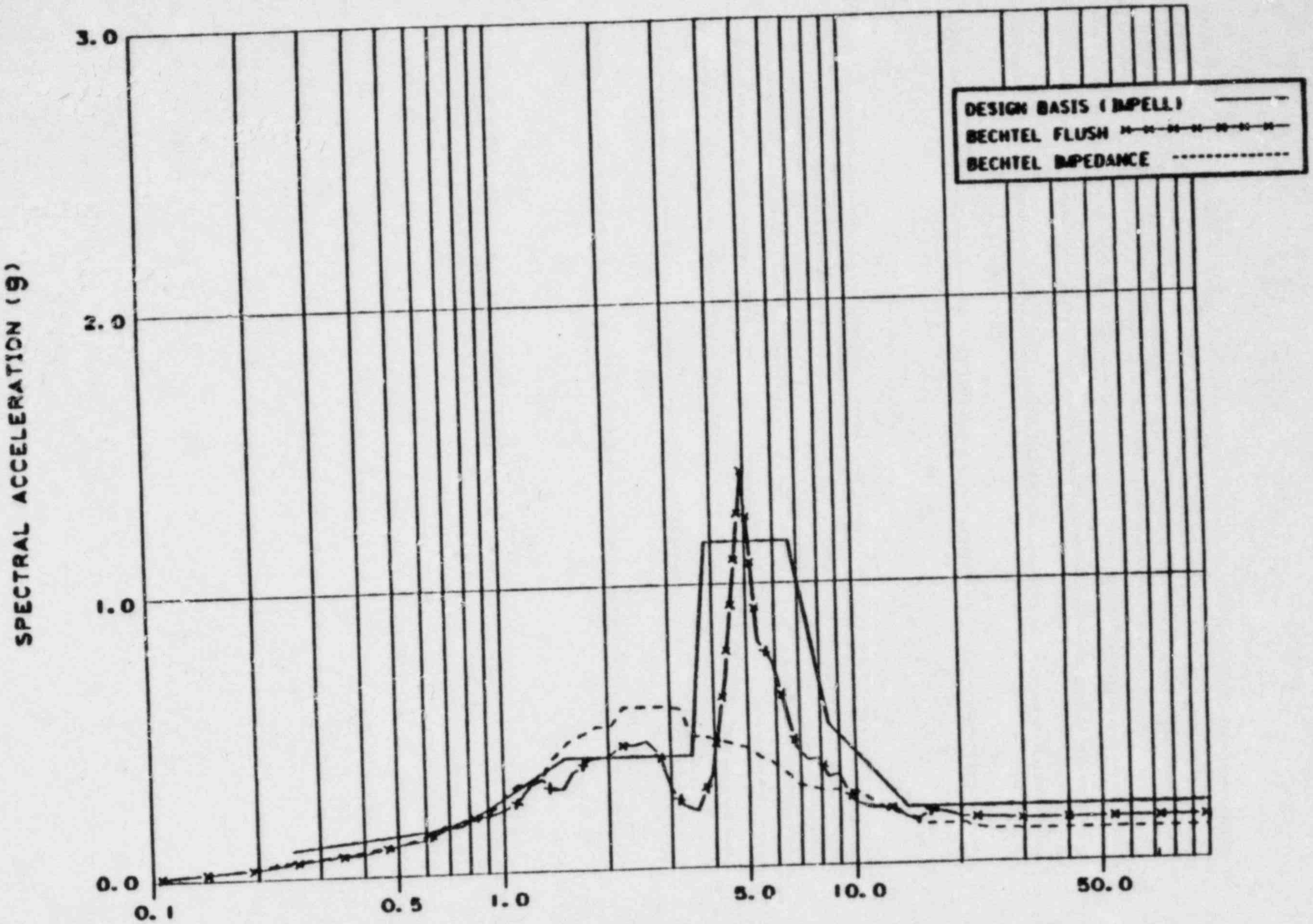


FIGURE A-13-29

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 54'-0",
N-S, OBE, 2% DAMPING

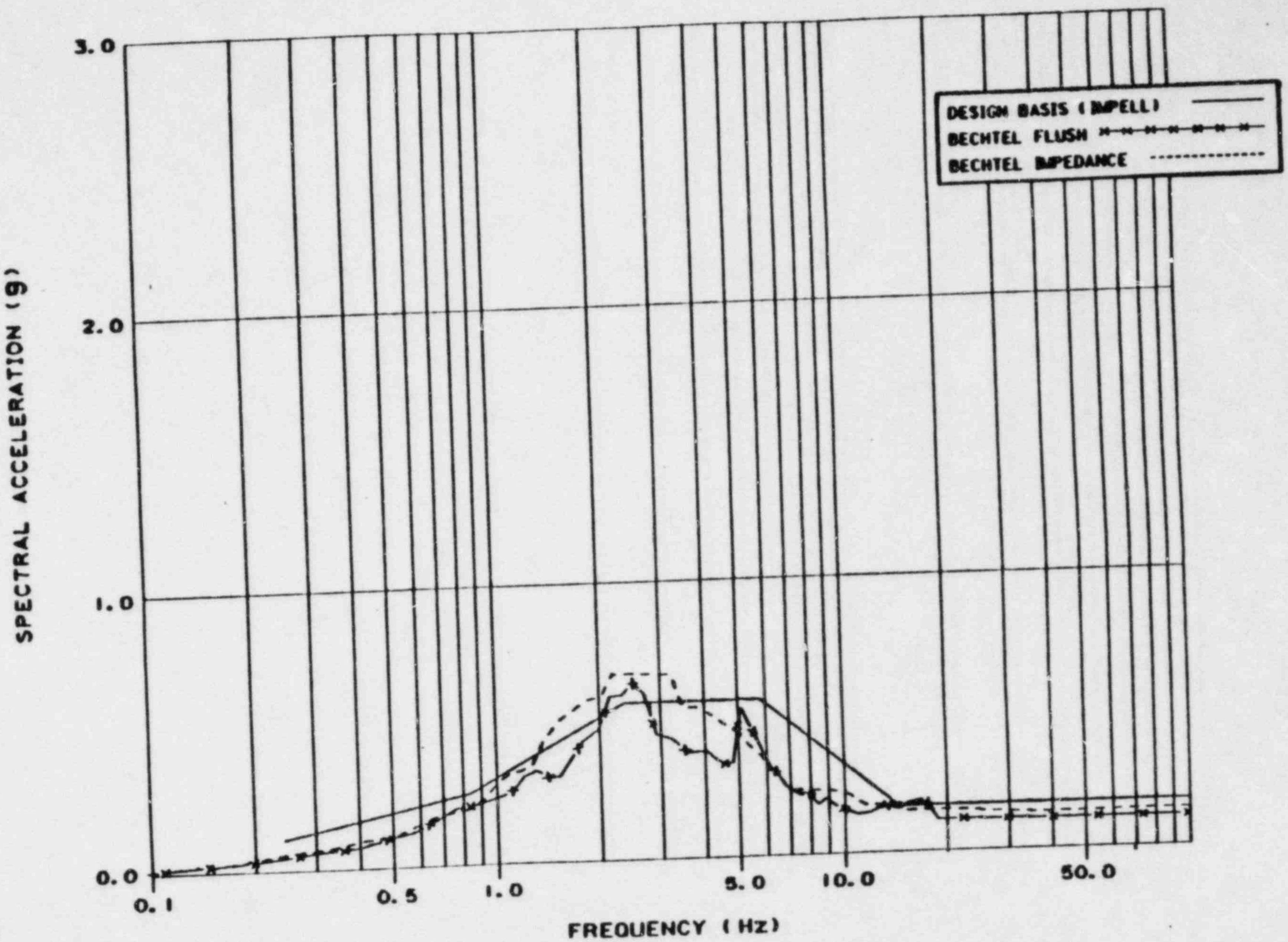


FIGURE A-13-30

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 102'-0",
N-S, OBE, 2% DAMPING

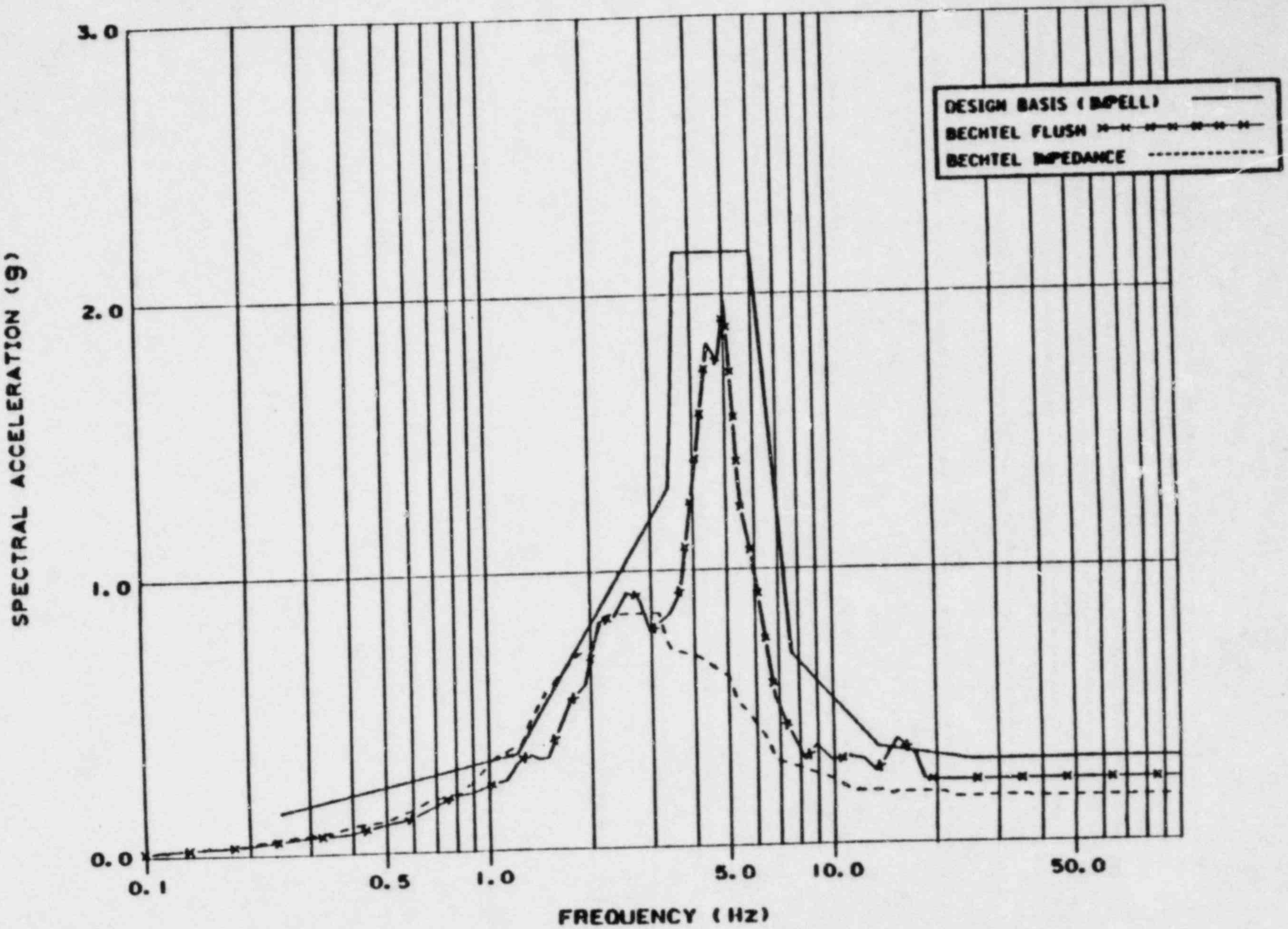


FIGURE A-13-31

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 178'-0",
N-S, OBE, 2% DAMPING

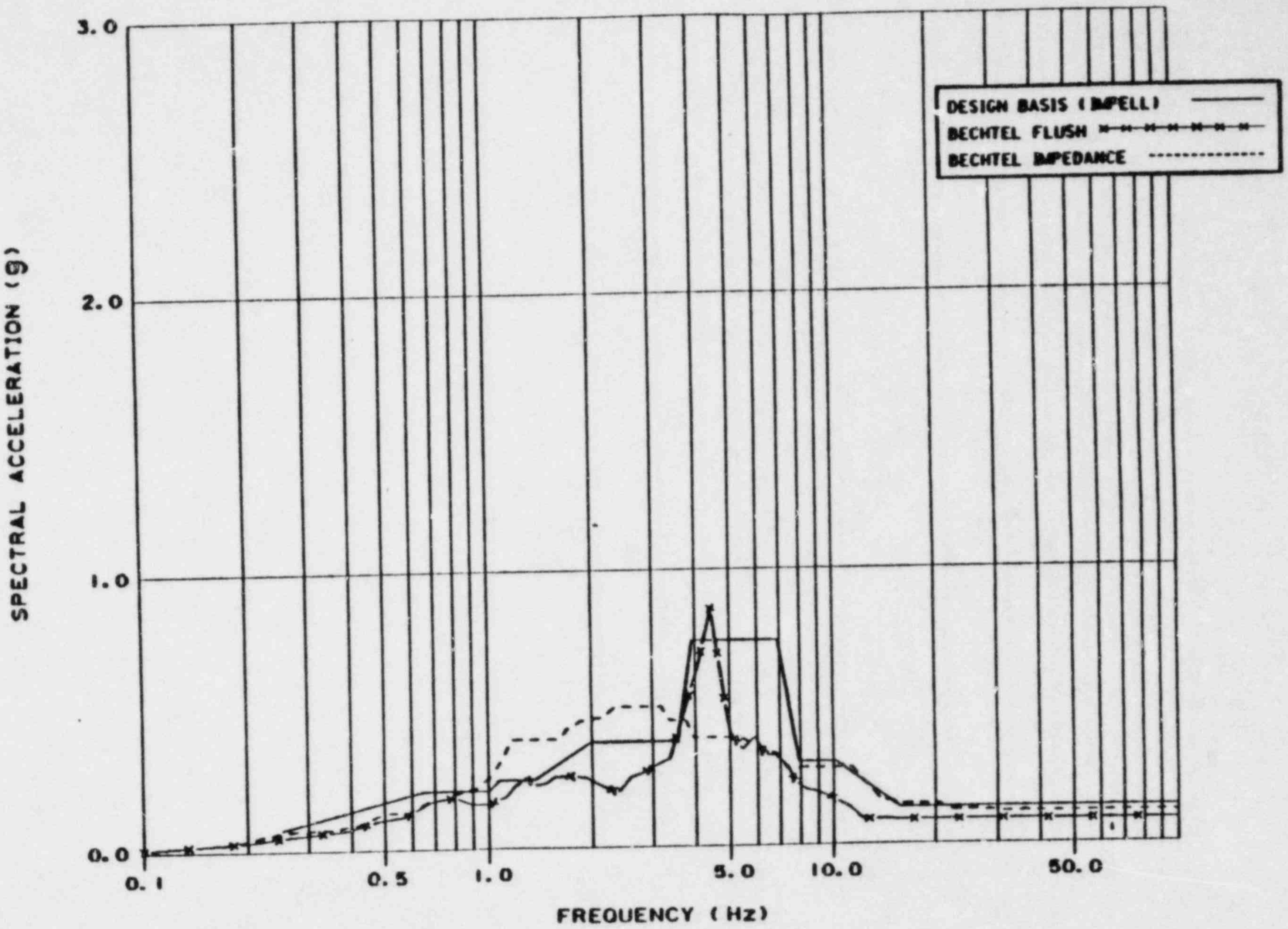


FIGURE A-13-32

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 54'-0",
E-W, OBE, 2% DAMPING

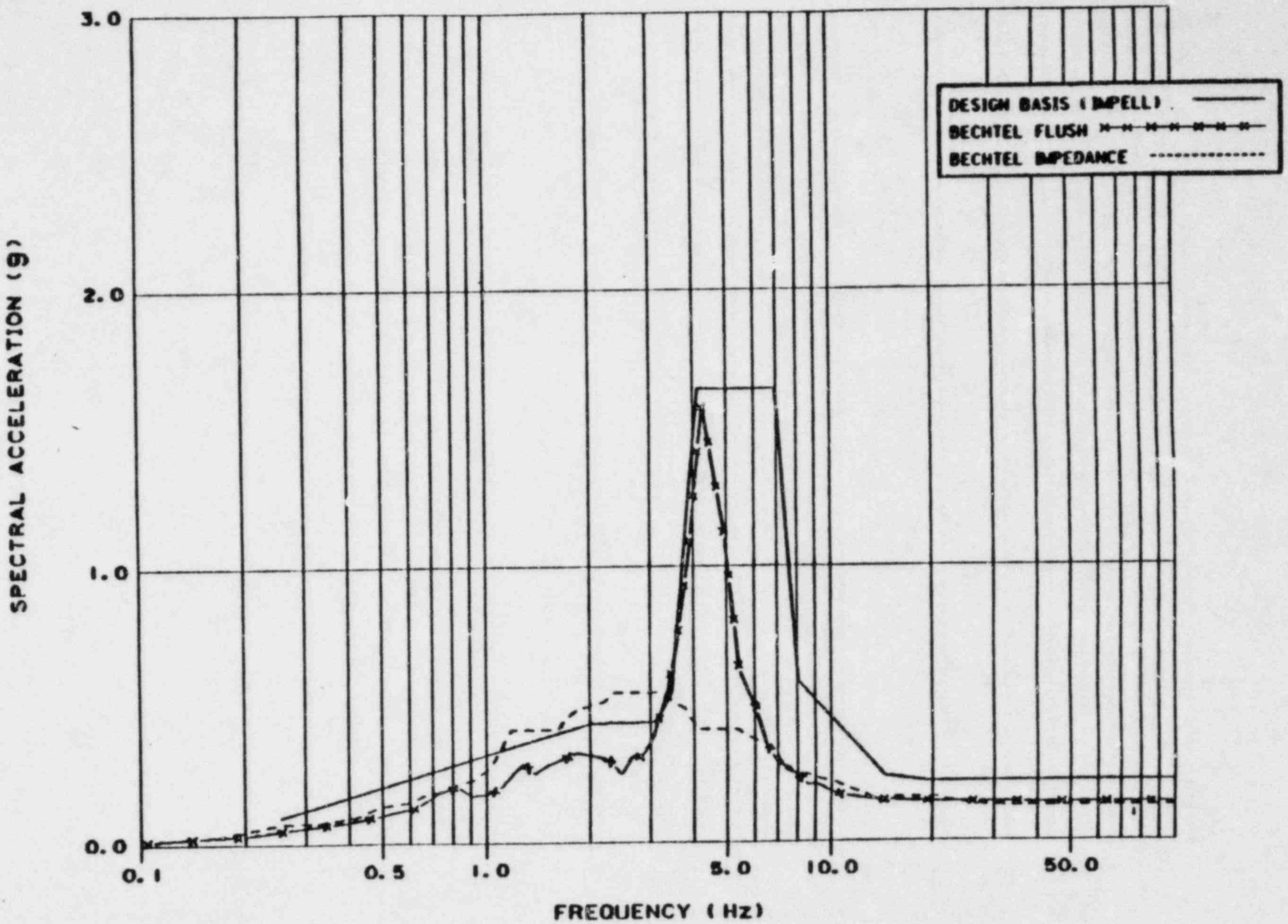


FIGURE A-13-33

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 102'-0",

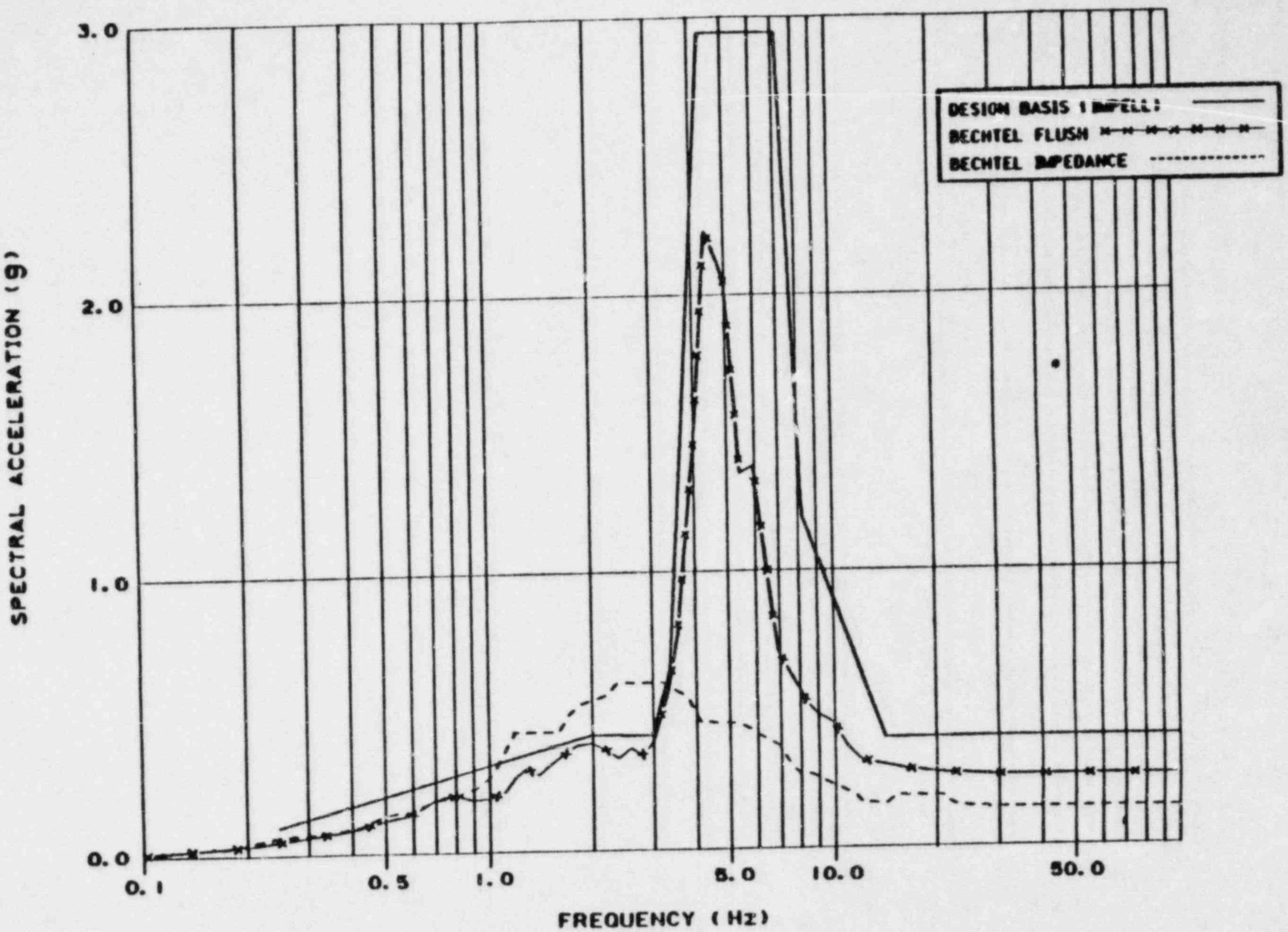


FIGURE A-13-34

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 178'-0",
E-W, OBE, 2% DAMPING

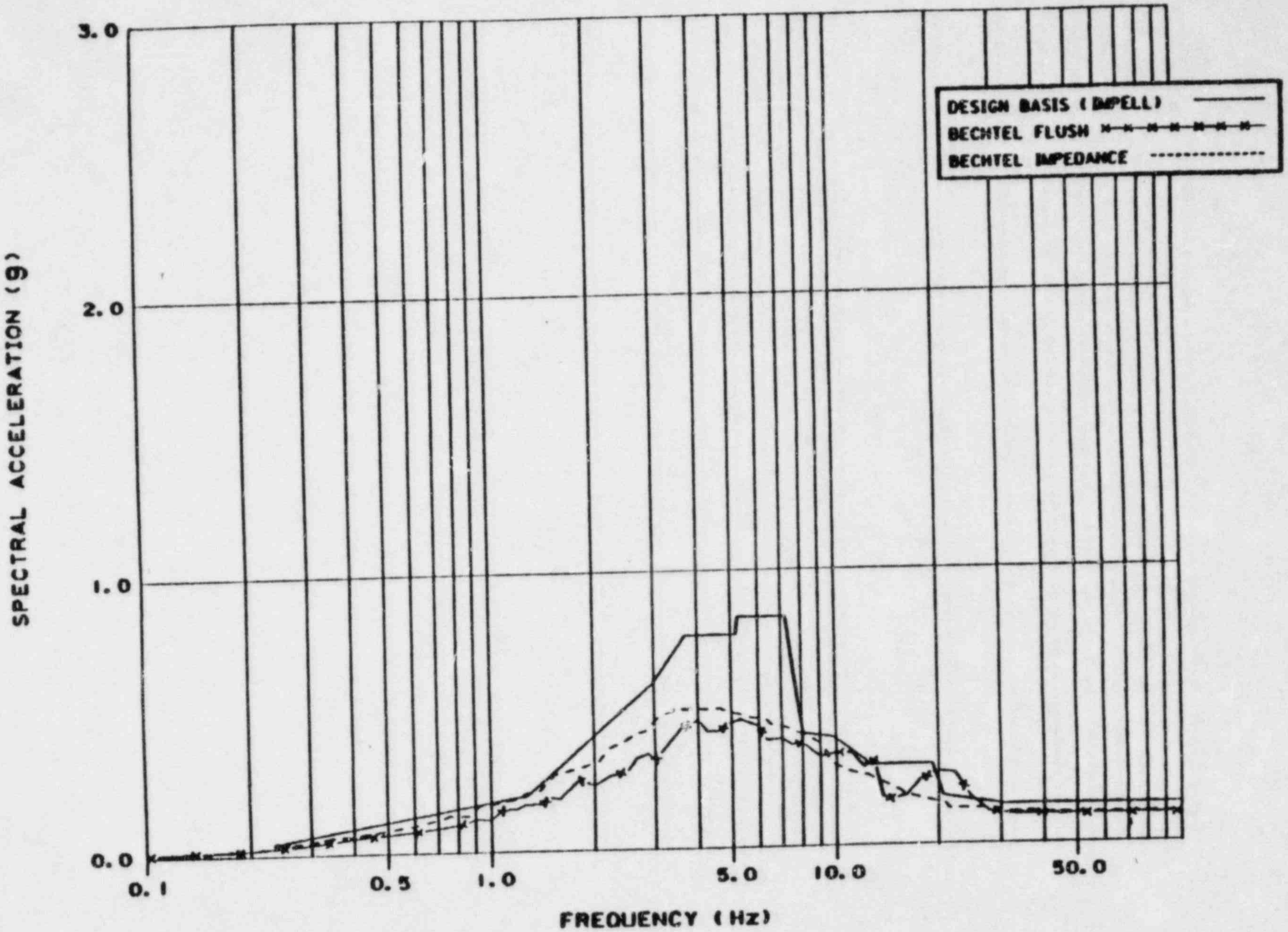


FIGURE A-13-35

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 54' -0",
VERTICAL, OBE, 2% DAMPING

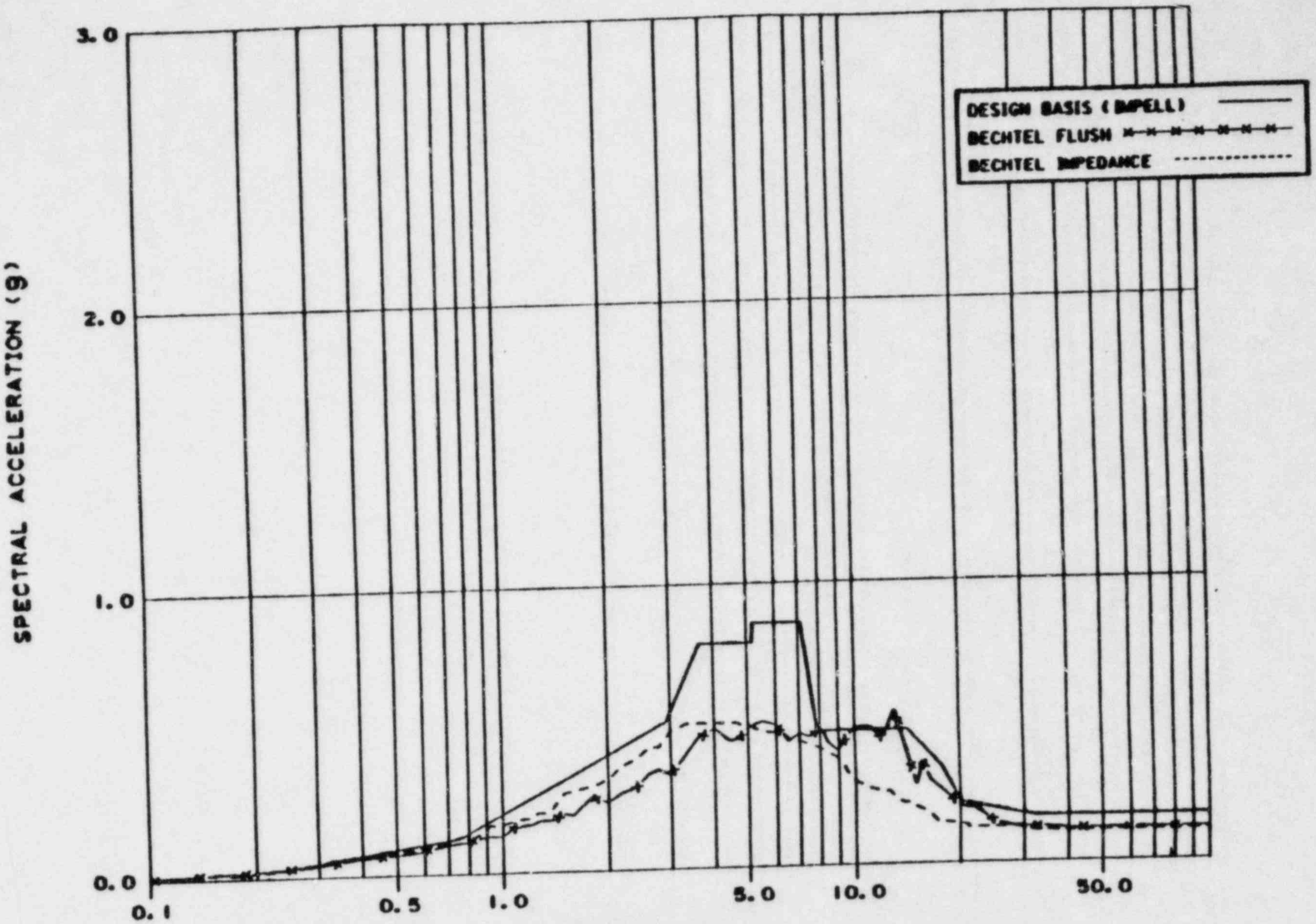


FIGURE A-13-36

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 102'-0",
VERTICAL, OBE, 2% DAMPING

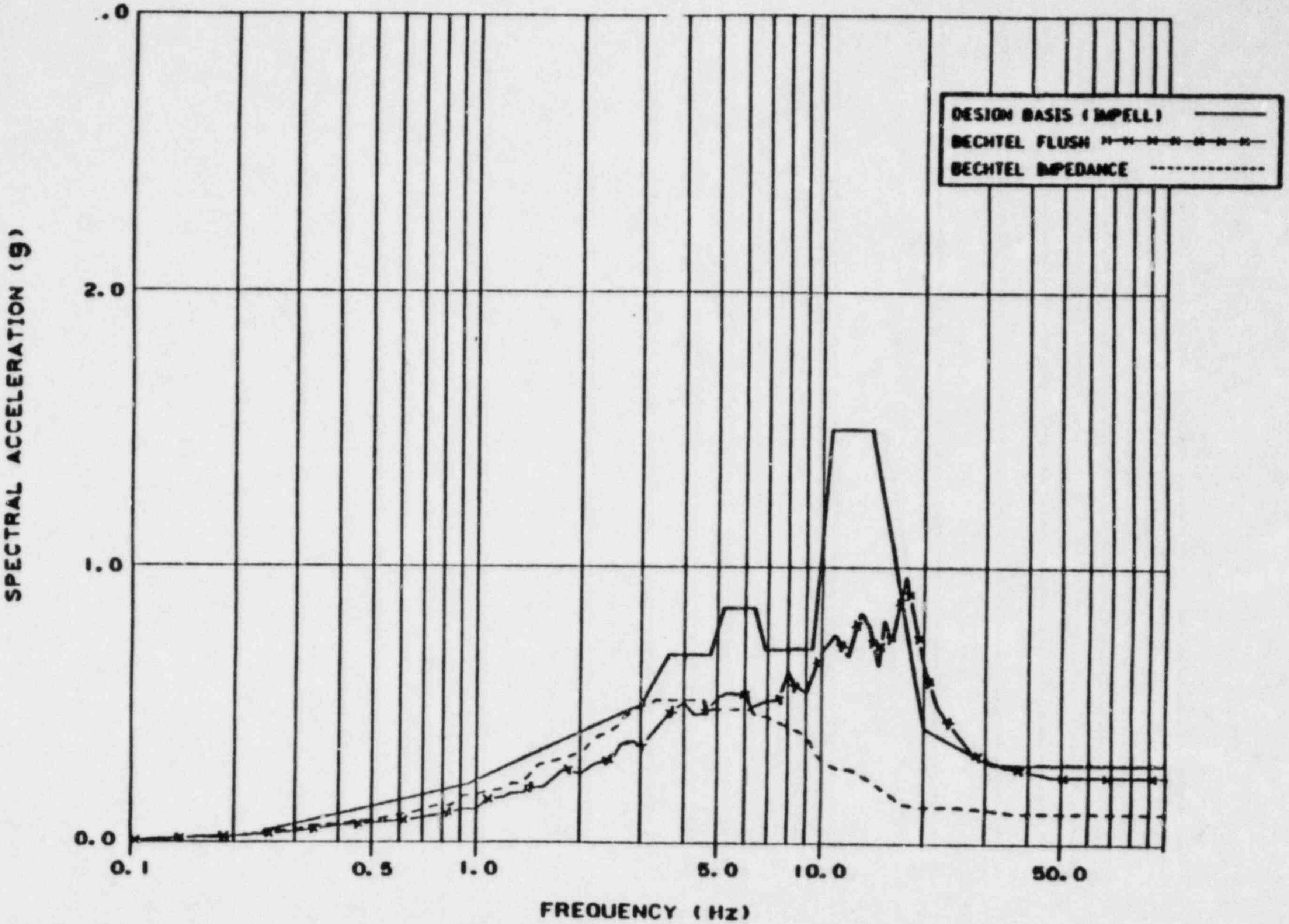


FIGURE A-13-37

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 178'-0",
VERTICAL, OBE, 2% DAMPING

OPE CREEK UNIT 1
AD. L36A, REV 1
4/20/84

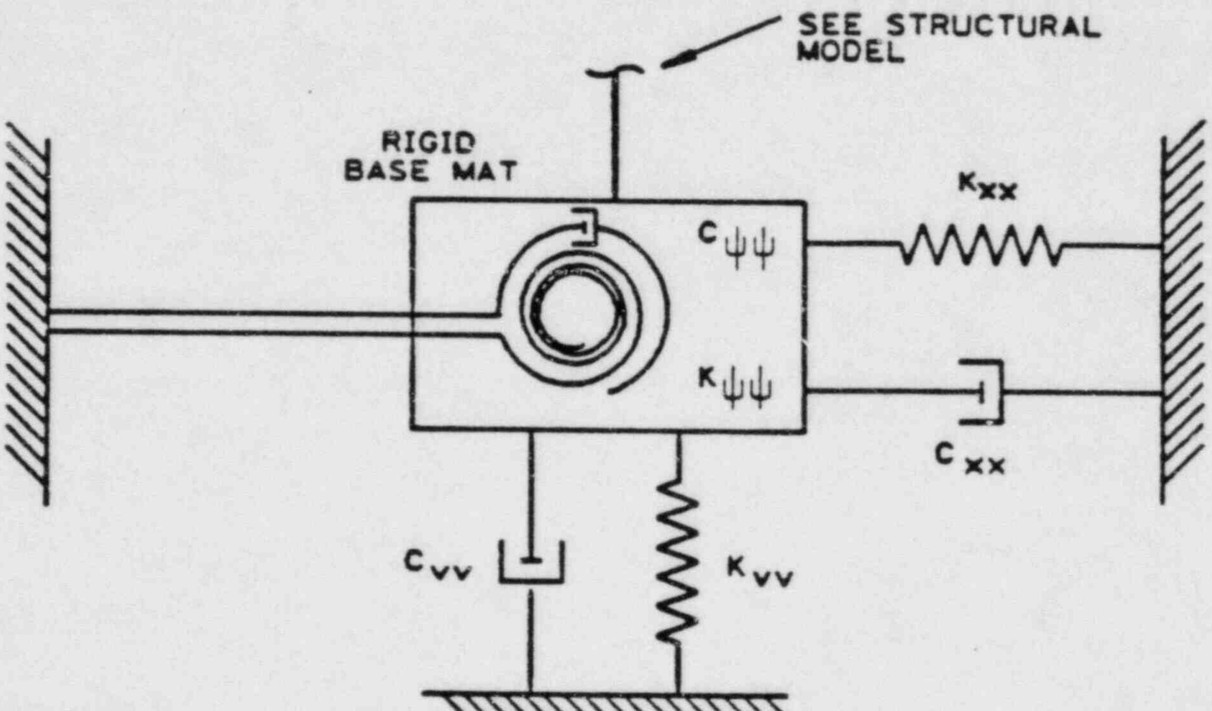
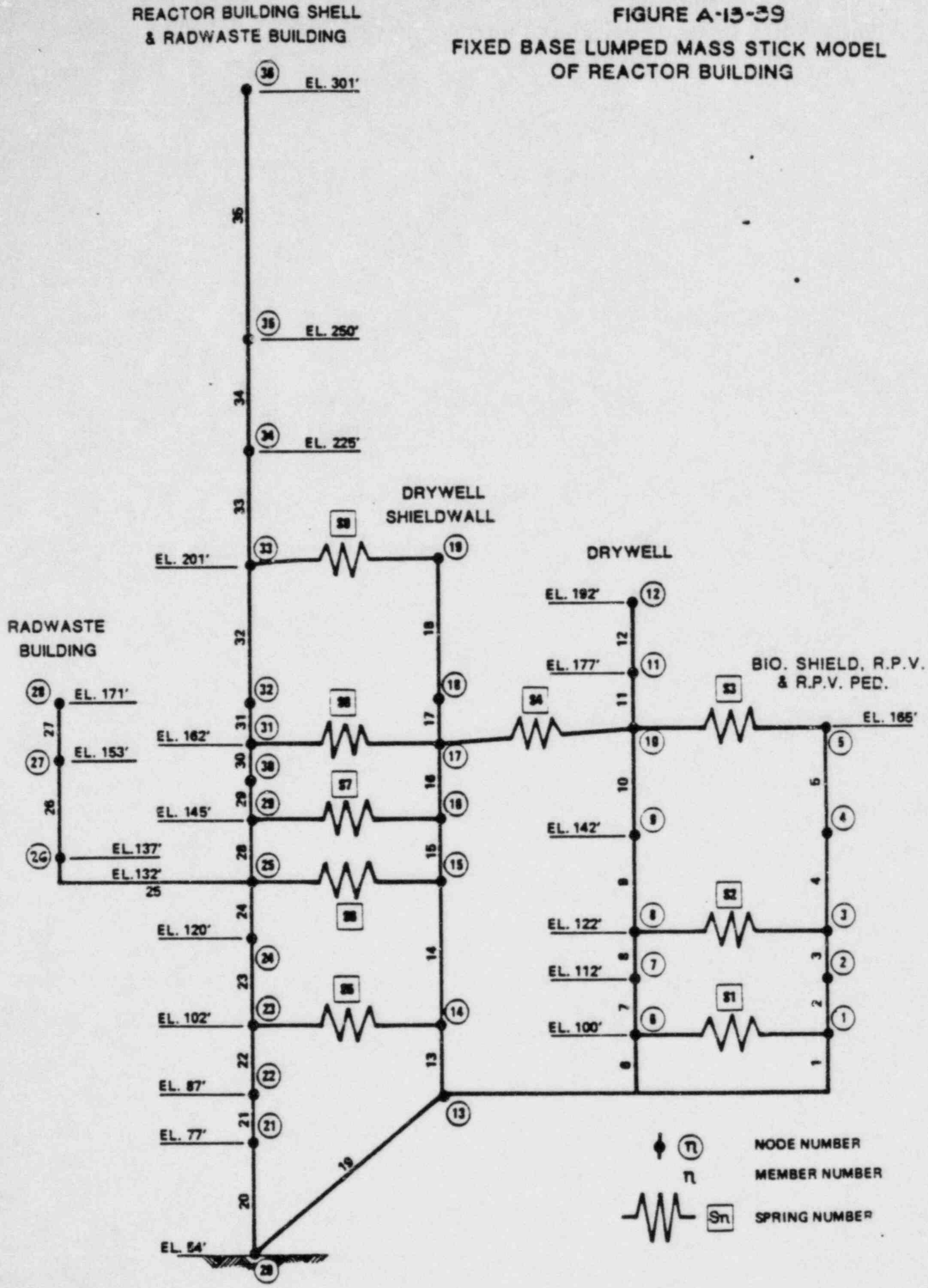


FIGURE A-13-38 IMPEDANCE APPROACH FOUNDATION MODEL

FIGURE A-13-39

FIXED BASE LUMPED MASS STICK MODEL OF REACTOR BUILDING



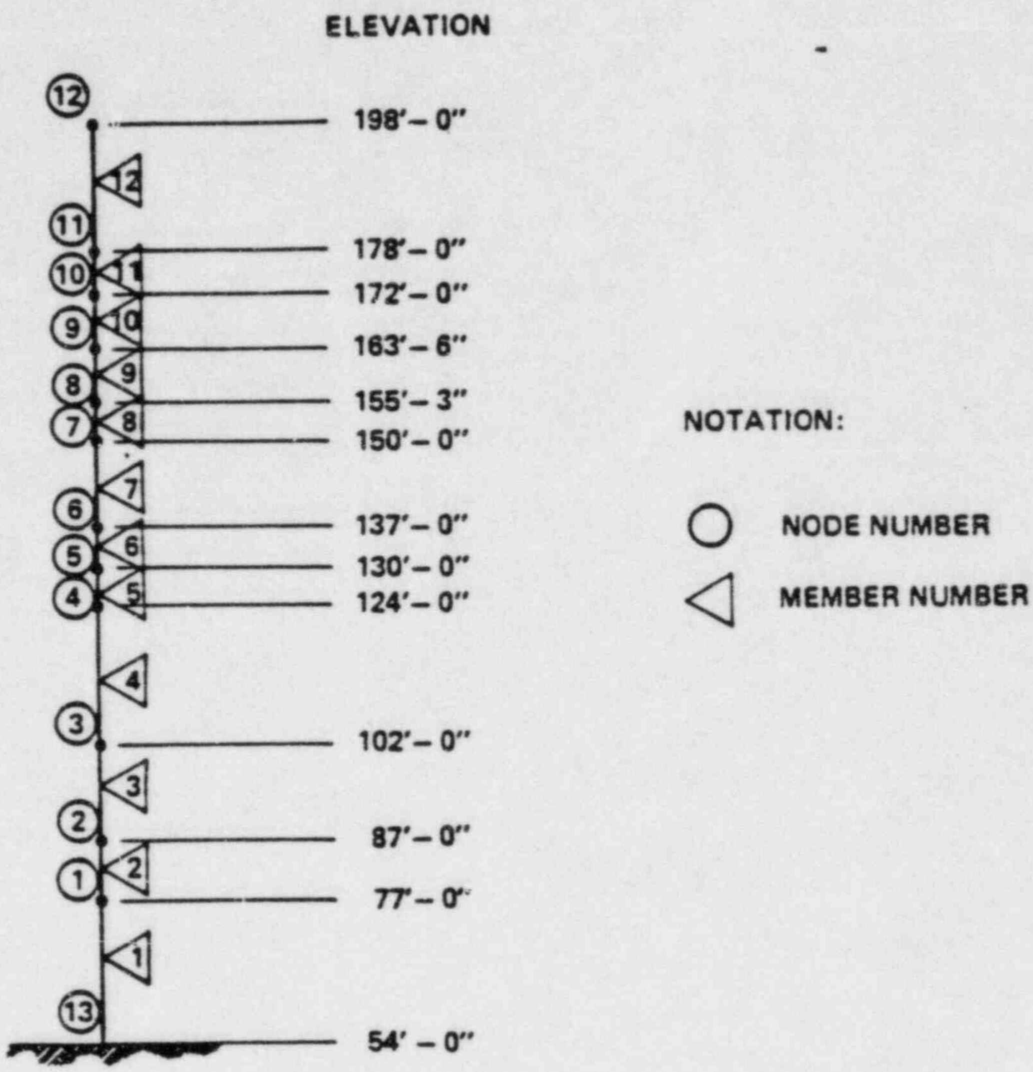


FIGURE A-13-40
FIXED BASE LUMPED MASS STICK MODEL FOR AUXILIARY BUILDING

QUESTION 220.21 (SECTION 3.7.2)

The Hope Creek soil-structure interaction analysis is performed by applying a specified earthquake input motion at the base of the finite-element model. The base level input motion is generated through a deconvolution analysis. The applicable SRP criteria for soil-structure interaction analysis require that both the half-space and finite-element approaches should be considered. (For details refer to SRP 3.7.2-II.4). Public Service should perform necessary analyses to demonstrate conformance to the criteria.

RESPONSE

As described in Amendment 1 of the FSAR (Section 3.7.2.5.2), seismic soil-structure interaction analysis is performed using both the impedance (half-space) and finite element approaches. The results of the impedance analysis are used to assess the adequacy of the finite element analysis results.

BLP 1577

~~As discussed during the NRC Structural Audit on January 10-12, 1984, a comparison of response spectra results from the finite-element and half-space methods will be provided in August 1984 for the reactor and auxiliary buildings and the service water intake structure.~~

DELETE

REPLACE WITH "INSERT A"

Insert A

Figures 220.21-1 through 220.21-54 show a comparison of the 2 percent damping response spectra obtained from the design basis finite element and the impedance analyses for the reactor building, auxiliary building and the service water intake structure (SWIS).

For the reactor and auxiliary buildings, the peak spectral accelerations obtained from the impedance analysis are generally lower than those obtained from the design basis analysis. However, the impedance analysis response spectra are not completely enveloped by those obtained from the design basis analysis, especially in the frequency range from 1.0 to 3.5 Hz. Also, there are some local exceedances in the higher frequency range, as shown in Figures 220.21-1 through 220.21-36.

For the SWIS (Figures 220.21-37 through 220.21-54), the impedance analysis response spectra are generally enveloped by those obtained from the design basis analysis at elevation 114.0 feet. For other elevations, the impedance analysis spectral accelerations exceed the design basis spectral accelerations in some frequency ranges. These ranges vary approximately between 1.5 and 10.0 Hz.

For the reactor building and the SWIS, seismic shear forces and moments obtained from the impedance analysis exceeded the design basis values at various locations within these buildings. The auxiliary building seismic shear forces and moments obtained from the impedance analysis are less than the design basis shears and moments.

Since the impedance analysis results are not completely enveloped by the design basis finite element analysis results, sampling studies are conducted to confirm the adequacy of the plant design. These sampling studies include an evaluation of the following items:

- Structures
- Equipment
- Cable tray and HVAC supports
- Piping and pipe supports

In all cases, the items reviewed during these sampling studies can accommodate the loads resulting from the impedance analysis. Therefore, the impedance analysis results have no impact on the plant design.

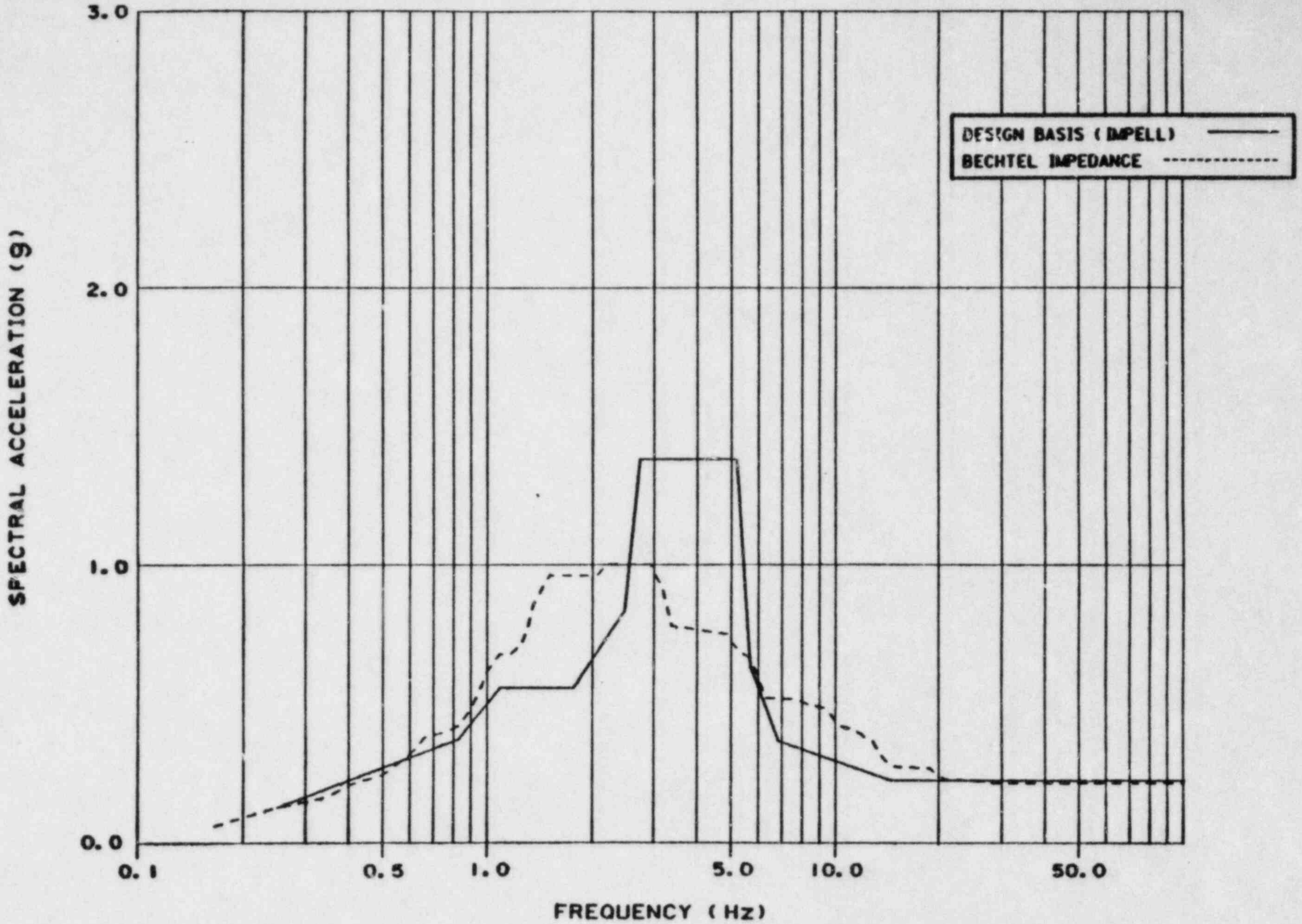


FIGURE 220.21-1

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 54'-0",
N-S, SSE, 2% DAMPING

HOPE CREEK UNIT I
CAD, LIA SHT 2, REV 0
07/27/84

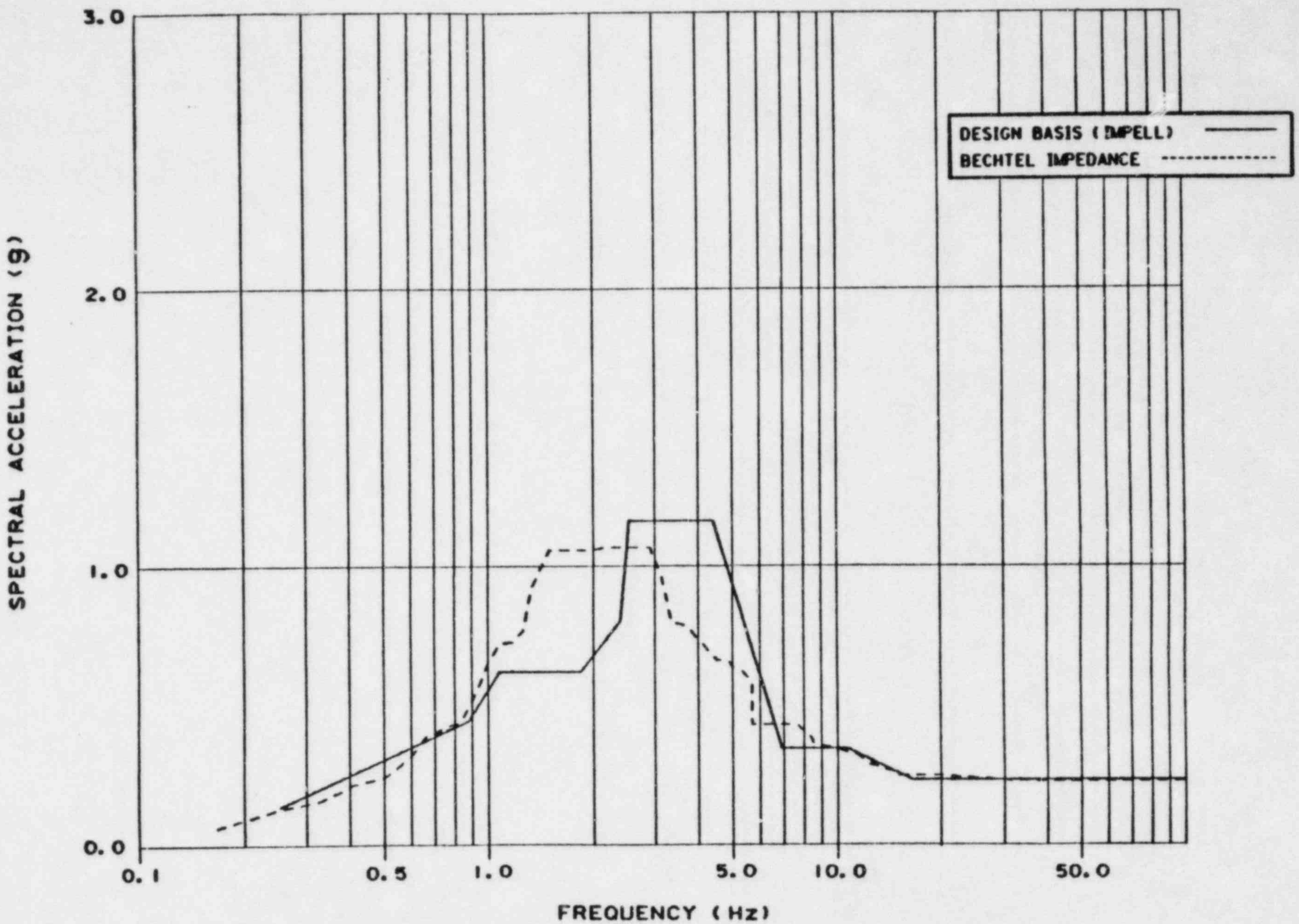


FIGURE 220.21-2

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 102' -0".
N-S, SSE, 2% DAMPING

HOPE CREEK UNIT I
CAD, L2A SHT 2, REV 0

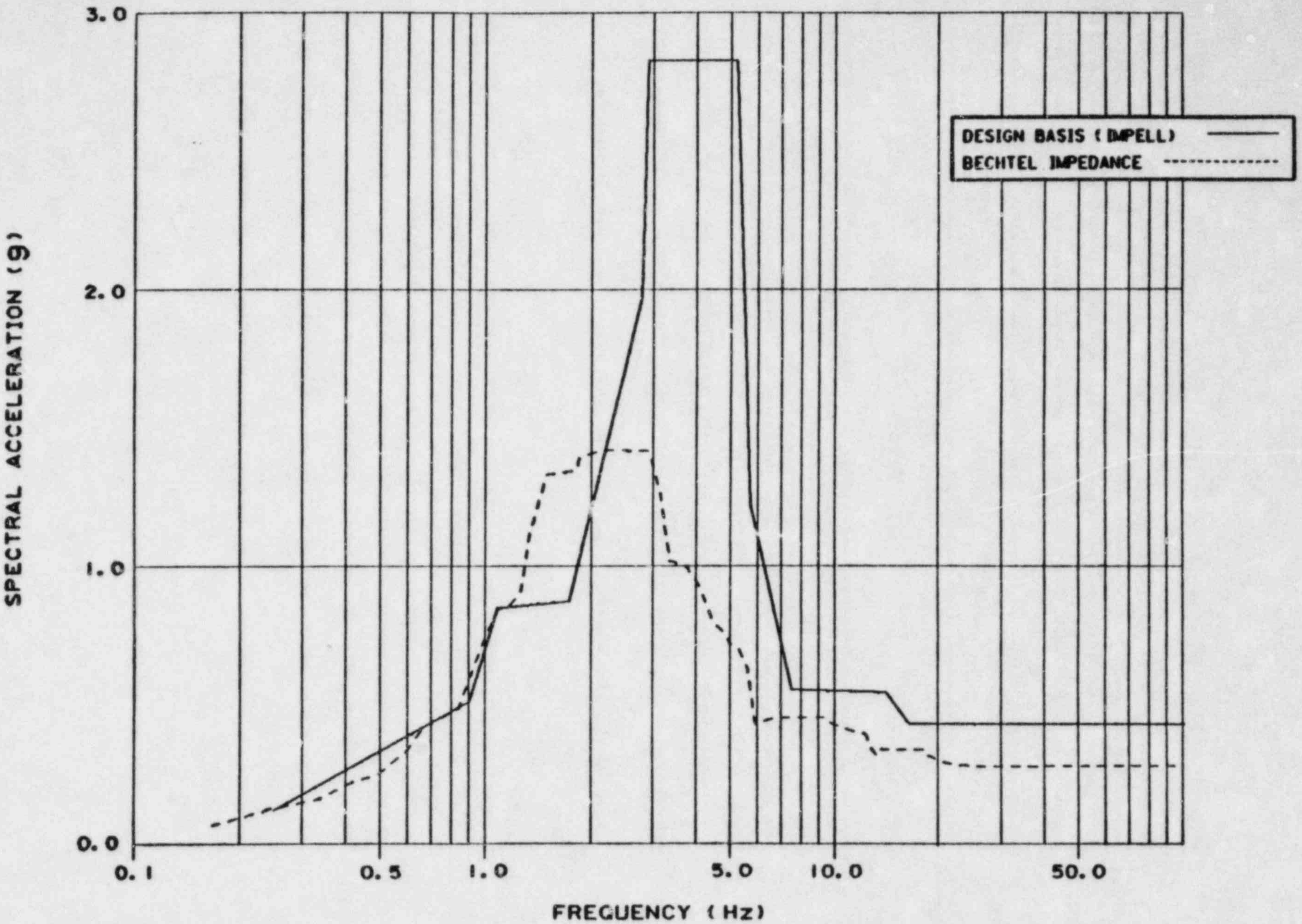


FIGURE 220.21-3

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 201'-0",
N-S, SSE, 2% DAMPING

HOPE CREEK UNIT I
CAD. L3A SHT 2, REV 0
07/27/84

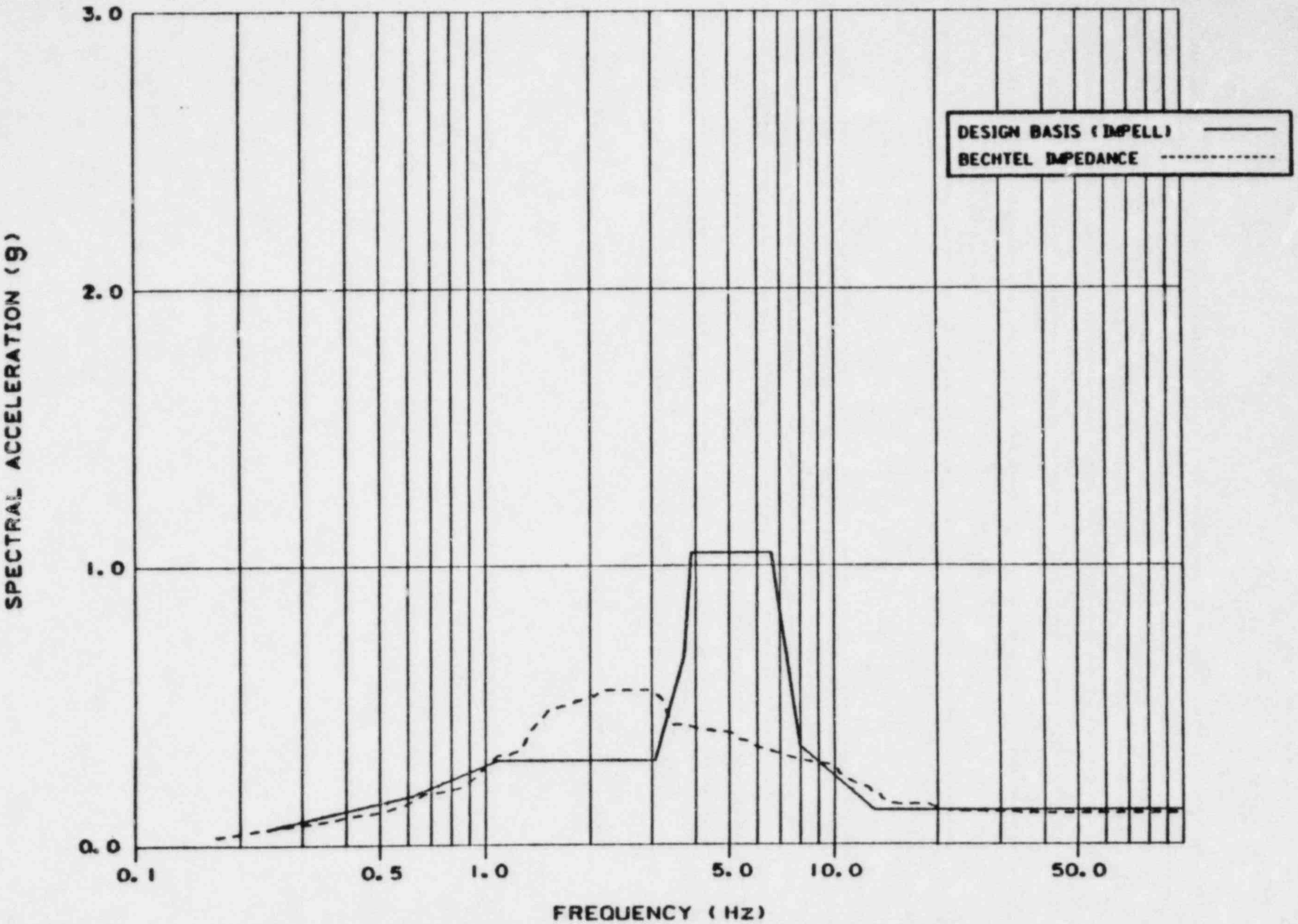


FIGURE 220.21-4

RESPONSE SPECTRA COMPARISON,
UNIT 1 REACTOR BUILDING AT ELEV. 54'-0"
N-S, OBE, 2% DAMPING.

HOPE CREEK UNIT 1
CAD. L4A SHT 2. REV 0

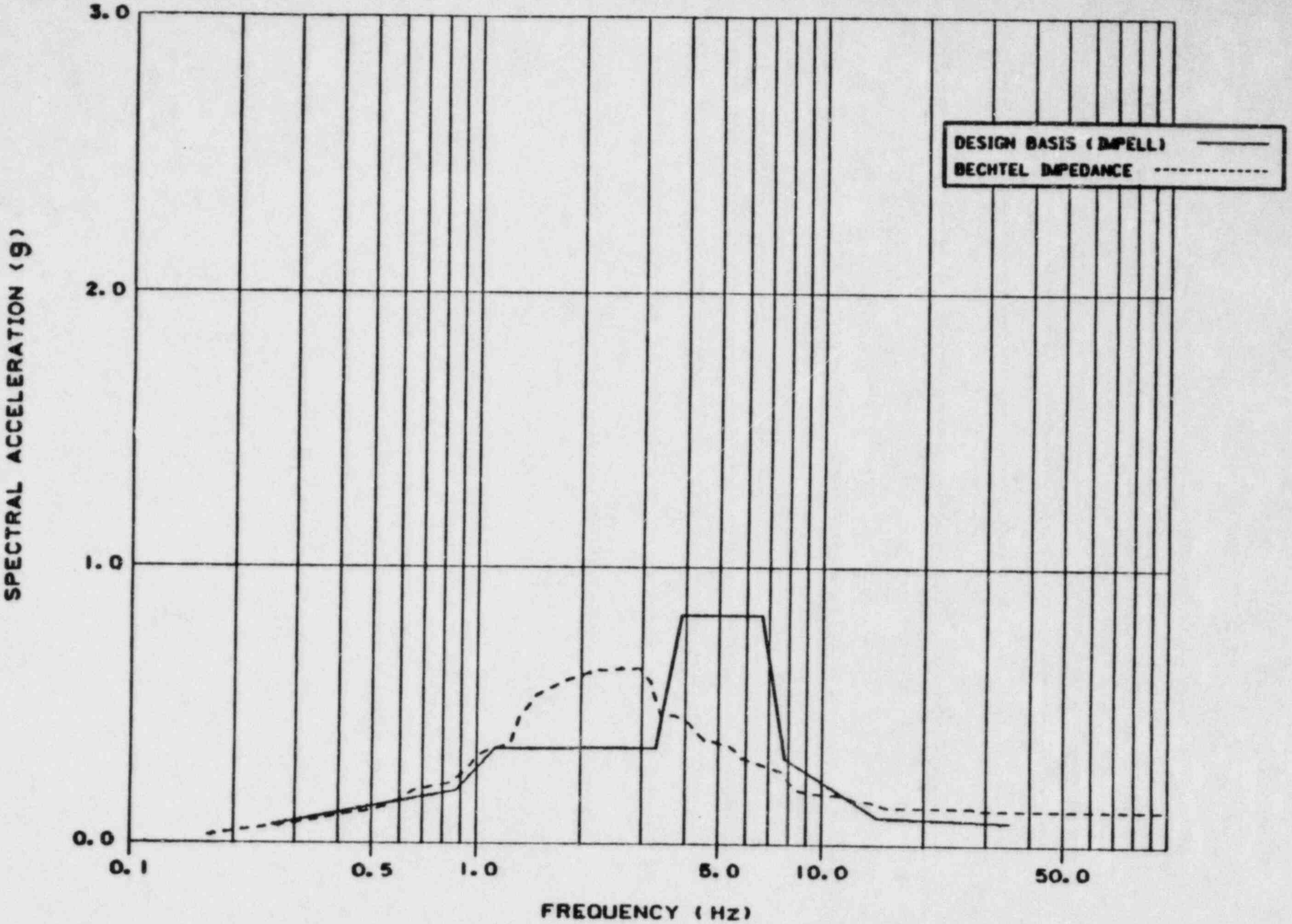


FIGURE 220.21-5

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 102'-0",
N-S, OBE, 2% DAMPING.

HOPE CREEK UNIT I
CAD.L5A SHT 2, REV 0
07/27/84

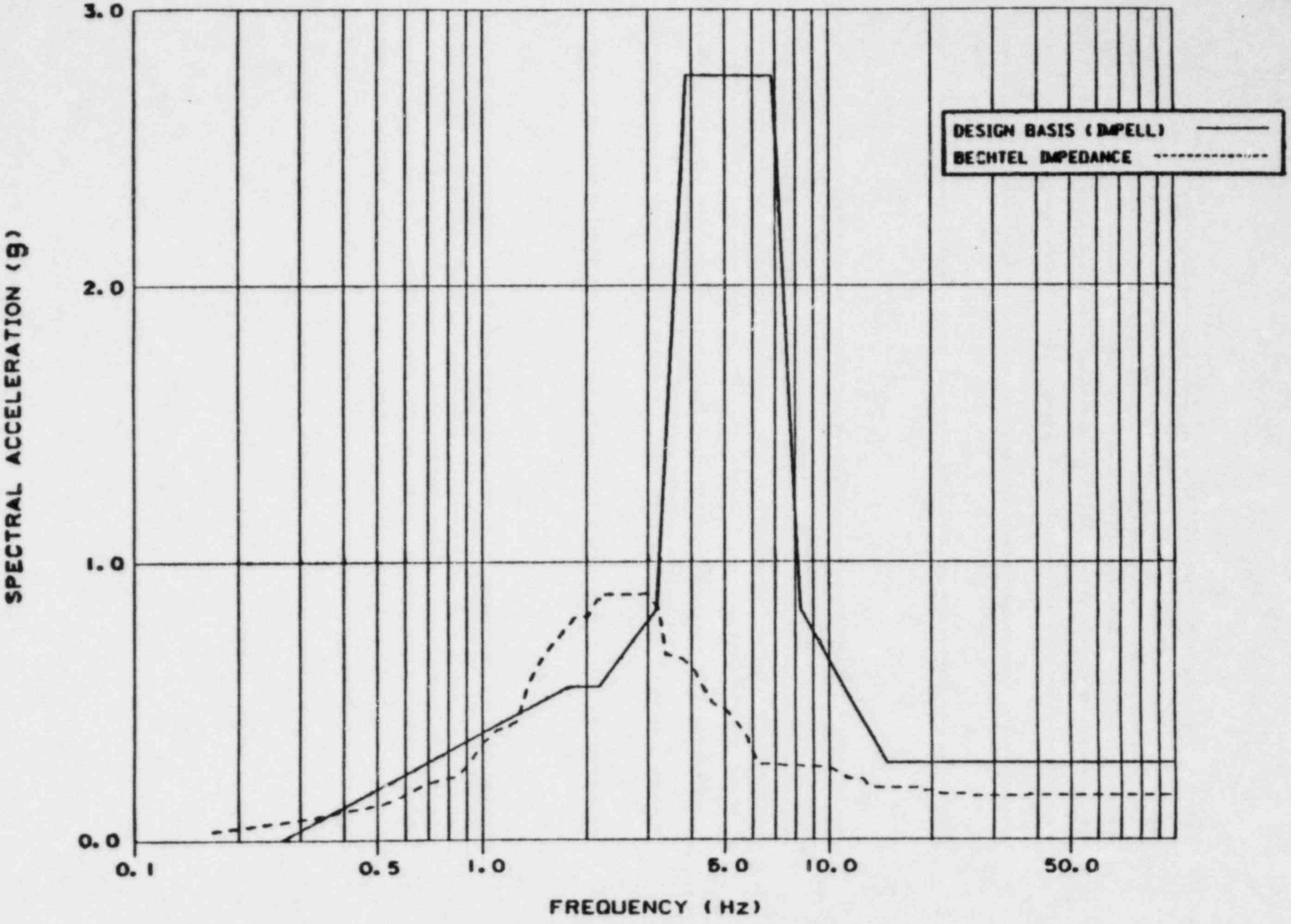


FIGURE 220.21-6

RESPONSE SPECTRA COMPARISON.
UNIT I REACTOR BUILDING AT ELEV. 201'-0",
N-S, OBE, 2% DAMPING

HOPE CREEK UNIT I
CAD. L6A SHT 2. REV 0

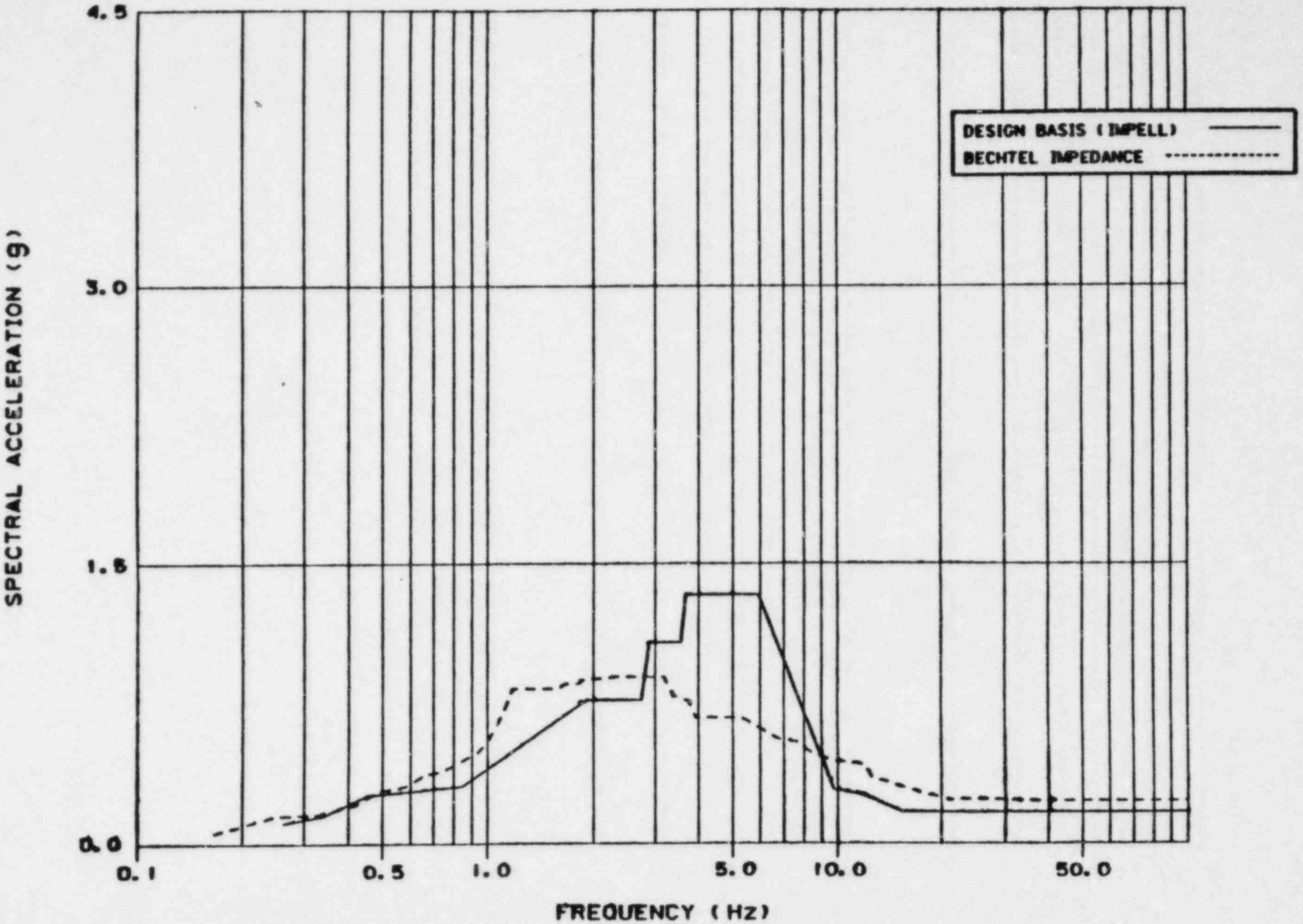


FIGURE 220.21-7

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 54'-0",
E-W, SSE, 2% DAMPING

HOPE CREEK UNIT I
CAD. L7A SHT 2, REV 0
07/27/84

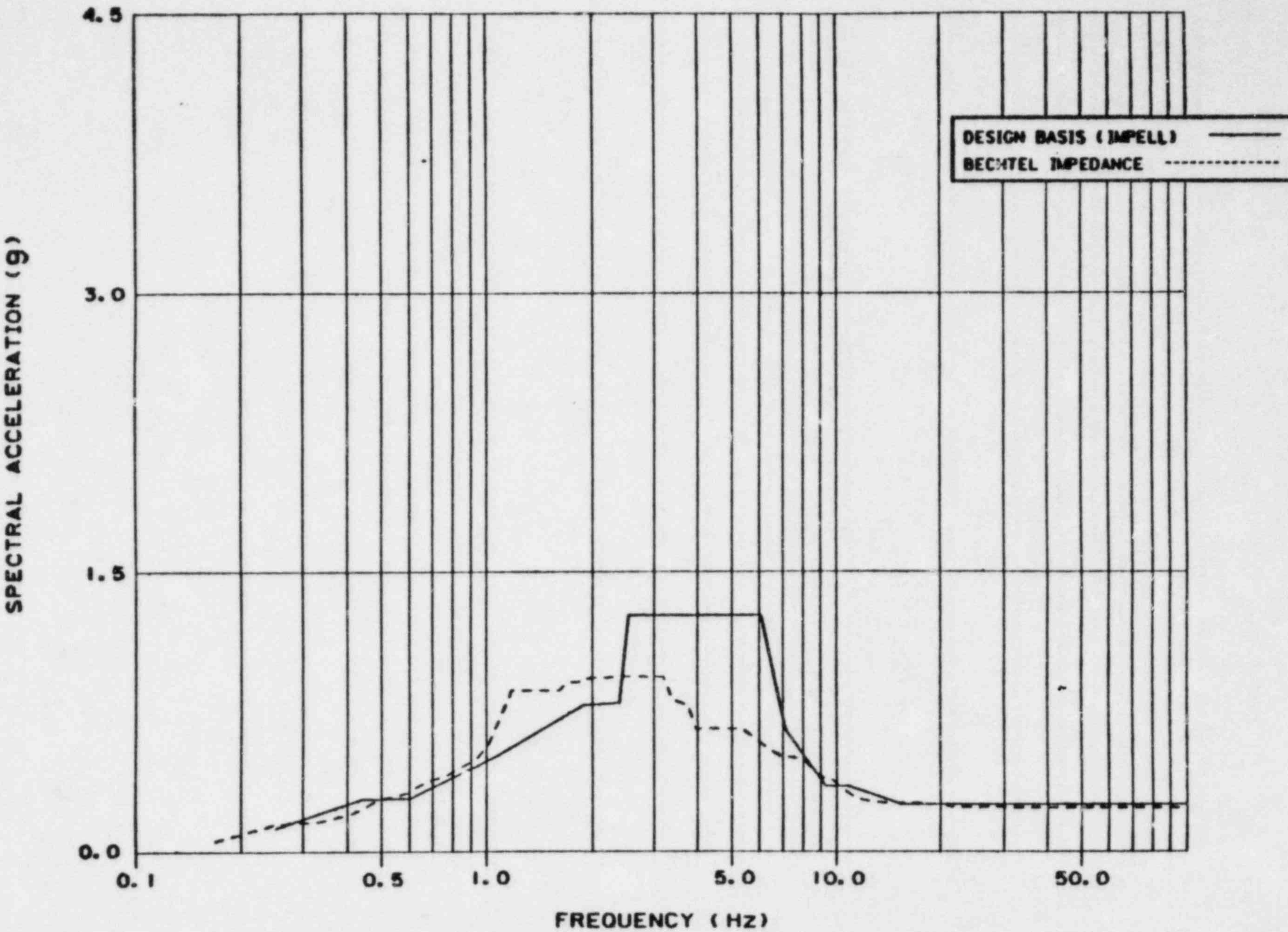


FIGURE 220.21-8

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 102'-0",
E-W, SSE, 2% DAMPING

HOPE CREEK UNIT I
CAD. LBA SHT 2. REV 0

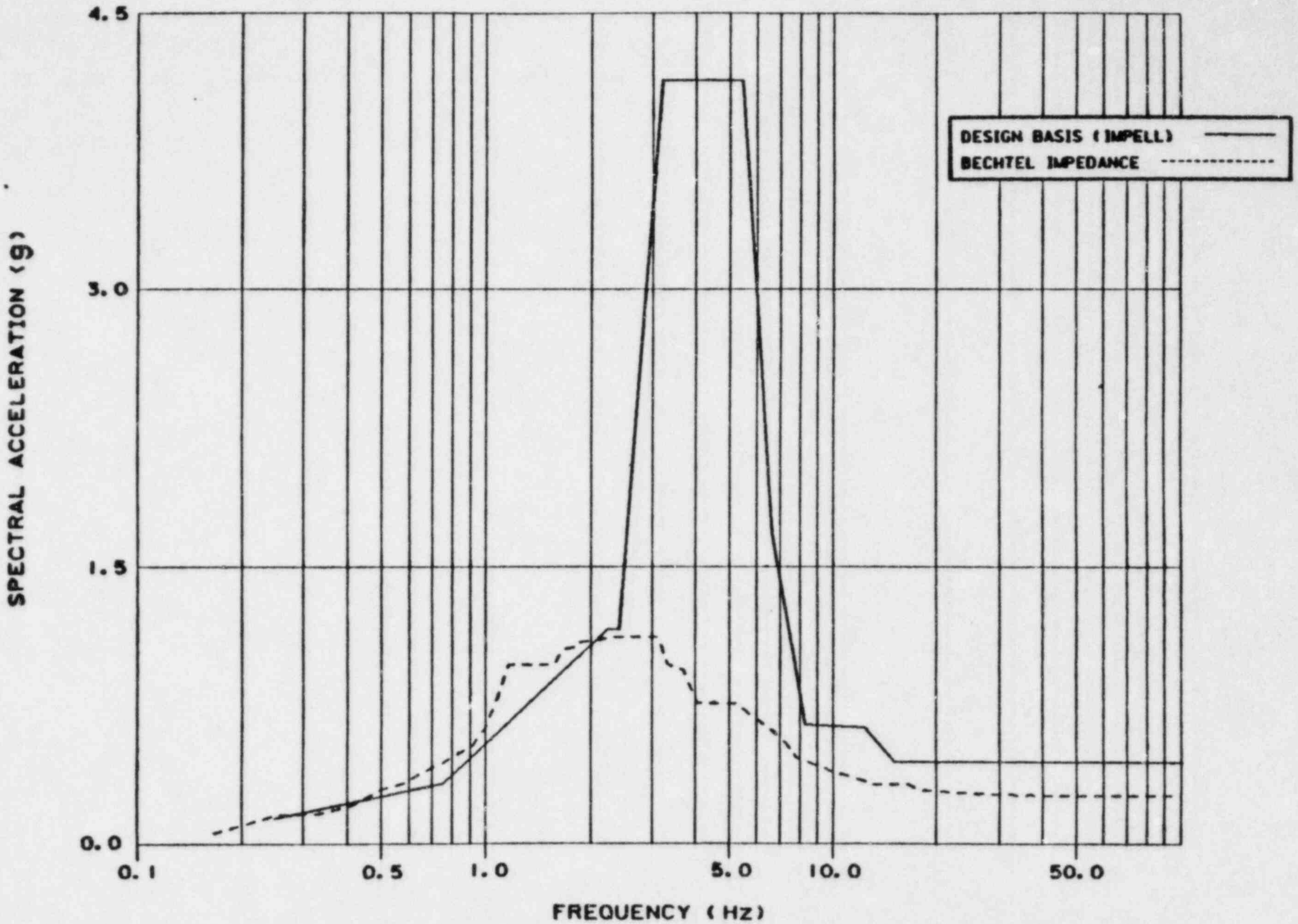


FIGURE 220.21-9

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 201'-0",
E-W, SSE, 2% DAMPING

HOPE CREEK UNIT I
CAD, L9A SHT 2, REV 0

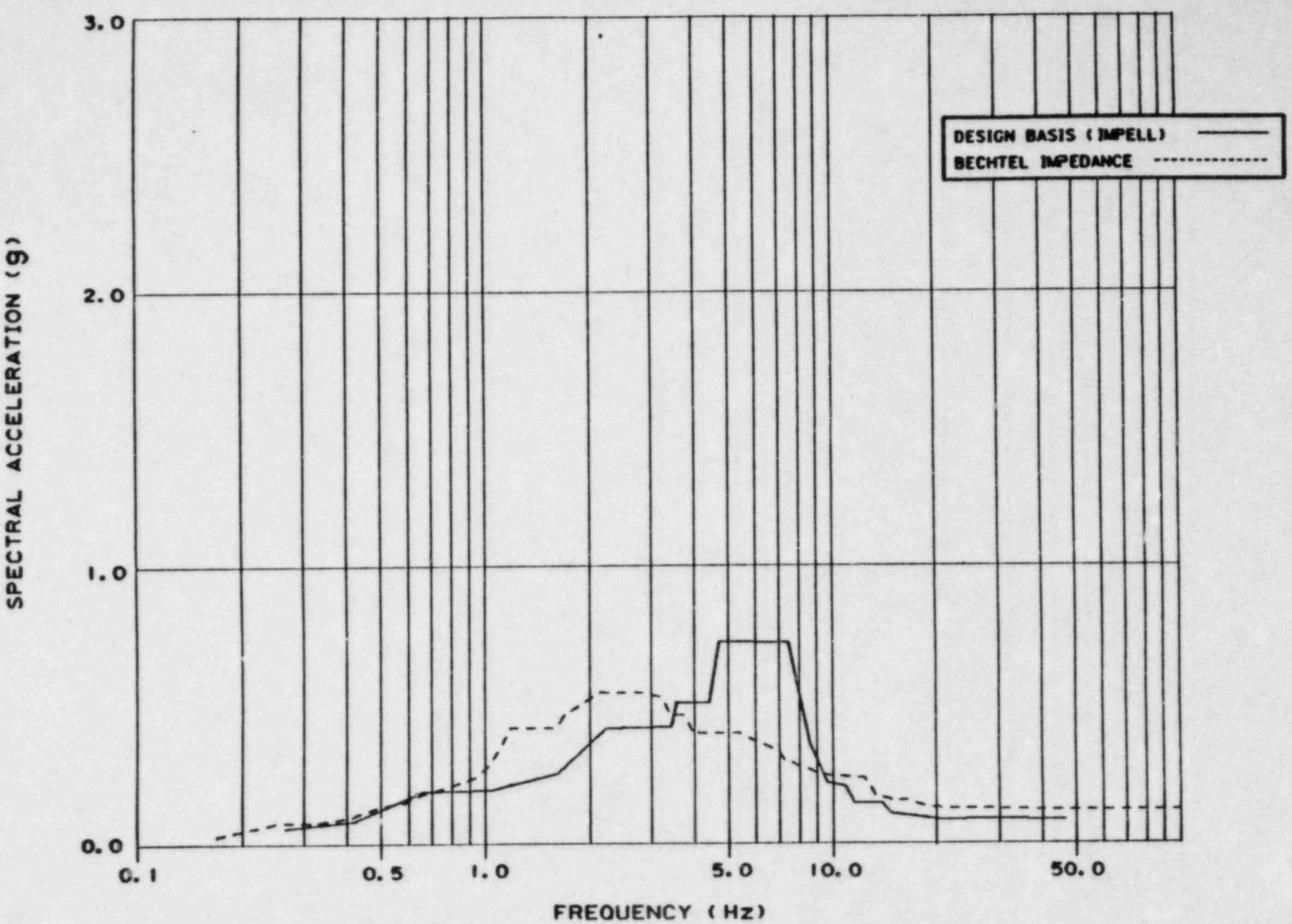


FIGURE 220.21-10

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 54'-0",
E-W, OBE, 2% DAMPING

HOPE CREEK UNIT I
CAD, LIOA SHT 2, REV 0
07/27/84

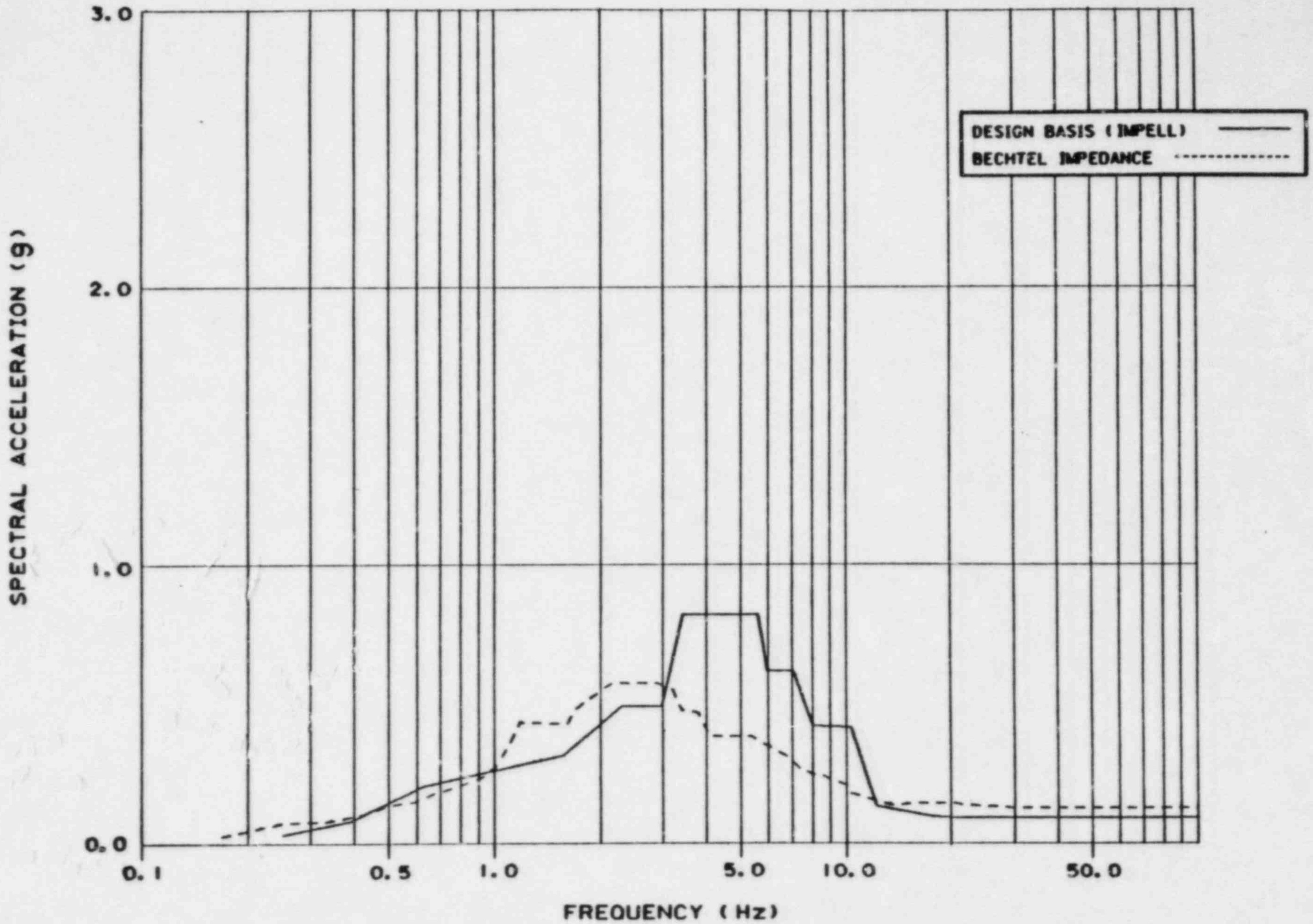
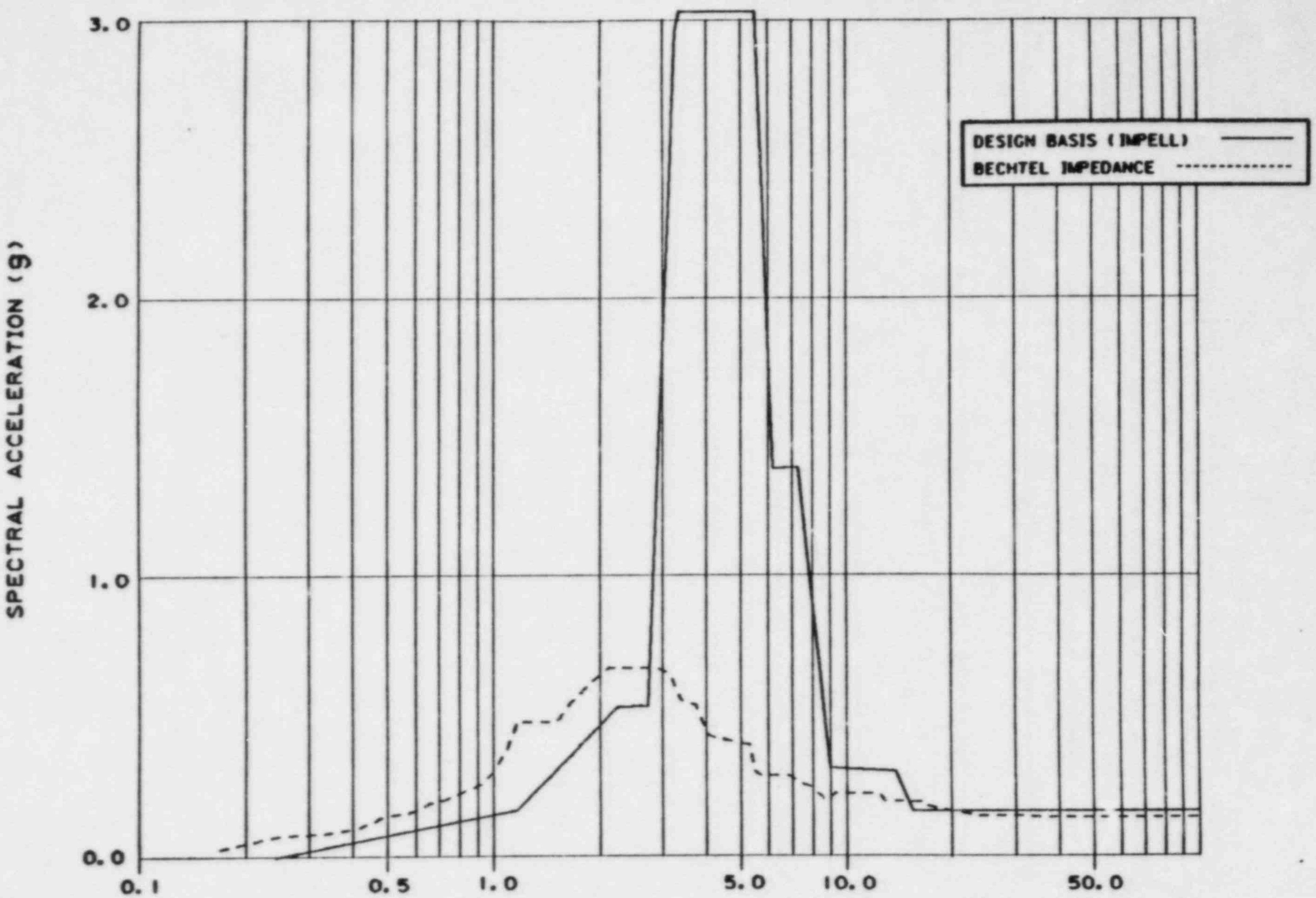


FIGURE 220.21-11

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 102'-0",
E-W, OBE, 2% DAMPING

HOPE CREEK UNIT I
CAD, LIIA SHT 2, REV 0
07/27/84



FIGL 220.21-12

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 201'-0",
E-W, OBE, 2% DAMPING

HOPE CREEK UNIT I
CAD. L12A SHT 2, REV 0
07/27/84

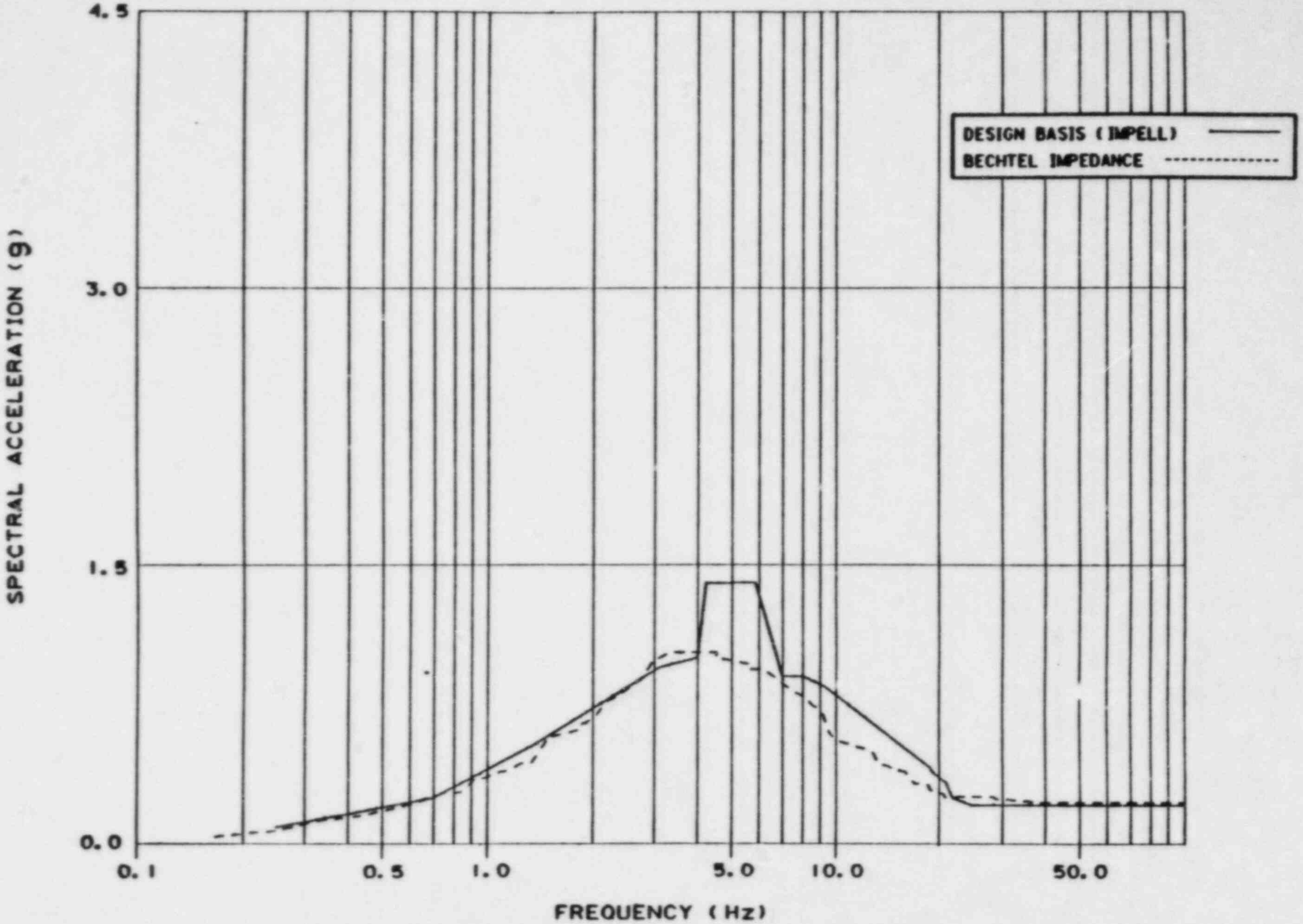


FIGURE 220.21-13

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 54'-0",
VERTICAL, SSE, 2% DAMPING

HOPE CREEK UNIT I
CAD, L25A SHT 2, REV 0
07/27/84

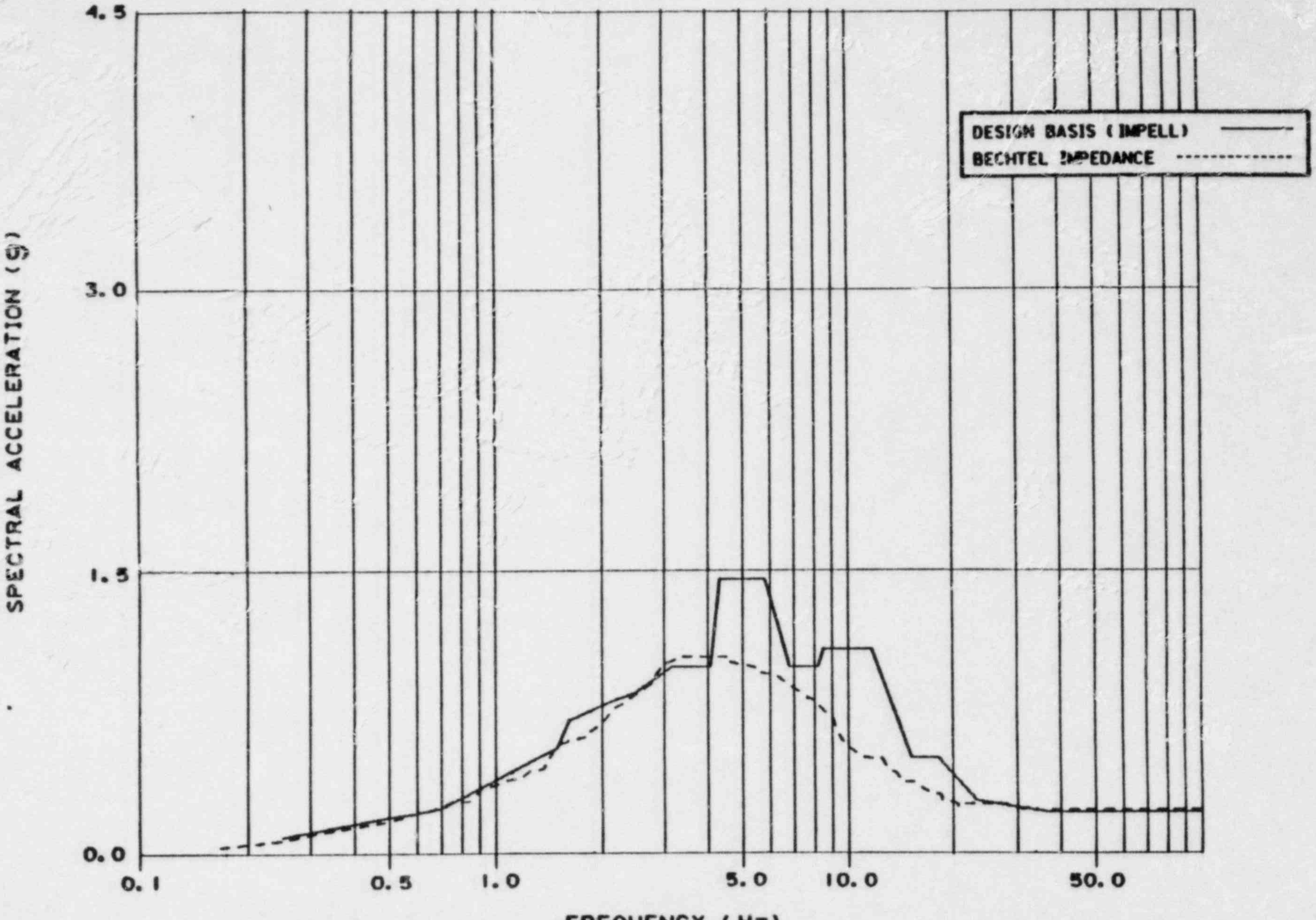


FIGURE 220.21-14

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 102' -0",
VERTICAL SSE, 2% DAMPING

HOPE CREEK UNIT I
CAD. L26A SHT 2, REV 0
07/27/84

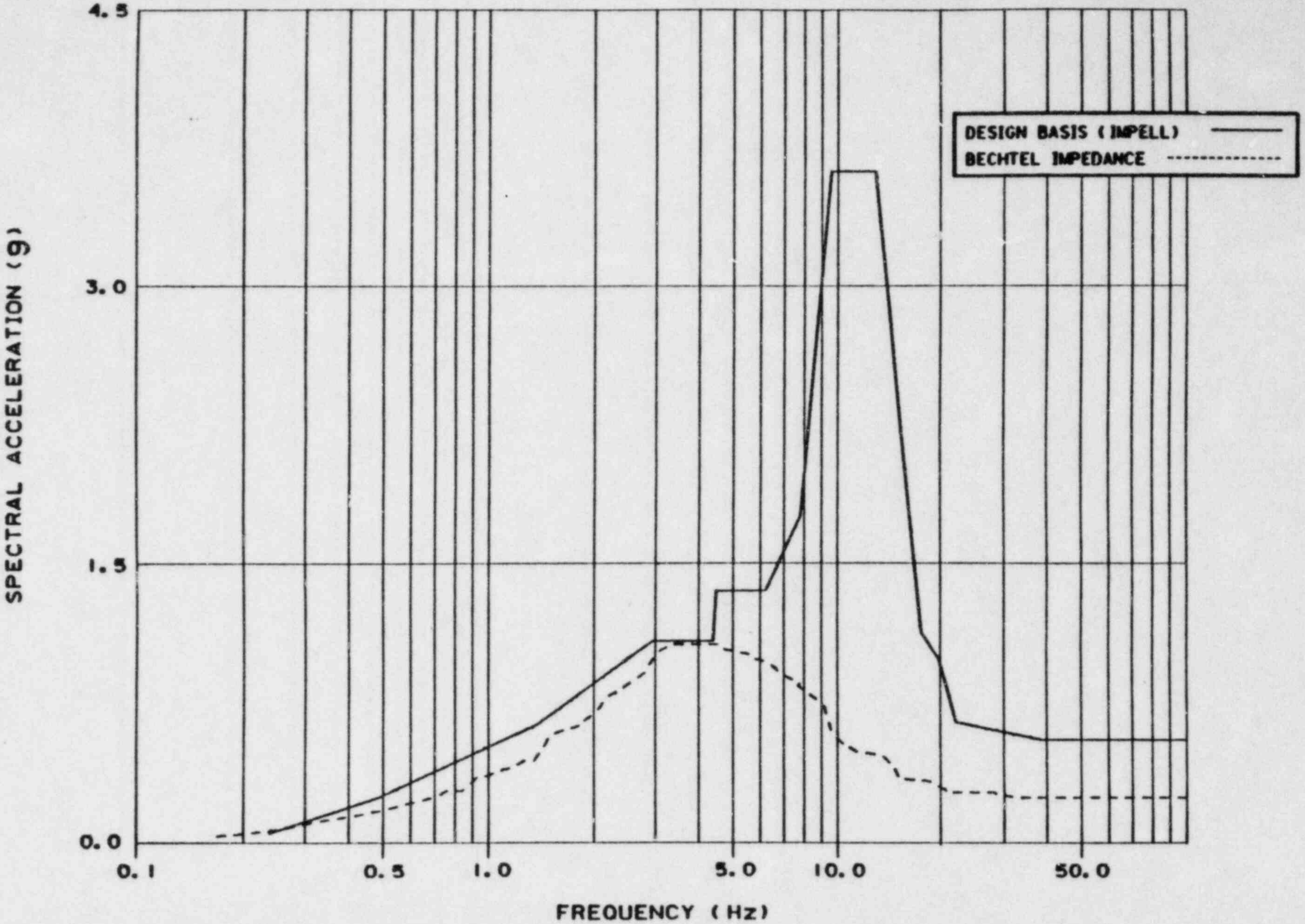


FIGURE 220.21-15

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 201'-0",
VERTICAL, SSE, 2% DAMPING

HOPE CREEK UNIT I
CAD, L27A SHT 2, REV 0
07/27/84

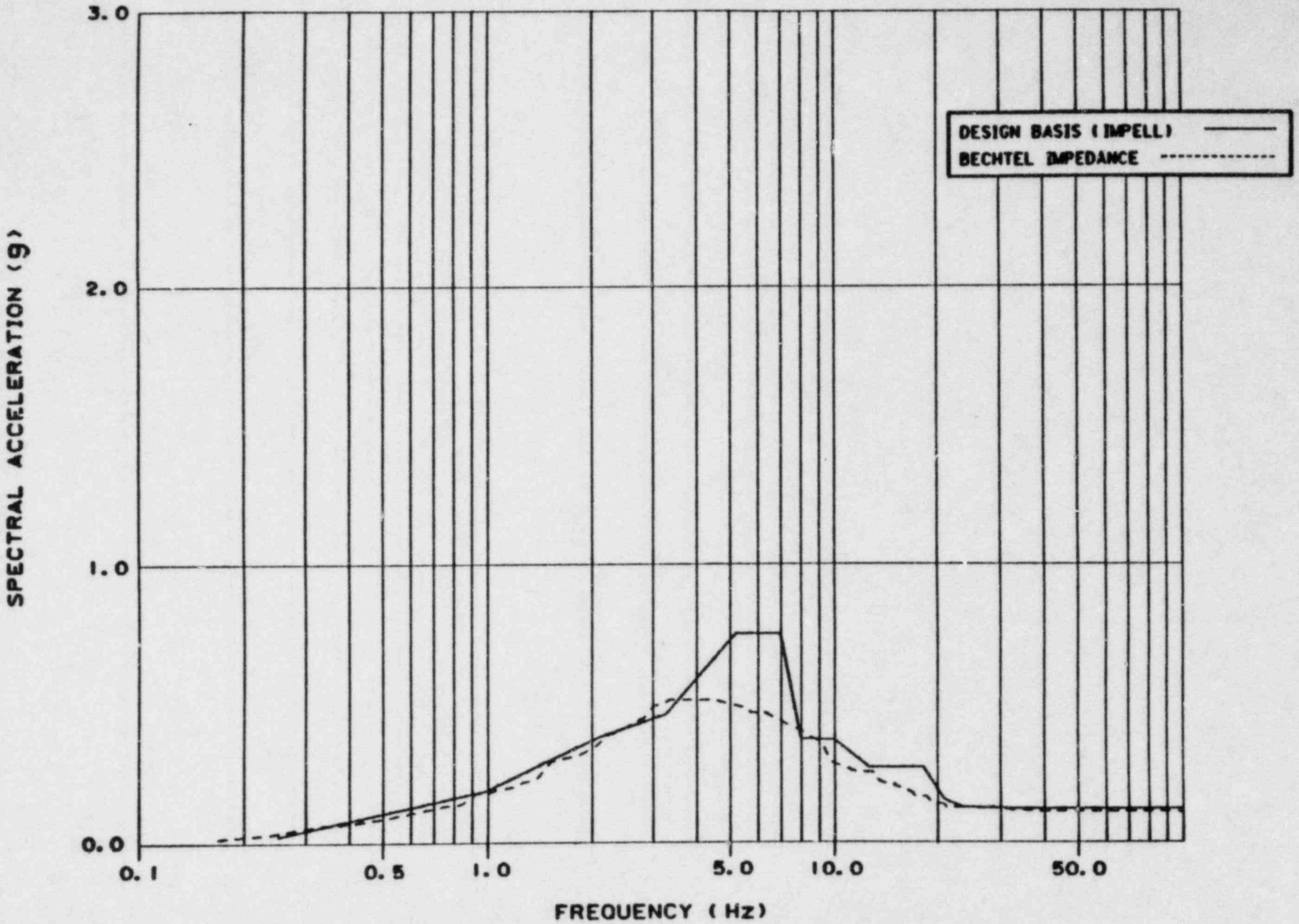


FIGURE 220.21-16

RESPONSE SPECTRA COMPARISON,
UNIT 1 REACTOR BUILDING AT ELEV. 54'-0",
VERTICAL, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD. L28A SHT 2, REV 1
07/27/84

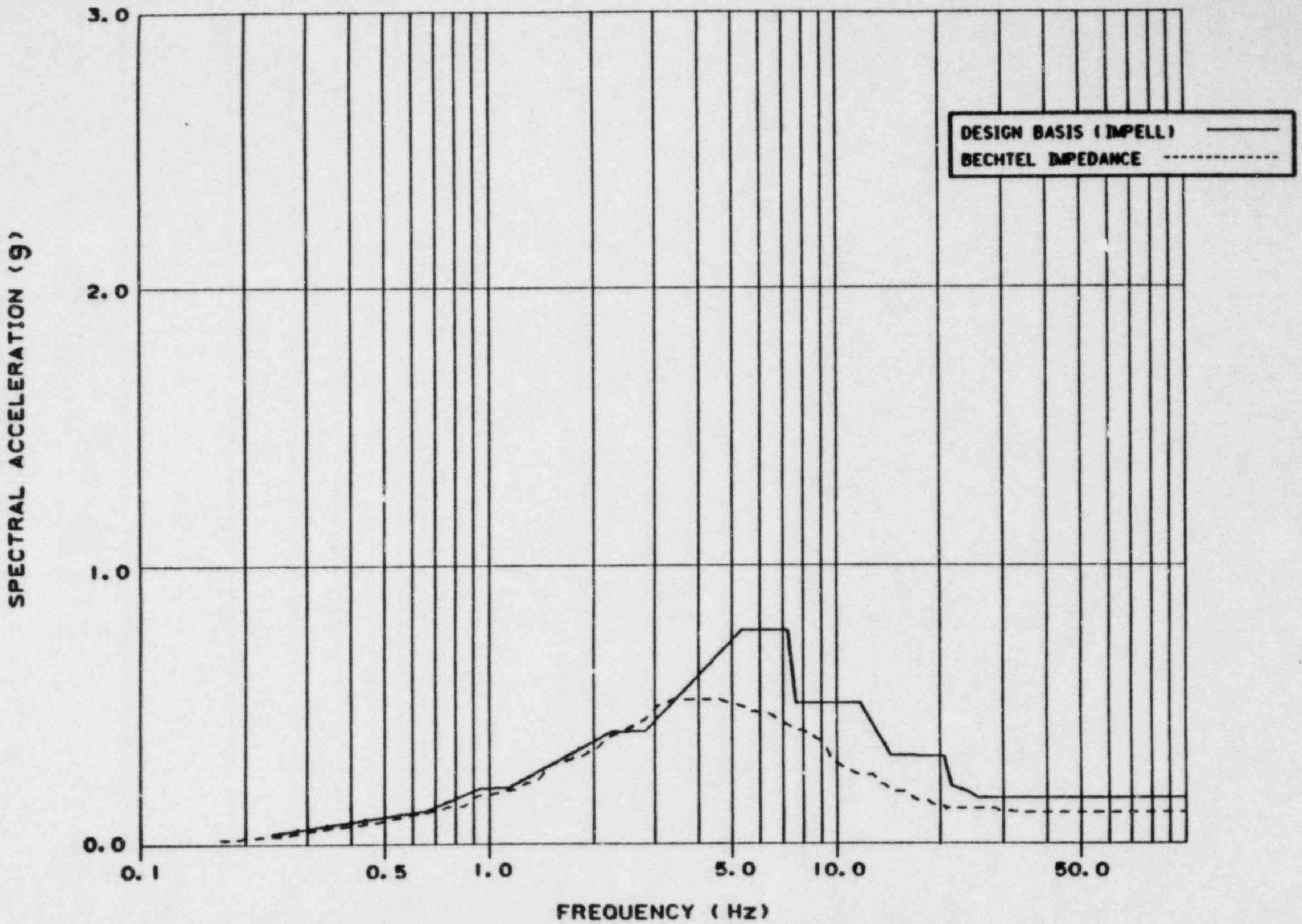


FIGURE 220.21-17

RESPONSE SPECTRA COMPARISON,
UNIT I REACTOR BUILDING AT ELEV. 102'-0",
VERTICAL, OBE, 2% DAMPING

HOPE CREEK UNIT I
CAD. L29A SHT 2, REV I
07/27/84

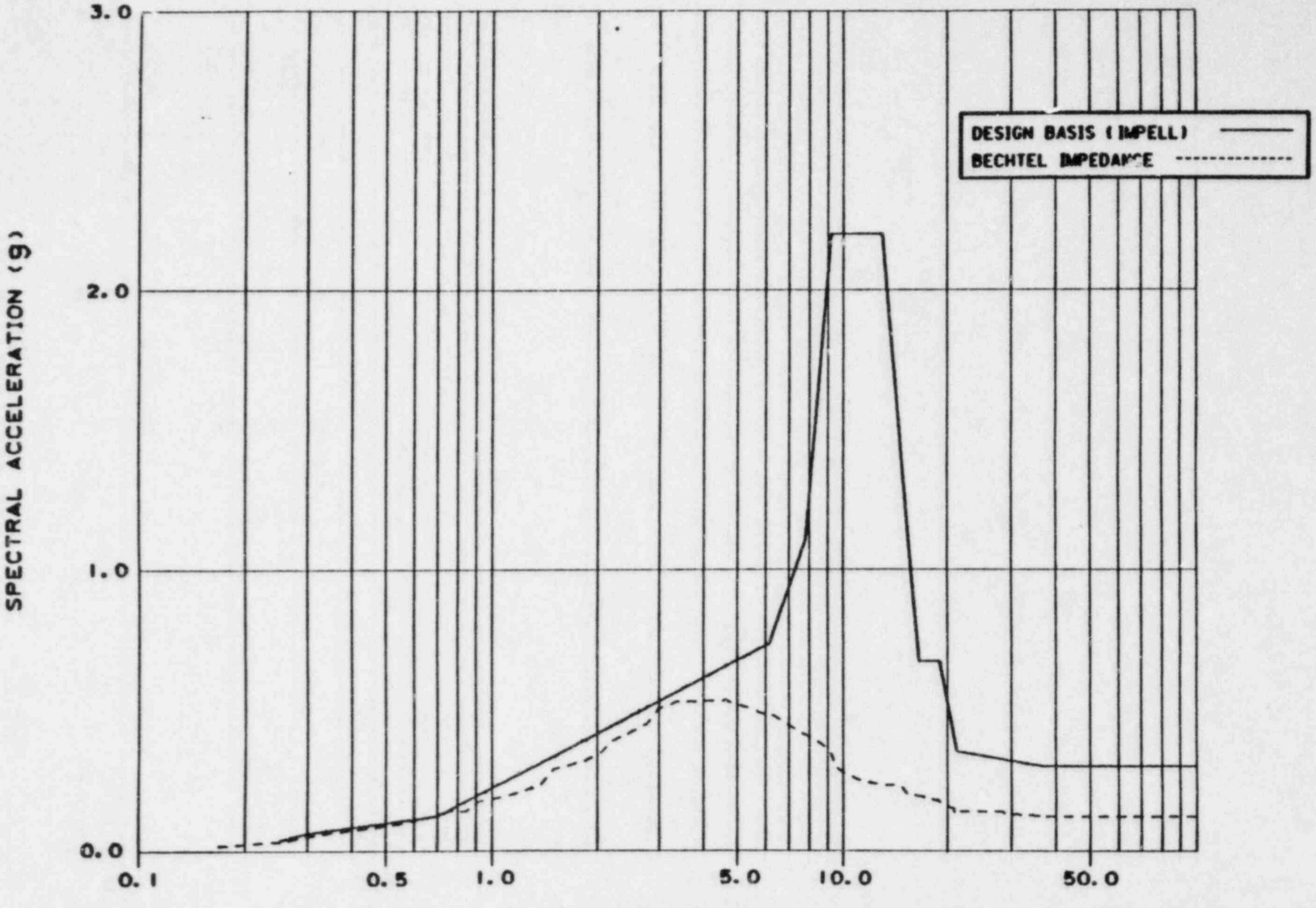


FIGURE 220.21-18

RESPONSE SPECTRA COMPARISON,
UNIT 1 REACTOR BUILDING AT ELEV. 201'-0",
VERTICAL, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD. L30A SHT 2. REV 1
07/27/84

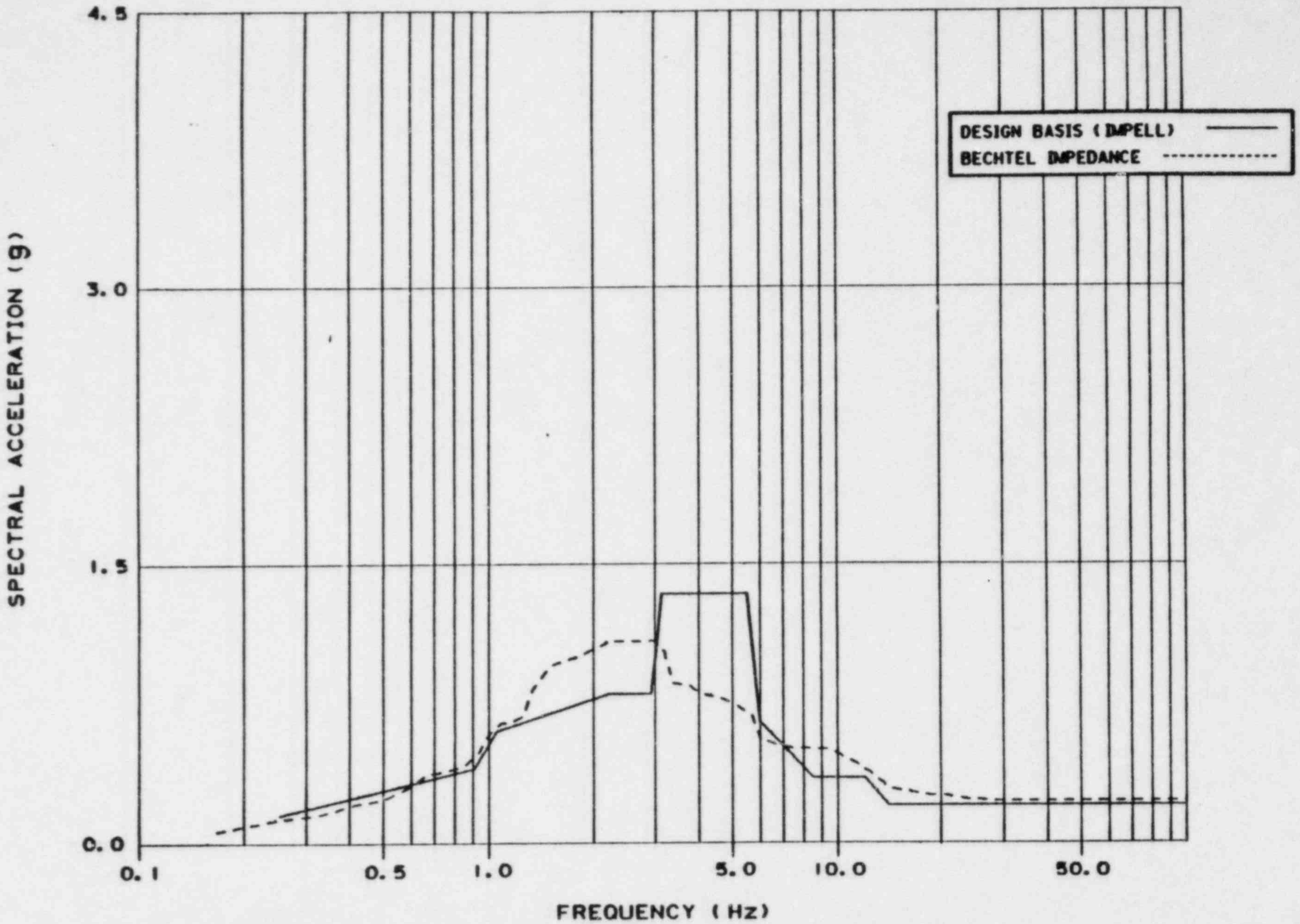


FIGURE 220.21-19

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 54'-0",
N-S, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD. L13A SHT 2, REV 0
07/27/84

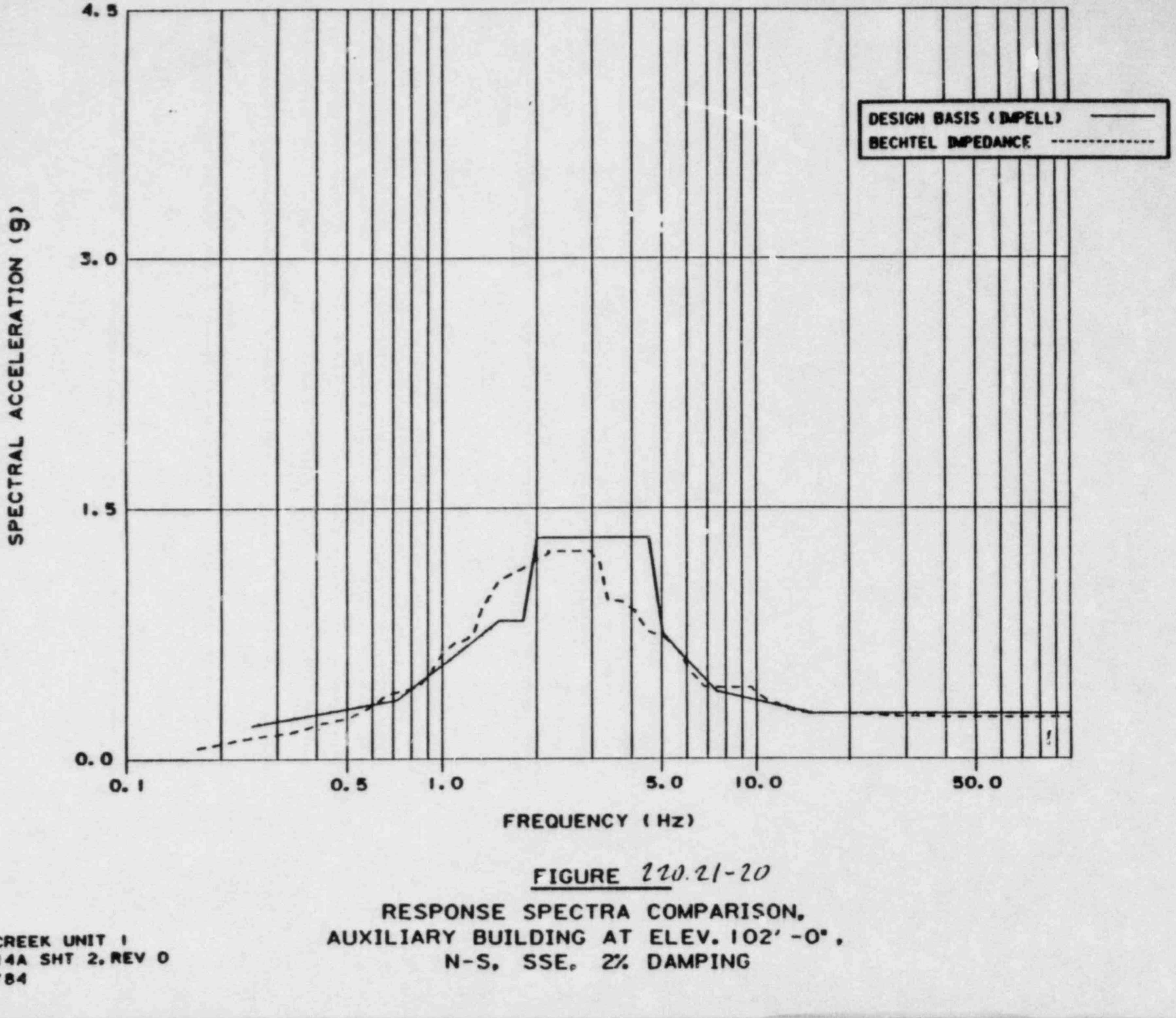


FIGURE 220.21-20

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 102'-0",
N-S, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD. L14A SHT 2, REV 0
07/27/84

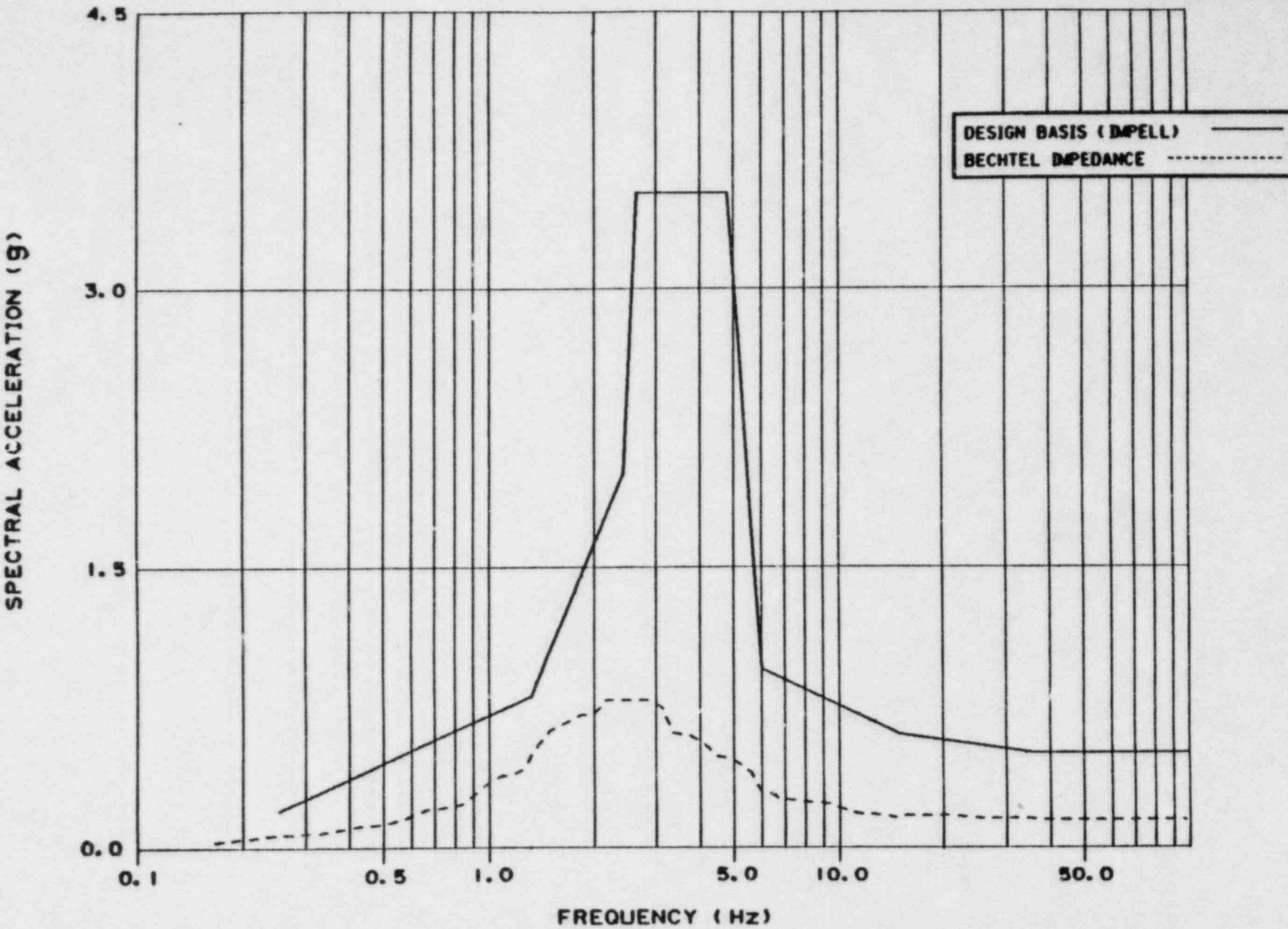


FIGURE 220.21-21

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 178'-0",
N-S, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD. L15A SHT 2, REV 0
07/27/84

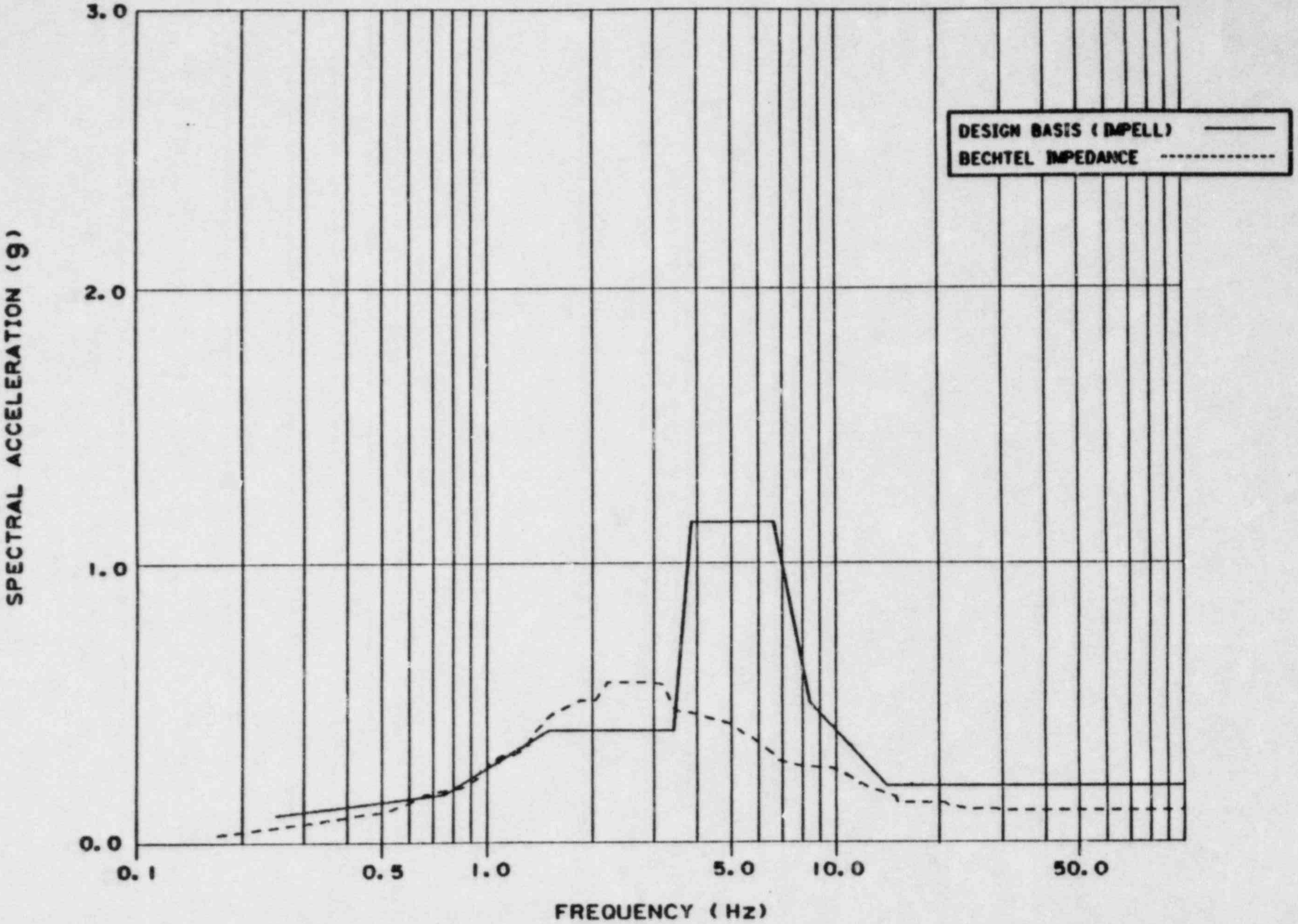


FIGURE 220.21-22

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 54'-0",
N-S, OBE, 2% DAMPING

HOPE CREEK UNIT I
CAD. L16A SHT 2, REV 0
07/27/84

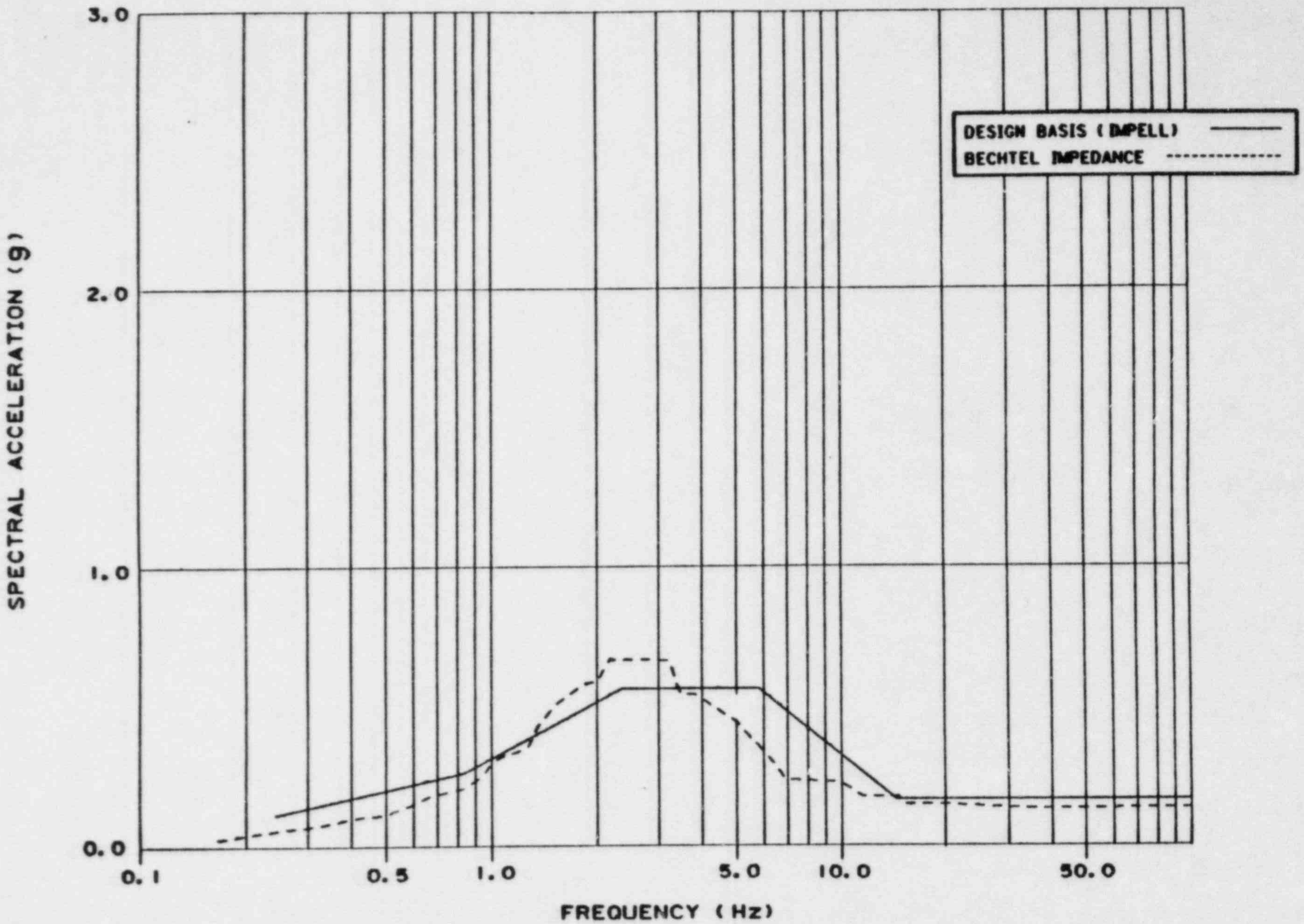


FIGURE 220.21-23

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 102'-0",
N-S, OBE, 2% DAMPING

HOPE CREEK UNIT I
CAD, LITA SHT 2, REV 0
07/27/84

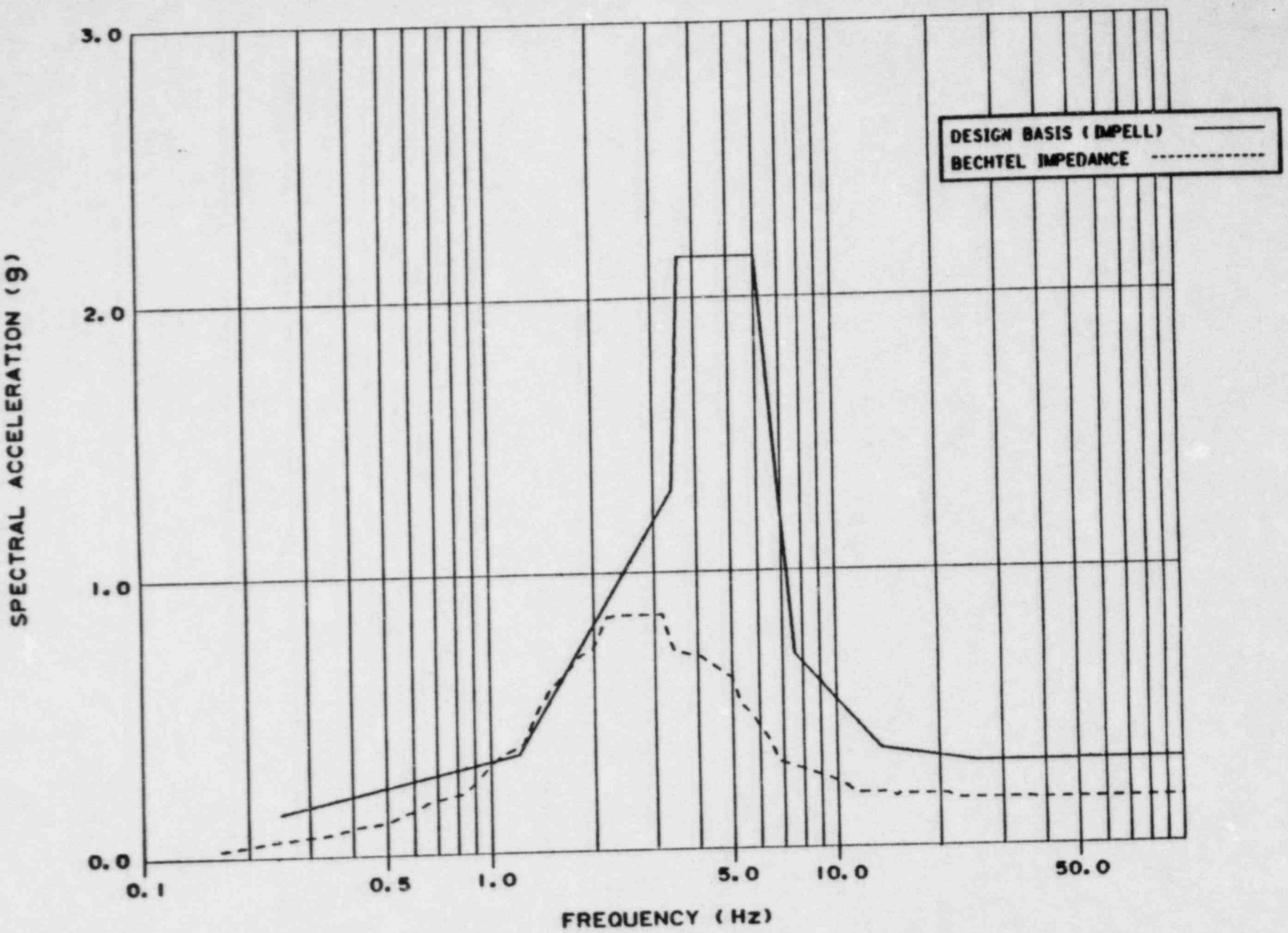


FIGURE 220.21-24

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 178'-0",
N-S, OBE, 2% DAMPING

HOPE CREEK UNIT I
CAD, L18A SHT 2, REV 0
07/27/84

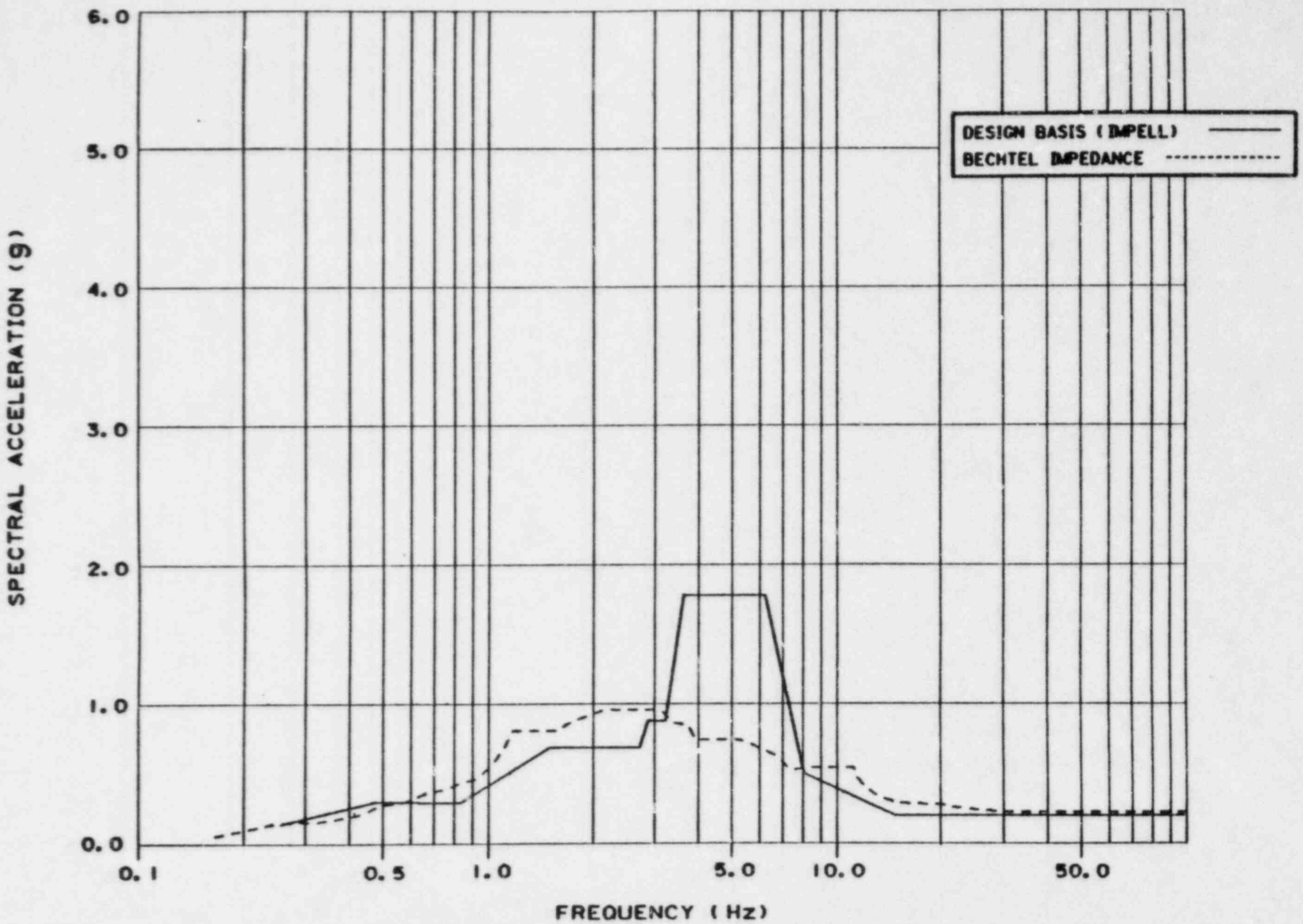


FIGURE 220.21-25

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 54'-0",
E-W, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L19A SHT 2, REV 0
07/27/84

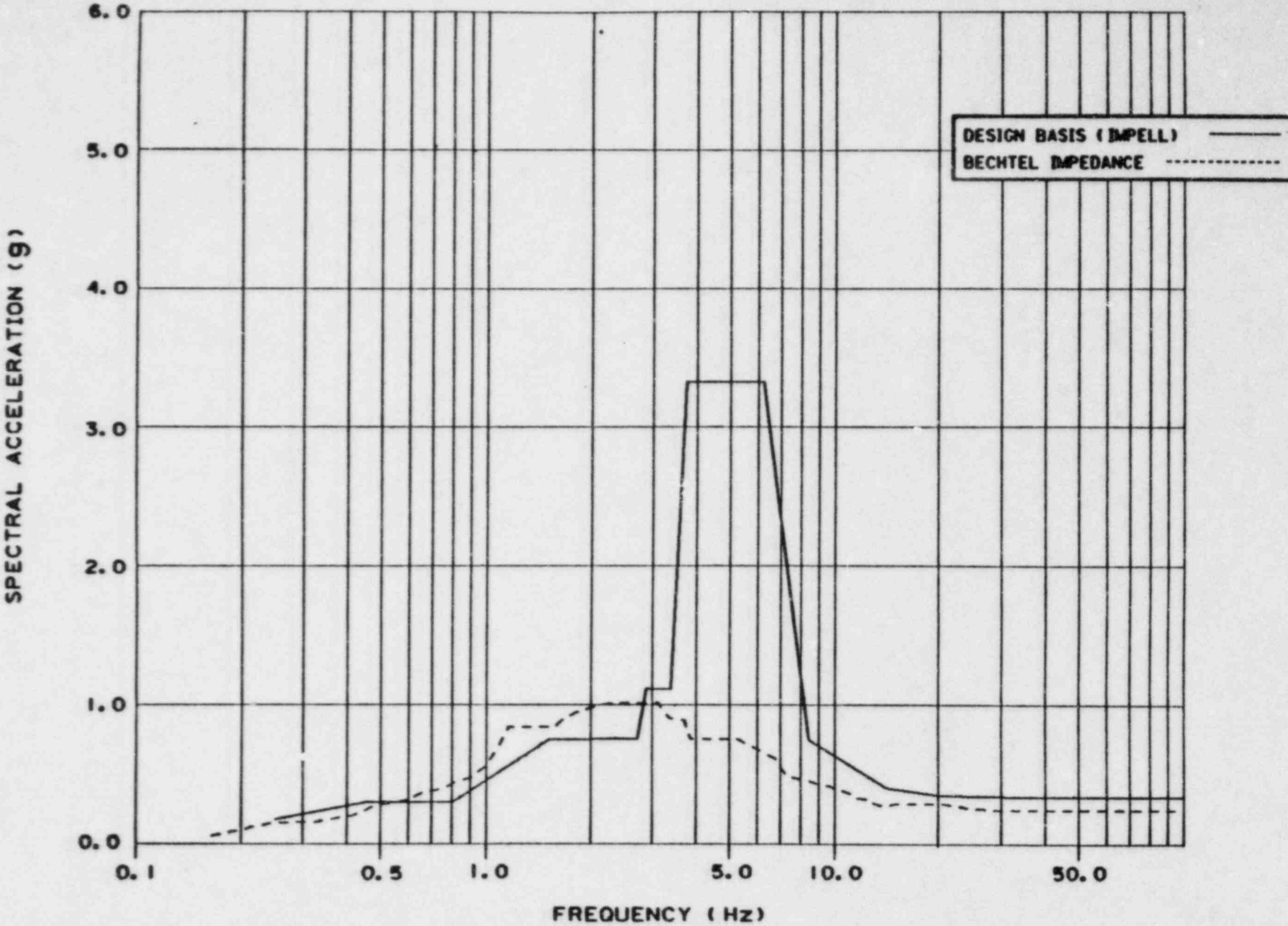


FIGURE 220.21-26

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 102'-0",
E-W, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L20A SHT 2, REV 0
07/27/84

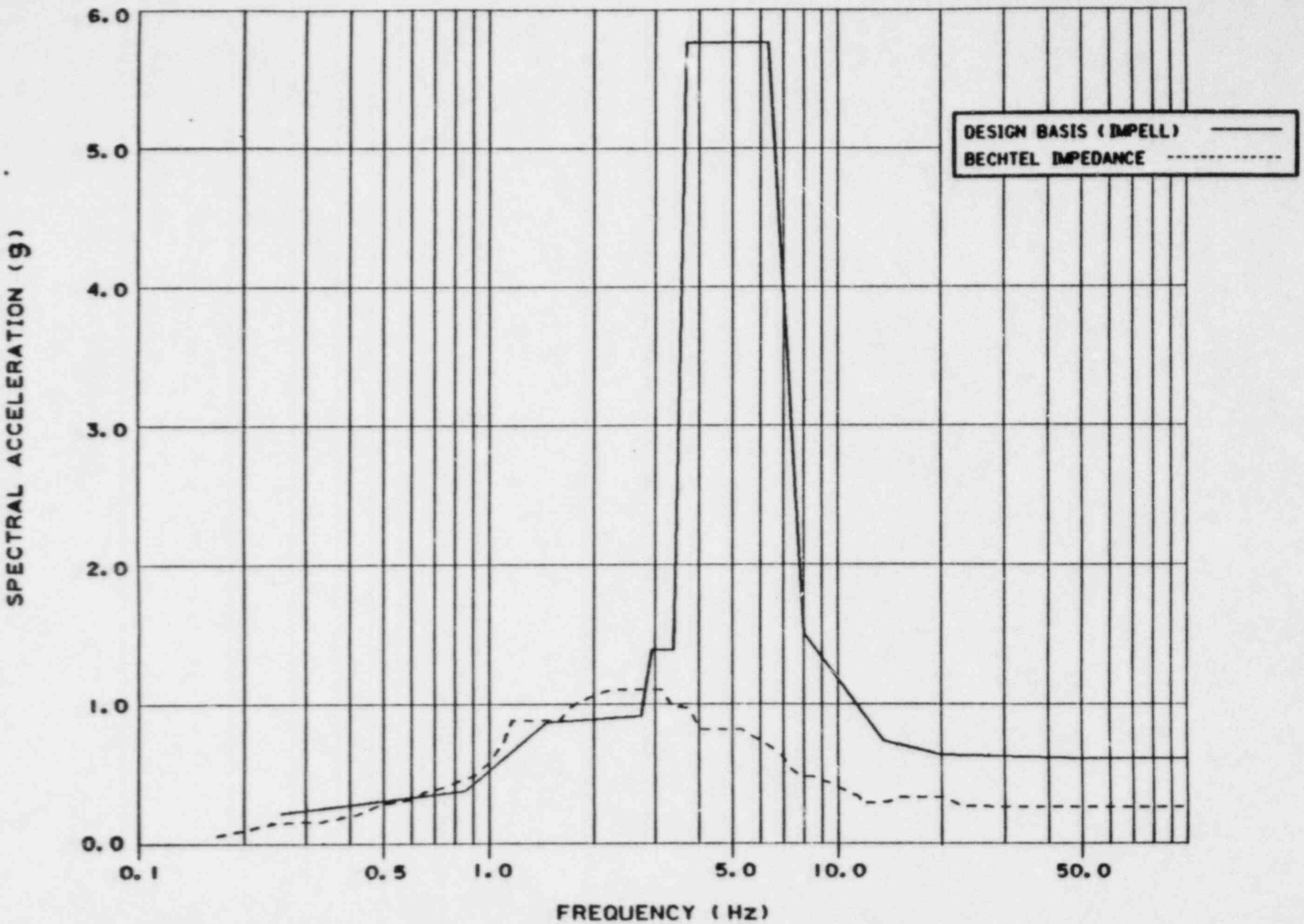


FIGURE 220.21-27

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 178'-0",
E-W, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L21A SHT 2, REV 0
07/27/84

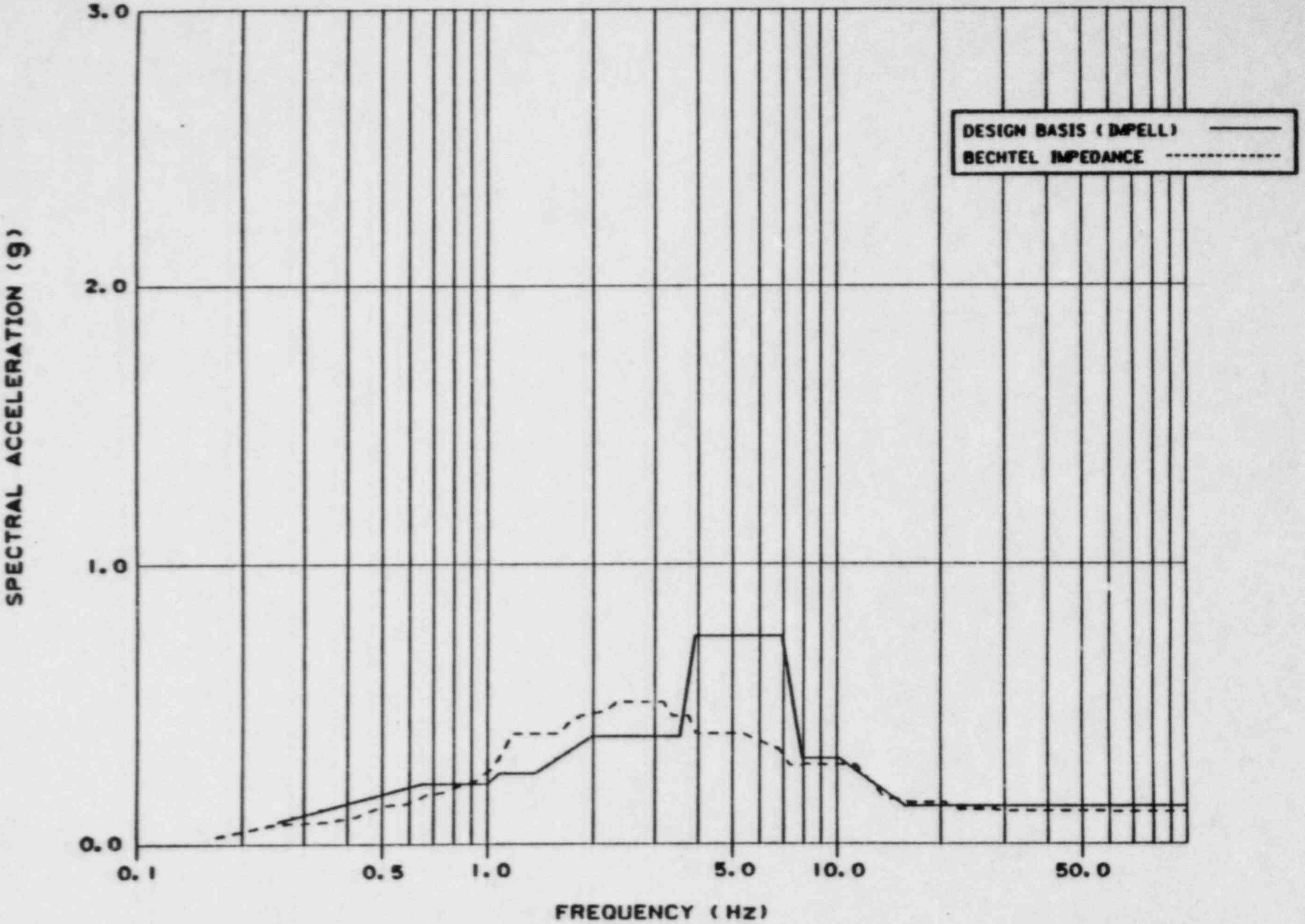


FIGURE 220.21-2B

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 54'-0",
E-W, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD. L22A SHT 2, REV 0
07/27/84

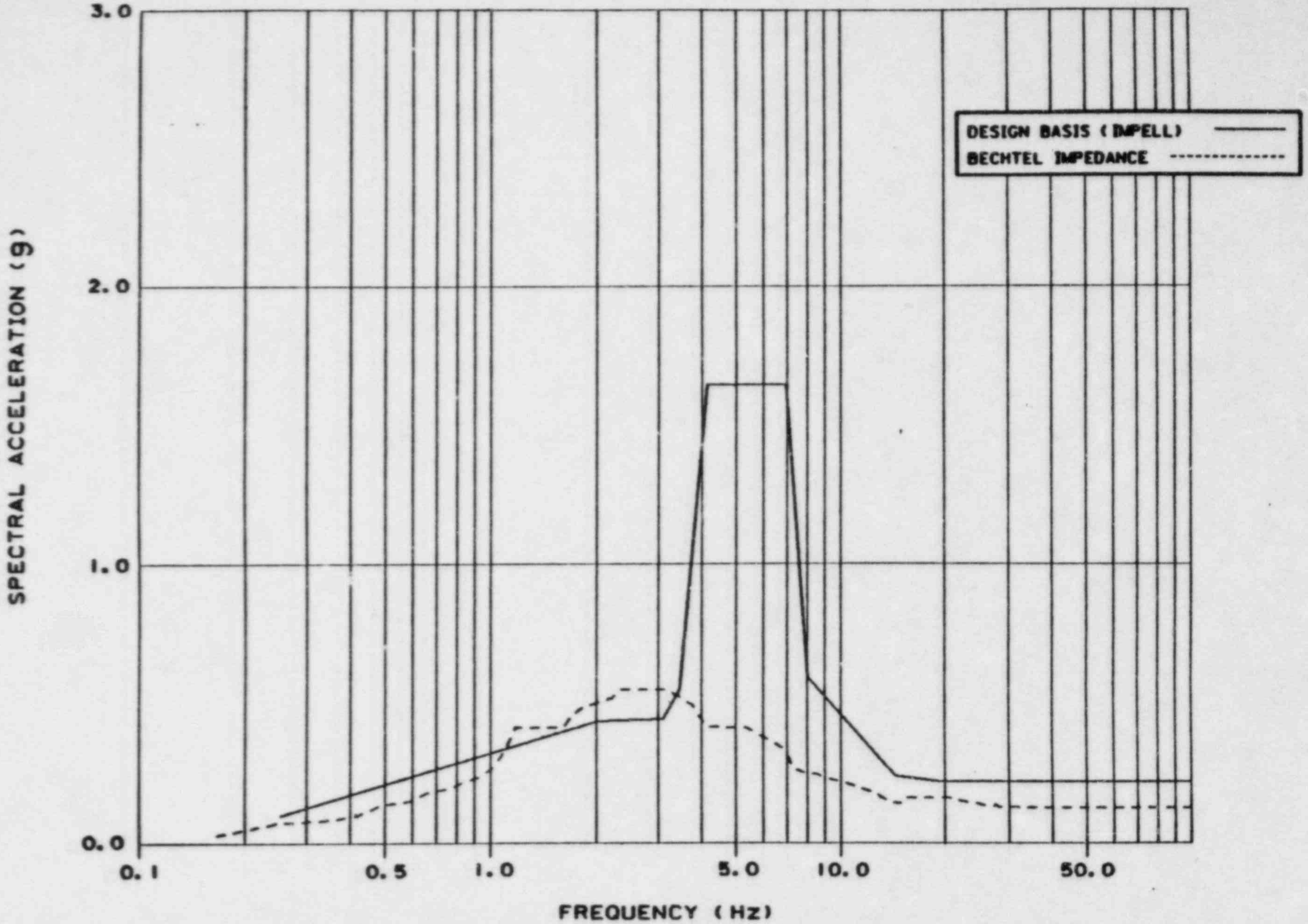


FIGURE 220.21-29

RESPONSE SPECTRA COMPARISON, AUXILIARY BUILDING AT ELEV. 102' -0", E-W, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L23A SHT 2, REV 0
07/27/84

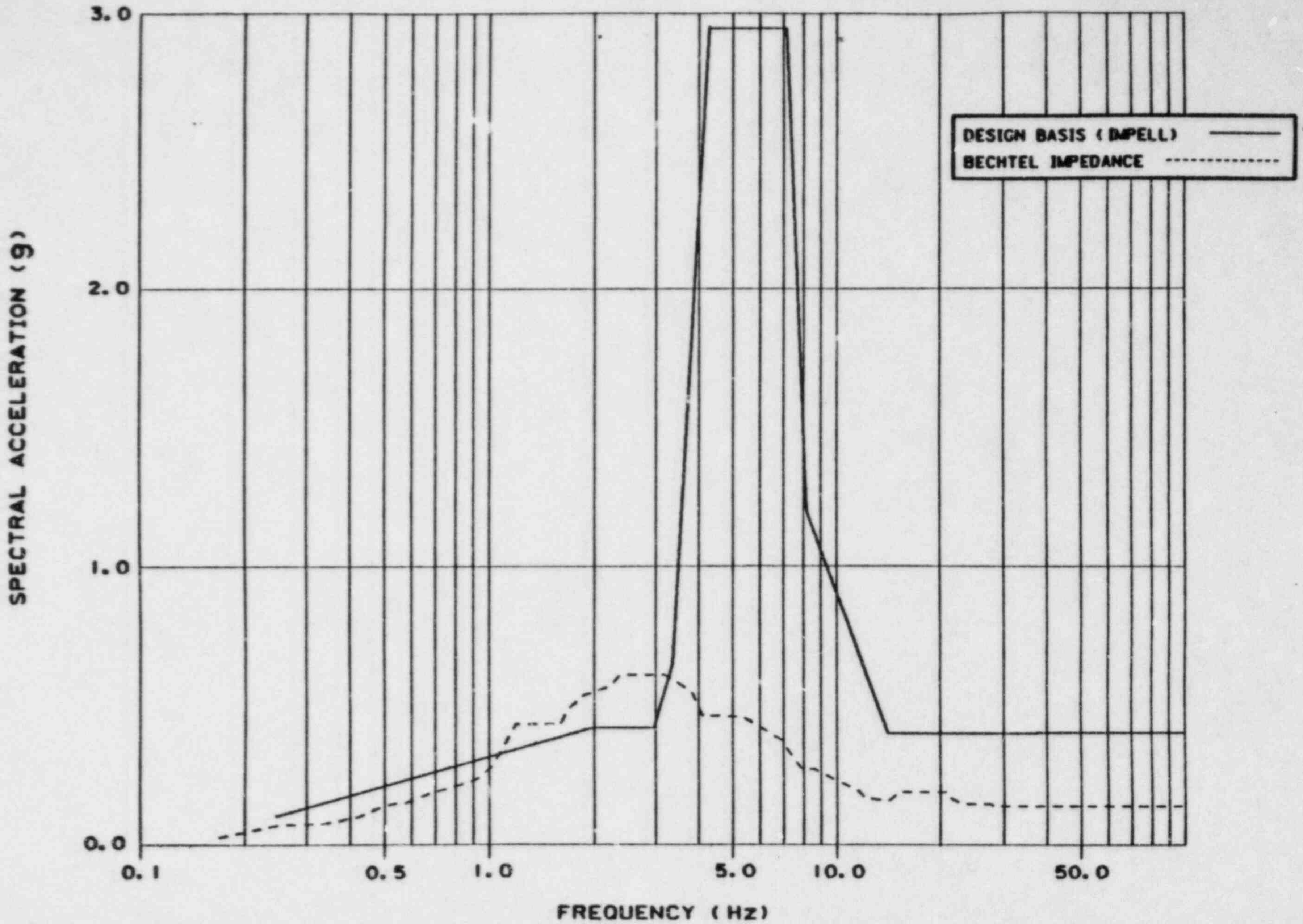


FIGURE 220.21-30

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 178'-0",
E-W, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD. L24A SHT 2, REV 0
07/27/84

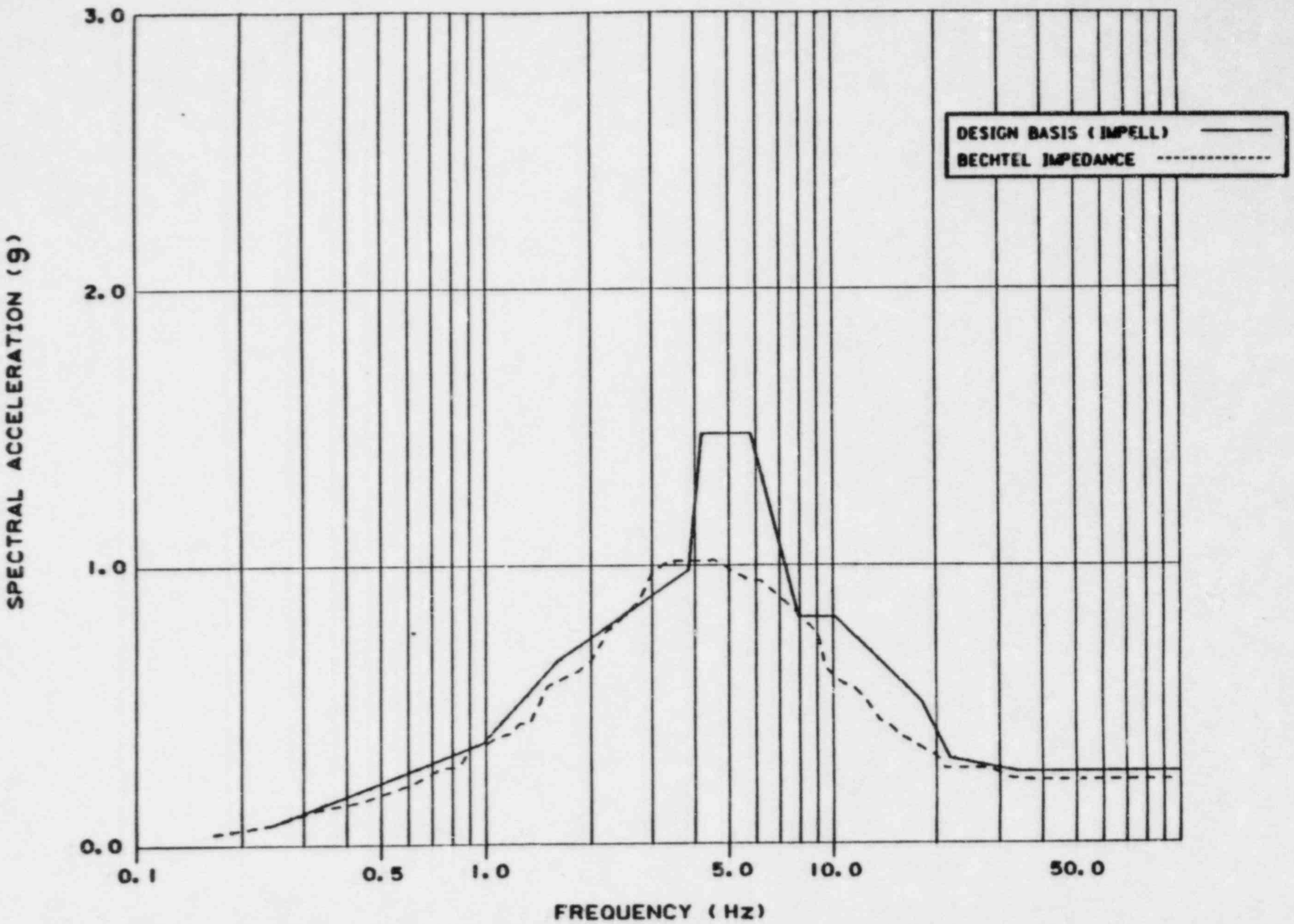


FIGURE 220.21-31

RESPONSE SPECTRA COMPARISON, AUXILIARY BUILDING AT ELEV. 54'-0", VERTICAL, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD. L31A SHT 2, REV 0
07/27/84

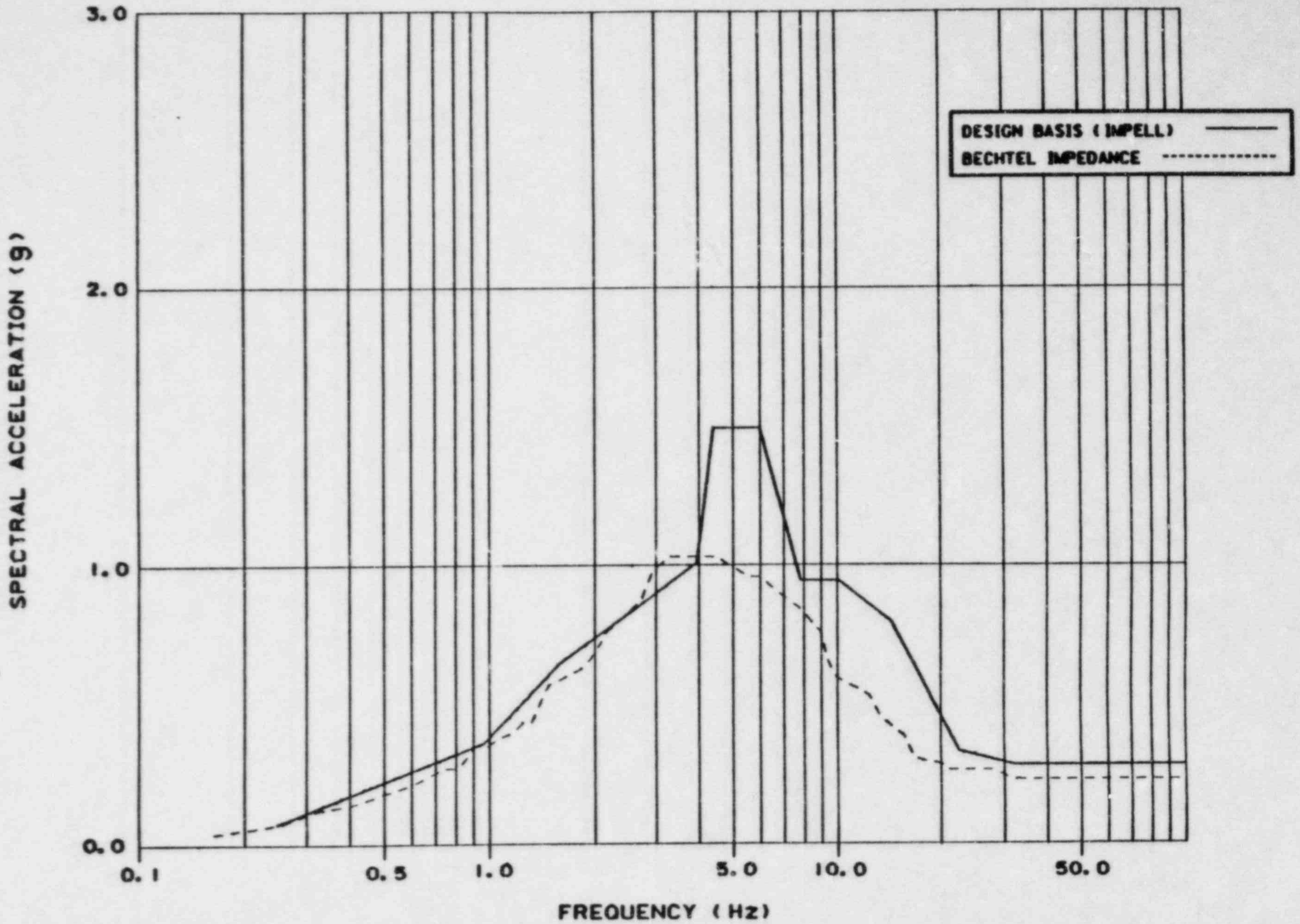


FIGURE 220.21-32

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 102'-0",
VERTICAL, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L32A SHT 2, REV 0
07/27/84

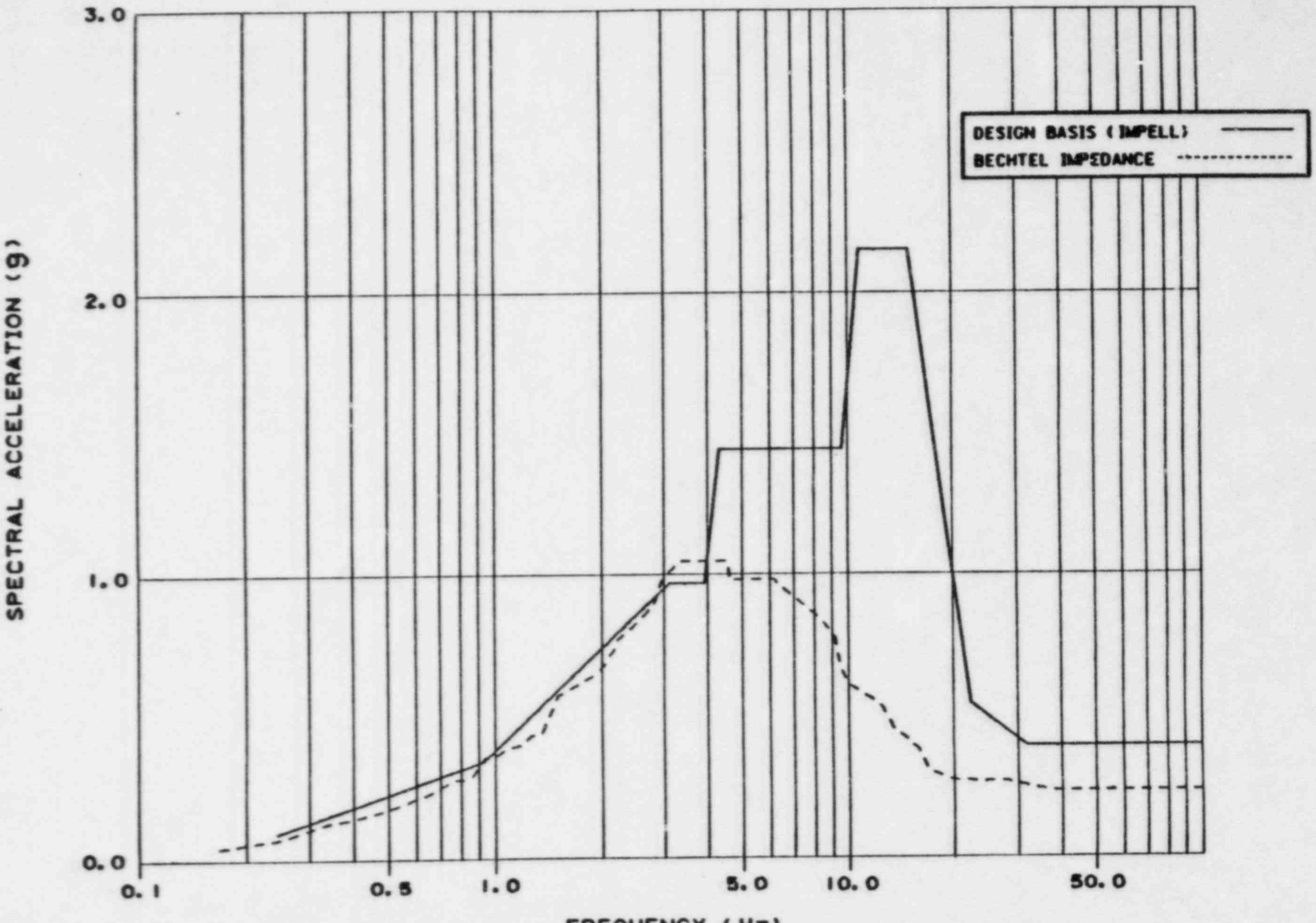


FIGURE 220.21-33

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 178'-0",
VERTICAL. SSE, 2% DAMPING

HOPE CREEK UNIT I
CAD, L33A SHT 2, REV 0
07/27/84

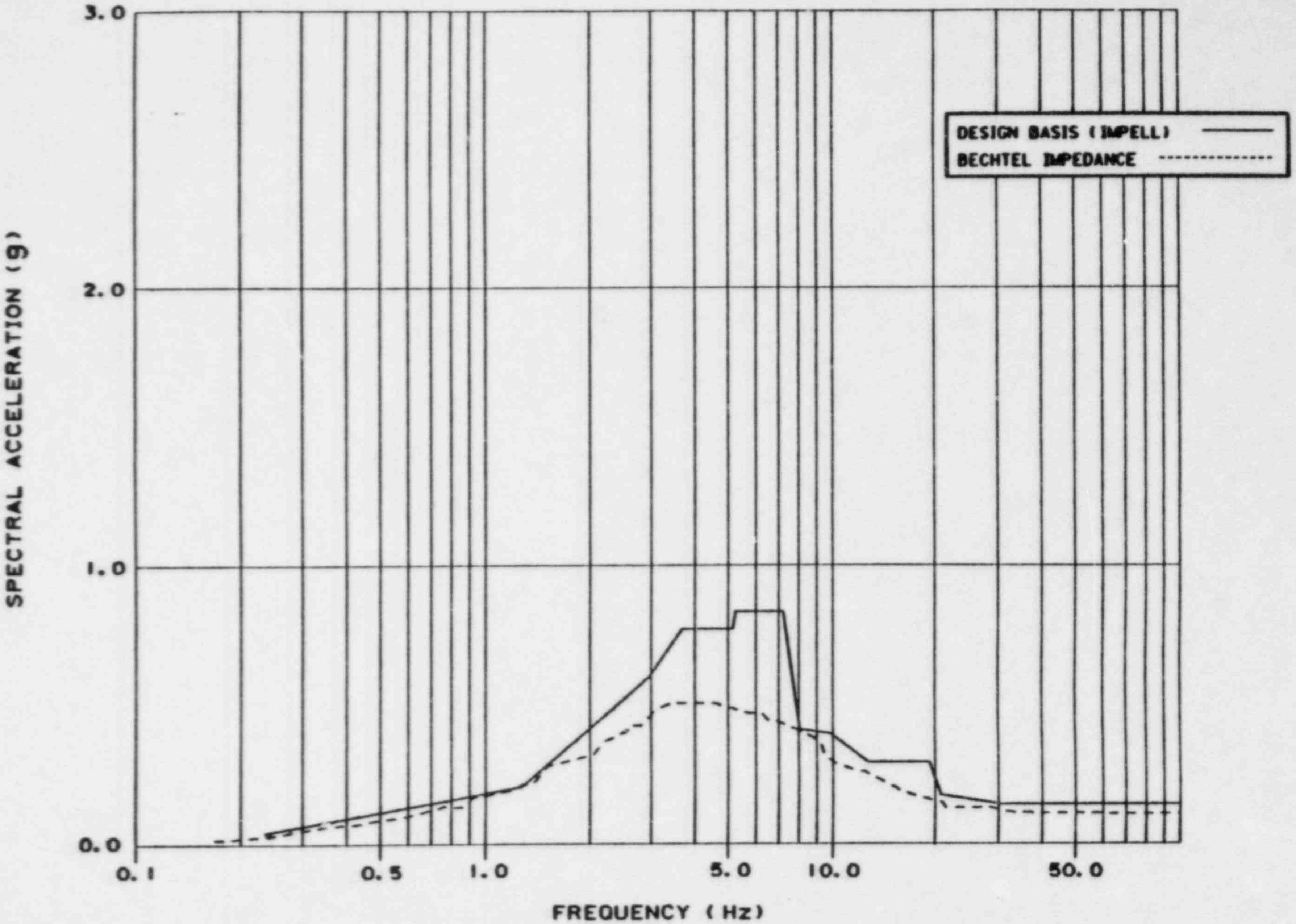


FIGURE 220.21-34

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 54'-0",
VERTICAL, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD. L34A SHT 2, REV 0
07/27/84

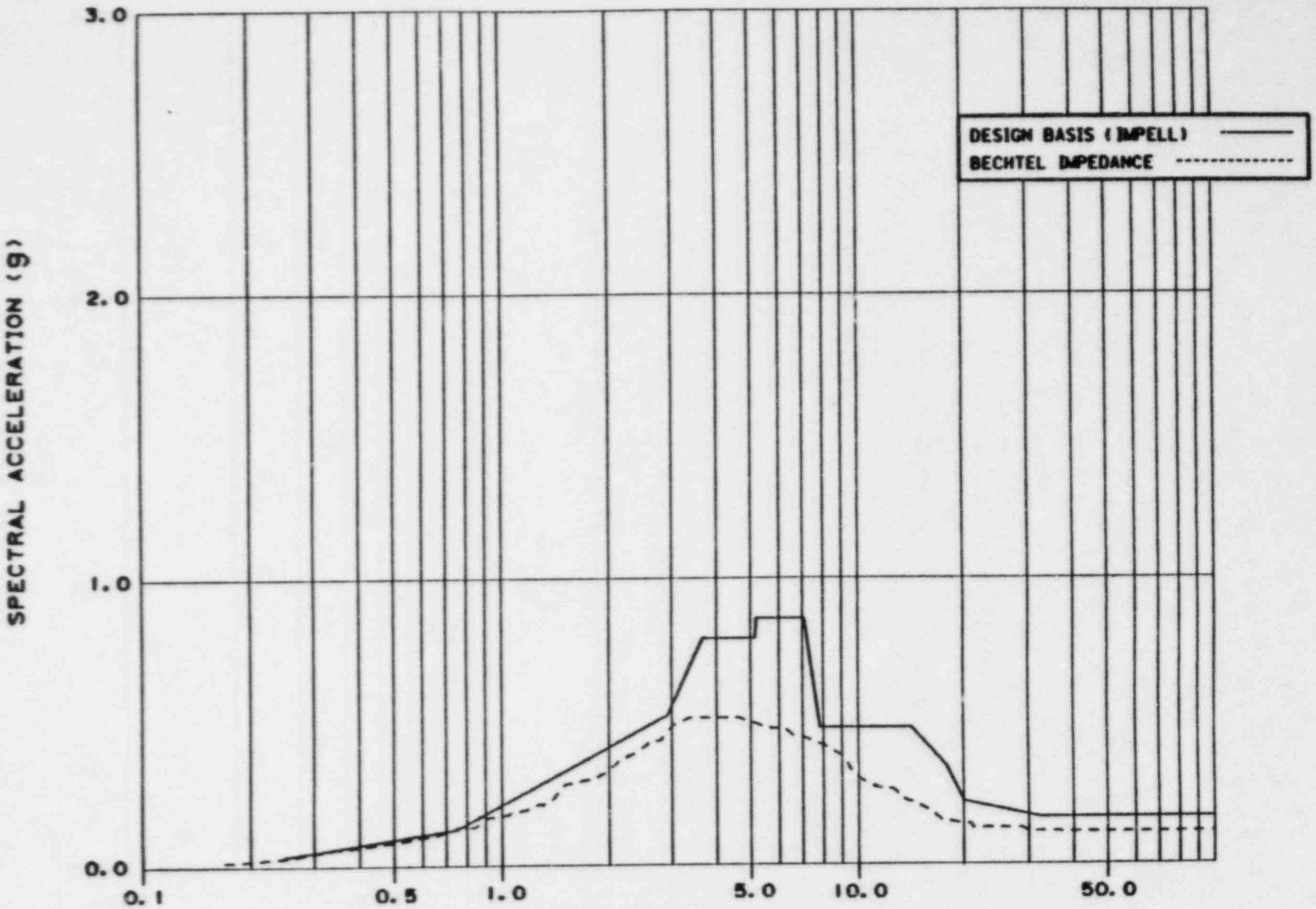


FIGURE 220.21-35

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 102' -0",
VERTICAL, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD. L35A SHT 2, REV 0

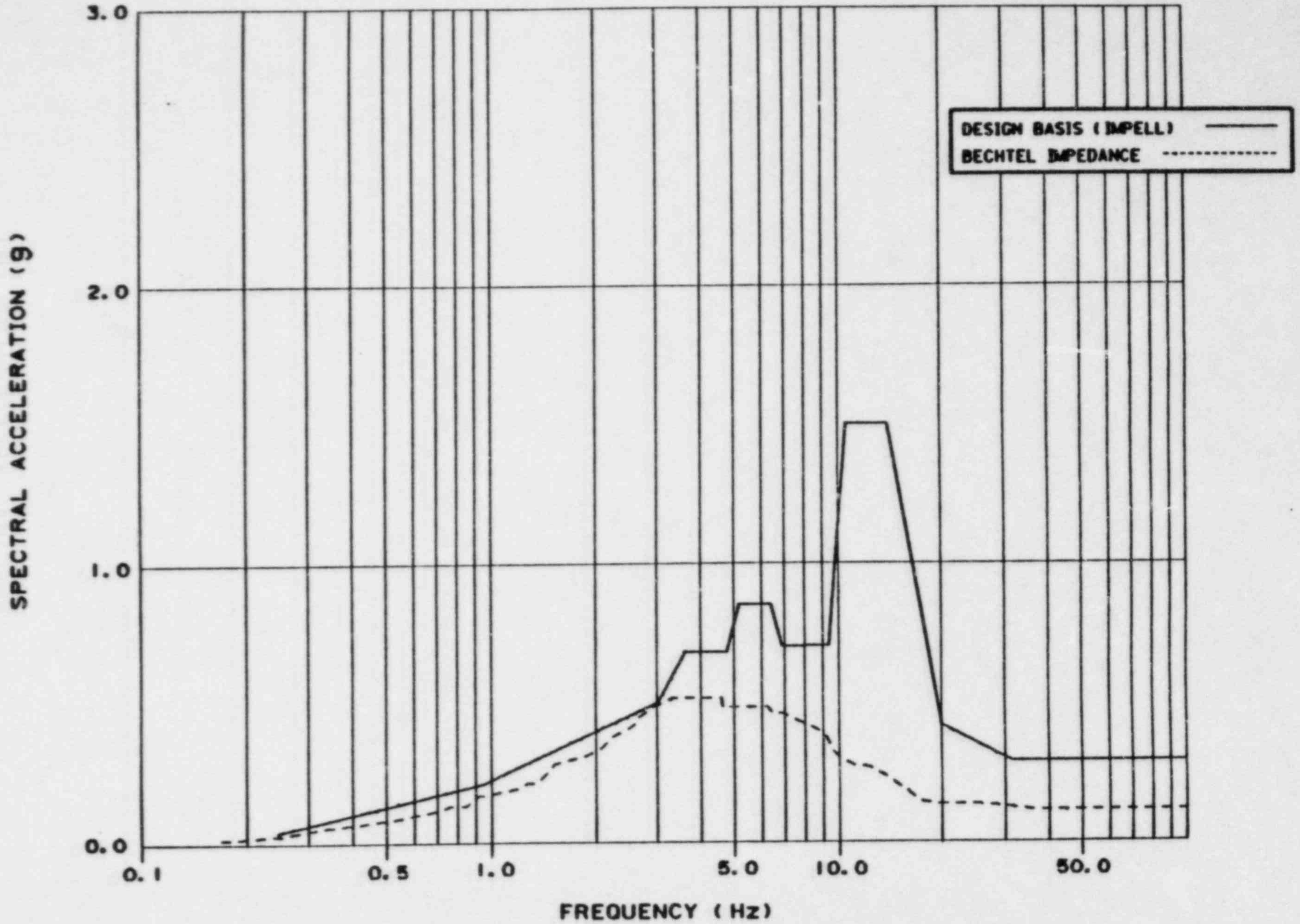


FIGURE 220.21-36

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 178'-0",
VERTICAL, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L36A SHT 2, REV 0
07/27/84

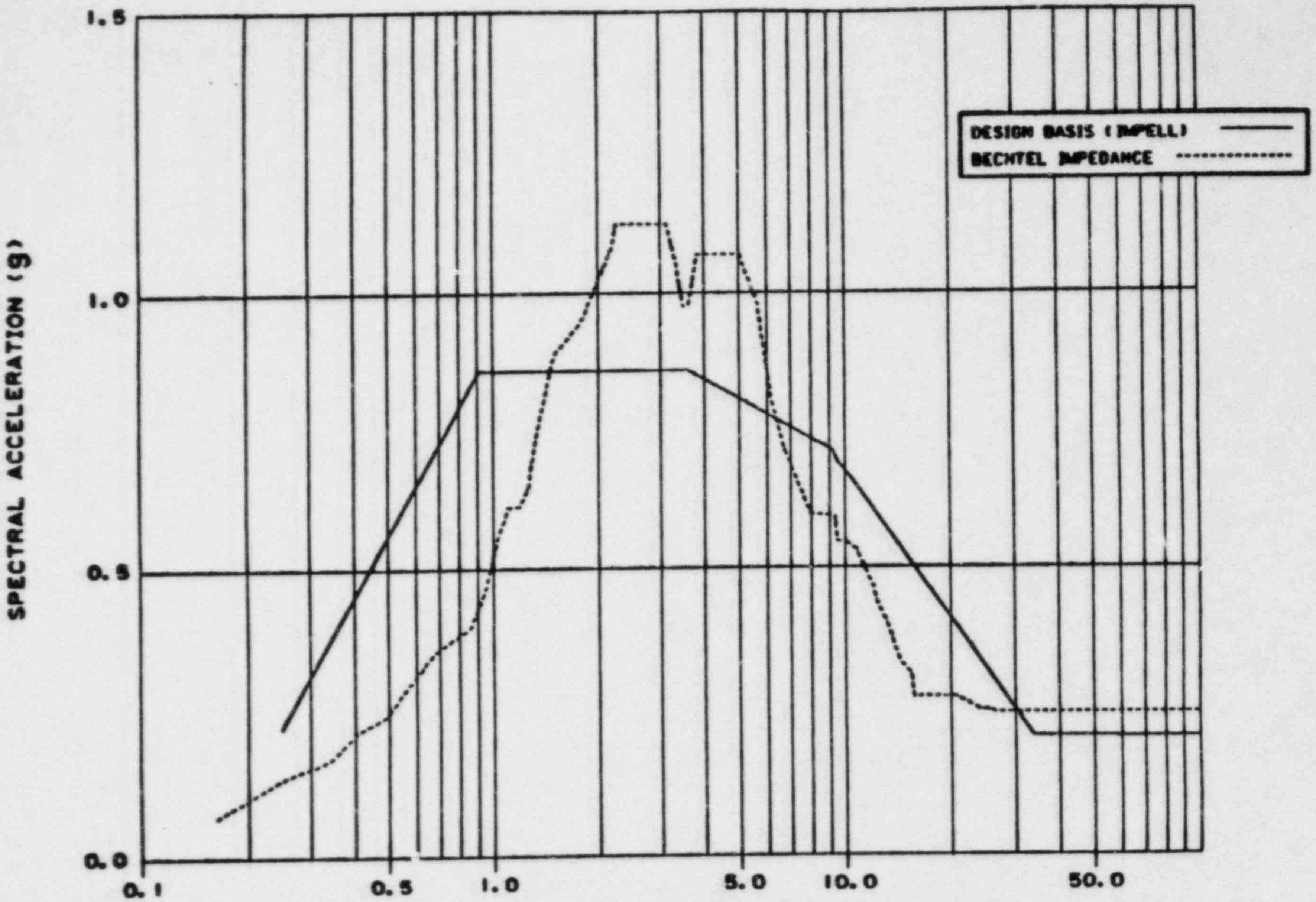


FIGURE ~~A-16-1~~ 220.21-37

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 93'-0",
N-S, SSE, 2% DAMPING

HOPE CREEK UNIT I
CAD, L39, REV 0

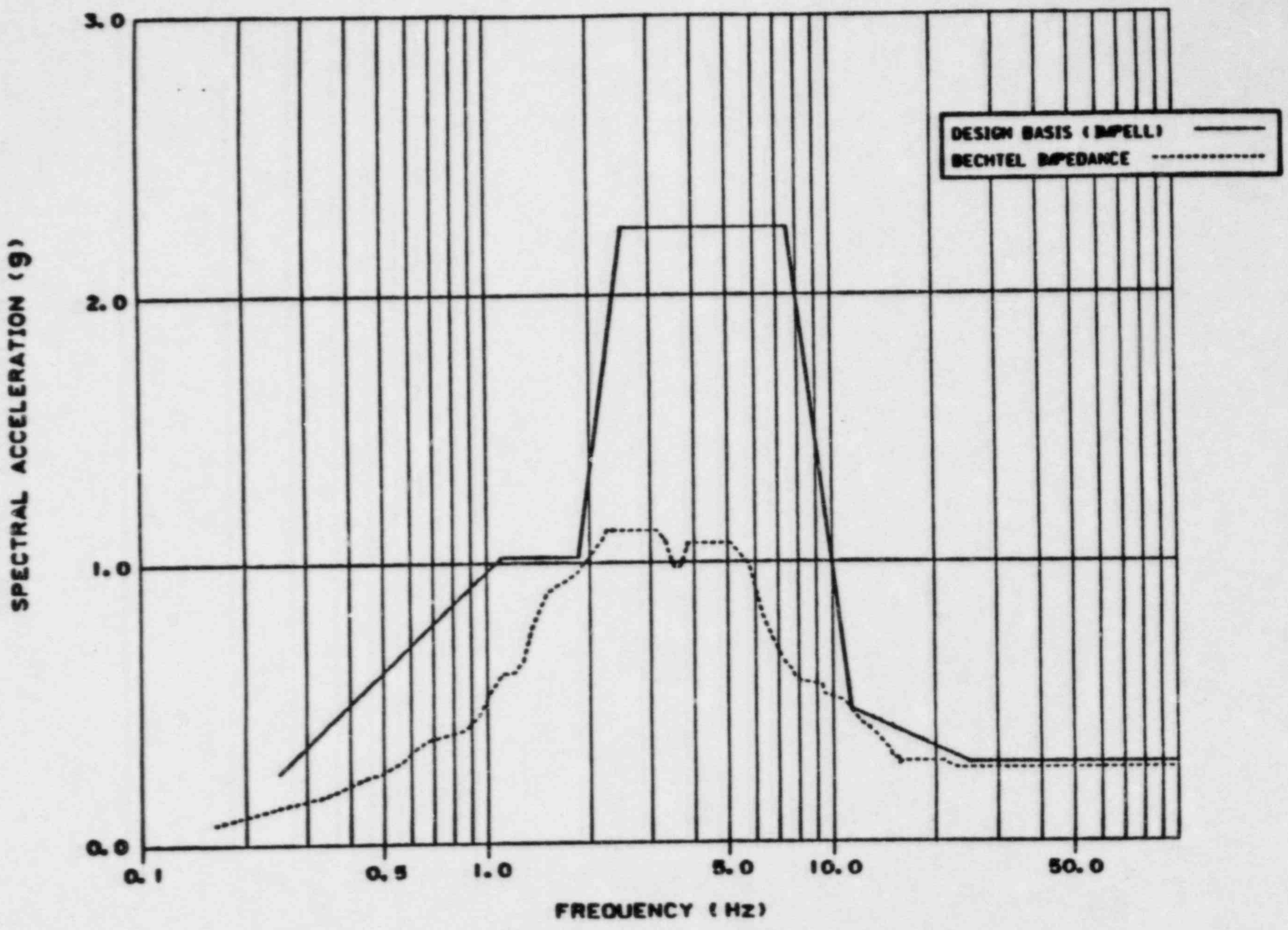


FIGURE ~~A-16-2~~ 220.21-3B

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 114' -0",
N-S, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L44, REV 0

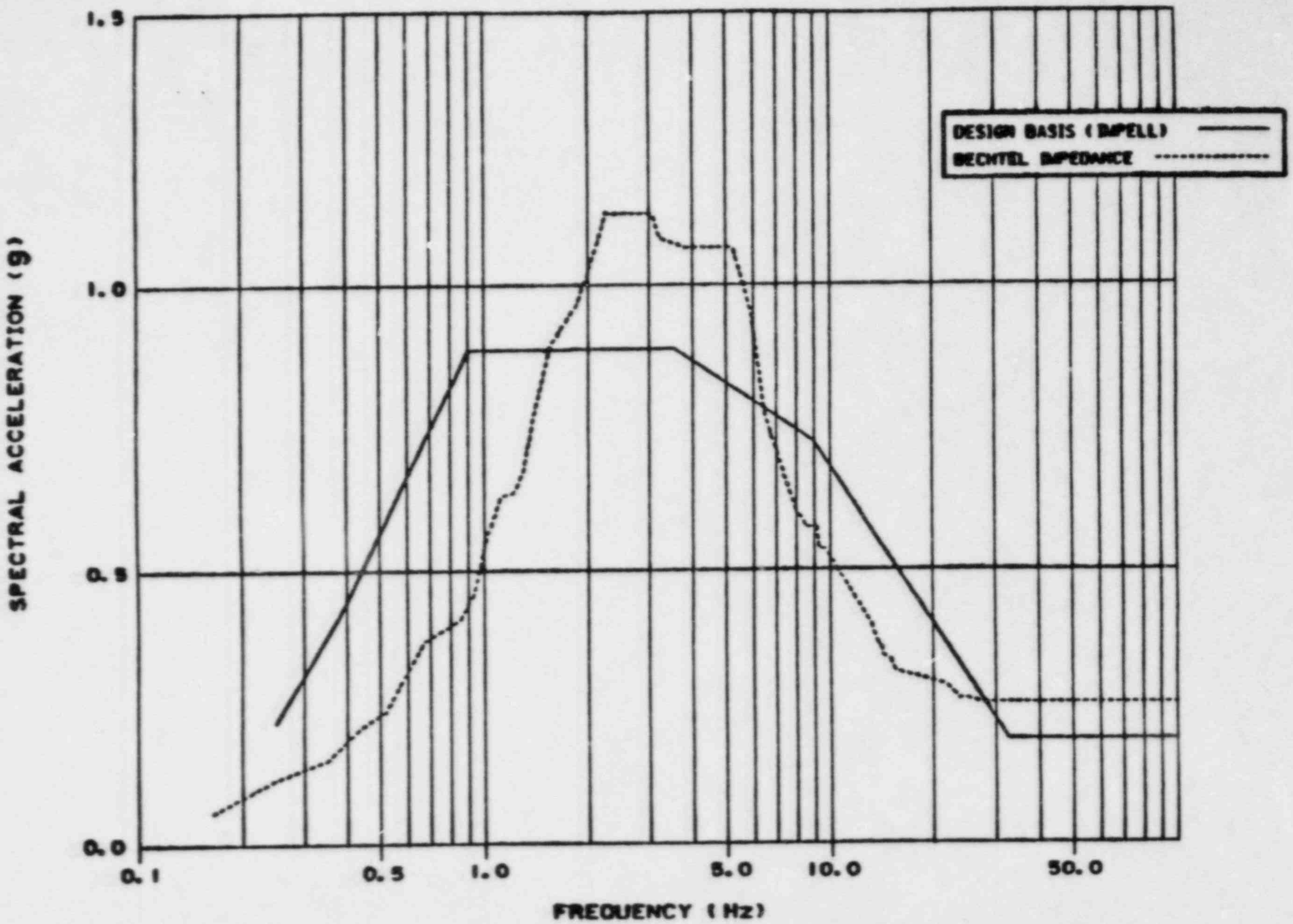


FIGURE ~~[illegible]~~ 220.21-39

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 135'-0",
N-S, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L52, REV 0

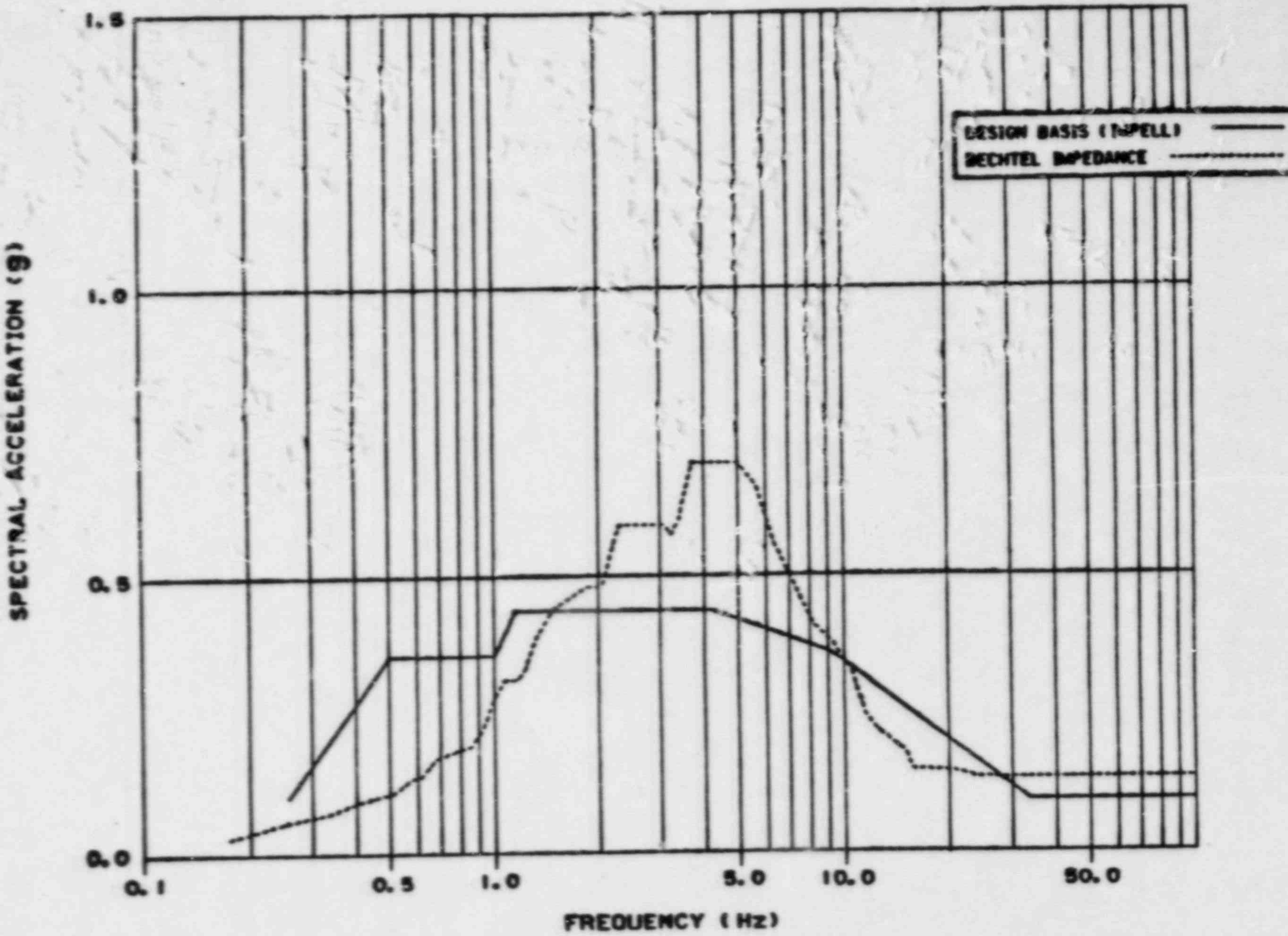


FIGURE ~~2.16-10~~ 220.21-40

RESPONSE SPECTRA COMPARISON, INTAKE STRUCTURE AT ELEV. 93'-0", N-S, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD. LST. REV 0

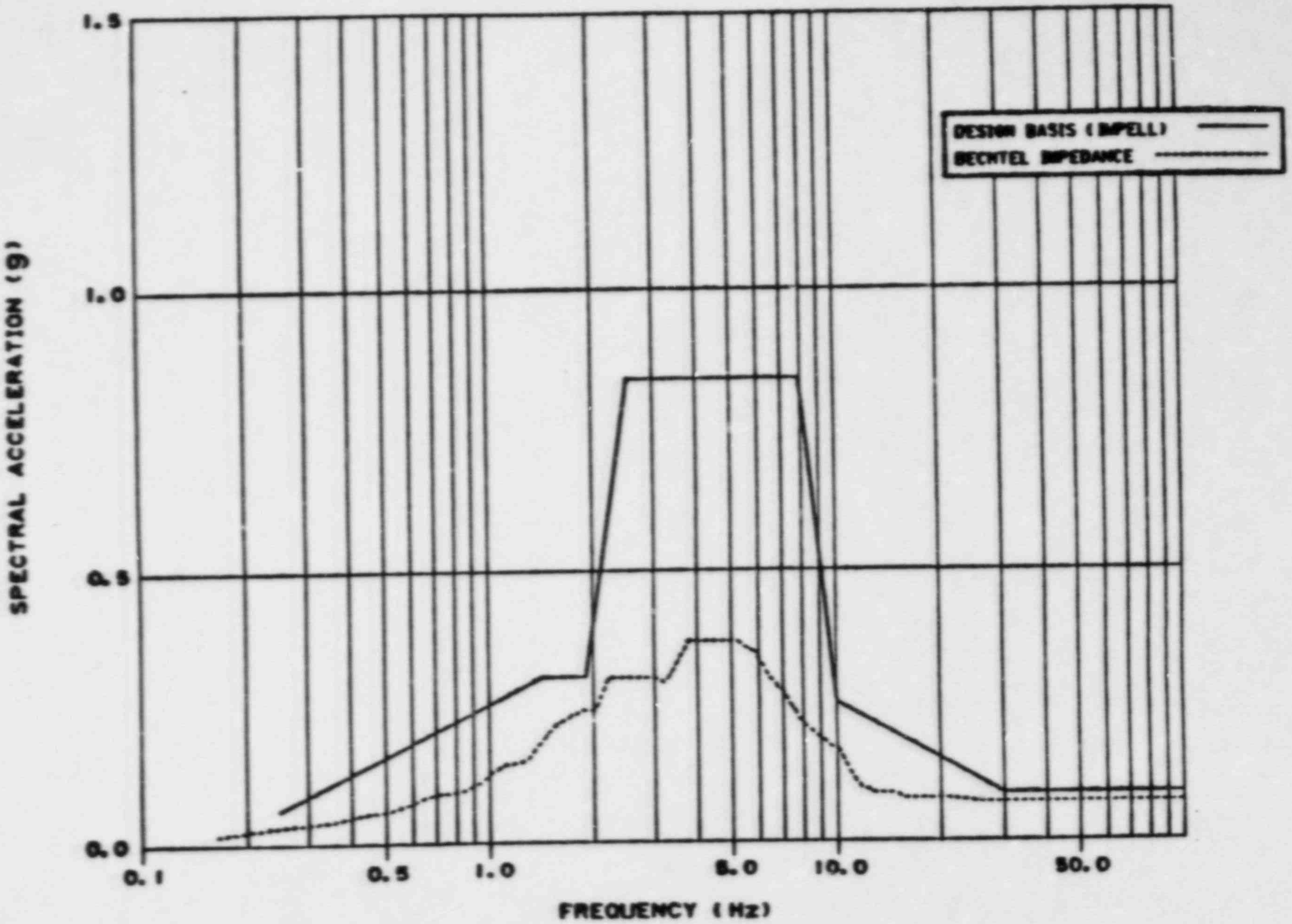


FIGURE ~~[illegible]~~ 220.21-41

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 114'-0",
N-S, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L62, REV 0

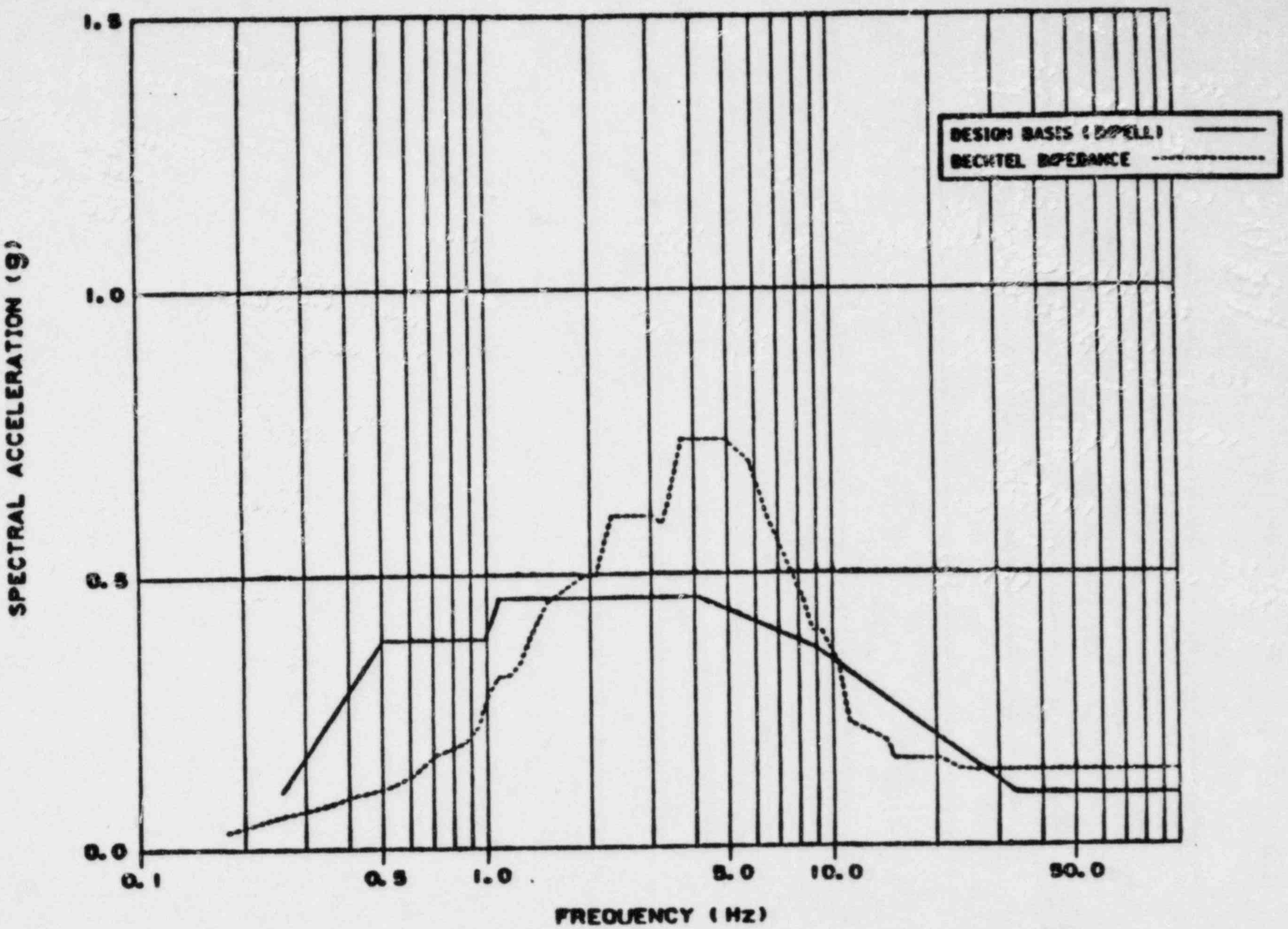


FIGURE ~~A-16-12~~ 220.21-42

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 135'-0",
N-S, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L70, REV 0

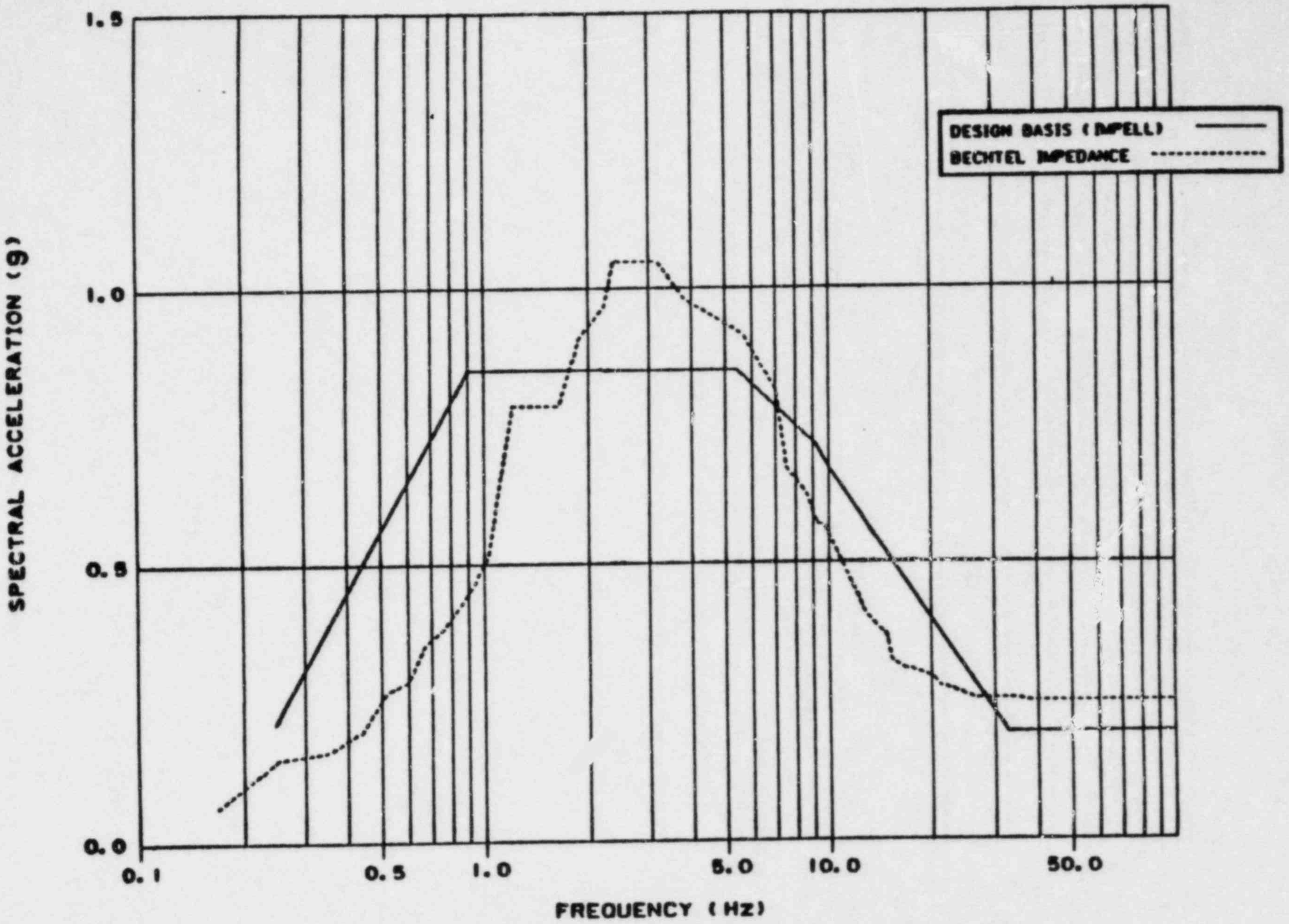


FIGURE ~~R-16md~~ 220.21-43

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 93' -0",
E-W, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD. L37, REV 0

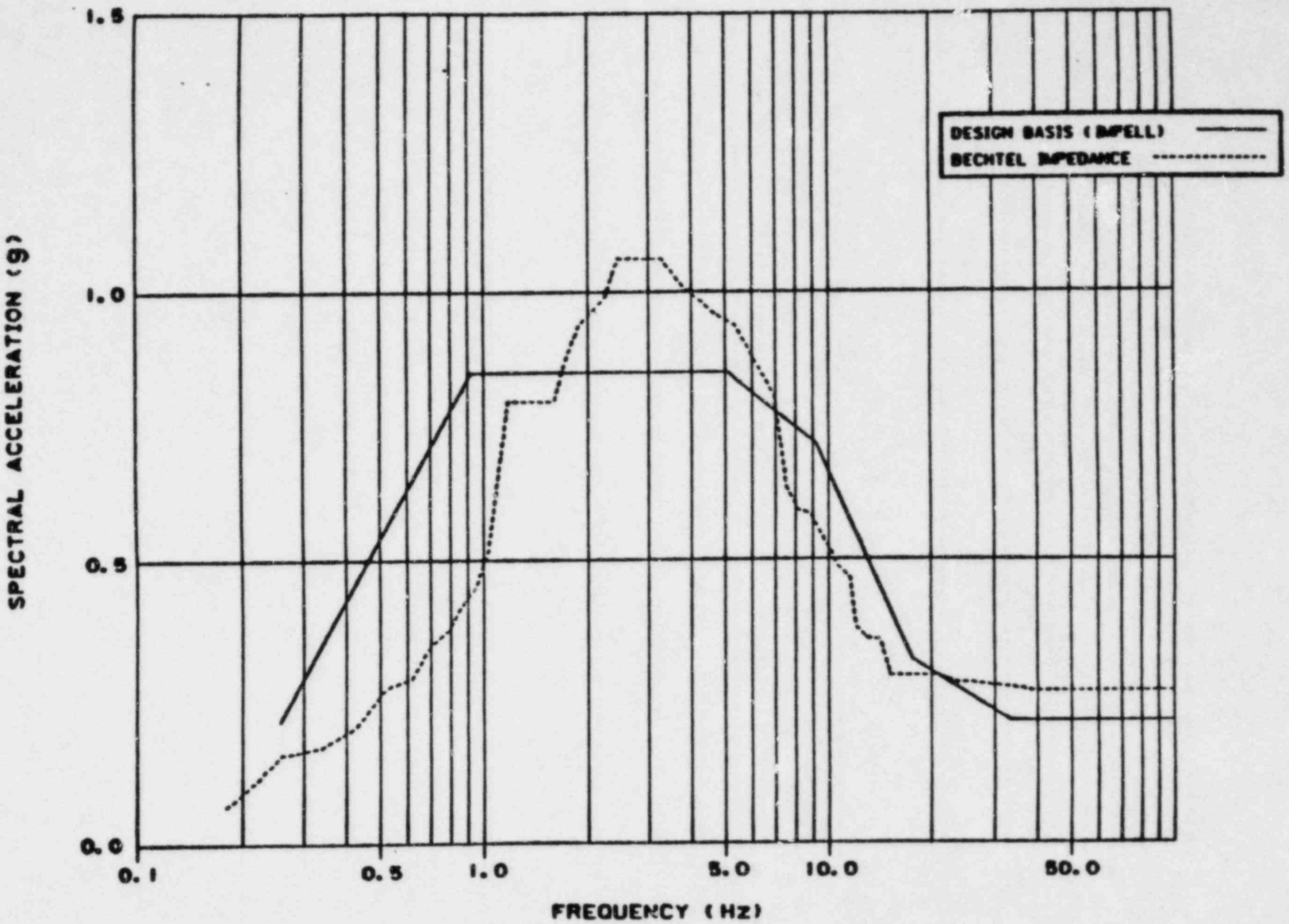


FIGURE ~~3-18-3~~ 220.21-44

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 114'-0",
E-W, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L43, REV 0

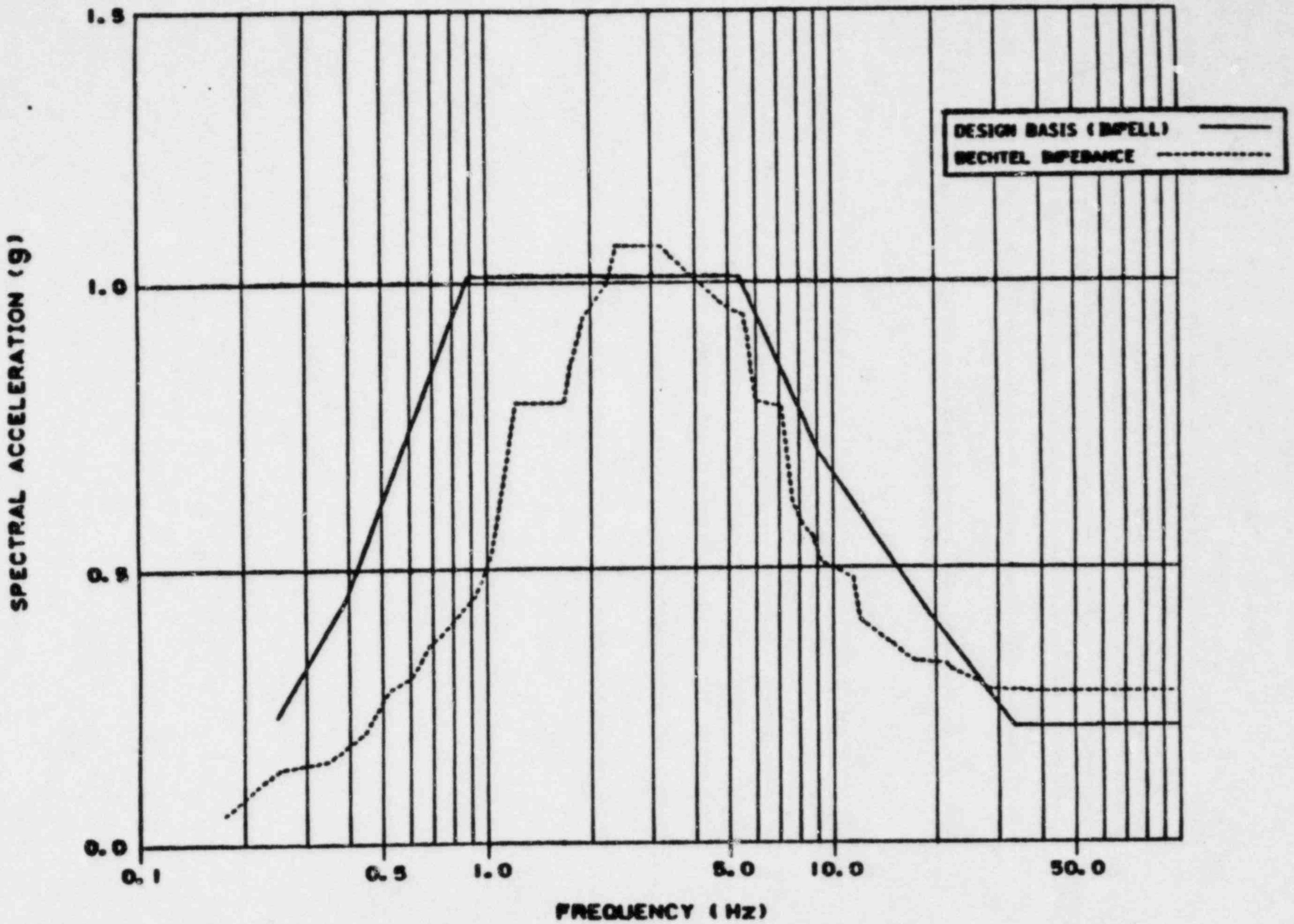


FIGURE 220.21-45

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 135'-0",
E-W, SSE, 2% DAMPING

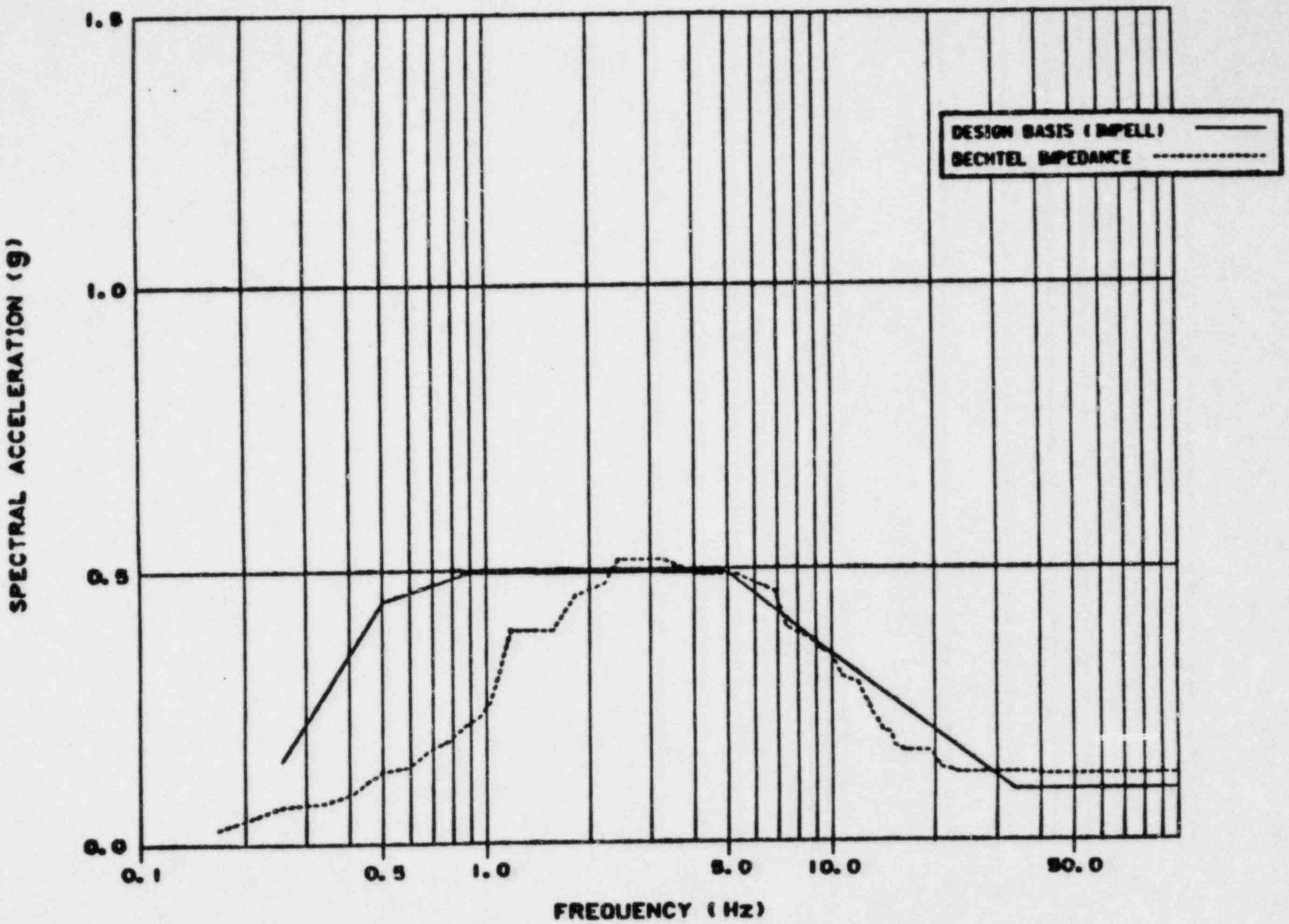


FIGURE ~~A-16-13~~ 220.21-46

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 93'-0",
E-W, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L55, REV 0

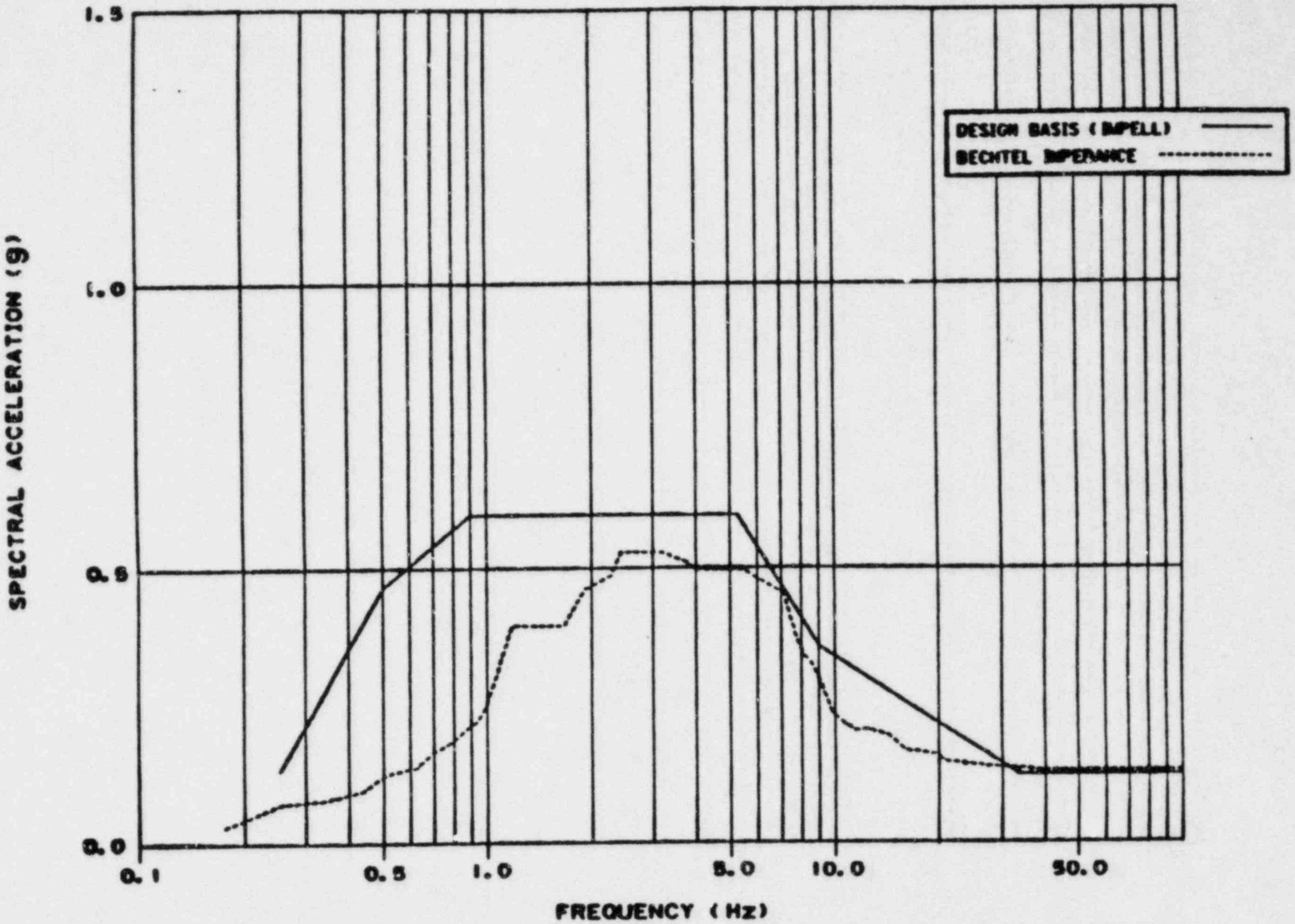


FIGURE ~~A-16-14~~ 220.21-47

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 114'-0",
E-W, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L61, REV 0

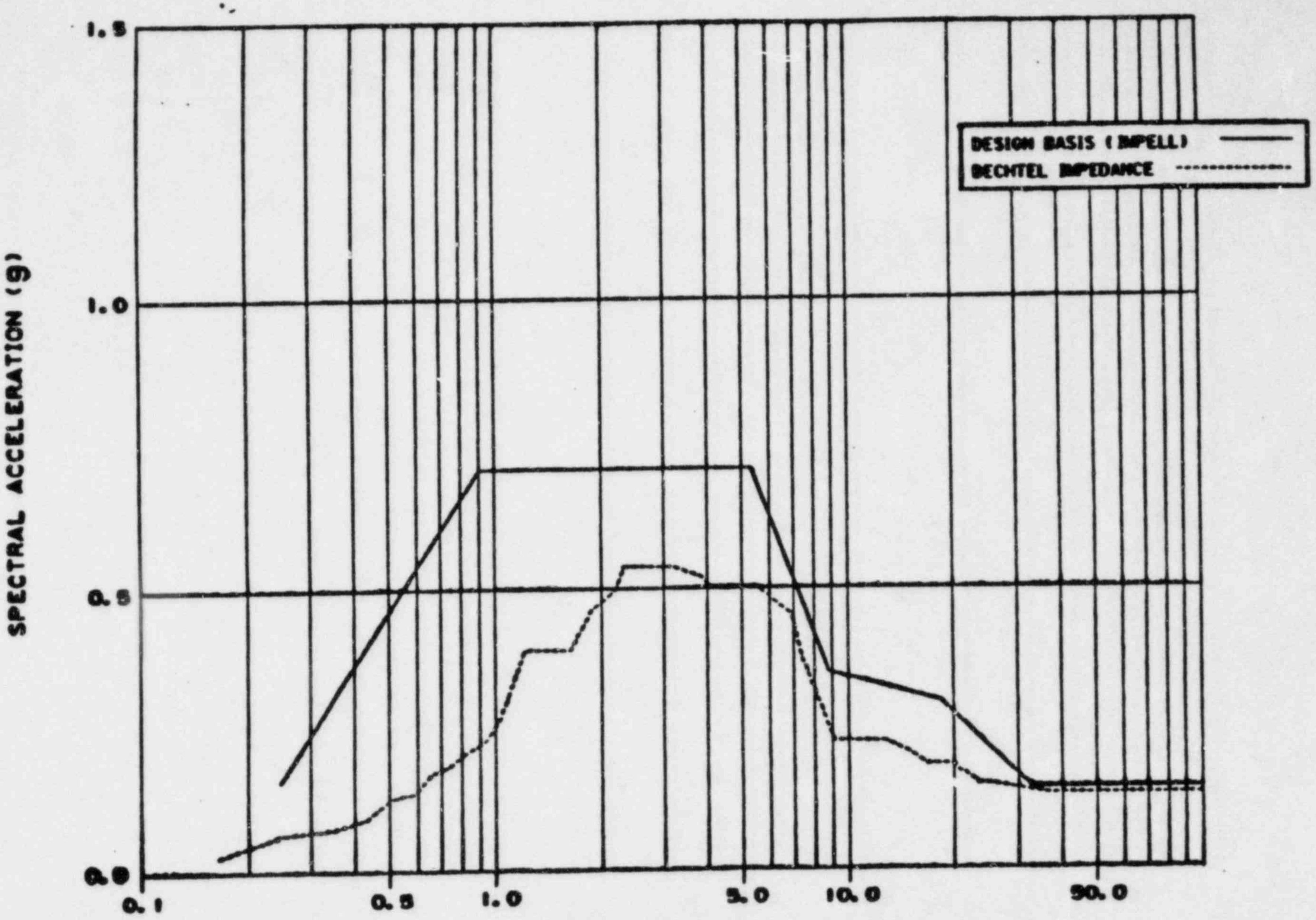


FIGURE ~~A-16-15~~ 220.21-48

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 135' -0",
E-W, OBE, 2% DAMPING

HOPE CREEK UNIT I
CAD, L71, REV 0

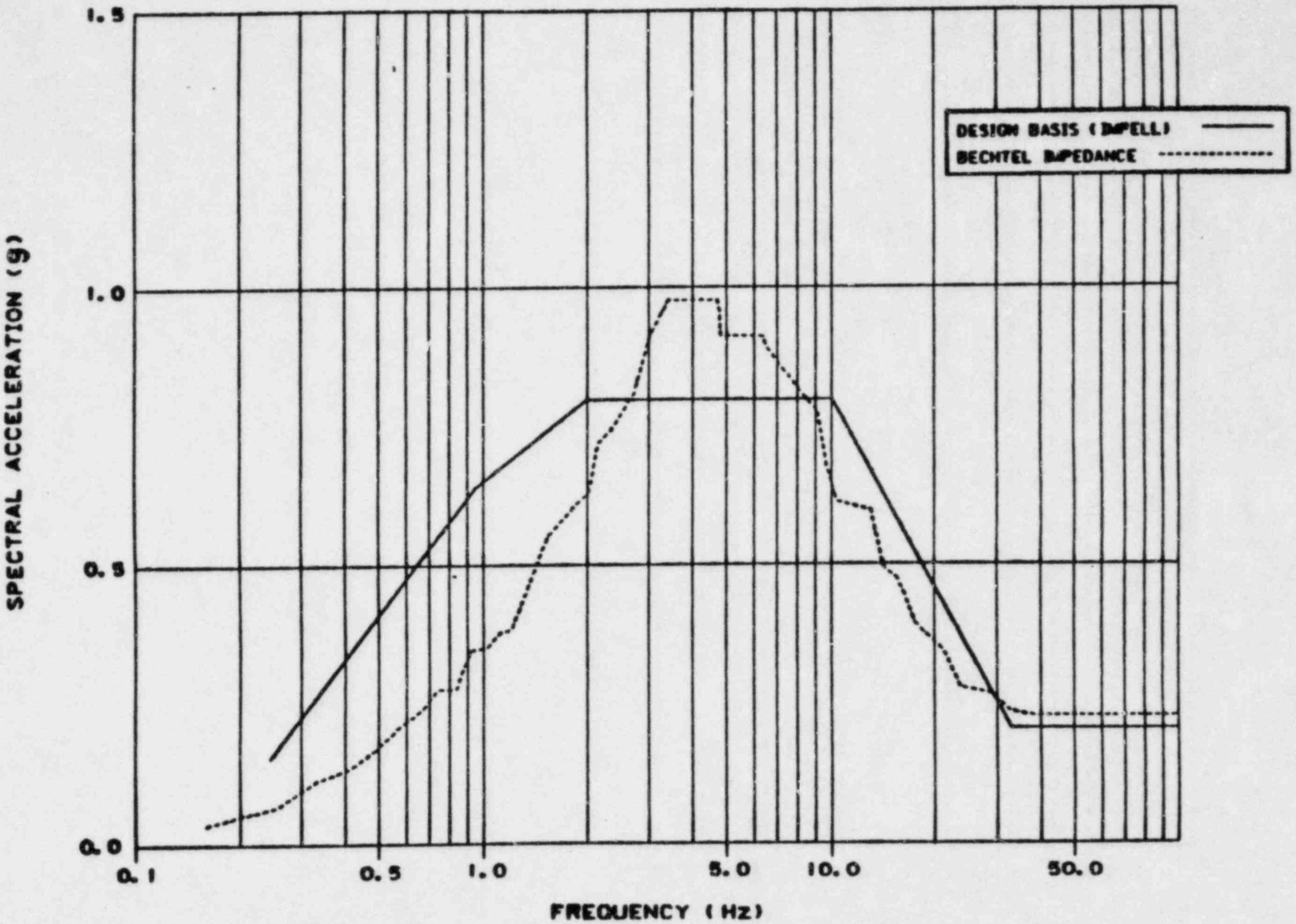


FIGURE ~~[illegible]~~ 220.21-49

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 93'-0",
VERTICAL, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L38, REV 0

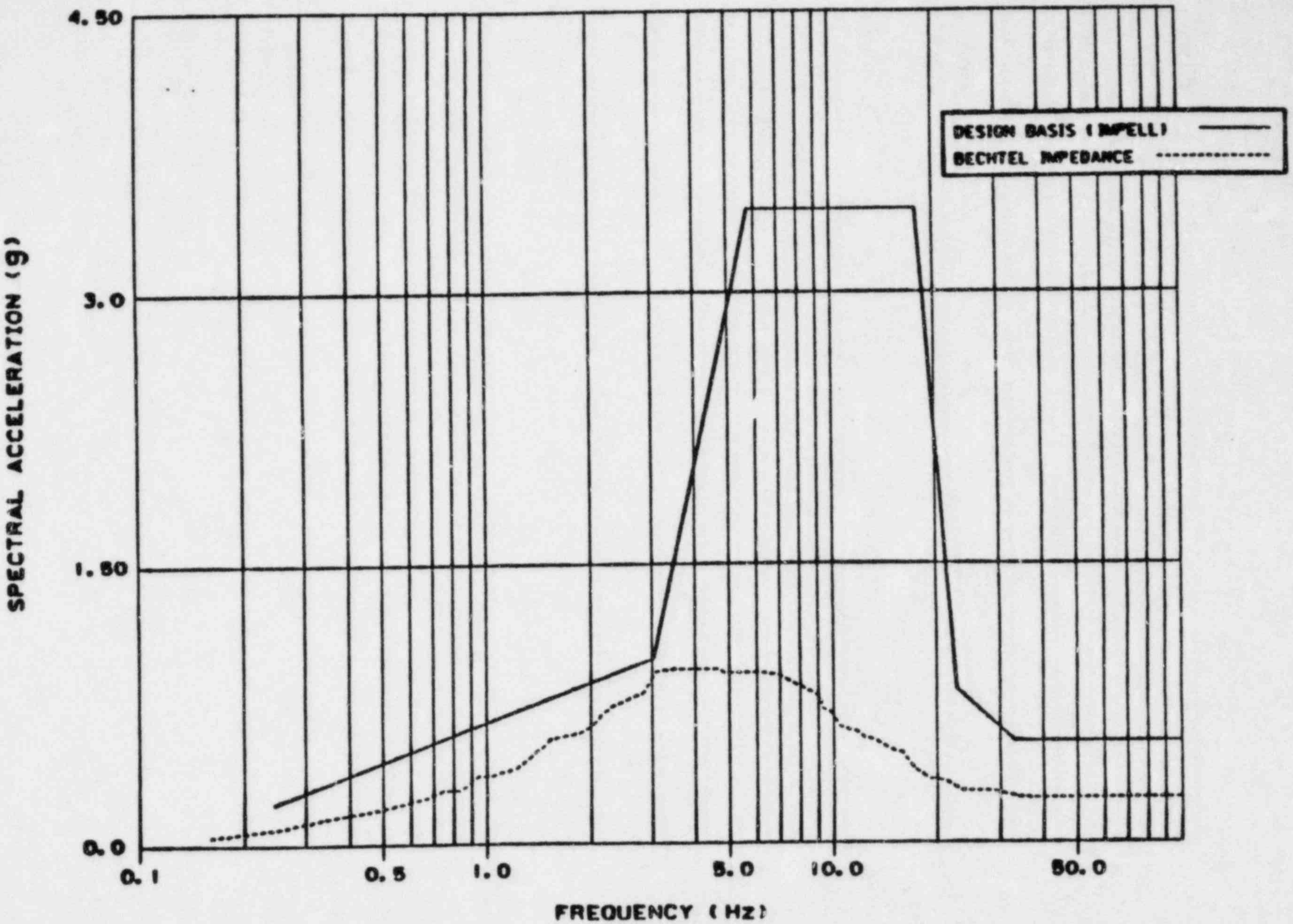


FIGURE ~~A-10-8~~ 220.21-5G

RESPONSE SPECTRA COMPARISON, INTAKE STRUCTURE AT ELEV. 114'-0", VERTICAL, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD. L45, REV 0

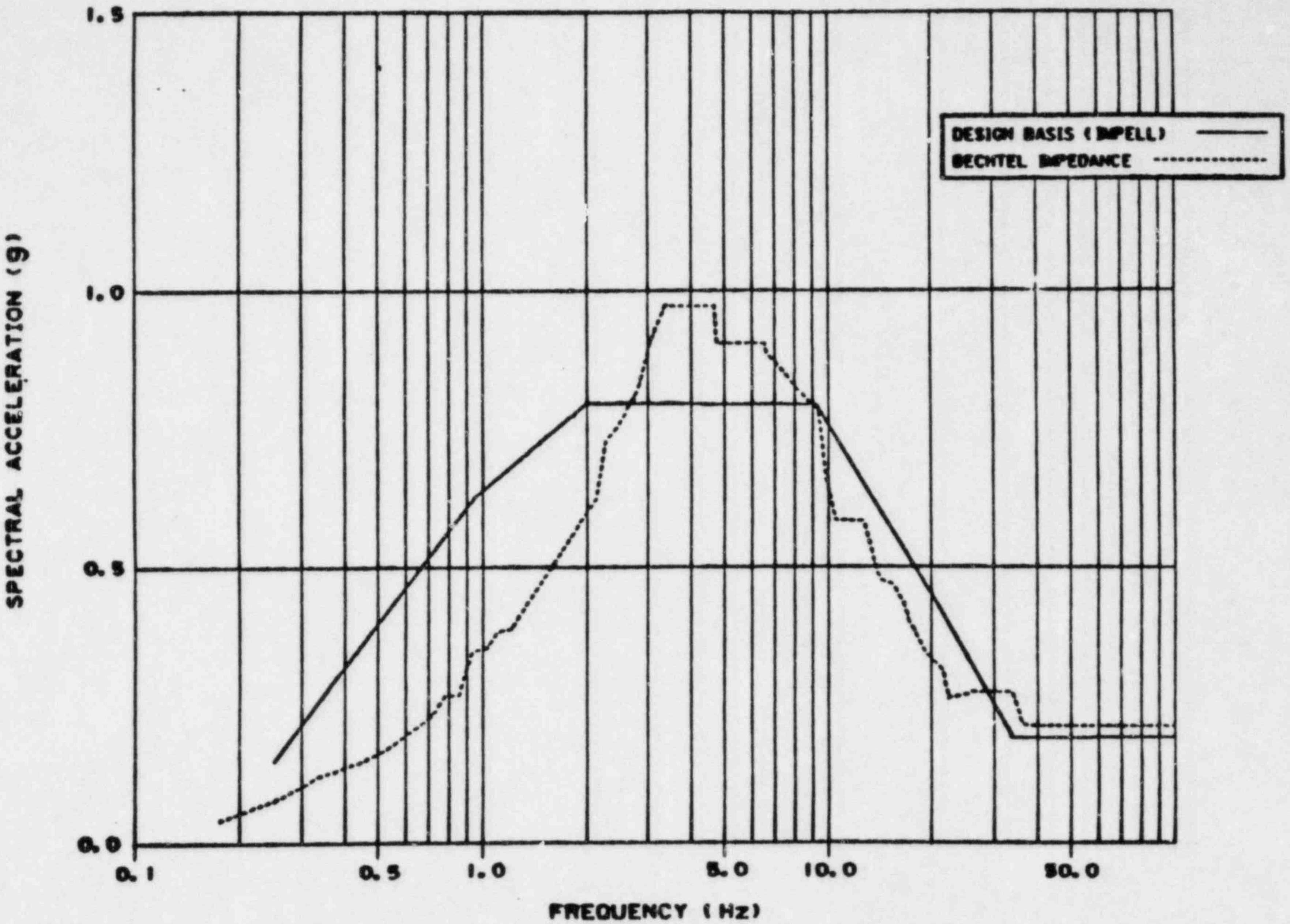


FIGURE 220.21-51

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 135'-0",
VERTICAL, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L54, REV 0

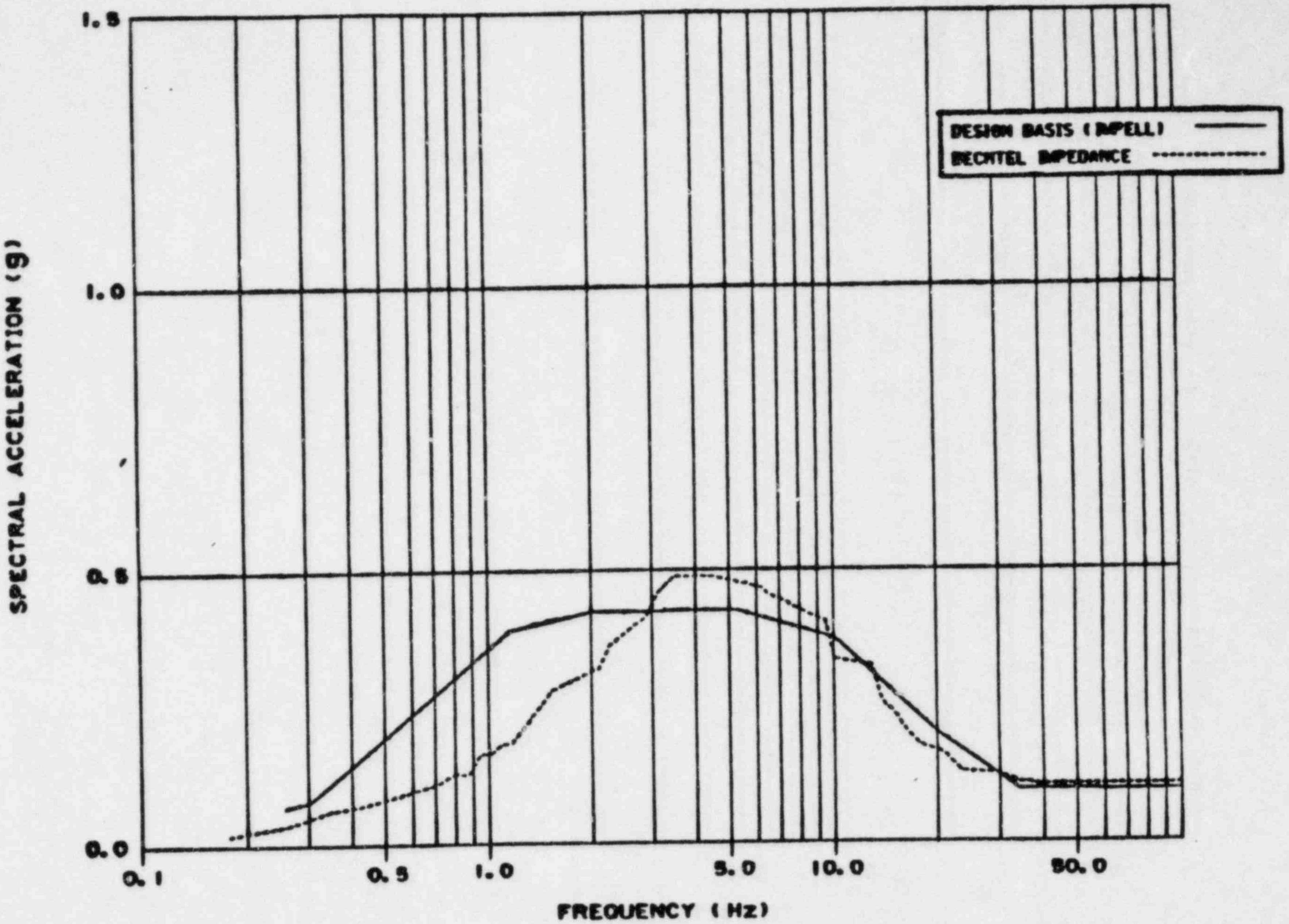


FIGURE ~~F-18-15~~ 220.21-52

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 93'-0",
VERTICAL, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L56, REV 0

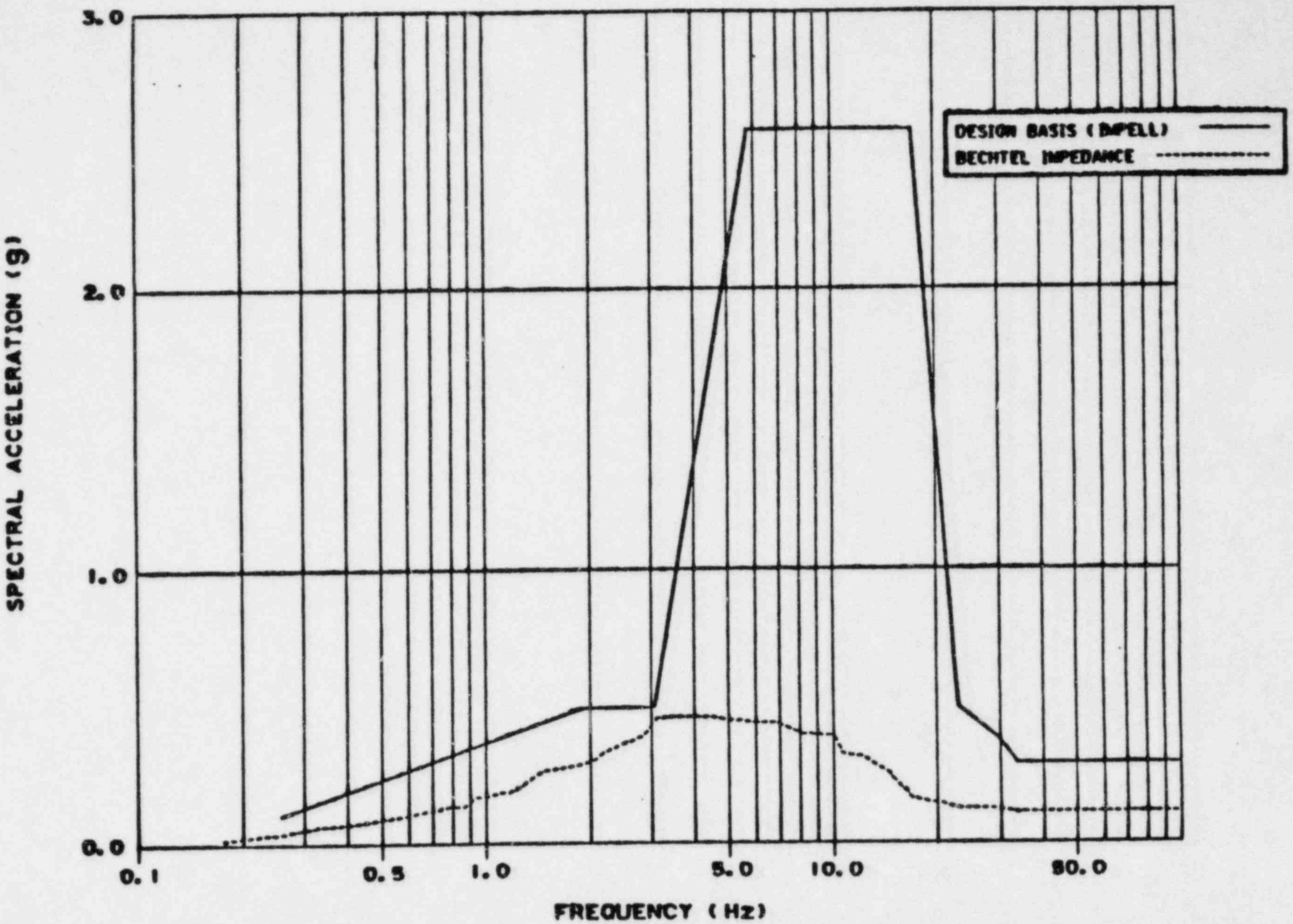


FIGURE ~~2-10-17~~ 220.21-53

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 114'-0",
VERTICAL, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD.L63, REV 0

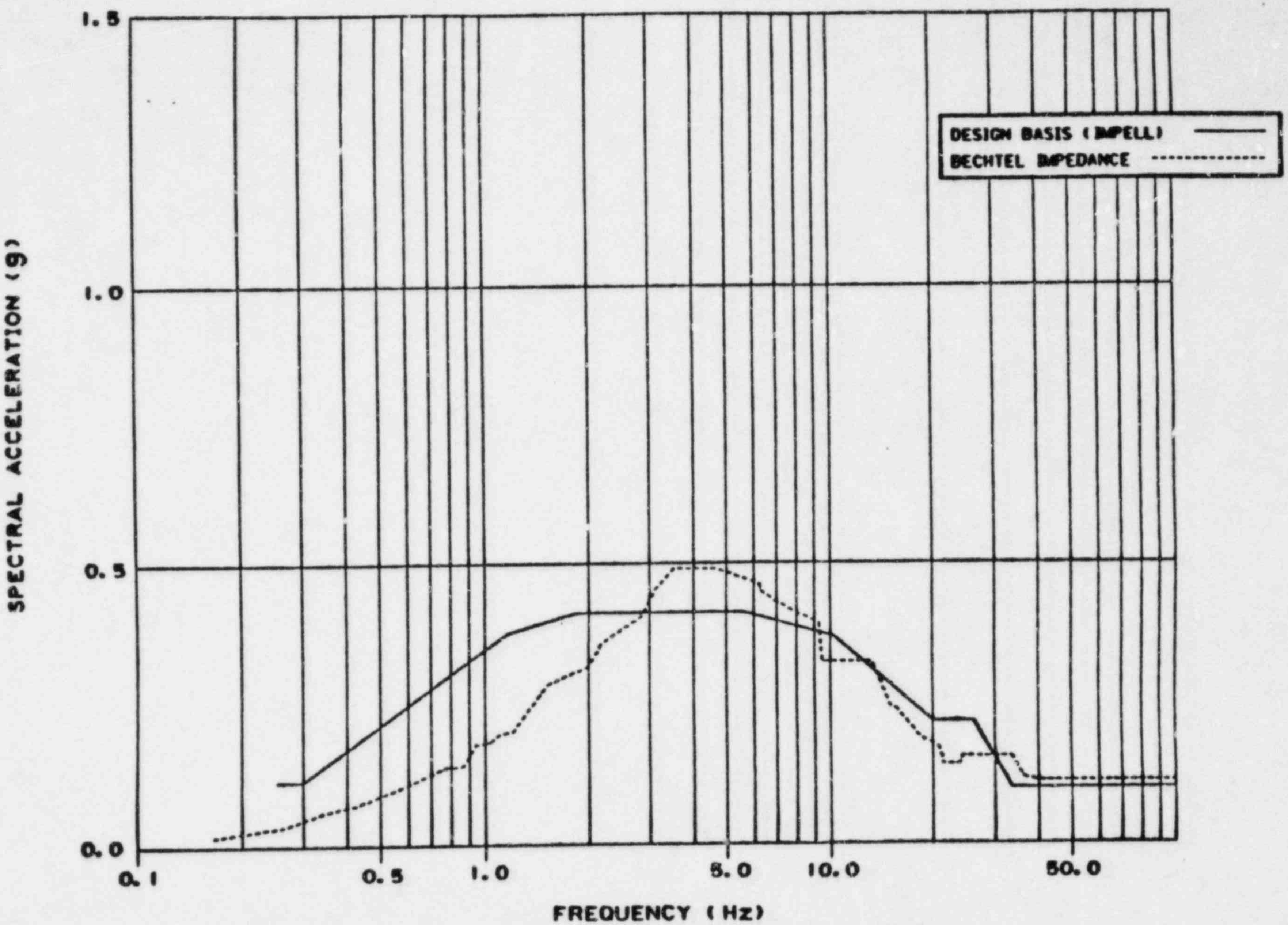


FIGURE ~~7-10-18~~ 220.21-54

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 135'-0",
VERTICAL, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, LT2, REV 0
06/20/84

DSER Open Item No. 53 (DSER Section 3.8.6)

DESIGN OF SEISMIC CATEGORY I TANKS

From January 10 through January 12, 1984, the sstaff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these repsonses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item A.4 from the NRC Structural/ Geotechnical meeting of January 12, 1984. A response to this item has been submitted to the NRC by a letter dated April 24, 1984, from R. L. Mittl to A. Schwencer. As a result of discussions with the NRC staff, a revised response to this item has been submitted to the NRC by a letter dated August 3, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 56 (DSER Section 3.8.6)

DRYWELL STICK MODEL DEVELOPMENT

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item B.9 from the NRC Structural/ Geotechnical meeting of January 10, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer. As a result of discussions with the NRC staff, a revised response to this item has been submitted to the NRC by a letter dated August 3, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 59 (DSER Section 3.8.6)

OVERTURNING MOMENT OF REACTOR BUILDING FOUNDATION MAT

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.7 from the NRC Structural/Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer. As a result of discussions with the NRC staff, a revised response to this item has been submitted to the NRC by a letter dated August 3, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 60 (DSER Section 3.8.6)

BSAP ELEMENT SIZE LIMITATIONS

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.8 from the NRC Structural/ Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer. As a result of discussions with the NRC staff, a revised response to this item has been submitted to the NRC by a letter dated August 3, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 64 (DSER Section 3.8.6)

SSI ANALYSIS $12H_3$ CUTOFF FREQUENCY

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.12 from the NRC Structural/Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984, from R. L. Mittl to A. Schwencer. As a result of discussions with the NRC staff, a revised response to this item has been submitted to the NRC by a letter dated August 3, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 77 (DSER Section 3.8.6)

FACTOR OF SAFETY AGAINST OVERTURNING OF INTAKE STRUCTURE

From January 10 through Januray 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item B.12 from the NRC Structural/Geotechnical meeting of January 11, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984 from R. L. Mittl to A. Schwencer. As a result of discussions with the NRC staff, a revised response to this item has been submitted to the NRC by a letter dated August 3, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 79 (DSER Section 3.8.6)

POST-MODIFICATION SEISMIC LOADS FOR TORUS

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item A.1 from the NRC Structural/Geotechnical meeting of January 12, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984 from R. L. Mittl to A. Schwencer. As a result of discussions with the NRC staff, a revised response to this item has been submitted to the NRC by a letter dated August 3, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 81 (DSER Section 3.8.6)

SEISMIC DISPLACEMENT OF TORUS

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action action items will be needed before the issuance of the Final SER.

RESPONSE

This item corresponds to Item A.3 from the NRC Structural/ Geotechnical meeting of January 12, 1984. A response to this item has been submitted to the NRC by a letter dated January 26, 1984, from R. L. Mittl to A. Schwencer. As a result of discussions with the NRC staff, a revised response to this item has been submitted to the NRC by a letter dated August 3, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 82 (DSER Section 3.8.6)

REVIEW OF SEISMIC CATEGORY I TANK DESIGN

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item A.4 from the NRC Structural/ Geotechnical meeting of January 12, 1984. A response to this item has been submitted to the NRC by a letter dated April 24, 1984, from R. L. Mittl to A. Schwencer. As a result of discussions with the NRC staff, a revised response to this item has been submitted to the NRC by a letter dated August 3, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 84 (DSER Section 3.8.6)

ULTIMATE CAPACITY OF CONTAINMENT (MATERIALS)

From January 10 through January 12, 1984, the staff met with the applicant and his consultants to conduct the structural audit. The audit covered each major safety-related structure at the Hope Creek Generating Station.

As a result of the audit, the staff identified 39 action items. The applicant has submitted preliminary responses to 22 of the 39 action items. The staff is in the process of reviewing these responses. The final resolution of the action items and any additional questions, which may be raised further, will be reported in the Final SER. The resolution of these action items will be needed before the issuance of the final SER.

RESPONSE

This item corresponds to Item B.2 from the NRC Structural/Geotechnical meeting of January 12, 1984. A response to this item has been submitted to the NRC by a letter dated February 17, 1984 from R. L. Mittl to A. Schwencer. As a result of discussions with the NRC staff, a revised response to this item has been submitted to the NRC by a letter dated August 3, 1984 from R. L. Mittl to A. Schwencer.

DSER Open Item No. 86 (DSER Section 3.9.1)

COMPUTER CODE VALIDATION

Analysis of mechanical components by the use of computer programs was performed by the applicant. A list showing all compute programs used by the applicant for statis and dynamic analyses to determine the structural integrity and functional integrity of seismic Category I Code and non-Code items and the analyses to determine stresses along with a description of the program is included in the FSAR. Design control measures to verify the adequacy of the design of safety-related components is required to 10 CFR 50, Appendix B.
Assurance is needed that all computer codes used in the analysis of piping have been appropriately validated.

RESPONSE

For the information requested above, see response to Question 210.28.

DSER Open Item No. 87 (DSER Section 3.9.1)

INFORMATION ON TRANSIENTS

Additional information on the transients used in the evaluation of some components is required. This is an open item.

RESPONSE

For the information requested above, see the response to Question 210.25.

Rev 1

HCGS

DSER Open Item No. 96 (DSER Section 3.9.3.3)

IE INFORMATION NOTICE 83-80

The staff's review of Section 3.9.3.4 of the applicant's FSAR relates to the methodology used by the applicant in the design of ASME Class 1, 2, and 3 component supports. The review includes assessment of design and structural integrity of the supports. The review addresses three types of supports: plate and shell, linear, and component standard types. More information regarding the design and construction of ASME Class 1, 2, and 3 component supports is required.

The applicant should address its actions taken in response to IE Information Notice 83-80.

RESPONSE

For the information requested above see response to Question 210.53.

QUESTION 210.53 (SECTION 3.9.3)

Describe what actions have been taken to address the staff concerns regarding stiff pipe clamps as described in IE Information Notice 83-80.

RESPONSE

The applications of stiff pipe clamps on HCGS ~~will be~~ were reviewed based on IE Information Notice 83-80. Section III of the ASME B&PV Code does not provide rules for evaluating stresses due to loadings from nonintegral attachments such as clamps; however, clamp-induced stresses ~~will be~~ were evaluated by methods consistent with the intent of the Section III of the ASME B&PV Code. The procedure ~~will~~ included the following:

1. Identif~~y~~ication of ~~the~~ locations of "stiff" clamps installed on ASME Section III Nuclear Class 1 piping systems.

2. Identif~~y~~ication of the types of clamps, the loads acting on the clamps and the bolt pre-load values used in their installation. In-piping stresses due to all loading conditions at the locations of stiff clamps ~~will~~ were also ~~be~~ identified and reviewed.

3. Addition of the primary membrane and bending stresses caused by the snubber load being transmitted to the pipe through the clamp to the stresses caused by internal pressure and bending computed by equation 9 of NB-3652. Clamp-induced stresses caused by the constraint of the expansion of the pipe due to the internal pressure ~~will be~~ were added to other secondary and peak stresses by calculating the effective increases in the C_1 and K_1 stress indices in accordance with NB-3681. Clamp induced stresses due to differential-temperature and differential-thermal-expansion coefficients ~~will be~~ were accounted for by computing the effective C_3 and K_3 stress indices. Clamp-induced stresses on elbows caused by the constraint of pipe wall ovalization ~~will be~~ were accounted for by computing the effective increases in C_2 and K_2 bending indices. The fatigue usage from clamp-induced plus other stresses ~~will be~~ were calculated at governing locations.

Although bolt preloads are not addressed under the ASME B&PV Code rules for piping, bolt preloads could result in damage to a pipe if a clamp were poorly designed. Calculations ~~will be~~ were made to ensure that bolt preloads could not result in plastic deformation of the pipe walls.

No problems were identified by the evaluations and calculations described above. The limits of Section III of the ASME B&PV Code were not violated. GE's Design Memo's # 170-107 and # 170-109 and Bechtel's Stress Report SR-10855-SS-27 were submitted under separate cover & letter from R.L. Mittl, PSEG, to A. Schwencer, NRC, dated August 14, 1984.

RESPONSE (Cont'd)

HCGS clamps were not used to meet stiffness criteria. They were designed to meet the requirements for strength and load distribution using a minimum of space.

The clamp design utilizes a double nut arrangement to prevent the nuts from backing off. The low temperature ($<600^\circ$F) and stresses in the bolt from preloads will not cause a relaxation of the material. Consequently, no lift off from the piping will occur.

DSER Open Item No. 100b (DSER Section 3.9.6)

10 CFR 55a, PARAGRAPH (g)

In Section 3.9.2 and 3.9.3 of the Safety Evaluation Report, the staff discussed the design of safety-related pumps and valves in the Hope Creek plant. The load combinations and stress limits used in the design of pumps and valves assure that the component pressure boundary integrity is maintained. In addition, the applicant will periodically test and perform periodic measurements of all its safety-related pumps and valves. These tests and measurements are performed in accordance with the rules of Section XI of the ASME Code. The tests verify that these pumps and valves operate successfully when called upon. The periodic measurements are made of various parameters and compared to baseline measurements in order to detect long-term degradation of the pump or valve performance. The staff reviews the applicant's program for preservice and inservice testing of pumps and valves using the guidance of SRP Section 3.9.6, and gives particular attention to the completeness of the program and to those areas of the test program for which the applicant requests relief from the requirements of Section XI of the ASME Code. The applicant must provide a commitment that the inservice testing of ASME Class 1, 2, and 3 components will be in accordance with the rules of 10 CFR 50.55a, Paragraph (g).

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reaactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure system. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems.

Pressure isolation valves are required to be Category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code, except as discussed below.

Limiting conditions for operation (LCO) are required to be added to the technical specifications which will require corrective action; i.e., shutdown or system isolation when the final approved leakage limits are not met. Also, surveillance requirements, which will state the acceptable leak rate testing frequency, shall be provided in the technical specifications.

DSER Open Item No. 100b (Cont'd)

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated as less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average to be approximately 1 year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance, and so forth.

The staff's position on leak rate limiting conditions for operation is that leak rates must be equal to or less than 1 gallon per minute (GPM) for each valve to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting value would be an indication of valve degradation from one test to another.

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM from the previous test leak rate or system design precludes mesasuring 1 GPM with sufficient accuracy. These items will be reviewed on a case-by-case basis.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves. In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valve provide isolation, only two of the valves need to be leak tested.

RESPONSE

An evaluation of the pressure isolation features is provided in the response to Question 210.56.

QUESTION 210.56 (SECTION 3.9.6)

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specifications which will require corrective action; i.e., shutdown or system isolation when the final approved leakage limits are not met. Also, surveillance requirements which will state the acceptable leak rate testing frequency shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average to be approximately one year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance, etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute (GPM) for each valve to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting value would be an indication of valve degradation from one test to another.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves.

Also discuss in detail how your leak testing program will conform to the above staff position.

RESPONSE

The reactor coolant pressure boundary has been reviewed for interconnecting safety-related low pressure systems. Table 210.56-1 summarizes the results of this review. The table identifies the reactor coolant system pressure isolation valves and details the extent of compliance with the staff's position. Also identified in Table 210.56-1 are those pressure isolation valves that are leakage tested.

INSERT A

Four sets of full size P&IDs were submitted under separate cover.

The P&IDs that the NRC staff will need to review this response are identified in Table 210.56.

INSERT A

THE H.C.G.S USES TWO ISOLATION VALVES. THE ISOLATION VALVES ARE PERIODICALLY LEAK RATE TESTED AS 10CFR50, APPENDIX J, TYPE C VALVES OR ASME, SECTION XI, CATAGORY A VALVES. IN THE EVENT OF ISOLATION VALVE LEAKAGE A SAFETY RELIEF VALVE WILL FURTHER PROTECT THE LOW PRESSURE SYSTEM.

TYPE C LEAK RATE TESTING IN ACCORDANCE WITH 10CFR50, APPENDIX J USING GAS IS MORE CONSERVATIVE THAN LEAK RATE TESTING ISOLATION VALVES WITH SYSTEM LIQUID IN ACCORDANCE WITH ASME CODE, SECTION XI.

TABLE 210.56-1

SAFETY-RELATED LOW PRESSURE SYSTEMS
CONNECTED TO THE RCPB

INSERT B

RPV Nozzle	Connecting Line Description	Pressure Isolation Valve	Leak Tested(9)
N1B	RHR Shutdown Cooling Suction	BC-V071 BC-V164	Yes Yes
N2A-E	RHR Shutdown Cooling Return	BC-V013(1)(2)	Yes
N2F-K	RHR Shutdown Cooling Return	BC-V110(1)(2)	Yes
N4A-C	RCIC Discharge	BD-V005 AE-V003 AE-V002	Yes Yes Yes
N4D-F	HPCI Feedwater Discharge	BJ-V059 AE-V007 AE-V006	Yes Yes Yes
N5A	Core Spray	BE-V003(1)(5)	Yes
N5B	Core Spray	BE-V007(1)(6)	Yes
	HPCI Core Spray Discharge	BJ-V001(1)(4)	Yes
N6A	RHR Headspray	BC-V021 BC-V020	Yes Yes
N17A	LPCI	BC-V004(1)(8)	Yes
N17B	LPCI	BC-V016(1)(2)	Yes
N17C	LPCI	BC-V101(1)(7)	Yes
N17D	LPCI	BC-V113(1)(3)	Yes

INSERT C

(1) HCGS uses one pressure isolation valve. The isolation valve is periodically leak rate tested and in the event of valve leakage, a safety-relief valve will protect the low pressure system.

(2) Safety-relief valve BC-PSV-F025B provides overpressure protection. It has a 410 psig set pressure and a 10 gpm capacity.

TABLE 210.56-1 (Cont'd) Page 2 of 2

(3) Safety-relief valve BC-PSV-F025A provides overpressure protection. It has a 410 psig set pressure and a 10 gpm capacity.

(4) BJ-V003 provides pressure isolation but is not required to be leak rate tested in order to prevent overpressurization of the low pressure (pump suction) portion of the HPCI system. Should BJ-V003 leak excessively, safety-relief valve BJ-PSV-F020 will prevent the system from being overpressurized. BJ-PSV-F020 has a 100 psig setpoint and a 15 gpm capacity.

(5) Safety-relief valve BE-PSV-F012B provides overpressure protection. It has a 500 psig setpoint and a 100 gpm capacity.

(6) Safety-relief valve BE-PSV-F012A provides overpressure protection. It has a 500 psig setpoint and a 100 gpm capacity.

(7) Safety-relief valve BC-PSV-F025C provides overpressure protection. It has a 410 psig setpoint and a 10 gpm capacity.

(8) Safety-relief valve BC-PSV-F025D provides overpressure protection. It has a 410 psig setpoint and a 10 gpm capacity.

(9) Leak rate tested in accordance with 10 CFR 50, Appendix J, requirements.

INSERT B

HCGS FSAR

TABLE 210.56-1
SAFETY-RELATED LOW PRESSURE SYSTEMS CONNECTED TO RCPB

RPV Nozzle	Containment Penetration	Connecting Line Description	Pressure Isolation Valve	Leak Tested	Safety-Relief Protection
N1B	P-3	RHR Shutdown Cooling Suction	BC-V164 (2)	(3)	(6)
			BC-V071 (1)	(3)	
N2A-E	P-4A	RHR Shut Down Cooling Return	BC-V013 (2)	(3)	(7)
			BC-V014 (1) (5)	(3)	
			BC-V118 (1)	(3)	
N2F-K	P4B	RHR Shut Down Cooling Return	BC-V110 (2)	(3)	(8)
			BC-V111 (1)(5)	(3)	
			BC-V117 (1)	(3)	
N4A-C	P2A	RCIC to Feedwater	BD-V005 [illegible]	(4)	(9)
			AE-V002 (2)(5)	(4)	
			AE-V003 (1)(5)	(4)	
N4D-F	P2B	HPCI to Feedwater	BJ-V059 (15)	(4)	(10)
			AE-V006 (2)(5)	(4)	
			AE-V008 (1)(5)	(4)	
N5A	P5A	Core Spray	BE-V003 (2)	(3)	(11)
			BE-V002 (1)(5)	(3)	
			BE-V072 (1)	(3)	
N5B	P5B	Core Spray	BE-V007 (2)	(3)	(12)
		HPCI to Core Spray	BJ-V001 (2)	(3)	(10)
			BE-V006 (1)(5)	(3)	
			BE-V071 (1)	(3)	
N6A	P10	RHR, RPV Head Spray	BC-V020 (2)	(3)	(7)
			BC-V021 (1)	(3)	
N17A	P6A	RHR, LPCI	BC-V004 (2)	(3)	(14)
			BC-V005 (1)(5)	(3)	
			BC-V122 (1)	(3)	

INSERT B (cont'd)

N17B	P6B	RHR, LPCI	BC-V016 (2)	(3)	(7)
			BC-V017 (1)(5)	(3)	
			BC-V120 (1)	(3)	
N17C	P6D	RHR, LPCI	BC-V101 (2)	(3)	(13)
			BC-V102 (1)(5)	(3)	
			BC-V121 (1)	(3)	
N17D	P6C	RHR, LPCI	BC-V113 (2)	(3)	(8)
			BC-V114 (1)(5)	(3)	
			BC-V119 (1)	(3)	

INSERT C

(1) 1ST PRESSURE ISOLATION VALVE
(2) 2ND PRESSURE ISOLATION VALVE
(3) LEAK RATE TESTED IN ACCORDANCE WITH 10CFR50, APPENDIX J.
(4) LEAK RATE TESTED IN ACCORDANCE WITH ASME, SECTION XI.
(5) FUNCTIONALLY TESTED AS A CATAGORY C CHECK VALVE IN ACCORDANCE WITH ASME, SECTION XI.
(6) SAFETY RELIEF VALVE BC-PSV-F029 PROVIDES OVERPRESSURE PROTECTION. IT HAS A 170 PSIG SET POINT AND A 10 GPM CAPACITY.
(7) SAFETY RELIEF VALVE BC-PSV-F025B PROVIDES OVER PRESSURE PROTECTION. IT HAS A 410 PSIG SET POINT AND A 10 GPM CAPACITY.
(8) Safety-relief valve BC-PSV-F025A provides overpressure protection. It has a 410 psig set pressure and a 10 gpm capacity.
(9) SAFETY RELIEF VALVE BD-PSV-F017 PROVIDES OVERPRESSURE PROTECTION. IT HAS A 100 PSIG SET POINT AND A 10 GPM CAPACITY.
(10) SAFETY RELIEF VALVE BJ-PSV-F020 PROVIDES OVERPRESSURE PROTECTION. IT HAS A 100 PSIG SET POINT AND A 15 GPM CAPACITY.
(11) Safety-relief valve BE-PSV-F012B provides overpressure protection. It has a 500 psig setpoint and a 100 gpm capacity.
(12) Safety-relief valve BE-PSV-F012A provides overpressure protection. It has a 500 psig setpoint and a 100 gpm capacity.
(13) Safety-relief valve BC-PSV-F025C provides overpressure protection. It has a 410 psig setpoint and a 10 gpm capacity.
(14) Safety-relief valve BC-PSV-F025D provides overpressure protection. It has a 410 psig setpoint and a 10 gpm capacity.
(15) 3RD PRESSURE ISOLATION VALVE

DSER OPEN ITEM NO. 101 (Section 3.9.6)

PST AND IST PROGRAMS FOR PUMPS AND VALVES

The applicant has not yet submitted his program for the preservice and inservice testing of pumps and valves.

RESPONSE

The HCGS pre-service pump and valve test program will be conducted during the start-up phase of systems and component testing. The inservice testing base line data will be established and any testing requirements, for which relief requests are appropriate, will be identified.

Upon completion of preservice testing, the inservice test program will be established and submitted 6 months prior to the effective date of the initial operating license. The IST program submitted will include any request(s) for relief from pump and valve testing requirements.

JES:dh

8/20/84

M P84 132/17 1-dh

RESPONSE TRANSMITTED
TO NRC ON: JUN 29 1984

DSER Open Item No. 102 (Section 3.9.6)

LEAK TESTING OF PRESSURE ISOLATION VALVES

The applicant has not yet responded to the staff's concern regarding the leak testing of pressure isolation valves.

RESPONSE

For the information requested above, see the response to Question 210.56.

QUESTION 210.56 (SECTION 3.9.6)

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specifications which will require corrective action; i.e., shutdown or system isolation when the final approved leakage limits are not met. Also, surveillance requirements which will state the acceptable leak rate testing frequency shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average to be approximately one year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance, etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute (GPM) for each valve to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting value would be an indication of valve degradation from one test to another.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves.

Also discuss in detail how your leak testing program will conform to the above staff position.

RESPONSE

The reactor coolant pressure boundary has been reviewed for interconnecting safety-related low pressure systems. Table 210.56-1 summarizes the results of this review. The table identifies the reactor coolant system pressure isolation valves and details the extent of compliance with the staff's position. Also identified in Table 210.56-1 are those pressure isolation valves that are leakage tested.

INSERT A

Four sets of full size P&IDs were submitted under separate cover.

The P&IDs that the NRC staff will need to review this response are identified in Table 210.56.

INSERT A

THE H.C.G.S USES TWO ISOLATION VALVES. THE ISOLATION VALVES ARE PERIODICALLY LEAK RATE TESTED AS 10CFR50, APPENDIX J, TYPE C VALVES OR ASME, SECTION XI, CATAGORY A VALVES. IN THE EVENT OF ISOLATION VALVE LEAKAGE A SAFETY RELIEF VALVE WILL FURTHER PROTECT THE LOW PRESSURE SYSTEM.

TYPE C LEAK RATE TESTING IN ACCORDANCE WITH 10CFR50, APPENDIX J USING GAS IS MORE CONSERVATIVE THAN LEAK RATE TESTING ISOLATION VALVES WITH SYSTEM LIQUID IN ACCORDANCE WITH ASME CODE, SECTION XI.

TABLE 210.56-1

SAFETY-RELATED LOW PRESSURE SYSTEMS
CONNECTED TO THE RCPB

INSERT B

RPV Nozzle	Connecting Line Description	Pressure Isolation Valve	Leak Tested[9]
N1B	RHR Shutdown Cooling Suction	BC-V071 BC-V164	Yes Yes
N2A-E	RHR Shutdown Cooling Return	BC-V013[1][2]	Yes
N2F-K	RHR Shutdown Cooling Return	BC-V110[1][2]	Yes
N4A-C	RCIC Discharge	BD-V005 AE-V003 AE-V002	Yes Yes Yes
N4D-F	HPCI Feedwater Discharge	BJ-V059 AE-V007 AE-V006	Yes Yes Yes
N5A	Core Spray	BE-V003[1][5]	Yes
N5B	Core Spray	BE-V007[1][6]	Yes
	HPCI Core Spray Discharge	BJ-V001[1][4]	Yes
N6A	RHR Headspray	BC-V021 BC-V020	Yes Yes
N17A	LPCI	BC-V004[1][8]	Yes
N17B	LPCI	BC-V016[1][2]	Yes
N17C	LPCI	BC-V101[1][7]	Yes
N17D	LPCI	BC-V113[1][3]	Yes

INSERT C

(1) HCGS uses one pressure isolation valve. The isolation valve is periodically leak rate tested and in the event of valve leakage, a safety-relief valve will protect the low pressure system.

(2) Safety-relief valve BC-PSV-F025B provides overpressure protection. It has a 410 psig set pressure and a 10 gpm capacity.

TABLE 210.56-1 (Cont'd)

(3) Safety-relief valve BC-PSV-F025A provides overpressure protection. It has a 410 psig set pressure and a 10 gpm capacity.

(4) BJ-V003 provides pressure isolation but is not required to be leak rate tested in order to prevent overpressurization of the low pressure (pump suction) portion of the HPCI system. Should BJ-V003 leak excessively, safety-relief valve BJ-PSV-F020 will prevent the system from being overpressurized. BJ-PSV-F020 has a 100 psig setpoint and a 15 gpm capacity.

(5) Safety-relief valve BE-PSV-F012B provides overpressure protection. It has a 500 psig setpoint and a 100 gpm capacity.

(6) Safety-relief valve BE-PSV-F012A provides overpressure protection. It has a 500 psig setpoint and a 100 gpm capacity.

(7) Safety-relief valve BC-PSV-F025C provides overpressure protection. It has a 410 psig setpoint and a 10 gpm capacity.

(8) Safety-relief valve BC-PSV-F025D provides overpressure protection. It has a 410 psig setpoint and a 10 gpm capacity.

(9) Leak rate tested in accordance with 10 CFR 50, Appendix J, requirements.

INSERT B

HCGS FSAR

TABLE 210.56-1
SAFETY-RELATED LOW PRESSURE SYSTEMS
CONNECTED TO RCPB

RPV Nozzle	Containment Penetration	Connecting Line Description	Pressure Isolation Valve	Leak-Tested	Safety-Relief Protection
N1B	P-3	RHR Shutdown	BC-V164 (2)	(3)	(6)
		Cooling Suction	BC-V071 (1)	(3)	
N2A-E	P-4A	RHR Shut Down	BC-V013 (2)	(3)	(7)
		Cooling Return	BC-V014 (1) (5)	(3)	
			BC-V118 (1)	(3)	
N2F-K	P4B	RHR Shut Down	BC-V110 (2)	(3)	(8)
		Cooling Return	BC-V111 (1)(5)	(3)	
			BC-V117 (1)	(3)	
N4A-C	P2A	RCIC to	BD-V005 (1)	(4)	(9)
		Feedwater	AE-V002 (2)(5)	(4)	
			AE-V003 (1) (5)	(4)	
N4D-F	P2B	HPCI to	BJ-V059 (1)	(4)	(10)
		Feedwater	AE-V006 (2)(5)	(4)	
			AE-V007 (1)(5)	(4)	
N5A	P5A	Core Spray	BE-V003 (2)	(3)	(11)
			BE-V002 (1)(5)	(3)	
			BE-V072 (1)	(3)	
N5B	P5B	Core Spray	BE-V007 (2)	(3)	(12)
		HPCI to Core Spray	BJ-V001 (2)	(3)	(10)
			BE-V006 (1)(5)	(3)	
			BE-V071 (1)	(3)	
N6A	P10	RHR, RPV Head	BC-V020 (2)	(3)	(7)
		Spray	BC-V021 (1)	(3)	
N17A	P6A	RHR, LPCI	BC-V004 (2)	(3)	(14)
			BC-V005 (1)(5)	(3)	
			BC-V122 (1)	(3)	

INSERT B (cont'd)

N17B	P6B	RHR, LPCI	BC-V016 (2)	(3)	(7)
			BC-V017 (1)(5)	(3)	
			BC-V120 (1)	(3)	
N17C	P6D	RHR, LPCI	BC-V101 (2)	(3)	(13)
			BC-V102 (1)(5)	(3)	
			BC-V121 (1)	(3)	
N17D	P6C	RHR, LPCI	BC-V113 (2)	(3)	(8)
			BC-V114 (1)(5)	(3)	
			BC-V119 (1)	(3)	

INSERT C

(1) 1ST PRESSURE ISOLATION VALVE

(2) 2ND PRESSURE ISOLATION VALVE

(3) LEAK RATE TESTED IN ACCORDANCE WITH 10CFR50, APPENDIX J.

(4) LEAK RATE TESTED IN ACCORDANCE WITH ASME, SECTION XI.

(5) FUNCTIONALLY TESTED AS A CATAGORY C CHECK VALVE IN ACCORDANCE WITH ASME, SECTION XI.

(6) SAFETY RELIEF VALVE BC-PSV-F029 PROVIDES OVERPRESSURE PROTECTION. IT HAS A 170 PSIG SET POINT AND A 10 GPM CAPACI

(7) SAFETY RELIEF VALVE BC-PSV-F025B PROVIDES OVER PRESSURE PROTECTION. IT HAS A 410 PSIG SET POINT AND A 10 GPM CAPACITY

(8) Safety-relief valve BC-PSV-F025A provides overpressure protection. It has a 410 psig set pressure and a 10 gpm capacity.

(9) SAFETY RELIEF VALVE BD-PSV-F017 PROVIDES OVERPRESSURE PROTECTION. IT HAS A 100 PSIG SET POINT AND A 10 GPM CAPACITY

(10) SAFETY RELIEF VALVE BJ-PSV-F020 PROVIDES OVERPRESSURE PROTECTION. IT HAS A 100 PSIG SET POINT AND A 15 GPM CAPACITY.

(11) Safety-relief valve BE-PSV-F012B provides overpressure protection. It has a 500 psig setpoint and a 100 gpm capacity.

(12) Safety-relief valve BE-PSV-F012A provides overpressure protection. It has a 500 psig setpoint and a 100 gpm capacity.

(13) Safety-relief valve BC-PSV-F025C provides overpressure protection. It has a 410 psig setpoint and a 10 gpm capacity.

(14) Safety-relief valve BC-PSV-F025D provides overpressure protection. It has a 410 psig setpoint and a 10 gpm capacity.

(15) 3RD PRESSURE ISOLATION VALVE

HCGS

DSER Open Item No. 103 (DSER Section 3.10.1 & 3.10.2)

SEISMIC AND DYNAMIC QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

Seismic and Dynamic Qualification

1. The applicant needs to address the effects on seismic and dynamic qualification of equipment due to the hydrodynamic loads associated with the suppression pool. In addition, the rigid frequency as referred to in the qualification program will need to be redefined.

2. The applicant needs to assess the fatigue cycling effect on equipment performance due to safety relief valve and other vibratory loadings. This should be addressed for both equipment qualified by analysis and by testing methods, in both NSSS and non-NSSS scopes.

3. The applicant needs to clarify how the conservative restrictions placed on allowable piping loads transmitted to the pump and valve bodies for NSSS supplied items have been demonstrated not to cause detrimental deflections of the active components. The applicant should also clarify how this issue is resolved for BOP equipment.

4. The methods for handling aging and sequential testing in the seismic qualification of both electrical and mechanical equipment should be clarified. In addition, the applicant should commit to establish a maintenance and surveillance program to maintain equipment in a qualified status throughout the life of the plant.

5. The applicant should clarify how General Electric generically qualified equipment is verified as being applicable to Hope Creek.

6. In cases where equipment was qualified by using single axis and/or single frequency testing, the equipment should be identified and, in light of applicable seismic and hydrodynamic loads, the justification of the use of these procedures should be given in each case.

7. There should be a list of equipment types which clearly shows the methods used for qualification. This list should also address which standards are met, in particular those cited in SRP 3.10.

Pump and Valve Operability Assurance

1. There should be a list of equipment types which clearly shows the methods used for qualification. This list should also address which standards are met, in particular those cited in SRP 3.10.

DSER Open Item No. 103 (Cont'd).

2. Clarification of how aging was incorporated in the qualification process should be contained in the FSAR. In addition, the applicant should commit to establish a maintenance and surveillance program to maintain equipment in a qualified status throughout the life of the plant. The criteria for the maintenance and surveillance program should be contained in the FSAR.

3. Identify in the FSAR which valves will be subjected to frequencies higher than 33 Hz (from hydrodynamic loads) and discuss the impact of these dynamic loads on the valve qualification and performance.

4. The FSAR should be amended to clearly show the loads and conditions considered in the qualification of safety related pumps and valves.

5. The extent to which draft standards ANSI/ASME QNPE-1 (N551.1), QNPE-2 (N551.2) QNPE-3 (N551.3), QNPE-4 (N555.4) and N41.6 and issued standard ANSI/ASME B.16.41 are used needs to be clearly stated in the FSAR. In addition, the applicant's position with respect to Regulatory Guide 1.148 must also be indicated in the FSAR.

6. The FSAR should be amended to show the extent to which operational testing is being used to meet the requirements of SRP Section 3.10. The extent to which operational testing is performed at full flow and temperature conditions should be shown.

Equipment Qualification

The following areas under the purview of the Equipment Qualification Branch are open:

(1) Dependability of Containment Isolation

(2) II.D.1, Performance Testing of Boiling Water Reactor and Pressurized Water Reactor Relief and Safety Valves.

(3) II.K.3.28, Verify Qualification of Accumulators on Automatic Depressurization System Valves.

(4) Long Term Operability of Deep Draft Pumps (IE Bulletin 79-15)

RESPONSE

Seismic and Dynamic Qualification

1. HCGS is a BWR/4 with a Mark I containment and a torus pressuresuppression design. The Mark I design limits hydrodynamic loads to the torus itself, effectively insulating equipment and the remainder of the plant from hydrodynamic loads. The only exception is the annulus pressurization

that would follow a pipe break inside the biological shield wall. This dynamic load would be localized to the RPV and attached piping and pipe-mounted equipment. Therefore, for the majority of equipment in the NSSS seismic qualification program, the only significant dynamic load is seismic with its 33-Hz ZPA. For equipment that does experience annulus pressurization, a higher ZPA in the range of 60 to 100 Hz is used depending on the dynamic characteristics of the equipment and its installation.

The hydrodynamic frequencies and frequency range is discussed in the response to FSAR Question 271.7.

2. It should be noted that HCGS is a BWR/4 with a Mark I containment and a torus pressure-suppression design. Hydrodynamic loads due to SRV actuation are limited to the torus itself. In accordance with the Mark I Long Term Program, non-NSSS equipment attached to the torus have been evaluated for appropriate hydrodynamic loads including fatigue effects.

   Vibration fatigue-cycle effects for NSSS equipment designed to ASME B&PV Code requirements were reviewed by NRC consultants from Battelle Pacific Northwest Laboratories at General Electric on October 7, 1980. The consultants stated satisfaction with the General Electric approach, which encompasses OBE, SRV where applicable, thermal, and pressure cycles.

   Non-ASME B&PV code components in the NSSS scope are qualified by tests that address the "strong motion" phase of seismic (and, if applicable, SRV) dynamic motion sufficient to generate the maximum equipment response. These loads are controlling. General Electric testing generally consists of five OBE tests and one SSE test of 30 seconds each. This is about 50% greater than is required to address strong-motion vibration.

   Non-ASME B&PV Code components in the NSSS scope are also qualified by analyses that generally have not, in the past, had to address vibration fatigue-cycle effects. In most cases, such effects are not now part of the qualification record. In accordance with the Mark I Long Term Program, the torus attached piping, piping components and the equipment inside the torus are designed for hydrodynamic loads, including fatigue effects.

   The plant operating vibration loads are insignificant compared to seismic loads considered for equipment qualification. In most cases, the effects of these small vibrations need not be a part of the qualification program since the induced stresses are generally well below the endurance limit of the material. In this case, static plus seismic rather than fatigue qualification governs the design. The operability of equipment is not impaired by these vibrations.

DSER Open Item No. 103 (Cont'd)

3. Deflections due to piping loads and dynamic loads are addressed for active, essential pumps and valves by several methods, depending on the situation. Methods used include static deflection analysis, dynamic deflection analysis, static bend testing, and dynamic seismic testing. Each active, essential pump and valve is evaluated by one or more of the above methods on a case-by-case basis, and the results are summarized in design record files for each component. This information will be available during the SQRT/PVORT on-site audit for each active, essential pump and valve selected for audit. This applies to both NSSS and non-NSSS pumps and valves.

4. The NSSS seismic qualification program for HCGS utilizes seismic data generated over a number of years. Since it was not a licensing requirement at the time, most of these data were developed in earlier years without pre-aging or sequential testing of the equipment. However, NSSS equipment located in harsh environments that has been qualified in recent years has generally been pre-aged and sequentially tested in accordance with the guidelines of IEEE 323-1974.

   NSSS equipment on HCGS is being seismically evaluated using pre-aged and sequential testing data where it is available. Otherwise, the earlier data wihout pre-aging and sequential testing are being used.

   IEEE 323 is used as a basis for establishing an aging program for safety-related non-NSSS electrical equipment. The electrical equipment that must meet aging requirements is subjected to an aging program designed to place the equipment in its end of qualified life condition before performing the seismic tests. The significant aging mechanisms expected to be present during actual service are identified. Age related degradation resulting from exposure to elevated temperature, radiation and cyclical mechanical and electrical stresses under spedified normal, abnormal (excluding DBE) and test conditions anticipated during the installed life of the equipment, are considered. If it is demonstrated that no age-related failure mechanisms exist that can impair the product's ability to perform its safety function throughout its installed life, the equipment to be tested can be considered exempt from age conditioning. The need for aging of a particular piece of equipment is determined based on an evaluation of the specified design, and application.

   The Class 1E electrical equipment located in the harsh environment are qualified per test sequence IEEE 323-1971 and NUREG-0588, category II. However the HCGS environmental qualification program is attempting to upgrade to IEEE 323-1974 and NUREG-0588 category I requirements.

The qualified life of the component materials of the safety-related mechanical equipment (limited to pumps, valves, fans, snubbers, and turbines) located in harsh environments is established by analysis. The analysis is performed only for materials with "significant aging mechanism" as defined in Section 4.4.1 of IEEE Std. 627-1980. Non-metallic materials with a qualified life greater than 40 years are not considered to be susceptible to significant age degradation. Non-metallic parts used in mechanical equipment include gaskets, diaphragms, seals, lubricating oil or grease, fluids for hydraulic systems, flexible hoses, and packing.

In addition, a maintenance and surveillance program to maintain equipment in a qualified status through the life of the plant will be provided for HCGS.

5. PSE&G has contracted with General Electric to reevaluate the seismic qualification of all essential NSSS equipment according to the requirements or recommendations of IEEE 344-1975, Reg. Guides 1.92 and 1.100, and Standard Review Plans 3.9.2 and 3.10. Each item of essential equipment is being reevaluated against Hope Creek's specific requirements, and the results are being summarized in design record files for each component. This information will be available during the SQRT on-site audit for each essential item of equipment selected for audit.

6. The method of qualification for each item of NSSS essential equipment is included in the information being furnished in response to SARI (5). Some equipment is shown to be qualified by single-axis and/or single-frequency testing. The item-by-item reevaluations discussed in the response to Comment No. 5 provide justifications for use of these forms of testing.

   In general, single-axis and/or single-frequency testing was performed in earlier years when testing was performed to the requirements of IEEE 344-1971. Much of this testing has never been repeated using multi-axis and/or multi-frequency techniques. Therefore, the reevaluations discussed in the response to Comment No. 5 reassess this single-axis and/or single-frequency test data to determine if it can be accepted in light of the current criteria of IEEE 344-1975, Regulatory Guides 1.92 and 1.100, and Standard Review Plans 3.9.2 and 3.10.

   In most instances, use of single-axis test data is restricted to equipment with a response that shows a predominant single mode of vibration in each direction with minimal cross coupling. In some cases, if the response shows a single mode of vibration in each direction but also has cross coupling, tha existing single-axis test data are still used if the TRS can be shown to exceed the RRS by a factor of 1.4 over all frequencies.

In most instances, use of single-frequency test data is restricted to cases where the required input motion is dominated by one frequency, where response of the equipment is adequately represented by one mode, or where the input motion has sufficient intensity and duration to produce sufficiently high levels of stress tq assure structural integrity where structural integrity is the determinant requirement. In some cases, if the input motion is sufficiently high so as to excite secondary modes, such that modal responses can be shown to occur out of phase and at a high enough levels, existing single-frequency test data are also used to demonstrate operability.

All pipe mounted non-NSSS valve operators (motor, air, and hydraulic) and accessories are qualified by using a single axis, single frequency testing (RIM test). This is justified on the ground that the seismic floor motion is filtered through the piping system, which generally has one predominent structural mode. Thus the resulting motion that reaches the line mounted equipment is predominently a single frequency and single axis motion. The test is performed by using Required Input Motion (RIM) in each of the three axes independently. No other non-NSSS equipment has been qualified using single axis/single frequency testing.

7. The requested list of equipment types which clearly shows the methods used for qualification is being furnished in response to SRAI (5). This list is updated regularly and includes information received from GE.

Pump and Valve Operability Assurance

1. The list of equipment types and qualification methods is discussed in SQRT Item 7 above.

2. The discussion of how aging is incorporated into the qualification process is discussed in SQRT Item 4 above.

The qualified life of the component materials of the safety-related pumps and valves located in harsh environments is established by analysis. This analysis is performed only for materials with "significant aging mechanism" as defined in Section 4.4.1 of IEEE Std. 627-1980. Non-metallic materials with a qualified life greater than 40 years are not considered to be susceptible to significant age degradation. Non-metallic parts used in mechanical equipment include gaskets, diaphragms, seals, lubricating oil or grease, fluids for hydraulic systems, flexible hoses and packing. In addition, a maintenance and surveillance program to maintain equipment in a qualified status throughout the life of the plant will be provided for HCGS.

DSER Open Item No. 103 (Cont'd)

3. The only active NSSS valves subjected to hydrodynamic loads are the safety relief valves (B21-F013) and the main steam isolation valves (B21-F022). Both of these valve types are being dynamically qualified by test up to the hydrodynamic ZPA.

Non-NSSS six-inch and larger valves subjected to frequencies higher than 33 Hz from hydrodynamic loads are listed in Table 103-1. These valves are contained in piping systems attached to the torus.

The maximum permissible accelerations were developed by analysis for these valves by limiting valve assembly stresses to normal (Level A) allowables. Analysis data was also used to determine the fundamental frequency of each valve which was in turn input into the dynamic analysis of the piping system.

Valve qualification and performance is maintained by limiting dynamic valve accelerations for all load combinations and service levels below the normal (Level A) allowables.

4. The response to FSAR Question 210.52 and DSER open item 93 provides the requested information concerning the loads and conditions considered in the qualification of safety-related pumps and valves.

5. *The PVORT forms for all active, essential pumps and valves selected for audit will be made available at the time of the audit. In addition,* the extent to which draft standards ANSI/ASME QNPE-1 (N551.1), QNPE-2 (N551.2), QNPE-3 (N551.3), QNPE-4 (N551.4) and N41.6 and issue standard ANSI/ASME B.16.41 are used will *also be provided at the time of the audit.*

See Section 1.8.1.148 for PSE&G's position with respect to Regulatory Guide 1.148.

6. *The response to this item will be provided by September, 1984.*

DSER Open Item No. 103 (Cont'd)

Equipment Qualification Branch

Per discussions between Jim Ashley of PSE&G and Dave Wagner of the NRC, the issues of testing containment isolation dependability, SRV performance testing, ADS accumulator qualification, and deep draft pump operability are four open items that are awaiting the NRC's action. During the plant site audit, they will review the HCGS efforts that addressed these issues and determine if any further action by Bechtel or PSE&G is required. No specific Bechtel or PSE&G actions are required at this time.

TABLE 103-1

Six Inch and Larger Non-NSSS Valves Subjected To Hydrodynamic Loads

TAG NOS.

1-BE-HV-F001A to D
1-FC-HV-F059
1-BJ-HV-F004
1-BJ-HV-F042
1-FD-HV-F071
1-BD-HV-F010
1-EE-HV-4680
1-EE-HV-4681
1-BD-HV-F031
1-BC-HV-F004A to D
1-BC-HV-F006A & B
1-EE-HV-4652
1-EE-HV-4679
1-GS-HV-4958
1-BE-HV-F015A & B
1-BC-HV-F024A & B

1-BC-HV-4421
1-BC-HV-4420A & B
1-BC-HV-F027A & B

Rei1

HCGS

DSER Open Items 105, 106 (Section 4.2)

(1) The mechanical fracturing analysis is usually done as a part of the seismic-and-LOCA loading analysis (see Item (2)). The staff has reviewed and approved the generic analytical method used by GE (described in NEDE-21175-3) to determine that fuel-rod mechanical fracturing will not occur as a result of combined seismic-and-LOCA loadings. However, the applicant has not demonstrated that this generic report is applicable to Hope Creek or presented an acceptable alternative. In either case, we require a plant-specific analysis.

(2) Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. SRP Section 4.2 and associated Appendix A state that fuel assembly coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during these low probability accidents. The SRP recommends acceptance criteria to achieve these objectives.

The entire seismic-and-LOCA loading evaluation has been described by GE in the approved topical report NEDE-21175-3.

This Item is similar to Item (1). The applicant must demonstrate that NEDE-21175-3 is applicable to Hope Creek or provide an acceptable alternative along with a plant-specific analysis to show that the criteria given in SRP Section 4.2, Appendix A, are met.

RESPONSE

In accordance with the methods described in NEDE-21175-3 (LTR), the HCGS fuel design was analyzed for the plant-unique seismic and annulus pressurization (AP) loadings. However, the seismic and AP loadings for Hope Creek were calculated by a linear dynamic analysis using the HCGS reactor building model with GE's detailed RPV model.

To address the fuel lift, a screening assessment was performed comparing the Hope Creek unique combined (seismic and AP) input loads at the top of the RPV support skirt (the load input point to the LTR model) with the input loads of other similar BWR plants for which plant-unique nonlinear LTR analyses were performed.

The screening assessment showed that the HCGS plant-unique input loads are well below the input loads of the comparison plants. Since the nonlinear-analysis fuel-lift values for these plants were well below the acceptable fuel-design limits, the HCGS fuel-lift values are expected to be negligible. FSAR Sections 3.9.1.4.10, 3.9.2.3.2.5, and 4.2 have been revised to reflect the results of the screening assessment.

3.9.1.4.8 Control Rod Drive Housing Supports

Examples of the calculated stresses, and the allowable stress limits for the faulted condition for the CRD housing supports, are shown in Table 3.9-5cc.

3.9.1.4.9 Fuel Storage Racks

Examples of the calculated stresses and stress limits for the faulted conditions for the new fuel storage racks are shown in Table 3.9-5u.

3.9.1.4.10 Fuel ~~Channels~~ assembly (Including Channel)

GE boiling water reactor (BWR) fuel ~~channel~~ assembly (including Channel) design bases, analytical methods, and evaluation results, including those applicable to the faulted conditions, are contained in References 3.9-10 and 3.9-11. The acceleration profiles are summerized in Table 3.9-5ee.

3.9.1.4.11 Refueling Equipment

Refueling and servicing equipment important to safety are classified as essential components, per the requirements of 10 CFR 50, Appendix A. This equipment, and other equipment whose failure would degrade an essential component, are defined in Section 9.1 and are classified as Seismic Category I. These components are subjected to an elastic, dynamic, finite-element analysis to generate loadings. This analysis uses appropriate seismic floor response spectra and combines loads at frequencies up to 33 hertz in three directions. Imposed stresses are generated and combined for normal, upset, and faulted conditions. Stresses are compared, depending on the specific safety class of the equipment, to industrial codes; ASME, ANSI or industrial standards, or AISC allowables. The calculated and allowable stresses are summarized in Table 3.9-5u.

3.9.1.4.12 Non-NSSS Seismic Category I System Components

The stress allowables of Appendix F of the ASME B&PV Code, Section III, in effect at the award of each purchase order, were used for Code components. For non-Code components, allowables

3.9.2.3.2.2 Control Rod Drive and Control Rod Drive Housing

The seismic qualification of the control rod drive (CRD) housing, with the CRD enclosed, for an operating basis earthquake (OBE) and a safe shutdown earthquake (SSE), is done analytically, and the stress results of the analyses establish the structural integrity of these components. Preliminary tests were conducted to verify the operability of the CRD during a seismic event. A simulated test, imposing a static bow in the fuel channels, was performed to show the CRD function satisfactorily.

3.9.2.3.2.3 Core Support - Fuel Support and Control Rod Guide Tube

No dynamic testing of the control rod guide tube is conducted. However, a detailed analysis imposing dynamic effects due to seismic events shows that the maximum stresses developed during these events are much lower than the maximum allowed for the component material.

3.9.2.3.2.4 Hydraulic Control Unit

The seismic loads adequacy of the hydraulic control unit (HCU), for the faulted condition, is demonstrated by test and analysis. With the HCUs mounted on a seismic support structure, the dynamic loads are 1.8g vertical at the natural frequency of 7 to 30 Hz, and 1.75g horizontal at 2 to 6 Hz and 4g horizontal at 10 Hz. At these frequencies, the maximum HCU capability (by test) for dynamic loads is 20g vertical at 7 to 30 Hz, and greater than 4g horizontal at 2 to 6 Hz and 8g horizontal at 10 Hz.

3.9.2.3.2.5 Fuel ~~Channels~~ assembly (Including Channel)

Refer to Section 3.9.1.4.10

~~GE boiling water reactor (BWR) fuel channel design bases, analytical methods, and evaluation results, including seismic considerations, are contained in References 3.9-10 and 3.9-11.~~

3.9.2.3.2.6 Recirculation Pump and Motor Assembly

Calculations are made to ensure that the recirculation pump and motor assembly is designed to withstand the specific static equivalent seismic forces. The flooded assembly is analyzed as a

Analysis," Nuclear Engineering and Design, 4, 1966.

3.9-7 E. Wilson, "Structural Analysis of Axisymmetric Solids," AIAA Journal, 3 (112), December 1965.

3.9-8 P.J. Schneider, Temperature Response Charts, John Wiley and Sons, Inc., 1963.

3.9-9 Sample Analysis of a Class I Piping System, prepared by the Working Group on Piping (SDG, ScIII) of the ASME Boiler and Pressure Vessel Code, December 1971.

3.9-10 General Electric, BWR Fuel Channel Mechanical Design and Deflection, NEDE-21354-P, September 1976.

3.9-11 General Electric, BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings, NEDE-21175-P, November 1976, and NEDE-21175-3-P, July 9

3.9-12 General Electric, Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants, NEDE-24057-P (Class III) and NEDO-24057 (Class I), November 1977.

3.9-13 General Electric, Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves, APED 5750, March 1969.

3.9-14 General Electric, Atomic Power Equipment Department, Design and Performance of GE BWR Jet Pumps, APED-5460, July 1968.

3.9-15 H.H. Moen, Testing of Improved Jet Pumps for the BWR/6 Nuclear System, NEDO-10602, General Electric, Atomic Power Equipment Department, June 1972.

TABLE 3.9-5 (cont)

v. High Pressure Coolant Injection Pump

w. Control Rod Drive

x. Control Rod Drive Housing

y. Jet Pumps

z. LPCI Coupling

aa. Control Rod Guide Tube

bb. In-core Instrument Housing

cc. Reactor Vessel Support Equipment CRD Housing Support

dd. HPCI Turbine

ee. Fuel Assembly (Including Channel)

TABLE 3.9-5ee

FUEL ASSEMBLY (INCLUDING CHANNEL)

ACCEPTANCE CRITERIA	LOADING	PRIMARY LOAD TYPE	CALCULATED PEAK (2) ACCELERATION	EVALUATION BASIS ACCELERATION
Acceleration Envelope	Horizontal Direction: 1. Peak Pressure 2. Safe Shutdown Earthquake 3. Annulus Pressurization	Horizontal Acceleration Profile	2.5 G	(1)
	Vertical Direction: 1. Peak Pressure 2. Safe Shutdown Earthquake	Vertical Accelerations	0.3G	(1)

NOTES:

(1) Evaluation Basis Accelerations and Evaluations are contained in NEDE-21175-P and NEDE-21175-3-P.

(2) For the most limiting load combination, the fuel assembly gap opening for Hope Creek is expected to be negligible. This is based on an assessment comparing the positive net hold down forces to those of other plants for which the calculated fuel assembly gap opening is found to be negligible.

4.2 FUEL SYSTEM DESIGN

The fuel system design for the HCGS is identical to that which the NRC reviewed and approved for GESSAR II (Reference 4.2-1). Methods and criteria used to evaluate fuel system performance are also identical to those used for GESSAR II, The results of the NRC review of Section 4.2 of GESSAR II documented in References 4.2-2 and 4.2-3 are therefore applicable to the HCGS.

4.2.1 REFERENCES

4.2-1 General Electric Standard Safety Analysis Report, Docket No. 50-447

4.2-2 NUREG-0979, "Safety Evaluation Report Related to the Final Design Approval of the GESSAR II BWR/6 Nuclear Island Design", April, 1983

4.2-3 NUREG-0979 (Supplement No. 1), "Safety Evaluation Report Related to the Final Design Approval of the GESSAR II BWR/6 Nuclear Island Design", July, 1983

except for the evaluation of combined fuel-lift loadings from a safe shutdown earthquake and a loss-of-coolant accident. See Section 3.9.1.4.10 and Table 3.9-5ee for the results of the fuel-lift evaluation.

DSER Open Item No. 109 (DSER Section 4.4.7)

TMI-2, ITEM II.F.2

The NRC staff has reviewed SLI-8211, which provides the BWROG evaluation of existing water level instruments and recommendations for their improvement. The preliminary conclusion of the NRC staff review indicates that present procedural changes identified in the procedure guidelines are adequate for the short term, but permanent physical improvements for the existing water level systems should be made to reduce the burden on the operator. The results of the NRC staff review as they apply to Hope Creek are summarized as follows:

1. The applicant should consider the BWROG recommendations for upgrading the water level instrumentation to reduce the errors caused by high drywell temperature.

2. The applicant should evaluate the water level systems for Hope Creek to determine if operator action is needed to mitigate the consequences of a break in a reference leg and a single failure in a protection channel associated with an intact reference leg. If operator action is needed to mitigate the consequences of the cited event, the applicant should consider changing the protection system logic (for reactor trip and/or ESF system(s) actuation on reactor vessel low water level) so that this is accomplished automatically.

3. The applicant should identify the type of water level indication equipment used for Hope Creek. If the mechanical level indication equipment is used, the applicant should develop a plan for replacing it with analog level transmitters and trip units to reduce the vulnerability to failures or malfunctions.

The second BWROG report, SLI-8218, presents evaluation results of additional instrumentation as diverse indicators of ICC and provides recommendations regarding the need for such additional instrumentation (including incore thermocouples) for BWR plant monitoring systems. The NRC staff is still reviewing the reports, SLI-8211 and SLI-8218, and will provide the evaluation results in a supplement to this SER.

For the NRC staff to reach conclusions concerning the instrumentation requirements for the Hope Creek reactor, the applicant must submit a plant-specific evaluation addressing the applicant's position with respect to the BWROG recommendations and the results of the NRC staff review of SLI-8211 (discussed above) for upgrading the existing water level instruments. If the applicant chooses not to modify the existing water level systems, the applicant must address the adequacy and reliability of the existing water level systems for responding to excessive drywell temperature,

DSER Open Item No. 109 (Cont'd)

reactor depressurization, rupture of a water level reference leg, failure of water level transmitter, and setpoint trip mechanism drag, as identified in SLI-8211. The evaluation should also address the applicability of the BWROG findings in SLI-8218 regarding the need for additional instrumentation for Hope Creek. The review of Item II.F.2 will continue and the NRC staff evaluation results will be included in a supplement to this report.

RESPONSE

The information requested above has been ~~will be~~ addressed in the responses to Questions 421.21 and 421.23.

QUESTION 421.21 (SECTIONS 7.2, 7.3, 7.4, 7.5)

Provide an evaluation of the effects of high temperatures on reference legs of water level measuring instruments subsequent to high-energy line breaks, including the potential for reference leg flashing/boil off, the indication/annunciation available to alert the control room operator of erroneously high vessel level indications resulting from high temperatures, and the effects on safety systems acuation (e.g., delays).

RESPONSE

~~An evaluation of this issue is in progress. Based on the results of this analysis, proposed modifications, if any, to the HCGS level monitoring instrumentation design will be provided to the NRC when available. This is estimated to be about August, 1984.~~

—INSERT A—

~~QUESTION 421.21 (SECTIONS 7.2, 7.3, 7.4, 7.5)~~

~~Provide an evaluation of the effects of high temperatures on reference legs of water level measuring instruments subsequent to high-energy line breaks, including the potential for reference leg flashing/boiloff, the indication/annunciation available to alert the control room operator of erroneously high vessel level indications resulting from high temperatures, and the effects on safety systems actuation (e.g., delays).~~ e

— INSERT A —

~~RESPONSE~~ e

An evaluation of the effects of high temperatures on reference legs of water level measuring instruments subsequent to High Energy Line Breaks (HELB) is divided into two parts: 1) the effects of temperature alone, and 2) the effects of flashing/boiloff.

High Temperature Effects (without flashing/boiloff)

An increase in the temperature of the drywell will cause a heat-up of the fluid in the instrument sensing lines, contributing to sensor error. The HCGS instrument sensing line design reduces this error by routing the variable leg and the reference leg lines with equivalent elevation drops in the drywell. The only exceptions to this design are the Upset Range transmitters reference leg sensing lines. Physical configuration prevents equivalent routing of these lines. However, these transmitters are used exclusively for indication and will not present any challenges to plant safety.

A high drywell temperature alarm is computer generated from isolated outputs of class 1E temperature transmitters. Class 1E temperature recorders located in the main control room provide a continuous display of drywell temperature.

Flashing/Boiloff Effects

The effect of flashing/boiloff of the instrument line reference leg is to cause the level instruments to indicate erroneously high levels. The amount of error is directly related to the drop in elevation of piping physically located within the drywell and subject to flashing.

HCGS has rerouted two channels of reactor pressure vessel (RPV) level instrumentation sensing lines to provide a maximum 3-ft elevation drop in the drywell (maximum 1-ft drop for the reference legs). A worst case analysis of the effects of boiloff of that portion of the sensing line inside the drywell, indicates the instruments using the rerouted lines will indicate a level that is 1.3 ft higher than actual. During and after an HELB the operator is required to maintain RPV level within the normal operating range, 18 ft above the top of active fuel. The 1.3 ft error is negligible with respect to the operating requirements.

Transmitters used for post accident monitoring use the rerouted lines. Threfore, the wide, narrow, and fuel zone range recorders and indicators will provide an unambiguous display of level even after partial flashing of the reference legs.

As a result of an HELB in containment, the drywell temperature may reach a maximum of 340°F. Flashing/boiloff of the sensing lines may occur when the RPV pressure is less than 118 PSIA when the drywell temperature is 340°F. At the 118 PSIA RPV pressure the high pressure coolant injection system (HPCI) and the automatic depressurization system (ADS) are not required.

In response to a HELB of a large or intermediate sized line (see figure 15.9-43) low pressure coolant injection (LPCI) and core spray are initiated by low water Level 1 (L1) or high drywell pressure signals. For these postulated events, HPCI and ADS are not required.

Two different response paths must be considered for a small break accident (SBA).

The first response path considers an SBA with HPCI available. The emergency core cooling system (ECCS) response to an SBA is outlined in FSAR Chapter 15 in response to event 42 (Figure 15.9-43). Core spray and LPCI are initiated by high drywell pressure. HPCI is initiated on receipt of a low Level 2 or high drywell pressure signal. HPCI continues to operate until the reactor vessel pressure is below the pressure at which LPCI or core spray operation can maintain core cooling. LPCI and core spray are designed to begin injecting water into the RPV when the differential pressure between the RPV and the suppression chamber is approximately 300 psid per design requirements (see FSAR Chapter 6.3).

The second response path considers a HPCI line SBA that incapacitates HPCI. Accident mitigation requires the actuation of the automatic depressurization system (ADS), LPCI, and core spray. LPCI and core spray are initiated on high drywell pressure or a L1 signal. ADS is initiated by a L1 and high drywell pressure and a L3 permissive signal when low pressure ECCS pumps are running. At the point flashing could occur, the RPV pressure will be low enough that ADS will not be required; before that point level signals/actuations will remain accurate.

In the event of any credible HELB inside containment, the capability of the ECCS to mitigate the accident is not compromised by high drywell temperature or flashing of the RPV level instrumentation line reference legs.

GE

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QUESTION 421.23 (SECTIONS 7.2, 7.3, 7.4)

Operating reactor experience indicates that a number of failures have occurred in BWR reactor vessel level sensing lines and that in most cases the failures have resulted in erroneously high reactor vessel level indication. For BWRs, common sensing lines are used for feedwater control and as the basis for establishing vessel level channel trips for one or more of the protective functions (reactor scram, MSIV closure, RCIC, LPCI, ADS or HPCS initiation). Failures in such sensing lines may cause a reduction in feedwater flow and consequential defeat of a trip within the related protective channel.

If an additional failure, perhaps of electrical nature, is assumed in a protective channel not dependent on the failed sensing line, protective action may not occur or may be delayed long enough to result in unacceptable consequences. This depends on the logic for combining channel trips to achieve protective actions.

Identify each case where a reactor vessel water level tap or sensing line failure concurrent with an additional random single electrical failure induces a transient and precludes the automatic operation of reactor scram and/or engineered safety feature system. For each case identified provide an evaluation which demonstrates how the redundancy or diversity of the plant design provides for reactor scram or safety system operation within acceptable limits. Where manual action is required by the operators discuss the instrumentation and time available for the operator to take such corrective action.

To reduce the consequences of sensing line failures in combination with a single failure in a protection channel not dependent on the failed sensing line, a modification of the protection system logic may be required.

BWROG generic report SLI-8211 indicates that early operator action would be required to initiate either HPCI or RCIC in the event of a loss of the reference leg connected to the level sensor which is controlling feedwater combined with the failure of a level sensor, control component or power supply bus associated with the intact reference leg instruments. The specific level sensors are N091, A, B, C, D (Figure 5.1-4) and the buses are 125 Vdc A and B. Provide a description of the modifications implemented at Hope Creek as a result of this concern or provide justifications why the modifications discussed in the generic report are not necessary to reduce the consequence of sensing line failures.

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RESPONSE

~~The concerns addressed above are being evaluated against the HCGS design. A justification why modifications are not necessary or a description of proposed modifications will be provided by July 1984.~~

An analysis was conducted based on the following assumption that simultaneously:

1. An instrument reference line fails (breaks),

2. A single electrical device also fails (but there is no power supply failure), and

3. There is no operator action.

These postulated multifailures are beyond the design basis for the HCGS; however, an assessment of the plant responses to these types of events was provided.

The instrument reference lines common to feedwater control and to protective system sensors were identified. All the various failure combinations were examined. Two failure combinations that represent the worst postulated failure paths were identified. These two failure combinations are described in what follows.

<u>Failure Combination 1</u> would be the failure of the division 1* instrument reference line connected to condensing chamber B21-D004A combined with a failure such that ~~it~~ indicates a high water level. In the analysis of this combination, it was assumed that the manual selection switch for feedwater control is on the failed instrument line (division 1) and that the operator does not switch the control to the other instrument line (division 2) as would be expected. This would cause the feedwater controller to respond to the erroneous high-level signal by reducing the feedwater flow.

Following the loss of feedwater flow, the decrease of the water level to level 4 would initiate a low water level alarm. After the water level decreased to level 3, a second low water level alarm would be initiated, but a reactor scram would not occur due to the assumed failures.

level transmitter B21-N080C

* ie Channel A

When the water level decreased to level 2, a reactor scram would occur due to the alternate rod insertion system, and a third low water level alarm would be initiated. The RCIC system would then automatically start, and both recirculation pumps would trip. However, HPCI system would be unavailable (tripped) due to the assumed failures.

Core uncovery analyses were performed using the REDY program and simulations that represent the beginning-of-cycle (BOC) and end-of-cycle (EOC) void-reactivity coefficients.

The case with the EOC void-reactivity coefficient showed that the minimum water level would be between level 1 and level 2. Figures 421.23-1 and 421.23-2 show the REDY plots for the cases with the BOC void-reactivity coefficient and the EOC void-reactivity coefficient, respectively. The case with the BOC void-reactivity coefficient showed that the minimum water level would be below level 1 outside the shroud and would trigger the closure of the MSIVs.

For the BOC void-reactivity case, a further analysis, based on realistic assumptions, was performed to evaluate the potential for core heatup. This analysis applied the power history that resulted from the core uncovery analysis until the level-2 scram signal occurred at approximately 42 seconds. After 42 seconds, the ANS 1979 best-estimate decay-heat values were used.

Figures 421.23-3 through 421.23-5 show the system pressure, water level inside the shroud, and peak cladding temperature (PCT), respectively, calculated from the core-heatup analysis. The minimum water level in the core would be 2.5 feet below the top of the active fuel (inside the shroud). This uncovery level would result in a PCT of 450°F. Since this PCT is less than the initial cladding temperature of 644°F and well below the 2200°F limit, these results are acceptable from an ECCS viewpoint.

<u>Failure Combination 2</u> would be the failure of the division 2* instrument reference line connected to condensing chamber B21-D004B combined with a B21-N097 D- or H-level transmitter failure such that it indicates a high water level. In the analysis of this combination, it was assumed that the manual selection switch for feedwater control is on the failed instrument line (division 2) and that the operator does not switch the control to the other instrument line (division 1) as would be expected. This would cause the feedwater controller to respond to the erroneous high-level signal by reducing the feedwater flow. Following the loss of feedwater flow, the water level would decrease to level 4, initiating a low water level alarm. After the water level decreased to level 3, a second low water level alarm would be initiated, and reactor scram would occur. After the water level decreased to level 2, a third low water level alarm would initiate, the HPCI system would automatically start, and both recirculation pumps would trip. The RCIC system would be unavailable (tripped) due to the assumed failures.

A core uncovery analysis was performed using the REDY program, and simulating the BOC void-reactivity coefficient only, since it presents the worse reactor condition for this scenario.

Figure 421.23-6 shows the REDY plot for this case. It can be seen that the minimum water level outside the shroud would be about 10 feet above the top of the active fuel. No core uncovery was found.

*ie Channel B

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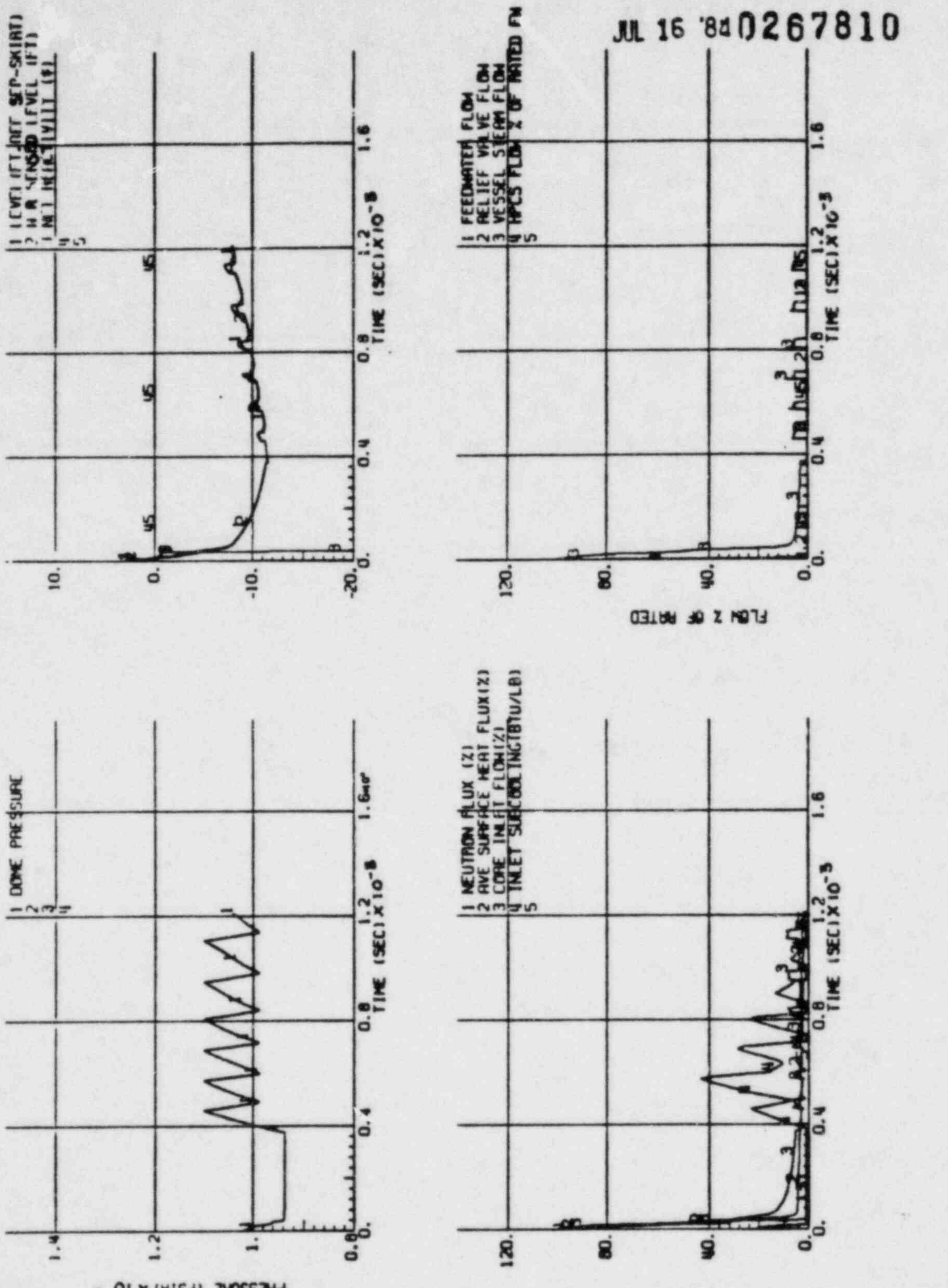

HOPE CREEK
GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

INSTRUMENT LINE BREAK
ANALYSIS, BOC; NO HPCI;
RCIC AVAILABLE WITH ARI

FIGURE 421.23-1 AMENDMENT 7, 08/84

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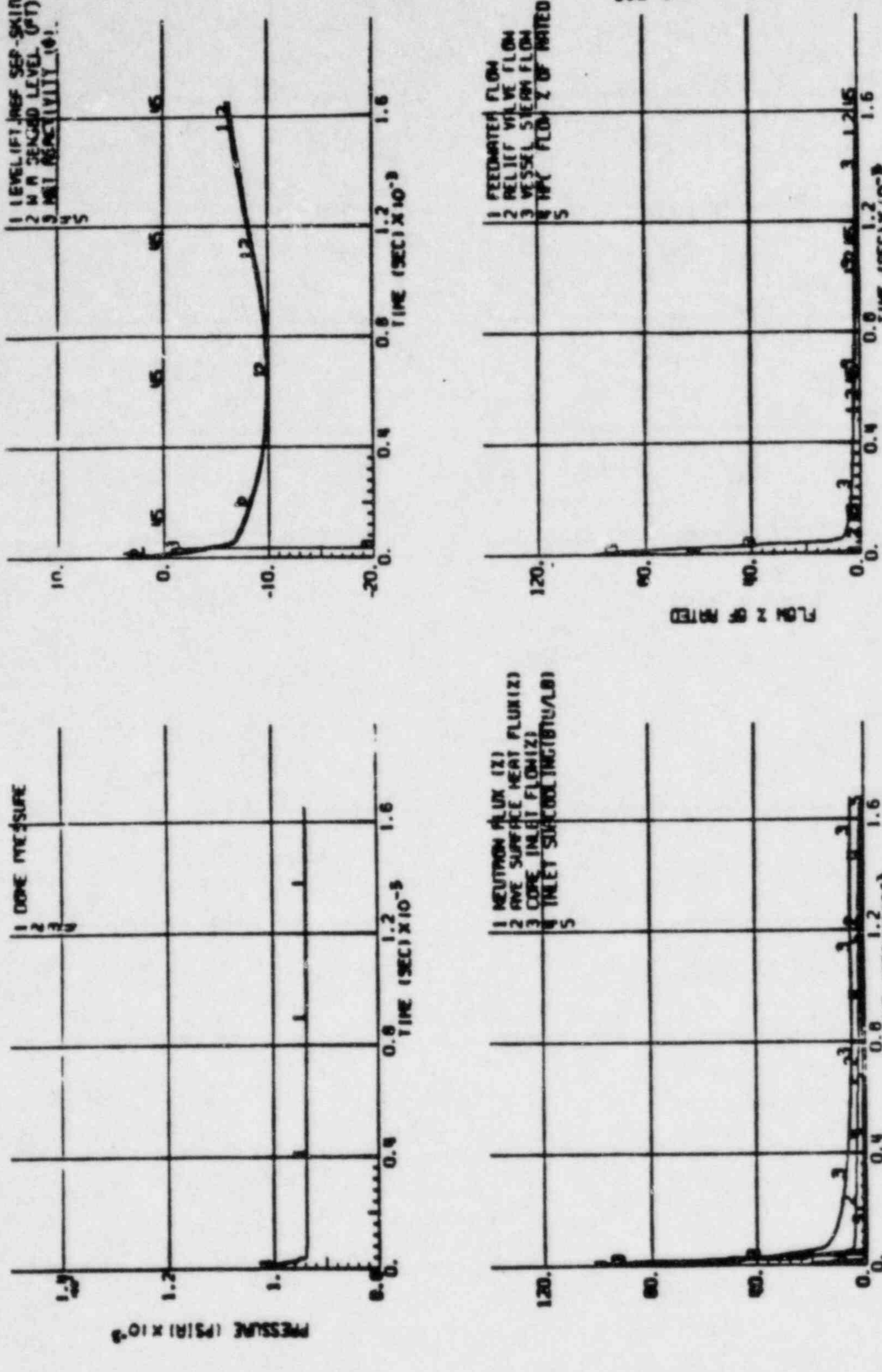


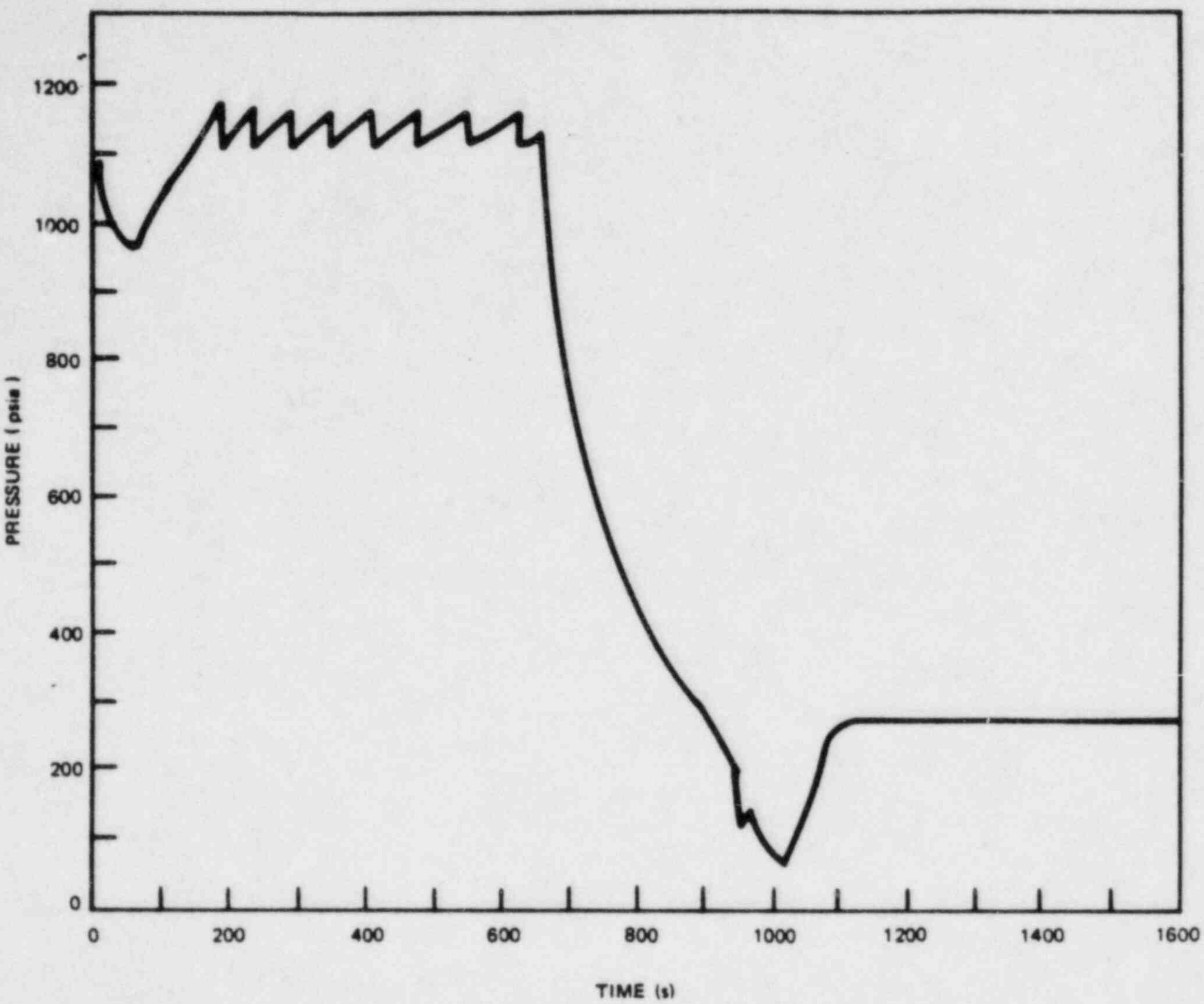
HOPE CREEK
GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

INSTRUMENT LINE BREAK
ANALYSIS, EOC; NO HPCI;
RCIC AVAILABLE WITH ARI

FIGURE 421.23-2 AMENDMENT 7, 08/84

JUL 16 '84 0267810



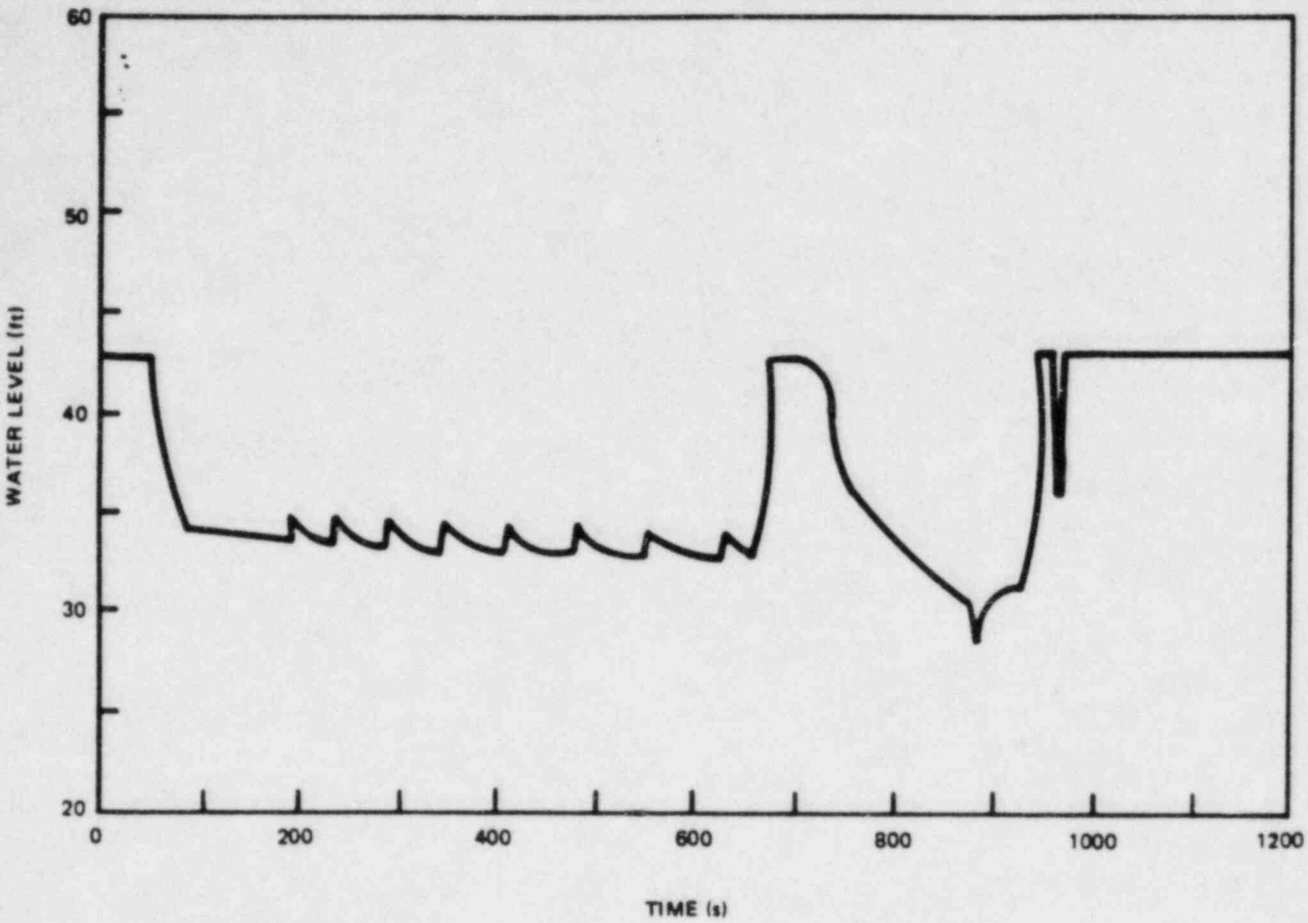
HOPE CREEK
GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

VESSEL PRESSURE CALCULATED
FROM THE CORE - HEATUP
ANALYSIS

FIGURE 421.23-3 AMENDMENT 7, 08/84

JUL 16 '84 0267810

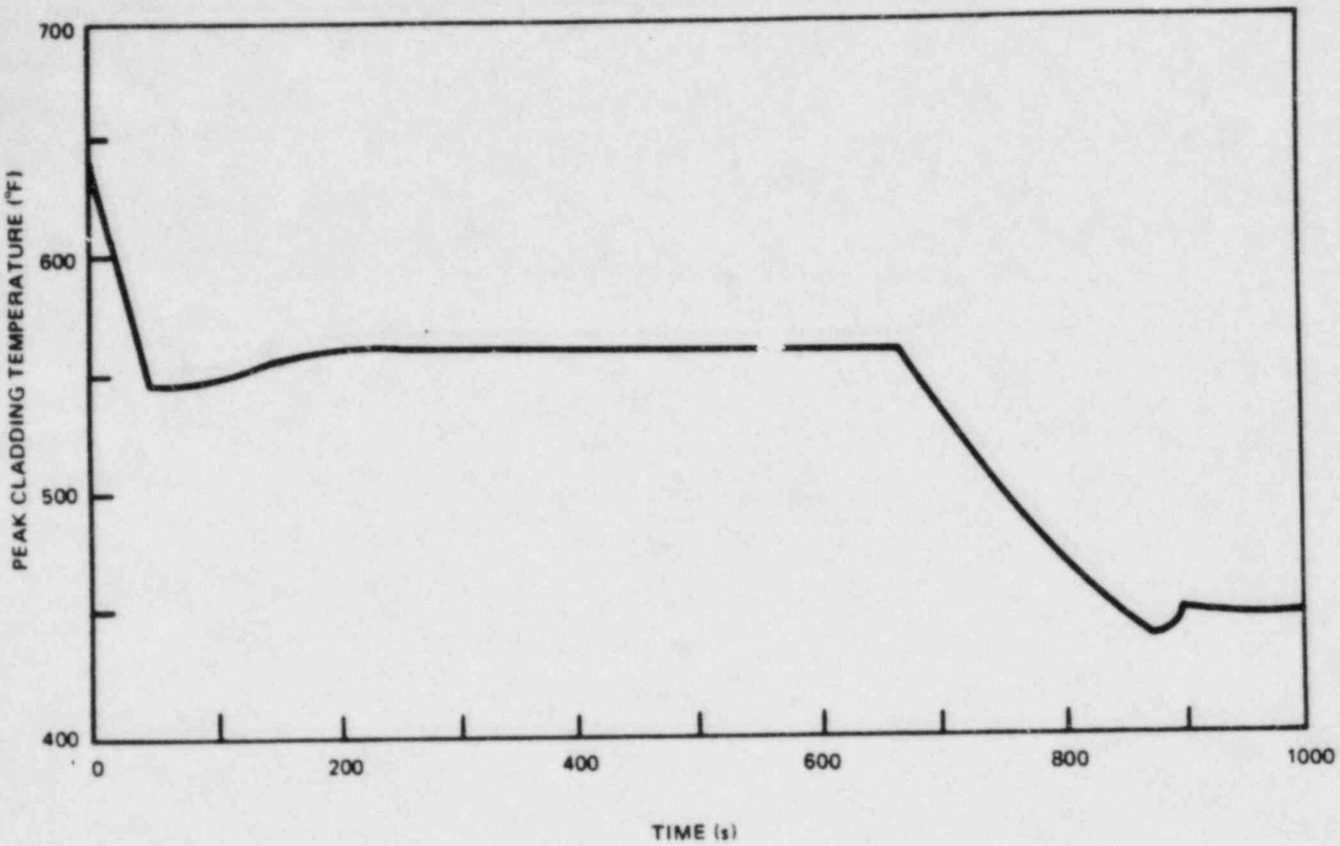


HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

WATER LEVEL CALCULATED FROM THE CORE - HEATUP ANALYSIS

FIGURE 421.23-4 AMENDMENT 7, 08/84

JUL 16 '84 0267810



HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

PEAK CLADDING TEMPERATURE
CALCULATED FROM THE
CORE - HEATUP ANALYSIS

FIGURE 421.23-5 AMENDMENT 7, 08/84

HCGS

DSER Open Item No. 117 (DSER Section 5.3.4)

COMPLIANCE WITH NB 2332 OF WINTER 1972 ADDENDA OF THE ASME CODE

To demonstrate that the ferritic RCPB materials in the MSIV meet the requirements of Paragraph NB 2332 of the Winter 1972 Addenda of the ASME Code, provide:

(a) thickness of MSIV bodies and covers

(b) connecting pipe sizes

(c) lowest service metal temperature

RESPONSE

For the information requested above, see the response to Question 251.6.

QUESTION 251.6

TÕ demonstrate that the ferritic RCPB materials in the MSIV meet the requirements of Paragraph NB 2332 of the Winter 1972 Addenda of the ASME Code, provide:

a. Thickness of MSIV bodies and covers

b. Connecting pipe sizes

c. Lowest service metal temperature

RESPONSE

The thickness of MSIV bodies and covers are 1.925 and 5.095 inches, respectively; the connecting pipe size is 26 inches; and the lowest service metal temperature is 70°F

JUN -5 '84 0265854

DSER Open Item No. 118 (DSER Section 5.3.4)

LEAD FACTORS AND NEUTRON FLUENCE FOR SURVEILLANCE CAPSULES.

Provide lead factors and predicted neutron fluence to be received by each surveillance capsule at the time of its withdrawal.

RESPONSE

For the information requested above, see the response to Question 251.7.

RUS -0 8408208214

HOPE CREEK FSAR

QUESTION 251.7: (REVISED AUGUST 1, 1984)

Provide lead factors and predicted neutron fluence to be received by each surveillance capsule at the time of their withdrawal.

RESPONSE

Lead factors have been calculated using the base locations of the sample and nominal dimensions of the vessel. The lead factors are defined as the ratio of the neutron flux at the surveillance sample to the highest neutron flux at the wall of the vessel. The lead factor at the vessel inside diameter is 0.86 and the lead factor at one quarter of the vessel thickness is 1.20.

The END-OF-LIFE (EOL) calculated peak fluence at the inside diameter of the vessel is 1.7 x 10^{18} n/cm² (E>1.0MeV) and at one quarter of the vessel thickness is 1.1 x 10^{18} n/cm² (E>1.0MeV). The withdrawal of the capsules will be according to the following criteria:

a. The first set will be withdrawn when its exposure corresponds to ~~the calculated exposure of the reactor vessel wall at 25% of the reactor design life.~~ SIX EFFECTIVE FULL-POWER YEARS OR AT THE TIME WHEN THE ACCUMULATED NEUTRON FLUENCE OF THE CAPSULE EXCEEDS 5X10^{18} n/cm² OR AT THE TIME WHEN THE HIGHEST PREDICTED ΔRT_{NDT} OF ALL THE ENCAPSULATED MATERIALS IS APPROX. 50°F WHICHEVER COMES FIRST.

b. The second set will be withdrawn when its exposure corresponds to ~~the calculated exposure of the reactor vessel wall at 75% of the reactor design life.~~ WHEN THE ACCUMULATED NEUTRON FLUENCE OF THE CAPSULE CORRESPONDS TO THE APPROXIMATE EOL FLUENCE AT THE REACTOR VESSEL INNER WALL LOCATION, WHICHEVER COMES FIRST.

c. The third set will be ~~a spare to be withdrawn based on previously developed data.~~ HELD TO THE EOL (NOT LESS THAN ONCE, NOR MORE THAN TWICE THE PEAK EOL VESSEL FLUENCE AT THE VESSEL INSIDE SURFACE

~~Based on these criteria, the first specimens would be withdrawn after 11.6 years of operation with a fast neutron fluence of 4.2 x 10^{17} n/cm². The second set would be withdrawn with a fast neutron fluence of 1.3 x 10^{18} n/cm².~~

The construction tolerances on the reactor vessel required that the minimum (nominal) radius of the vessel be maintained. The applicable version of the ASME B&PV Code did allow for areas of the vessel to have larger radii. The measurement acceptance techniques for the vessel were either the use of a template to test the minimum diameter or a series of measurements to determine the diameter at various points. The measurement technique did not require the identification of the locations where the vessel diameter is longer than nominal. Hence the lead factors were calculated for the nominal dimension.

If an area of increased vessel diameter were to coincide with a location of the surveillance sample specimens, the correct fluence at the samples would be less than that predicted from measurements on the samples. If these data were used to predict the peak fluences, the values would be less than the calculated peak fluences. The calculated peak fluences using nominal dimensions will be conservative.

RWB
8/2/84

DSER Open Item No. 120 (DSER Section 6.2)

TMI ITEM II.E.4.2

The staff has reviewed the HCGS for compliance with the requirements of Item II.E.4.2 (NUREG-0737). Its evaluation and findings are summarized as follows:

Item (1), (2), (3), and (7) cannot be reviewed until the information required for the review of Section 6.2.4 "Containment Isolation System" is provided by the applicant.

Item (6) cannot be reviewed until the information required for the review of Section 6.2.4.1 (Containment Purge System) is provided by the applicant.

RESPONSE

For the information requested above see the response to DSER open item 132 (items 1, 2, 3 & 7) and 133 (item 6).

Rev1

HCGS

DSER Open Item No. 124 (DSER Section 6.2.1.5.1)

RPV SHIELD ANNULUS ANALYSIS

The applicant's analysis resulted in pressures in the shield annulus that peak at approximately 90 psia in the volumes surrounding the recirculation line break and approximately 100 psia in the volumes surrounding the feedwater line break. The applicant has not provided a graphical presentation of the differential pressures (psi) responses as a function of time for a selected number of nodes, as requested.

In addition, the applicant has not provided the peak and transient loading on the major components used to establish the adequacy of the supports design. This should include the load forcing functions (e.g., fx(t), fy(t), fz(t)) and transient moments (e.g., Mx(t), My(t), Mz(t)) as resolved about a specific identified coordinate system. The applicant also has not provided the projected areas used to calculate these loads. This information was also previously requested. The staff intends to perform confirmatory analysis using the COMPARE code upon receipt of this information.

RESPONSE

The graphical presentation of differential pressure is not required per March 30, 1984 conference call between the NRC and Bechtel. Bechtel noted that the initial containment pressure could be considered constant during the transient and thus differential pressure can be determined by subtracting a constant intial pressure from the already provided graphical presentations of absolute pressures.

The forcing functions and moments from a recirculation line break for the reactor pressure vessel from annulus pressurization are provided in Tables 124-2 and 124-3 and Figures 124-1 through 124-7. These values are not used in the analysis performed for the HCGS design. They have been generated for your information and are based on the RPV Shield Annulus Analysis discussed in Appendix 6B of the FSAR.

124-3

The forces are calculated without a Z component as there is no uplifting force. The moments are calculated using the top of the pedestal, elevation 112 feet-8 1/2 inches, as the reference. Figure 124-8 shows the coordinate system used in the analysis.

The requested projected areas for the RPV Shield Annulus Analysis are provided in Table 124-1.

TABLE 124-1

Projected Areas (in^2)

NODE	VESSEL AX	VESSEL AY	SHIELD AX	SHIELD AY
1	5747	1540	6875	1842
2	4207	4207	5033	5033
3	1540	5747	1842	6875
4	1540	5747	1842	6875
5	4207	4207	5033	5033
6	5747	1540	6875	1842
7	5747	1540	6875	1842
8	4207	4207	5033	5033
9	1540	5747	1842	6875
10	1540	5747	1842	6875
11	4207	4207	5033	5033
12	5747	1540	6875	1842
13	8281	2219	9906	2654
14	6062	6062	7252	7252
15	2219	8281	2654	9906
16	2219	8281	2654	9906
17	6062	6062	7252	7252
18	8281	2219	9906	2654
19	8281	2219	9906	2654
20	6062	6062	7252	7252
21	2219	8281	2654	9906
22	2219	8281	2654	9906
23	6062	6062	7252	7252
24	8281	2219	9906	2654
25	6343	1700	7588	2033
26	4644	4644	5555	5555
27	1700	6343	2033	7588
28	1700	6343	2033	7588
29	4644	4644	5555	5555
30	6343	1700	7588	2033
31	6343	1700	7588	2033
32	4644	4644	5555	5555
33	1700	6343	2033	7588
34	1700	6343	2033	7588
35	4644	4644	5555	5555

~~JUL 13 '84 0267763~~

TABLE 124-1
(cont.)

	VESSEL		SHIELD	
NODE	AX	AY	AX	AY
36	6343	1700	7588	2033
37	6343	1700	7588	2033
38	4644	4644	5555	5555
39	1700	6343	2033	7588
40	1700	6343	2033	7588
41	4644	4644	5555	5555
42	6343	1700	7588	2033
43	6343	1700	7588	2033
44	4644	4644	5555	5555
45	1700	6343	2033	7588
46	1700	6343	2033	7588
47	4644	4644	5555	5555
48	6343	1700	7588	2033
49	8347	2237	9985	2676
50	6111	6111	7310	7310
51	2237	8347	2676	9985
52	2237	8347	2676	9985
53	6111	6111	7310	7310
54	8347	2237	9985	2676
55	8347	2237	9985	2676
56	6111	6111	7310	7310
57	2237	8347	2676	9985
58	2237	8347	2676	9985
59	6111	6111	7310	7310
60	8347	2237	9985	2676
61	3876	1038	4636	1242
62	2837	2837	3394	3394
63	1038	3876	1242	4636
64	1038	3876	1242	4636
65	2837	2837	3394	3394
66	3876	1038	4636	1242
67	3876	1038	4636	1242
68	2837	2837	3394	3394
69	1038	3876	1242	4636
70	1038	3876	1242	4636
71	2837	2837	3394	3394
72	3876	1038	4636	1242

GK/em
F2(9)

TABLE 124-2

PAGE 1

HCGS - ANNULUS PRESSURIZATION ANALYSIS *Recirculation Line Break*
FORCE-TIME HISTORIES

TIME SEC	X-FORCE LBF	Y-FORCE LBF	NET FORCE LBF	ANGLE DEG
.000000	-6.738281-02	1.574707-02	6.919836-02	1.797704+02
1.000000-03	7.608848+03	-4.006628+01	7.608953+03	3.599947+02
2.000000-03	2.573807+03	-4.006628+01	2.574118+03	3.599844+02
3.000000-03	-1.060102+04	-4.007019+01	1.060109+04	1.800038+02
4.000000-03	-3.148812+04	-4.007019+01	3.148815+04	1.800013+02
5.000000-03	-5.818239+04	-4.006238+01	5.818240+04	1.800007+02
6.000000-03	-9.067675+04	-4.007410+01	9.067676+04	1.800004+02
7.000000-03	-1.259239+05	-4.006628+01	1.259239+05	1.800003+02
8.000000-03	-1.703845+05	-3.862097+01	1.703845+05	1.800002+02
9.000000-03	-2.342563+05	-3.788269+01	2.342563+05	1.800002+02
1.000000-02	-3.126419+05	-2.729675+01	3.126419+05	1.800001+02
1.100000-02	-4.025834+05	-4.257019+01	4.025834+05	1.800001+02
1.200000-02	-5.014111+05	5.729309+01	5.014111+05	1.799999+02
1.300000-02	-6.056673+05	5.129309+01	6.056673+05	1.799999+02
1.400000-02	-7.138278+05	1.143868+02	7.138278+05	1.799998+02
1.500000-02	-8.324846+05	1.851700+02	8.324846+05	1.799998+02
1.600000-02	-9.611132+05	2.367915+02	9.611132+05	1.799998+02
1.700000-02	-1.093677+06	2.641883+02	1.093677+06	1.799995+02
1.800000-02	-1.230277+06	3.393215+02	1.230277+06	1.799967+02
1.900000-02	-1.372130+06	3.411167+02	1.372130+06	1.799998+02
2.000000-02	-1.427276+06	3.677637+02	1.427276+06	1.799997+02
2.100000-02	-1.456728+06	3.324355+02	1.456728+06	1.799998+02
2.200000-02	-1.478755+06	3.473865+02	1.478755+06	1.799998+02
2.300000-02	-1.498403+06	2.958733+02	1.498403+06	1.799998+02
2.400000-02	-1.512664+06	2.465719+02	1.512664+06	1.799998+02
2.500000-02	-1.527458+06	2.299933+02	1.527458+06	1.799999+02
3.000000-02	-1.546753+06	5.040229+02	1.546753+06	1.799997+02
3.500000-02	-1.492878+06	-1.322065+03	1.492879+06	1.800009+02
4.000000-02	-1.440640+06	-6.678142+03	1.440656+06	1.800046+02
4.500000-02	-1.412026+06	-1.455829+04	1.412101+06	1.800103+02
5.000000-02	-1.448091+06	-2.015793+04	1.448231+06	1.800139+02
5.500000-02	-1.538435+06	-1.281496+04	1.538488+06	1.800083+02
6.000000-02	-1.646604+06	-7.532260+03	1.646621+06	1.800046+02
7.000000-02	-1.779800+06	1.911333+03	1.779801+06	1.799989+02
8.000000-02	-1.731093+06	-3.313389+03	1.731097+06	1.800019+02

TABLE 124-2

PAGE 2

HCGS - ANNULUS PRESSURIZATION ANALYSIS *RECIRCULATION LINE BREAK*
FORCE-TIME HISTORIES

TIME SEC	X-FORCE LBF	Y-FORCE LBF	NET FORCE LBF	ANGLE DEG
9.000000-02	-1.540031+06	6.827487+03	1.540046+06	1.799956+02
1.000000-01	-1.364762+06	6.030278+02	1.364762+06	1.799996+02
1.100000-01	-1.294004+06	-5.872842+02	1.294005+06	1.800005+02
1.200000-01	-1.248021+06	-1.364286+03	1.248022+06	1.800011+02
1.300000-01	-1.223457+06	-6.683634+03	1.223475+06	1.800055+02
1.400000-01	-1.208038+06	-6.540907+03	1.208056+06	1.800054+02
1.500000-01	-1.196622+06	-1.491383+04	1.196715+06	1.800125+02
1.600000-01	-1.173453+06	-1.886393+04	1.173604+06	1.800161+02
1.700000-01	-1.145289+06	-6.046721+03	1.145305+06	1.800053+02
1.800000-01	-1.114820+06	5.027324+03	1.114831+06	1.799955+02
1.900000-01	-1.095348+06	1.110712+04	1.095404+06	1.799899+02
2.000000-01	-1.079972+06	3.432707+03	1.079977+06	1.799968+02
2.200000-01	-1.059611+06	6.579709+03	1.059631+06	1.799938+02
2.400000-01	-1.037077+06	1.592520+04	1.037199+06	1.799848+02
2.600000-01	-1.017690+06	2.675903+01	1.017690+06	1.800000+02
2.800000-01	-1.002975+06	8.853946+03	1.003014+06	1.799912+02
3.000000-01	-9.833149+05	1.350421+04	9.834076+05	1.799863+02
3.200000-01	-9.677014+05	-2.257962+04	9.679647+05	1.800233+02
3.400000-01	-9.611205+05	-2.985006+04	9.615839+05	1.800310+02
3.600000-01	-9.425995+05	-7.789616+03	9.426317+05	1.800083+02
3.800000-01	-9.560509+05	2.220758+04	9.563088+05	1.799768+02
4.000000-01	-9.532052+05	8.411509+02	9.532056+05	1.799991+02
4.200000-01	-9.244254+05	-1.450998+04	9.245393+05	1.800157+02
4.400000-01	-9.546061+05	2.517601+04	9.549381+05	1.799736+02
4.600000-01	-9.573411+05	1.729713+04	9.574973+05	1.799819+02
4.800000-01	-9.054635+05	-1.947732+04	9.056729+05	1.800215+02
5.000000-01	-9.041225+05	-1.581827+04	9.042609+05	1.800175+02
5.000000-01	-9.041225+05	-1.581827+04	9.042609+05	1.800175+02

TABLE 124-3

HCGS - ANNULUS PRESSURIZATION ANALYSIS RECIRCULATION LINE BREAK
MOMENT-TIME HISTORIES

TIME SEC	X-MOMENT LBF-FT	Y-MOMENT LBF-FT	Z-MOMENT LBF-FT	NET MOMENT LBF-FT
.000000	-2.078125-01	-1.500000+00	1.884365+00	2.422250+00
1.000000-03	1.555242+03	9.376312+04	1.862798+00	9.377602+04
2.000000-03	1.555242+03	3.110012+04	1.870886+00	3.113899+04
3.000000-03	1.555242+03	-1.329369+05	1.881669+00	1.329460+05
4.000000-03	1.555242+03	-3.917769+05	1.887060+00	3.917800+05
5.000000-03	1.555242+03	-7.218684+05	1.892452+00	7.218700+05
6.000000-03	1.554992+03	-1.124040+06	1.914018+00	1.124041+06
7.000000-03	1.555242+03	-1.562459+06	1.959847+00	1.562460+06
8.000000-03	1.535492+03	-2.100455+06	1.946368+00	2.100455+06
9.000000-03	1.523492+03	-2.845888+06	1.976022+00	2.845888+06
1.000000-02	1.301992+03	-3.791418+06	2.013763+00	3.791418+06
1.100000-02	1.519367+03	-4.876779+06	2.067679+00	4.876780+06
1.200000-02	-8.222578+02	-6.070025+06	2.089245+00	6.070025+06
1.300000-02	-5.672578+02	-7.331540+06	2.135074+00	7.331540+06
1.400000-02	-1.455289+03	-8.640870+06	2.172815+00	8.640870+06
1.500000-02	-2.867914+03	-1.002592+07	2.238862+00	1.002592+07
1.600000-02	-2.667523+03	-1.155430+07	2.303561+00	1.155430+07
1.700000-02	-2.238570+03	-1.311140+07	2.404654+00	1.311140+07
1.800000-02	-3.544336+03	-1.472658+07	2.468005+00	1.472658+07
1.900000-02	-3.268047+03	-1.647536+07	2.519226+00	1.647536+07
2.000000-02	-3.375187+03	-1.716434+07	2.652668+00	1.716434+07
2.100000-02	-1.627875+03	-1.753045+07	2.575837+00	1.753045+07
2.200000-02	-6.815859+02	-1.780010+07	2.575837+00	1.780010+07
2.300000-02	1.534203+03	-1.804640+07	2.658059+00	1.804640+07
2.400000-02	4.097758+03	-1.827855+07	2.775326+00	1.827856+07
2.500000-02	8.033437+03	-1.856911+07	2.779370+00	1.856911+07
3.000000-02	4.804337+04	-1.994980+07	3.007165+00	1.994985+07
3.500000-02	1.327244+05	-2.040606+07	3.177001+00	2.040649+07
4.000000-02	3.318842+05	-1.986198+07	3.287529+00	1.986475+07
4.500000-02	5.822771+05	-1.809751+07	3.418275+00	1.810688+07
5.000000-02	7.816637+05	-1.643715+07	3.536890+00	1.645573+07
5.500000-02	4.808140+05	-1.601492+07	3.782208+00	1.602214+07
6.000000-02	2.493209+05	-1.653144+07	4.022134+00	1.653332+07
7.000000-02	-8.011866+04	-1.809439+07	4.375284+00	1.809457+07
8.000000-02	2.346889+05	-1.839364+07	4.714955+00	1.839514+07

TABLE 124-3

HCGS - ANNULUS PRESSURIZATION ANALYSIS RECIRCULATION LINE BREAK
MOMENT-TIME HISTORIES

TIME SEC	X-MOMENT LBF-FT	Y-MOMENT LBF-FT	Z-MOMENT LBF-FT	NET MOMENT LBF-FT
9.000000-02	-5.727282+04	-1.578037+07	4.758088+00	1.578048+07
1.000000-01	1.810631+05	-1.350112+07	5.003405+00	1.350234+07
1.100000-01	5.716190+04	-1.308533+07	5.151675+00	1.308545+07
1.200000-01	-9.743656+04	-1.293597+07	5.254115+00	1.293634+07
1.300000-01	9.407658+04	-1.303808+07	5.507520+00	1.303842+07
1.400000-01	7.520003+04	-1.310165+07	5.798667+00	1.310187+07
1.500000-01	3.170262+05	-1.330942+07	5.936153+00	1.331319+07
1.600000-01	4.823748+05	-1.306301+07	5.903803+00	1.307191+07
1.700000-01	7.849003+04	-1.277479+07	6.052072+00	1.277503+07
1.800000-01	-1.368890+05	-1.236260+07	6.122183+00	1.236335+07
1.900000-01	-2.778115+05	-1.208259+07	6.256953+00	1.208578+07
2.000000-01	-4.916672+04	-1.194426+07	6.475313+00	1.194437+07
2.200000-01	-2.845628+05	-1.187566+07	6.455138+00	1.187907+07
2.400000-01	-4.579718+05	-1.183391+07	6.852725+00	1.184277+07
2.600000-01	2.992488+05	-1.116353+07	6.866204+00	1.116754+07
2.800000-01	-7.905656+04	-1.123117+07	6.930903+00	1.123145+07
3.000000-01	-5.846664+05	-1.098900+07	6.990211+00	1.100350+07
3.200000-01	4.144294+05	-1.063826+07	7.084564+00	1.064633+07
3.400000-01	4.774502+05	-1.061063+07	7.184308+00	1.062137+07
3.600000-01	-4.425219+04	-1.034831+07	7.192396+00	1.034841+07
3.800000-01	-6.534471+05	-1.098374+07	7.351193+00	1.100316+07
4.000000-01	-6.563872+04	-1.072063+07	7.405364+00	1.072074+07
4.200000-01	1.485773+05	-1.004840+07	7.332577+00	1.004950+07
4.400000-01	-7.592507+05	-1.106722+07	7.170829+00	1.109324+07
4.600000-01	-4.061152+05	-1.134717+07	7.246312+00	1.135440+07
4.800000-01	5.772636+05	-1.001576+07	7.187004+00	1.003238+07
5.000000-01	4.100969+05	-9.710417+06	7.456584+00	9.719073+06
5.000000-01	4.100969+05	-9.710417+06	7.456584+00	9.719073+06

FIGURE 124-1

HCGS - A.P. ANALYSIS X-FORCE RECIRCULATION LINE BREAK

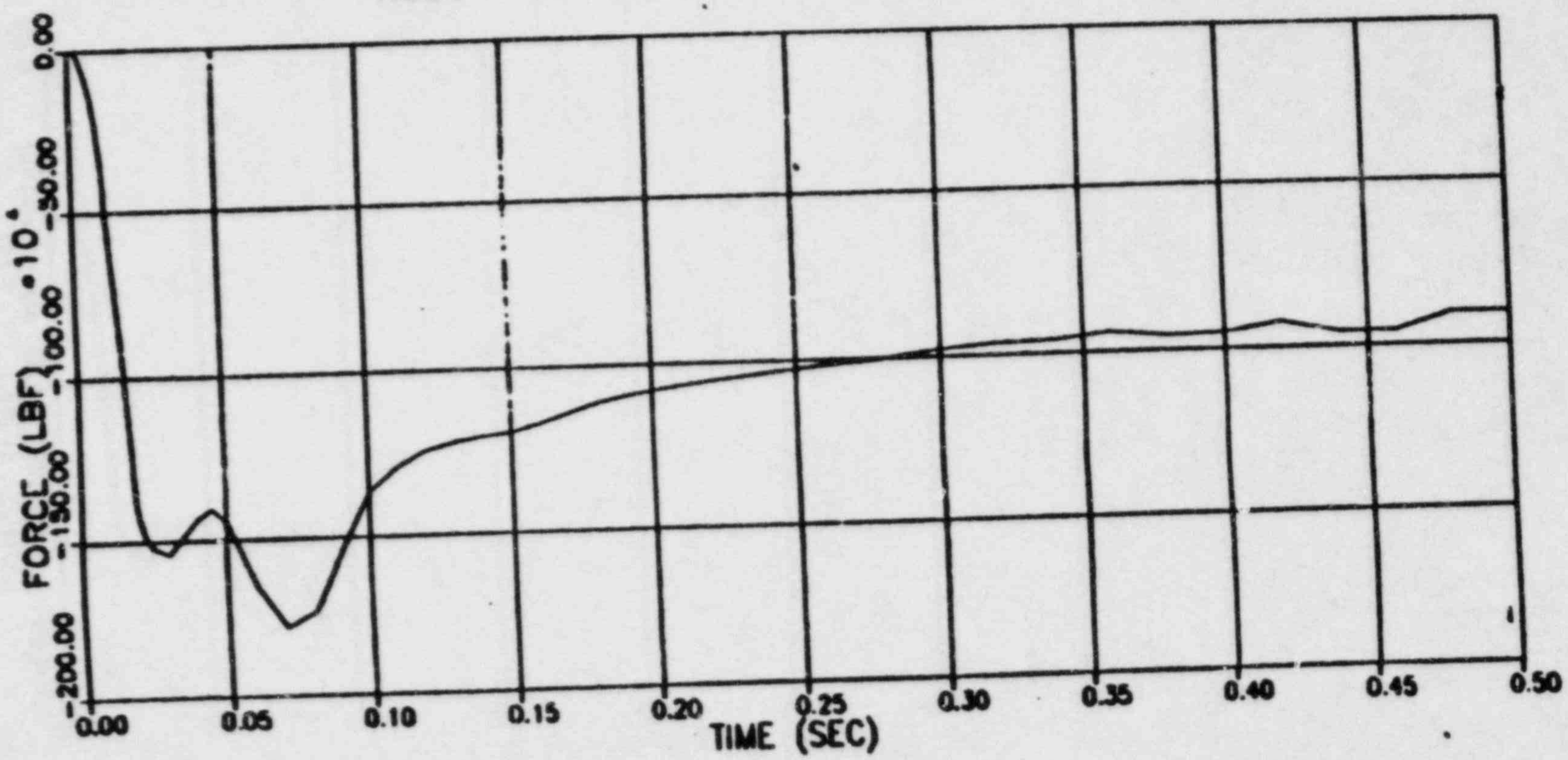


FIGURE 124-2

HCGS - A.P. ANALYSIS Y-FORCE RECIRCULATION LINE BREAK

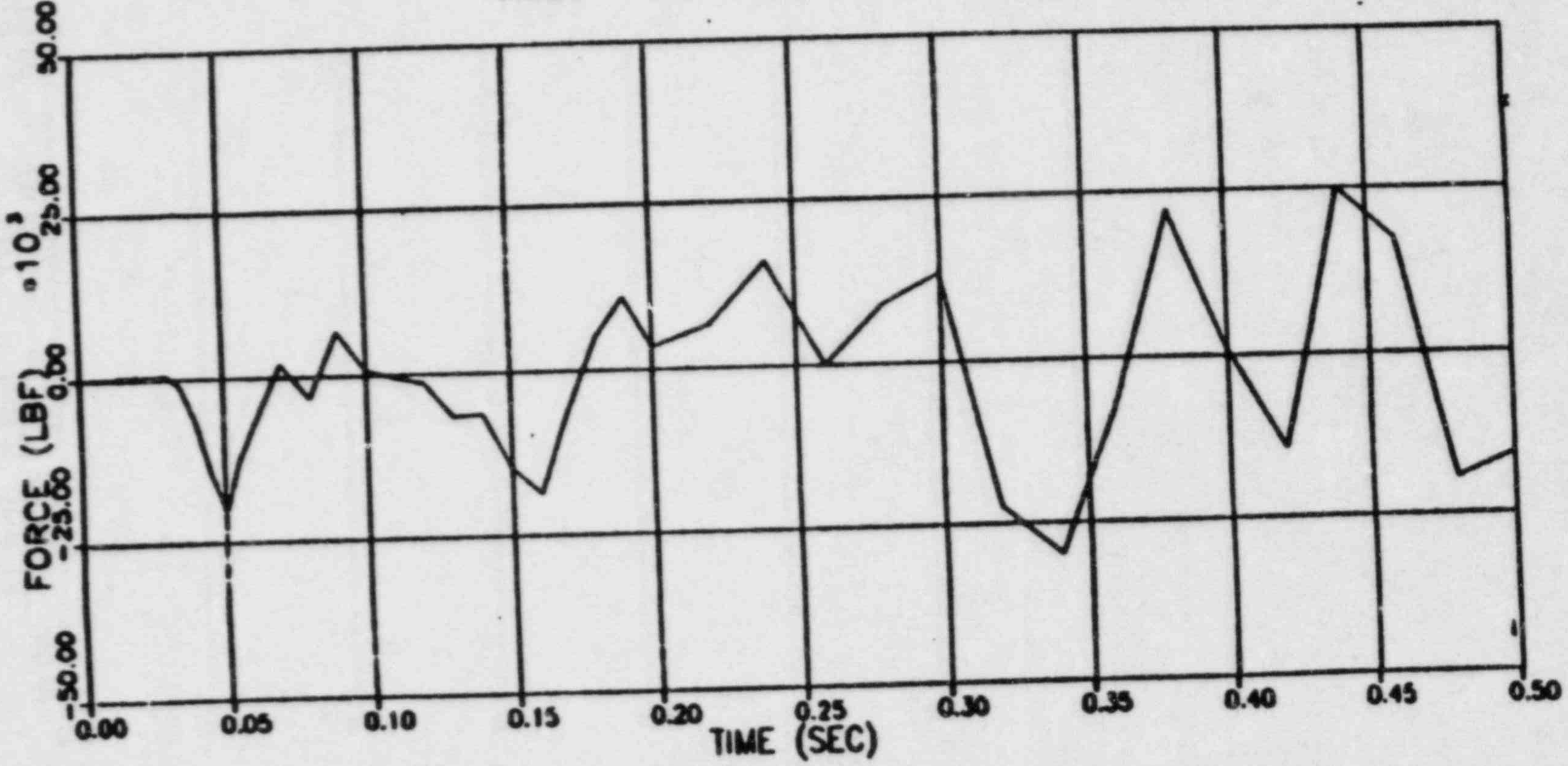
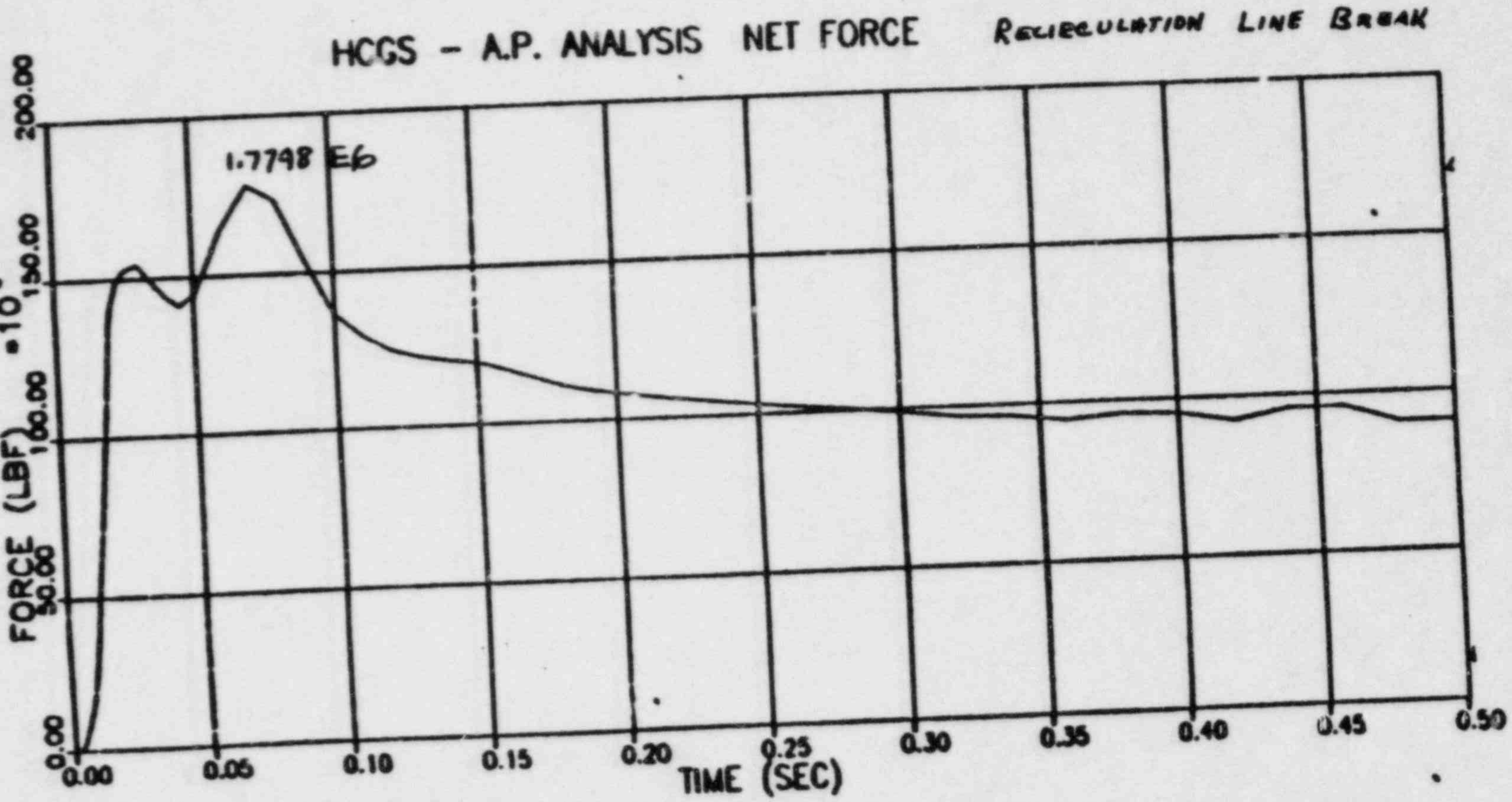


FIGURE 124-3



1 △

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FIGURE 124-4

HCGS - A.P. ANALYSIS X-MOMENT RECIRCULATION LINE BREAK

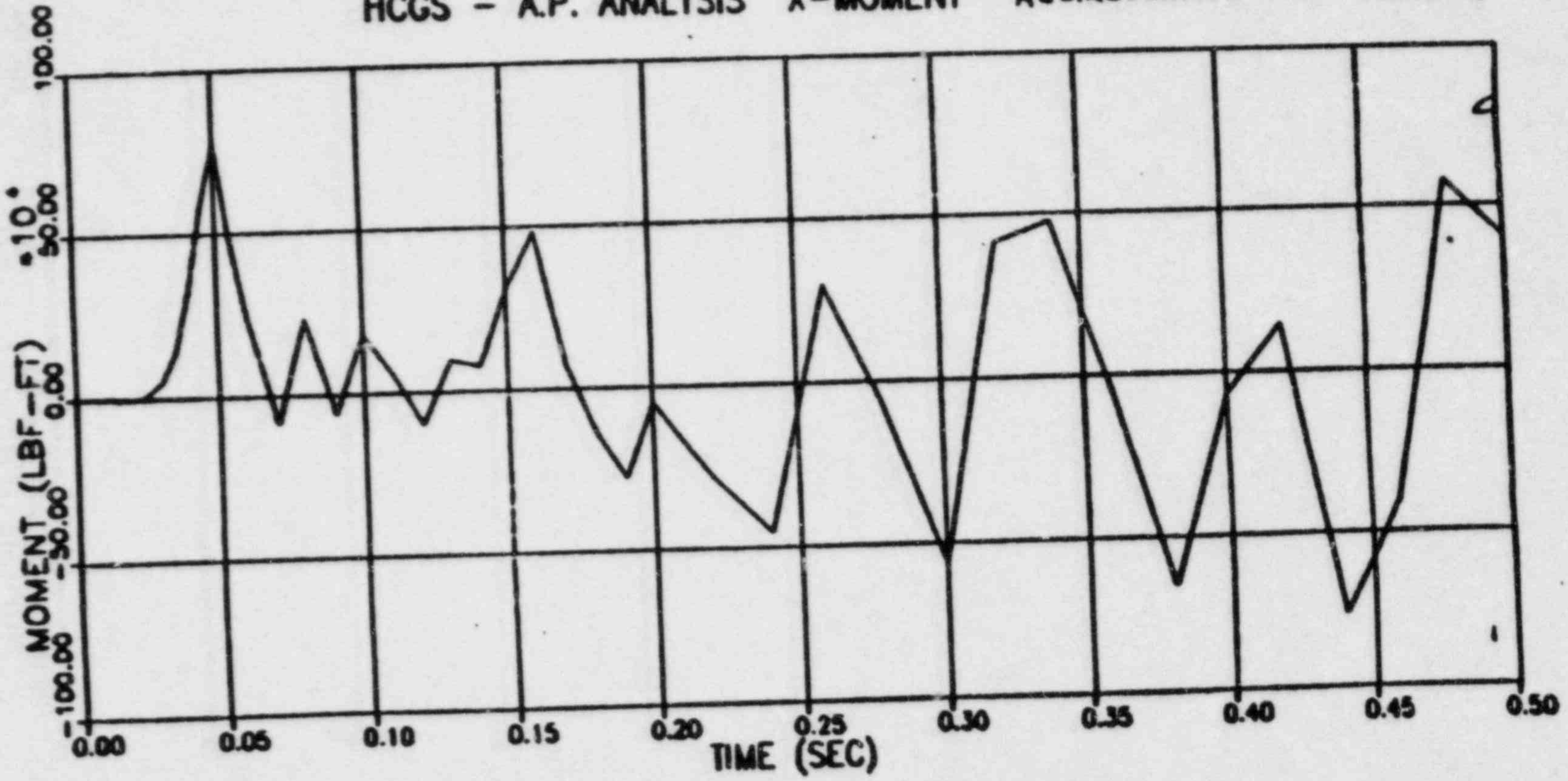

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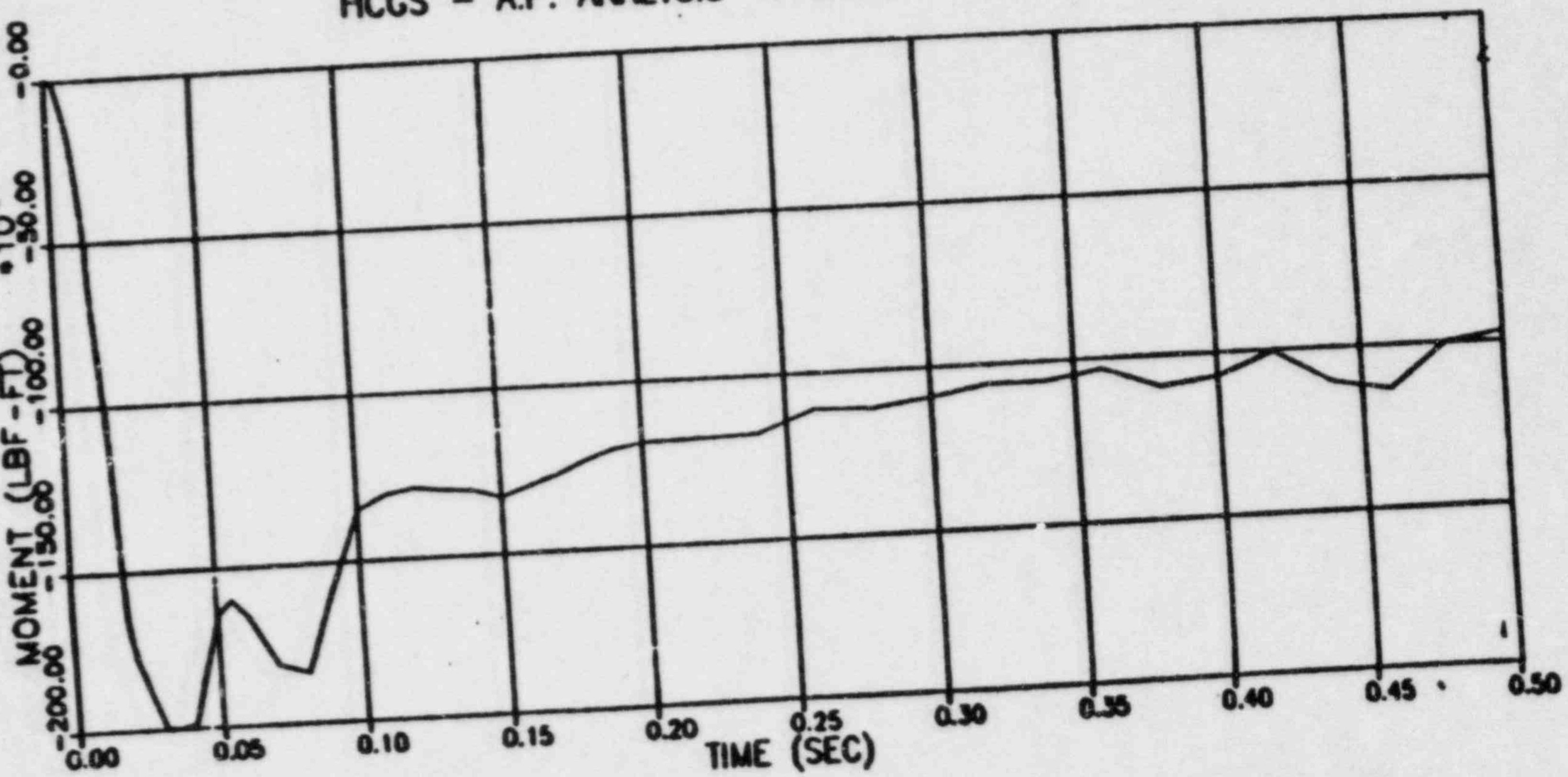
FIGURE 124-5

1 △

HCGS - A.P. ANALYSIS Y-MOMENT Recirculation Line Break



8409260276

FIGURE 124-6

1

HCGS - A.P. ANALYSIS Z-MOMENT RECIRCULATION LINE BREAK

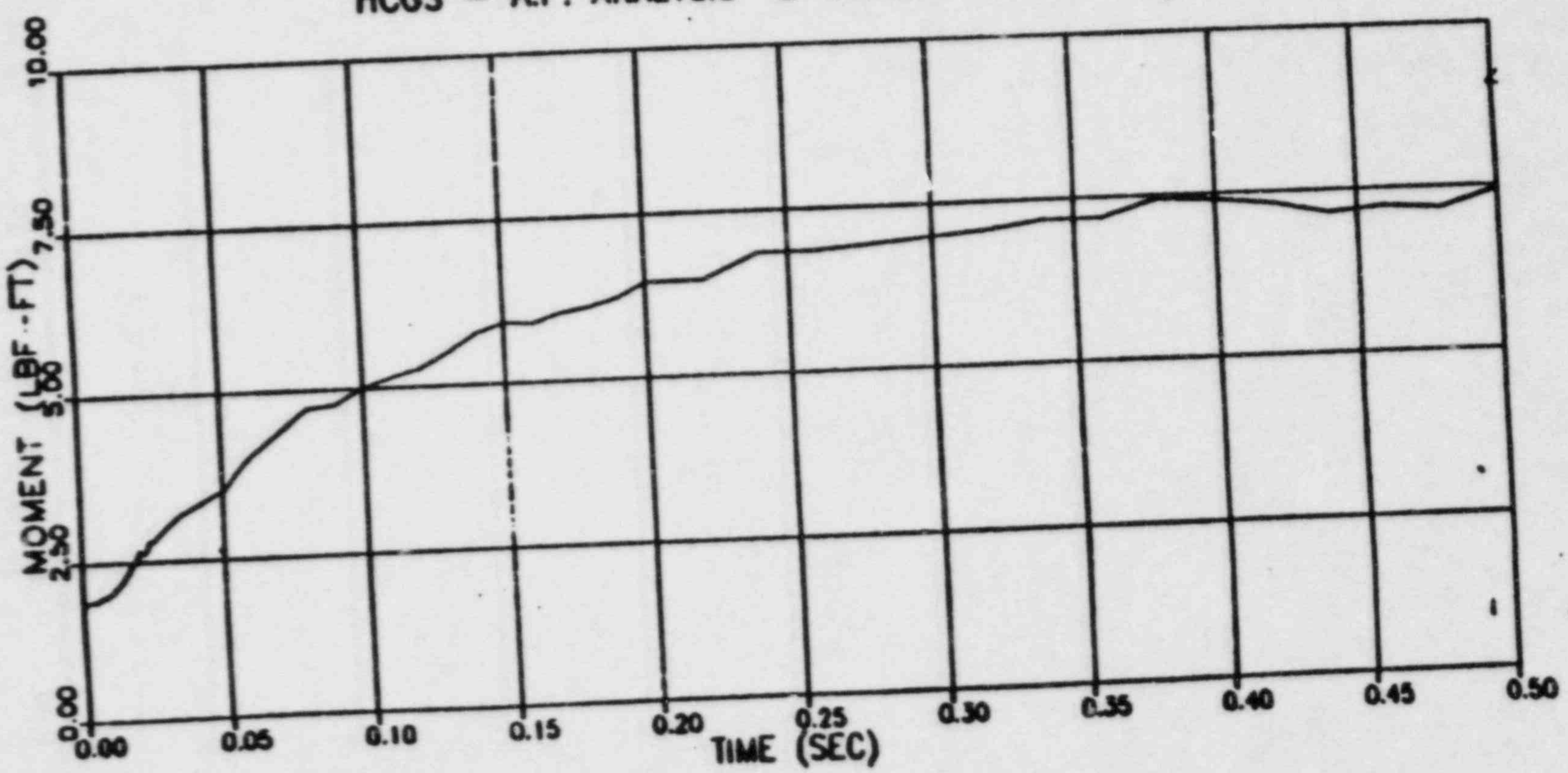
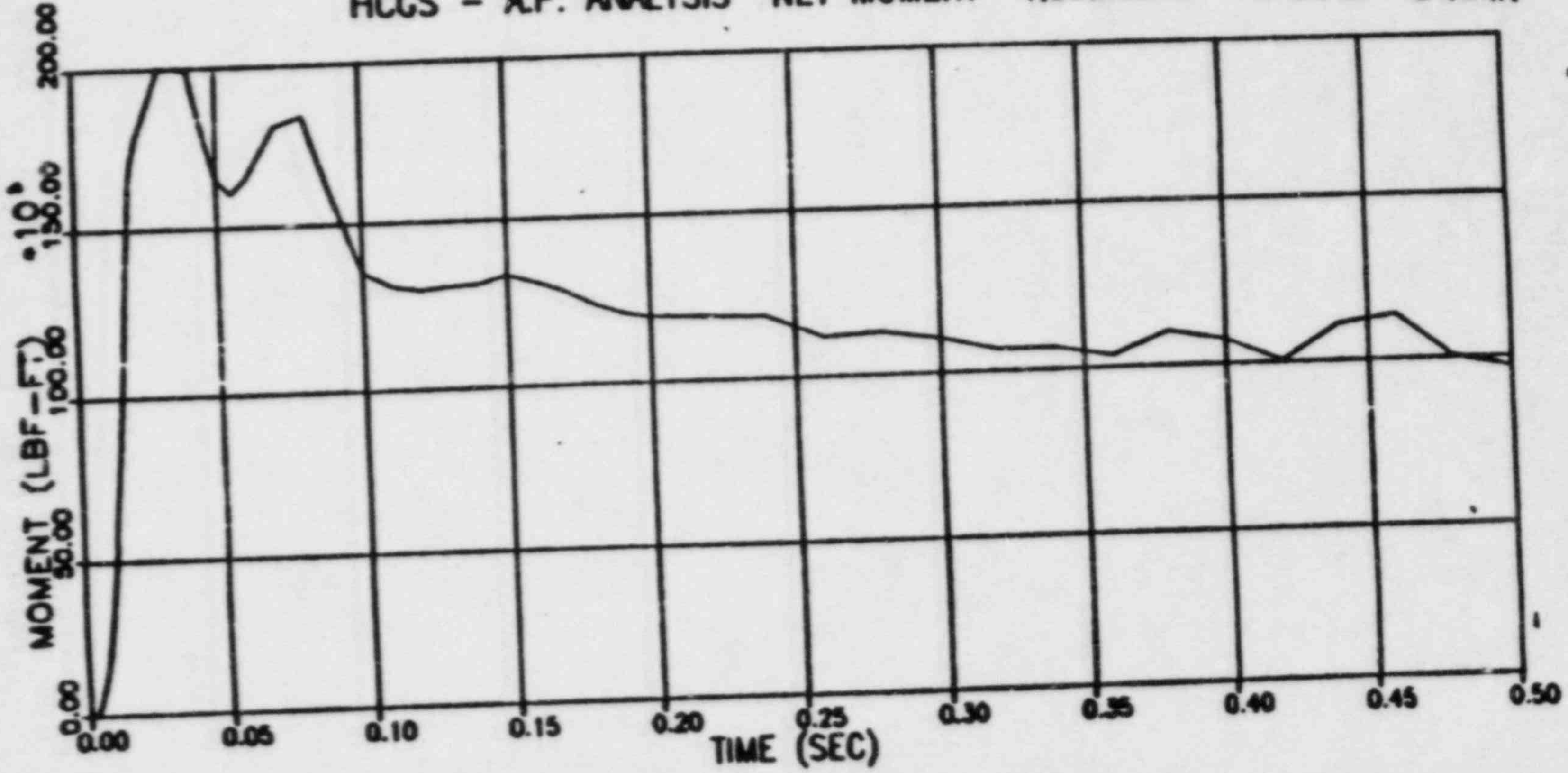


FIGURE 124-7

HCGS - A.P. ANALYSIS NET MOMENT RECIRCULATION LINE BREAK



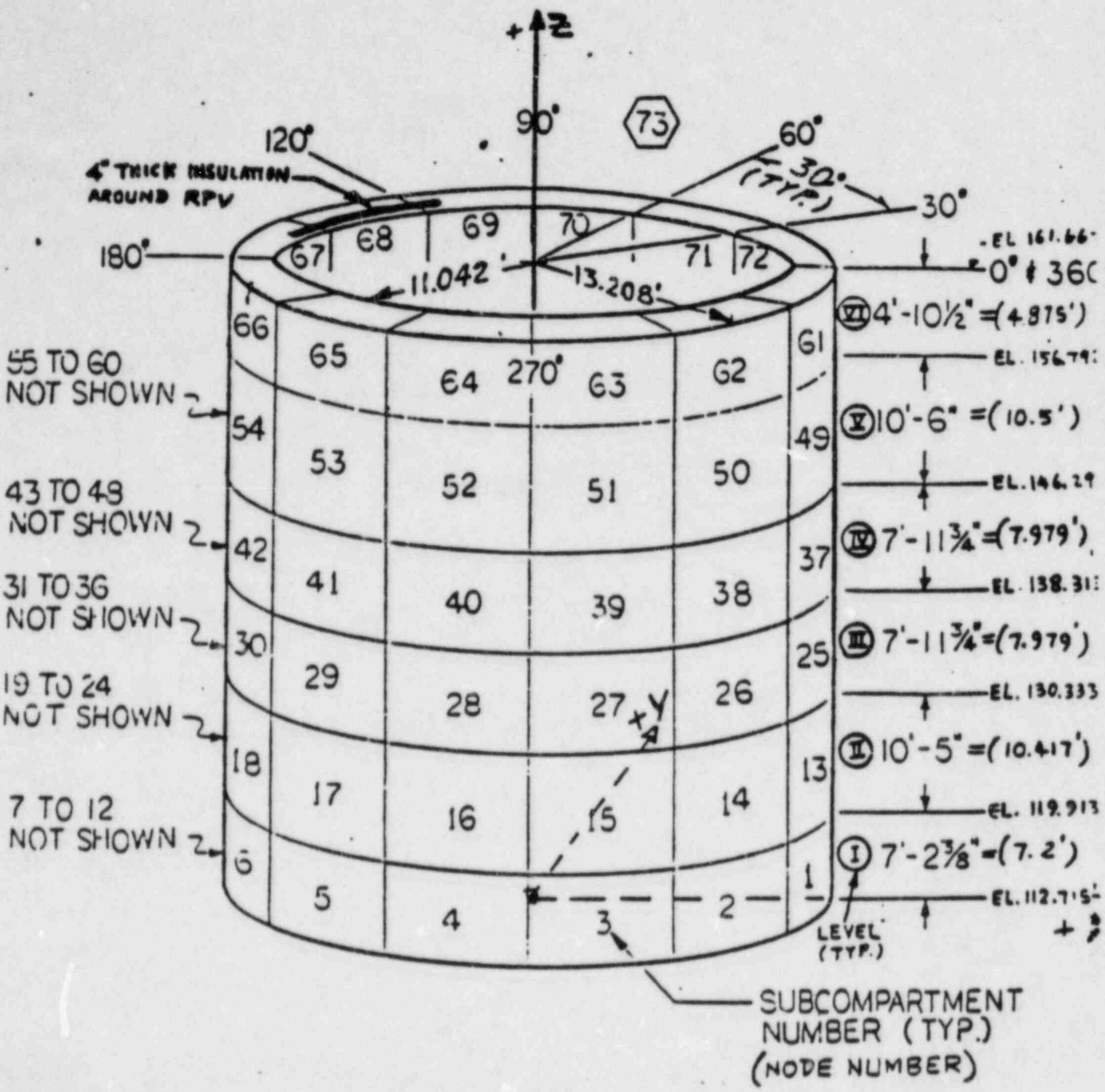


Figure 124-8

DIVISION & DESIGNATION OF SUBCOMPARTMENTS IN ANNULUS FOR PRESSURIZATION ANALYSIS

DSER Open Item No. 126a,b (DSER Section 6.2.1.6)

REDUNDANT POSITION INDICATORS FOR VACUUM BREAKERS (AND CONTROL ROOM ALARMS)

Appendix A to SRP 6.2.1.1.C, "Steam Bypass for Mark I, II, and III Containments" requires leakage tests and periodic surveillance.

A post-operational low pressure test should be performed to detect leakage in the drywell to suppression chamber. For Mark I containments the acceptance criterion is that the measured leakage is not greater than the leakage that could result from a one inch diameter opening. A visual inspection should also be conducted at each refueling outage to detect leak paths.

Appendix A also requires that redundant position indicators should be placed on all vacuum breakers with redundant indicators and an alarm system in the control room. The vacuum breaker position indicator system should be designed to provide the plant operators with continuous surveillance of the vacuum breaker position. 126a The indicators should have adequate sensitivity to detect valve opening which would not result in leakage greater than that from a one inch diameter opening if all valves were at this maximum offset. 126b We will require the applicant to verify this sensitivity. Also, the vacuum breakers should be operability tested at monthly intervals to assure free movement of the valves. 127

To minimize the potential for steam bypass, we will require the applicant to commit to (1) perform operational testing of the torus to drywell vacuum breakers once each month; and (2) perform a leakage test of the drywell to torus vent system at the end of each refueling outage. We will include these periodic tests in the Technical Specification. We will also require the applicant to commit to having alarms in the control room for the redundant vacuum breaker position indicators. We will report on this matter in a supplement to the SER.

RESPONSE

FSAR Section 6.2.5.2.3 has been revised to address redundant position indicators and their sensitivity.

the suppression chamber pressure. The VRVS vacuum valves are fully open when the drywell pressure falls below that of the suppression chamber by 0.25 psid.

The VRVS also limits the pressure differentials between the reactor building and the suppression chamber to less than 3.0 psid through the operation of reactor-building-to-suppression chamber vacuum relief valves. The two 24-inch valves are fully open at 0.25 psid and will vent air into the suppression chamber from the reactor building.

The VRVS operability can be demonstrated by exercising vacuum relief valves to the open position. Each valve is equipped with redundant valve position indicators that indicate and alarm in the main control room. ~~Closure of each valve is to a tolerance of 0.01 inch.~~ The sensitivity of the indicators is sufficient to detect an offset greater than 0.01 inch at the valve centerline. The flow through the eight suppression pool-to-drywell vacuum breakers with each offset at 0.01 inch is equivalent to the flow through a 0.5 inch diameter hole.

insert

6.2.5.2.4 Containment Hydrogen Recombiner System

CHRS is part of the CACS. The CHRS consists of two redundant hydrogen recombiner packages, each of which has adequate processing capacity to control the quantity of the hydrogen and oxygen postulated to be generated in the primary containment after a LOCA. The recombiners are thermal recombination type and are described in Reference 6.2-13.

A schematic diagram of one recombiner package is shown on Figure 6.2-30. Each hydrogen recombiner package consists of three modules: the recombiner skid assembly, the power cabinet, and the control cabinet. The recombiner skid assembly, which is shown on Figure 6.2-31, contains the process components. The process components include flow control valves, canned motor/blower assembly, gas heater pipe, reaction chamber, water-spray cooler, and water separator and associated instrumentation. The gas heater pipe and the reaction chamber are located within an insulated enclosure that also contains electric heater elements. The recombiner skid assembly is located outside the primary containment in the reactor building.

The power cabinet houses the power distribution components for the recombiner package. The cabinet is located near to its associated recombiner skid assembly and contains the 480-V power supply, control transformer, blower motor starter, circuit breakers, control relays, and the silicon-controlled rectifiers (SCRs) that control electrical power to the heater elements.

insert

The redundant position indicators are visually observed and verified at each refueling outage inaccordance with ASME Section XI, Article IWV-3300, to confirm that remote valve indications accurately reflect valve operation. The accuracy verification shall be in accordance with the instructions provided by the valve supplier.

DSER Open Item 127 (Section 6.2.1.6)

OPERABILITY TESTING OF VACUUM BREAKERS

Also, the vacuum breakers should be operability tested at monthly intervals to assure free movement of the valves. To minimize the potential for steam bypass, we will require the applicant to commit to (1) perform operational testing of the torus to drywell vacuum breakers once each month; and (2) perform a leakage test of the drywell to torus vent system at the end of each refueling outage. We will include these periodic tests in the technical specification.

RESPONSE

(1) H.C.O will commit to perform operational testing of the torus to drywell vacuum breakers at a frequency of once per 31 days. This requirement should be included in the HCGS Technical Specifications.

(2) A leakage test of the drywell to torus vent system will be performed at the end of each refueling outage.

DSER Open Item No. 133 (DSER Section 6.2.4.1)

CONTAINMENT PURGE SYSTEM

The Hope Creek Generating Station drywell air purge inlet and vent outlet lines are 26 inches in diameter, while the suppression chamber purge lines are 24 inches in diameter. One 2-inch bypass vent path is provided for the drywell and another is provided for the suppression chamber. The valve arrangements consist of all isolation valves located outside the primary containment for both the drywell and the suppression chamber purge systems. However, until the applicant provides the information outlined in Section 6.2.4, the valve arrangement for these systems cannot be properly described or reviewed.

The applicant has indicated that an evaluation has been made against the branch Technical Position CSB 6-4 - "Containment Purging During Normal Plant Operation". However, we find this evaluation to be incomplete.

The applicant proposed to use the 26-inch purge lines to periodically permit venting of the primary containment during normal plant operation. We will require the applicant to commit to limiting the use of the purge system to less than 90 hours per year while the plant is in Modes 1, 2, and 3 of operation, or provide justification for use of the system beyond 90 hours. The applicant proposes to use the 2-inch bypass vent path on both the drywell and suppression chamber to reduce containment pressure during normal plant operation. The applicant has not indicated how frequently the 2-inch bypass vent paths are to be used. We are awaiting information regarding the above matters from the applicant and will report our findings in a supplement to the SER.

As a result of the numerous reports on unsatisfactory performance of the resilient seats for the isolation valves in containment purge and vent lines (addressed in OIE Circular 77-11, dated September 6, 1977), Generic Issue B-20, "Containment Leakage Due to Seal Deterioration," was established to evaluate the matter and establish an appropriate testing frequency for the isolation valves. Excessive leakage past the resilient seats of isolation valves in purge/vent lines is typically caused by severe environmental conditions and/or to frequent use. Consequently, the leakage test frequency for these valves should be keyed to the occurrence of severe environmental conditions and the use of the valves.

DSER Open Item No. 133 (Cont'd)

RESPONSE

The penetration details in Figure 6.2-28 have been revised to show the proper valve arrangements.

FSAR Section 6.2.4.3.2.1 has been revised to limit the 24-inch and 26-inch containment isolation valves operation in accordance with the Technical Specifications

set

FSAR Section 1.14.1.71.2 has been revised to include the frequency of operation of the 2-inch bypass vent~~s~~. The plant operator will open the 2-inch bypass vent paths if the drywell normal operating pressure ~~exceeds~~ approaches the Technical Specification limit.

Per discussions with the NRC on July 19, 1984, the following information is also being provided.

At HCGS the nitrogen line from the vaporizer connects to the drywell and torus purge lines. The torus purge line is also located above the vent header at HCGS but offset from the centerline of the header by about three feet.

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 is the temperature drops below 40°F. These two control loops are independent of each other. No single failure of a senser, fuse, power supply, etc. could, therefore, lead to a nitrogen temperature below 40°F.

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

Administrative procedures will also be in place to assure that, if the outlet temperature of the vaporizer drops below 40°F, the system will be isolated.

1.14.1.71 Containment Purge System, LRG I/CSB-3

1.14.1.71.1 Issue

Containment purge systems often have small vent lines that are used to bleed off excess primary containment pressure during normal operation. Because the lines provide an open path from the containment to the environs, they must be evaluated against the requirements of Branch Technical Position CSB 6-4.

1.14.1.71.2 Response

The containment inerting and purge system (CIPS) is sized to purge the primary containment during refueling operations after cooldown and cold shutdown. The requirements outlined in BTP CSB 6-4 pertain to the use of CIPS during normal power operation. During normal operation the 24- and 26-inch containment isolation valves will be sealed closed as defined in SRP 6.2.4, Section II.6.f. They will be administratively controlled to assure that they are not inadvertently opened.

To relieve the initial containment pressure buildup caused by the temperature increase during reactor power ascension and to reduce pressure as required during other normal operating transients, the first containment isolation valve from the drywell, may be opened under administrative control to permit the use of the 2-inch vent lines that bypass the second isolation valve.

Insert →

The containment isolation valves and the bypass lines are shown on Figure 6.2-29.

The following is an evaluation of CIPS with respect to the criteria specified in BTP CSB 6-4, when used during normal power operation to bleed off excess primary containment pressure. The evaluation is keyed to the Criteria of BTP CSB 6-4.

1.14.1.71.2.1 Criterion 1.a

The reliability and performance capabilities of the containment isolation valves should be commensurate with the importance to safety of isolating the system penetrating the primary containment boundary.

Insert

The frequency of operation of the 2 inch bypass vent paths used to reduce containment pressure during normal plant operation will depend on operating experience at HCGS. The operator will open the 2 inch bypass vent paths if the drywell normal operating pressure approaches the technical specification limit.

In addition to the containment isolation for the main drywell purge vent line, there is an inlet line to the A train containment hydrogen recombiner that connects to the vent line between the primary containment and the first containment isolation valve. This line is isolated by two motor-operated gate valves. All isolation valves receive a containment isolation signal.

Also connected to the primary containment purge vent line is a 2-inch exhaust line that connects to the vent line between the two main isolation valves. This line is isolated by the isolation valve on the purge line and by a motor-operated globe valve. The valve is normally closed and is maintained closed by a containment isolation signal. For a detailed evaluation of the primary containment venting operation against BTP CSB 6-4 requirements see Section 1.14.1.71.

During normal operation, the 26- and 24-inch containment purge valves are sealed closed except for the inboard valve on the drywell purge outlet vent line (GS-V024). This 26-inch valve can be periodically opened to permit venting of the primary containment to relieve pressure during power ascension from cold shutdown. All the 26-inch and 24-inch containment isolation valves will be under administrative control to assure that they cannot be inadvertently opened. The valve position indicating lights in the main control room will be checked periodically to verify that the sealed closed valves remain closed. The limiting condition for opening the 24 inch and 26 inch containment isolation valves will be in accordance with the technical specifications.

To prevent the unlikely event of a containment purge valve being prevented from closing by debris that could be entrained in the containment purge lines, the drywell purge lines discussed in Section 6.2.4.3.2.14 are provided with debris screens. Debris screens are not provided for the suppression pool purge lines for the following reasons:

a. There are no high energy lines in the suppression pool.

b. There is no insulation or other loose debris in the suppression pool to become entrained in exiting fluid.

The debris screens are designed based on the following criteria:

DSER Open Item No. 135 (DSER Section 6.3.3)

CORE SPRAY AND LPCI INJECTION VALVE INTERLOCKS

The core spray and LPCI (RHR) systems are not designed to withstand normal reactor operating pressure. Each of the low pressure lines that interface with the reactor coolant system has a testable check valve inside primary containment backed up by a normally closed motor-operated gate valve outside containment. Relief valves are provided in the low pressure lines to protect against leakage from the reactor coolant system. An interlcok is provided on the motor-operated valves that prevents their opening until the differential pressure across the valve is below a specified value. We have requested the applicant to verify this valve is interlocked so that it does not open until the reactor coolant pressure is below the system design pressure. We will report on this item in the final SER.

RESPONSE

Illustrated in Figure 135-1 are the present design and a ~~proposed~~ new design for the Hope Creek LPCI injection valve pressure interlocks.

The present design permits the injection valves to open when the differential pressure across the valves is equal to or less than 730 psi. Therefore, the injection valves can open when the reactor pressure is equal to 1080 psig (ie: 730 psi plus the LPCI pump discharge pressure of approximately 350 psi = 1080 psig). The NRC has established the position that this arrangement is inacceptable because a single failure of the inboard testable check valve could result in overpressurization of the LPCI low-pressure piping upstream of the injection valve.

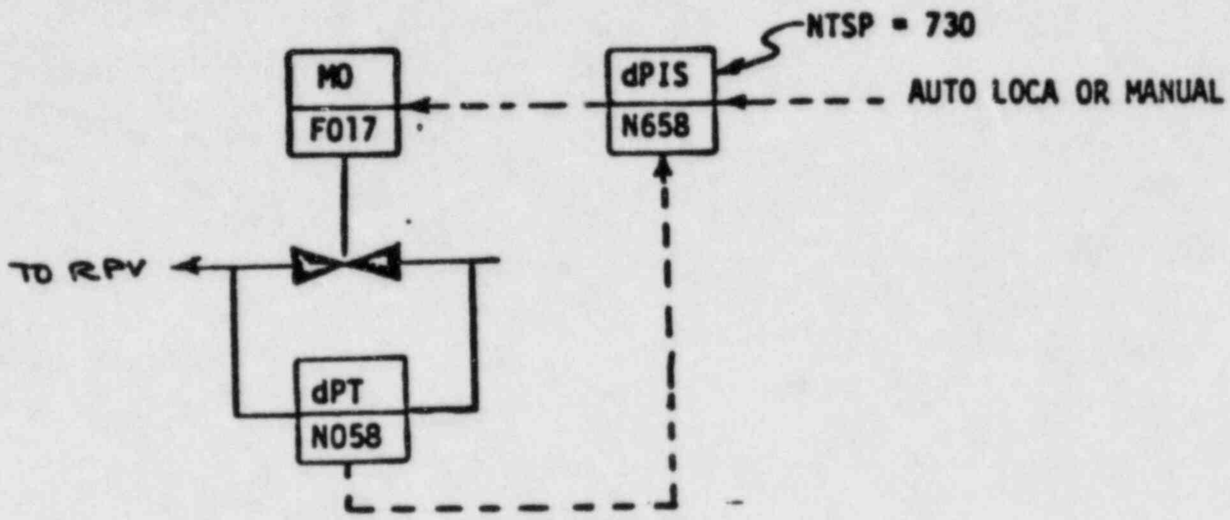
The ~~proposed~~ new design ~~would~~ eliminates this single failure concern by preventing the injection valve from opening when the pressure downstream of the injection valve is greater than the design pressure of the LPCI piping upstream of the injection valve. The pressure indicating switch would have a nominal trip setpoint (NTSP) of 460 psi. Pressure downstream of the injection valve would have to be equal to or less than this NTSP before the automatic or manual open signal would be transmitted to the injection valve. Therefore, the LPCI low-pressure piping that has a design pressure of 500 psi could not be overpressurized by injection valve opening.

Figure 135-1

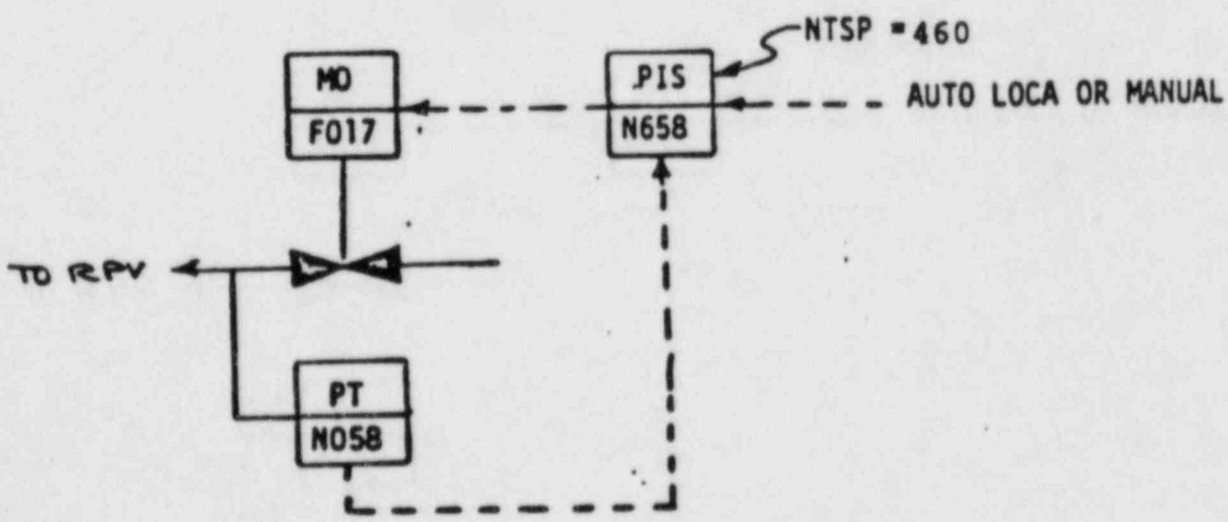
~~PROPOSED~~ HOPE CREEK LPCI

PRESSURE INTERLOCK

PRESENT DESIGN



new ~~PROPOSED~~ DESIGN



Rev 1

DSER OPEN ITEM 136 (Section 6.3.5, 15.9.13)

PLANT SPECIFIC LOCA ANALYSIS

The LOCA analyses reported in the FSAR were for a lead plant representative of Hope Creek. The applicant has committed to supply plant-specific LOCA analyses in a later amendment to the FSAR before fuel loading. The NRC staff will report the results of its review of the plant-specific analyses in a supplement to this report.

The applicant has included small-break LOCA calculations in FSAR Section 6.3.3 that were performed for a lead plant representative of Hope Creek. The applicant has committed to supply plant-specific LOCA analyses in a later amendment before fuel load. The staff will report on its review of the plant-specific analyses in a supplement to this report.

Response

The plant-specific LOCA analysis will be provided in July 1985 and will utilize the evaluation model described in Reference 1 and accepted by the NRC staff in Reference 2. The response to Question 440.27 has been revised to reflect this response.

References

1. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K," NEDE-20566P, November 1975.

2. Letter to G.G. Sherwood (General Elecric) from R.L. Tedesco (NRC), "Acceptance for Referencing of Topical Reports-20566P, NEDO-20566-1 Revision 1, and NEDE-20566-4 Amendment 4," February 4, 1981.

QUESTION 440.27 (SECTION 6.3)

The references provided for the ECCS analysis must include references for the latest model changes and corrections used in the HCGS analysis.

RESPONSE

The HCGS-specific ECCS analysis will be provided in July 1985 and will utilise ~~initiated in the fourth quarter of 1984.~~ The LOCA evaluation models ~~available for the HCGS analysis that are currently~~ approved by the NRC in Reference 1 ~~are~~ and described in Reference 2. ~~It is anticipated that near the end of 1983 the NRC may approve a more realistic LOCA evaluation model which will be consistent with the requirements of 10 CFR 50, Appendix K. This could demonstrate a peak cladding temperature margin in the range of 600°F to 1000°F and will also be available for the HCGS evaluation.~~

~~In August 1984, Public Service Electric and Gas will revise this response to document which evaluation model will be used. If ther more realistic model is selected, the references applicable to it will be included in the revision.~~

REFERENCES

1. Letter to G. G. Sherwood (General Electric) from R. L. Tedesco (NRC), "Acceptance for Referencing of Topical Reports NEDE-20566P, NEDO-20566-1 Revision 1, and NEDE-20566-4 Amendment 4, "February 4, 1981.

2. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K," NEDE-20566P, November 1975.

DSER Open Item No. 137 (DSER Section 6.4)

CONTROL ROOM HABITABILITY

The staff has evaluated the control room doses following a postulated LOCA in accordance with SRP Section 6.4. Page 6.4-8 of the Hope Creek FSAR states that the design-basis LOCA is the worst-case accident scenario for control room habitability. The basis for this conclusion has not been provided in the FSAR. Although the amount of radiation released may be lower following a design-basis accident (DBA) other than a LOCA, the control room dose could be higher because of unfavorable building-release-receptor relationships resulting in less atmospheric dispersion (higher X/Q values). In order for the staff to complete its control room dose evaluation, the following information is needed from the applicant: radiation release locations for each DBA including main steam line failure outside containment, rod drop accident, failure of small lines carrying primary coolant outside containment, LOCA-containment leakage, LOCA-ECCS leakage outside containment, LOCA MSIV leakage, and a fuel handling accident; and control room X/Q for each release location. Until this information is evaluated, these will be open items.

With regard to the applicant taking exception to the X/Q determination method recommended in SRP Section 6.4 and the 1974 Murphy-Campe paper, the staff cannot endorse the method proposed by the applicant without wind tunnel tests performed for the specific building-release-receptor relationships of the Hope Creek facility (open item). This is because of the uncertainties involved in estimating atmospheric dilution between sources and receptors in close proximity to major structures.

With respect to toxic gas protection, the staff's evaluation in accordance with SRP Section 6.4, RGs 1.78 and 1.95 indicated that there is no danger to control room personnel from toxic chemicals, including chlorine, stored onsite or offsite, or transported nearby (See Section 2.2.3).

Based upon the foregoing, the applicant has not demonstrated that the control room habitability systems will adequately protect the control room operators in accordance with the requirements of GDC 19 and, therefore, compliance with NUREG-0737, Item III.D.3.4 cannot be established.

RESPONSE

The additional information requested by the staff for design basis accidents other than the LOCA and the basis for our conclusion that the LOCA is the worst case accident scenario has been provided in the attached revision to Section 6.4.

DSER Open Item No. 137 (Cont'd)

As a result of the meeting with NRC on April 10, 1984, it was agreed that the NRC will independently calculate control room operator doses for HCGS. If these doses are within GDC-19 limits, no further justification of our X/Q determination method by way of wind tunnel tests will be required.

system consists of two 100%-capacity trains, each supplied by a separate Class 1E power system and interlocked with one of the CRS/CRRA systems. Each CREF train consists of an outside air connection to the CRS outside air intake plenum, radiation sensor, tornado damper, smoke detector, outside and return air dampers, fan, 80 to 85% ASHRAE dust spot efficiency filters, electric heating coil for humidity control, upstream high efficiency particulate air (HEPA) filter, charcoal adsorber, a downstream HEPA filter, and a discharge damper. The CREF system may operate in one of the two following modes:

a. A pressurizing mode in which 1000 cfm of outside air is mixed with 3000 cfm of control room return air before entering the CREF unit, thus pressurizing the control room envelope above the surrounding space This mode is an automatic mode following a detection of high airborne radioactivity in the control room normal air intake.

b. The operator can override the control room pressurization mode to initiate isolation mode by manually closing the outside air intake isolation damper for the operating CREF unit. However, this mode is not used following a radiological accident. The recirculation (isolation) mode is circulating 4000 cfm of return air, without introduction of outside air, through a CREF unit.

6.4.4 DESIGN EVALUATIONS

The control room habitability system is designed with redundancy and separation of active components to provide reliable operation under normal conditions and to ensure operation under accident conditions. The design basis accident (DBA) radiation source terms used for control room dose evaluation are in accordance with NUREG-0737, Item III.D.3.4., Control Room Habitability Requirements.

6.4.4.1 Radiological Protection

A detailed discussion of the dose calculation model for control room operators following the postulated DBA is provided in Section 6.4.7. ~~The worst-case accident scenario for control room habitability design purposes is the DBA loss of coolant accident (LOCA), as indicated in Section 15.6.5.~~ The resulting calculated doses for control room occupancy on a rotating shift basis are shown in Table 6.4-4 and are less than 5 rem to the whole body or

Insert

Insert

The design basis accidents have been evaluated to determine the worst case accident scenario for control room habitability design purposes. The control room operator doses can be derived for each of the accidents by the methodology described in Section 6.4.7 with the radiation source terms defined in the appropriate sections in chapter 15 and the release locations relative to the control room intake shown in Figure 6.4-2 and Table 6.4-5.

The release location for the DBA LOCA, fuel handling accident, main steam line break accident and instrument line break accident which occur inside primary containment and the reactor building is the FRVS exhaust vent at the top the reactor building. The release location for the control rod drop accident, main steam line break accident and off gas system failure which occur in the turbine and radwaste buildings is the south plant vent. The offgas system failure could also result in releases from the north plant vent depending upon the actual accident location in the building. The release location for the main steam line break accident which occurs in the main steam tunnel is the blowout panels located between the reactor building and the South plant vent.

The release location for a HPCI steam supply line break accident, which occurs in the reactor building at el. 63'-0", is the reactor building blowout panels located in the west wall of the reactor building. The radiation source term for this accident can be conservatively assumed equivalent to the main steam line break for purposes of this evaluation. The expected mass release is approximately 50% of the main steam line break which would result in a much lower source term.

Of all the accident releasing from the FRVS exhaust vent, the highest source term and longest duration results from the DBA LOCA. Releases from the south plant vent occur from accidents which have lower source terms and a smaller atmospheric dispersion factor with respect to control room intake LOCA. Further, although the main steam line break and HPCI line break have higher source terms, the duration of the accident is shorter in comparison to the DBA LOCA therefore, the consequences are less. Consequently, the DBA LOCA has been determined to result in the controlling accident conditions and has been designated the worst case accident scenario for control room habitability design purposes.

TABLE 6.4-2

DBA LOCA ACCIDENT ATMOSPHERIC DILUTION FACTORS -- X/Q
(FRVS EXHAUST)

Time Period (h)	Wind Speed Factor	Wind Direction Factor	X/Q (s/m^3)
0 to 8	1	1	4.39×10^{-5}
8 to 24	0.67	0.88	2.59×10^{-5}
24 to 96	0.5	0.75	1.65×10^{-5}
96 to 720	0.33	0.5	7.24×10^{-6}

TABLE 6.4-4

DBA LOCA CONTROL ROOM DOSE

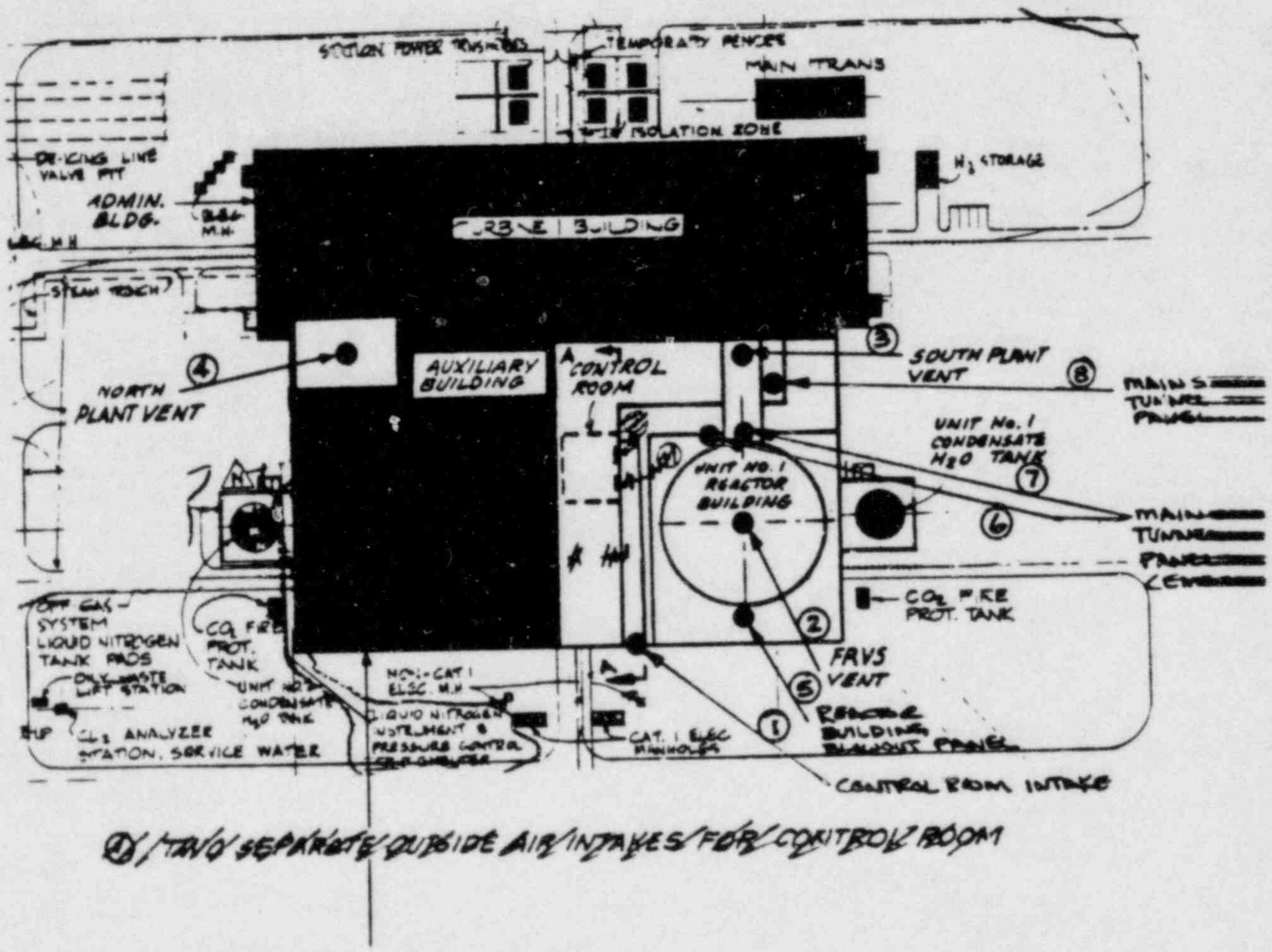
	Control Room Dose, rem
Thyroid	.026
Whole-body	.013
Skin	.356

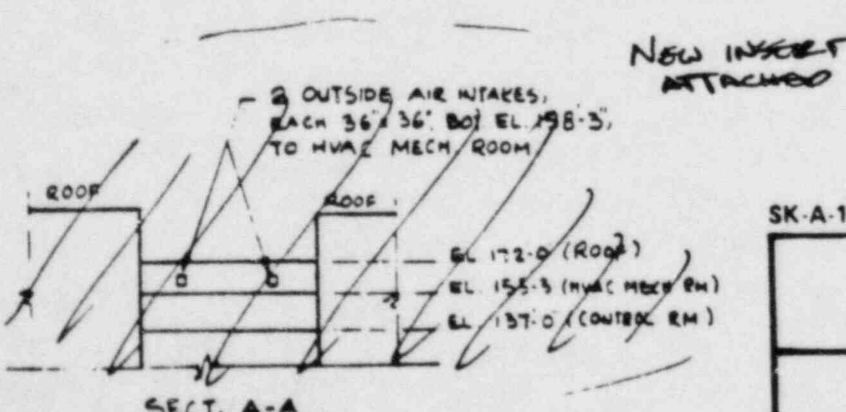
HCGS FSAR

TABLE 6.4-5

RADIOACTIVE RELEASE LOCATIONS
RELATIVE TO CONTROL ROOM INTAKE

RECEPTOR/ RELEASE ~~LOCATION~~ POINT	ELEVATION OF ℄ (ft)	DISTANCE FROM CONTROL ROOM INTAKE TO RELEASE ℄ (ft) X (NORTH/SOUTH)	Y (EAST/WEST)
① CONTROL ROOM INTAKE	156'-9"	0'	0'
② FRVS EXHAUST VENT	301'-4"	111'-2"(S)	129'-0"(E)
③ SOUTH PLANT VENT	217'-0"	111'-2"(S)	303'-3"(E)
④ NORTH PLANT VENT	217'-0"	266'-1"(N)	306' 3"(E)
⑤ REACTOR BUILDING BLOWOUT PANEL	143'-9"	111'-2"(S)	45'-0"(E)
⑥ MS TUNNEL NORTH BLOWOUT PANEL (EMERGENCY)	161'-11"	94'-2"(S)	226'-2"(E)
⑦ MS TUNNEL SOUTH BLOWOUT PANEL (EMERGENCY)	162'-0"	137'-5"(S)	224'-11"(E)
⑧ MS TUNNEL BLOWOUT PANEL	143'-6"	151'-2"(S)	274'-6"(E)





SK-A-1006, REV N

HOPE CREEK GENERATING STATION
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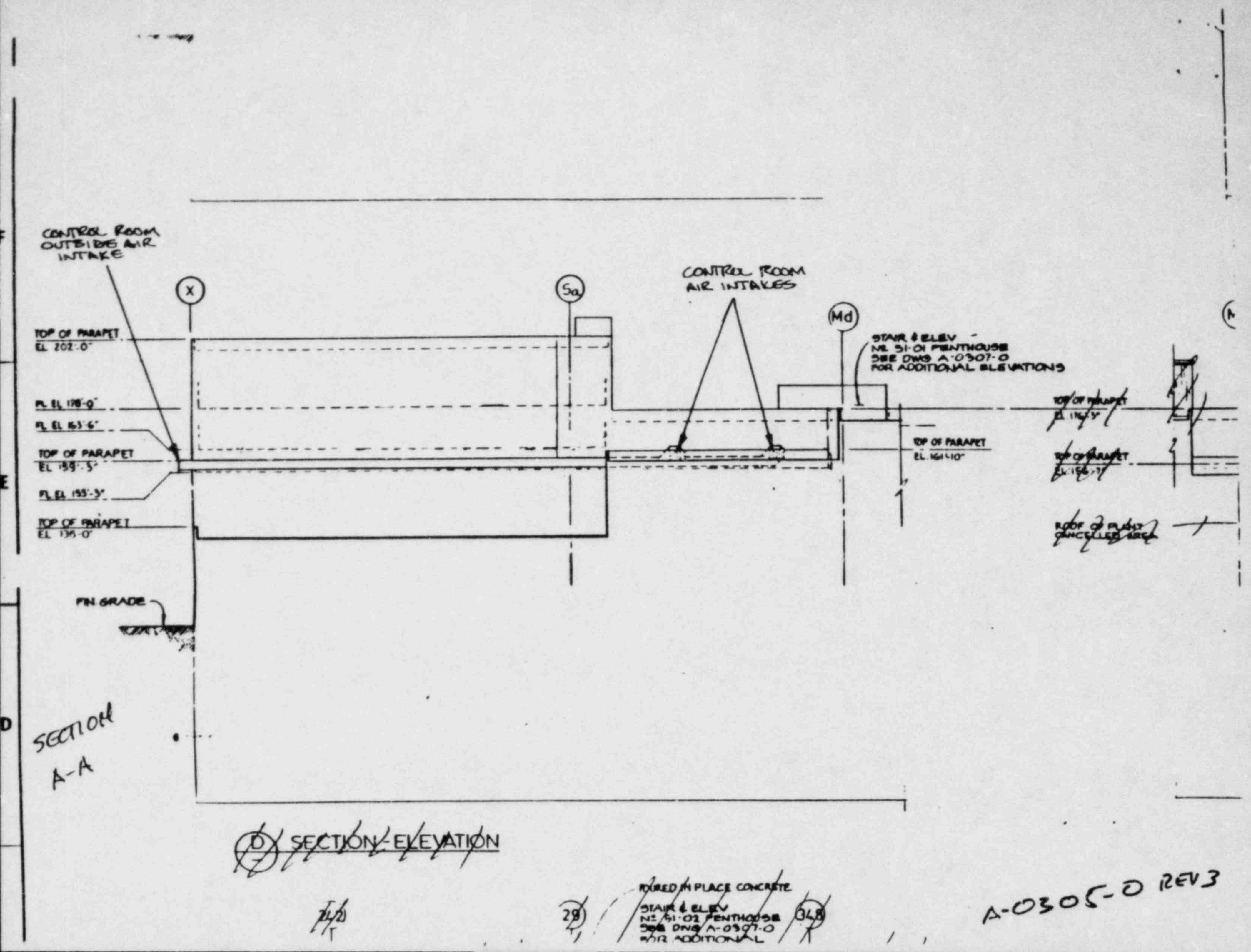
PLANT LAYOUT WITH RESPECT TO CONTROL ROOM INTAKE

FIGURE 6.4-2



CONTROL ROOM OUTSIDE AIR INTAKE
X
Sa
CONTROL ROOM AIR INTAKES
Md
STAIR & ELEV - Nº 51-01 PENTHOUSE SEE DWG A-0307-0 FOR ADDITIONAL ELEVATIONS
TOP OF PARAPET EL 202'-0"
PL EL 178'-0"
PL EL 165'-6"
TOP OF PARAPET EL 159'-5"
PL EL 155'-3"
TOP OF PARAPET EL 135'-0"
FIN GRADE
TOP OF PARAPET EL 161'-10"
TOP OF PARAPET EL 176'-5"
TOP OF PARAPET EL 156'-7"
ROOF OF PLANT CANCELLED AREA
SECTION A-A
D SECTION-ELEVATION
POURED IN PLACE CONCRETE
STAIR & ELEV Nº 51-02 PENTHOUSE SEE DWG A-0307-0 FOR ADDITIONAL
29
34B
A-0305-0 REV3

HCGS

DSER Open Items No. 148 (DSER Section 9.3.2)

Postaccident Sampling System, TMI-2 Action Plan Item II.B.3

The information provided through Amendment 3 was not sufficient for the staff to complete its evaluation. This is an open item.

To meet the criteria of NUREG-0737, Item II.B.3, the guidelines of Appendix C to this SER should be implemented.

RESPONSE

For the information requested above, see the response to question 281.15.

In addition, the following information was requested in discussions with the NRC on July 13, 1984:

- The only inaccessible values are located in the Reactor Building and are discussed in FSAR Section 9.3.2.2.2.2.
- Heat tracing of the sample line is discussed in Section 9.3.2.2.2.2.

QUESTION 281.15 (SECTION 9.3.2)

The information provided on the Post Accident Sampling System (PASS) is inadequate to demonstrate compliance with NUREG-0737, Item II.B.3. Provide information that satisfies the criteria in the attachment.

RESPONSE

Section 9.3.2 has been revised to provide the information responding to the attachment transmitted with this question.

~~Additional information on the following will be provided in June 1984.~~

- ~~Equipment used to ship samples for offsite analyses~~
- ~~Time to analyze samples~~
- ~~Quantification methods~~
- ~~Chloride analysis~~
- ~~Compliance with GDC19 for PASS sample analysis~~

delete

In addition, HCGS will meet the requirements of GDC 19 and a discussion of HCGS compliance with GDC 19 will be provided by September, 1984.

1.8.1.97 Conformance to Regulatory Guide 1.97, Revision 2, December 1980: Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Envircns Conditions During and Following an Accident

HCGS complies with the BWR Owner's Group position (Reference 1.8-4) on Regulatory Guide 1.97 with the following clarifications and exceptions:

a. Suppression chamber spray flow (Type D variable) - The BWR Owner's Group has recommended not implementing this variable. HCGS has implemented this variable as Category 2.

b. Drywell spray flow (Type D variable) - The BWR Owner's Group has recommended not implementing this variable. HCGS has implemented this variable as Category 2.

c. Condenser cooling water flow (BWR Owner's Group recommended Type D variable) - HCGS deviates from the BWR Owner's Group position on this variable by using the cooling water temperature rise (delta T) across the condenser to provide this information.

Insert E →

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

1.8.1.98 Conformance to Regulatory Guide 1.98, Revision 0, March 1976: Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor

HCGS complies with Branch Technical Position ETSB 11-5, Revision 0, July 1981, in lieu of Regulatory Guide 1.18.

For further discussion, see Section 15.7.1.

Insert E

d. Total Dissolved Gas Analysis - The guidelines recommended by the BWR owners Group and GE shall be followed. This was agreed to in a meeting between NRC Management (R. Vollmar, et. al.) and GE (J. Quirk, et al.), dated December 12, 1983.

The post-accident sampling system (PASS) is designed to provide specific samples in the event of a loss-of-coolant accident (LOCA) in compliance with ~~Regulatory Guide 1.97~~ the requirements for accident sampling capability in Item II.B.3 of NUREG-0737.

Radiation monitoring of gaseous and liquid process streams is discussed separately in Section 11.5.

9.3.2.1 Design Basis

9.3.2.1.1 Process Sampling System

The PSS is designed to provide representative samples of all process streams related to plant power operation and liquid radwaste processing.

The system is designed to allow for the collection of data or a grab sample without hazard to the operator or contamination of general working areas.

Sample line size, length, and routing are designed to provide a representative sample by maintaining turbulent flow.

The PSS is designed to ensure representative sampler from liquid and gaseous processes in accordance with Regulatory Guide 1.21, Position C.6.

Isolation valves fail in the closed position, in accordance with the requirements of GDC 60 in 10 CFR 50, Appendix A, to control the release of radioactive materials to the environment. Isolation valves are provided to limit reactor coolant loss from a rupture of the sample line in accordance with ALARA provisions in 10 CFR 20.1(c) and GDC 60 in 10 CFR 50, Appendix A, to control the release of materials to the environment.

9.3.2.1.2 Post-Accident Sampling System

The PASS is designed to meet the requirements of Item II.B.3 of NUREG 0737. ~~and Regulatory Guide 1.97, Revision 2.~~

A gaseous radwaste storage tank is not part of the HCGS design. Offgas treatment system radioactivity is monitored downstream of the offgas system charcoal adsorbers, upstream of the offgas system discharge valve. This monitor does not provide for sample removal. It is described in Section 11.5.2.2.6.

Sample flow rates to the analyzer and grab sample panels are designed to provide turbulent flow and to supply a representative sample. The liquid sample stations have flush and blowdown capabilities built into the system to reduce radiation exposure of the operator to as low as reasonably achievable (ALARA). The various sample points and design parameters provided to meet the acceptance criteria are listed in Table 9.3-3.

9.3.2.2.2 Post-Accident Sampling System

The post-accident sampling system~~s~~ (PASS) ~~are~~ is designed to obtain representative liquid and gas grab samples from the primary coolant system and from within the primary and secondary containments for radiological and chemical analysis under accident conditions. The grab samples are subsequently transported to the laboratory for chemical and radiosotopic analysis or shipped offsite for analysis.

The system design minimizes operating complexities and "in-line" instrumentation, is modular for maintenance and contamination control purposes, and is compact in size to reduce the amount of shielding required. The system can be used to provide samples under all plant conditions, ranging from normal shutdown and power operation to post-accident conditions.

Figures 9.3-5 and 9.3-6 show the piping and instrumentation diagrams and the logic diagrams respectively for the PASS. The equipment includes isolation and control valves, piping racks, shielded sample stations (gas and liquid), liquid chillers, and control panels for the sampling stations and the isolation valves. The seismic category, quality group classification, and corresponding codes and standards that apply to the design of the PASS are as shown on Table 3.2-1. Demineralized water, nitrogen gas ~~and tracer gas~~ are provided as support systems for the PASS.
and

plate. As reactor pressure decays, low pressure coolant injection (LPCI) is initiated into the core region. This water volume supplies more coolant than is boiled off by the decay heat. X This excess water will flow down past the core, up through the jet pumps, and out through the postulated break, assuring a representative sample at the sample point.

To ensure a representative liquid sample from the jet pumps at low (<1%) power conditions for small break or non-break events, the reactor water level will be raised to the level of the moisture separator when this action is not inconsistent with station emergency procedures. This will fully flood the separators and will provide a thermally-induced recirculation flow path for mixing.

Samples will be taken from the reactor via the jet pump pressure instrument lines as long as possible. This allows a more direct and therefore faster response to core conditions. Upon decay or loss of reactor pressure, the jet pump sample point is lost, and the RHR loops sample points must be employed for sampling. Reactor coolant and/or suppression pool samples may be taken from the RHR sample lines, depending on the mode of RHR operation. These modes are:

1. LPCI: Suppression pool water is injected into the core, flows up through the jet pumps, and back to the suppression pool via the postulated break. The system will be operated for an estimated 30 minutes minimum prior to sampling of the suppression pool water to ensure that a representative sample is obtained at the sample taps.

2. Shutdown Cooling: The RHR system, aligned in the shutdown cooling mode, provides cooling and circulation of reactor coolant through the core, resulting in a representative sample at the RHR sample taps.

3. Suppression Pool Cooling: The RHR system, aligned in the suppression pool cooling mode, provides cooling and circulation of the suppression pool

water. The system will be operated for an estimated 30 minutes minimum prior to sampling of the suppression pool water to ensure that a representative sample is obtained at the RHR sample taps.

These sample lines tap off upstream of the first isolation valve in the RHR system sample lines at the discharge of each RHR heat exchanger.

9.3.2.2.2.2 Isolation Valves and Sample Lines

Containment isolation for the drywell/suppression chamber gas sample lines, the jet pump instrument liquid sample line, and the gas/liquid sample return lines is provided by the isolation valves noted in Section 9.3.2.2.2.1. System isolation for the RHR liquid sample lines is provided by the isolation valves discussed in Section 9.3.2.2.2.1. All PASS isolation valves in the reactor building are environmentally qualified for the conditions in which they must operate.

The gas sample lines are heat traced to prevent precipitation of moisture and the resultant loss of iodine in the sample lines. Sample line routings are as direct and short as practical. Recirculation flow rates in the liquid sample lines are maintained in the turbulent flow regime.

The liquid sample lines have top or side takeoff taps to minimize the possibility of line plugging.

Primary containment gas/liquid sample lines and secondary containment gas sample lines are designed Seismic Category I up to and including each lines' piping-to-tubing reducer which is located immediately downstream of the restriction orifice. All sample lines beyond the piping-to-tubing reducers conform to quality group D, meet the requirements of ANSI B31.1, Power Piping Code, and are non-Seismic Category I. All isolation valves are located in the Seismic Category I portion of the sample lines.

PASS control instrumentation is installed in two control panels mounted side by side. One of these panels contains the conductivity and radiation level readouts. The other control panel contains the flow, pressure, and temperature indicators, and various valve controls and switches. A graphic display is provided directly on this control panel which shows the status of the pumps and valves. These panels are located directly outside the PASS room to minimize operator exposure while operating the PASS.

The PASS isolation valve control panels are located adjacent to the PASS control panel outside the PASS room. Once the MCR permissive keylock switch is activated, the isolation valves can be operated from the these panels. Valve status indication is provided on the control panels; 100% closed valve status signals are provided to the computer. The valves close if the MCR permissive is removed.

9.3.2.2.2.5 Gas Sampler

The gas sample system is designed to operate at pressures ranging from sub-atmospheric to the design pressures to the primary containment one hour after a LOCA. The gas samples may be passed through a particulate filter and silver zeolite cartridge for determination of particulate activity and total iodine activity by subsequent spectroscopic analysis. A radiation monitor is mounted close to the filter tray to measure the activity buildup on the cartridges. Alternatively, the sample flow bypasses the iodine sampler, is chilled to remove moisture, and a
15 ~~10~~ milliliter grab sample can be taken for determination of gaseous activity and gas composition by gas chromatography. The gas is collected in an evacuated vial using hypodermic needles. When purging the drywell and suppression chamber gas sample lines to obtain a representative sample, the flow is returned to the suppression chamber. During purging of the secondary containment line and when flushing the sample panel lines with nitrogen, flow is returned to secondary containment. The sample station design allows for sample gas or nitrogen flushing of the entire sample panel line downstream of the four-position selector valve. This capability will minimize cross-contamination between the various samples.

9.3.2.2.2.6 Liquid Sampler

The liquid sample system is designed to operate at pressures from -0 to 1150 psi. The design recirculation flow rate of 1 gpm is sufficient to maintain turbulent flow in the sample line and serves to minimize cross-contamination between samples. The recirculation flow is returned to the suppression pool. The liquid sampling system is designed to allow demineralized water flushing of the system lines from a point in the piping station through the sampling needles.

9.3.2.2.2.6.1 Diluted Liquid Sample

The small volume ~~All~~ liquid samples are taken into 15 milliliter septum bottles mounted on sampling needles. In the sampling lineup, the sample flows through a conductivity cell (0.1 to 1000 micromhos/cm) and through a ball valve bored to 0.10 milliliter volume. After flow through the sample is established, the ball valve is rotated 90 degrees, and a syringe is used to flush the sample and a measured volume of diluent (generally 10 milliliters) through the valve and into the sample bottle. This provides an initial dilution of up to 100:1. The sample bottle is contained in a shielded cask and remotely positioned on the sample needles through an opening in the bottom of the sample enclosure.

9.3.2.2.2.6.2 Non-Diluted Liquid and Dissolved Gas Samples

Alternatively, the sample can be diverted through a 70 milliliter holdup cylinder to obtain depressurized samples of primary coolant gas and liquid phases. A coolant sample is circulated through a holdup cylinder, the cylinder is then isolated and contents circulated through a ~~gas~~ loop. ~~containing a measured amount of inert krypton.~~ The gases are vented to an evacuated gas collection chamber, and a fraction of the gas is expanded into a sample vial for analysis of individual species by gas chromatography. ~~The concentration of krypton in the sample is used to calculate the fraction of the dissolved gases recovered. The krypton also serves as a stripping agent at low gas concentrations.~~ insert A Ten milliliter aliquots of degassed liquid can then be taken for offsite (or onsite depending on activity level) analyses which require a relatively large undiluted sample. This sample is obtained remotely using the large volume cask and cask positioner through needles on the underside of the sample station enclosure.

9.3.2.2.2.7 Piping and Sample Station Ventilation

The sample station enclosure will be vented into the piping station area. The ventilation rate required for heat removal and proper sweep velocity during operation is about 40 scfm. A

Insert A

The concentration of total dissolved gas is determined by measuring the pressure rise in the gas collection chamber following gas expansion and applying the ideal gas law.

pressure gauge is attached to the sample station enclosure to monitor the pressure differential between the enclosure and the PASS room. The pressure differential will assure the operator that airborne activity in the sample enclosure will be swept into the piping area.

The piping area is vented into the auxiliary building radwaste area exhaust system discussed in Section 9.4.3.2.2. The nominal exhaust rate is 200 scfm.

Any potential liquid leakage in the piping station area will be collected and processed in accordance with Section 9.3.3 and 11.2.2 respectively.

9.3.2.2.2.8 Sample Station Sump

The sample station is provided with a bottom sump to collect liquid leakage. This sump can be isolated, pressurized, and discharged into the sample station liquid return line to the suppression pool.

9.3.2.2.2.9 Sample Handling Tools and Transport Containers

Appropriate sample handling tools and transporting casks are used. Gas vials are installed and removed by use of a vial positioner through the front of the gas sampler. The vial is manually lowered into a shielded cask directly from the positioning tool. This allows the operator to maintain a distance of about three feet from the unshielded vial. The cask provides about 1-1/8 inches of lead shielding. A 1/8-inch diameter hole is drilled in the cask so that an aliquot can be withdrawn from the vial with a gas syringe without exposing the analyst to the unshielded vial.

The particulate and iodine cartridges are removed via a drawer arrangement. The quantity of activity accumulated on the cartridge is limited by controlling the line flow using a flow orifice and by timing the sample duration either manually or by use of preset timer. In addition, the radioactivity level is monitored during sampling using a radiation probe installed adjacent to the cartridge. These samples will be limited to activity levels that will not require shielded sample carriers.

concentrations are monitored by one of two hydrogen/oxygen analyzers. The Combustible Gas Analyzers are discussed in Section 6.2.5.2.5.

3. Dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids.

Total Dissolved Gas analysis will be performed by the method recommended by the BWR Owners Group and GE (as discussed in FSAR Section 1.8.1.97).

Chloride analysis will be performed by Ion Chromatography, Boron by Specific Ion Electrode.

4. Alternatively, have inline monitoring capabilities to perform all or part of the above analyses.

Inline monitoring capabilities (radiation monitors and conductivity cell) are discussed in Section 9.3.2.5.2.

HCGS will have the capability of sending samples offsite. ~~Arrangements will be made with offsite facilities to perform analyses and an appropriate shipping cask will be obtained prior to core load.~~

Insert C

c. Reactor coolant and containment atmosphere sampling during postaccident conditions shall not require an isolated auxiliary system (e.g., the letdown system, reactor water cleanup system) to be placed in operation in order to use the sampling system.

Isolated auxiliary systems are not required for PASS operation. The PASS is described in Section 9.3.2.2.2.

d. Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

The method of gathering pressurized and non-pressurized reqactor coolant samples is discussed in Section 9.3.2.2.2.

e. The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above

Insert C

Arrangements will be made with offsite facilities to perform analyses and a licensed shipping cask will be obtained (such as recommended by the BWR Owners group) prior to core load.

conditions the licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

A chloride analysis will need to be performed within 4 days of the sample being taken because 1) the plant has brackish coolant water and 2) two barriers are provided between primary containment systems and the cooling water (see Figure 9.2-3).

f. The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). (Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979 letter from H.R. Denton to all licensees).)

The PASS radiation shielding design will be in accordance with Section 12.3.2.2.6 to keep personnel exposures as low as practicable and within the limits established by GDC 19.

g. The analysis of primary coolant samples for boron is required for PWRs. (Note that Revision 2 of Regulatory Guide 1.97 specifies the need for primary coolant boron analysis capability at BWR plants.)

HCGS will develop a procedure for Boron analysis ~~(Procedure No. CH-CA-22-25)~~ prior to core load.

h. If inline monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least

Sample aliquots are taken from the septum bottles for analysis or further dilution. Aliquoting and transfer will be performed using shielded containers, or behind a lead brick pile. Calibrated hypodermic syringes will be used for aliquoting the higher activity samples. Tongs or other holding/clasping devices will be available for holding the sample bottle during the transfer and dilutions to reduce hand and body exposure. Unless prohibited by the intended analysis, dilutions will be done using very dilute (about 0.01N) nitric acid as the diluent to minimize sample plateout problems.

Primary coolant samples obtained from the sampling station are diluted by a factor of 100 (0.1 ml coolant diluted to 10 ml). Under severe accident conditions, a calibrated syringe would be used to obtain an aliquot for this sample for further dilutions. At the maximum expected primary coolant activity level (3 Ci/cc), a dilution factor of 1 X 10^5 would be required for gamma spectroscopy.

Insert A1 →

~~Direct counting of the initial 100:1 dilution sample would allow analysis at coolant activity levels down to 1 Ci/cc. In addition, the degassed, undiluted 10 ml sample available from the sample station could be used for analysis of samples in the 10^{-2} to 10^{-3} Ci/cc range.~~ Thus, useful samples may be obtained from the post-accident sampling station for coolant activity levels ranging from design basis accident source terms to well below the maximum level that can be tolerated at the normal reactor sample station.

j. Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

Insert B1 →

1. Offsite provisions for chloride analysis will be accurate ±10 percent over the range 0.5 to 20 ppm and ±0.05 ppm below concentrations of 0.5 ppm.

2. Onsite chloride will be determined by Ion Chromatography. Table 9.3-7 lists range and accuracy for onsite analysis. No radiation damage is anticipated with resins based on experience developed at Battelle. Resins are conventionally

Insert A1

Gross activity measuremements are accurate within a factor of 2.

The onsite radiological and chemical laboratory facilities are equipped with gamma spectral analysis equipment to quantify the radionuclides present in gas and liquid samples. Shielding is provided for the radiation detectors to minimize the effect of background radiation. Initial dilutions are performed in the process of taking liquid samples at the sample stations. Any additional dilutions required will be performed in the laboratory fume hood behind a lead brick pile.

If the levels of noble gases in the ambient atmosphere surrounding the detector are high enough to cause significant interference or to overload the detector, a compressed air or nitrogen purge of the detector shield volume will be maintained.

Insert B1

The analytical methods selected by HCGS were based on research done by NUS, Exxon Nuclear, General Electric, and EPRI using the NRC Standard Test Matrix.

on the reasonable assumption that chloride level in the primary coolant will generally be below 10 ppm. Sensitivity will be in accordance with EPRI NP-3513.

3. A combination electrode will be used to measure the pH of coolant samples. Testing performed by GE has verified that expected levels of irradiation result in a shift of less than 0.3 pH units.

4. The boron determination is made on a 1:100 dilution of reactor water, the 5 ml sample radiation level is on the order of 30 R/hr at 1 cm two hours after the accident. The total dose to the fluoroborate electrode during the analyses sequence will be on the order of tens to hundreds of rads. The level of exposure is not anticipated to have any significant effect on the accuracy of measurement or operating lifetime of the probe.

5. The post-accident sample station is equipped with a 0.1 uS conductivity cell. The conductivity meter has a linear scale with a six-position range of 0-3, 0-10, 0-30, 0-100, 0-300 and 0-1000 uS when using the 0.1 uS cell. This conductivity measurement system will be used to determine the primary coolant or suppression pool conductivity. During normal operation the BWR technical specifications require maintaining the primary coolant below 1.0 uS/cm, and conductivity measurements are the primary method of coolant chemical control.

   Conductivity measurements are, of course, non-specific, but they serve the important function of indicating changes in chemical concentrations and conditions. Perhaps even more important, in the case of the BWR primary coolant, the conductivity measurements can establish upper limits of possible chemical concentrations and can eliminate the need for additional analyses.

   The conductivity measurement can also be used to bound the possible range of pH values.

9.3.2.6 SRP Rule Review

In SRP Section 9.3.2, Revision 2, Acceptance Criterion II.5.a implies that the PASS should have the capability for verifying dissolved oxygen concentration in the reactor coolant. The PASS was designed prior to the issuance of SRP Section 9.3.2, Revision 2, and Regulatory Guide 1.97, Revision 2, which both now call for verification of dissolved oxygen.

Insert B →

After a December 12, 1983 meeting between the NRC staff and GE personnel, the NRC staff concluded that the accuracy guidelines for the measurement of total dissolved gas could be relaxed, and that the dissolved oxygen measurement is not necessary. The concentration of total dissolved gas in the reactor coolant will be based solely on the readings from the pressure transducer in the gas collection chamber before and after the expansion of the dissolved gas into that chamber. The concentration of dissolved hydrogen will be inferred from the concentration of total dissolved gas. These conclusions are described in a letter dated January 18, 1984 to D.G. Eisenhut of the NRC staff from G.G. Sherwood of GE.

9.3.3 EQUIPMENT AND FLOOR DRAINAGE SYSTEMS

The plant equipment and floor drainage systems consist of the radioactive and nonradioactive waste drainage and collection systems. The radioactive and nonradioactive drainage systems are segregated to prevent transfer of radioactive contamination to the nonradioactive liquid wastes and uncontrolled access areas.

The nonradioactive waste drainage systems consist of the normal waste, oily waste, chemical (acid/caustic) waste, sanitary, and plant storm drainage systems. The radioactive waste drainage systems consist of the clean radwaste (CRW), dirty radwaste (DRW), high conductivity radwaste (ARW), decontamination radwaste (DECRW), detergent radwaste (DERW), and oily radwaste (ORW) drainage systems. All of the radwaste drainage systems shown in detail on Figure 9.3-7 collect and transfer potentially radioactive liquid wastes.

The equipment and floor drainage systems are provided throughout the plant to collect liquid wastes from their sources and transfer them to sumps or tanks following selective collection.

Insert B

During a meeting between GE and the NRC staff on May 2, 1984, GE agreeded to include the capability for a dissolved-gas grab sample in the PASS. The accuracies of the dissolved-oxygen and dissolved-total-gas measurements was accepted by the NRC staff in a letter from W. V. Johnston (NRC) to G.G. Sherwood (GE) dated July 7, 1984.

TABLE 9.3-7

RANGE AND ACCURACY FOR ONSITE ANALYSES

ANALYSIS	EQUIPMENT	SUITABILITY	METHOD	RANGE & ACCURACY
1. Chloride	Dionex 2020-I	Recommended by EPRI (NP- 3513)	Ion Chromatography	0.5 to 20 ppm $\pm$ 10% < 0.5ppm$\pm$ .05 ppm
2. Conductivity	G E PASS Conductivity Meter	Verified by NRC Exxon Study and G E	Direct Measurement by in-line Conductivity cell	0.54 to 2.0 µS $\pm$ 10% > 2.0 µS$\pm$ 20%
3. Radiochemical Gross Gamma Isotopic	G E Detector Multichannel	Recommended by EPRI (NP-3513)	Gamma Spectroscopy	Within a factor of two across the entire range
4. pH	pH Meter	Recommended by EPRI (NP-3513)	Potentiometry with glass Electrode	5 to 9 $\pm$ 0.3 pH units < 5 or > 9 $\pm$ 0.5
5. Boron	Selective Ion Electrode	Recommended by EPRI (NP-3513)	Potentiometry	50-1000$\pm$ 50 ppm
6. Dissolved Gas Measurement	G E PASS	Verified by G E & Accepted by the NRC	G E PASS system	< 400 cc/kg as recommended by BWR owners group and accepted by the NRC

DSER Open Item No. 157 (DSER Section 9.5.1-4.e)

REPORT ON CABLE CONCENTRATIONS

Safety-related cable trays outside the cable spreading room are not provided with automatic water suppression systems. The staff is concerned that the level of fire protection in areas containing high concentrations of cables and cable trays is not adequate. Without the added benefit of an automatic sprinkler system in these areas, the high fuel load in the form of concentrated cable trays could lead to a severe fire exposure that may ultimately threaten safety-related equipment in other areas. The staff will require the applicant to provide automatic sprinklers in accordance with the guidelines in Section C.5.e of BTP CMEB 9.5-1.

RESPONSE

Fire Areas/Fire Zones with six (6) or more cable trays within ten (10) feet of each other were reviewed. The concerned area of cable concentrations are bounded by walls or a reduction in cable concentration for at least ten (10) feet. In fire zones where cable tray distribution was considered to be evenly distributed by a qualified fire protection engineer, the entire floor area of the zone was used. In order to gage relative fire severities, an equivalent fire severity of the cable concentration was determined.

The total heat content of fully loaded cable trays in the area bounding the concentration was determined (refer to FSAR Appendix 9A, for cable heat content). This was divided by the area of cable concentration, as defined above, to determine the equivalent fire severity of the cable concentration. Electrical cable in conduit, covered cable trays or enclosed raceways were not considered as contributing to the fire load. All trays were considered to be filled to maximum capacity regardless of actual fill.

All areas of cable concentration were reviewed for accessibility and the capability to manually suppress a fire. Changes, including adding suppression systems, improving access for manual fire fighting and reducing in situ combustibles were recommended. Areas of cable concentration with an equivalent fire severity of greater than 60,000 BTU/ft^2 (3/4 hour), were reviewed closely for possible addition of suppression systems or cable tray covers, if conditions warranted. All areas of cable concentration with an equivalent fire severity of greater than 80,000 BTU/ft^2 (1 hour) were covered by an automatic suppression system except in two cases where manual fire fighting was considered adequate. Suppression systems, when added, are automatic preaction sprinkler systems over cable concentration areas per NFPA 13.

DSER Open Item No. 157 (cont'd)

To test the above criteria, BTP CMEB 9.5.1 criteria of seven horizontally stacked, 24 inch wide trays, filled to maximum capacity, were analytically placed six inches from a wall (a typical HCGS configuration). This yielded approximately 80,000 BTU/ft^2 assuming they contained all control cable and approximately 60,000 BTU/ft^2 with a mixture of control, instrument and power cable. Therefore, since all areas above 60,000 BTU/ft^2 were reviewed, this definition is a conservative approach to comparing and evaluating cable concentration.

All areas were reviewed for effect of the cable concentration fire on safe shutdown. There are no adverse effects on redundant safe shutdown equipment or cable due to a fire in any one zone. Refer to discussion on safe shutdown in Appendix 9A.

As the postulated fire event is considered to originate on or near the floor, fog nozzles can be used to control and extinguish the fire at the point of origin. The capability to cover all trays with an effective hose stream was also reviewed. A fog nozzle can be used to reach high cable trays when adjusted for narrow fog.

Attached are Tables 9.5-DSER-157-1, 9.5-DSER-157-2 and 9.5-DSER-157-3 which summarize cable concentration areas in 1E buildings and areas. Also attached are raceway drawings showing each cable tray concentration and its bounding area which is outlined in orange. Design changes are being initiated to accomplish the recommendations made therein.

Attachment:

1. Table 9.5-DSER-157-1, Auxiliary Building - Radwaste/Service Area
2. Table 9.5-DSER-157-2, Auxiliary Building - Control/DG area
3. Table 9.5-DSER-157-3, Reactor Building
4. Raceway Drawings, Auxiliary Building - Radwaste/Service Area (see attached list #1)
5. Raceway Drawings, Auxiliary Building - Control/DG Area (see attached list #2)
6. Raceway Drawings, Reactor Building (see attached list #3).

HCGS

AUXILIARY BUILDING-RADWASTE/SERVICE AREA

Dwg. No.	Sheet No.	Revision
E-1732-0	-	17
E-1734-0	-	11
E-1736-0	-	13
E-1752-0	-	14
E-1754-0	1	11
E-1756-0	-	14
E-1767-0	-	12
E-1769-0	-	7

HCGS

AUXILIARY BUILDING - CONTROL/DG AREA

Dwg. No.	Sheet No.	Revision
E-1651-1	2	22
E-1652-1	1	17
E-1653-1	-	15
E-1654-1	4	10
E-1661-1	2	19
E-1662-1	1	23
E-1663-1	1	15
E-1664-1	[illegible]	10
E-1665-1	-	14
E-1672-1	-	13
E-1673-1	-	14
E-1675-1	-	12
E-1676-1	-	10
E-1682-1	-	14
E-1683-1	-	16
E-1685-1	-	12
E-1686-1	-	11

HCGS

REACTOR BUILDING

Dwg. No.	Sheet No.	Revision
E-1511-1	2	23
E-1512-1	-	14
E-1513-1	1	13
E-1514-1	1	8
E-1515-1	1	8
E-1516-1	-	6
E-1523-1	1	13
E-1524-1	-	9
E-1525-1	-	11
E-1531-1	2	21
E-1532-1	1	15
E-1533-1	1	10
E-1535-1	-	9
E-1543-1	1	14
E-1544-1	-	8
E-1545-1	-	5
E-1555-1	1	10
E-1563-1	1	13
E-1565-1	-	10
E-1566-1	1	7
E-1566-1	2	5
E-1571-1	2	20
E-1572-1	-	15
E-1573-1	1	12
E-1574-1	-	4

HCGS

REACTOR BUILDING (cont'd)

Dwg. No.	Sheet No.	Revision
E-1575-1	-	5
E-1583-1	1	10
E-1586-1	-	8
E-1591-1	2	20
E-1592-1	-	16
E-1593-1	1	12
E-1595-1	-	8
E-1596-1	-	9
E-1611-1	2	17
E-1612-1	-	14
E-1613-1	1	12
E-1631-1	2	16
E-1632-1	-	15

TABLE 9.5-DSER-157-1

AUXILIARY BUILDING - RADWASTE/SERVICE AREA

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
1	El. 77'-0" Room 3204. Elect access. Areas 71, 72 and 73	E-1732-0 E-1734-0 E-1736-0	2-IE tray stacks of 4 trays. One tray stack turns up through ceiling. 2-Non IE tray stack of 4 and 5 trays 1-Non IE tray stack 2 trays 2-Non IE risers that go from tray stack down.	Good	Marginal. H&V ducts, cat walk & arrangement of trays make it difficult.	78,500	None	Provide sprinkler system. Combine with adjacent item 2 of control/DG area.
2	El. 102'-0" Room 3314. Elect access. Area 73	E-1752-0	1-IE-tray stack of 4 trays 1-Non IE tray stack of 4 trays 1-IE group of 4 vertical risers	Good	Yes	68,500	None	Manual fire hose adequate because of good accessibility and arrangement of trays.
3	El. 102'-0" Room 3342. Hot water heater room. Area 72	E-1754-0	2-IE tray stacks of 4 trays each. One tray stack turn up through ceiling	Good	Yes	69,000	None	Manual fire hose adequate because of accessibility and arrangement of trays.
4	El. 102'-0" Room 3302. Corridor. Area 72 & 73	E-1754-0 E-1756-0	2-IE tray stacks of 4 trays each	Good	Yes. Trays are above suspended ceiling.	48,000	None	Manual fire hose adequate.

TABLE 9.5-DSER-157-1

AUXILIARY BUILDING - RADWASTE/SERVICE AREA

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
5	El. 102'-0" Rooms 3304 & 3303. Janitor rm & Men's toilet. Area 72 & 73	E-1754-0 Sht 1 E-1756-0	1-1E tray stack of 4 trays	Good	Yes. Trays are located above suspended gypsum board ceiling. Access panel is provided in ceiling.	63,000	None	Manual fire hose adequate because of good accessibility and arrangement of trays.
6	El. 124'-0" Room 3.25. Elect access. Area 72 & 73	E-1767-0 E-1769-0	2-1E tray stacks of 4 & 5 trays each which turn down thru floor. 4-Single 1E trays	Entrance to area from control area is blocked	No, because of accessibility problem.	73,000	None	Provide sprinkler system with adjacent item 7 of control/DG area.

TABLE 9.5-DSER-157-2

AUXILIARY BUILDING - CONTROL/DG AREA

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft2)	Effect On Safe Shutdown	Comment
1	El. 54'-0" Room 5102. Elect equip room. Area 25 & 26	E-1651-1 Sht. 2 E-1661-1 Sht. 2	Several Non-1E tray stack of 1, 2, & 3 trays each. Also vert. tray risers.	Good	Yes	42,500	None	Manual fire hose adequate.
2	El. 77'-0" Rm 5207. Corridor. Area 25 & 26	E-1652-1 Sht. 1 E-1662-1 Sht. 1	2-1E tray stacks of 4 & 5 trays 3-Non-1E tray stacks of 4, 5 & 7 trays.	Good	Top trays difficult to reach.	95,000	None	Provide sprinkler system with adjacent item 1 of radwaste/ service area.
3	El. 77'-0" Rm. 5237. Corridor. Area 26, 27, & 28	E-1662-1 Sht. 1 E-1672-1 E-1682-1	1-1E tray stack of 5 trays which turn up. 3-Non-1E tray stacks of 4, 5 and 7 trays. 1-1E tray stack of 4 trays with turn up thru ceiling.	Good	Arrangement of trays make some trays not accessible.	65,000	None	Provide sprinkler system.
4	El. 102'-0" Rm. 5301. Elect access. Area 25	E-1653-1	1-1E tray stack of 4 trays 1-Non-1E tray stack of 4 trays. Both tray stacks turn down and go thru floor.	Good	Yes	62,000	None	Manual fire hose adequate because of accessibility and arrangement of trays.

AUXILIARY BUILDING - CONTROL/DG AREA

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
5	El. 102'-0" Rm 5302. Lower control equipment room. Area 25 & 26	E-1653-1 E-1663-1	12 rows of IE vertical tray risers from cabinets up thru ceiling.	Good	Yes	29,000	None	Manual fire hose adaquate.
6	El. 102'-0" Rm 5339. Elect access. Area 26, 27 & 28	E-1663-1 E-1673-1 E-1683-1	4-IE tray stacks of 4 & 5 trays which turn up or down. 1-IE tray stack of 5 trays with risers	Fair, some places head room 7' high & conduit support in the way.	No. In some places the reach is difficult.	55,000	None	Provide sprinkler system.
7	El. 124'-0" Room 5401. Elect access. Area 25 & 26	E-1654-1 Sht. 4 E-1664-1 Sht. 2	1-non-IE tray stack of 2 trays with vertical risers. 2-IE tray stacks of 5 trays that turn up.	Fair. Aisle in 2 places 2'-0" to 2'-6" wide. Also, new arch encl. around cable trays risers impairs aisle. Head room for about 30' along aisle is 6'-2" to 6'-4" high because of cable tray that runs above aisle.	No. Access limited.	71,000	None	Provide sprinkler system with adjacent item No. 6 of radwaste/ service area.

AUXILIARY BUILDING - CONTROL/DG AREA

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
8	El. 130'-0" Room 5423. H&V Wing. Area 27.	E-1675-1	1-IE tray stacks with 3 trays. 4-single non-IE trays which turn down. 1-Non-IE tray stack with 2 trays.	Good	Yes.	60,000	None	Manual fire hose adequate.
9	El. 77'-0", 102'-0", 130'-0", 137'-0" & 150'-0" Rm 5203,5323, 5331, 5405, 5419 & 5531. Cable chase. Areas 26,27 & 28.	E-1676-1 E-1686-1 E-1662-1 Sht. 1 E-1663-1 E-1664-1 Sht. 2 E-1665-1 E-1685-1 E-1683-1	1-IE tray stack with with 6 trays at El. 150'-0" Vertical cable chase at each end of 5531 connect 77'-0" to 150'-0".	Access from east end of chase at El. 150'-0" is by ladder from 137'-0" elev. the aisle space in chase good. Doors at each elevation of vertical sections.	No. Would need to use vertical access ladder at east end of chase. Trays accessible from floor, but length of chase impairs manual fire fighting.	180,000	None	Provide sprinkler system because of high load.

AUXILIARY BUILDING - CONTROL/DG AREA

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
10	El. 77'-0", 102'-0", 130'-0", 137'-0" & 150'-0" Rm 5204,5324, 5332,5406, 5420 & 5532. Cable chase. Area 26, 27 & 28.	E-1676-1 E-1686-1 E-1662-1 Sht. 1 E-1663-1 E-1664-1 Sht. 2 E-1665-1 E-1685-1 E-1683-1	1-1E tray stack with with 6 trays at El. 150'-0". Vertical cable chase at each end of 5532 connect 77'-0" to 150'-0".	Access from east end of chase at El. 150'-0" is by ladder from 137'-0" elev. the aisle space in chase good. Doors at each elevation of vertical sections.	No. Would need to use vertical access ladder at east end of chase trays accessible from floor but length of chase impairs manual fire fighting.	177,000	None	Provide sprinkler system because of high load.
11	El. 77'-0", 102'-0", 130'-0" 137'-0" & 150'-0" Rm. 5205,5325, 5333,5407, 5421 & 5533. Cable chase. Area 26, 27 & 28.	E-1676-1 E-1686-1 E-1662-1 Sht. 1 E-1663-1 E-1664-1 Sht. 2 E-1665-1 E-1685-1 E-1683-1	1-1E tray stack with with 6 trays at El. 150'-0". Vertical cable chase at each end of 5533, connect 77'-0" to 150'-0".	Access from east end of chase at El. 150'-0" is by ladder from 137'-0" elev. the aisle space in chase good. Doors at each elevation of vertical sections.	No. Would need to use vertical access ladder at east end of chase trays accessible from floor but length of chase impairs manual fire fighting.	177,000	None	Provide sprinkler system because of high load.

AUXILIARY BUILDING - CONTROL/DG AREA

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft²)	Effect On Safe Shutdown	Comment
12	El. 77'-0" 102'-0" 130'-0" 137'-0" & 150'-0" Rm. 5206,5326, 5334, 5408, 5422 & 5534. Cable chase. Area 26, 27, & 28.	E-1676-1 E-1686-1 E-1662-1 Sht. 1 E-1663-1 E-1664-1 Sht. 2 E-1665-1 E-1685-1 E-1683-1	1-IE tray stack with 6 trays at El. 150'-0". Vertical cable chase at each end of 5534 connect 77'-0" to 150'-0".	Access from east end of chase at El. 150'-0" is by ladder from 137'-0" elev. the aisle space in chase good. Doors at each elevation of vertical sections.	No. Would need to use vertical access ladder at east end of chase trays accessible from floor but length of chase impairs fire fighting.	177,000	None	Provide sprinkler system because of high load.
13	El. 150'-0" Rm 5535. HVAC Chase. Area 27 & 28.	E-1676-1 E-1686-1	1-Non IE tray stack with 2 trays. Various risers connect trays to equipment above and below.	Access limited because of ductwork and supports.	No. Access limited.	62,500	None	Although actual cable quantity is low, provide a sprinkler system because of inaccessibility.

REACTOR BUILDING

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
1	El. 54'-0" Room 4104. Core spray pump room. Area 13	E-1531-1 Sht. 2	1-1E tray stack of 3 trays. 3-1E risers that go from tray stack up thru ceiling.	Good	Yes. All concentrated trays are accessible from floor or H&V cooler unit platform.	39,000	None	Manual hose adequate.
2	El. 54'-0" Room 4105. Core spray pump room. Area 13	E-1531-1 Sht. 2	1-1E tray stack of 4 trays which turn up thru ceiling. 3-1E tray risers that go from tray stack up through ceiling.	Good	Yes. All concentrated trays are accessible from floor or H&V cooler unit platform.	43,000	None	Manual hose adequate.

REACTOR BUILDING

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
3	El. 54'-0" Room 4116. Core spray pump room. Area 15	E-1511-1 Sht. 2	1-IE tray stack of 2 trays. 2-single IE trays. 3-IE risers that go from tray stack down.	Good	Yes. All concentrated trays are accessible from floor or H&V cooler unit platform.	26,000	None	Manual fire hose adequate.
4	El. 54'-0" Room 4116. Core spray Pump room. Area 15	E-1511-1 Sht. 2	1-IE tray stack of 2 trays 2-single IE trays which turn thru ceiling 1-IE riser	Good	Yes. All concentrated trays are accessible from floor or H&V cooler unit platform.	30,500	None	Manual fire hose adequate.
5	El. 54'-0" Room 4118. Core spray pump room. Area 15	E-1511-1 Sht. 2	1-IE tray stack of 3 trays. 3-IE risers that go from tray stack down.	Good	Yes. All concentrated trays accessible from floor or H&V cooler unit platform.	35,000	None	Manual fire hose adequate.
6	El. 54'-0" Room 4107. RHR pump room. Area 19	E-1591-1 Sht. 2	1-IE tray stack of 3 trays. 3-IE risers that go from tray stack down.	Good	Yes. All concentrated trays accessible from floor.	29,000	None	Manual fire hose adequate.

REACTOR BUILDING

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
7	El. 54'-0" Room 4114. RHR pump room. Area 19	E-1571-1 Sht. 2	1-1E tray stack of 3 trays. 3-1E risers that go from tray stack down.	Cood	Yes	24,500	None	Manual fire hose adequate.
8	El. 54'-0" Room 4112. Elect. equip. room. Area 24	E-1611-1 Sht. 2	1-1E tray stack of 5 trays that turn up through ceiling. Various 1E trays stacks of 2, 3, and 4 trays.	Good	Yes	36,000	None	Manual fire hose adequate.
9	El. 54'-0" Room 4108. Elect. equip. room. Area 22.	E-1631-1 Sht. 2	Various 1E trays stacks of 2 and 3 trays. Also risers that go up thru ceiling and down to floor & MCC	Good	Yes	37,000	None	Manual fire hose adequate.
10	El. 54'-0" Room 4109. RHR heat exch. room. Area 22	E-1631-1 Sht. 2	1-1E tray stack of 3 trays 3-1E risers that go from tray stack up thru ceiling.	Good	Yes	32,000	None	Manual fire hose adequate.

REACTOR BUILDING

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
11	El. 77'-0" Room 4201. MCC area. Area 13	E-1532-1 Sht. 1	Various IE tray stacks of 2, 3, & 4 trays with risers which go down from tray stacks to MCC. Also 3-IE trays which turn up through ceiling.	Good	No. Due to arrangement of trays and equipment, manual fire fighting would be difficult.	71,000	None	Provide sprinkler system.
12	El. 77'-0" Room 4218. MCC area. Area 15	E-1512-1	1-Non-IE tray stack of 4 trays 1-Non-IE tray stack of 3 trays with Non-IE risers which go down from tray stack to MCC. Also 1 riser from tray down thru floor and 3 non-IE risers that go from tray stack up thru ceiling	Good	Yes	58,000	None	Manual fire hose adequate.
13	El. 77'-0" Room 4205. MCC area. Area 19	E-1592-1	1-Non-IE tray stack of 5 trays 2-Single non-IE trays which turn up through ceiling, 1-Non-IE riser from tray stacks to MCC.	Good	Yes. The single trays are high and somewhat difficult to reach because H&V duct and catwalk in way.	48,000	None	Manual fire hose adequate.

REACTOR BUILDING

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
14	El. 77'-0" Room 4215. MCC area. Area 21 & 24	E-1572-1 E-1612-1	1-Non-IE tray stack of 2,3 & 5 trays. 1-Non-IE tray stack of 3 & 2 Trays which turn up through ceiling. 5-Non-IE risers which go down from tray stacks to MCC & floor.	Good	Yes	36,500	None	Manual fire hose adequate.
15	El. 77'-0" Room 4207. Passage. Area 22	E-1632-1	1-Non-IE tray stack of 3 trays 3-Risers from floor to ceiling.	Good	Yes	86,000	None	Not a high concentration of trays. Fire load high due to trays located in small confined space. Manual fire hose adequate.
16	El. 77'-0" Room 4213. Passage. Area 24	E-1612-1	1-Non-IE tray stack of 4 & 5 trays. 5-IE tray risers from floor to ceiling.	Good	Yes	80,000	None	Not a high concentration of trays. Fire load high due to trays located in small confined space. Manual fire hose adequate.

REACTOR BUILDING

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concen-tration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
17	El. 102'-0" Room 4301. Corridor. Area 13 & 14	E-1533-1 Sht. 1 E-1523-1 Sht. 1	1-Non-1E tray stack of 5 trays 1-1E tray stack of 5 trays which turn down 1-1E tray stack of 4 trays which turn down.	Good	Marginal due to con-gestion.	118,000	None	Provide sprinkler system. Item 17 & 21 are adjacent and will be covered by a single sprinkler system.
18	El. 102'-0" Room 4303. MCC area. Area 13 & 16	E-1533-1 Sht. 1 E-1563-1 Sht. 1	2-1E tray stack of 4 trays. 1-1E tray stack of 4 & 5 trays. 1-1E single tray. Various risers from tray stack down to MCC and floor.	Good	Yes	65,000	None	Manual fire hose adequate.
19	El. 102'-0" Room 4317. CRD master control area. Area 13 & 14	E-1533-1 Sht. 1 E-1523-1 Sht. 1	1-Non-1E tray stack of 6 trays 1-1E tray stack of 2,3,& 4 trays. 8-Risers that go from that stack up through ceiling	Good	Yes	45,000	None	Manual fire hose adequate.

REACTOR BUILDING

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
20	El. 102'-0" Room 4331. Pers & equip. access area. Area 15	E-1523-1 E-1513-1 Sht. 1	1-Non-1E tray stack of 3 trays. 1-Non-1E tray stack of 2 trays. 3-Non-1E risers that go up through ceiling from tray stack. 4-Non-1E risers that go up through ceiling from tray stack.	Good	Yes	37,500	None	Manual fire hose adequate.
21	El. 102'-0" Room 4301. Corridor. Area 14 & 15	E-1523-1 Sht. 1 E-1513-1 Sht. 1	1-Non-1E tray stack of 5 trays which turn down. 1-Non-1E tray stack of 3 & 5 trays 1-Single non-1E tray 3-Non-1E risers that go from tray stack down.	Good	Yes. All trays are accessible.	66,500	None	Provide sprinkler system. Item 17 and 21 are adjacent and will be covered by a single sprinkler system.

REACTOR BUILDING

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
22	El. 102'-0" Room 4310. MCC area. Area 15 & 18	E-1513-1 Sht. 1 E-1543-1 Sht. 1	2-1E tray stack of 3 trays 1-1E tray stack of 2 trays. 4-1E risers that go down from tray stack. 6-1E riser that go down from tray stack to MCC.	Good	Yes	46,000	None	Manual fire hose adequate.
23	El. 102'-0" Room 4320. CRD hydr. contr. room. Area 16	E-1563-1 Sht. 1	1-1E tray stack of 4 trays. 1-1E tray stack of 2 trays. 4-1E risers that go down from tray stack.	Good	Yes	33,000	None	Manual fire hose adequate.
24	El. 102'-0" Room 4328 & 4331. Pers. & equip. access area and CRD hydr. contr. room. Area 15 & 18	E-1513-1 Sht. 1 E1543-1 Sht. 1	1-1E tray stack of 4 & 5 trays with 4 risers that go up from tray stack.	Good	Yes. All concentrated trays accessible.	53,000	None	Manual fire hose adequate.

REACTOR BUILDING

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
25	El. 102'-0" Room 4328. CRD hydr. contr. room. Area 18 & 21	E-1543-1 Sht. 1 E-1573-1 Sht. 1	1-Non-1E tray stack of 5 trays 3-Single 1E trays	Good	Yes. All concentrated trays accessible.	45,500	None	Manual fire hose adequate.
26	El. 102'-0" Rooms 4326 & 4333. Removal & repair room & storage room. Area 18, 20 & 21	E-1543-1 Sht. 1 E-1573-1 Sht. 1 E-1583-1 Sht. 1	1-Non-1E tray stack of 5 trays 2-1E tray stack of 2 & 3 trays 1-Single 1E tray 4-1E risers that go up from tray stack.	Good	Yes. All trays accessible.	60,000	None	Manual fire hose adequate.
27	El. 102'-0" Room 4310. MCC area. Area 18	E-1543-1 Sht. 1	1-Non-1E tray stack of 5 trays 1-Single Non-1E tray.	Good	Yes.	37,000	None adequate.	Manual fire hose
28	El. 102'-0" Room 4322. Pers. & equip. access area. Area 19	E-1593-1 Sht. 1	2-1E tray stacks of 2 trays, 1-Non-1E tray stack of 2 trays. 2-Non-1E tray riser that go up from tray stack.	Good	Yes.	35,000	None	Manual fire hose adequate.

REACTOR BUILDING

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concen-tration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
29	El. 102'-0" Room 4309. SACS heat exch. & pump room. Area 21 & 24	E-1573-1 Sht. 1 E-1613-1 Sht. 1	3-1E tray stacks of 3,4,&5 trays 2-1E tray stack of 2 trays. 1-1 single 1E tray. 7-1E riser that go down to MCC from tray stack. 5-1E risers from tray stack down.	Good	Yes. All trays are accessible.	44,000	None	Manual fire hose adequate.
30	El. 132'-0" Room 4401. Equip area. Area 14	E-1524-1	1-Non-1E tray stack of 3 & 4 trays 4-Non-1E single trays. 4-Non-1E tray risers down from tray stack.	Good	Yes	24,000	None	Manual fire hose adequate.
31	El. 132'-0" Room 4411. FRVC recirc unit area. Area 15	E-1514-1 Sht. 1 E-1515-1 Sht. 1	2-Non-1E tray stack of 2 trays 1-Non-1E tray stack of 3 trays 4-Risers	Access difficult due to pipe and supports.	Yes. If interference in access aisle is removed lower horizontal trays are accessible from floor. H&V duct makes it difficult to spray upper trays.	62,500	None	Improve access around pipe supports. Manual fire hose will be adequate.

REACTOR BUILDING

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
32	El. 132'-0" Room 4411. FRVS recirc. unit area. Area 15 & 18	E-1515-1 Sht. 1 E-1544-1 E-1545-1	1-IE tray stack of 4 trays. 1-Non-IE tray stack of 2 trays 4-Risers that go down from tray stack.	Access aisle width in one place is 2'-0 to 2'-6" wide	Yes.	48,500	None	Manual fire hose will be adequate. 50 feet of additional hose is needed to cover all areas.
33	El. 132'-0" Room 4410. FRVS recirc. unit area. Area 18 & 21.	E-1544-1 E-1545-1 E-1575-1	1-Non-IE tray stack of 4 trays with risers which go down. 1-Non-IE tray stack of 2 trays.	Access difficult due to H&V unit ductwork & small pipe.	Yes, if access is improved.	48,000	None	Improve access. Manual fire hose will be adequate.
34	El. 132'-0" Room 4410. FRVS recirc unit area. Area 18 & 21	E-1544-1 E1574-1 E1575-1	4-Single Non-IE trays with risers which go down 1-IE tray stack of of 2 trays. 1-IE tray stack of 3 trays. 1-IE tray stack of 4 trays. 1-IE riser	Good, head room of access aisle 6'-8" in some places.	Yes.	44,000	None	Manual fire hose adequate.

REACTOR BUILDING

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
35	El. 132'-0" Room 4412. Instr gas compr. rooms. Area 14	E-1524-1 E-1525-1	1-Non-IE tray stack of 3 trays. 3-Risers that go down from tray stack	Access difficult due to pipe & supports.	Fair, even if access is available, instr. gas compr. and tank make some trays difficult to reach.	71,500	None	Improve access. Cable concentration analytically high due tc confined area. Manual fire hose will be adequate.
36	El. 145'-0" Room 4501. Elect equip room. Area 13	E-1535-1	1-IE tray stack of 2 & 4 trays. 3-Single Non-IE trays. 7-IE risers that go up and down from tray stack.	Good, head room in some places is 6'-9" high	Yes	39,000	None	Manual fire hose adequate.
37	El. 145'-0" Room 4501. Elect equip room. Area 14.	E-1525-1	2-Non-IE tray stacks stacks of 2 & 3 trays 3-Single Non-IE trays 4-Non-IE riser that go down.	Good, head room in some places is 6'-8" high	Yes	41,000	None	Manual fire hose adequate.
38	El. 145'-0" Room 4504. Passage way. Area 19	E-1595-1	1-Non-IE tray stack of 3 trays. 1-Non-IE tray stack of 2 trays 2-Single non-IE trays. Also risers that go down to floor and MCC.	Good, head room is 6'-9" to 7'-0" in some places.	Yes	30,500	None	Manual fire hose adequate.

REACTOR BUILDING

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
39	El. 162'-0" Room 4605. Equip area. Area 19	E-1566-1 Sht. 1 E-1596-1	1-Non-IE tray stack of 4 & 3 trays 1-Single Non-IE tray. 3-Riser that go down to MCC from tray stack.	Good	Yes. Trays are 26' to 30' high. Manual fire hose will reach with new higher throw fog nozzle.	41,500	None	Manual fire hose adequate with higher throw fog nozzle.
40	El. 162'-0" Room 4604. Recombiner area. Area 19	E-1596-1	1-IE tray stack of 3 trays. 1-Single Non-IE tray. 2-Non-IE tray risers	Good	Yes. Horizontal trays are 28' to 31' high. Risers extend full height of 40'. Existing fire hose will not reach full height.	37,000	None	Manual fire hose adequate with new higher throw fog nozzle.
41	El. 162'-0" Room 4605 and 4604. Equip area & standby liquid control. area. Area 20	E-1586-1	1-Non-IE tray stack of 3 trays. 3-Non-IE risers that go down from tray stack	Good	Yes. All trays accessible.	55,500	None	Manual fire hose adequate.
42	El. 162'-0" & El. 178'-6" Room 4602 & 4504. Recombiner area. Area 16	E-1566-1 Sht. 1 E-1566-1 Sht. 2	1-IE tray stacks of 4 trays. 2-Non-IE risers. (one riser is spare)	Good	Yes. All trays accessible from el. 162'-0" or el. 178'-6".	46,500	None	Manual fire hose adequate.

REACTOR BUILDING

Item No.	Area	Elect Dwg. No.	Cable Tray Concentration Description	Accessibility (Aisle Space)	Manual Fire Fighting Capability For Trays	Cable Concentration Fire Load (Btu/Ft^2)	Effect On Safe Shutdown	Comment
43	El. 178'-6" Room 4616. FRVS recirc unit room. Area 15	E-1516-1	2-IE tray stacks of 2 trays. 2-Non-IE risers.	Good	Yes	24,500	None	Manual fire hose adequate. 50 foot of additional hose is needed to cover all areas.
44	El. 102'-0" Room 4318. Neutron monitoring system area. Area 13,14 & 16	E-1523-1 Sht. 1 E-1533-1 Sht. 1 E-1563-1 Sht. 1	1-Non-IE tray stack of 6 trays 1-Non-IE tray stack of 2 trays 2-IE tray stacks of 1, 2 & 3 trays.	Good	Yes	56,500	None	Manual fire hose adequate.
45	El. 145'-0" Room 4504 Passageway. Area 16	E-1565-1	5-Single Non-IE trays 1-Non-IE tray stack of 2 trays 3-Non-IE risers that go from horizontal trays down 1-Non-IE riser that go up thru ceiling and down thru floor. 4-IE risers that go up thru ceiling and down thru floor.	Good	Yes. Structural beam makes it somewhat difficult to spray upper portion of one riser.	33,500	None	Manual fire hose adequate.
46	El. 132'-0" Room 4411 FRVS recirc. unit area. Area 15, 17 & 18.	E-1514-1 Sht. 1 E-1524-1 E-1544-1 E-1545-1 E-1555-1 Sht. 1	1-IE tray stack of 3 & 4 trays 1-Non-IE tray stack of 2 trays 3-IE risers	Good	Yes. H&V ducts make it somewhat difficult to spray risers.	56,500	None	Manual fire hose adequate.

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DSER Open Item 176g (Section 14.2)

INITIAL PLANT TEST PROGRAM

Provide a response to Q640.17.

RESPONSE

The response to Question 640.17 is attached.

Attachment 1

14.2.12.3.40 CONFIRMATORY INPLANT TEST OF SAFETY-RELIEF VALVE DISCHARGE

a. OBJECTIVE

The objective of this test is to confirm assumptions and methodologies used in the plant unique analysis (PUA) (see a summary report in Appendix 3B) and show that the loads and structural responses documented in the PUAR for SRV discharge related loads are conservative compared to the responses which occur during actual SRV discharges.

b. PREREQUISITES

1. Power level should be sufficient to support steady steam flow, during the test duration, through SRV discharge line with normal plant operating pressure at the SRV.

2. Instrumentation for monitoring loads and structural responses has been installed and calibrated.

c. TEST METHOD

A shakedown test will be conducted to verify the test set-up is functioning properly. The testing will consist of single valve actuations (SVA) and subsequent consecutive valve actuations (CVA) of the same valve. Selection of the SRV discharge line used for testing will be based on NUREG-0763, "Guidelines for Confirmatory Inplant Tests of Safety-Relief Valve Discharges for BWR Plants," recommendations. Data will be collected and analyzed by computer code to verify design analysis.

d. ACCEPTANCE CRITERIA

Level 1

The peak pool boundary pressure during air clearing and steam discharge during the valve actuation is less than the predicted valve specified in the PUAR.

QUESTION 640.17 (SECTION 14.2.12)

Modify FSAR Subsection 14.2.12.3.24 (Relief Valves) to describe or reference any confirmatory in-plant tests of safety-relief valves to be performed in compliance with NUREG-0763 "Guidelines for Confirmatory Inplant Tests of Safety-Relief Valve Discharges for BWR Plants."

RESPONSE

A description of the confirmatory in-plant tests ~~to be performed at HCGS will be available by January 1985.~~

is in FSAR Subsection 14.2.12.3.40.

Rev 1

DSER OPEN ITEM 177 (Section 15.1.1)

The applicant was asked to justify that operation with partial feedwater heating to extend the cycle beyond the normal end-of-cycle condition would not result in a more limiting change in minimum critical power ratio than that obtained using the assumption of normal feedwater heating. The staff requires that analyses be provided before operation in this mode if a decision is made to operate in this mode. Until such analyses are provided, the staff will condition the license from operation in this mode.

Response

PSE&G will accept a license condition as described above until such time as a HCGS analysis is submitted. The response to question 440.34 has been revised to reflect this response.

QUESTION 440.34 (SECTION 15.1.1)

Operation of HCGS with partial feedwater heating might occur during maintenance or as a result of a decision to operate with lower feedwater temperature near end of cycle.

We require analyses to justify that this mode of operation will not result in (1) greater maximum reactor vessel pressure than those obtained with the assumptions used in FSAR Section 5.2.2, or (2) a more limiting MCPR than would be obtained with the assumptions used in FSAR Chapter 15.0.

Otherwise, the staff will condition the license to prohibit operation in this mode.

Provide the basis for the maximum reduction in feedwater heating.

RESPONSE

~~Reducing the feedwater temperatue before the rated EOC will result in less severe transients. The peak pressures will be lower due to the reduced steam production. The ΔCPRs will be smaller due to a less negative dynamic void coefficient and to a stronger scram caused by additional insertion of the control rods to keep the reactor power within licensed limits. The basis for the maximum reduction in feedwater heating is provided in Section 15.1.1.~~

PSE&G will accept a license condition as described above until such time as a HCGS-specific analysis is submitted.

DSER Open Item No. 198 (DSER Section 7.3.2.6)

TMI ITEM II.K.3.18 - ADS ACTUATION

Pending staff acceptance of the revised ADS logic which will be submitted as an FSAR amendment.

RESPONSE

THE SECTIONS LISTED BELOW HAVE BEEN REVISED TO REFLECT CHANGES TO THE ADS LOGIC:

SECTIONS 1.10.2
6.3.1.2.2
7.3.1.1.1.2
7.3.2.1.1
15.6.4.2.1

II.K.3.18 MODIFICATION OF ADS LOGIC - FEASIBILITY FOR INCREASED DIVERSITY FOR SOME EVENT SEQUENCES

Position

The automatic depressurization system (ADS) actuation logic should be modified to eliminate the need for manual actuation to assure adequate core cooling. A feasibility and risk assessment study is required to determine the optimum approach. One possible scheme that should be considered is ADS actuation on low reactor-vessel water level provided no high pressure coolant injection or high pressure core spray flow exists and a low pressure emergency core cooling (ECC) system is running. This logic would complement, not replace, the existing ADS actuation logic.

Response

GE performed a study for the BWR Owners Group and a revised report, NEDO - 24951, which identified eight optional means for resolution of this issue and was submitted to the NRC on October 28, 1982. The NRC judged acceptable either Option 2 (Eliminate High Drywell Pressure Trip and Add Manual Inhibit Switch) or Option 4 (Bypass High Drywell Pressure Trip and Add Manual Inhibit Switch). HCGS design incorporates Option 4.

This further automates the ADS system by providing initiation, if required, for events that result in loss of coolant without an increase in the drywell pressure such as a pipe break outside the drywell or stuck open SRVs.

The manual inhibit switch allows the operator to inhibit ADS operation without having to repeatedly press the reset switch.

These modifications have been incorporated in the HCGS design. ~~will be completed prior to fuel load~~.

• II.K.3.21 RESTART OF CORE SPRAY AND LPCI SYSTEMS

Position

The core spray and LPCI system flow may be stopped by the operator. These systems will not restart automatically on loss of water level if an initiation signal is still present. The core spray and LPCI system logic should be modified so that these systems will restart if required to assure adequate core cooling. Because this design modification affects several core cooling modes under accident conditions, a preliminary design should be

6.3.1.2.1 High Pressure Coolant Injection

The HPCI system pumps water through one of the core spray spargers and one of the feedwater spargers. The primary purpose of HPCI is to maintain reactor vessel inventory after small breaks that do not depressurize the reactor vessel. The HPCI system is also used to maintain reactor vessel inventory following a reactor isolation and coincident failure of the non-ECCS reactor core isolation cooling (RCIC) system.

6.3.1.2.2 Automatic Depressurization System

or isolated

The ADS uses a number of the reactor safety/relief valves to reduce reactor vessel pressure during small breaks, in the event HPCI fails. When reactor vessel pressure is reduced to within the design capability of the low pressure systems (core spray and LPCI), these systems provide reactor vessel coolant inventory makeup, so that acceptable post-accident reactor core coolant temperatures are maintained.

6.3.1.2.3 Core Spray

The two core spray system loops pump water into peripheral ring spray spargers mounted above the reactor core. The primary purpose of core spray is to provide reactor vessel inventory makeup and spray cooling during large breaks in which the reactor core is calculated to uncover. After ADS initiation, core spray also provides inventory makeup following a small break.

6.3.1.2.4 Low Pressure Coolant Injection

LPCI is an operating mode of the RHR system. Four pumps deliver water from the suppression chamber to four separate reactor vessel nozzles and inject directly into the core shroud region. The primary purpose of LPCI is to provide reactor vessel coolant inventory makeup following large breaks. After ADS initiation, LPCI also provides inventory makeup following a small break.

6.3.2 SYSTEM DESIGN

Detailed descriptions of the individual emergency core cooling system (ECCS) subsystems, including individual design

TABLE 6.3-2 (cont)

Variable	Value
Number of ADS valves used in analysis	5
Total minimum flow capacity, at a vessel pressure	4.0 x 10^6 lbm/h, at 1125 psid (vessel to suppression pool)
Initiating signals[1]	
a) Low water level, ~~and~~ e	≤1.0 feet above top of active fuel
High drywell pressure, and a	≥2.0 psig
Signal that at least one RHR pump or one core spray system is running (pump discharge pressure)	145 psig
Insert A	
Delay time from all initiating signals completed to the time valves are open	≤120 seconds
C. Fuel Parameters	
Fuel type	Initial core
Fuel bundle geometry	8 x 8
Lattice	C
Number of fueled rods per bundle	62
Peak technical specification linear heat generation rate	13.4 kW/ft
Initial minimum critical power ratio	1.2
Design axial peaking factor	1.4

(1) This analysis is binding for initiating signals within the indicated range.

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INSERT

A

or b) Low water level, ~~and~~	≤1.0 feet above top of active fuel
High-drywell-pressure bypass timer timed out and a	≥ 8 minutes
Signal that at least one RHR pump or one core spray system is running (pump discharge pressure)	145 psig

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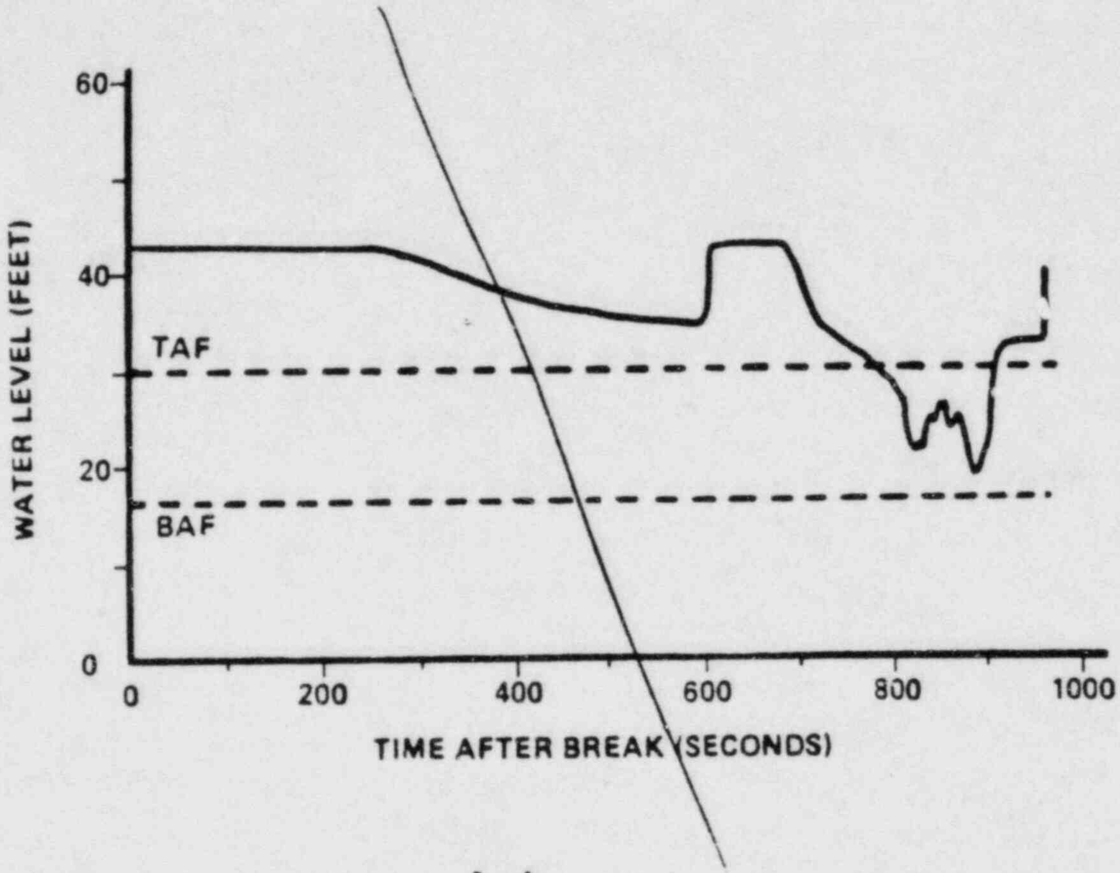


* Replace with attached figure.

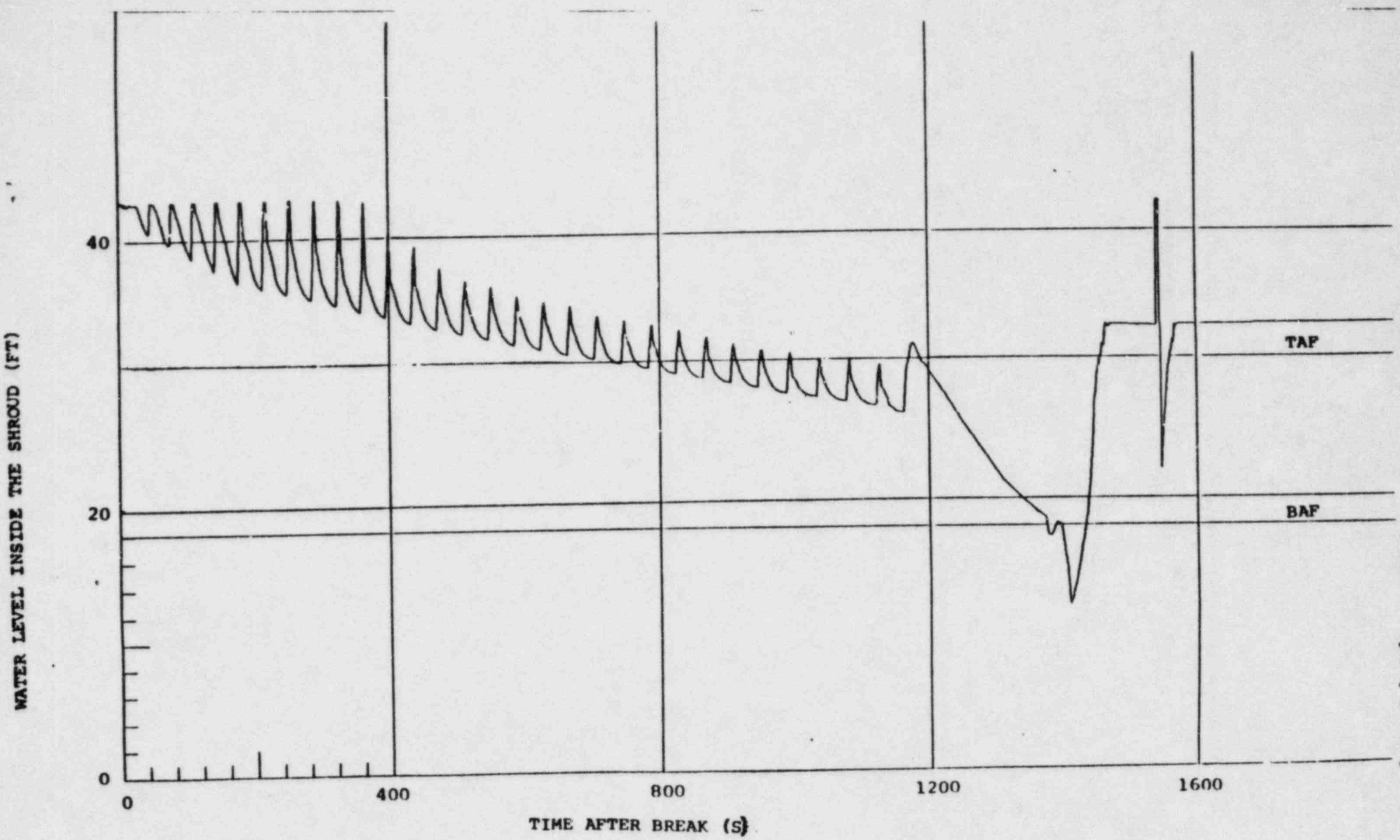
HOPE CREEK GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

WATER LEVEL INSIDE SHROUD vs TIME AFTER BREAK (MAXIMUM MAIN STEAM LINE BREAK OUTSIDE PRIMARY CONTAINMENT, FAILURE OF CHANNEL A DC SOURCE)

FIGURE 6.3-56

Fig 6.3-56



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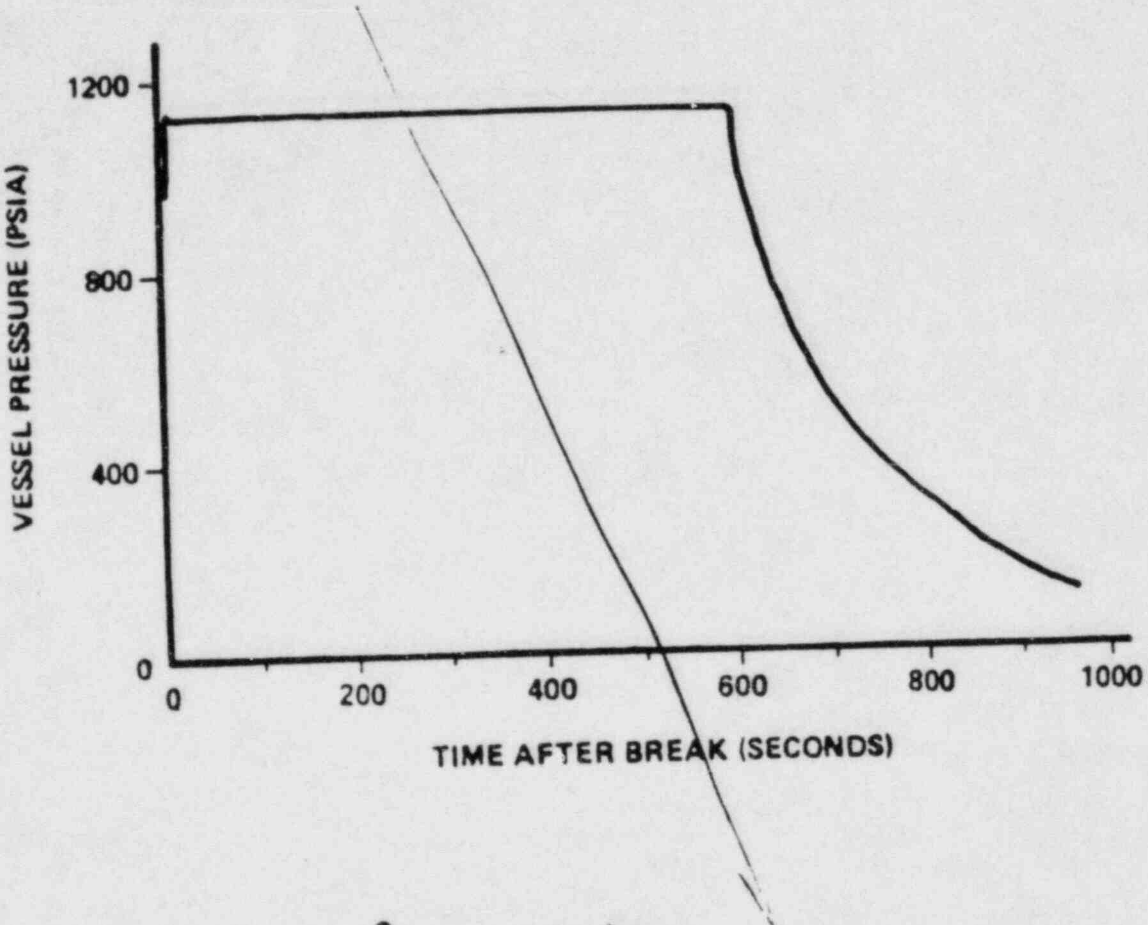


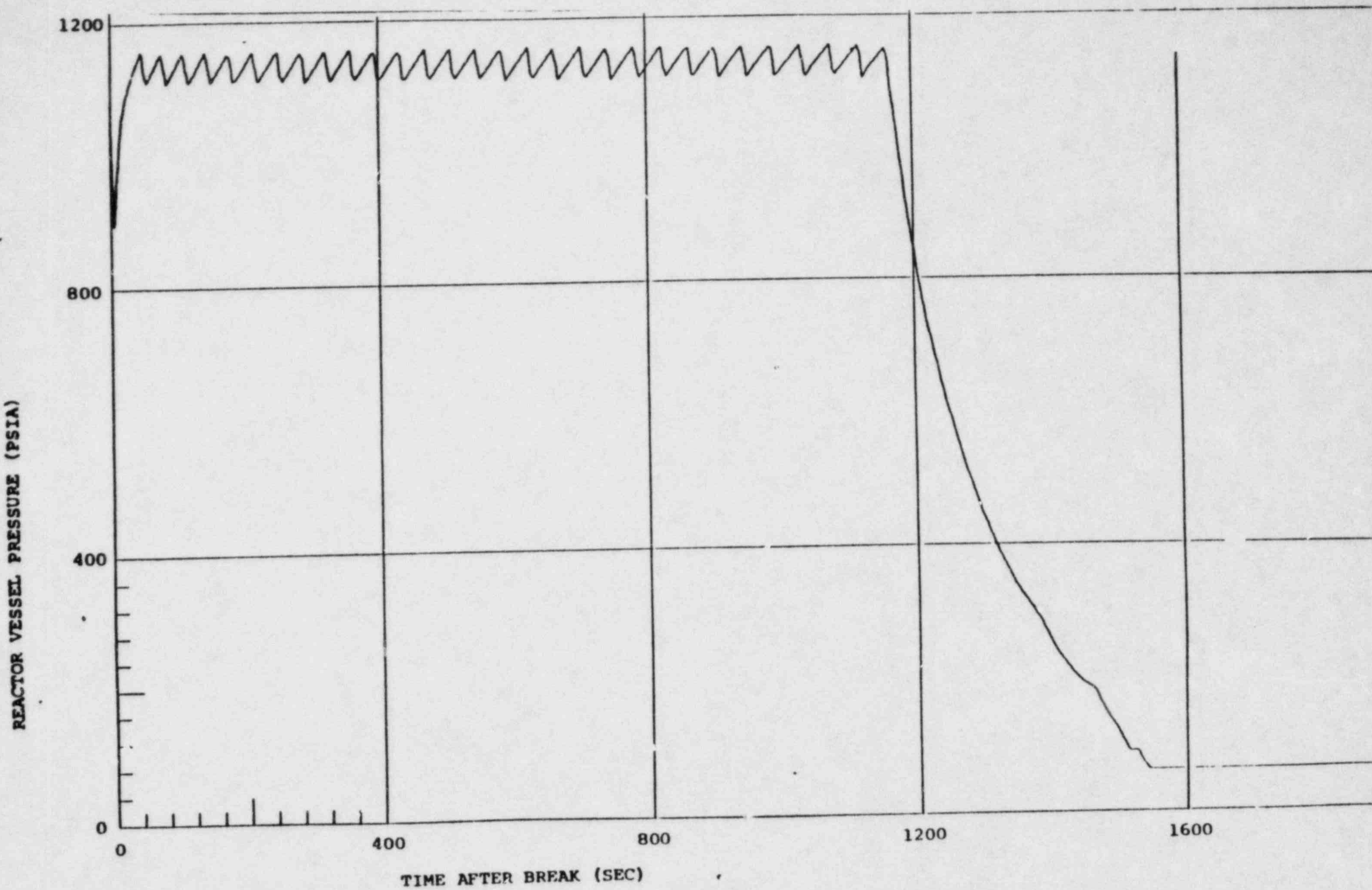
* Replace with attached figure

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

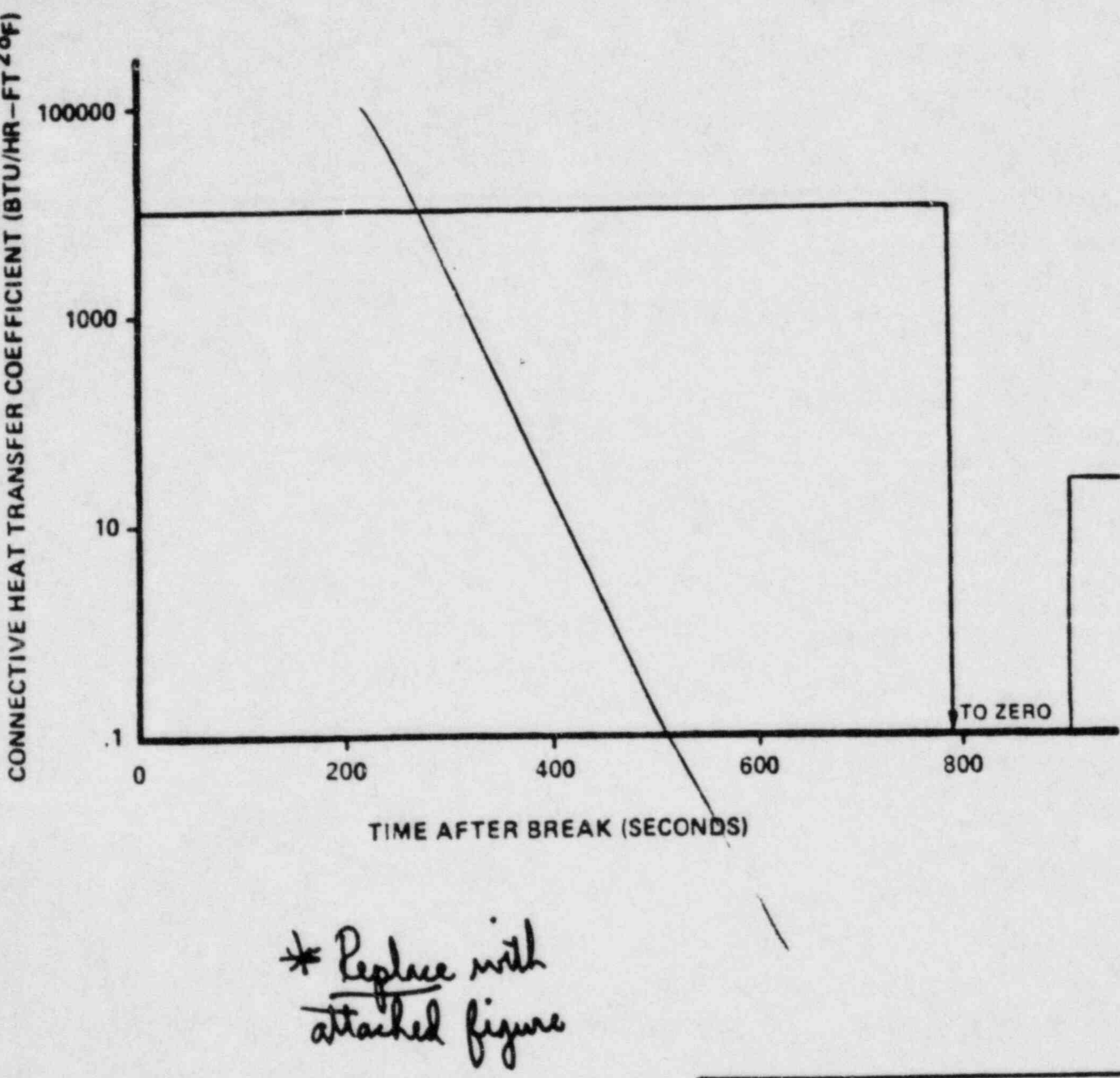
REACTOR VESSEL PRESSURE
vs TIME AFTER BREAK
(MAXIMUM MAIN STEAM LINE BREAK
OUTSIDE PRIMARY CONTAINMENT,
FAILURE OF CHANNEL A DC SOURCE)

FIGURE 6.3-57

Fig 6.3-57



REACTOR VESSEL PRESSURE (PSIA)
1200
800
400
0
0
400
800
1200
1600
TIME AFTER BREAK (SEC)



* Replace with attached figure

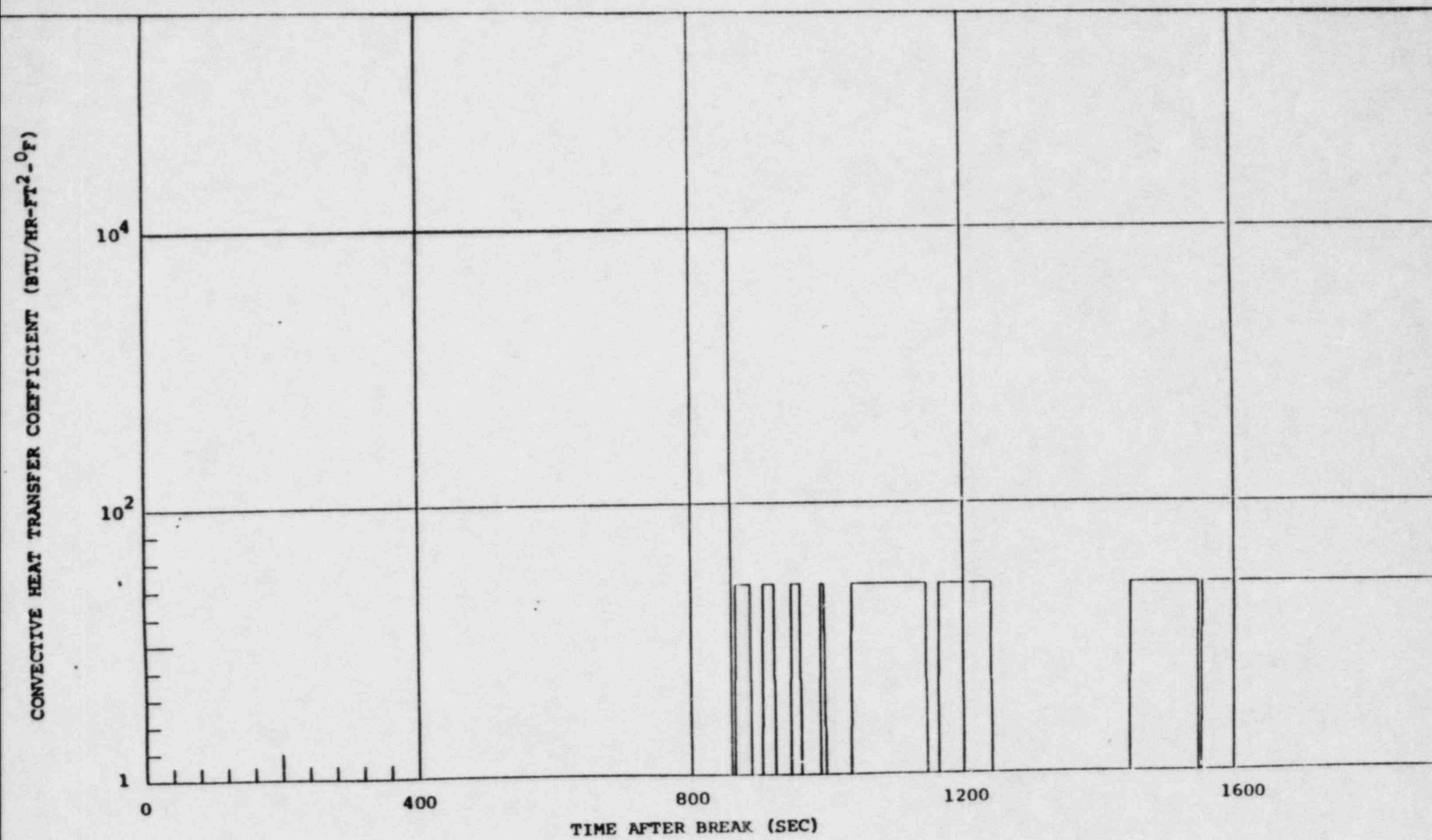
HOPE CREEK
GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

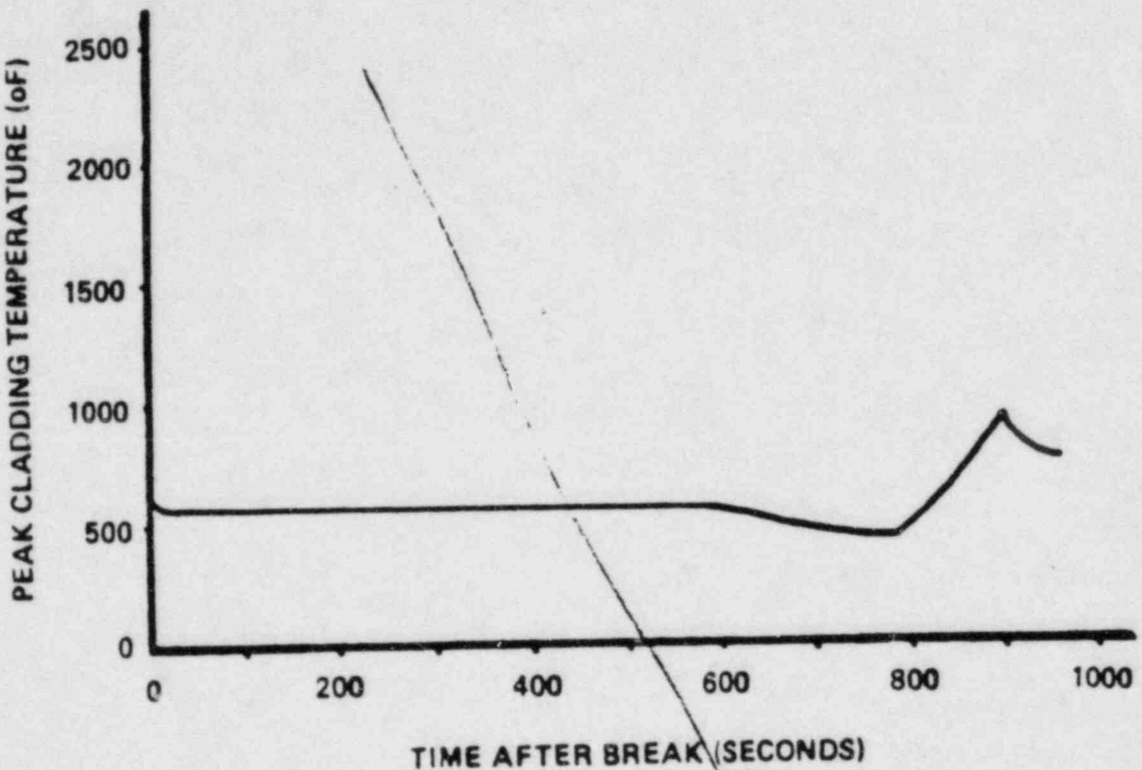
FUEL ROD CONVECTIVE HEAT TRANSFER COEFFICIENT vs TIME AFTER BREAK (SMALL BREAK MODEL) (MAXIMUM MAIN STEAM LINE BREAK OUTSIDE PRIMARY CONTAINMENT, FAILURE OF CHANNEL A DC SOURCE)

FIGURE 6.3-58

Fig 5.3-58



CONVECTIVE HEAT TRANSFER COEFFICIENT (BTU/HR-FT^2-^{0}F)
10^4
10^2
1
0
400
800
1200
1600
TIME AFTER BREAK (SEC)



* Replace with attached figure

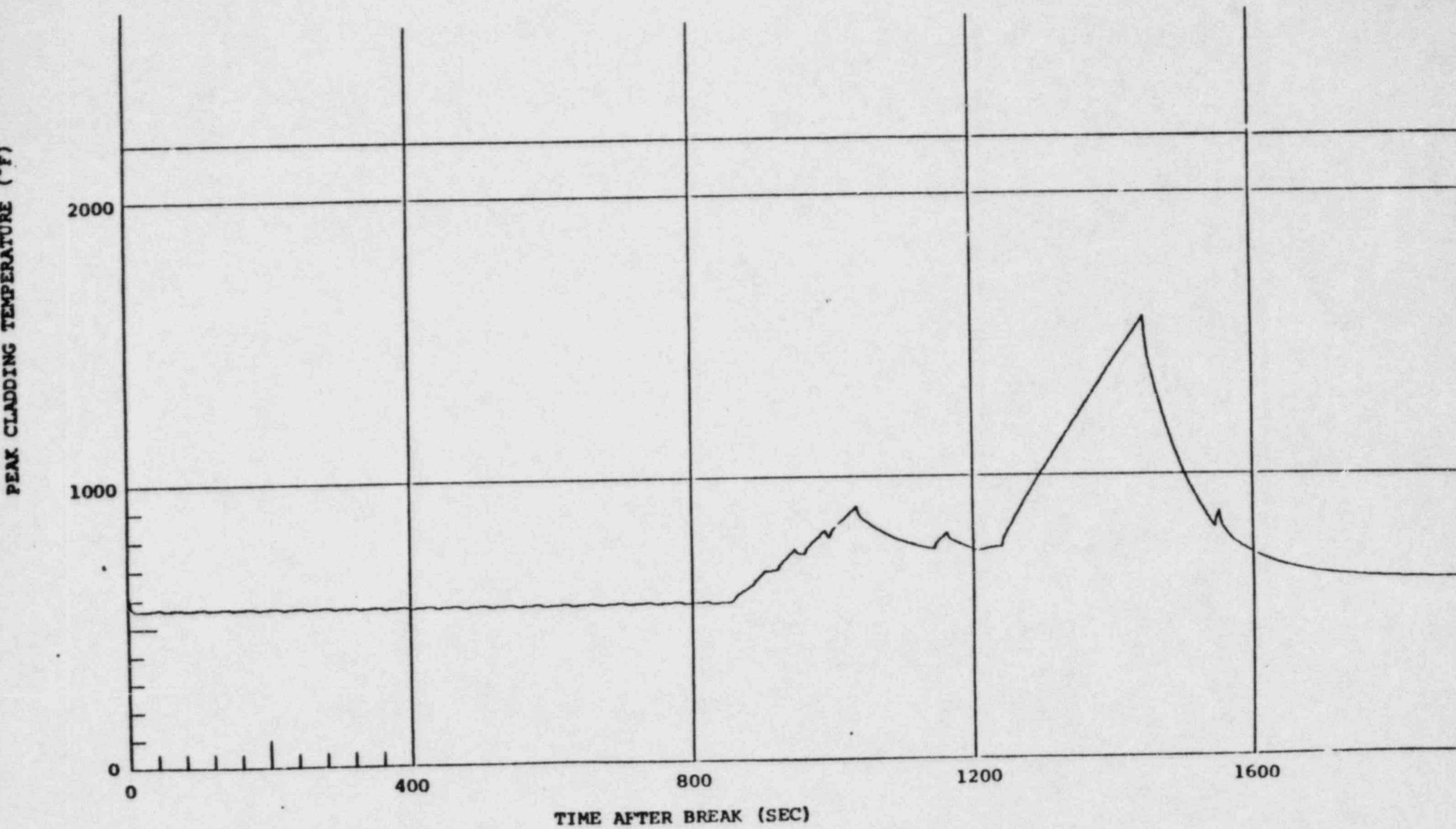
HOPE CREEK
GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

PEAK CLADDING TEMP. vs TIME AFTER BREAK (SMALL BREAK MODEL) (MAXIMUM MAIN STEAM LINE BREAK OUTSIDE PRIMARY CONTAINMENT, FAILURE OF CHANNEL A DC SOURCE)

FIGURE 6.3-59

Fig. 6.3-59


PEAK CLADDING TEMPERATURE (^{0}F)
2000
1000
0
0
400
800
1200
1600
TIME AFTER BREAK (SEC)

b. ADS operation - Schematic arrangements of system mechanical equipment is shown on Figure 5.1-3, Nuclear Boiler P&ID. ADS control logic is shown on Figures 7.3-3, Nuclear Boiler System FCD, and 7.3-4, Nuclear Boiler Logic Diagram. Instrumentation specifications are listed in Table 7.3-2 and Chapter 16. Instrument location drawings and electrical schematics are identified in Section 1.7. Operator information displays are shown on Figures 5.1-3, Nuclear Boiler P&ID, and 7.3-3, Nuclear Boiler System FCD.

To prevent inadvertent actuation of the ADS, two channels of logic for each ADS trip system (B and D) are used. Both channels must function to actuate an ADS trip system. Refer to Figure 7.3-3 for a schematic representation of the ADS initiation logic.

Each channel contains a single input from a drywell high pressure sensor. In addition, one channel includes two differential pressure sensor inputs monitoring reactor vessel low water level (L3 and L1). The second low water level trip (L3) provides confirmation of a reactor vessel low water level condition. The other channel, in addition to drywell high pressure, includes a single reactor vessel low water level (L1) input.

Insert "A"

the automatic initiation logic

~~To ensure that adequate makeup water is available after the vessel has been depressurized,~~ each logic channel includes a pump discharge pressure permissive signal indicating LPCI and/or core spray system availability for providing reactor vessel makeup water.

The automatic ADS logic trip system B requires the following ECCS pump running configurations to function:

RHR pump B or D or core spray pump B, and RHR pump B or D or core spray pump D.

The automatic ADS logic trip system D requires the following ECCS pump running configurations to function:

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RHR pump A or C or core spray pump A, and RHR pump A or C or core spray pump C.

After receipt of the initiation signals and after a delay provided by time delay relays, each of the two solenoid pilot air valves for all ADS valves are energized. This allows pneumatic pressure from each ADS valve accumulator to act on the air cylinder operator of its respective ADS valve. Each ADS trip system timer can be reset manually to delay system initiation. If reactor vessel water level is restored by the HPCI system prior to the end of the time delay, ADS initiation will be prevented.

Insert "B" →

The ADS trip system B actuates the A solenoid pilot valve on each ADS valve. Similarly, the ADS trip system D actuates the B solenoid pilot valve on each ADS valve. Actuation of either solenoid pilot valve causes the ADS valve to open to provide depressurization.

Manual initiation of the ADS trip systems or individual ADS valves is possible from the main control room. To manually initiate an ADS trip system, the control room operator must actuate two armed pushbutton switches, one for each of the logic channels associated with that trip system. Manual initiation bypasses ~~the ADS trip system time delay and all the trip logic except that the requirement that a LPCI and/or core spray system be in operation must still be satisfied.~~ The control room operator can manually open an individual ADS valve by depressing one of the two pushbutton switches (one for each pilot solenoid) that will bypass the trip logic and energize the associated pilot solenoid allowing air to open the valve.

c. ADS testability - The ADS has two complete trip systems, one in trip system B and one in trip system D. Each trip system has two channels, both of which ~~must~~ two ~~operate to~~ one initiate ADS. One channel contains a time-delay relays, to delay ADS and give the HPCI system an opportunity to restore reactor vessel level, and the second to bypass the high drywell pressure trip. Four test jacks are provided, one for each channel. To prevent spurious actuation of ADS during testing, only one channel will be tested at a time. An annunciator is provided in the main control room to indicate that a test plug is inserted in both channels of a trip system

8. Derivation of System Inputs, Paragraph 4.8 - The ESF variables are direct measures of the desired variables requiring protective actions. Refer to Sections 7.3.1.1.1 through 7.3.1.1.4 for NSSS systems. See Sections 7.3.1.1.5 through 7.3.1.1.11 for non-NSSS systems.

9. Capability for Sensor Checks, Paragraph 4.8 - Refer to Section 7.3.2.1.3, Regulatory Guide 1.22, for NSSS systems. See Sections 7.3.1.1.5 through 7.3.1.1.11 for non-NSSS systems.

10. Capability for Test and Calibration, Paragraph 4.10 - Refer to Section 7.3.2.1.3, Regulatory Guide 1.22 for NSSS systems. See Sections 7.3.1.1.5 through 7.3.1.1.11 for non-NSSS systems.

11. Channel Bypass or Removal from Operation, Paragraph 4.11 - During periodic testing of any one ESF system channel, a sensor may be isolated, tested, and returned to service under administrative control procedures. Since only one sensor that may be common to more than one system is isolated at any given time during the test interval, protective action capability for ESF system automatic initiation is maintained through the remaining redundant instrument channels.

12. Operating Bypasses, Paragraph 4.12 - The ESF system contains the following operating bypasses.

    The primary containment and reactor vessel isolation control system (PCRVICS) has ~~two~~ operating bypasses:

    STET

    (a) The first is the main steam line low pressure operating bypass, which is imposed by means of the reactor mode switch. The reactor mode switch cannot be left in any position except "run", above 10% of rated reactor power, without initiating a reactor trip. Therefore, the bypass is removed by the normal reactor operating sequence.

(b) The second is the low condenser vacuum bypass, which is imposed by means of a manual bypass switch in conjunction with closure of the main stop valves, the reactor mode switch in any position other than "run", and reactor pressure below the low pressure setpoint. Bypass removal is accomplished automatically by the opening of the main stop valves or by raising reactor pressure above the interlock pressure setpoint, and manually by placing the bypass switch in "normal" position or by placing the mode switch in the "run" position.

Insert "C" →

13. Indication of Bypasses, Paragraph 4.13 - For a discussion of bypass and inoperability indication, refer to Section 7.1.2.4, Regulatory Guide 1.47, for NSSS systems. See Sections 7.3.1.1.5 through 7.3.1.1.11 for non-NSSS systems.

14. Access to Means for Bypassing, Paragraph 4.14 - Access to means of bypassing any safety action or function for the ESF systems is under the administrative control of the control room operator. The control room operator is alerted to bypasses as described in Section 7.1.2.4, Regulatory Guide 1.47, for NSSS systems. See Sections 7.3.1.1.5 through 7.3.1.1.11 for non-NSSS systems.

Control switches that allow safety system bypasses are keylocked, with the exception of the low pressure coolant injection (LPCI) injection valve manual override switches, which are pushbutton. All keylock switches in the main control room are designed such that their key can only be removed when the switch is in the "normal" position. All keys will normally be removed from their respective switches during operation and maintained under the control of the shift supervisor. Furthermore, the key locker will be audited once per day. To change a valve position, the control switch key will be obtained from the shift supervisor via approved key control procedures.

Insert A

Two initiation signals and one permissive signal are used for the ADS. These signals are reactor vessel low water level, high drywell pressure, and RHR and/or core spray pumps running. If all these signals are present, the ADS safety relief valves will open after the ADS timer runs out; but if the high-drywell-pressure signal is not present, the ADS safety relief valves will open after the high-drywell-pressure bypass timer and the ADS timer run out.

Insert B

A manual inhibit switch is provided in each division of the ADS initiation logic. By placing this switch in the inhibit position, the operator will inhibit automatic depressurization. This will be indicated by a white status light and an annunciator window in the main control room. If the ADS has already begun and the initiation signal is sealed in, the inhibit switch will not break the seal-in, and the operation of the ADS will not be terminated.

Insert C

~~The third is the~~ (ADS) has a high-drywell-pressure trip bypass. When the high-drywell-pressure bypass timer runs out, the high-drywell-pressure trip will be bypassed, and the ADS will be initiated with only a low water level signal.

Each trip logic channel of the automatic depressurization system

allowing the ADS to initiate under low water level conditions caused by a pipe break outside containment.

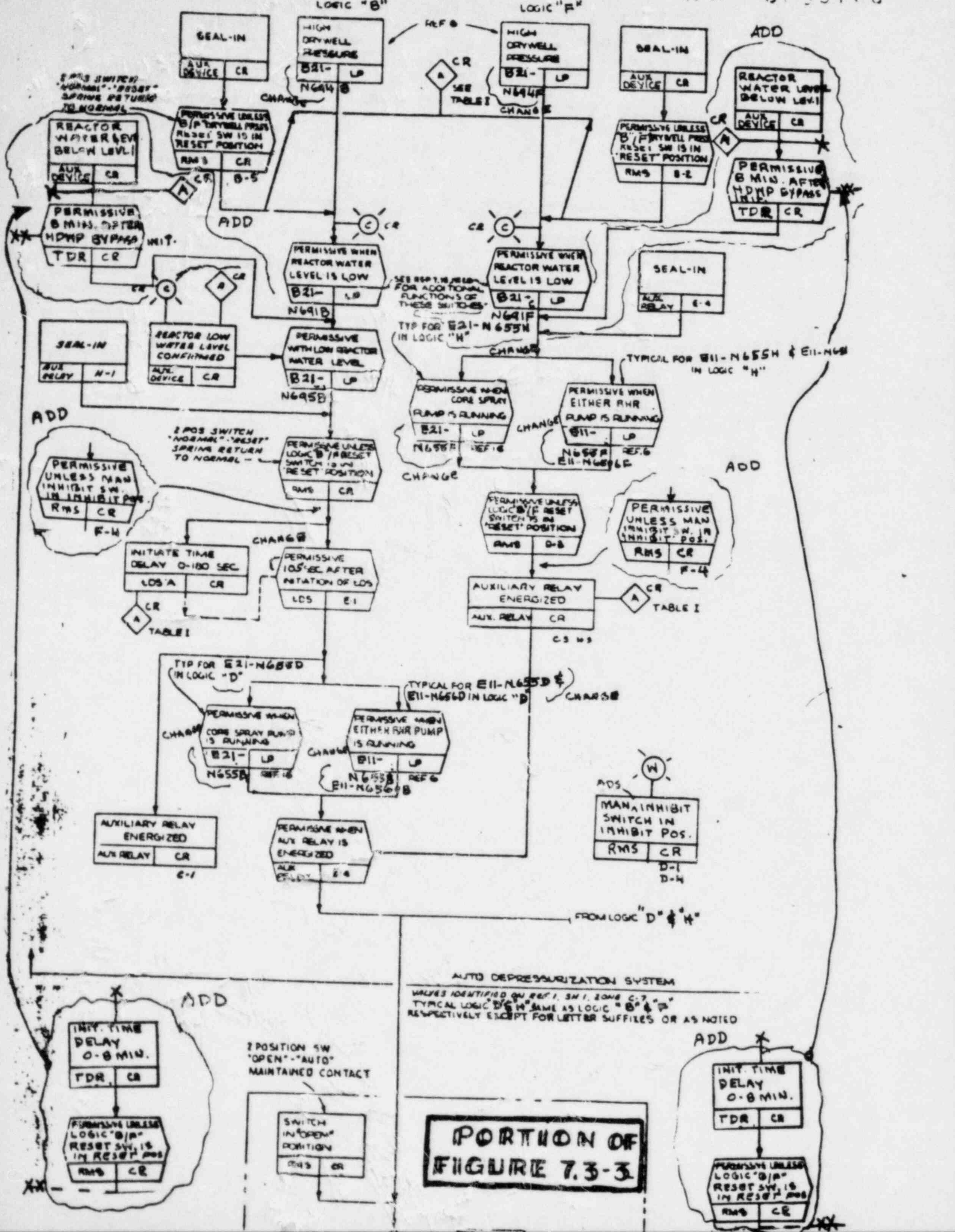
TABLE 7.3-2

AUTOMATIC DEPRESSURIZATION SYSTEM
INSTRUMENTATION RANGES

ADS Functions	Instrument	Instrument Range
Reactor vessel low water level (L1)	Level transmitter	0-125 to 0-750 in. w.g.
Reactor vessel low water level (L3)	Level transmitter	0-125 to 0-750 in. w.g.
Drywell high pressure	Pressure transmitter	0-125 to 0-750 in. w.g.
Core spray permissive	Pressure transmitter	0-1000 psig
RHR permissive	Pressure transmitter	0-1000 psig
ADS time delay	Time-delay relay	105 seconds
High-drywell-pressure bypass time delay	Time-delay relay	0-30 Minutes

(1) See the plant Technical Specifications, Chapter 16, for instrument setpoints and allowable values.

0-120



PORTION OF FIGURE 7.3-3

indicates that the limiting fault event for breaks outside the primary containment is a complete severence of one of the four main steam lines. The sequence of events and approximate times required to reach the events are given in Table 15.6-7.

Normally the operator maintains the vessel inventory and core cooling with the reactor core isolation cooling (RCIC) system. Following main steam isolation valve (MSIV) closure, the RCIC system initiates automatically on a signal of low water level. The core is covered throughout the accident, and there is no fuel damage. Without taking credit for the RCIC water makeup capability, and assuming high pressure coolant injection (HPCI) system failure, ~~the operator will initiate,~~ the automatic depressurization system (ADS) ~~or manual relief valve system to ensure termination of the accident without fuel damage.~~ will automaticlly actuate at low water level, L1, to reduce reactor pressure. The subsequent actuations of the low pressure ECCS systems will reestablish water level above the core and terminate the accident without fuel damage.

15.6.4.2.2 Systems Operation

The postulated break of one of the four main steam lines outside the containment results in mass loss from both ends of the break. The flow from the upstream side is initially limited by the flow restrictor upstream of the inboard isolation valve. Flow from the downstream side is initially limited by the total area of the flow restrictors in the three unbroken lines. Subsequent closure of the main steam isolation valves (MSIVs) further limits the flow when the valve area becomes less than the limiter area and finally terminates the mass loss when full closure is reached.

A discussion of the responses of the plant, the reactor protection system (RPS), and engineered safety features (ESF) is given in Sections 6.3, 7.3, and 7.6.

15.6.4.2.3 The Effect of Single Failures and Operator Errors

The effect of single failures has been considered in analyzing this event. The emergency core cooling system (ECCS) aspects are covered in Section 6.3. The break detection and isolation considerations are defined in Sections 7.3 and 7.6. Refer to Section 15.9 for further details.

[illegible]17 310260728

HCGS FSAR

TABLE 15.6-7

SEQUENCE OF EVENTS FOR A STEAM LINE BREAK OUTSIDE PRIMARY CONTAINMENT (1)

Approximate Time, s	Event
0	Break of one main steam line outside primary containment
~~Approx~~ 0.5	High steam line flow signal initiates closure of MSIVs
<1	Reactor scrams
≤5.5	MSIVs fully closed
~~10~~ 46	SRVs open upon high vessel pressure. The valves open and close to maintain vessel pressure at approximately 1100 psi.
~~330~~	~~Reactor water level above the core begins to drop slowly due to loss of steam through the SRVs. Reactor pressure remains at approximately 1100 psi~~
~~600~~	~~Operator initiates ADS or manually controls relief valves, vessel depressurizes rapidly~~
1160	ADS initiates on low water level, L1, following time delays imposed by both the ADS timer and the high-drywell-pressure bypass timer. Vessel depressurizes rapidly.
1370 ~~530~~	Low-pressure ECCS systems begin injection. ~~initiated, Reactor fuel~~ ~~core~~ ~~uncovered partially~~
1450 ~~See Section 6.3.3~~	Core reflooded and clad temperature heatup terminated; no fuel rod failure.

(1) The event times presented here are typical of BWR 4 plants with ADS logic modification (See [illegible] Section 1.10.2.II.K.3.18). HCGS-unique values will be provided when the HCGS-unique ECCS analysis is submitted in July, 1985.

Item II.K.3.18

See revised Section 1.10.2.II.K.3.18. ~~for the status of the resolution of this item.~~

Item II.K.3.21

In TMI Item II.K.3.21, the NRC staff requested automatic restart of the core spray and LPCI systems. Justification for making no changes to these system is provided in Section 1.10.2.II.K.3.21.

Item II.K.3.22

Resolution of TMI Item II.K.3.22 has resulted in a modification of the RCIC system to allow an automatic switchover of pump suction from the condensate storage tank to the suppression pool if the level in the condensate storage tank falls to a preset low level. Redundant level switches monitor the level in the tank. If either switch senses the low level, suction is automatically transferred to the suppression pool. A signal is provided to open the suppression pool suction valve. After the suppression pool suction valve is fully open, a signal is provided to close the suction valve on the condensate storage tank.

The HPCI and RCIC elementary diagrams (791E420AC and 791E421AC) show circuitry details. Section 7.4.1.1.2 and Table 7.4-1 have been revised to be consistent with the modifications.

DSER 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

For the information requested above, see the response to Question 430.145

QUESTION 430.145 (SECTION 8.3.1, 9.5.6)

Diesel generators for nuclear power plants should be capable of operating at maximum rated output under various service conditions. Under no load and light load operations, the diesel generator may not be capable of operating for extended periods of time under extreme service conditions or weather disturbances without serious degradation of the engine performance. This could result in the inability of the diesel engine to accept full load or fail to perform on demand. Provide the following:

a. The environmental service conditions for which your diesel generator is designed to deliver rated load including the following:

   Service Conditions

   (a) ambient air intake temperature range-°F

   (b) humidity, max-%

b. Assurance that the diesel generator can provide full rated load under the following weather disturbances:

   (1) A tornado pressure transient causing an atmospheric pressure reduction of 3 psi in 1.5 seconds followed by a rise to normal pressure in 1.5 seconds.

   (2) A low pressure storm such as a hurricane resulting in ambient pressure of not less than 26 inches Hg for a minimum duration of two (2) hours followed by a pressure of no less than 26 to 27 inches Hg for an extended period of time (approximately 12 hours).

c. In light of recent weather conditions (subzero temperatures), discuss the effects low ambient temperature will have on engine standby and operation and effect on its output particularly at no load and light load operation. Will air preheating be required to maintain engine performance? Provide curve or table which shows, performance verses ambient temperature for your diesel generator at normal rated load, light load, and no load conditions. Also provide assurance that the engine jacket water and lube oil preheat systems has the capacity to maintain the diesel engine at manufacturer's recommended standby temperatures with minimum expected ambient conditions. If the engine jacket water and lube oil preheat systems' capacity is not sufficient to do the above, discuss how this equipment will be maintained at ready stand-by status with minimum ambient temperature.

d. Provide the manufacturer's design data for ambient pressure vs engine derating.

e. Discuss the effects of any other service and weather conditions will have on engine operation and output, i.e., dust storm, air restruction, etc.
(SRP 8.3.1, Parts II & III; SRP 9.5.5, Part III, SRP 9.5.7, Parts II & III; and SRP 9.5.8, Parts II & III)

RESPONSE

~~This question is being reviewed by the diesel engine manufacturer and a response will be provided in March 1984~~

ATTACHED

0261889

430.145

~~ITEM~~ (a) The environmental service conditions are:

(a) ambient air intake range: outdoor

winter - -4°F RH 25 to 95%

summer +102°F RH 25 to 95%

(b) The diesel engine is not sensitive to humidity. The unit will tolerate, with no effect on load capability or rating, any relative humidity from 0 to 100%

~~ITEMS~~ (b) (1)&(2), & (c) Engine Rating Capability During Adverse Weather Conditions.

Engines are rated on a basis of the long term effects on the life of the engine due to altitude, ambient temperatures, and so forth. Hurricanes and tornadoes are considered short term conditions and are of no consequence to the rating or capability of these units.

The diesels are designed to operate over the ~~full~~ range of operating loads under the environmental conditions described in ~~part a)(a) & (b)~~ item a. (a) and (b).

add Insert I

cont.

(1)

430.145

INSERT I

C. It is Colt's position that there, as stated in the revised response to questions 430.111 and documented in Colt's letter from Mr. V.T. Stonehocker to Mr. Clemenson (NRC) dated September 11, 1975, is enough conservativity in the ~~direction~~ operating instructions given on Low Load (Idle) operation, ~~given~~ ~~the~~ ~~letter to the NRC~~, that the SDG should operate successfully ~~direction stands~~ regardless of any ambient temperature conditions expected at the HCGS ~~Hope Creek~~ site.

If in the unlikely event the standby diesel engine keepwarm systems fail and the systems temperatures fall to the low temperature point an alarm will be sounded in the control room. Operating / maintenance personnel will be dispatched to investigate and remedy the problem.

430.145 - Cont

of the engine.

Keepwarm systems is unable to

be placed back into service and/or the HVAC fails to keep the room at the proper temperatures

the engine can be started and idled ~~as a~~

~~last step~~ to maintain temperatures

in the standby range.

It is not anticipated that the Colt Industries

supplied diesel engines would not start or operate

at temperatures below the specified low temperatures

Colt Industries has supplied diesel engines which

have preformed successfully in much more severe climate

having similar equipment.

0261889

30.145 CONT.

(Figure 430.145-1), 12 CYL. PC2 class

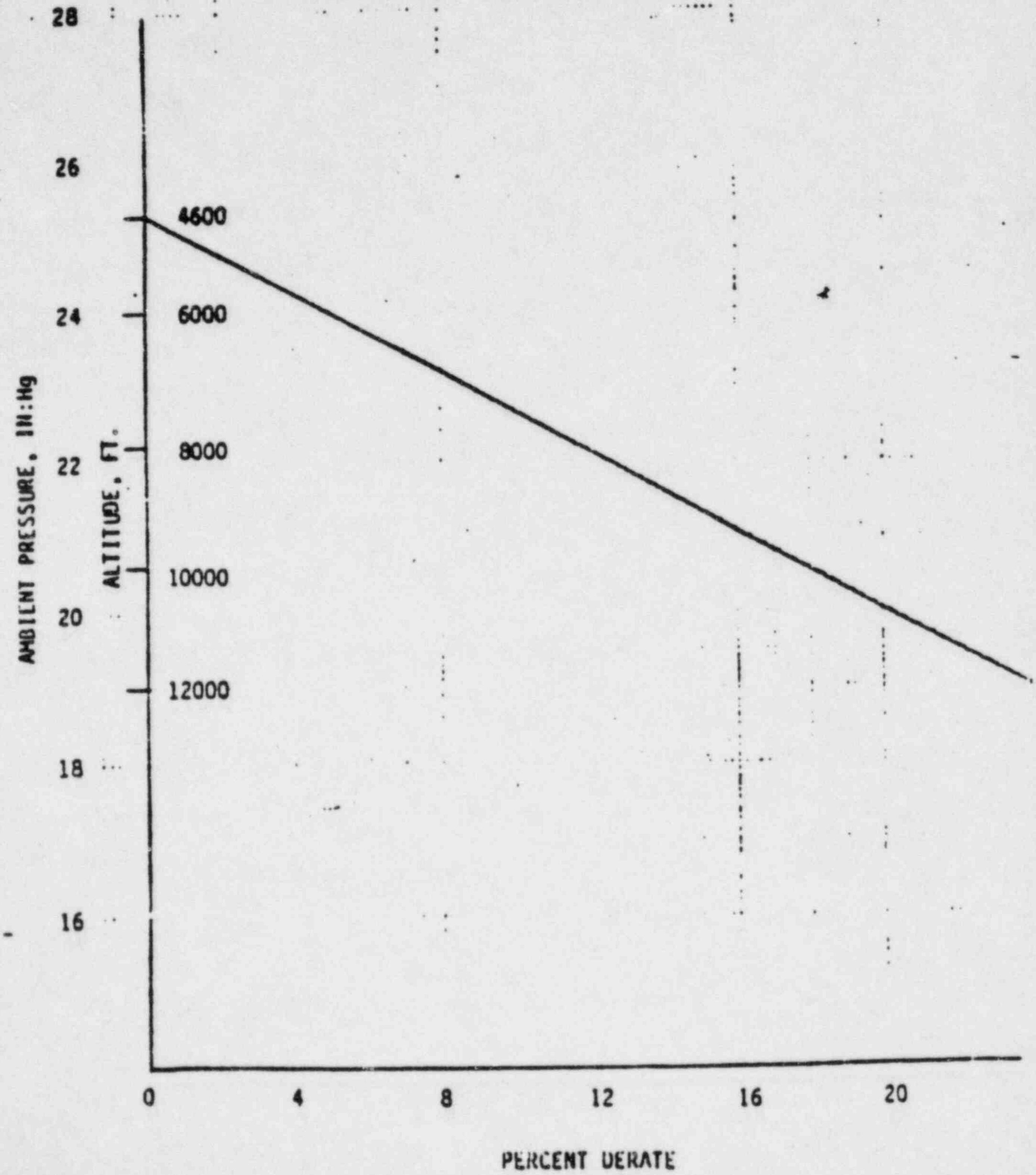
d.) A curve of the engine derating for ambient pressure (altitude) is attached. It should be noted that this curve is applicable on the long term basis - altitude derating - and is not applicable to short term phenomena such as tornadoes, hurricanes, tropical storms, or other weather depressions.

e.) The diesel engine manufacturer confirms

that as long as the unit is adequately maintained (air intake filters kept cleaned, etc), there are no other conditions adverse to the engine.

0261889

12 CYL. PC2 (4238KW AT 514 RPM)
AMBIENT PRESSURE VS. PERCENT DERATE



10/20/83 R.C.H.

430 145 ... FIGURE 430.145-1

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

Each circuit breaker for the loads described above is provided with a RESET-OFF-TEST switch at the applicable 125 Vdc switchgear. When the test switch is placed in the TEST position, the shunt trip breaker TDPU relay (40 or 60 minutes, as applicable) is energized, and the circuit breaker will trip after the time delay. Test-tripping ~~any of~~ the applicable 125 Vdc breakers can be performed during normal power operation and will not affect the operability as of the 120 Vac Class 1E inverters or BOP computer inverters ~~in that~~ 125 Vdc power is one of two backup power sources for the 120 Vac inverters.

Electrical schematics of the 125 Vdc circuitry are provided in Figure 430.28-1.

The actual circuit breaker trip tests will be performed as a non-technical specification surveillance during refueling outages. The class 1E 120VAC Distribution System is preoperational tested per Reg. Guide 1.68 Rev. 2 and Hope Creek FSAR Chapter 14 Section 14.2.12.1.37 which requires 100% logic overlap testing of all "Q" systems.

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

Section 7 does not describe or analyze the 40 minute time from loss of battery charger to loss of Class 1E instrument power supply because there is no requirement for maintaining instrumentation ac power if both the offsite power and the onsite standby diesel generator (SDG) are lost. However, the dc source for the Class 1E instrument power supply is designed for a 40 minute capability to ensure that the instrument power is uninterrupted from the time of loss of offsite power to the time that the SDG-backed motor control center (MCC) can supply power, which is on the order of a few seconds.

Figure 8.3-11, Sheet 1 shows the design of the Class 1E 120 V ac instrumentation power supply system of which there are four independent power supplies and distribution panels, corresponding to the four SDGs. The dc supply is an alternate power source and the normal power supply is from a regulated rectifier which is, in turn, supplied from an SDG-backed 480 V MCC. Thus, the dc supply (batteries and battery chargers) is only necessary during the brief interval before an SDG is started and ready to accept load upon loss of offsite power.

Section 8.3.2.1.2.2 has been revised to clarify the design intent of the dc supply to the Class 1E instrument power supply.

As indicated above the design intent of the dc supply to the Class 1E instrument power supply is to furnish power during the period when ac power is unavailable. The condition that causes unavailability of ac power is a loss of offsite power event plus failure of a SDG to start and accept load. For the loss of offsite power event, all four SDGs are automatically started and sequentially loaded as indicated on Table 8.3-1. In particular, the Class 1E instrument power supply is energized at 13 seconds after the event (Item 19 of this table). Once ac power to the Class 1E instrument power supplies is established, the alternate dc power is automatically disconnected. Since the instrument power remains uninterrupted throughout the event, there is no impact on plant safety even when considering a single failure of one channel of the Class 1E onsite ac and dc power sources.

RESPONSE CONTINUED

In the event that more than one SDG has failed to start and accept loads after loss of offsite power or if there is a station blackout, emergency operating procedures, ~~which will be developed,~~ will require use of safety sytems and components that are not ac powered. As indicated in the response to Question 430.31, the procedures will include the requirements of Generic Letter 81-04.

DSER Open item 262 (11.4.2)

Insufficient information has been provided regarding the details of a solid waste process control program and solid waste requirements of 10 CFR Part 61. Therefore, as a licensing condition prior to solid waste processing, the applicant must obtain NRC approval of the solid waste process control program addressing the requirements of 10 CFR Part 61 and Branch Technical Position ETSB 11-3.

Response:

PSE&G will contract Waste Chem to develop the solid waste process control program (PCP). The PCP will address the solid waste requirements delineated in 10 CFR Part 61 and Branch Technical Position ETSB 11-3. The PCP is tentatively scheduled for completion in March 1985 and will be submitted for the staff's review and approval at least 6 months prior to fuel load. This will be accomplished in order for the staff to complete its review of the proposed Radiological Effluent Technical Specifications (RETS). The RETS are presently scheduled for NRC submittal in the fourth quarter of 1984.

DSER Open Item No. 264 (Section 6.2.5)
SOURCES OF OXYGEN

We will require the applicant to comment on whether there are other sources of oxygen in the containment; e.g., personnel oxygen bottles, which might result in additional oxygen sources post-LOCA.

RESPONSE

The only sources of oxygen in the containment post-LOCA, other than radiolysis, are leakage from the MSIV sealing system (MSIVSS) and makeup to the main steam safety relief valve accumulators from the primary containment instrument gas system. Because the MSIVSS only operates when the steam line pressure is less than 20 psig and since the SRV's are only required when the vessel pressure is high, these sources will not contribute at the same time. Therefore, only the larger of the two (MSIVSS leakage) has been considered in the combustible gas analyses of Section 6.2.5.

HCGS

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.

DSER Open Item No. 269 (DSER Section 1.8.1.84)

EXCEPTIONS TO CODE CASE N-242, N-242-1, REGULATORY GUIDE 1.85

Delete exceptions taken to Code Case N-242, N-242-1, Regulatory Guide 1.85.

RESPONSE

Question 210.9 was revised in Amendment 7 to reflect the agreements that were reached with the NRC at the MEB meetings during the week of May 7, 1984.

QUESTION 210.9 (SECTION 5.2)

The use of ASME Code Case N-242 and N-242-1 are not acceptable unless compliance is demonstrated with Regulatory Position C.1 of Regulatory Guide 1.85.

RESPONSE

Compliance with Regulatory Position C.1 of Regulatory Guide 1.85 is discussed in Section 1.8.1.85. As requested in the Mechanical Engineering Branch meetings on May 8-10, 1984, Table 210.9-1 provides a listing of the ASME Section III, Class 1 components in the reactor coolant pressure boundary, where Code Case N-242 was invoked.

CN 3?

TABLE 210.9-1

USE OF CODE CASE N-242 ON RCPB PIPING AND COMPONENTS

Item[1]	Tag/Spool No.	Application
Piping	1-AB-050-S19	Main Steam Line Drain
	1-AB-050-S20	Main Steam Line Drain
	1-BE-023-S01	Core Spray Injection
	1-BC-003-S07	RHR RPV Head Spray
	1-BC-003-S10	RHR RPV Head Spray
	1-BC-003-S11	RHR RPV Head Spray
Safety-relief Valve	1-BC-PSV-4425	RHR shutdown suction
Flow Orifices	1-FD-FO-N032	HPCI Steam Supply
	1-FC-FE-4155	RCIC Steam Supply

[1] Code Case N-242, Revision 1 is applicable to all items.

HCGS

DSER Open Item No. 270 (DSER Section 1.8.1.85)

EXCEPTIONS TO CODE CASE N-252, REGULATORY GUIDE 1.84

Delete exceptions taken to Code Case N-252, Regulatory Guide 1.84.

RESPONSE

Question 210.10 was revised in Amendment 7 to reflect the agreements that were reached with the NRC at the MEB meetings during the week of May 7, 1984.

QUESTION 210.10 (SECTION 5.2)

The use of ASME Code Case N-252 is not acceptable unless compliance is demonstrated with Regulatory Position C.1 of Regulatory Guide 1.84.

RESPONSE

Compliance with Regulatory Position C.1 of Regulatory Guide 1.84 is discussed in Section 1.8.1.84. As requested in the Mechanical Engineering Branch meetings, on May 7-10, 1984, information on the use of Code Case N252 on the reactor coolant pressure boundary has been provided below. C 30

Code case N-252 was invoked in the fabrication of nuclear service piping. The guidance in Code Case N-252 was applied to the attachment of thermocouples to materials for the monitoring of metal temperature during post-weld heat treatment. The material involved was carbon steel (ASME P No. 1) greater than 1-1/2-inch thick.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves.

Also discuss in detail how your leak testing program will conform to the above staff position.

RESPONSE

The reactor coolant pressure boundary has been reviewed for interconnecting safety-related low pressure systems. Table 210.56-1 summarizes the results of this review. The table identifies the reactor coolant system pressure isolation valves and details the extent of compliance with the staff's position. Also identified in Table 210.56-1 are those pressure isolation valves that are leakage tested.

INSERT A

Four sets of full size P&IDs were submitted under separate cover.

The P&IDs that the NRC staff will need to review this response are identified in Table 210.56.

INSERT A

THE H.C.G.S USES TWO ISOLATION VALVES. THE ISOLATION VALVES ARE PERIODICALLY LEAK RATE TESTED AS 10 CFR 50, APPENDIX J, TYPE C VALVES OR ASME, SECTION XI, CATAGORY A VALVES. IN THE EVENT OF ISOLATION VALVE LEAKAGE A SAFETY RELIEF VALVE WILL FURTHER PROTECT THE LOW PRESSURE SYSTEM.

TYPE C LEAK RATE TESTING IN ACCORDANCE WITH 10CFR50, APPENDIX J USING GAS IS MORE CONSERVATIVE THAN LEAK RATE TESTING ISOLATION VALVES WITH SYSTEM LIQUID IN ACCORDANCE WITH ASME CODE, SECTION XI.

TABLE 210.56-1

SAFETY-RELATED LOW PRESSURE SYSTEMS
CONNECTED TO THE RCPB

INSERT B

RPV Nozzle	Connecting Line Description	Pressure Isolation Valve	Leak Tested(9)
N1B	RHR Shutdown Cooling Suction	BC-V071 BC-V164	Yes Yes
N2A-E	RHR Shutdown Cooling Return	BC-V013(1)(2)	Yes
N2F-K	RHR Shutdown Cooling Return	BC-V110(1)(2)	Yes
N4A-C	RCIC Discharge	BD-V005 AE-V003 AE-V002	Yes Yes Yes
N4D-F	HPCI Feedwater Discharge	BJ-V059 AE-V007 AE-V006	Yes Yes Yes
N5A	Core Spray	BE-V003(1)(5)	Yes
N5B	Core Spray	BE-V007(1)(6)	Yes
	HPCI Core Spray Discharge	BJ-V001(1)(4)	Yes
N6A	RHR Headspray	BC-V021 BC-V020	Yes Yes
N17A	LPCI	BC-V004(1)(8)	Yes
N17B	LPCI	BC-V016(1)(2)	Yes
N17C	LPCI	BC-V101(1)(7)	Yes
N17D	LPCI	BC-V113(1)(3)	Yes

INSERT C

(1) ~~HCGS uses one pressure isolation valve. The isolation valve is periodically leak rate tested and in the event of valve leakage, a safety-relief valve will protect the low pressure system.~~

(2) ~~Safety-relief valve BC-PSV-F025B provides overpressure protection. It has a 410 psig set pressure and a 10 gpm capacity.~~

3/7

TABLE 210.56-1 (Cont'd) Page 2 of 2

(3) Safety-relief valve BC-PSV-F025A provides overpressure protection. It has a 410 psig set pressure and a 10 gpm capacity.

(4) BJ-V003 provides pressure isolation but is not required to be leak rate tested in order to prevent overpressurization of the low pressure (pump suction) portion of the HPCI system. Should BJ-V003 leak excessively, safety-relief valve BJ-PSV-F020 will prevent the system from being overpressurized. BJ-PSV-F020 has a 100 psig setpoint and a 15 gpm capacity.

(5) Safety-relief valve BE-PSV-F012B provides overpressure protection. It has a 500 psig setpoint and a 100 gpm capacity.

(6) Safety-relief valve BE-PSV-F012A provides overpressure protection. It has a 500 psig setpoint and a 100 gpm capacity.

(7) Safety-relief valve BC-PSV-F025C provides overpressure protection. It has a 410 psig setpoint and a 10 gpm capacity.

(8) Safety-relief valve BC-PSV-F025D provides overpressure protection. It has a 410 psig setpoint and a 10 gpm capacity.

(9) Leak rate tested in accordance with 10 CFR 50, Appendix J, requirements.

+/7

INSERT B

HCGS FSAR

TABLE 210.5-6-1
SAFETY-RELATED LOW PRESSURE SYSTEMS CONNECTED TO RCPB

RPV Nozzle	Containment Penetration	Connecting Line Description	Pressure Isolation Valve	Leak Tested	Safety-Relief Protection
N1B	P-3	RHR Shutdown	BC-V164 (2)	(3)	(6)
		Cooling Suction	BC-V071 (1)	(3)	
N2A-E	P-4A	RHR Shut Down	BC-V013 (2)	(3)	(7)
		Cooling Return	BC-V014 (1)(5)	(3)	
			BC-V118 (1)	(3)	
N2F-K	P4B	RHR Shut Down	BC-V110 (2)	(3)	(8)
		Cooling Return	BC-V111 (1)(5)	(3)	
			BC-V117 (1)	(3)	
N4A-C	P2A	RCIC to	BD-V005 (1)	(4)	(9)
		Feedwater	AE-V002 (2)(5)	(4)	
			AE-V003 (1)(5)	(4)	
N4D-F	P2B	HPCI to	BJ-V059 (1)	(4)	(10)
		Feedwater	AE-V006 (2)(5)	(4)	
			AE-V007 (1)(5)	(4)	
N5A	P5A	Core Spray	BE-V003 (2)	(3)	(11)
			BE-V002 (1)(5)	(3)	
			BE-V072 (1)	(3)	
N5B	P5B	Core Spray	BE-V007 (2)	(3)	(12)
		HPCI to Core Spray	BJ-V001 (2)	(3)	(10)
			BE-V006 (1)(5)	(3)	
			BE-V071 (1)	(3)	
N6A	P10	RHR, RPV Head	BC-V020 (2)	(3)	(7)
		Spray	BC-V021 (1)	(3)	
N17A	P6A	RHR, LPCI	BC-V004 (2)	(3)	(14)
			BC-V005 (1)(5)	(3)	
			BC-V122 (1)	(3)	

INSERT B (cont'd)

N17B	P6B	RHR, LPCI	BC-V016 (2)	(3)	(7)
			BC-V017 (1)(5)	(3)	
			BC-V120 (1)	(3)	
N17C	P6D	RHR, LPCI	BC-V101 (2)	(3)	(13)
			BC-V102 (1)(5)	(3)	
			BC-V121 (1)	(3)	
N17D	P6C	RHR, LPCI	BC-V113 (2)	(3)	(8)
			BC-V114 (1)(5)	(3)	
			BC-V119 (1)	(3)	

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INSERT C

(1) 1ST PRESSURE ISOLATION VALVE
(2) 2ND PRESSURE ISOLATION VALVE
(3) LEAK RATE TESTED IN ACCORDANCE WITH 10CFR50, APPENDIX J.
(4) LEAK RATE TESTED IN ACCORDANCE WITH ASME, SECTION XI.
(5) FUNCTIONALLY TESTED AS A CATAGORY C CHECK VALVE IN ACCORDANCE WITH ASME, SECTION XI.
(6) SAFETY RELIEF VALVE BC-PSV-F029 PROVIDES OVERPRESSURE PROTECTION. IT HAS A 170 PSIG SET POINT AND A 10 GPM CAPACITY.
(7) SAFETY RELIEF VALVE BC-PSV-F025B PROVIDES OVER PRESSURE PROTECTION. IT HAS A 410 PSIG SET POINT AND A 10 GPM CAPACITY.
(8) Safety-relief valve BC-PSV-F025A provides overpressure protection. It has a 410 psig set pressure and a 10 gpm capacity.
(9) SAFETY RELIEF VALVE BD-PSV-F017 PROVIDES OVERPRESSURE PROTECTION. IT HAS A 100 PSIG SET POINT AND A 10 GPM CAPACITY.
(10) SAFETY RELIEF VALVE BJ-PSV-F020 PROVIDES OVERPRESSURE PROTECTION. IT HAS A 100 PSIG SET POINT AND A 15 GPM CAPACITY.
(11) Safety-relief valve BE-PSV-F012B provides overpressure protection. It has a 500 psig setpoint and a 100 gpm capacity.
(12) Safety-relief valve BE-PSV-F012A provides overpressure protection. It has a 500 psig setpoint and a 100 gpm capacity.
(13) Safety-relief valve BC-PSV-F025C provides overpressure protection. It has a 410 psig setpoint and a 10 gpm capacity.
(14) Safety-relief valve BC-PSV-F025D provides overpressure protection. It has a 410 psig setpoint and a 10 gpm capacity.

(15) 2ND PRESSURE ISOLATION VALVE

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QUESTION 251.7: (REVISED AUGUST 1, 1984)

Provide lead factors and predicted neutron fluence to be received by each surveillance capsule at the time of their withdrawal.

RESPONSE

Lead factors have been calculated using the base locations of the sample and nominal dimensions of the vessel. The lead factors are defined as the ratio of the neutron flux at the surveillance sample to the highest neutron flux at the wall of the vessel. The lead factor at the vessel inside diameter is 0.86 and the lead factor at one quarter of the vessel thickness is 1.20.

The END-OF-LIFE (EOL) calculated peak fluence at the inside diameter of the vessel is 1.7 x 10^{18} n/cm² (E>1.0 MeV) and at one quarter of the vessel thickness is 1.1 x 10^{18} n/cm² (E>1.0 MeV). The withdrawal of the capsules will be according to the following criteria:

a. The first set will be withdrawn when its exposure corresponds to ~~the calculated exposure of the reactor vessel wall at 25% of the reactor design life.~~ SIX EFFECTIVE FULL-POWER YEARS OR AT THE TIME WHEN THE ACCUMULATED NEUTRON FLUENCE OF THE CAPSULE EXCEEDS 5x10^{18} n/cm² OR AT THE TIME WHEN THE HIGHEST PREDICTED ΔRT_{NDT} OF ALL THE ENCAPSULATED MATERIALS IS APPROX. 50°F

b. The second set will be withdrawn when its exposure corresponds to ~~the calculated exposure of the reactor vessel wall at 75% of the reactor design life.~~ 15 EFFECTIVE FULL-POWER YEARS OR AT THE TIME WHEN THE ACCUMULATED NEUTRON FLUENCE OF THE CAPSULE CORRESPONDS TO THE APPROXIMATE EOL FLUENCE AT THE REACTOR VESSEL INNER WALL LOCATION, WHICHEVER COMES FIRST.

c. The third set will be ~~a spare to be withdrawn based on previously developed data.~~ HELD TO THE EOL (NOT LESS THAN ONCE, NOR MORE THAN TWICE THE PEAK EOL VESSEL FLUENCE AT THE VESSEL INSIDE SURFACE

~~Based on these criteria, the first specimens would be withdrawn after 11.6 years of operation with a fast neutron fluence of 4.2 x 10^{17} n/cm². The second set would be withdrawn with a fast neutron fluence of 1.3 x 10^{18} n/cm².~~

The construction tolerances on the reactor vessel required that the minimum (nominal) radius of the vessel be maintained. The applicable version of the ASME B&PV Code did allow for areas of the vessel to have larger radii. The measurement acceptance techniques for the vessel were either the use of a template to test the minimum diameter or a series of measurements to determine the diameter at various points. The measurement technique did not require the identification of the locations where the vessel diameter is longer than nominal. Hence the lead factors were calculated for the nominal dimension.

If an area of increased vessel diameter were to coincide with a location of the surveillance sample specimens, the correct fluence at the samples would be less than that predicted from measurements on the samples. If these data were used to predict the peak fluences, the values would be less than the calculated peak fluences. The calculated peak fluences using nominal dimensions will be conservative.

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RAB
8/2/84

QUESTION 281.9 (SECTION 10.4.6)

In accordance with Regulatory Position C.1 of Regulatory Guide 1.56 revision 1, describe the sampling frequency, chemical analyses, and established limits for purified condensate dissolved and suspended solids that will be performed and the basis for these limits.

RESPONSE

Sampling frequency and chemical analysis will be in accordance with the regulatory guide recommendations and the Hope Creek Generating Station Technical Specifications. Limits for dissolved and suspended solids, and the basis for these limits ~~will be provided by June 1984~~ ~~December~~ have been provided in Section 10.4.6.1.

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QUESTION 281.11 (SECTION 9.1.3)

Regarding the Spent Fuel Pool Cleanup System, provide the following information:

Describe the samples and instrumentation and the frequency of the measurements that will be performed to monitor (a) the spent fuel pool water purity and (b) the need for ion exchanger resin and filter replacement. State the chemical and radio-chemical limits to be used in monitoring the spent fuel pool water for initiating corrective action. Provide the basis for establishing these limits. Your response should consider factors such as: gross gamma and iodine activity, demineralizer and/or filter differential pressure, decontamination factor, pH and crud level.

RESPONSE

Sections 9.1.3.5 and 9.1.3.2.2.4 ~~has~~ have been revised to provide this information. ~~Chemistry sampling procedures, which discuss sampling frequency, will be available by June 1984.~~ ~~September~~

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QUESTION 430.75 (SECTION 9.5.3)

In Section 9.5.2.4 of the FSAR you state that inservice inspection tests, preventative maintenance, and operability checks are performed periodically to prove the availability of the communication systems. However no description is provided for the inservice inspection tests, preventative maintenance and operability checks to prove the availability of the emergency lighting systems. Describe the tests and checks that will be performed on the emergency lighting systems and their frequency. (SRP 9.5.3, Parts I & II).

RESPONSE

~~The frequency and extent of the periodic maintenance and testing of the three subsystems comprising the emergency lighting system will be performed using written preventive maintenance procedures in accordance with the frequencies specified in the station inspection order/preventive maintenance system or Technical Specifications.~~

Testing of the Class 1E feed will be performed in conjunction with the standby diesel generator load testing.

replace

¶ The emergency lighting systems will be demonstrated operable by energizing the lighting systems. Visual inspections will be performed: (1) semiannually for those areas of the plant that are accessible, (2) within 72 hours of achieving cold shutdown for those areas of the plant that are not accessible during plant operation, unless emergency lighting operability has been demonstrated in those areas within the past 6 months.

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QUESTION 471.4 (SECTION 12.3.2)

All accessible areas in the vicinity of the spent fuel transfer canal which are capable of having radiation levels greater than 100 rads per hour shall be shielded during fuel transfer. Use of removable shielding for this purpose is acceptable. This shielding shall be such that the resulting contact radiation levels shall be no greater than 100 rads per hour. All accessible portions of the spent fuel transfer canal shall be clearly marked with a sign stating that potentially lethal radiation fields are possible during fuel transfer. Describe how you plan to comply with the above. If a "cattle chute" shield is used in the fuel transfer canal, provide the maximum dose rates to any potentially occupied portions of the upper drywell during fuel transfer. Describe precautions to prevent access to other plant areas having radiation levels greater than 100 rads per hour.

RESPONSE

Spent fuel transfer and storage is performed underwater in the fuel transfer canal and in the spent fuel pool. Since HCGS does not have a cattle chute shield design, administrative controls will be used to preclude access to the drywell during fuel movement.

upper elevations of the

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