PSE&G Public Service Electric and Gas Company

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Robert L. Mittl General Manager
Nuclear Assurance and Regulation

August 3, 1984

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, MD 20014

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, FSAR question responses and structrual geotechnical audit item responses, listed in Attachment 3.

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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Attachments

C D. H. Wagner
USNRC Licensing Project Manager

W. H. Bateman
USNRC Senior Resident Inspector

FM05 1/2

DATE: 8/3/84

ATTACHMENT 1

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

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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	Open	
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	Open	
26	2.5.4.4	Intake structure sliding stability	Open	
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	Open	
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	Open	
33	3.5.2	Structures, systems, and components to be protected from externally generated missiles	Open -	-
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	Open	
38	3.6.2	Design of pipe rupture restraints	Open	
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	8/4/84
41	3.8.2	Steel containment buckling analysis	Complete	6/1/84
42	3.8.2	Steel containment ultimate capacity analysis	Complete	6/1/84
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	6/1/84
46	3.8.5	ACI 349 deviations for foundations	Complete	6/1/84
47	3.8.6	Base mat response spectra	Complete	6/1/84
48	3.8.6	Rocking time histories	Complete	6/1/84
49	3.8.6	Gross concrete section	Complete	6/1/84
50	3.8.6	Vertical floor flexibility response spectra	Complete	6/1/84
51	3.8.6	Comparison of Bechtel independent verification results with the design-basis results	Complete	8/4/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
52	3.8.5	Ductility ratios due to pipe break	Complete	8/3/84
53	3.8.6	Design of seismic Category I tanks	Complete	6/1/84
54	3.8.6	Combination of vertical responses	Complete	6/1/84
55	3.8.6	Torsional stiffness calculation	Complete	6/1/84
56	3.8.6	Drywell stick model development	Complete	6/1/84
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	6/1/84
60	3.8.6	BSAP element size limitations	Complete	6/1/84
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	6/1/84
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	8/3/84
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	6/1/84
78	3.8.6	Dead load calculations	Complete	6/1/84
79	3.8.6	Post-modification seismic loads for the torus	Complete	6/1/84
80	3.8.6	Torus fluid-structure interactions	Complete	6/1/84
81	3.8.6	Seismic displacement of torus	Complete	6/1/84
82	3.8.6	Revi w of seismic Category I tank design	Complete	6/1/84
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	6/1/84
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	Open	
87	3.9.1	Information on transients	Open	
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	6/15/84
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
100b	3.9.6	10CFR50.55a paragraph (g)	Open	
101	3.9.6	PSI and ISI programs for pumps and valves	Open	
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	Open	
103a2	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103a3	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103a4	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103a5	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103a6	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103a7	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103b1	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103b2	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103b3	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103b4	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103b5	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	

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	Open	
103c1	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103c2	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103c3	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103c4	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
104	3.11	Environmental qualification of mechanical and electrical equipment	NRC Action	
105	4.2	Plant-specific mechanical fracturing analysis	Complete	7/18/84
106	4.2	Applicability of seismic andd LOCA loading evaluation	Complete	7/18/84
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	Open	
109b	4.4.7	TMI-2 Item II.F.2	Open	
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	Open	
118	5.3.4	Lead factors and neutron fluence for surveillance capsules	Open	

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	Open	
120b	6.2	TMI Item II.E.4.2	Open	
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	6/1/84
124b	6.2.1.5.1	RPV shield annulus analysis	Complete	6/1/84
124c	6.2.1.5.1	RPV shield annulus analysis	Complete	6/1/84
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)	Open	
126b	6.2.1.6	Redundant position indicators for vacuum breakers (and control room alarms)	Open	
127	6.2.1.6	Operability testing of vacuum breakers	Complete	7/18/84
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	Open	
133b	6.2.4.1	Containment purge system	Open	
133c	6.2.4.1	Containment purge system	Open	
134	6.2.6	Containment leakage testing	Complete	6/15/84
135	6.3.3	LPCS and LPCI injection valve interlocks	Open	
136	6.3.5	Plant-specific LOCA (see Section 15.9.13)	Complete	7/18/84
137a	6.4	Control room habitability	Open	
137b	6.4	Control room habitability	Open	
137c	6.4	Control room habitability	Open	
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	7/27/84
140b	9.1.2	Spent fuel pool storage	Complete	7/27/84
140c	9.1.2	Spent fuel pool storage	Complete	7/27/84
140d	9.1.2	Spent fuel pool storage	Complete	7/27/84
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 p[illegible] cooling and cleanup system	Complete	8/1/84
142a	9.1.4	Light load handling system (related to refueling)	Closed (5/30/84-Aux.Sys.Mtg.)	6/29/84
142b	9.1.4	Light load handling system (related to refueling)	Closed (5/30/84-Aux.Sys.Mtg.)	6/29/84
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	7/27/84
144b	9.2.1	Station service water system	Complete	7/27/84
144c	9.2.1	Station service water system	Complete	7/27/84
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
148	9.3.2	Post-accident sampling system (II.B.3)	Open	(Rev 1)
149a	9.3.3	Equipment and floor drainage system	Complete	7/27/84
149b	9.3.3	Equipment and floor drainage system	Complete	7/27/34
150	9.3.6	Primary containment instrument gas system	Complete	8/1/84 (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	Open	
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	Open	

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	Open	

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	Open	
176c	14.2	Initial plant test program	Complete	7/27/84
176d	14.2	Initial plant test program	Complete	7/27/84
176e	14.2	Initial plant test program	Complete	7/27/84
176f	14.2	Initial plant test program	Open	
176g	14.2	Initial plant test program	Open	
176h	14.2	Initial plant test program	Open	
176i	14.2	Initial plant test program	Complete	7/27/84
177	15.1.1	Partial feedwater heating	Complete	7/18/84
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	Open	
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	8/3/84

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	6/1/84
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	Open	
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	6/1/84
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	F. 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	6/1/84
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	Open	
218	9.5.1.1	Fire hazards analysis	Complete	6/1/84
219	9.5.1.2	Fire protection administrative controls	Open	
220	9.5.1.3	Fire brigade and fire brigade training	Open	
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	8/1/84
226	8.2.2.5	Grid stability	Complete	8/1/84
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	Open	
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	Open	
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	8/1/84
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 with-stand 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	Open	

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
TS-1	2.4.14	Closure of watertight doors to safety-related structures	Open	
TS-2	4.4.4	Single recirculation loop operation	Open	
TS-3	4.4.5	Core flow monitoring for crud effects	Complete	6/1/84
TS-4	4.4.6	Loose parts monitoring system	Open	
TS-5	4.4.9	Natural circulation in normal operation	Open	
TS-6	6.2.3	Secondary containment negative pressure	Open	
TS-7	6.2.3	Inleakage and drawdown time in secondary containment	Open	
TS-8	6.2.4.1	Leakage integrity testing	Open	
TS-9	6.3.4.2	ECCS subsystem periodic component testing	Open	
TS-10	6.7	MSIV leakage rate		

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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: 8/3/84

DRAFT SER SECTIONS AND DATES PROVIDED

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

CT:db

MP 84 95/03 01

DATE: 8/3/84

ATTACHMENT 3

OPEN ITEM	DSER SECTION NUMBER	SUBJECT
4	2.4.2.2	Ponding levels
6c	2.4.10	Stability of erosion protection structures
7a	2.4.11.2	Thermal aspects of ultimate heat sink
7b	2.4.11.2	Thermal aspects of untimate heat sink
29	3.5.1.1	Internally generated missiles (outside containment)
39	3.7.2.3	SSI analysis results using finite element method and elastic half-space approach for containment structure
40	3.7.2.3	SSI analysis results using finite element method and elastic half-space approach for intake structure
51	3.8.6	Comparison of Bechtel indeper-dent verification results with the design-basis results
52	3.8.6	Ductility ratios due to pipe break
66	3.8.6	Impedance analysis for the intake structure
147a	9.3.1	Compressed air systems
147b	9.3.1	Compressed air systems
147c	9.3.1	Compressed air systems
147d	9.3.1	Compressed air systems
150	9.3.6	Primary containment instrument gas system

OPEN ITEM	DSER SECTION NUMBER	SUBJECT
186	7.2.2.3	Testability of Plant protection systems at power
187	7.2.2.4	Lifting of leads to perform surveillance testing
201	7.4.2.3	RCIC/HPCI interactions
202	7.5.2.1	Level measurement errors as a result of environmental temperature effects on level instrumentation reference leg.
203	7.5.2.2.	Regulatory Guide 1.97

Attachment 3 con't.

QUESTION NUMBER	FSAR SECTION	STRUCTURAL/ GEOTECHNICAL AUDIT ITEM	MEETING DATE
430.67	9.5.2		
430.79	9.5.4	A.3	1/10/84
430.113	9.5.5	A.4	1/10/84
430.123	9.5.6	A.11	1/10/84
430.136	9.5.7	A.12	1/10/84
430.137	9.5.7	A.13	1/10/84
430.138	9.5.7	A.16	1/10/84
430.145	9.5.6	B.5	1/10/84
430.149	9.5.8	B.9	1/10/84
		A.7	1/11/84
		A.8	1/11/84
		A.12	1/11/84
		A.16	1/11/84
		B.12	1/11/84
		A.1	1/12/84
		A.3	1/12/84
		A.4	1/12/84
		B.2	1/12/84

ATTACHMENT 4

DSER OPEN ITEM No 4 (DSER SECTION 2.4.2.2)

PONDING LEVELS

The applicant states that safety-related buildings have been designed with either roof parapets and scuppers or no curbing at all so that if internal roof drains become clogged, the precipitation accumulation would overflow before the basic roof loading would be exceeded or roof hatches flooded. Because there was insufficient information available to enable the staff to reach the same conclusion, the applicant has been asked to provide additional information and detailed analysis of the roof drainage system including the ponding levels on roofs of safety-related structures. Until the additional information and analysis are available, the staff cannot conclude that the plant meets the requirements of GDC 2 with respect to the effect of local intense precipitation on roofs.

Response:

The response to this item is provided in the response to Question 240.13

QUESTION 240.13 (Section 2.4.2.3)

Provide your detailed analysis of the PMP ponding levels on roofs of safety-related structures requested in Q240.7. Details should identify and provide information on the roof area of each sub-drainage area for each safety-related structure; the size, number and distance above roof (elevation) of the invert of each scupper (overflow drain) for each drainage system, and the elevation of the curb of each roof hatch within each roof drainage area system. Also provide details used to conclude that the ponding resulting from PMP does not effect safety-related facilities.

RESPONSE

Section 3.4.1.1 has been revised to respond to this question.

DSER OPEN ITEM 4

b. Waterstops provided in exterior wall construction joints and seismic separation joints below flood level

c. A minimum number of openings in exterior walls a d slabs below flood level (these openings are designed to prevent intrusion of flood water.)

d. Water-pressure-tight doors installed in exterior walls below flood level

e. Exposed equipment hatches installed above flood level; those below flood level installed behind exterior walls designed to prevent intrusion of water

f. Continuous waterproofing systems applied to the underside of base slabs and on exterior walls to grade, as discussed below.

DELETE + REPLACE WITH "INSERT A"

Except for the intake structure, the HCGS safety-related structures are provided with roof drainage systems capable of handling a maximum rainfall rate of 4 inches per hour for a period of 20 minutes. ~~In the unlikely event that the roof drains become clogged, redundant overflow drains are provided approximately 6 inches above the main roof drain elevations except for the plant cancelled area, which has no parapets. The roof drainage system disposes water through the yard drainage system. To preclude ponding for significantly greater rainfall intensities segments of the parapets are removed where necessary.~~

The intake structure roof is designed without parapets or other continuous obstructions and is sloped to shed the water. Accordingly, no significant ponding will occur.

To prevent seepage into any Seismic Category I structure all roof openings are watertight and provided with either metal sleeves or concrete curbs of sufficient height to exceed any possible ponding levels.

As an additional margin of safety, all Seismic Category I roofs are designed to withstand a loading of 150 lb/ft^2, which is greater than the loading resulting from the maximum ponding on the roofs.

Doors and penetrations in exterior walls of the auxiliary and reactor buildings are protected against water inflow up to elevation 127 feet for parts of the south exterior walls and up to elevation 121 feet of other exterior walls. Interior drains from the radwaste areas are independently piped to the liquid waste disposal system and are not connected to the yard drainage system. Wall penetrations above elevations 121 feet and 127 feet

INSERT A

The roof drainage system consists of roof drains and 6-inch diameter scuppers located 6 inches above the roof drain elevations. Supplementing the roof drain system is a series of openings in the parapets of the roofs of the buildings. The 6-hour, local, all-season PMP was used to size these openings. The PMP, which is 27.5 inches, is distributed into 5-minute increments such that the maximum amounts for durations of 1 hour, 30 minutes, 15 minutes and 5 minutes are 18.1, 13.7, 9.5 and 6 inches respectively. Roof elevations, sub-drainage areas, and the dimension of parapet openings are shown in Table 3.4-3. A schematic of the roof drainage is shown on Figure 3.4-4.

The routing of the PMP assumes no losses, the roof drain system to be non-functional, and the ponding is allowed up to the limiting elevation of the top of the curb of each roof hatch within each roof drainage area system. Prior to the PMP, an initial level of ponding at the invert elevation of the parapet openings is assumed (invert elevation is 6 inches above the roof drain elevation).

A

[redacted] rating curve for each rectangular parapet opening was derived using the equation:

$$Q = CLH^{1.5}$$

where:

Q is the discharge in cubic feet per second

C is the discharge coefficient (3.0)

L is the length in feet of the parapet opening

H is the head in feet of water above the invert of the parapet opening

B

The flow capacity of the 8-inch diameter openings is derived using the following short culvert equations:

Inlet control flow for unsubmerged inlets:

$$\frac{H}{D} = \frac{H_c}{D} + k\left(1.273\ \frac{Q}{D^{5/2}}\right)^m$$

Inlet control flow for submerged inlets:

$$\frac{H}{D} = \frac{h_1}{D} + k_1\left(\frac{Q}{D^{5/2}}\right)^2$$

where:

H is the total head above the invert of the opening in feet

(A)

Rectangular parapet openings were analyzed as a broad-crested weir for upstream water-surface elevations below the top of the opening. For unsubmerged conditions the

(B) For conditions when the upstream water-surfac elevation was higher than the top of the openi the orifice equation was used:

$$Q = CA\sqrt{2gH}$$

where:

Q is the discharge in cubic feet per second
C is the discharge coefficient (taken as 0.6)
A is the area of the opening in square feet
H is the head measured from the centerline of the opening in feet
g is the acceleration of gravity (32.2 feet/second

4

DSER OPEN ITEM

Insert A (cont'd)

H_c is the specific energy

Q is the discharge in cubic feet per second

D is the opening diameter in feet

k, m, $\frac{h_1}{D}$ and k_1 are the inlet control performance coefficients. The experimentally determined values for a square edged entrance are:

$k = 0.0098$

$m = 2.0$

$\frac{h_1}{D} = 0.67$

$k_1 = 0.0645$

Since the limiting water depths are greater than the ponding levels resulting from the PMP (as shown in Table 3.4-3), the ponding levels do not effect safety-related facilities.

DSER OPEN ITEM

TABLE 3.4-3

Maximum Ponding Depths on Roofs
of Safety-Related Structures
for Local 6 Hour PMP

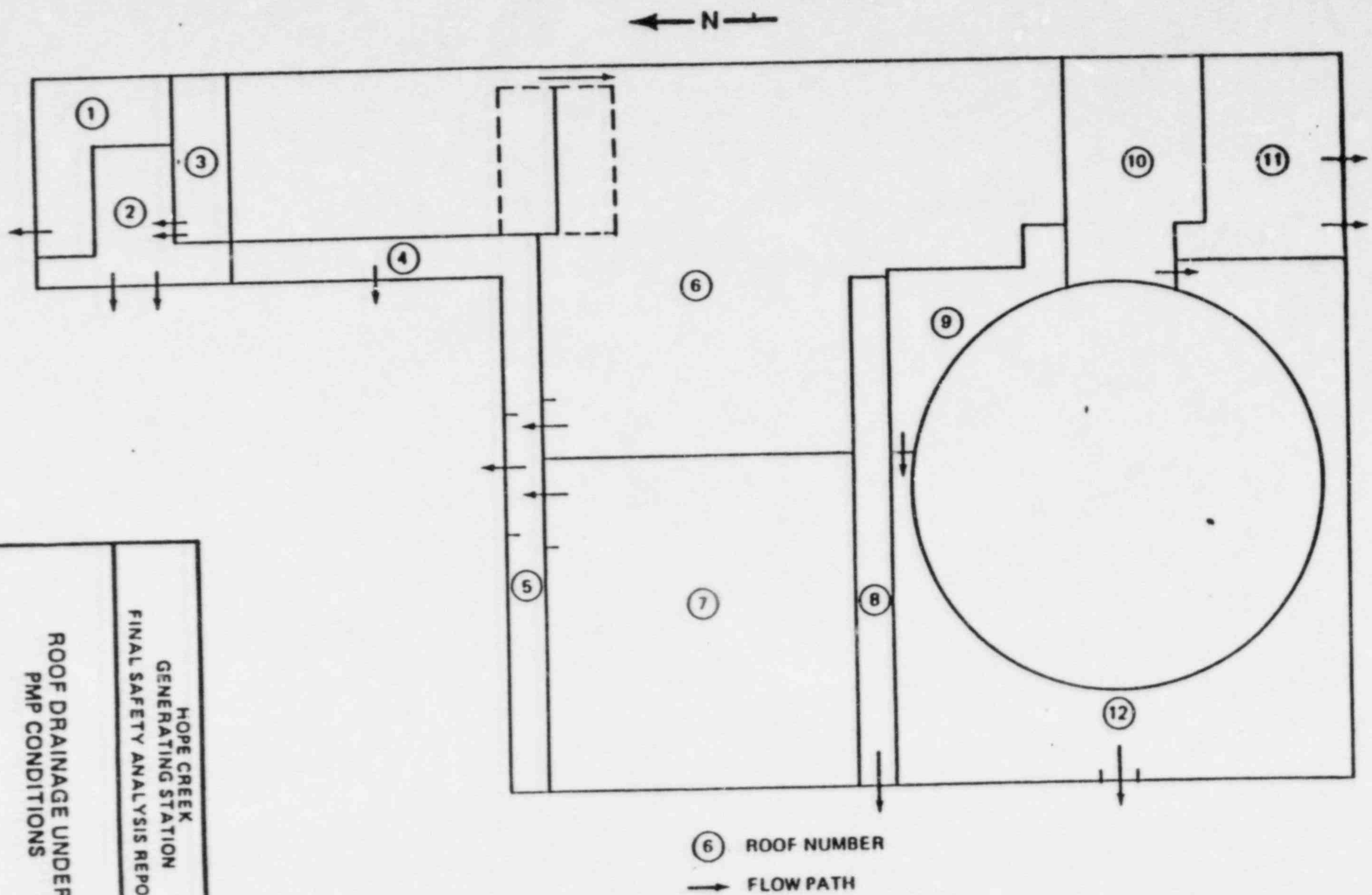
Roof No. (2)	Min. Roof Elevation (ft)	Sub-Drainage Area (ft^2)	Number of 8-inch Diameter Openings	Width of 8-inch High Slot (ft)	Width of Parapet Opening (ft)	Limiting Water Depth Over Roof Drain Elevation (in.)	Max. Water Depth Over Roof Drain Elevation (in.)
1	159	2720	-	2.5	-	12.0	11.5
2	137	2570	2	-	-	28.8	18.0
3	172	1530	2	-	-	15.0	13.6
4	153	1930	1	-	-	28.8	16.1
5	155.25	3700	-	-	50	12.0	11.9
6	172	38850	-	-	25	13.0	12.6
7	198	18420	-	-	35	10.0	9.8
8	155.25	3490	-	3.0	-	12.0	11.7
9	158.33	7380	-	2.5	-	19.0	18.1
10	172	5220	1	0.83	-	18.0	15.8
11	124	5030	2	-	-	18.0	17.5
12	132	33500	-	-	14	18.0	17.6

Notes:

1. The invert elevation of openings and the crest elevation of slots and parapet openings are 6 inches above the roof drain elevation.

2. See Figure 3.4-4.

DSER OPEN ITEM 4



HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

ROOF DRAINAGE UNDER
PMP CONDITIONS

FIGURE 3.4-4

DSER Open Item No. 6C (DSER Section 2.4.15)

PONDING LEVELS

Based on its review and analysis as described above, the staff has requested the applicant to provide additional information on roof ponding levels due to intense local precipitation (PMP), flood protection for the service water intake structure and power block, and flood protection structures adjacent to the intake structure. Until the applicant provides the additional information, the staff cannot conclude that the plant meets the requirements of GDC 2 with respect to flooding. The staff also cannot conclude that the plant meets the hydrologic criteria of GDC 44 with respect to the thermal aspects of the UHS.

Response

The response to this item is provided in the response to Question 240.13. Response to Question 240.13 is attached to DSER Open Item No. 4.

DSER Open Item No. 7 (DSER Section 2.4.11.2)

THERMAL ASPECTS OF ULTIMATE HEAT SINK

The applicant has analyzed the ability of the service cooling water supply to withstand the effect of such severe natural phenomena as ice blockage, flooding, low water, and thermal aspects of UHS. As indicated in Section 2.4.7, the effects of ice blockage would not obstruct the flow to safety-related pumps. Thus the staff concludes that the intake structure and essential service cooling water flow is adequately protected against ice effects. As indicated in Section 2.4.5, the ability of the service water intake structure to withstand the effects of PMH surge flooding and associated wave runup and overtopping remains an open item.

The applicant reported that the minimum historical low water level at the Reedy Point, Delaware, tide station is -8.6 ft msl. The applicant's analysis of the maximum setdown considered the PMH wind speed of 85 mph (the overland PMH wind speed for the direction resulting in maximum setdown) to be blowing down the estuary coincident with 10% exceedance low spring astronomical tide of -3.9 ft msl and the associated trough of the 6.0 ft maximum wind wave. The resultant low water level would be -13.0 ft msl. The applicant has stated that -13.0 ft msl is the design basis minimum low water level for service water pumps. Based on its independent analysis, the staff concurs that -13.0 ft msl is an appropriate design basis minimum low water level. The applicant has not identified the maximum intake temperature that will allow the plant to safely shut down under normal and emergency conditions as discussed in Regulatory Guide 1.27 nor the ability of the Delaware River to supply water below this temperature. Until this information is available, the staff cannot conclude that the plant meets GDC 44 with respect to the thermal aspects of UHS.

Based upon the evaluation described above, we conclude the hydrologic characteristics of the Ultimate Heat Sink meet the requirements of 10 CFR Part 100 and 10 CFR Part 100, Appendix A. As indicated above, certain aspects related to flooding level for the service water intake structure are unresolved. Therefore, the staff cannot conclude that the Ultimate Heat Sink System meets the requirements of General Design Criterion 2 with respect to hydrologic characteristics. In addition, the staff cannot conclude that the Ultimate Heat Sink meets the requirements of GDC 44 with respect to thermal aspects of the heat transfer system.

RESPONSE

For information on the ability of the service water intake structure to withstand the effects of PMH surge flooding and associated wave runup and overtopping, see the response to DSER Open Item Number 5.

The maximum intake temperature that will allow the plant to safely shut down under normal and emergency conditions is discussed in the response to FSAR Question 240.15.

QUESTION 240.15 (Section 2.4.11.6)

Identify the maximum temperature of the intake water that will allow the plant to safely shut down under normal and emergency conditions and discuss the ability of the Ultimate Heat Sink to supply service cooling water below this maximum intake temperature.

Response

Section 9.2.5.2 has been revised to provide this information.

9.2.5.2 System Description

The UHS is the Delaware River, which provides the source of cooling water to the SACS heat exchangers through the SSWS, as shown on Figure 9.2-1. The SACS, in turn, provides demineralized cooling water in a closed loop to the ESF components. The water from the SSWS is discharged into the CWS to provide makeup for that system.

Details of the safety-related and nonsafety-related systems and heat load dissipation are discussed in the following sections:

a. SSWS and intake structure - Section 9.2.1

b. Circulating water and cooling tower - Section 10.4.5

c. SACS - Section 9.2.2.

A discussion of Delaware River water temperatures is provided in the Hope Creek Generating Station Operating License Stage-Environmental Report.

ADD PARA FROM NEXT PAGE →

9.2.6 CONDENSATE AND REFUELING WATER STORAGE AND TRANSFER SYSTEM

9.2.6.1 Design Bases

The condensate and refueling water storage and transfer system has no safety-related function, except for that of supplying condensate to the suction line of the high pressure coolant injection (HPCI) and the reactor core isolation cooling (RCIC) pumps. The system is designed to perform the following functions:

DSER OPEN ITEM 7

a. Supply water to fill the reactor well, the dryer/separator storage pool via the reactor well, and the spent fuel cask storage pool during refueling operations, and provide storage for this water when refueling is completed

THE PLANT MAY BE SHUT DOWN UNDER NORMAL CONDITIONS WITH AN AVERAGE INTAKE WATER TEMPERATURE AS HIGH AS 91.6 °F. THE PLANT MAY BE SAFELY SHUT DOWN UNDER EMERGENCY CONDITIONS WITH AN AVERAGE INTAKE WATER TEMPERATURE AS HIGH AS 90.5 °F. THIS MAXIMUM TEMPERATURE FOR EMERGENCY SHUTDOWN IS AN EXTREME CONDITION FOR DESIGN PURPOSES. IT IS EXPECTED THAT THE AVERAGE RIVER WATER TEMPERATURE WILL RARELY, IF EVER, REACH THIS MAXIMUM.

DSER OPEN ITEM 7

Rev 1

<u>DSER Open Item No. 29</u> (Section 3.5.1.1)

INTERNALLY GENERATED MISSILES (OUTSIDE CONTAINMENT)

With respect to rotating equipment, the applicant has stated that the pumps and fans were manufactured to the same industry standards as Palo Verde and therefore the results of the Palo Verde's analysis for internally generated missiles is applicable to Hope Creek. In order to rely upon the analysis performed by Palo Verde, the applicant must verify that every rotating component (pumps, fans, motors, and turbines, except the main turbine-generator) is designed and constructed to exactly the same codes and standards (including addenda and editions), to be of the same manufacturer, size, and materials as the analyzed components at Palo Verde. Palo Verde relied mainly upon compartmentalization as the means to protect the redundant equipment. For each component where compartmentalization was relied upon at Palo Verde, the applicant must verify the identical components at Hope Creek provided with comparable compartmentalization.

Similarly, the applicant must verify the use of barriers, separation and orientation as was used by Palo Verde. For every component which is not identical with Palo Verde, the applicant must provide a discussion of the analysis which verifies that the casing would be capable of retaining the internally generated missile or that the missile would not strike safety-related components or generate a secondary missile. Unless the applicant either verifies comformance with the Palo Verde design (as outlined above) or provides the results of an analysis which shows that the casings will contain the internally generated missiles, the applicant must provide protection by any one or a combination of compartmentalization, barriers, separation, orientation, and equipment design. Safety-related systems must be verified to be physically separated from nonsafety-related systems and components of safety-related systems are physically separated from their redundant compartments.

Based on the above, we cannot conclude that the design is in conformance with the requirements of General Design Criterion 4 as it relates to protection against internally generated missiles until the applicant provides an acceptable discussion concerning rotating components as potential sources of internally generated missiles. We cannot determine that the design of the facility for providng protection from internally generated missiles meets the applicable acceptance criteria of SRP Section 3.5.1.1. We will report resolution of this item in a supplement to this SER.

RESPONSE

FSAR Section 3.5.1.1 has been revised to include the results of an analysis of the internally generated rotational missiles outside containment.

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CHAPTER 3

TABLES

CHAPTER 3

TABLES (cont)

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3.5 MISSILE PROTECTION

The Seismic Category I and safety-related structures, equipment, and systems are protected from postulated missiles through basic plant arrangement so that a missile does not cause the failure of systems that are required for safe shutdown or whose failure could result in a significant release of radioactivity. Where it is impossible to provide protection through plant layout, suitable physical barriers are provided to shield the critical system or component from credible missiles. Redundant safety-related Seismic Category I components are arranged so that a single missile cannot simultaneously damage a critical system component and its backup system.

A tabulation of safety-related structures, systems, and components, their locations, seismic category, quality group classification, and the applicable FSAR sections is given in Table 3.2-1. General arrangement drawings are included as Figures 1.2-2 and 1.2-41.

3.5.1 MISSILE SELECTION AND DESCRIPTION

3.5.1.1 Internally Generated Missiles (Outside Primary Containment)

The systems located outside the primary containment have been examined to identify and classify potential missiles. These systems and missiles are listed in Tables 3.5-1 and 3.5-13. Redundant systems are normally located in different areas of the plant or separated by missile-proof walls so that a single missile can not damage both systems.

~~Large pumps, such as~~ The residual heat removal (RHR) and core spray pumps, are located in separate missile-proof compartments, and their impellers are enclosed in a concrete structure therefore ~~and~~ are not considered a potential missile source or hazard to other systems.

Refer to Section 3.5.3 for barrier design procedure.

There are three general sources of postulated missiles:

a. Rotating component failure

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b. Pressurized component failure

c. Gravitationally generated missiles.

3.5.1.1.1 Rotating Component Failure Missiles

probable

Catastrophic failure of rotating equipment having synchronous motors, e.g., pumps, fans, and compressors, that could lead to the generation of missiles is not considered ~~credible.~~ Massive and rapid failure of these components is improbable because of the conservative design, material characteristics, inspections, and quality control during fabrication and erection. Also, the rotational speed is limited to the design speed of the motor, thereby precluding component failures due to runaway speeds.

Similarly, it is concluded that the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) pumps and turbines cannot generate credible missiles. These pumps are not in continuous use, but are periodically tested and otherwise operate only in the unlikely event of a postulated accident. They are classified as moderate energy systems. Overspeed tripping devices ensure that the turbines do not reach runaway speed, where failure leading to the ejection of a missile could take place.

~~Other rotating equipment does not constitute a missile hazard because of its small size and/or the unlikelihood that its rotating components would penetrate its housing.~~

Insert 1

3.5.1.1.2 Pressurized Component Failure Missiles

The following are potential internal missiles from pressurized equipment:

a. Valve bonnets

b. Valve stems

c. Temperature detectors

d. Nuts and bolts

INSERT 1

A tabulation of missiles generated by postulated failures of rotating components, their sources and characteristics, and a safety evaluation are provided in Table 3.5-13.

The evaluation identified one instance where a postulated missile, which could penetrate through the flexible connection of a vane-axial fan, could have the potential to damage safe-shutdown equipment in the room. In order to prevent the postulated missile from damaging safety-shutdown equipment, a missile shield has been added to the design to withstand the impact of the postulated fan blade missile.

The formulas used to predict the penetration resulting from missile impact are provided in Reference 3.5-4. The penetration and perforation formulas assume that the missile strikes the target normal to the surface, and the axis of the missile is assumed parallel to the line of flight. The rotating components is assumed to fail at 120 percent overspeed. These assumptions result in a conservative estimate of local damage to the target.

TABLE 3.5-1 PRESSURIZED COMPONENT

INTERNALLY GENERATED MISSILES OUTSIDE CONTAINMENT

System	FSAR Section	Missile Description	Protection Evaluation Codes(1)
HPCI	6.3	Test connection	c
		Startup flange	c
		Pressure indicator (PI-R003)	c
CRD hydraulic	4.6.1	Drains	c
		Pressure indicators (PI-R008, 4013 A, B)	c
		Pressure indicators (PI-R021, PI-N005, PI-R016, PI-R012, PI-R007, PI-R010, PI-R006)	c
		Test indicators (TI-4014, TE-4014, TE-N018)	c
		Test connections	c
		Vent	d
		Blind flange	c
Main steam	5.1	Test connections	c
		Temperature elements (TE-N040)	c
		Pressure indicators (PP-3632 A, B, C, D)	c
Main steam sealing	5.1	Temperature elements (TE-N057 A, B, C, D, E)	c
		Pressure transmitter (PT-5838)	c
		Blind flange or Y-strainer	c
		Test connection	c
		Temperature element (TE-N060)	c
Feedwater	5.1	Test connection	c
RWCU	5.4.8	Blind flange	c
		Temperature sensors/elements (TE-N007, TE-N019, TE-N015, TE-N004, TS-169, TS-170, TS-242 A, B)	c
		Pressure transmitter (PT-N005)	c
		Pressure point (PP-3876 A, B; PP-3875 A, B; PP-3916 A, B; PP-3917 A, B)	c
RWCU	5.4.8	Pressure indicators (PI-3377 A, B; PI-R009; PI-R004; PI-R008; PDIS-3987 A, B; PDIS-3988 A, B)	c
		Pressure switches (PSL-N013, PSH-N014)	c
		Flow elements (FE-3986 A, B)	c

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INTERNALLY GENERATED ROTATING COMPONENT MISSILES OUTSIDE CONTAINMENT

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS (IN.)	REMARKS
Fan Blade	Containment Pre-purge Cleanup Fan 10V-200 (Centrifugal Fan)	Reactor Bldg El. 162'	199.0	1.21	3.7	0.211	0.1406	Fan blade may penetrate fan casing. The surrounding concrete wall for the fan is 12" thick. The calculated depth of fan blade penetration into the concrete wall is 1.43". Therefore, missile has no effect on plant safe shutdown capability. Therefore protection is not needed.
Fan Blade	Diesel Generator Wing Area Exhaust Fan 1A, B-V414 (Centrifugal Fan)	Aux Bldg SDG Area El. 178'	116.0	1.24	4.05	0.1066	0.0781	Perforation of fan casing may occur. Due to the orientation of the fans, the postulated fan blade missile will not damage any safe shutdown equipment in the room. Therefore, protection is not needed.(2)
Fan Blade	Control Area Exhaust Fan 1A, B-V02 (Centrifugal Fan)	Aux Bldg Control Area El. 155'	105.0	0.969	0.614	0.034	0.0781	Casing perforation will not occur; however, fan blade may exit through the flexible connector on the fan discharge. There is no safe shut-shutdown equipment in the room.
Fan Blade	FRVS Recir. Fan 1A thru F-V213 (Centrifugal Fan)	Reactor Bldg El. 132', 162', and 178'	248.0	1.4	5.42	0.318	0.1406	Perforation of the fan casing or flexible connector may occur. However, due to the orientation of the fans, only ceiling and floor may be hit. The calculated depth of the fan blade penetration on the concrete is 3.61". Since there are no safe shutdown equipment impacted, protection is not needed.(2)

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS (IN.)	REMARKS
Fan Blade	FRVS Vent Fan 1A, B-V206 (Centrifugal Fan)	Reactor Bldg El. 145'	144.0	1.02	1.99	0.108	0.1406	Casing perforation will not occur; however, fan blade may exit through the flexible connector or the fan discharge. The calculated depth of the fan blade penetration into the concrete is 1.138". Due to the orientation of the fan, only the ceilling and floor could be hit. Therefore, protection is not needed.(1)
Fan Blade	Control Room Emerg. Filter Fan 1A, B-V400 (Centrifugal Fan)	Aux Bldg Control Area El. 155'	197	0.772	0.764	0.115	0.1406	Casing perforation will not occur; however, fan blade may exit through the flexible connector on the fan discharge. The calculated depth of the blade penetration into the concrete is 1.09". There is no safe shutdown equipment in the room. Therefore protection is not needed.
Fan Blade	Battery Room Exhaust Fan 1A thru D-V406 (Centrifugal Fan)	Aux Bldg SDG Area El. 163'	81	0.846	0.23	0.014	0.0625	Casing perforation will not occur; however, fan blade may exit through the flexible connector on the fan discharge. The calculated depth of the fan blade penetration in the concrete is 0.086". Due to orientation of the fan, safe shutdown equipment will not be impacted and protection is not needed.(1)
Fan Blade	Control Area Battery Exhaust Fan 1A, B-V410 (Centrifugal Fan)	Aux Bldg Control Area El. 178'	143	0.834	0.206	0.029	0.0625	Casing perforation will not occur; however, fan blade may exit through the flexible connector on the fan discharge. There are conduits that belong to A, C, and D channels in the room that may be needed for safe shutdown. However, the conduits are thicker than the calculated maximum steel perforation depth (0.029"), therefore, protection is not needed.(1)(2)

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TABLE 3.5-13 (Cont)

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS (IN.)	REMARKS
Fan Blade	Battery Room Exhaust Fan 1A, B-V416 (Centrifugal Fan)	Aux Bldg SDG Area El. 178'	81	0.846	0.23	0.014	0.0625	Casing perforation will not occur; however, fan blade may exit through the flexible connector on the fan discharge. There are conduits that belong to A, C, and D channels in the room that may be needed for safe shutdown. However, the conduits are thicker than the calculated maximum steel perforation depth (0.014"), therefore, protection is not needed. (1)(2)
Fan Blade	Aux Bldg Battery Exhaust Fan 1A, B-V417 (Centrifugal Fan)	Aux Bldg SDG Area El. 178'	78.5	0.984	0.792	0.027	0.0781	Casing perforation will not occur; however, fan blade may exit through the flexible connector on the fan discharge. There are conduits that belong to A, C, and D channels in the room that may be needed for safe shutdown. However, the conduits are thicker than the calculated maximum steel perforation depth (0.027"), therefore, protection is not needed(1)(2)
Fan Blade	Control Equipment Supply Fan 1A, B-VH-407 (Centrifugal Fan)	Aux Bldg SDG Area El. 178'	235	1.68	8.8	0.341	0.25	Perforation of fan casing may occur; however, the fan is inside a filter housing that is 3/16" thick. The calculated steel perforation after the fan blade penetration through the fan casing is 0.176". Therefore, the fan blade will not exit from the filter housing.
Fan Blade	Diesel Generator Panel Supply Unit Fan 1A, B-VH-408 (Centrifugal Fan)	Aux Bldg SDG Area El. 163'	149	1.37	3.16	0.115	0.1875 (filter housing thickness)	Filter housing perforation will not occur.

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MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS (IN.)	REMARKS
Fan Blade	Switchgear Room Unit Coolers 1A, B-VH-401 (Centrifugal Fan)	Aux Bldg SDG Area El. 163'	157	3.31	8.09	0.094	0.1875 (filter housing thickness)	Filter housing perforation will not occur.
Fan Blade	Control Room Supply Unit 1A, B-VH-403 (Centrifugal Fan)	Aux Bldg SDG Area El. 178'	174	1.45	4.867	0.178	0.1875	Casing perforation will not occur. Also, the fan is inside a filter housing.
Fan Blade	Control Area Smoke Vent Fan 10-V408 (Vane-Axial Fan)	Aux Bldg Control Area El. 178'	210	1.37	0.753	0.069	0.1875	Casing perforation will not occur. However, the fan blade may exit through the suction side flexible connector. There is no safe shutdown equipment within the room. Therefore, protection is not needed.
Fan Blade	Diesel Area Exhaust Fan 1A, B-V411 (Vane-Axial Fan)	Aux Bldg SDG Area El. 178'	281	1.72	0.902	0.092	0.1875	Casing perforation will not occur. However, the fan blade could exit through the suction side flexible connector. A 1/4" thick steel barrier is provided to enclose the section flexible connector.(2)
Fan Blade	Diesel Generator Room Recir. Fan 1A thru H-V812 (Vane-Axial Fan)	Aux Bldg SDG Area El. 77'	260	3.33	23.9	0.383	0.25	Fan blade will penetrate through the fan casing. However, there are no safe shutdown equipment in the room. Therefore protection is not needed.

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MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS (IN.)	REMARKS
Fan Blade	Control Room Return Air Fan 1A, B-VH-415 (Vane-Axial Fan)	Aux Bldg Control Area El. 155'	362	1.26	0.72	0.151	0.1719	Casing perforation blade will not occur. However, the fan ~~blade~~ may exit through the suction flexible connector. There are no safe shutdown equipment in the room. Therefore, protection is not needed.
Fan Blade	RCIC Room Coolers 1A, B-VH-208 (Vane-Axial Fan)	Reactor Bldg El. 54'	205	1.36	0.758	0.0684	0.1875	Casing perforation will not occur. There is a wire screen on the suction of the Fan Cooler which will prevent a fan blade from leaving the cooler at an oblique angle. insert 3
Fan Blade	RHR Room Coolers 1A thru H-VH-210 (Van-Axial Fan)	Reactor Bldg El. 54'	281	2.12	4.59	0.220	0.25	Casing perforation will not occur. There is a wire screen on the suction of the fan cooler which will prevent a fan blade from leaving the cooler at an oblique angle. insert 4
Fan Blade	SACS Room Coolers 1A thru D-VH-214 (Vane-Axial Fan)	Reactor Bldg El. 102'	215	1.46	1.05	0.084	0.1875	Casing perforation will not occur. There is a wire screen on the suction of the fan cooler which will prevent a fan blade from leaving the cooler at an oblique angle. insert 4
Fan Blade	Core Spray Pump Room Coolers 1A thru H-VH-211 (Van-Axial Fan)	Reactor Bldg El. 54'	230	1.61	1.598	0.11	0.1875	Casing perforation will not occur. There is a wire screen on the suction of the fan coolers which will prevent a fan blade from leaving the cooler at an oblique angle. insert 4
Fan Blade	HPCI Room Coolers 1A,B-VH-209 (Van-Axial Fan)	"	"	"	"	"	"	"

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TABLE 3.5-13 (Cont)

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS (IN.)	REMARKS
Fan Blade	Intake Structure Supply Fan 1A thru D-V503 (Van-Axial Fan)	Intake Structure El. 122'	250	2.72	8.49	0.22	0.25	Casing perforation will not occur. There is a wire screen on the suction of the fan cooler which will prevent a fan blade from leaving the cooler at an oblique angle. insert 3
Fan Blade	Intake Structure Exhaust Fan 1A thru D-V504 (Vane-Axial Fan)	Intake Structure El. 122'	250	2.72	8.49	0.22	0.25	Casing perforation will not occur. There is no flexible connector on the suction or the discharge side. insert 4
Fan Blade	Traveling Screen Motor Room Fan OA, B-V558 (Van-Axial Fan)	Intake Structure	138	1.368	0.746	0.04	0.1875	Casing perforation will not occur. The intake damper and vane guide on the suction of the fan prevents a fan blade from exiting in that direction and the vane guide on the discharge of the fan prevents a fan blade from leaving the fan housing on the discharge direction. Therefore, protection is not needed.
Impeller	SACS Pumps	Reactor Bldg El. 102'	98.8	16.1	1016.	0.267	0.625	No casing perforation.
Impeller	Fuel Pool Cooling Pump	Reactor Bldg El. 162'	121.6	5.3	46.4	0.136	0.59	No casing perforation.
Impeller	ECCS Jockey Pump	Reactor Bldg El. 54'	93.0	2.56	8.35	0.0529	0.43	No casing perforation.
Impeller	Torus Water Cleanup Pump	Reactor Bldg El. 54'	119.9	5.3	44.6	0.132	0.59	No casing perforation.

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS (IN.)	REMARKS
Impeller	Chilled Water Pump	Aux Bldg Control Area El. 155'	82.8	5.97	79.75	0.104	0.63	No casing perforation.
Impeller	D/G 1E Panel Chilled Water Pump	Aux Bldg Diesel Area El. 178'	94.5	3.04	11.79	0.068	0.39	No casing perforation.
Impeller	RACS Pump	Reactor Bldg El. 77'	79.6	6.24	83.66	0.096	0.77	No casing perforation.
Impeller	Service Water Booster Pump	Intake Structure El. 79'-8"	67.3	3.78	26.3	0.0596	0.51	No casing perforation.
Impeller	Service Water Pump	Intake Structure El. 93'	97.6	15.2	1215.5	0.314	0.75	No casing perforation.
Impeller	RWCU Recir. Pump	Reactor Bldg El. 132'	158.1	4.79	48.2	0.219	1.125	No casing perforation.
# Impeller	RWCU Precoat Pump	Reactor Bldg El. 145'	62.4	4.31	31.8	0.053	0.5	No casing perforation.
Impeller	RWCU Holdup Pump	Reactor Bldg El. 145'	57.6	4.09	25.6	0.0429	0.801	No casing perforation.
Impeller	RWCU Backwash Pump	Reactor Bldg El. 132'	70.4	4.04	15.9	0.0423	0.43	No casing perforation.
Impeller	CRD Pump	Reactor Bldg El. 77'	120.6	3.91	21.4	0.109	0.675	No casing perforation.

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TABLE 3.5-13 (Cont)

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBS)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS
Impeller	Service Water Dewater Pump	Reactor Bldg El. 54'	64.1	3.98	13.94	0.0347	0.5	No casing perforation
Impeller	RCIC Pump	Reactor Bldg El. 54'	168.8	4.04	29.3	0.204	0.5	No casing perforation.
Impeller	HPCI Booster Pump	Reactor Bldg El. 54'	169.9	3.77	55.82	0.339	0.625	No casing perforation.
Impeller	HPCI Main Pump	Reactor Bldg El. 54'	224	2.65	18.2	0.330	0.687	No casing perforation.
~~(Later)[2]~~ N/A	HPCI Turbine	Reactor Bldg El. 54'	~~(Later)[2]~~ N/A	~~(Later)[2]~~ N/A		~~(Later)[2]~~ N/A	~~(Later)[2]~~ N/A	~~(Later)[2]~~ insert 1
~~(Later)[2]~~ N/A	RCIC Turbine	Reactor Bldg El. 54'	~~(Later)[2]~~ N/A	~~(Later)[2]~~ N/A		~~(Later)[2]~~ N/A	~~(Later)[2]~~ N/A	~~(Later)[2]~~ insert 2
~~(Later)[2]~~ Flywheel	RPS MG Sets	Aux. Bldg Control Area El. 54'	~~(Later)[2]~~ 183.4	~~(Later)[2]~~ 12.5	283.33	~~(Later)[2]~~ 0.348	~~(Later)[2]~~ 0.094 (guard plate thickness)	~~(Later)[2]~~ insert 3

(1) The conduits have a maximum fill volume of 53 percent. If a missile should strike the conduit, crimping of the cable is improbable because there is sufficient volume in the conduit to accommodate the cables.

(2) The missile interaction evaluation will be confirmed by a field walk-down.

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Table 3.5-13 p8

Insert 1

The manufacturer of the turbine has performed an evaluation of the missile generation capability of the turbine. This evaluation demonstrates that it is not possible for the turbine rotor to achieve burst speed using saturated steam. Therefore, protection is not needed.

Insert 2

The post[illegible]ed missile is a 120° section of the flywheel [illegible] missile could perforate the guard plate and penetrate 2.3" into the surrounding concrete structure. There is no perforation or generation of spalling missiles caused by this impact.

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Insert 3

However, the postulated missile may exit through the inlet screen. The room does not contain any safe-shutdown equipment. Therefore, protection is not needed.

Insert 4

However, the postulated missile may exit through the inlet screen. There are no missile interactions which could prevent safe-shutdown of the plant. (2)

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QUESTION 410.11 (SECTION 3.5.1)

The FSAR states that fans are not considered as credible missile sources. Recently (Palo Verde, 1982) a fan at a nuclear facility generated a missile which penetrated the fan housing and damaged a safety-related structure. Provide a discussion of the effects of fan blades as a missile source and the means used to prevent damage of safety-related equipment for each fan.

RESPONSE

Insert Delete

~~As discussed in the HCGS response to Question 410.12, we do not consider through-fan-housing missiles that would damage safety-related structures to be credible. The condition that existed at Palo Verde involved workmanship deficiencies as the blade locknut torque and blade tip angle did not meet the supplier's specification. As a result, the blade experienced fatigue failure and was ultimately propelled out of the fan housing at an angle that penetrated the flexible connections of the fan and impinged the containment liner plate. HCGS has conducted a survey of vane-axial and centrifugal fans in safety-related areas employing flexible connectors. We identified one instance where a postulated missile through the flexible connection of a vane-axial fan may have the potential to damage safe-shutdown equipment in the room. In order to prevent the postulated missile from damaging safe-shutdown equipment, a missile shield has been added to the design to withstand the impact of the postulated missile.~~

INSERT

Section 3.5 has been revised to provide the results of an analysis which shows that internally generated rotating component missiles have no adverse effect on plant safe shutdown capability.

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QUESTION 410.12 (SECTION 3.5.1)

The FSAR states that rotating equipment which is not specifically identified does not constitute a missile hazard because of the "unliklihood" that a missile would penetrate the casing. Provide the results of a quantitative analysis to verify this conclusion.

RESPONSE

INSERT

~~The possibility that any pump or fan other than those identified in Section 3.5.1, will fail at HCGS and generate a missile which has sufficient energy to penetrate a component casing is remote. A review of the analyses of internally generated missiles performed for Palo Verde verified that postulated missiles from pumps and fans (e.g., a pump impeller or fan blade) typically do not have sufficient energy to penetrate the component casings. The formulae used by Palo Verde to predict the penetration resulting from missile impact are provided in Reference 3.5-4.~~ Dele

~~Since HCGS uses pumps and fans which are designed and constructed in accordance with the same recognized industry codes and standards as those installed at Palo Verde, results of the rigorous analyses conducted for Palo Verde are indicative of the structural integrity of the HCGS equipment.~~

INSERT

Section 3.5 has been revised to provide the results of an analysis which shows that internally generated rotating component missiles have no adverse effect on plant safe shutdown capability.

QUESTION 410.13 (SECTION 3.5.1)

Provide a discussion of an analysis for each rotating component which verifies that the casing would be capable of retaining an internally generated missile. For each rotating component whose casing cannot retain the internally generated missile, verify that no secondary missiles will be generated from any internally generated missile.

RESPONSE

Section 3.5 has been revised to provide the results of an analysis which shows that internally generated rotating component missiles have no adverse effect on plant safe shutdown capability. Secondary missiles are not considered within the affected compartment because their effects are bounded by the primary missiles. Where compartment walls, floors, or ceilings are impacted, the potential for generating secondary (spalling) missiles was included in the evaluation.

DSER Open Item No. 39 (DSER Section 3.7.2#3)

SSI analysis results using finite element method and elastic half-space approach for containment structure

Compare the results of the finite element soil-structure interaction analysis and the impedance soil-structure interaction analysis of the containment structure.

Response

For the information requested above, see the response to DSER open item 51.

DSER Open Item No. 40 (DSER Section 3.7.2.3)

SSI analysis results using finite element method and elastic half-space approach for intake structure

Response

For the information requested above, see the response to DSER open item 66.

DSER Open Item No. 51 (DSER Section 3.8.6)

Comparison of Bechtel independent verification results with the design-basis results.

RESPONSE

This item corresponds to item A.13 from the NRC structural/Geotechnical meeting of January 10, 1984. A comparison of Bechtel independant verification results with the design basis results is attached.

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 basi 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.

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.

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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. 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 exceeds the design basis moments at a few wall locations as indentified 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 approa

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2. (Cont'd)

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 acceleration exceeded the impedance approach spectral acceleration 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 tha 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' 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.

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<u>**Table A-13-1**</u>

Comparison of Design basis and Independent
<u>Finite Element Verification Response Spectra</u>

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)

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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 reconcile 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 spectr (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 observe 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 structure are superimposed on their corresponding design basis curves to identify those impedance curves which are not enveloped by desi 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 envelop 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 selec 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 attem to include response of all structures in the sample selecte

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.

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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.

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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	Applicab. 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

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TABLE A-13-6 (Cont'd)

POWER BLOCK SEISMIC CATEGORY I EQUIPMENT

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

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

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TABLE A-13-6 (Cont'd)

POWER BLOCK SEISMIC CATEGORY I EQUIPMENT

Equipment or Component	Tag No.	Location Bldg./El.	Equipment Frequencies (Hz)	Method of Seismic Qualification	Applic. Note
Standby Diesel Generator Set	1(A-D)G 400	D. G. El. 102	Horizontal - >15 Vertical - >15	Analysis	2
SACS Heat Exchanger	1A1E, 1A2E201 1B1E, 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. Nozzl Allowables M
				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
Drywall	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

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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

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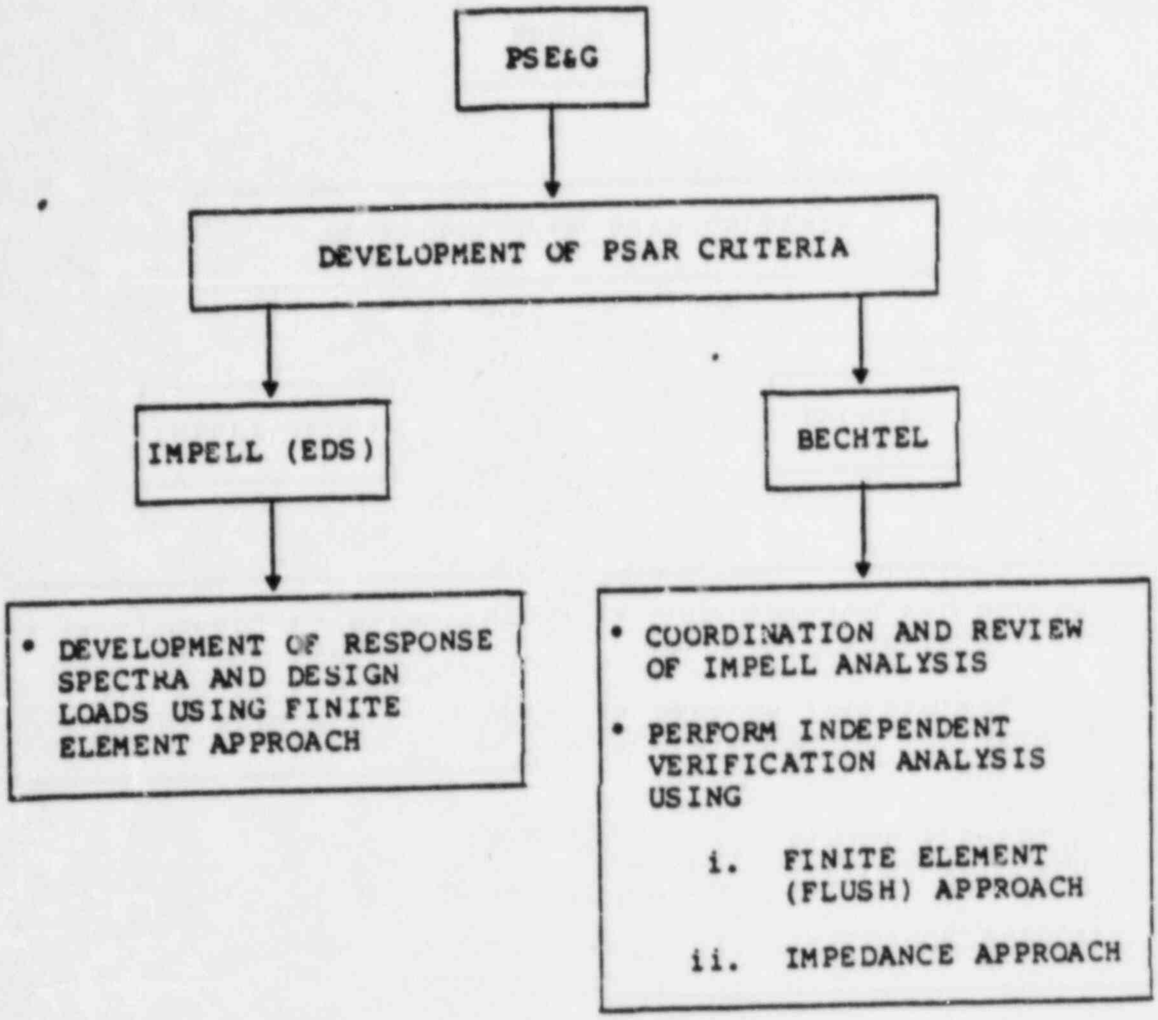

Figure A-13-1

Division of Responsibility

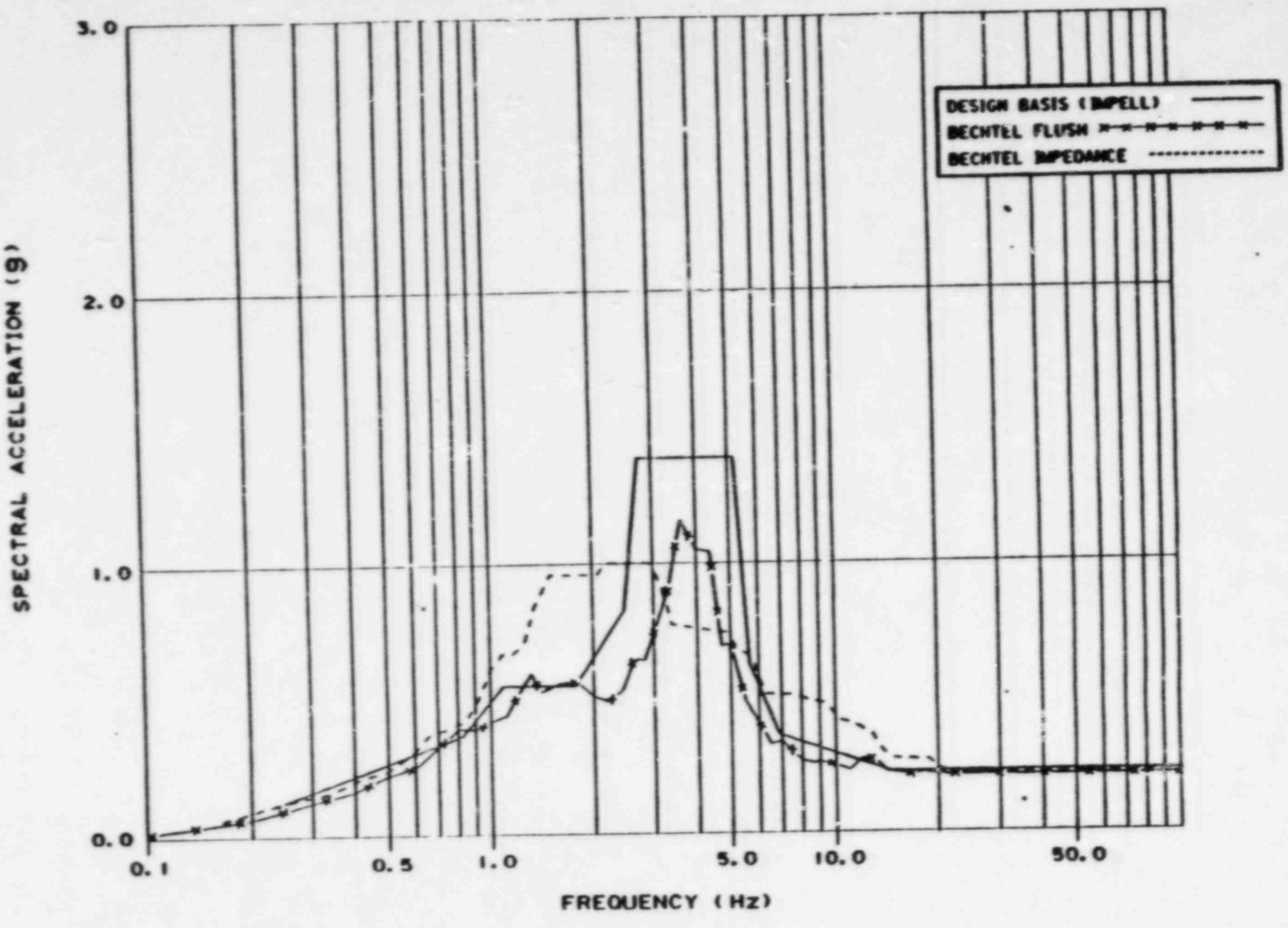

FIGURE A-13-2

RESPONSE SPECTRA COMPARISON.

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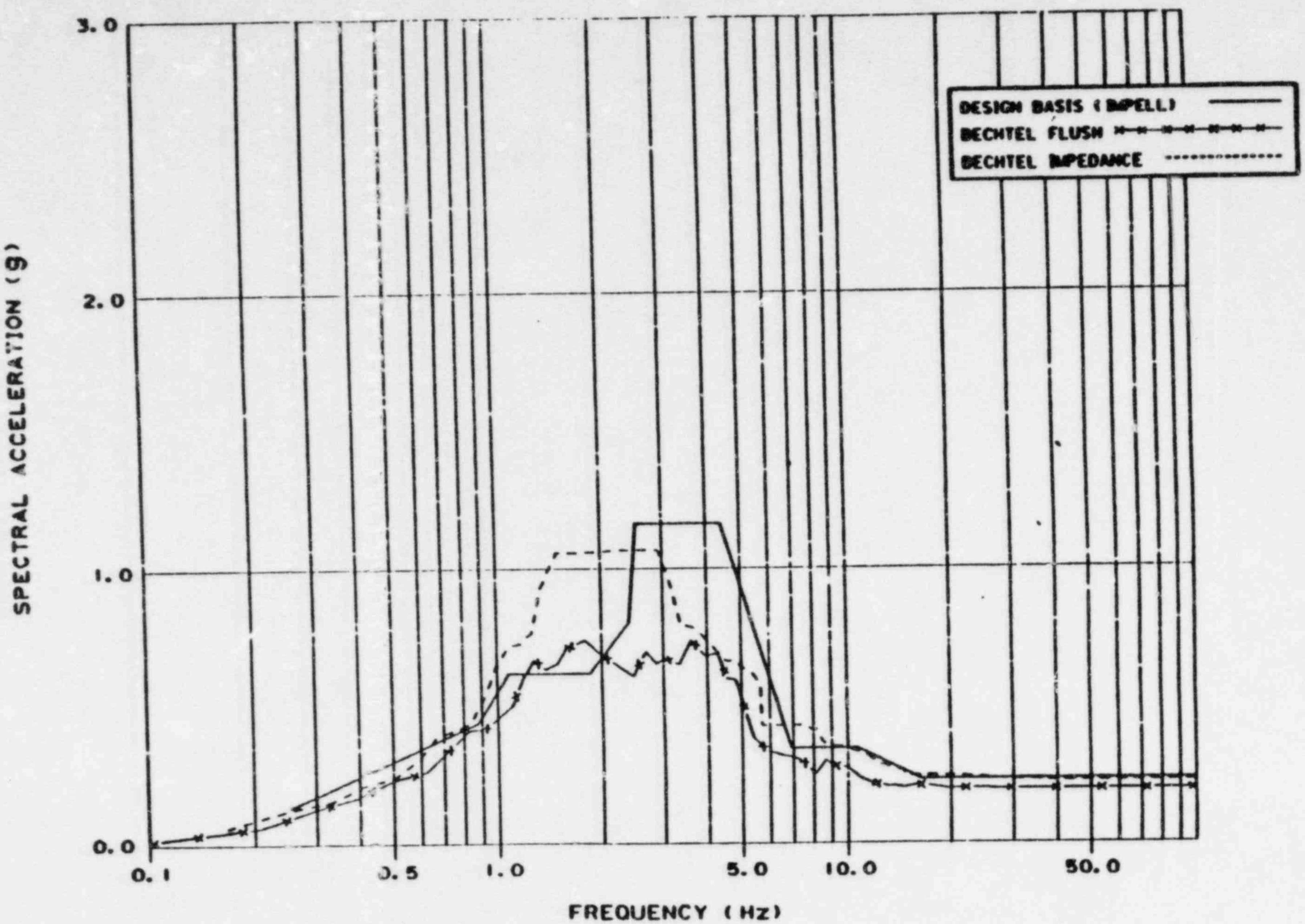

FIGURE A-13-3

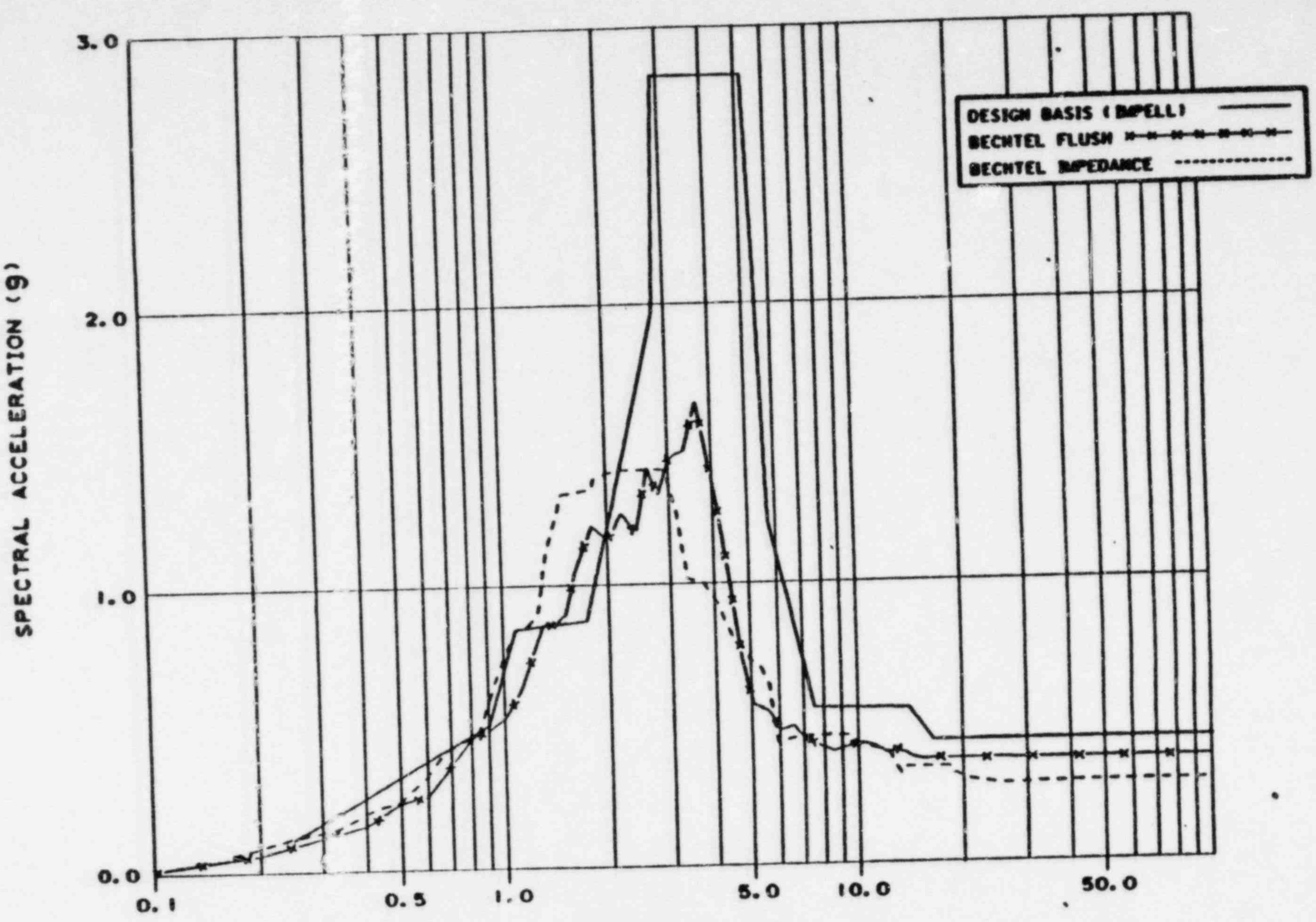

FIGURE A-13-4

RESPONSE SPECTRA COMPARISON

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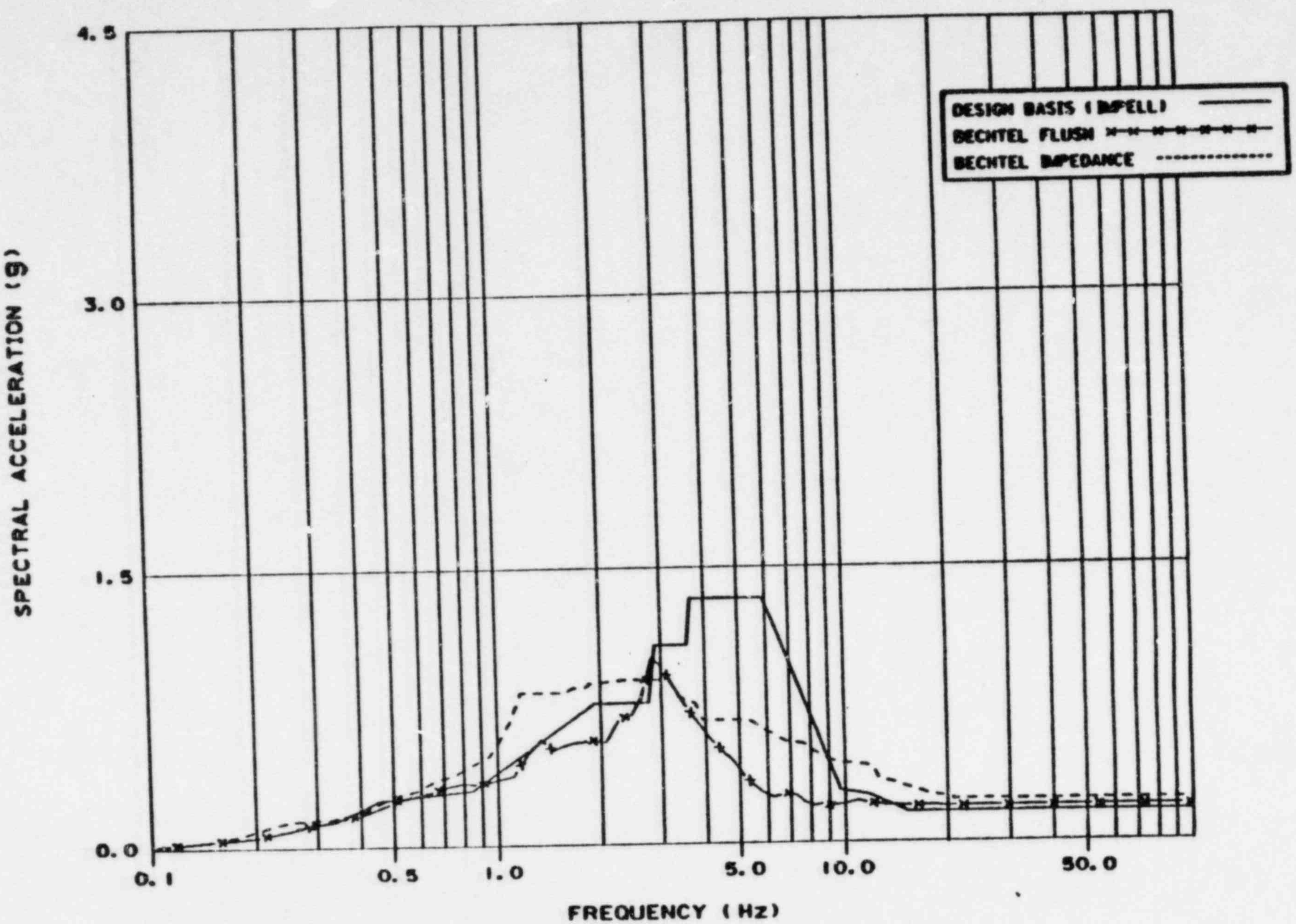

FIGURE A-13-5

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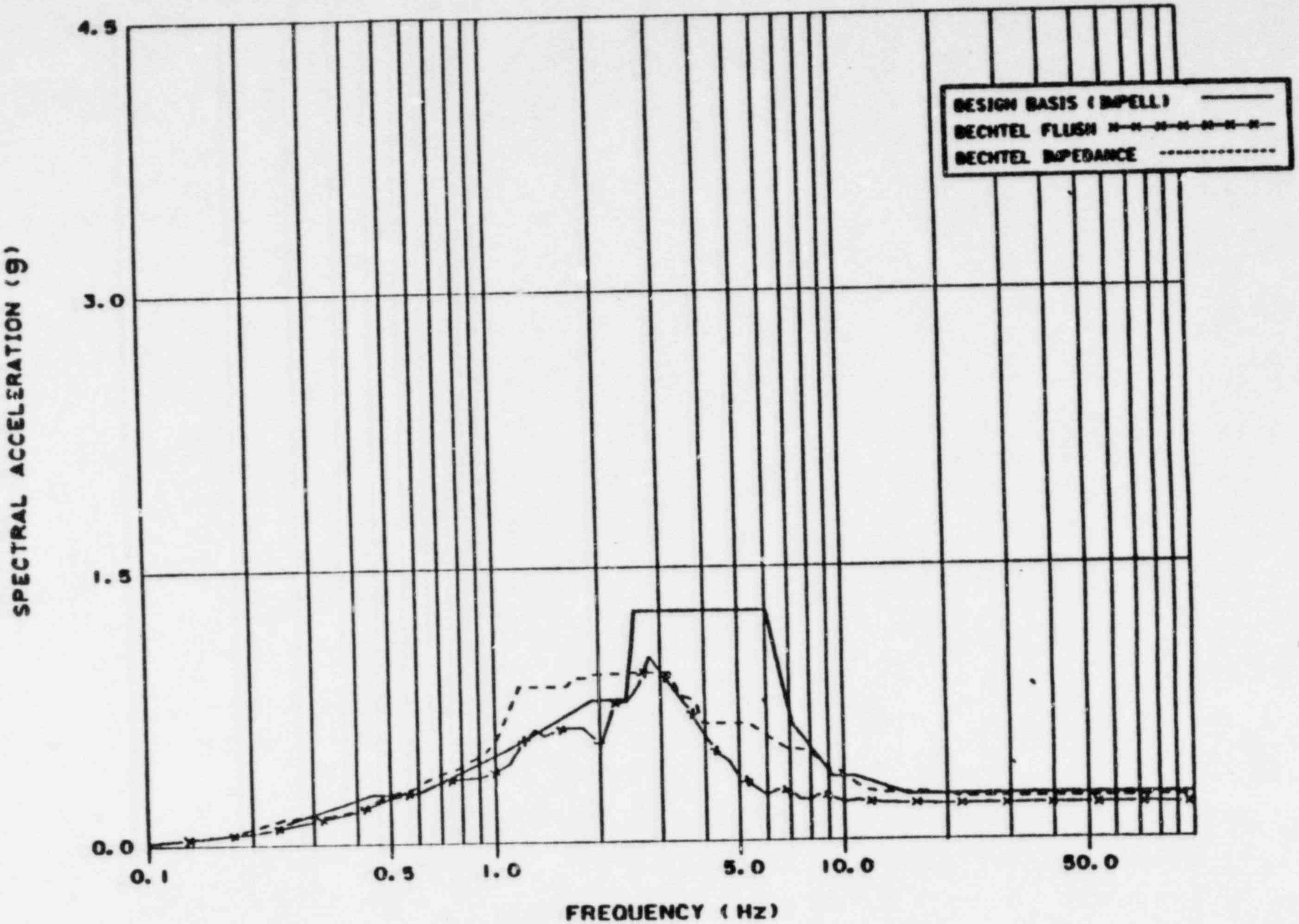

FIGURE A-13-6

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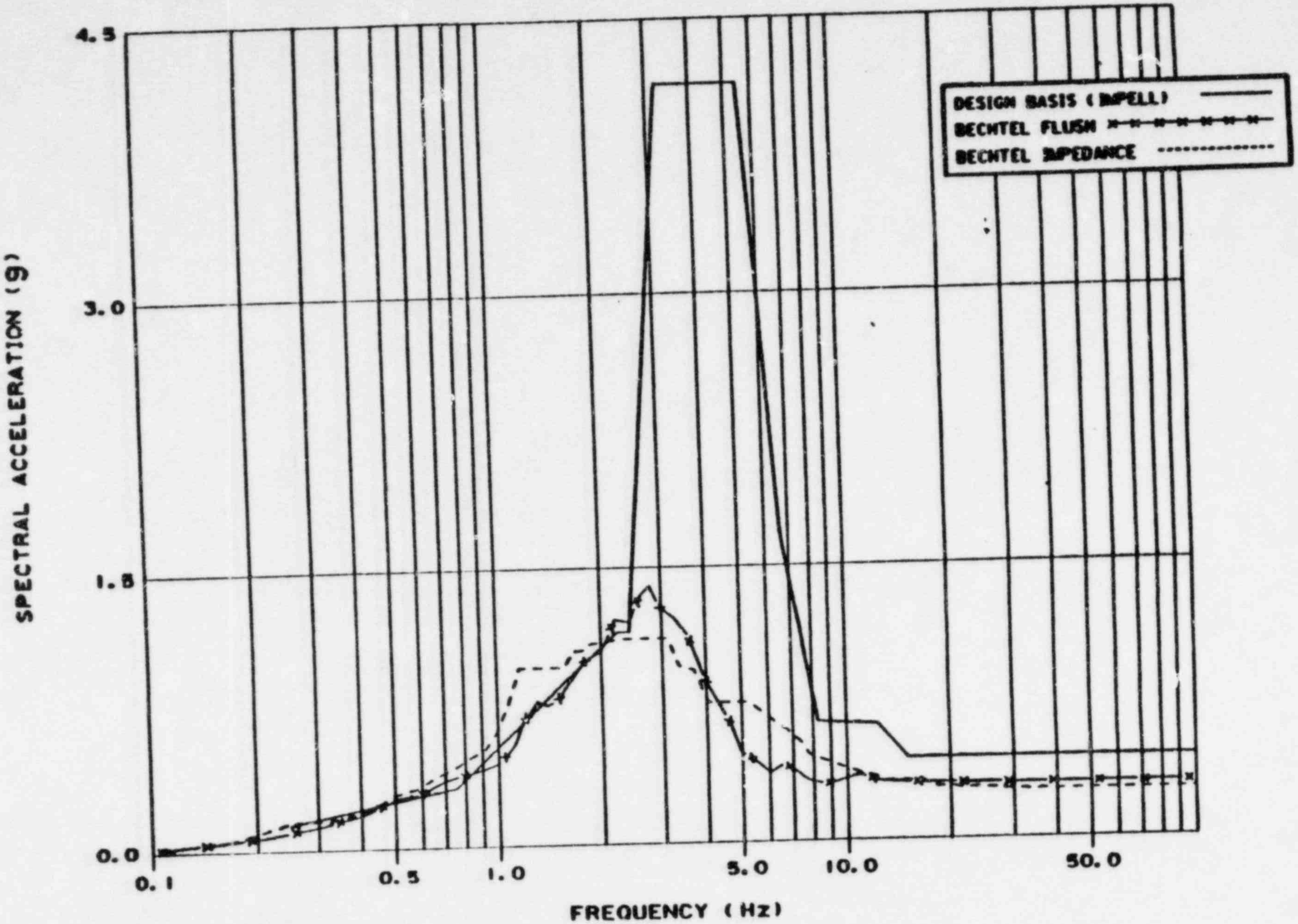

FIGURE A-13-7

RESPONSE SPECTRA COMPARISON.

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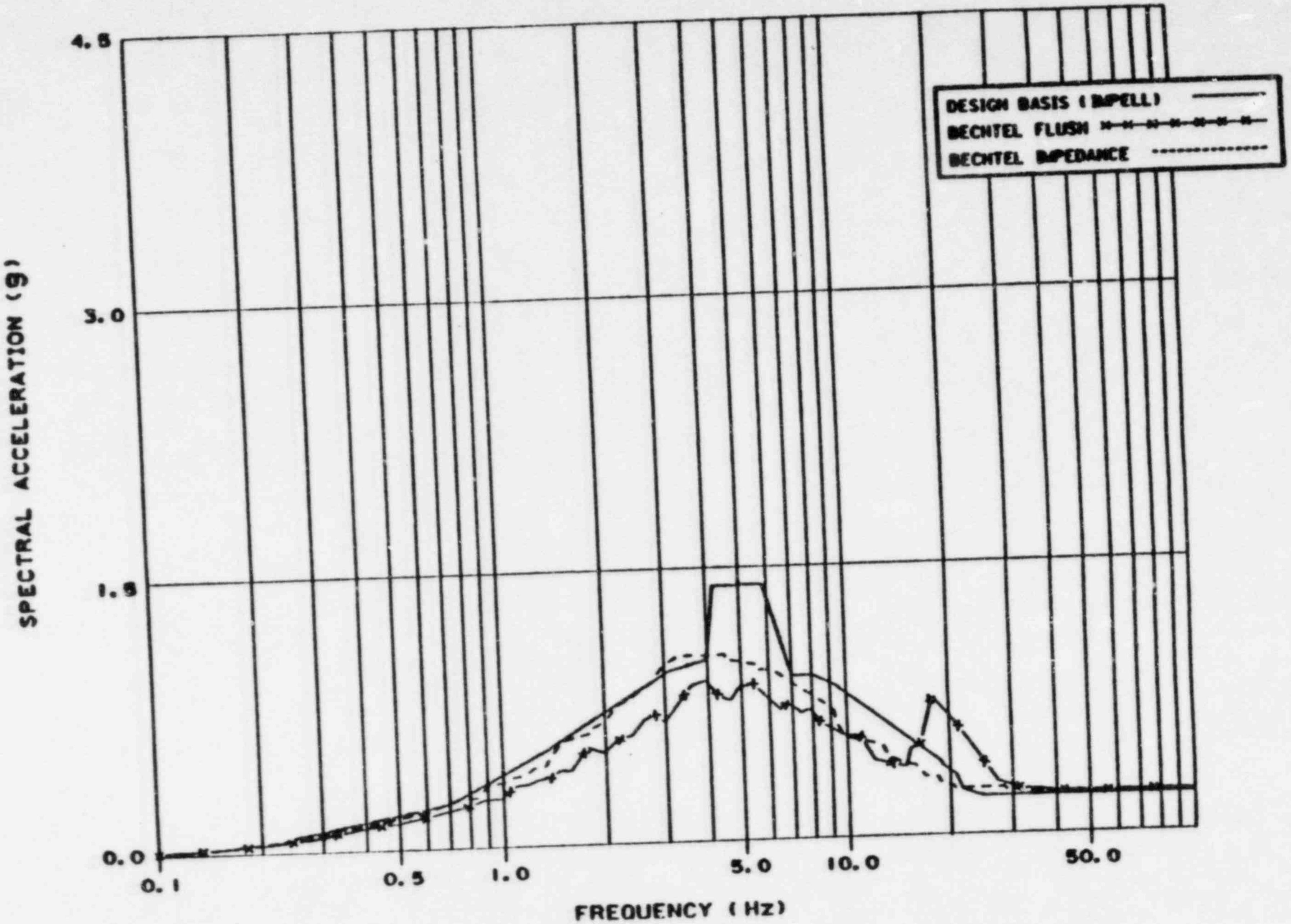

FIGURE A-13-8

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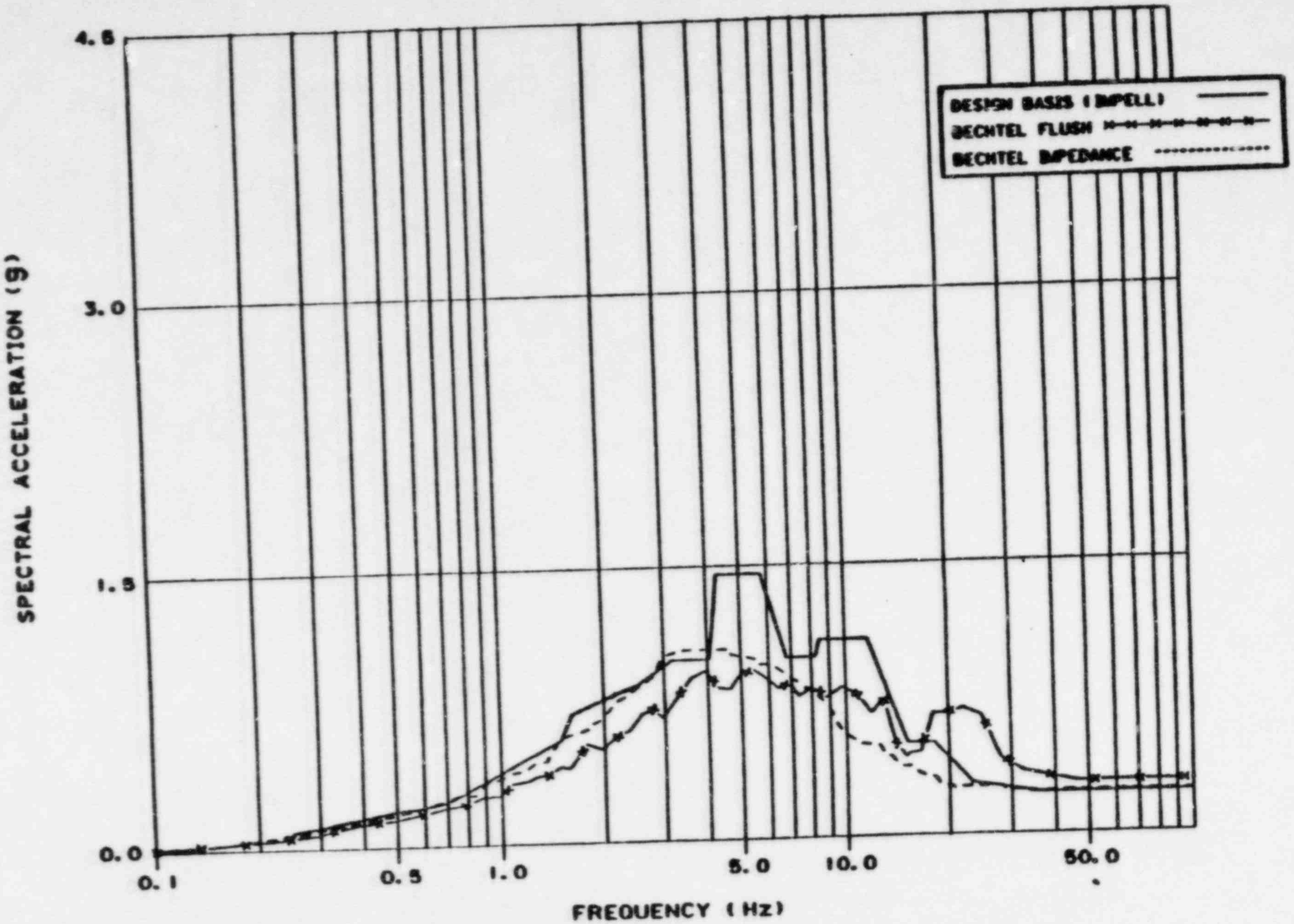

FIGURE A-13-9

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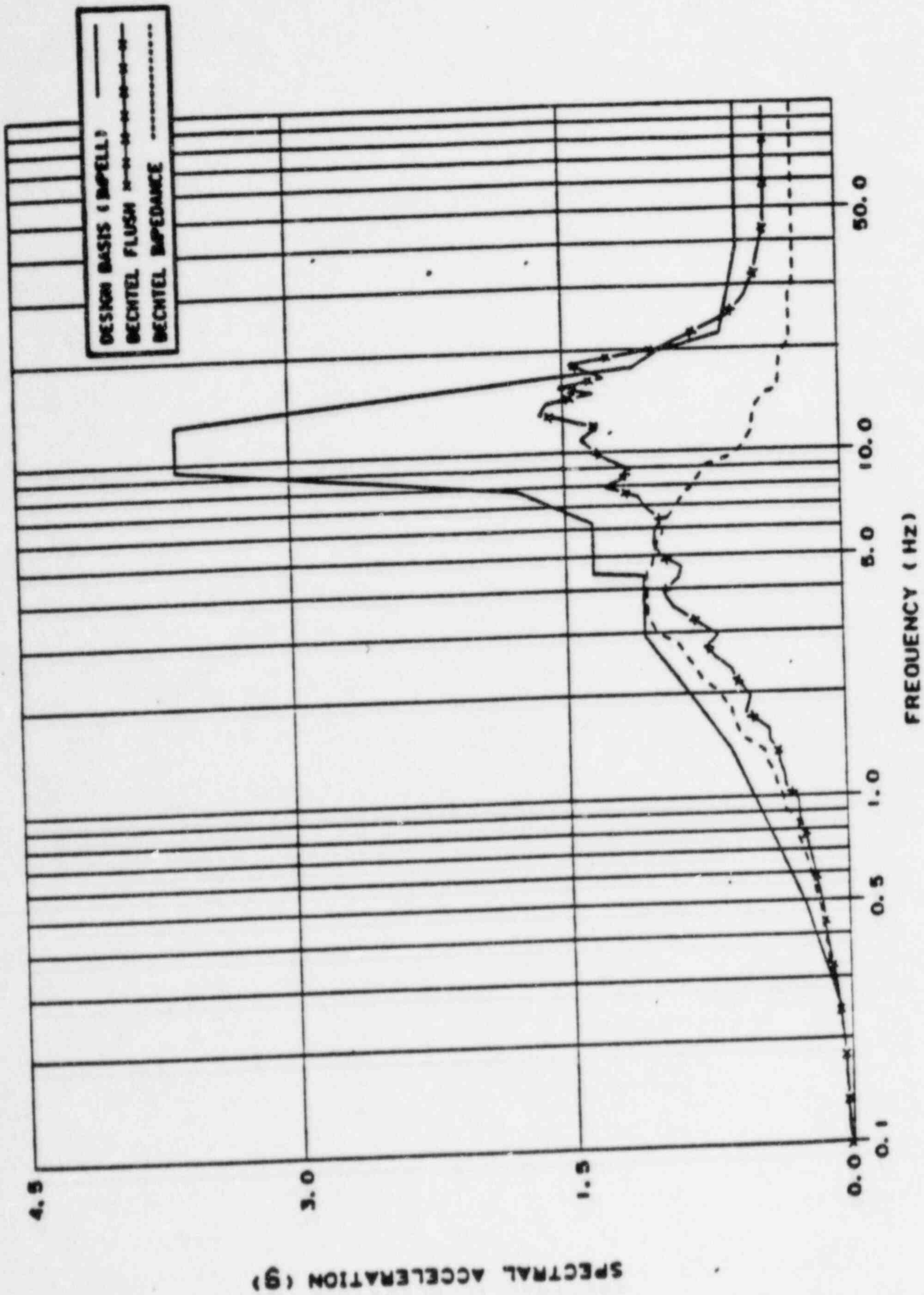

FIGURE A-13-10

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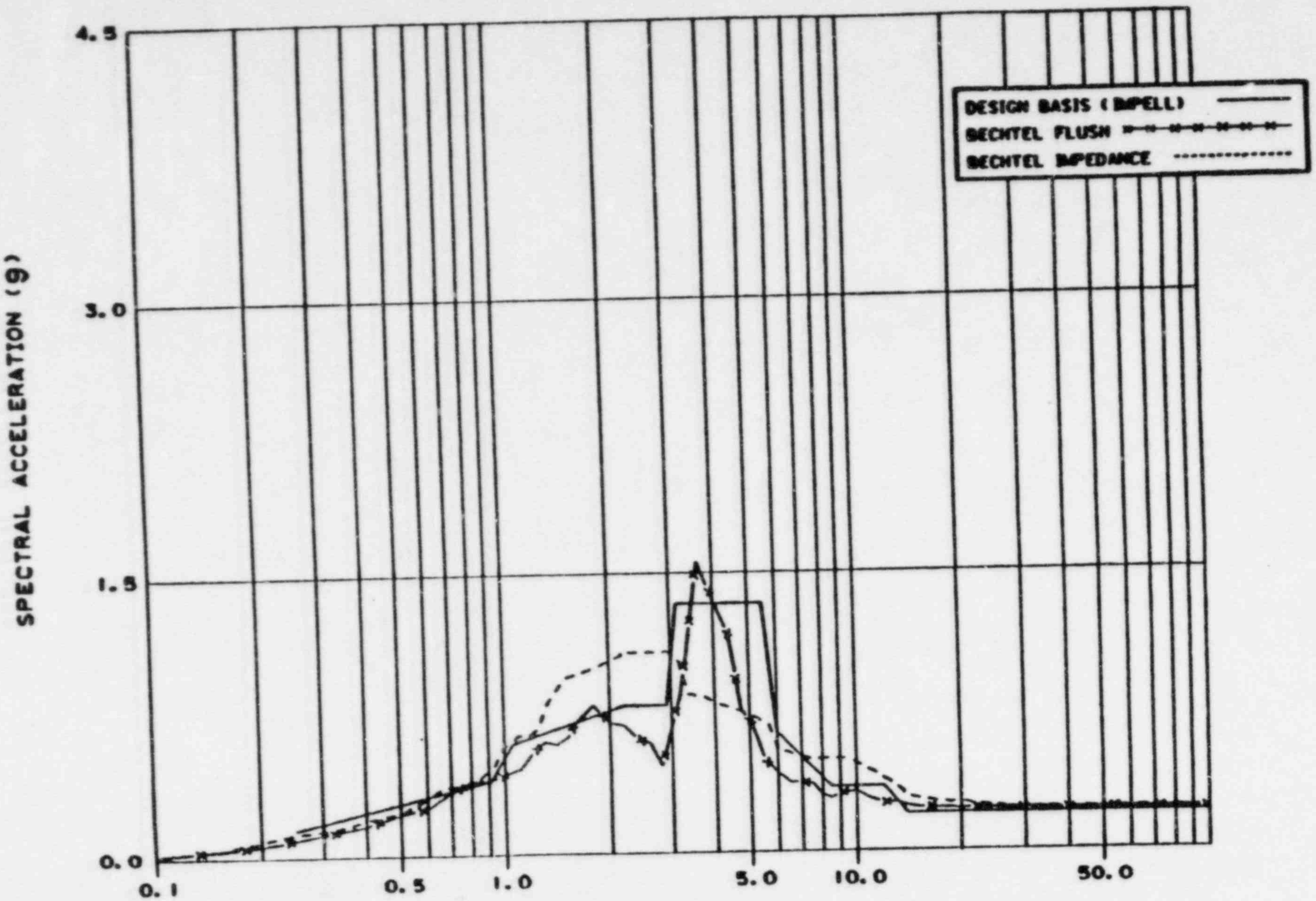

FIGURE A-13-11

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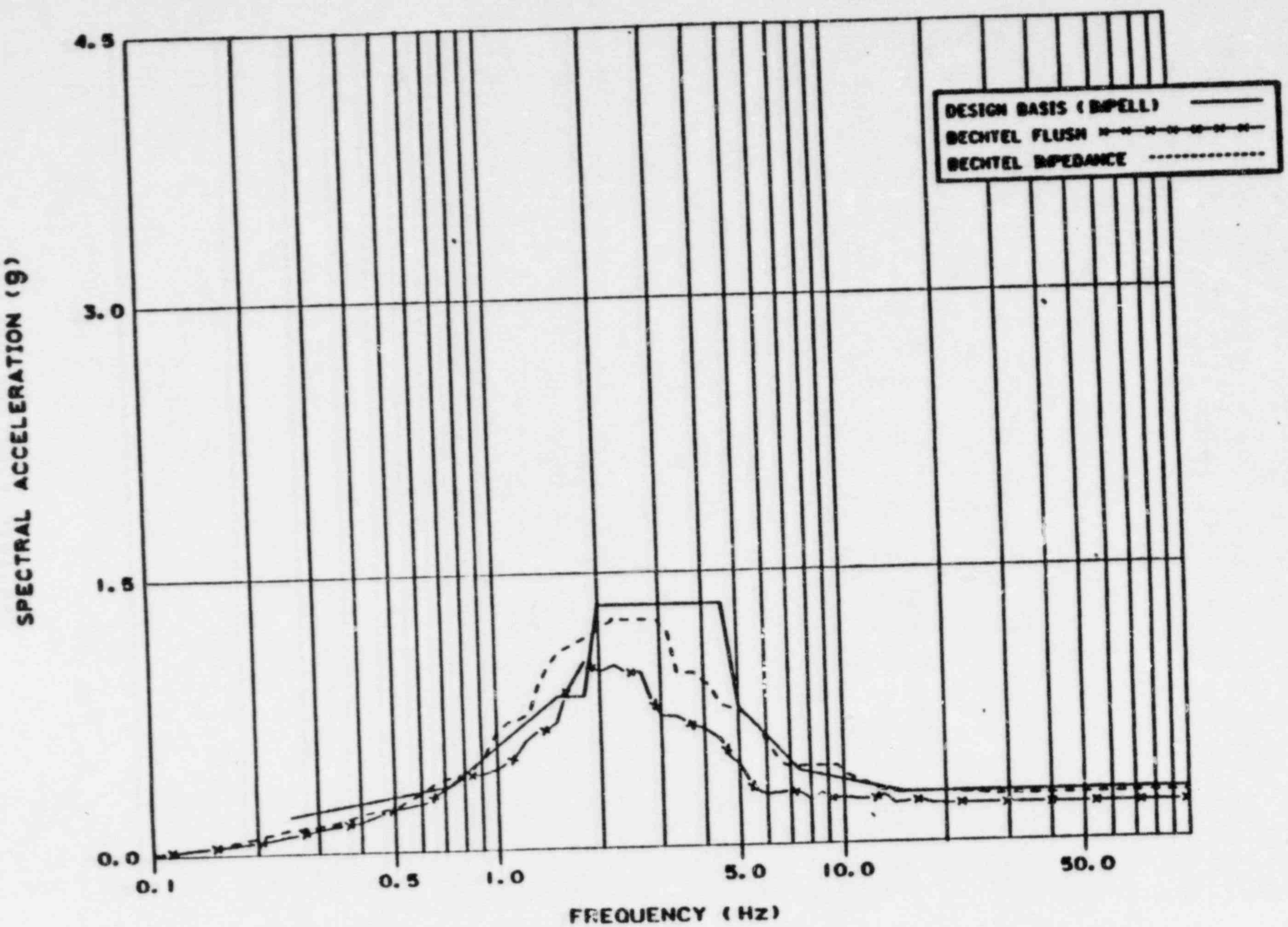

FIGURE A-13-12

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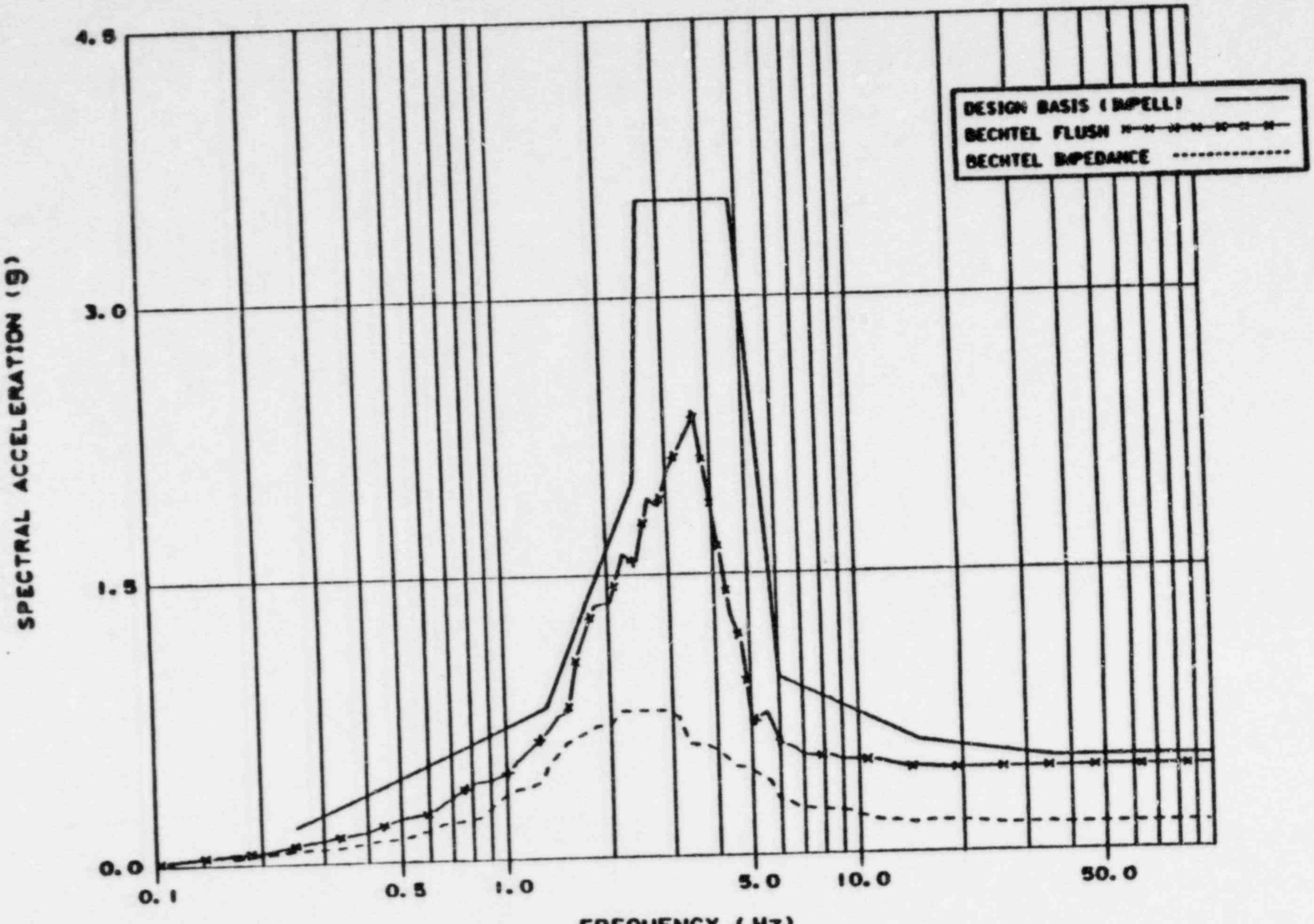

FIGURE A-13-13

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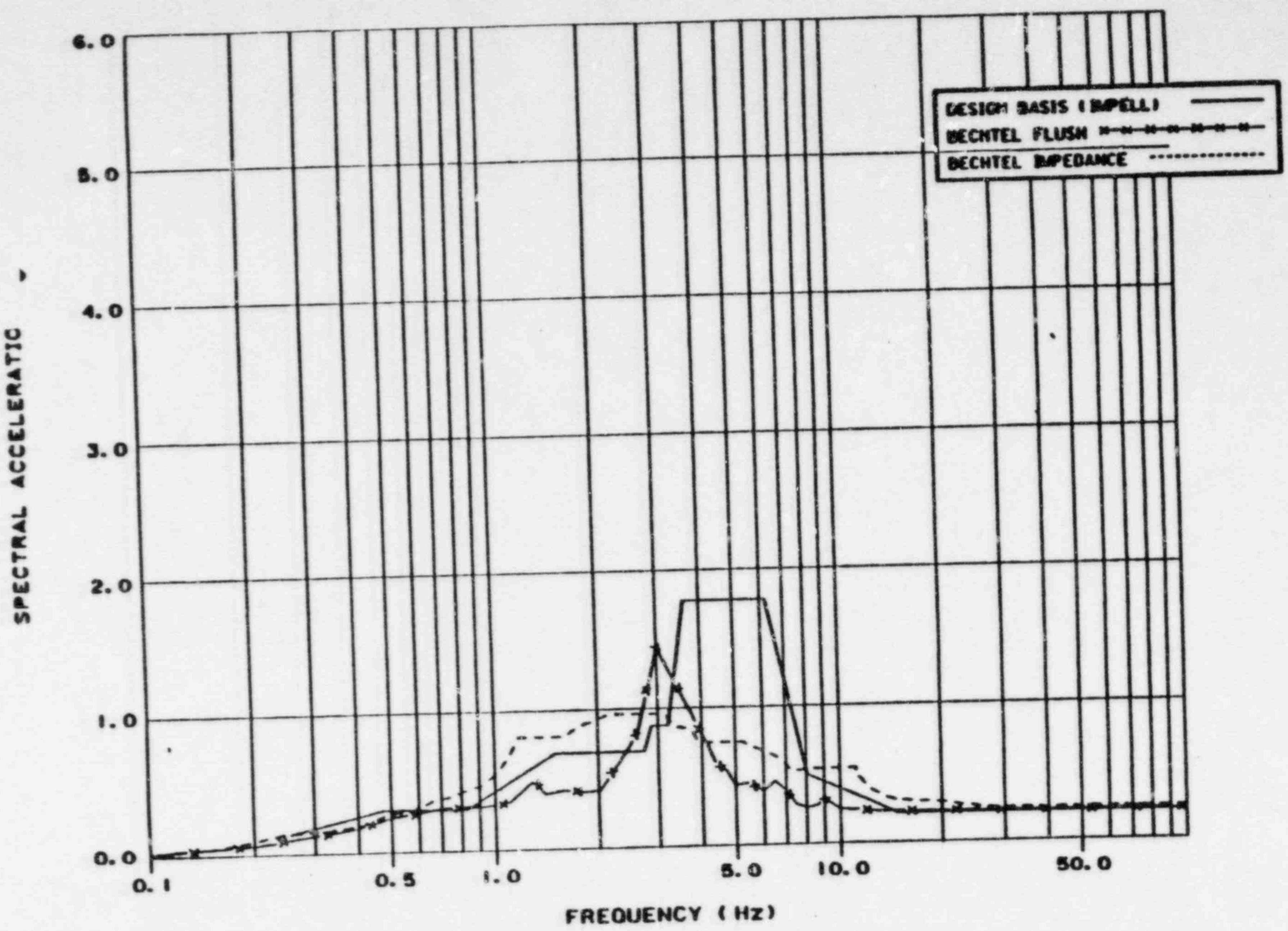

FIGURE A-13-14

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 54' -0",

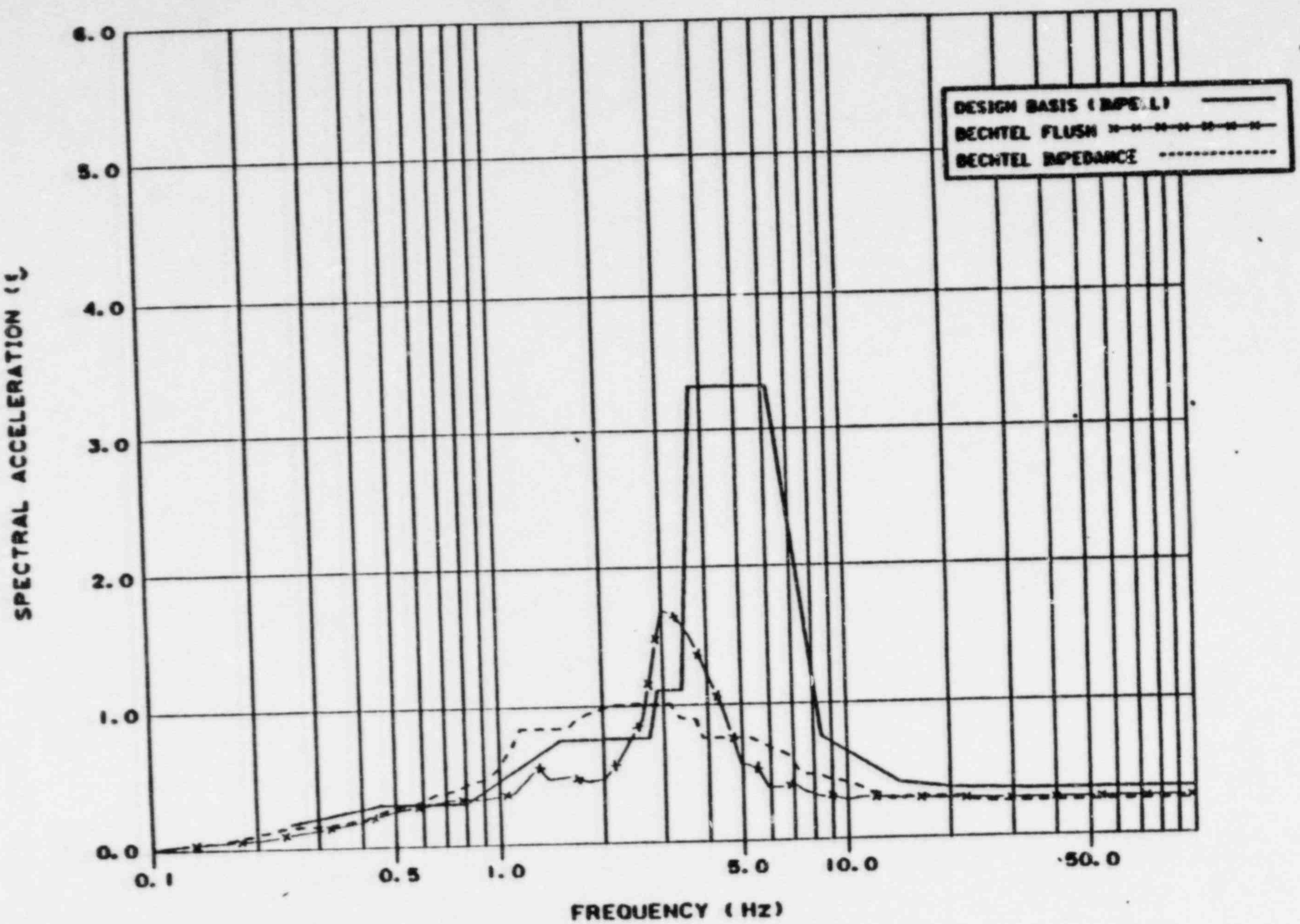

FIGURE A-13-15

RESPONSE SPECTRA COMPARISON,

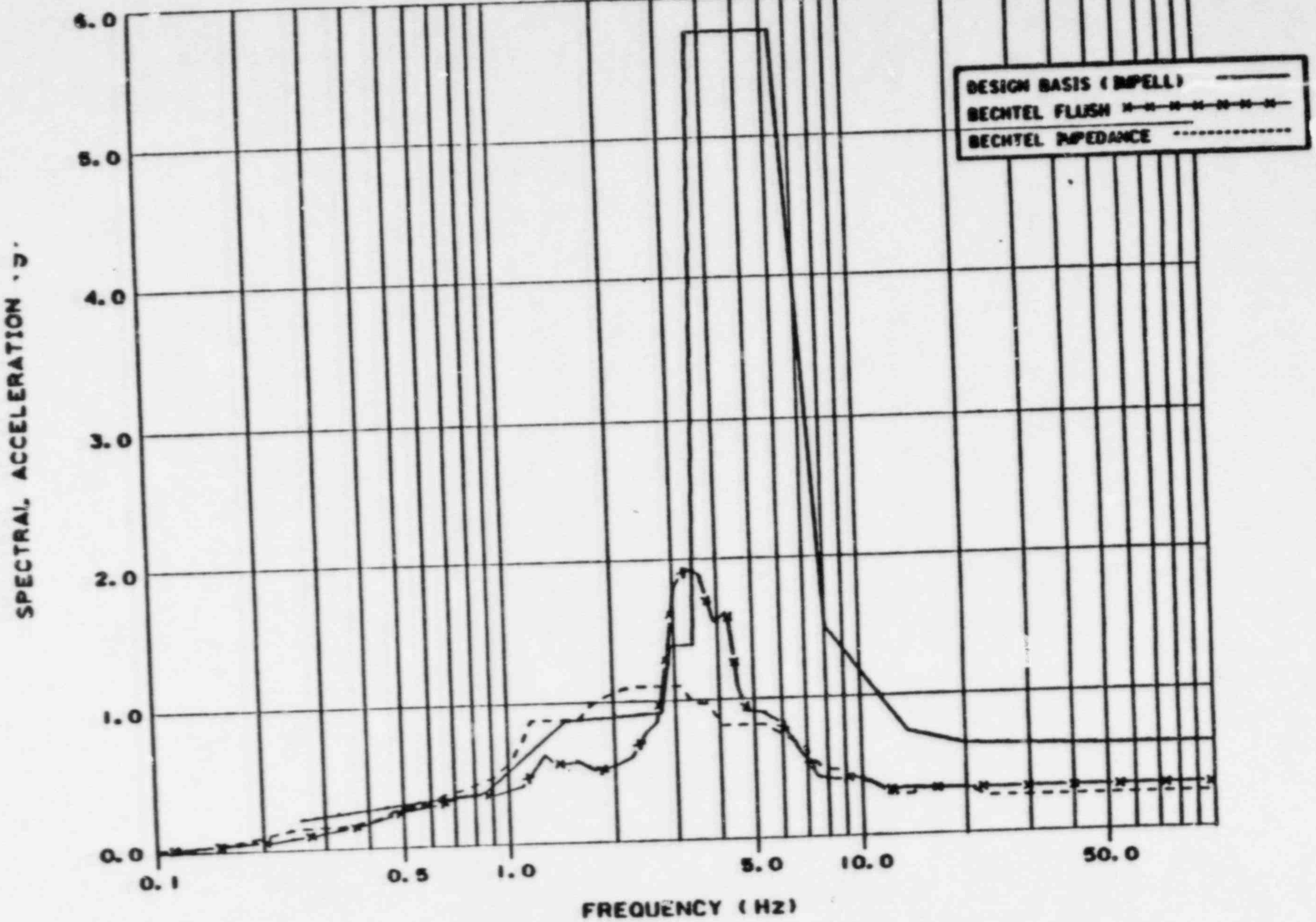

FIGURE A-13-16

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 178'-0".

DSER OPEN ITEM 51

DSER OPEN ITEM 51

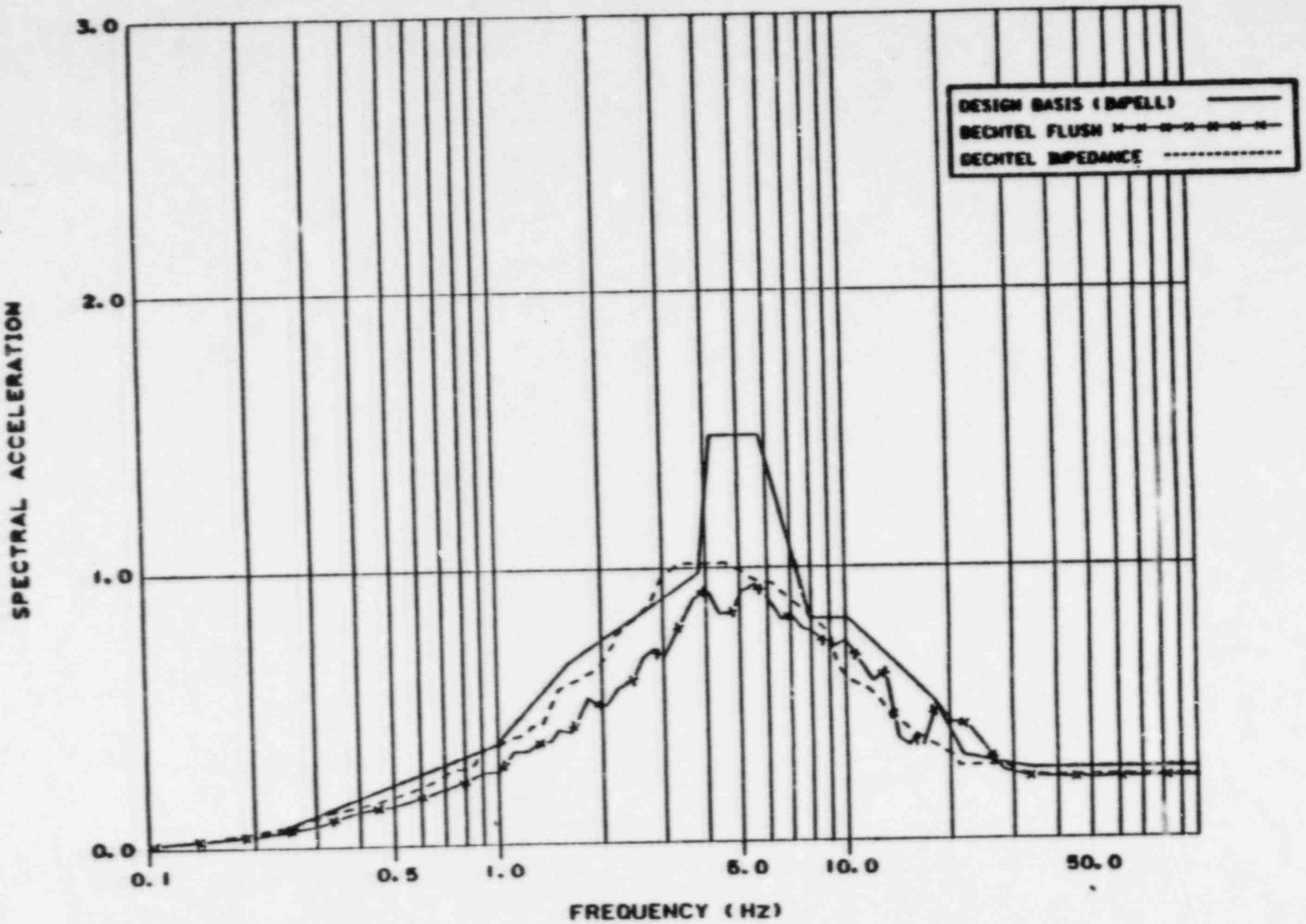

FIGURE A-13-17

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 54' -0",

DSER OPEN ITEM 51

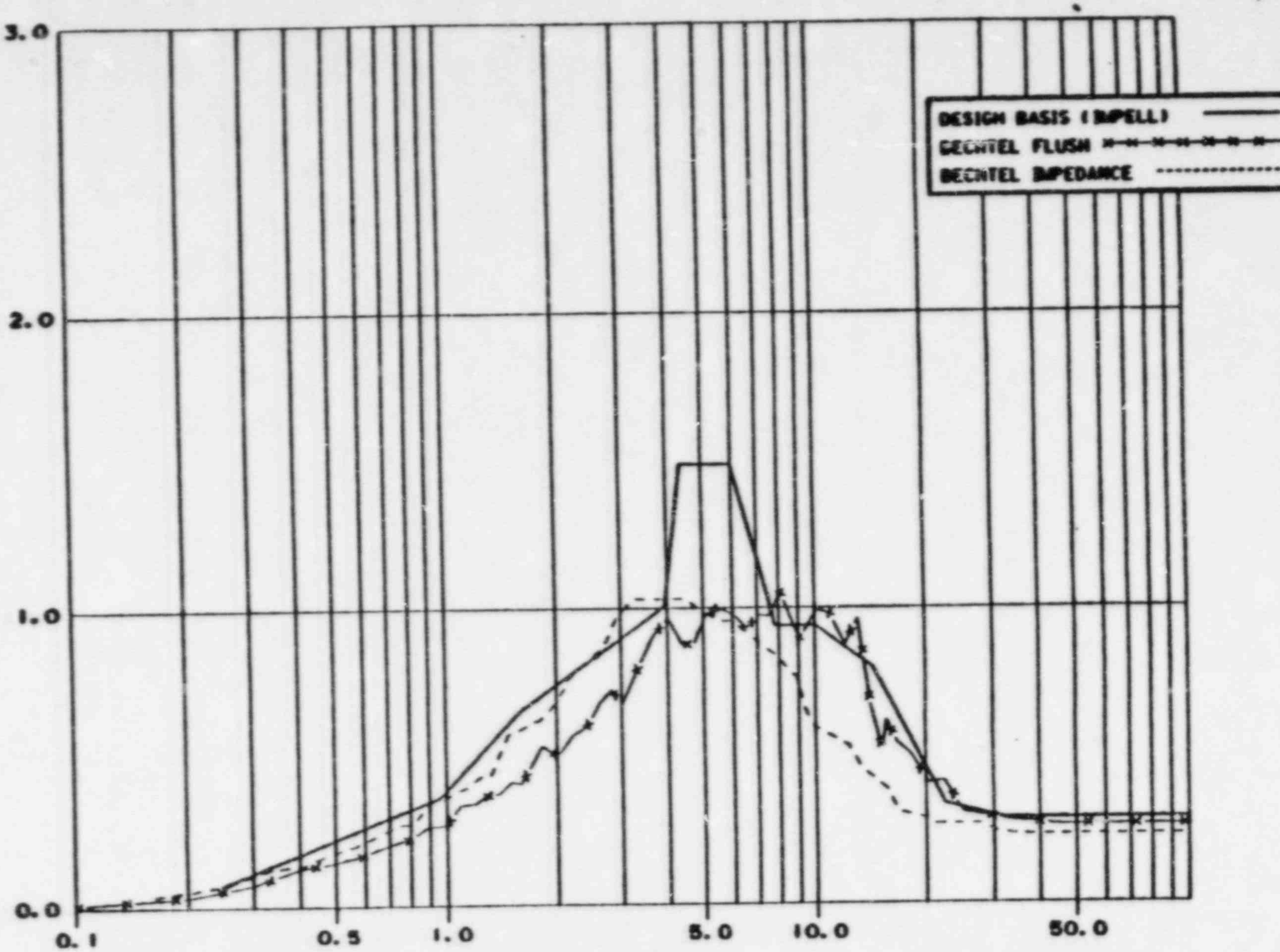

FIGURE A-13-18

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

DSER OPEN ITEM 5/

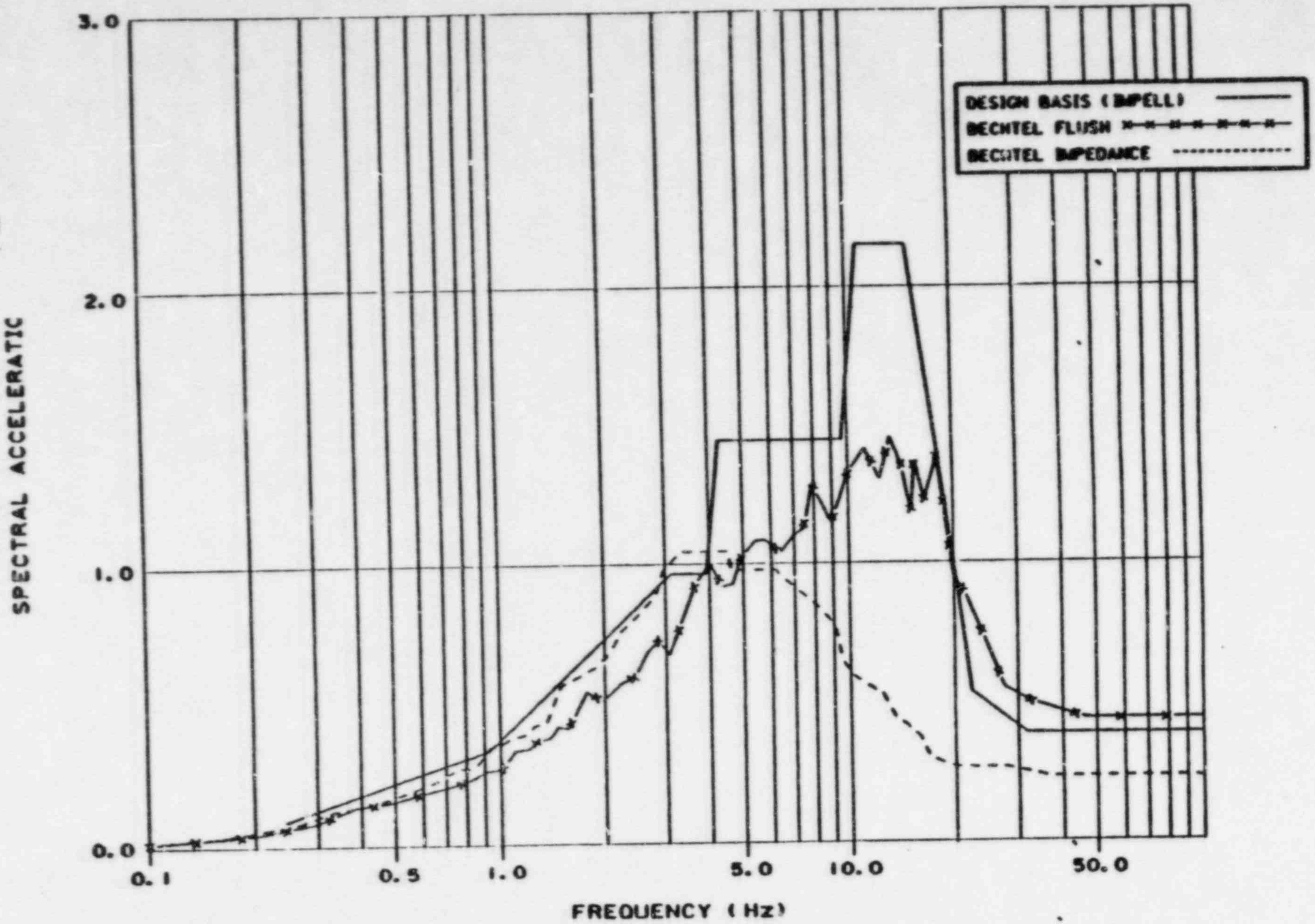

FIGURE A-13-19

RESPONSE SPECTRA COMPARISON,
AUXILIARY BUILDING AT ELEV. 178'-0".

HOPE CREEK UNIT 1

DSER OPEN ITEM 5/

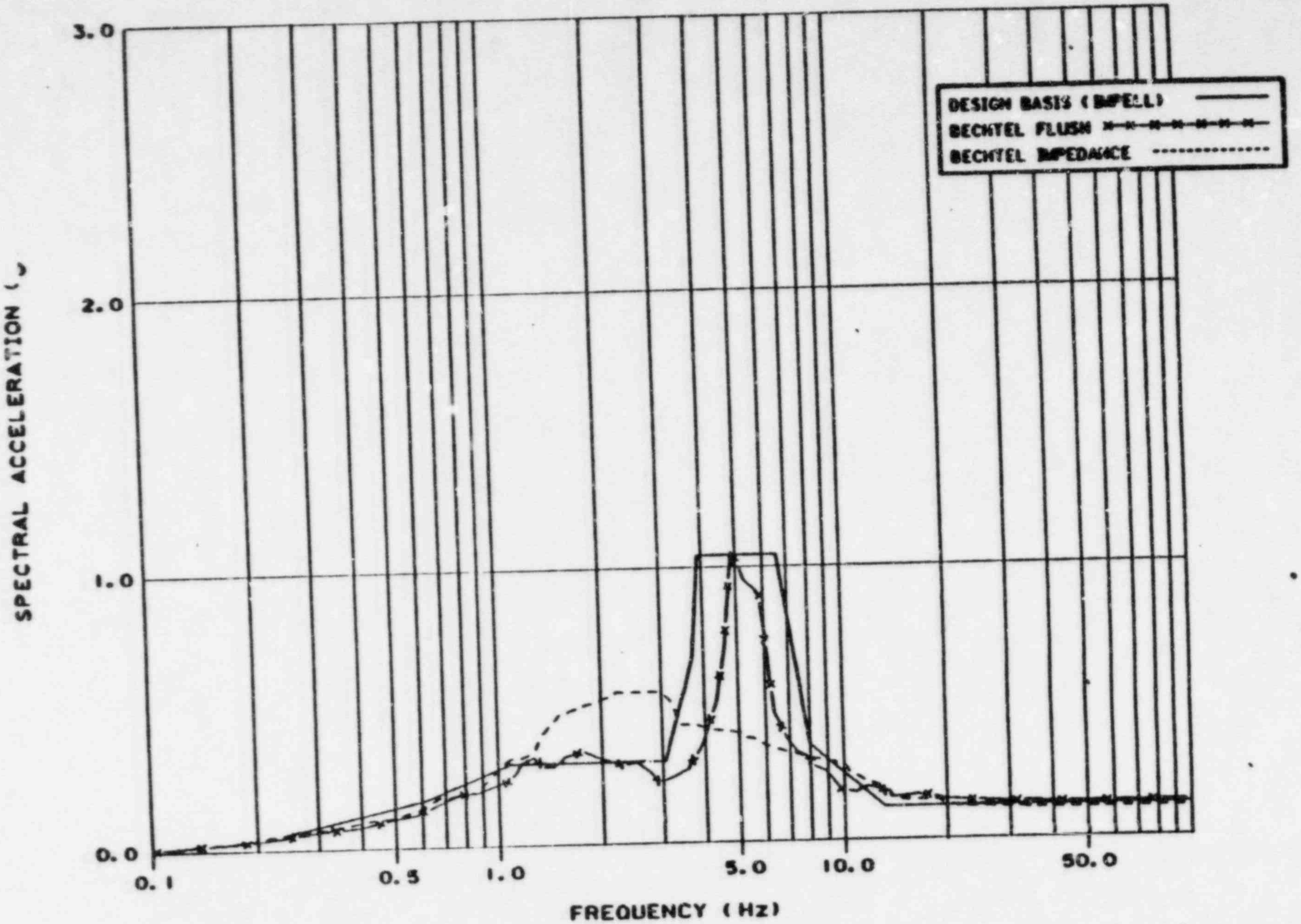

FIGURE A-13-20

RESPONSE SPECTRA COMPARISON,
UNIT 1 REACTOR BUILDING AT ELEV. 54'-0"

DSER OPEN ITEM 51

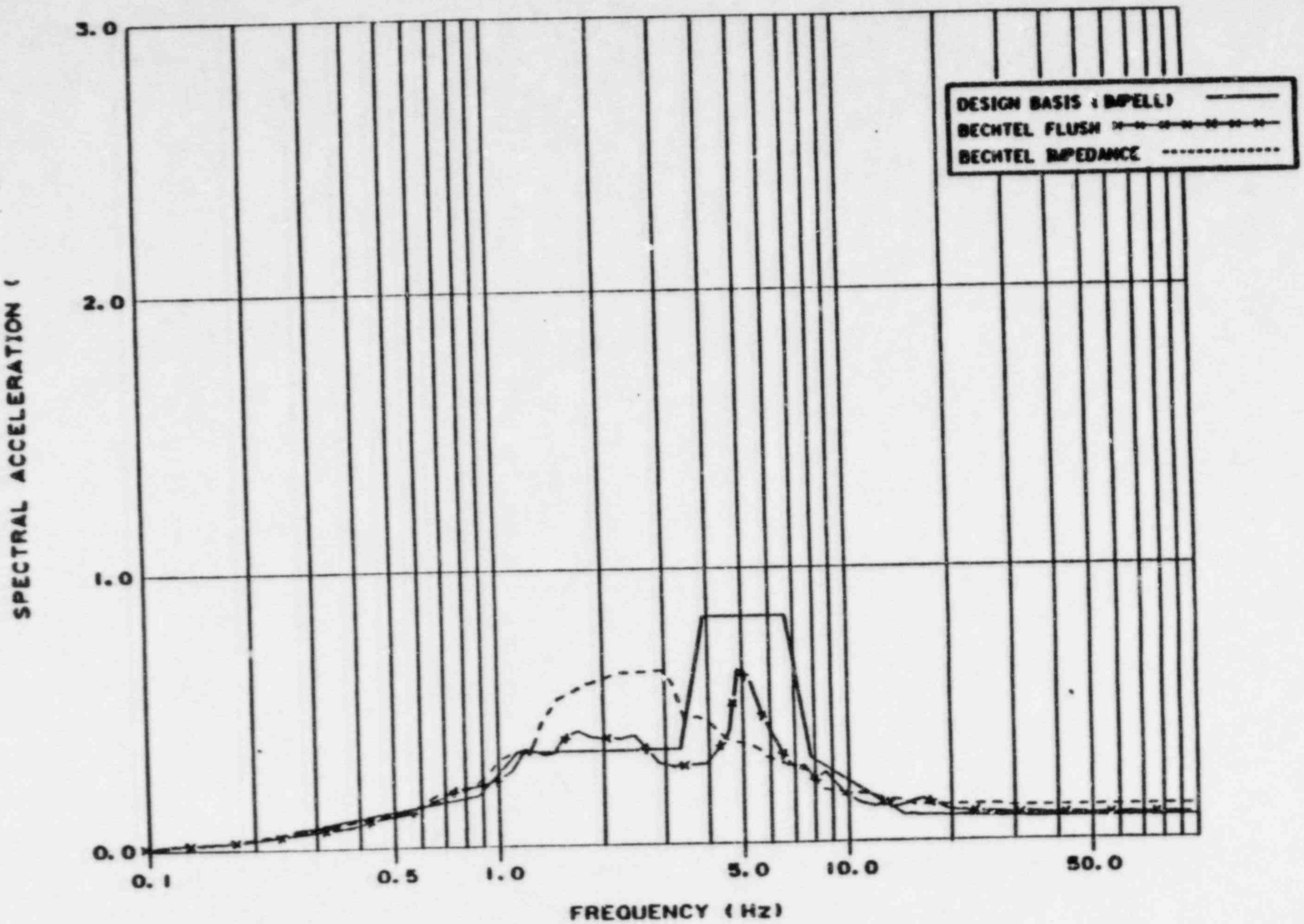

FIGURE A-13-21

RESPONSE SPECTRA COMPARISON.

UNIT 1 REACTOR BUILDING AT ELEV. 102'-0".

DSER OPEN ITEM 51

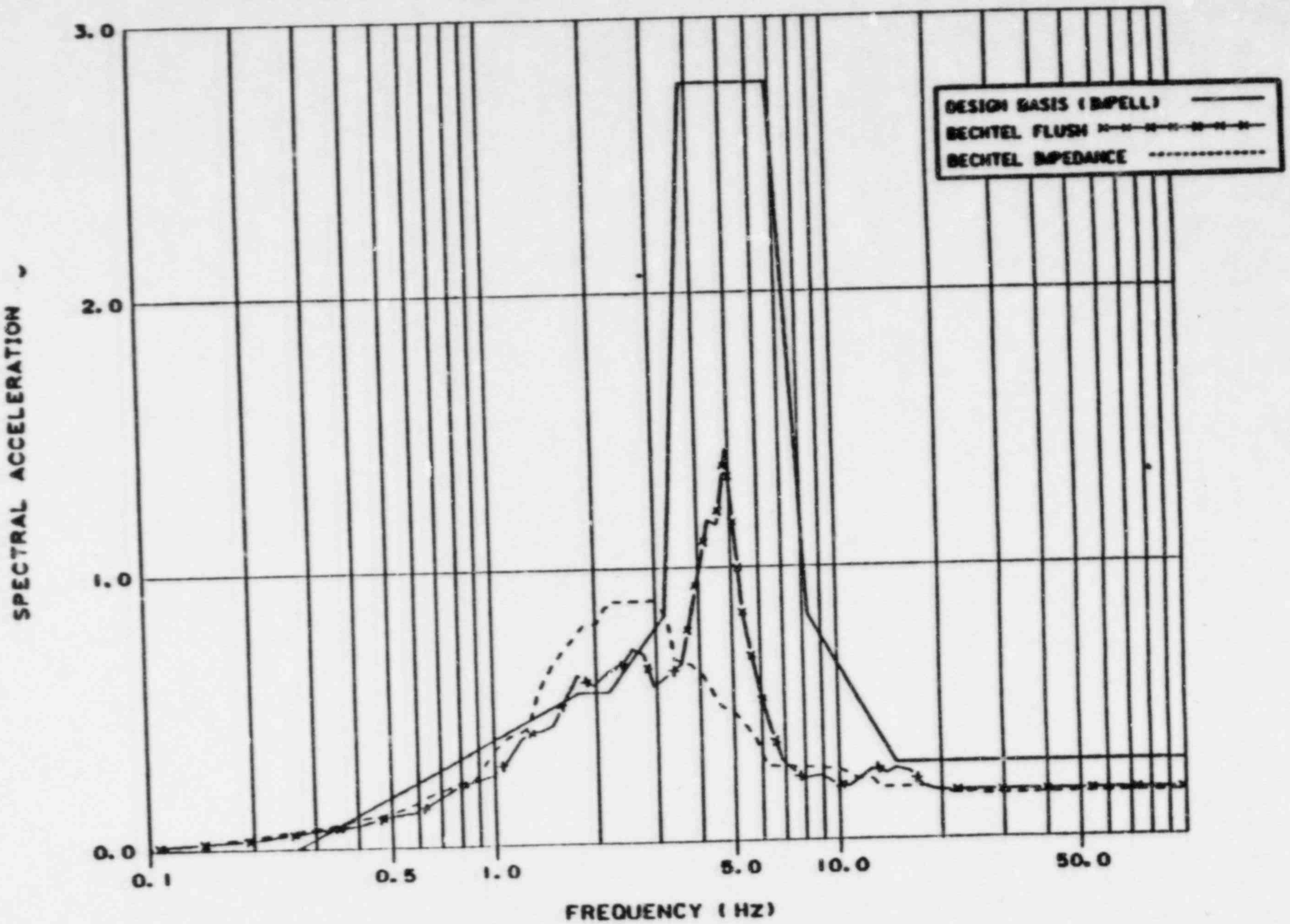

FIGURE A-13-22

RESPONSE SPECTRA COMPARISON.

REACTOR BUILDING AT ELEV. 201'-0".

DSER OPEN ITEM 51

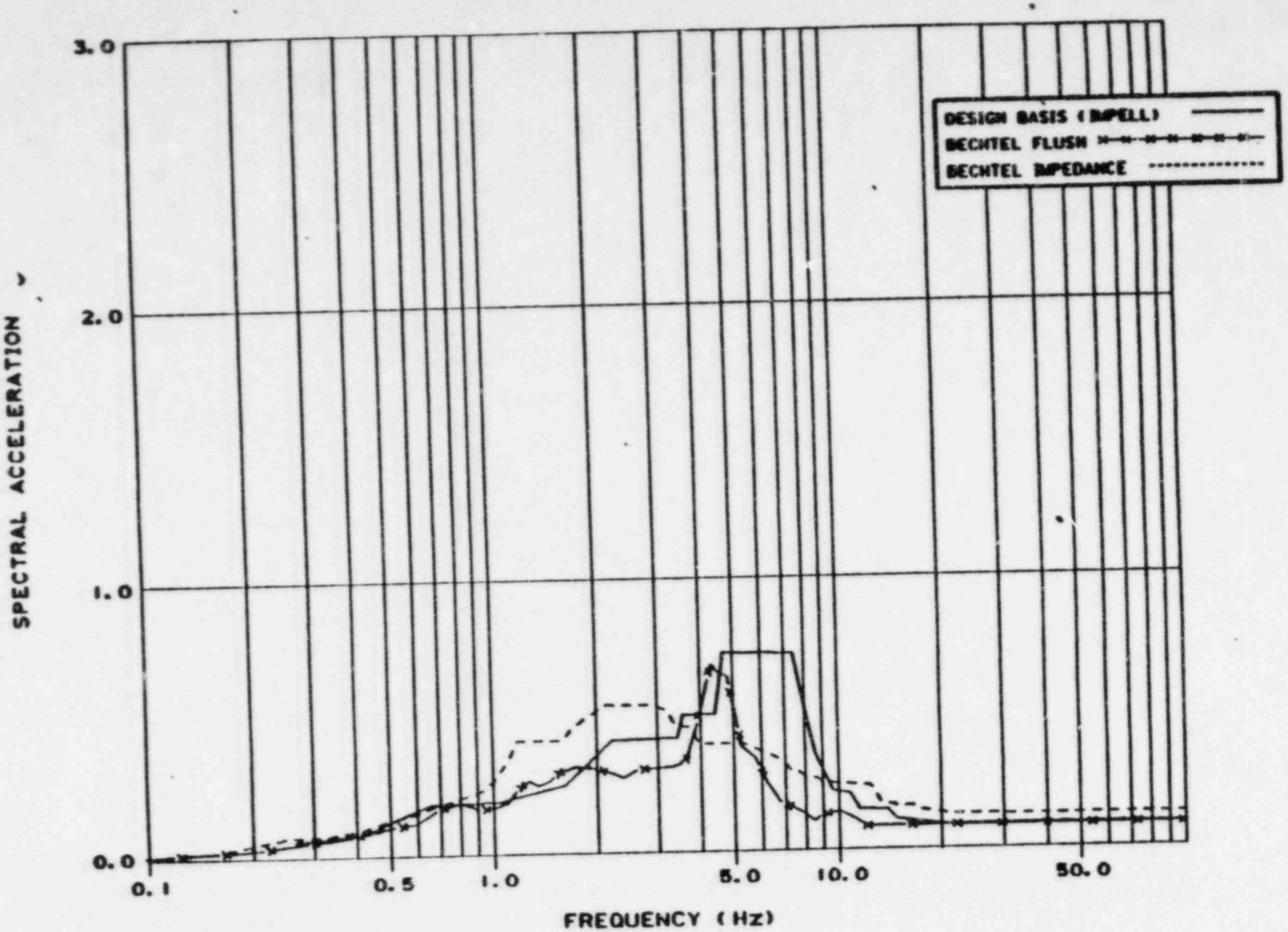

FIGURE A-13-23

RESPONSE SPECTRA COMPARISON,
UNIT 1 REACTOR BUILDING AT ELEV 54'-0".

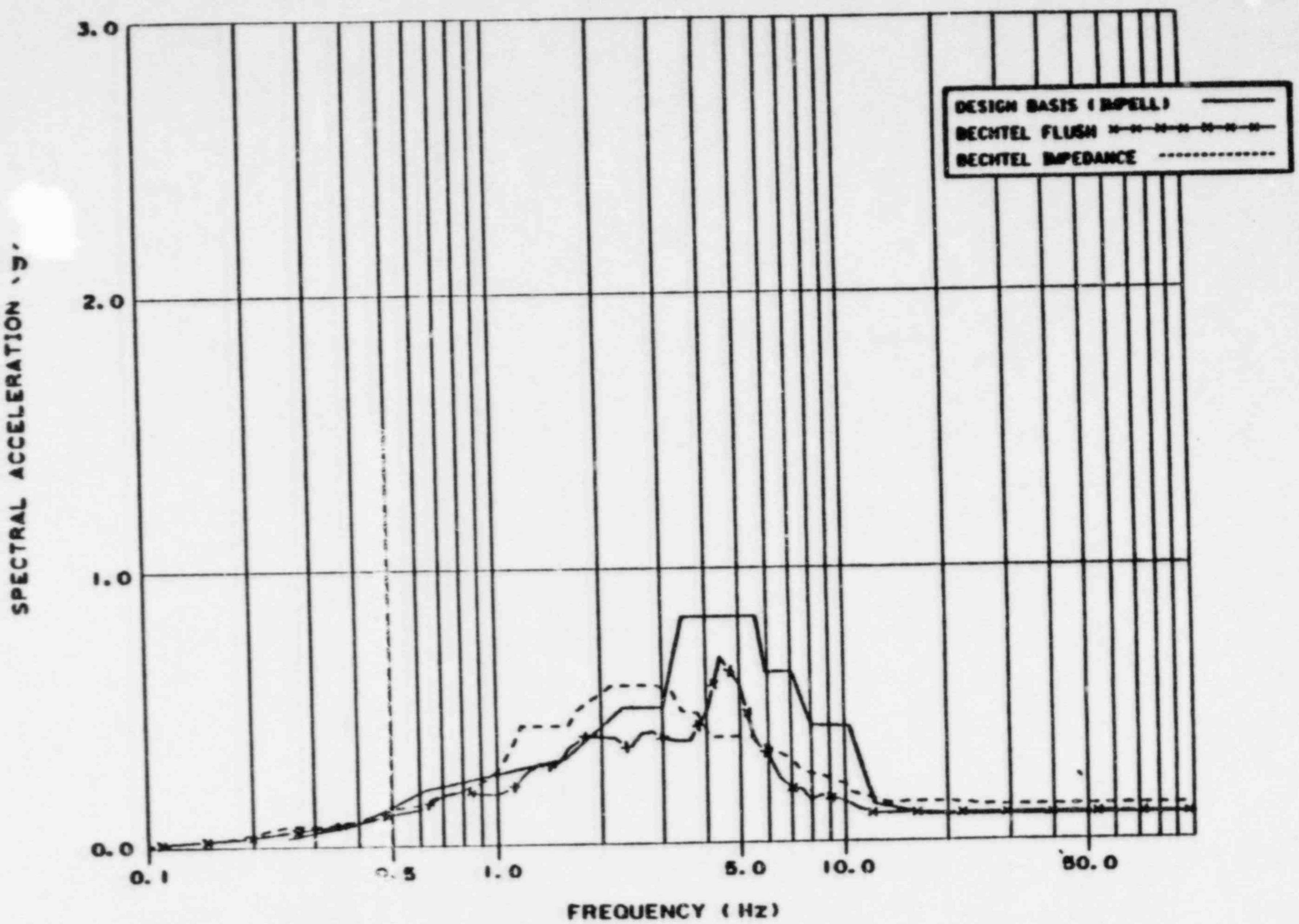

FIGURE A-13-24

RESPONSE SPECTRA COMPARISON,
UNIT 1 REACTOR BUILDING AT ELEV. 102' -0".

DSER OPEN ITEM 51

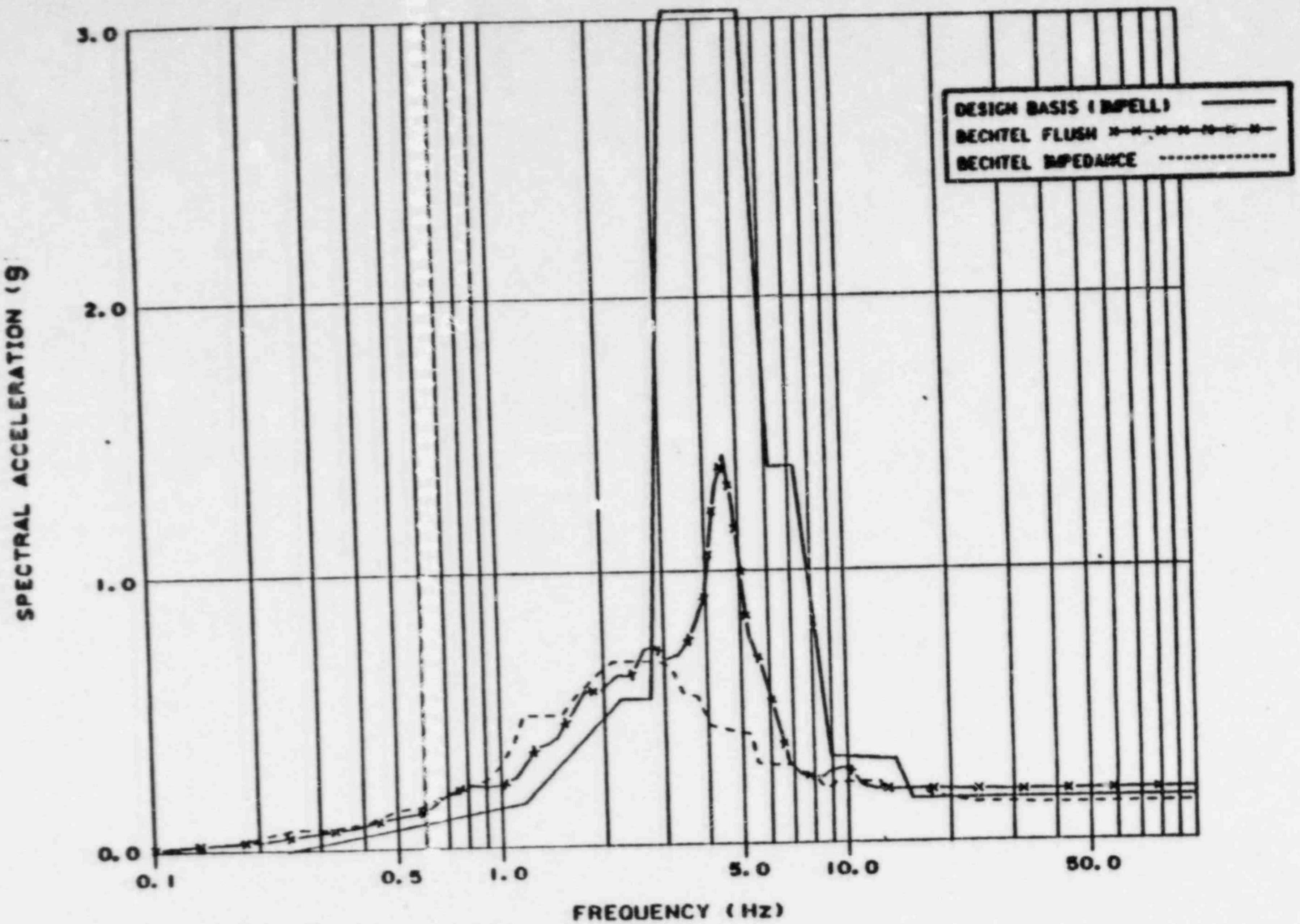

FIGURE A-13-25

RESPONSE SPECTRA COMPARISON.

DSER OPEN ITEM 51

DSER OPEN ITEM 51

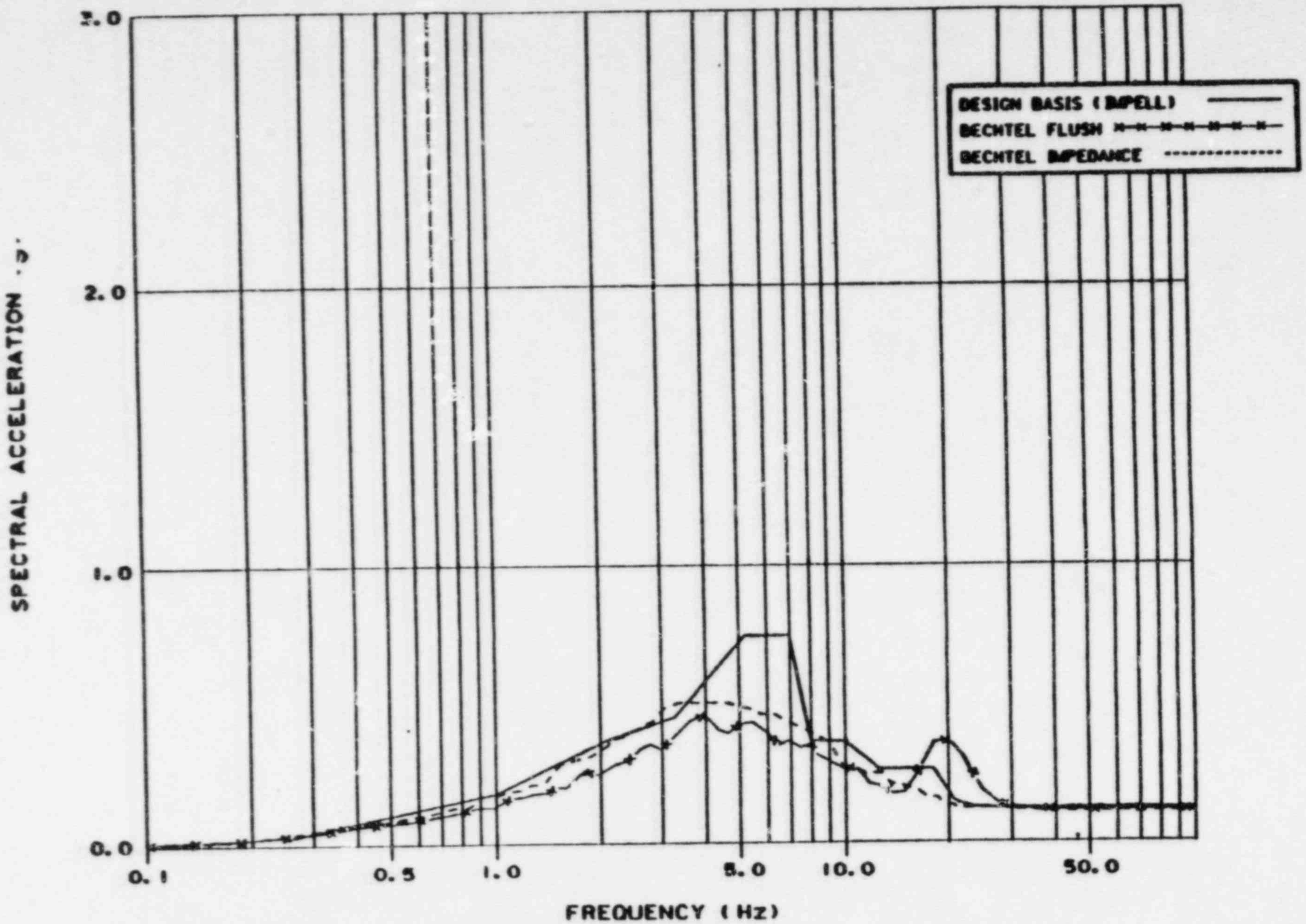

FIGURE A-13-26

RESPONSE SPECTRA COMPARISON,
UNIT 1 REACTOR BUILDING AT ELEV. 54' -0".

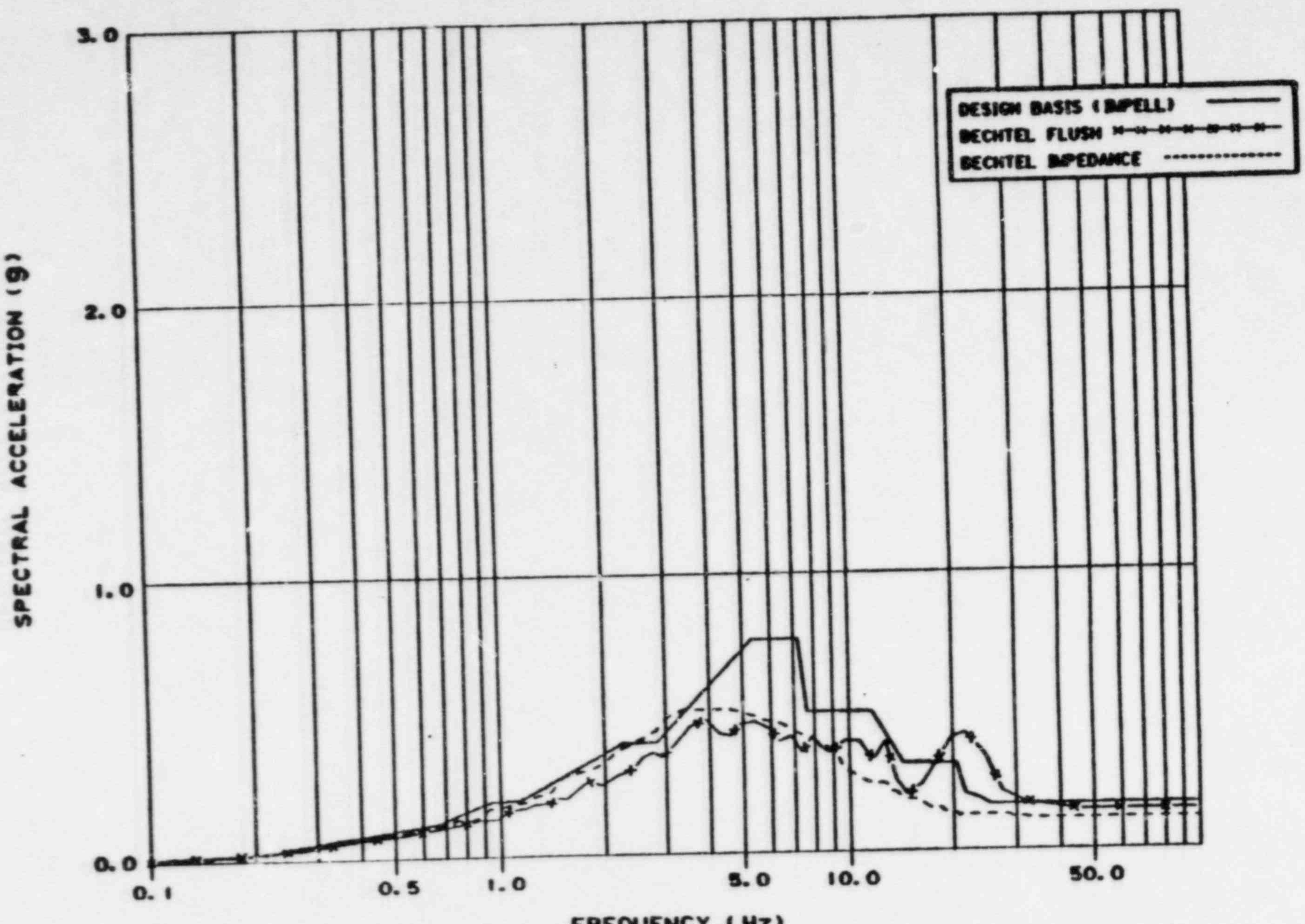

FIGURE A-13-27

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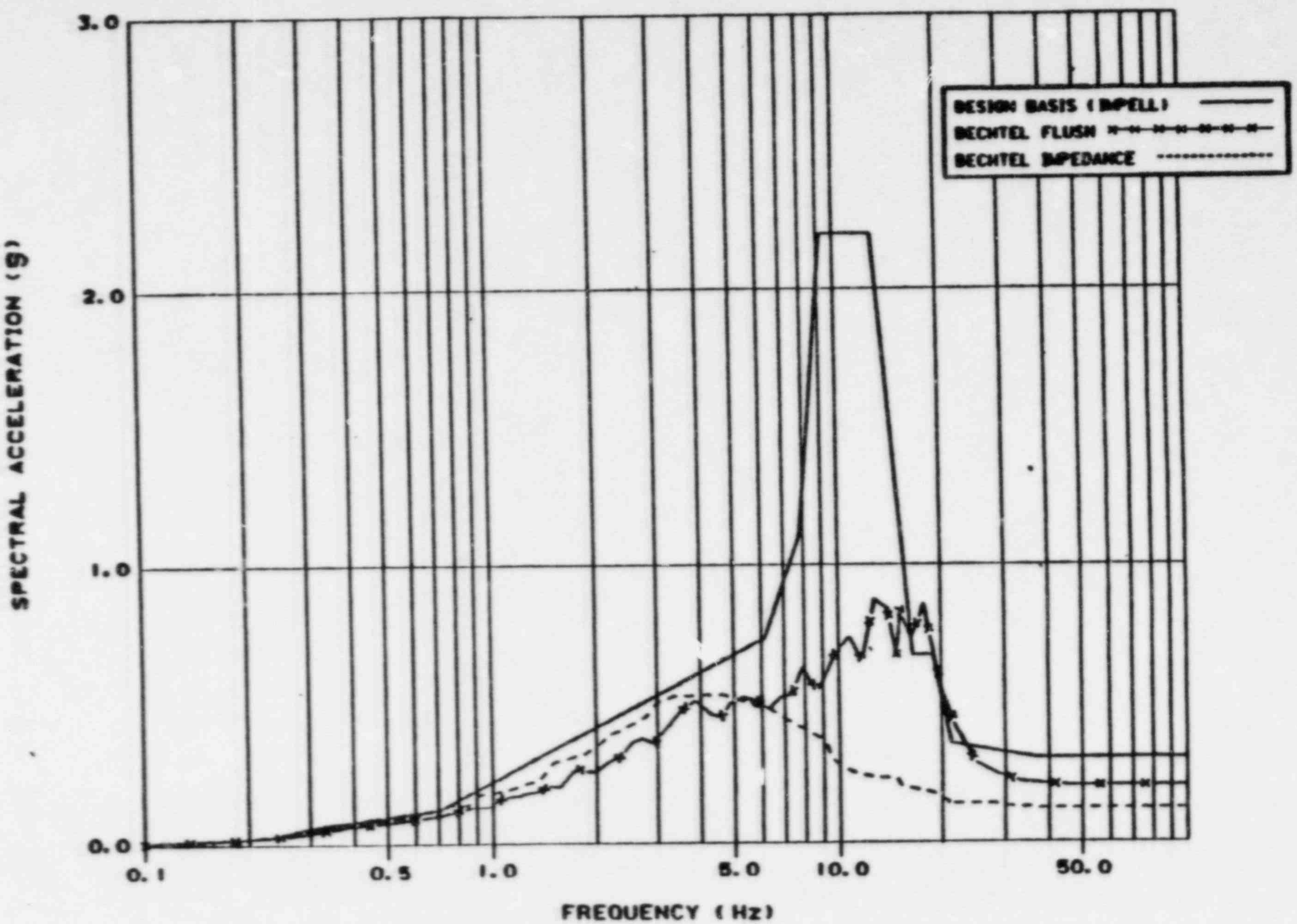

FIGURE A-13-20

DSER OPEN ITEM 51

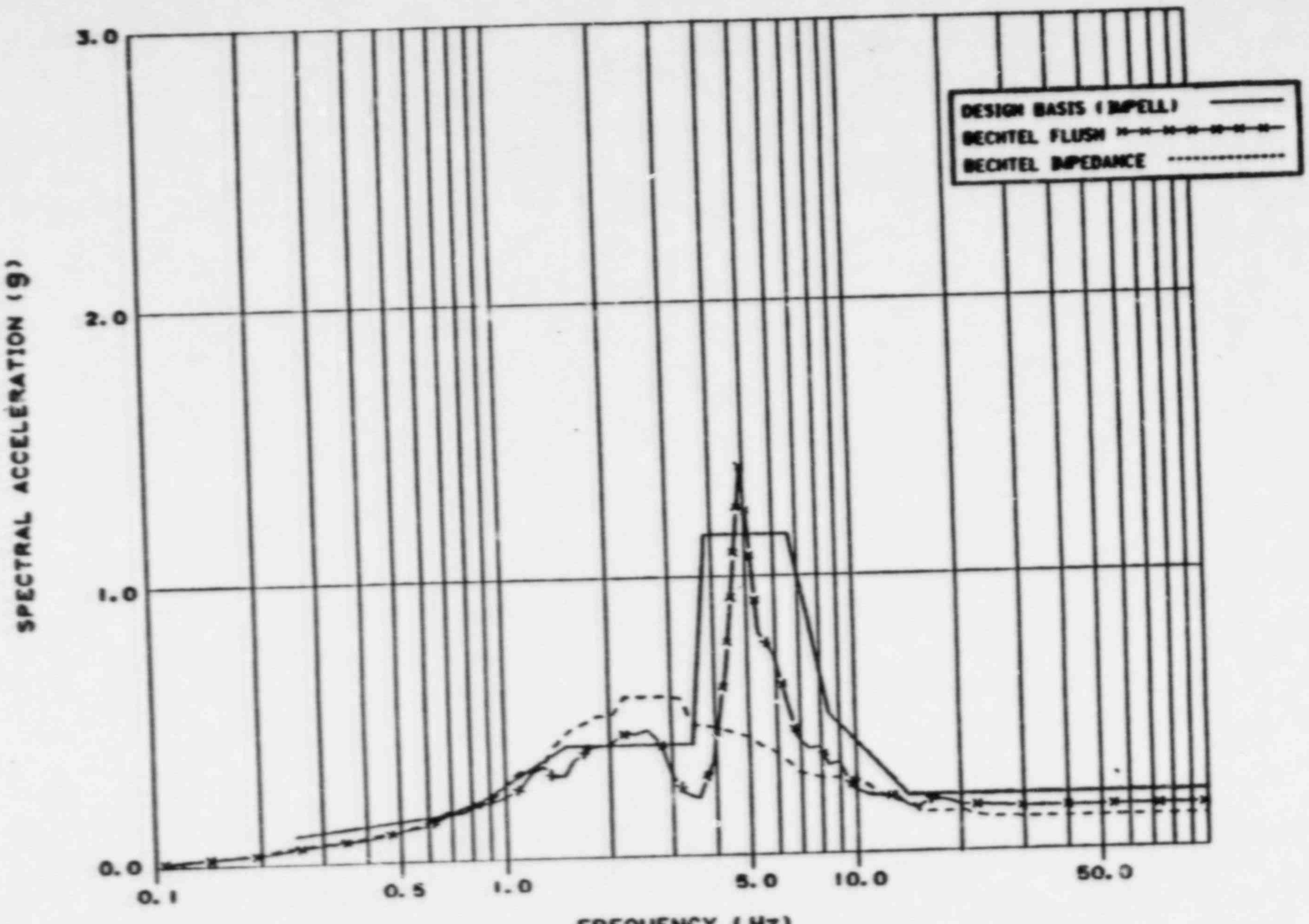

FIGURE A-13-29

DSER OPEN ITEM 51

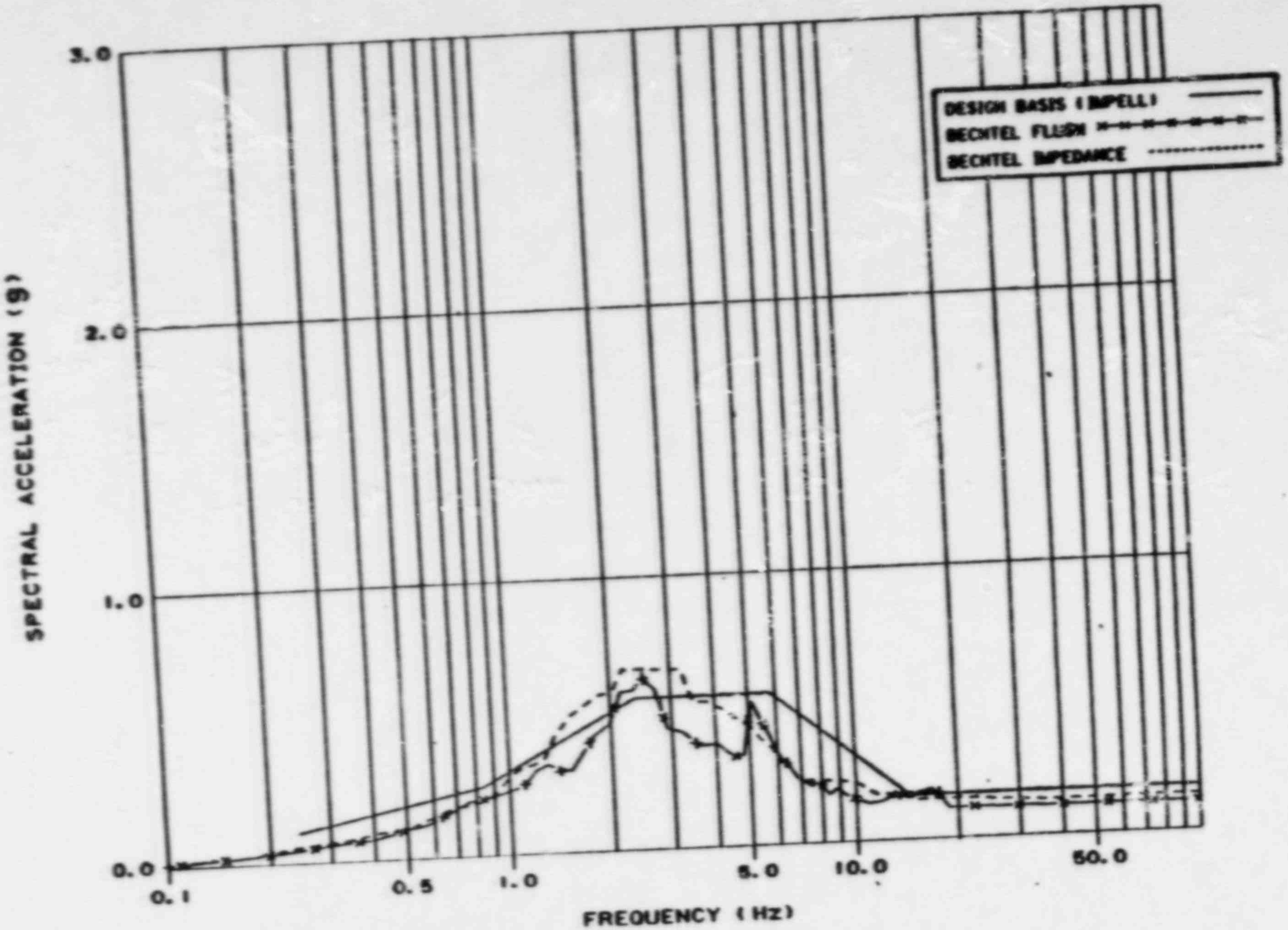

FIGURE A-13-30

DSER OPEN ITEM 51

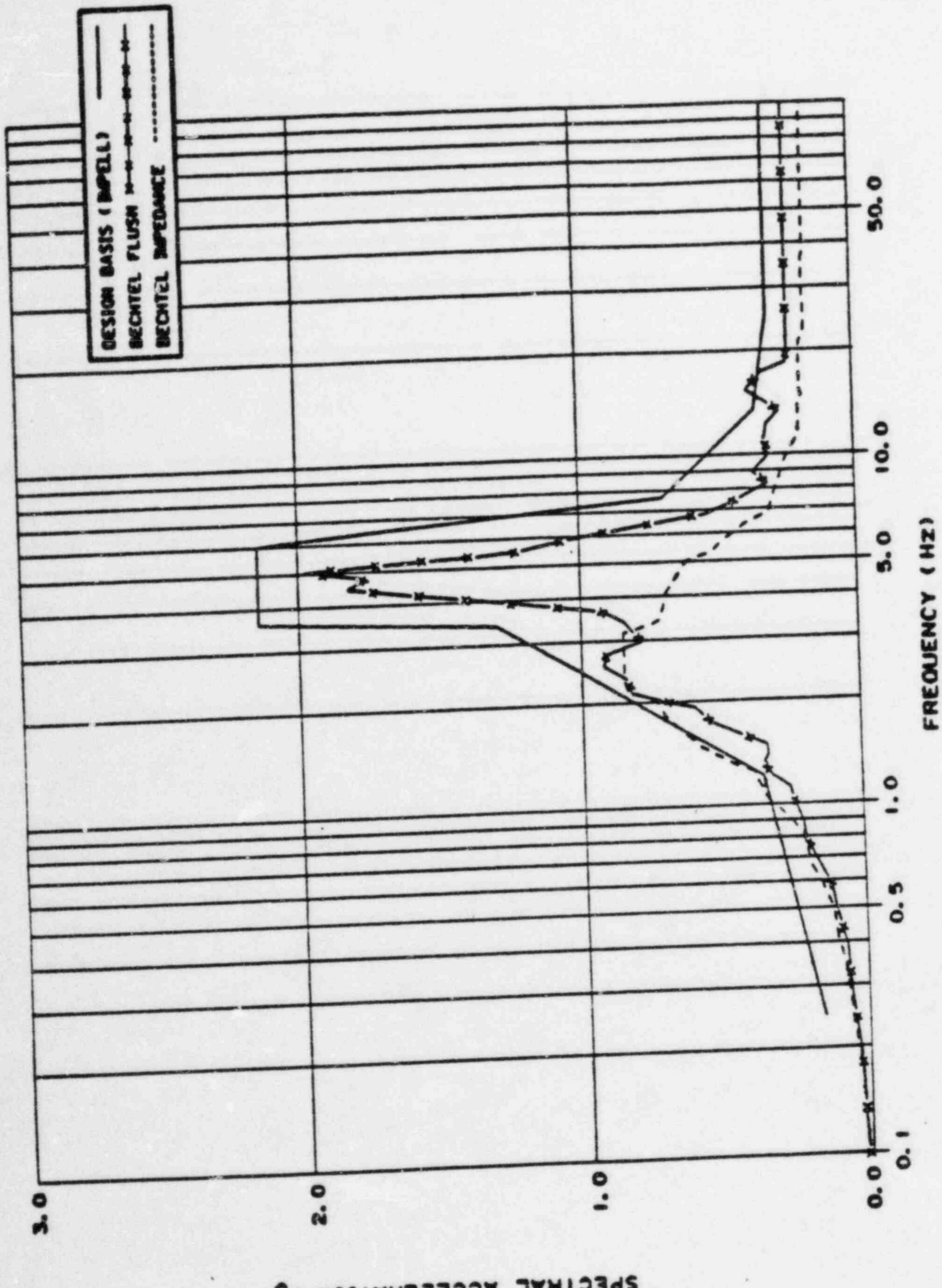

FIGURE A-13-31

DSER OPEN ITEM 51

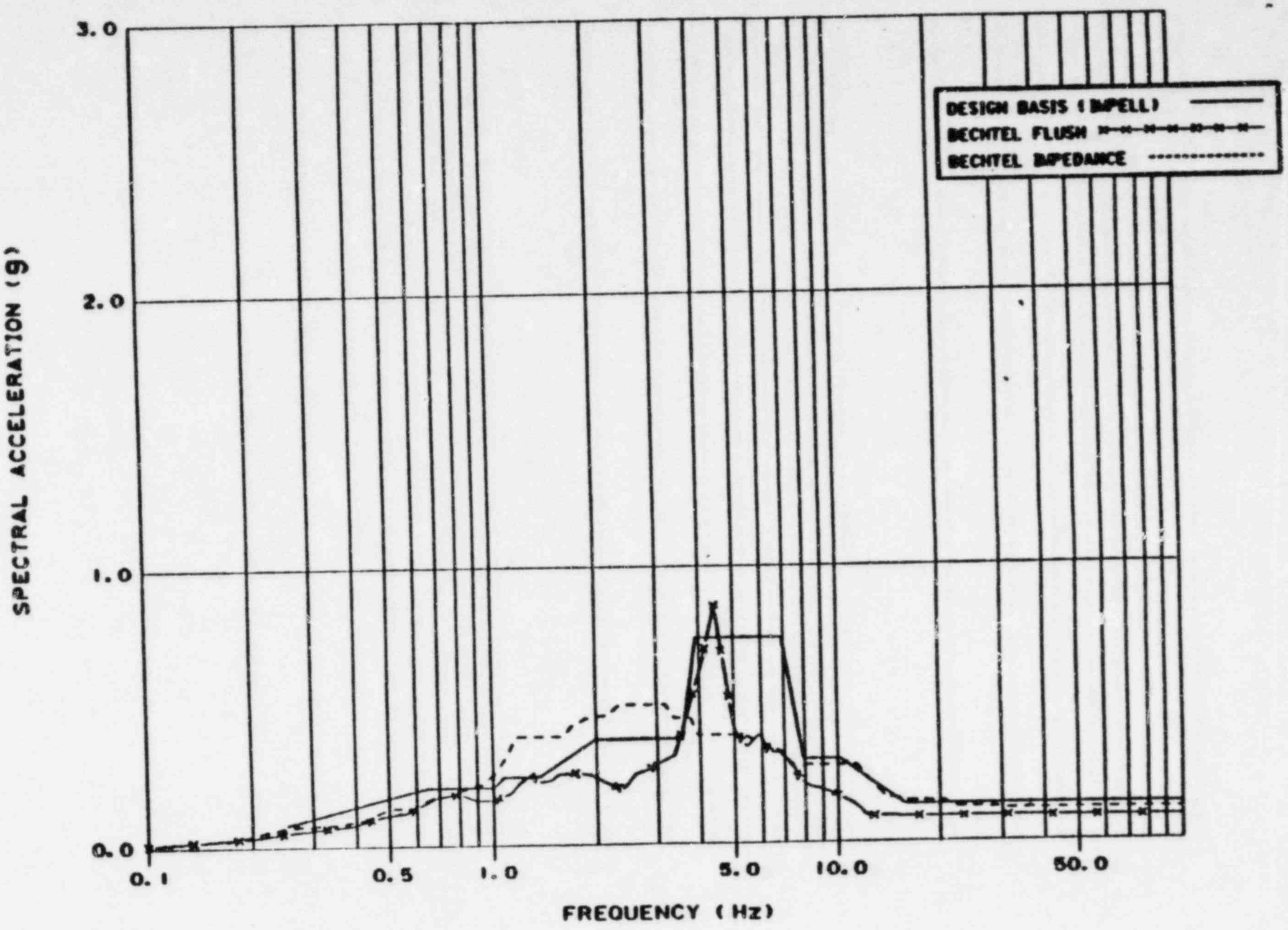

FIGURE A-13-32

DSER OPEN ITEM 51

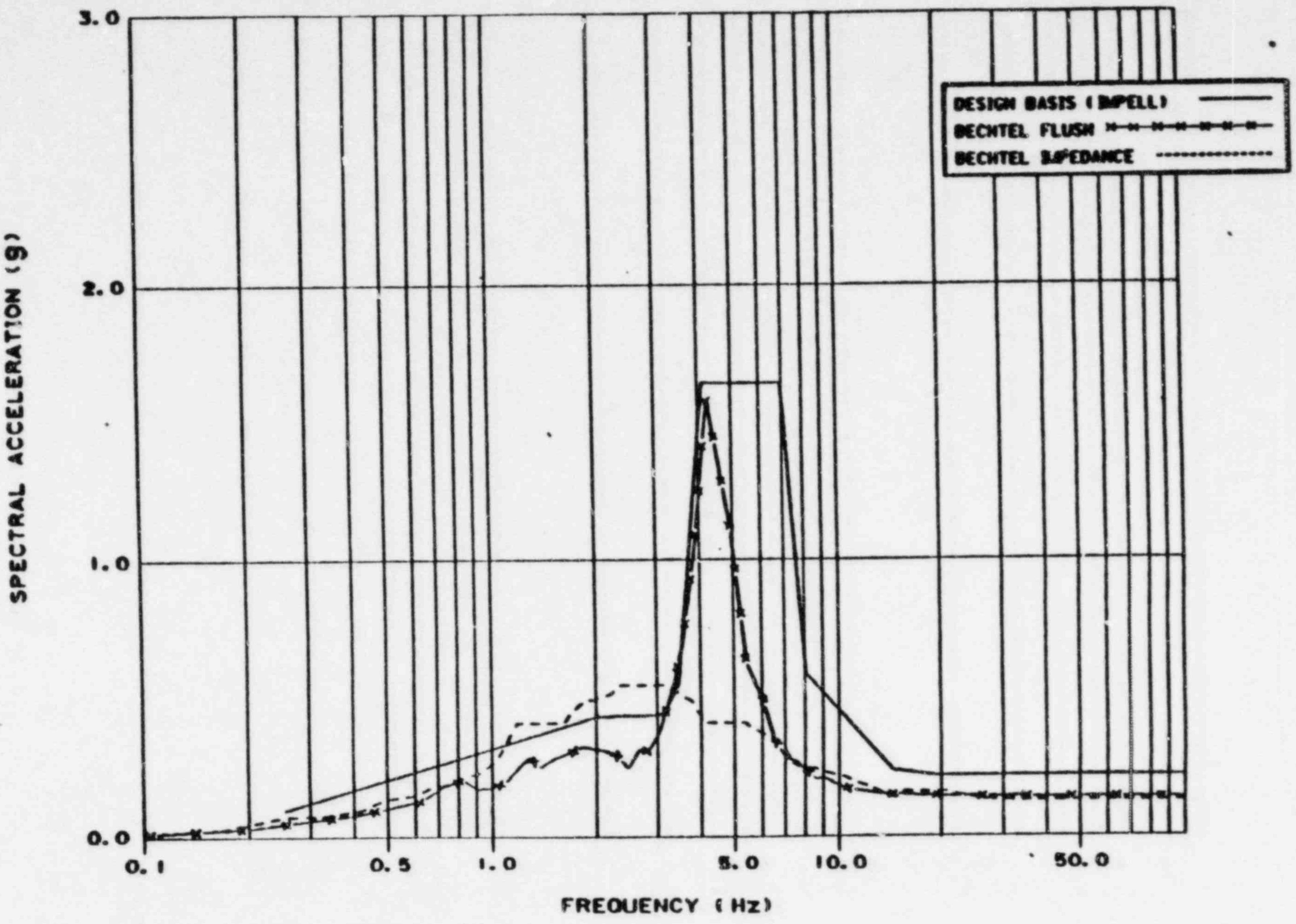

FIGURE A-13-33

DSER OPEN ITEM 51

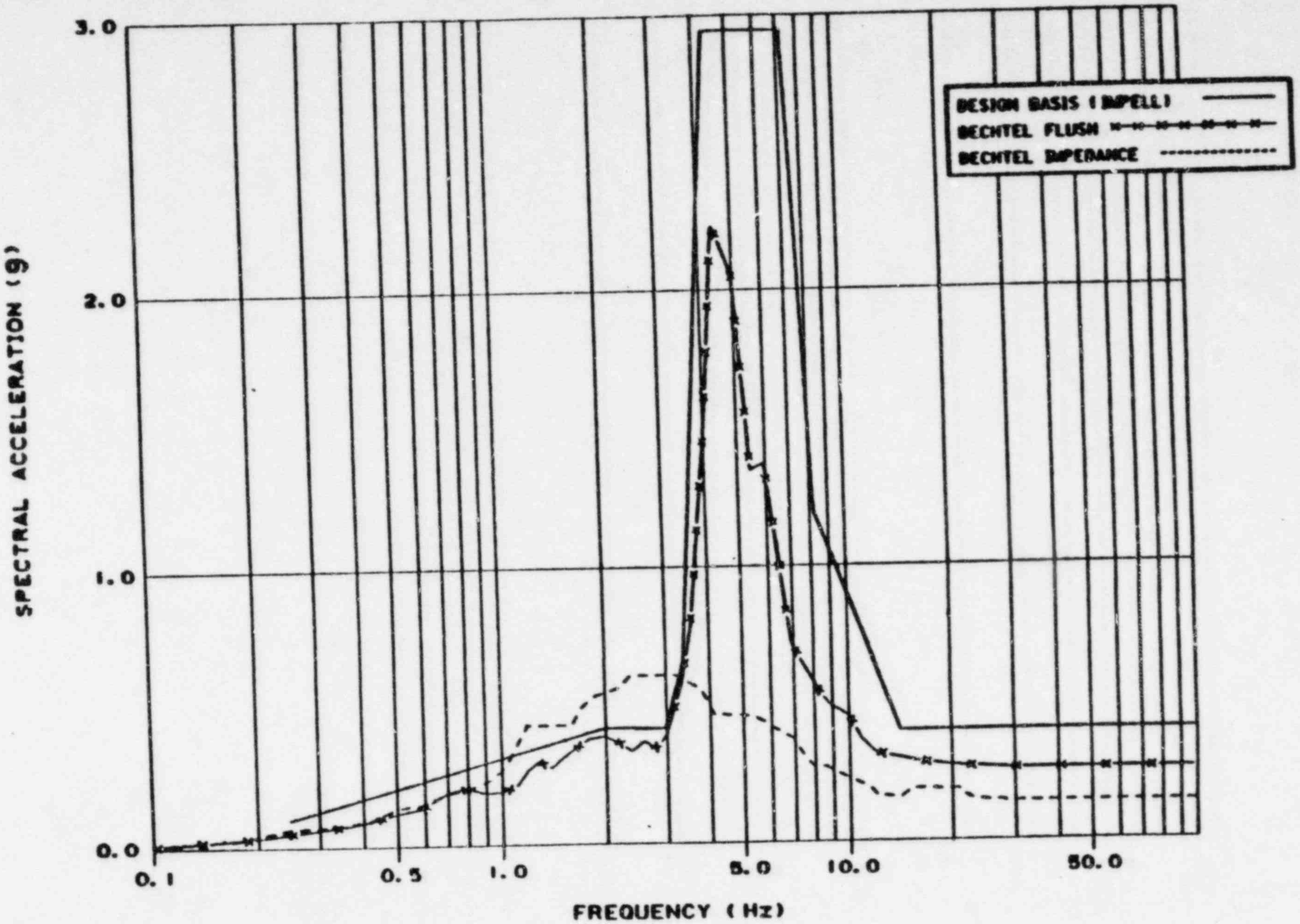

FIGURE A-13-34

DSER OPEN ITEM 51

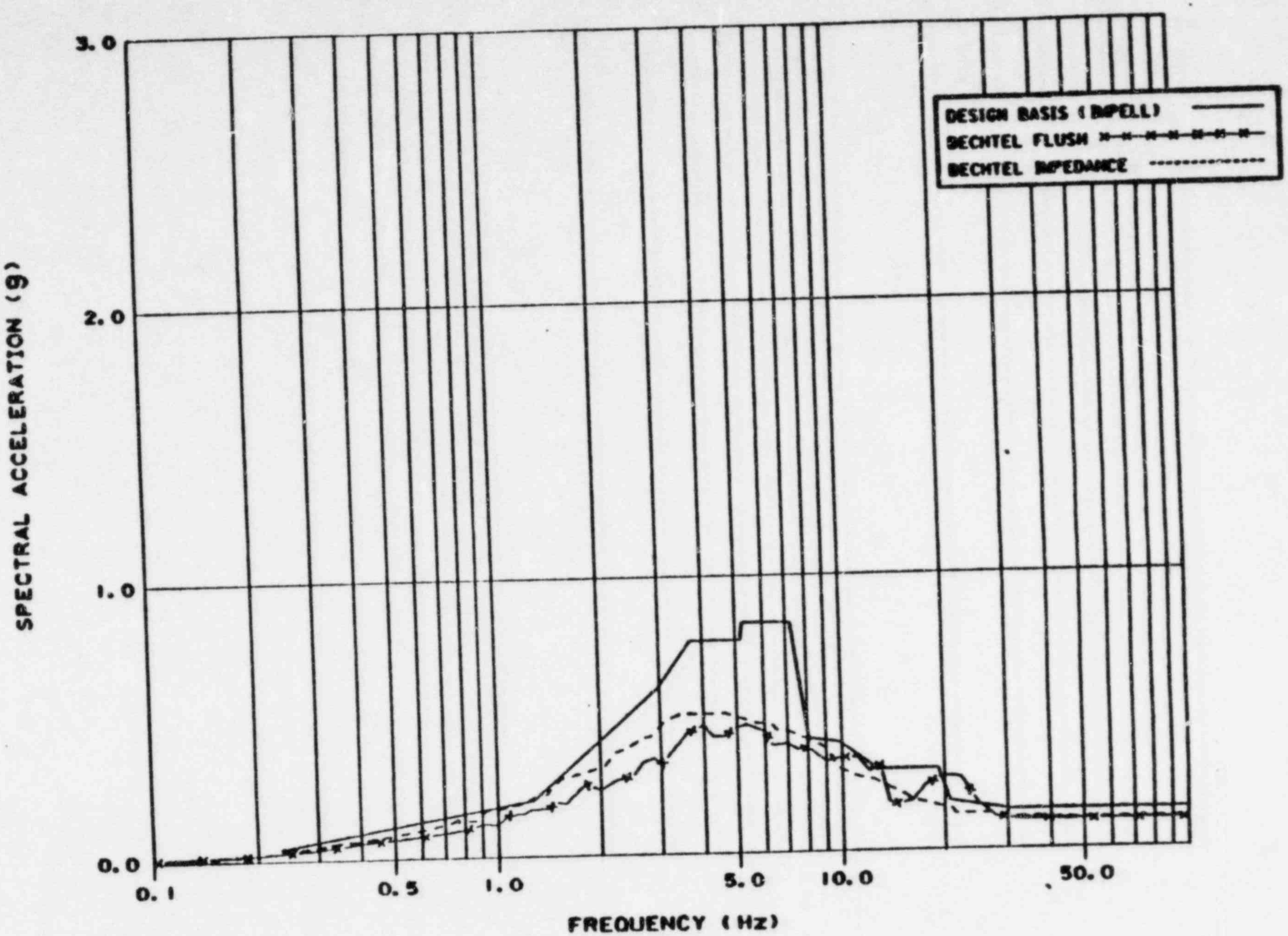

FIGURE A-13-35

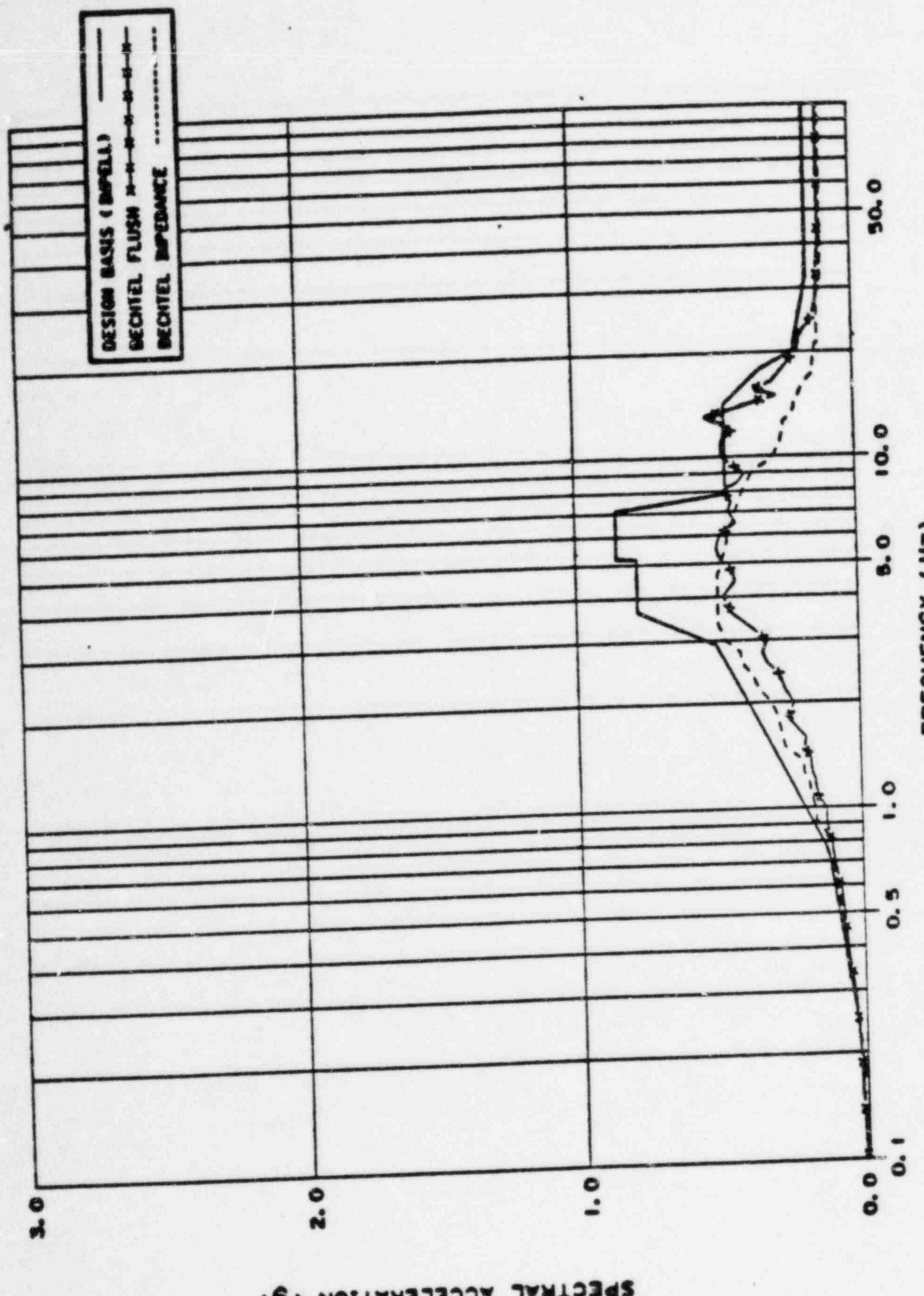

FIGURE A-13-36

DSER OPEN ITEM 51

DSER OPEN ITEM 5/

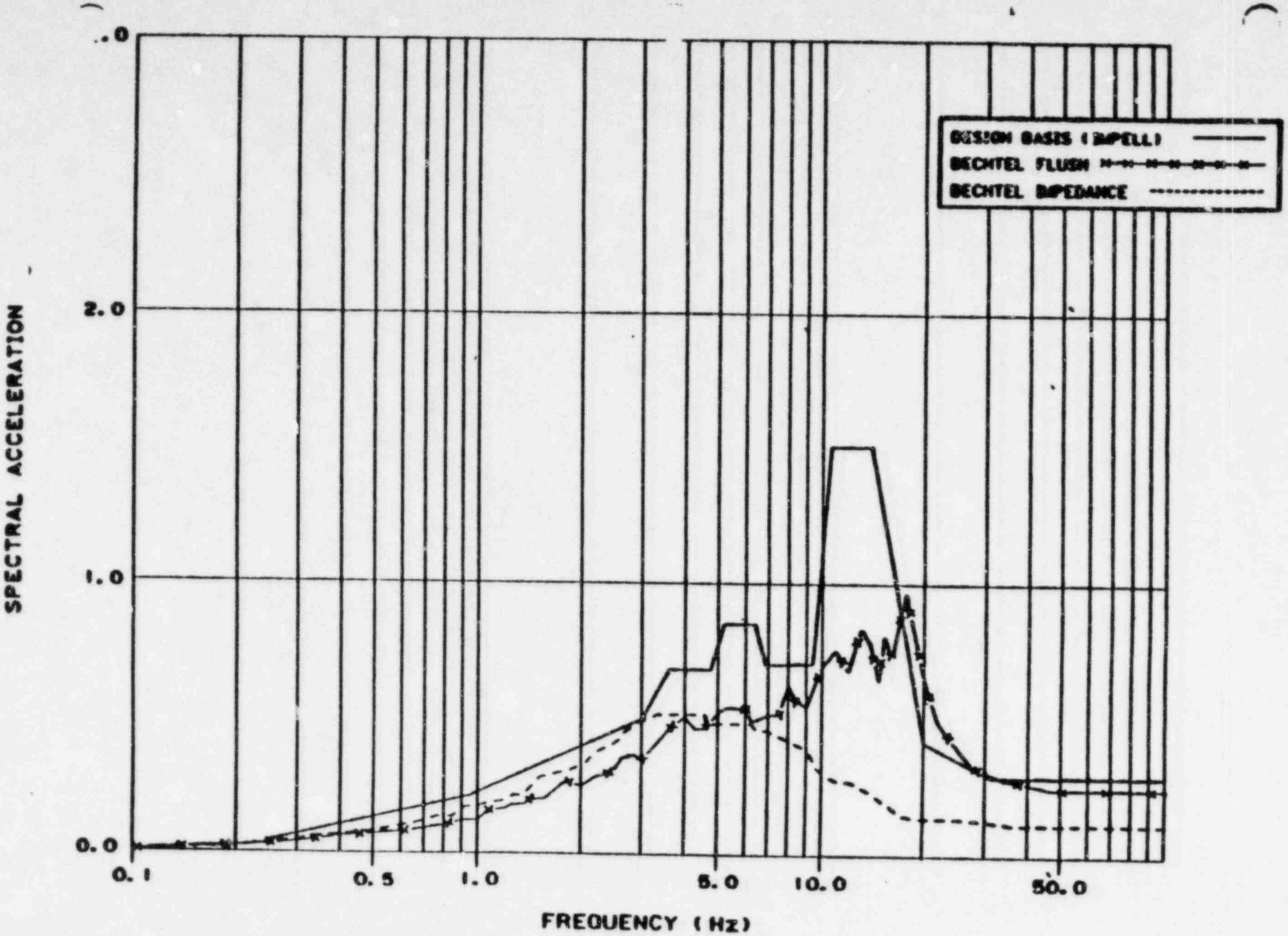

FIGURE A-13-37

RESPONSE SPECTRA COMPARISON,

AUXILIARY BUILDING AT ...

DSER Open Item No. 52 (DSER Section 3.8.6)

Ductility ratios due to pipe break

RESPONSE

This item corresponds to item A.16 from the NRC Structural/Geotechnical meeting of January 10, 1984. A discussion of the ductility ratios due to pipe breaks is attached.

Response to NRC Audit

Meeting Date: January 10, 1984

Question No.: A.16

Revised Response
Revision 1
July 10, 1984

QUESTION: Provide calculations of ductility ratios due to pipe break for key elements.

RESPONSE: FSAR Section 3.8.4.8.2 discusses the allowable ductility ratios used for the design of pipe whip restraints. For flexure in beams, an allowable ductility ratio of 20 is used.

As discussed with the NRC Staff, originally the majority of the pipe whip restraints had ductility ratios less than or equal to 10. However, the ductility ratios for approximately 25% of the pipe whip restraints exceeded 10 under the original design basis. These restraints have been reevaluated based on as-built conditions, final pipe break loads and actual hot gap requirements. This reevaluation revealed that all flexural members for pipe whip restraints have an actual ductility ratio of less than or equal to 10.

FSAR Section 3.8.4.8.2 will be revised to reflect compliance with SRP 3.5.3 for actual ductility ratios of flexural members.

DSER OPEN ITEM 52

DSER Open Item No. 66 (DSER Section 3.8.6)

Impedance analysis for the intake structure

RESPONSE

This item corresponds to item A.16 from the NRC structural/Geotechnical meeting of January 11, 1984. A comparison of the impedance analysis results with the design basis results ~~are~~ is attached.

Meeting Date: January 11, 1984

Question No.: A-16

Question: Perform an independent seismic verification analysis (impedance analysis) for the intake structure and compare the results with design basis results. Consider the effects of side boundaries, embedment and the presence of water masses in the analysis.

Response:

In accordance with the requirements of the Standard Review Plan, Section 3.7.2 (NUREG 0800), impedance approach (half-space) seismic soil-structure interaction verification analyses of the service water intake structure (SWIS) are performed by Bechtel. The analytical method used for the impedance approach seismic soil-structu interaction analyses of the SWIS is described in FSAR Section 3.7.2.1. The effects of side boundaries and embedment are consider using the method described in References A-16-1 to A-16-3. The effects of water masses are also accounted for by adding effective water mass to the related nodal points of the structural model in accordance with procedures described in Reference A-16-4.

Figures A-16-1 to A-16-18 show the comparison of the 2 percent damping response spectra obtained from the design basis finite element and the impedance approach seismic soil-structure interaction analyses. The impedance approach response spectra generally are enveloped by those obtained from the design basis analyses at elevation 114.0 feet of the SWIS. For other elevations, the impedance approach spectral accelerations exceed the design basis spectral accelerations in some frequency ranges. These ranges vary approximately between 1.5 and 10.0 Hz.

As discussed during the January 1984 NRC Structural Audit Meeting, sampling studies have been performed to confirm the adequacy of the SWIS design. The criteria used in selection of the samples for this study is given in Table A-16-1. The results of the sampli studies are as follows:

1. <u>Structure</u>

   All major reinforced concrete shear walls at the base of the intake structure have been evaluated for seismic forces and moments obtained from the impedance approach analyses. The she stresses resulting from the impedance approach analyses were compared with those of the design basis analyses. Table A-16-2 shows comparison of shear stresses. In all cases these revisec shear stresses were found to be within the allowables.

   The moments in the walls, obtained from the impedance analyses, were smaller than those of design basis analyses for both the East-West OBE and SSE cases, therefore, no further evaluation c these walls is required.

DSER OPEN ITEM 66

For North-South OBE and SSE cases, the moments obtained from impedance approach analyses exceeded the design basis moments. The increase in moments were mostly isolated to the eastern portion of the intake structure. This portion of the intake structure was reevaluated and the resulting moments were found to be less than the allowables.

Based on the above, it is concluded that the as-built SWIS can accommodate loads obtained from the impedance approach analyses

2. Equipment

The effects of the impedance approach response spectra was evaluated on 8 types of seismic category I equipment located in the areas where the impedance approach spectra were found to have higher spectral accelerations than those of the design basis response spectra. The equipment evaluated represents over 30% of all equipment located in the intake structure.

Table A-16-3 summarizes the results of the above evaluation for equipment in the Intake Structure. It is concluded that all category I equipment can accommodate the response spectra obtained from the impedance analyses.

3. Cable Tray and HVAC Supports

All cable tray and HVAC supports were evaluated using the impedance analysis results. All supports were found to meet the impedance approach spectral response requirements.

4. Piping and Piping Supports

Piping and pipe supports were evaluated using the screening techniques discussed in Table A-16-1. The results are summarized in Tables A-16-4 and A-16-5. The analysis results show that piping stresses and nozzle loads are within allowable limits. There was no load increase found on existing supports.

It is therefore concluded that the existing design margins associated with the present project design basis seismic loadir are not affected by the consideration of the loads generated from the impedance approach analyses as demonstrated by the SWI piping systems.

References: A-16-1, Apsel, R.J., (1979) "Dynamic Green's Functions for Layered Media and Applications to Boundary Value Problems", Ph.D Thesis, University of California, San Diego.

Response to Question A-16 (cont'd)

References: (Cont'd)

A-16-2, Wong, H.L., and Luco, J.E., (1978) "Tables of Impedance Functions and Input Motions for Rectangular Foundations", Report No. CE78-15; University of California, San Diego.

A-16-3, Barneich, J.A., Johns, D.H., and McNeill, R.L., (1974) "Soil-Structure Interaction Parameters for Aseismic Design of Nuclear Power Stations", Preprint 2182, ASCE National Meeting on Water Resources Engineer January 21-25.

A-16-4, Newmark, N. and Rosenblueth, E., "Fundamentals of Earthquake Engineering," Prentice-Hall, Englewood Cliffs, N.J. (1971)

TABLE A-16-1
PROCEDURES FOR EVALUATION OF INTAKE STRUCTURES, EQUIPMENT & COMPONENTS USING IMPEDANCE ANALYSIS RESULTS

INTRODUCTION

The results of the impedance analysis are used to assess the existing design of the HCGS intake structure, 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 structure, and the design margins for the upper elevations are greater than those of the base, the effects of these loads at the base of the structure are evaluated.

B. EQUIPMENT:

The impedance analysis spectra in general are not completely enveloped by the design basis spectra in the 1.5 to 10.0 Hz and in the ZPA range throughout the intake structure.

The following procedure is selected for review:

- Review the significant frequencies of at least 30% of equipment located in the areas where the impedance approac spectra were found to have higher spectral accelerations than those of the design basis response spectra.
- If the significant equipment frequencies fall in the range where the difference in the spectra exist, additional eva uation is necessary. No further evaluation is necessary the significant frequencies are outside the frequency ran in question.
- The evaluation is performed either by comparing the test response spectra of the equipment with the impedance spec (if the equipment is qualified by testing) or comparing t actual-to-allowable stress ratios with the spectrum excee ance 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 modifica tions, the sample size for this evaluation is expanded as required.

C. CABLE TRAY AND HVAC SUPPORTS

All cable tray and HVAC supports are evaluated for impedance analysis results.

D. PIPING AND PIPE SUPPORTS

In general, impedance curves resulted in significant reduction: in spectral accelerations as compared to those of the design basis curves. However, in some curves, the peak accele: ations showed small increases. To evaluate the effects of the increase in peak accelerations a "biased" sample of affected piping systems is reanalyzed and reevaluated. The sample is selected as follows:

Individual impedance curves for various elevations and structu are superimposed on their corresponding design basis curves to identify those impedance curves which are not enveloped by des 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 envelo response spectra curves affecting a calculation did not totall envelop all the corresponding impedance curves, that particula calculation is then identified as "affected" calculation 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 sele to include curves showing significant differences.

3. High equipment nozzle loads in existing calculation.

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 sampl
Intake Structure	11	11	5	1

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

Table A-16-2
Intake Structure Shear Stress at the Base

Base Elevation	Wall Location Column Line	Design Base (psi)	Impedance Approach (psi)	Allowable (psi)
79'-8"	Col. A (East Wall)	80	124	630
79'-8"	Col. Ac	66	98	630
79'-8"	Col. Ak	47	73	630
70'-0"	Col. C (West Wall)	47	77	126
79'-8"	Col. 5 (South Wall)	230	214	630
79'-8"	Col. 7	200	176	630
79'-8"	Col. 9 (North Wall)	230	214	630

Notes: 1. Concrete f'c = 4000 psi.
2. See FSAR Figures 1.2-40 and 1.2-41 for wall location.

Table A-16-[illegible]

<u>Intake Structure - Seismic Category I Equipment</u>

Equipment or Component	Tag No.	Elev.	Fundamental Frequencies (Hz)	Method of Seismic Qualification	Applicable Note
Travelling Water Screen (T.W.S.)	1(A-D)S501	70'-0" & 114'-0"	Horizontal - 7.4, 14 Vertical - >33	Analysis	2
Control Panel (for T.W.S.)	1(A-D)C515	107'-0"	Horizontal - 21, 30 Vertical - >33	Testing	1
Service Water Pumps	1(A-D)P502	93'-0"	Horizontal - 28.4 Vertical - >33	Analysis	3
Supply Fans	0AV558 0BV558	128'-0"	Horizontal - >33 Vertical - >33	Analysis	2
Vane Axial Fans	1AV-DV503 1AV-DV504	122'-0"	Horizontal - >33 Vertical - >33	Analysis	3
HVAC Control Panel	1(A-D)C581	93'-0"	Horizontal - 15, 22 Vertical - >33	Analysis	2
Travelling Screen Spray Water Booster Pumps	1AP-DP507	79'-8"	Horizontal - >33 Vertical - >33	Analysis	2
Transformer Panel Board	10Y501-504	93'-0"	Horizontal - 29,31 Vertical - >33	Testing	1

Notes:

1. TRS envelops 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-16-4

Intake Structure Pipe Stress Summary

Calc. No.	Max. Seismic Stress Ratios: Max. Impedance Stress / Max. Design Basis Stress		ASME Code Equation Evaluation		Vendor Equipmen[t] Nozzle Allowabl[e] Met
	OBE	SSE	Eq. 9B* / Code Allow. Upset	Eq. 9D* / Code Allow. Faulted	
C2019	0.46	0.51	0.26	0.14	Yes

*ASME Section III NC, ND-3652

Table A-16-5

Intake Structure Pipe Support Load Summary

Calc. No.	Total No. of Supports	No. of Supports with load increase	Average Percentage increase in load		Support Design Adequate
			Upset	Faulted	
C2019	15	0	N/A	N/A	Yes

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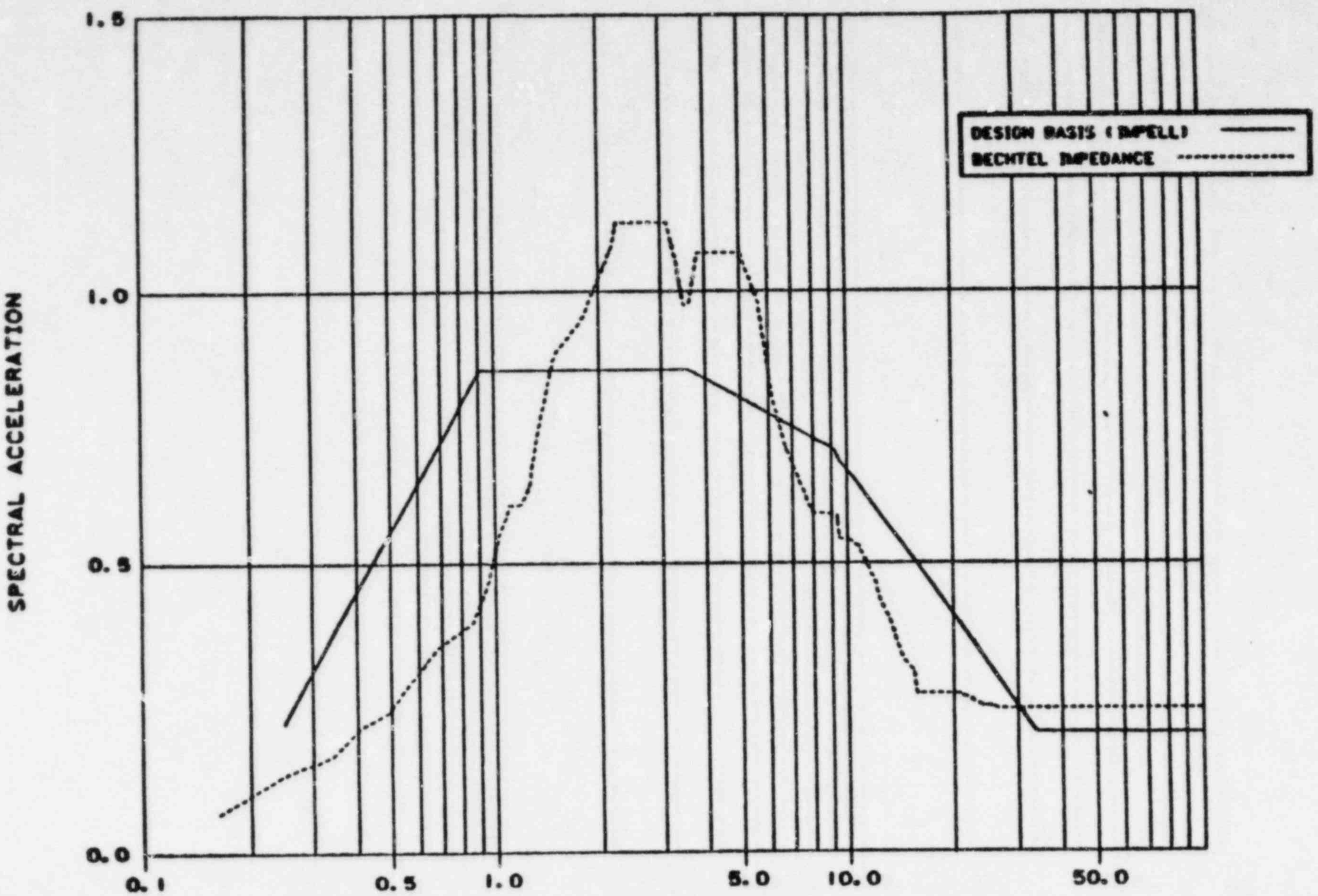

FIGURE A-16-1

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 93' -0".

DSER OPEN ITEM 66

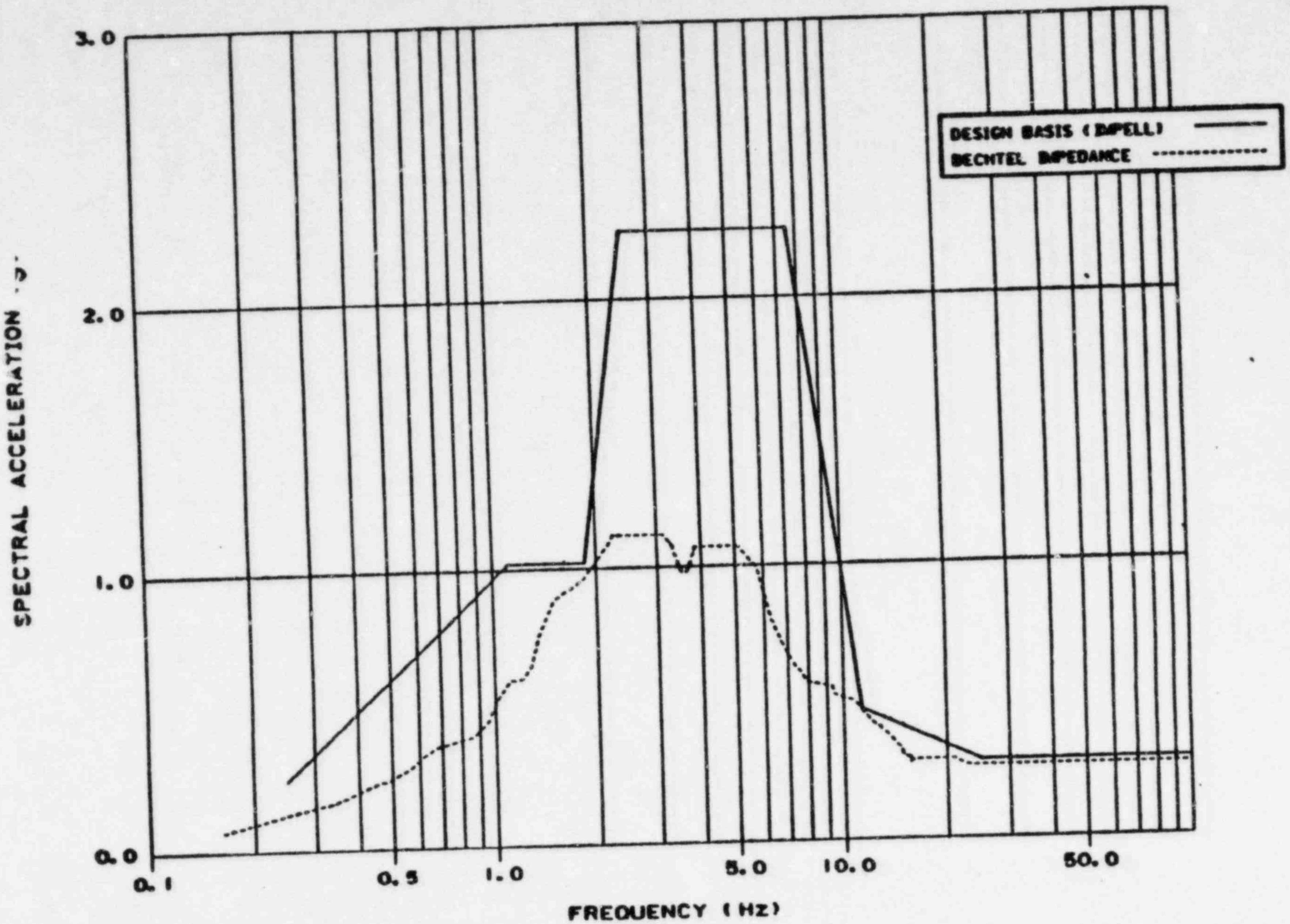

FIGURE A-16-2

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 114'-0",

HOPE CREEK UNIT 1

DSER OPEN ITEM 66

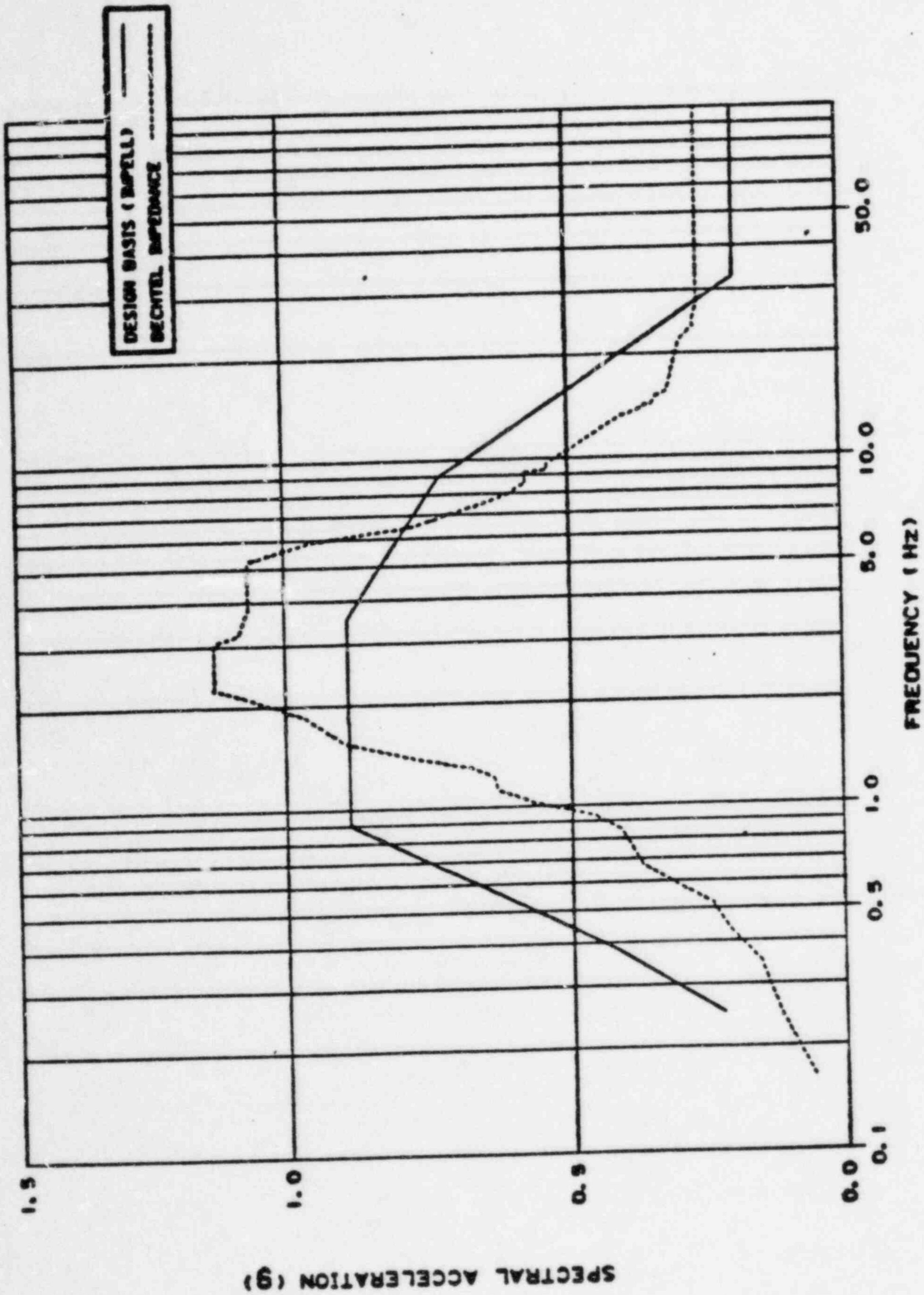

FIGURE A-16-3

DSER OPEN ITEM 66

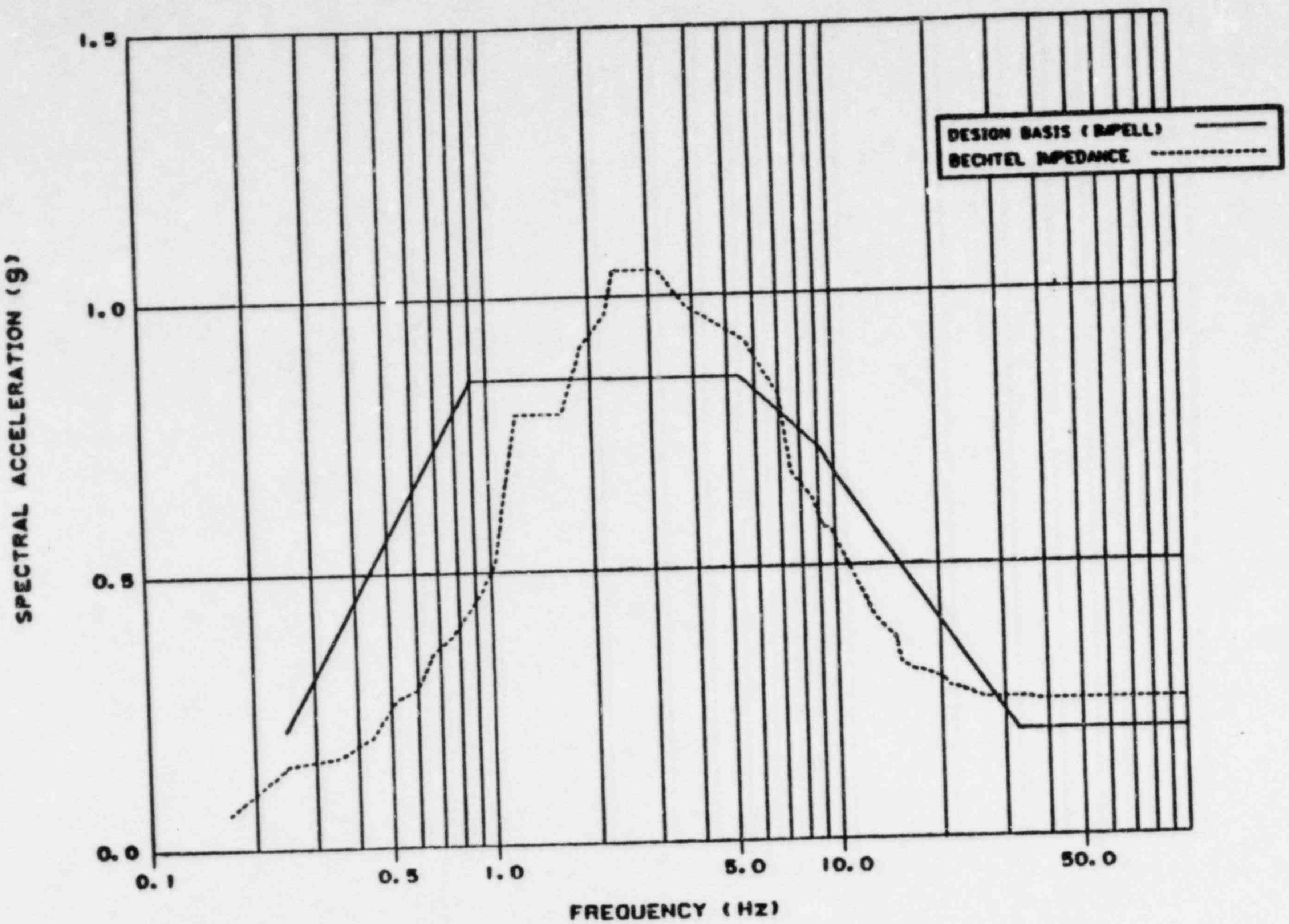

FIGURE A-16-4

DSER OPEN ITEM 66

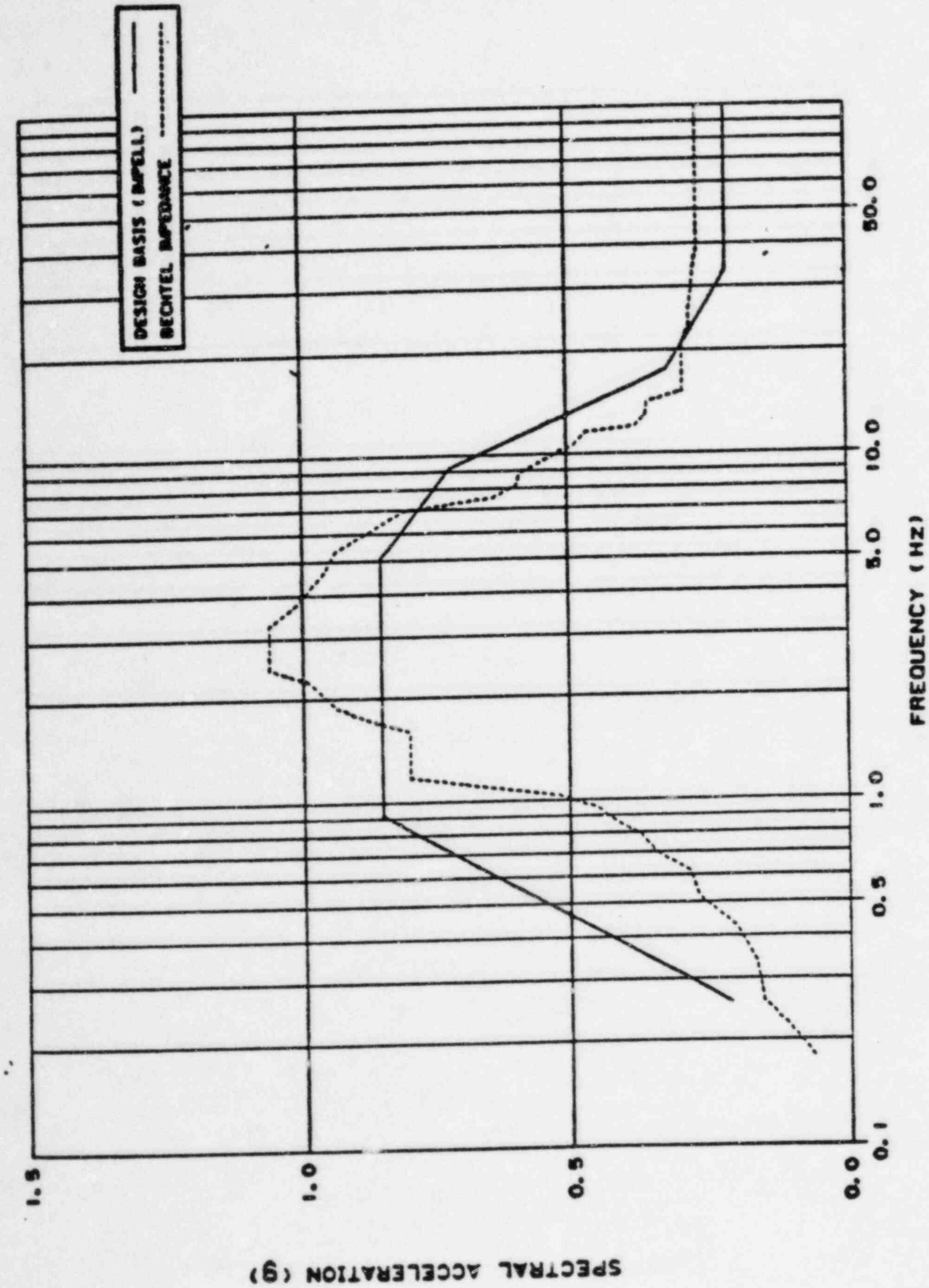

FIGURE A-16-5

DSER OPEN ITEM 66

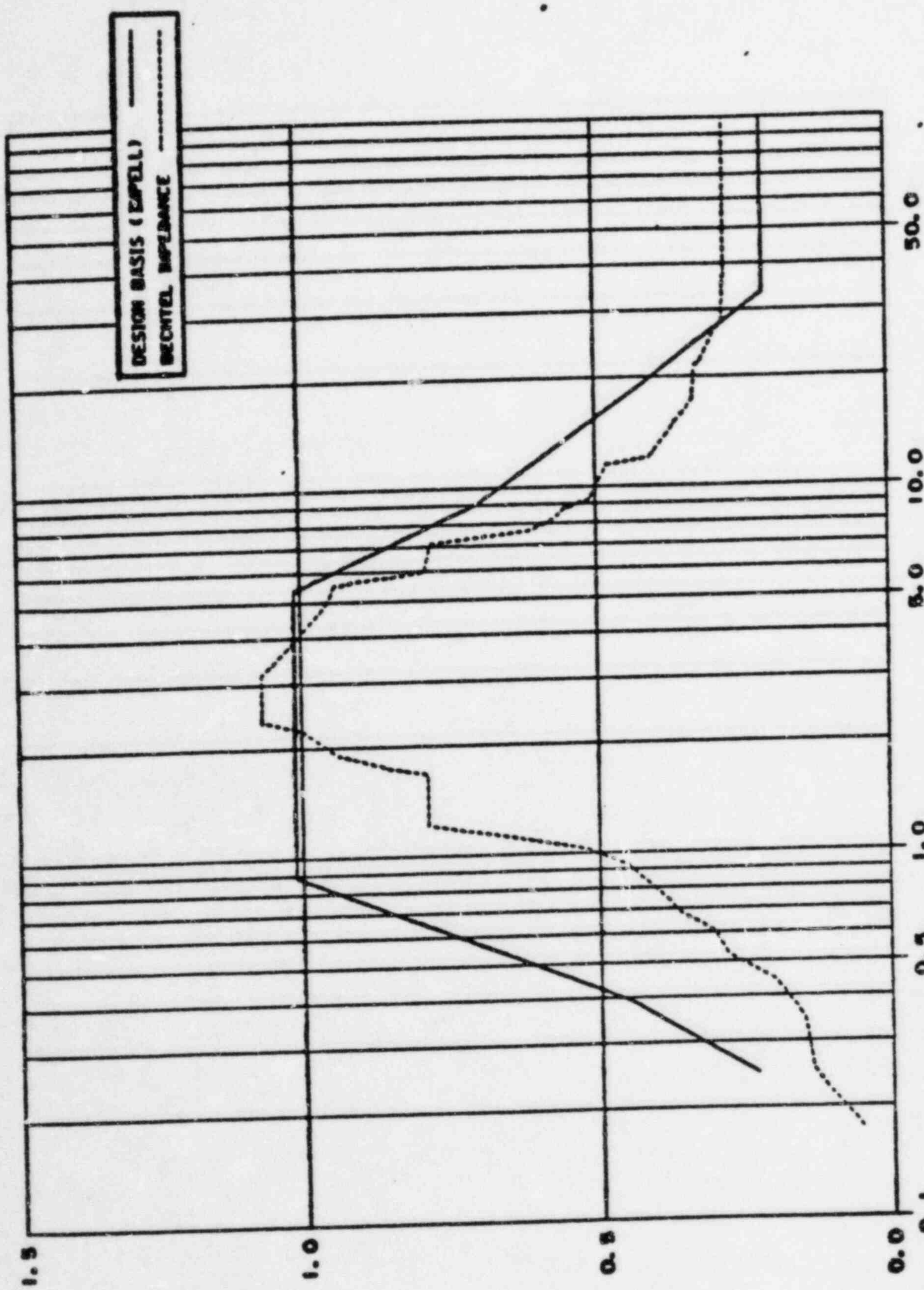

FIGURE A-16-6

DSER OPEN ITEM 66

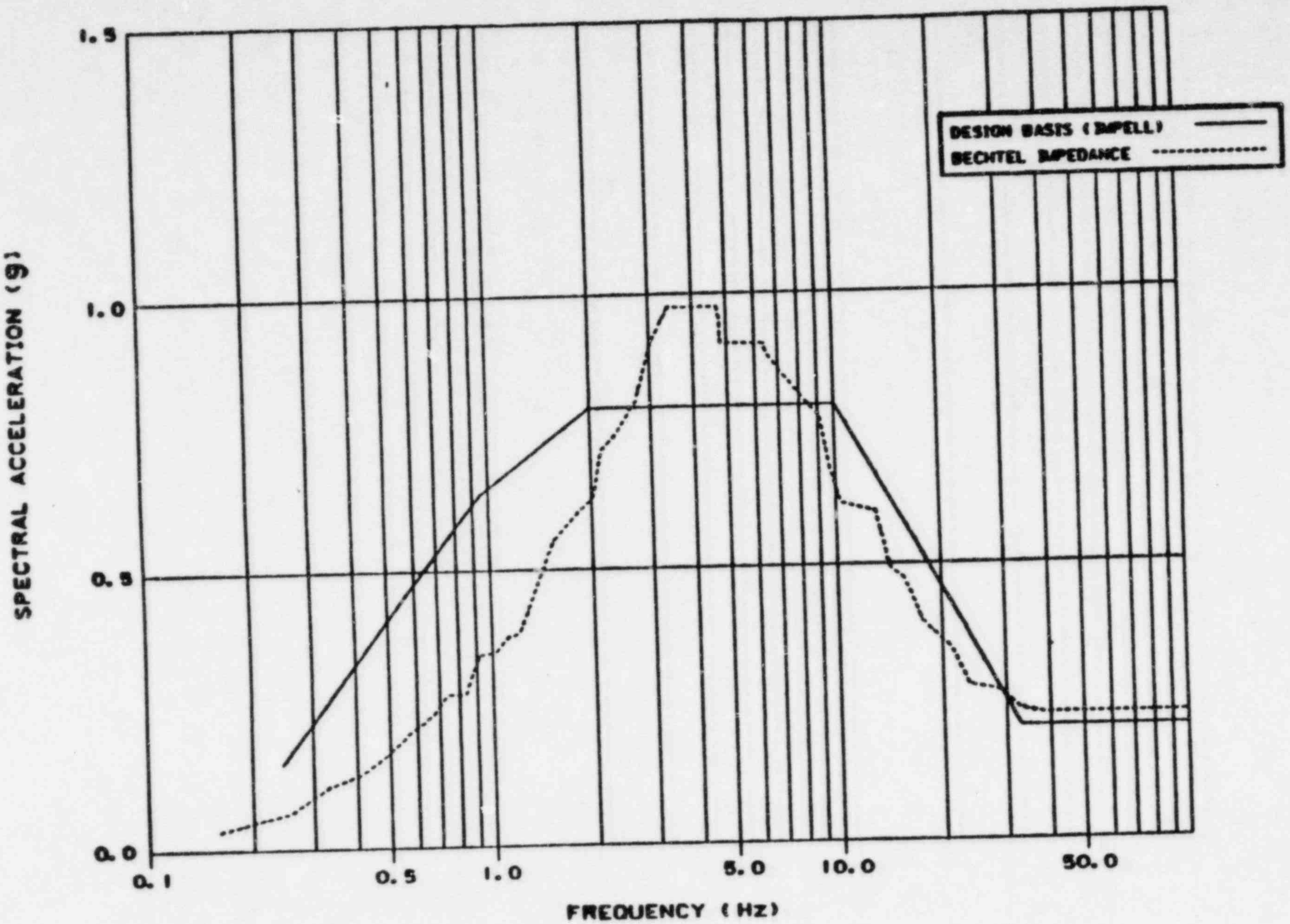

FIGURE A-16-7

DSER OPEN ITEM 66

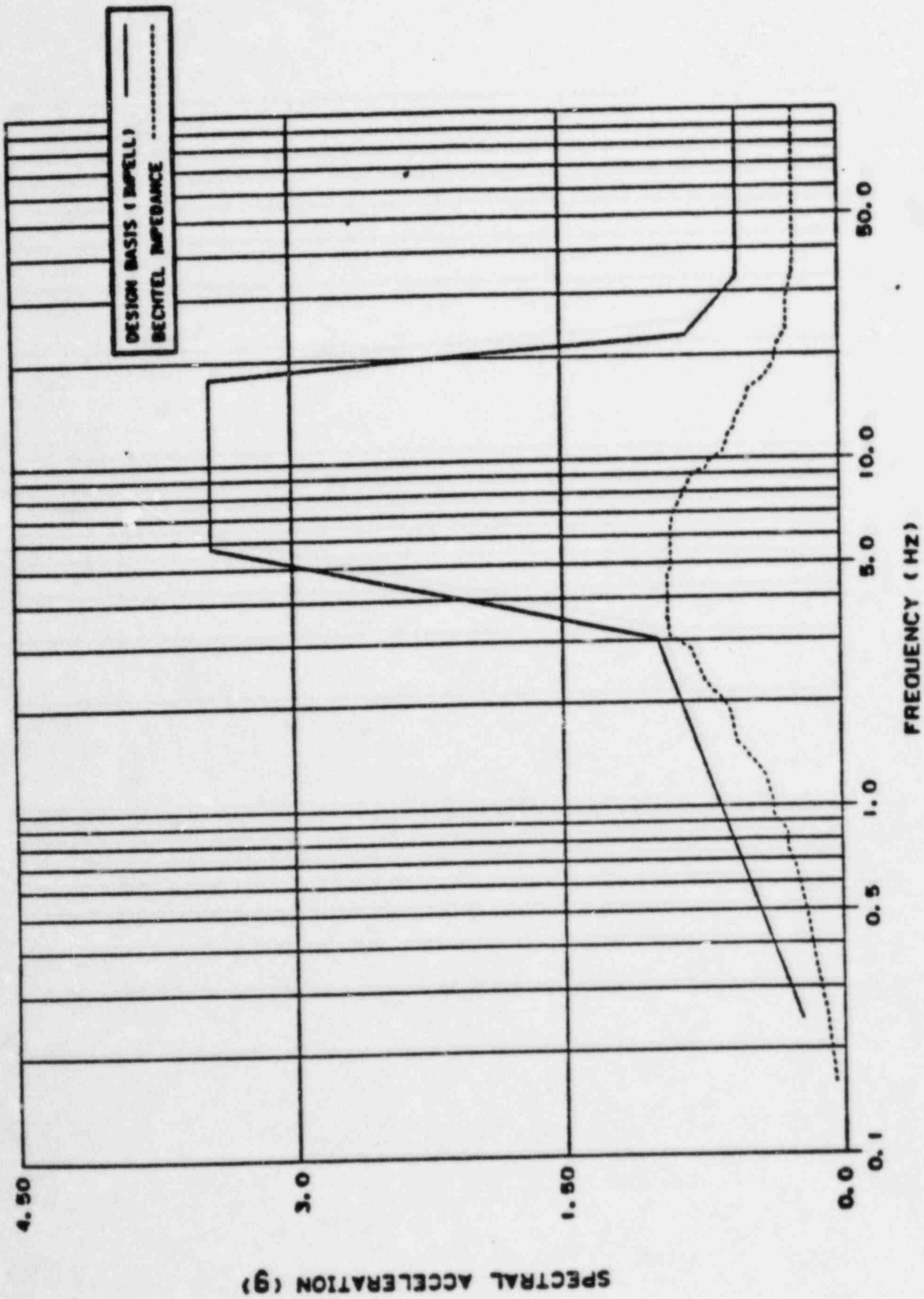

FIGURE A-16-8

DSER OPEN ITEM 66

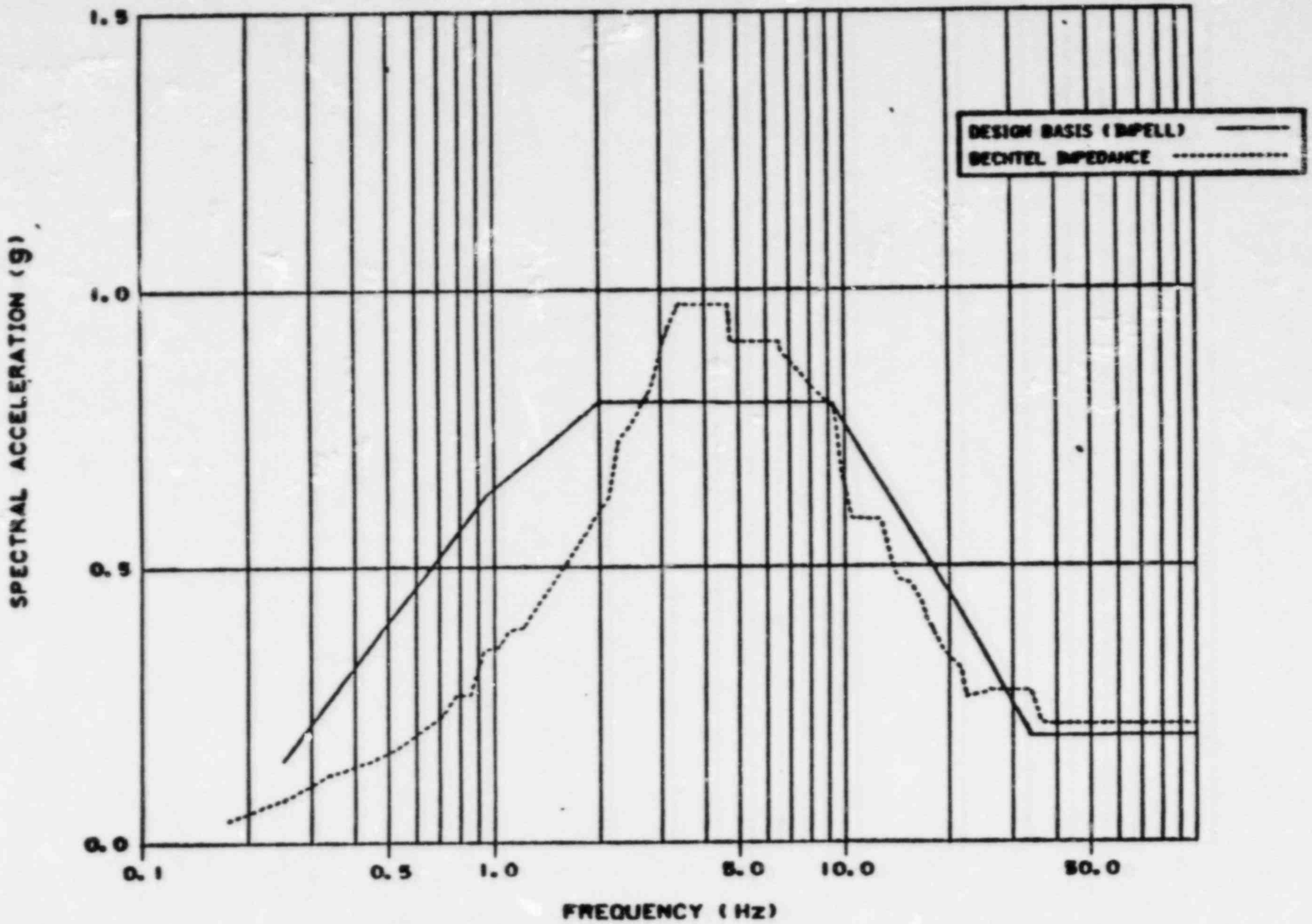

FIGURE A-16-9

DSER OPEN ITEM 66

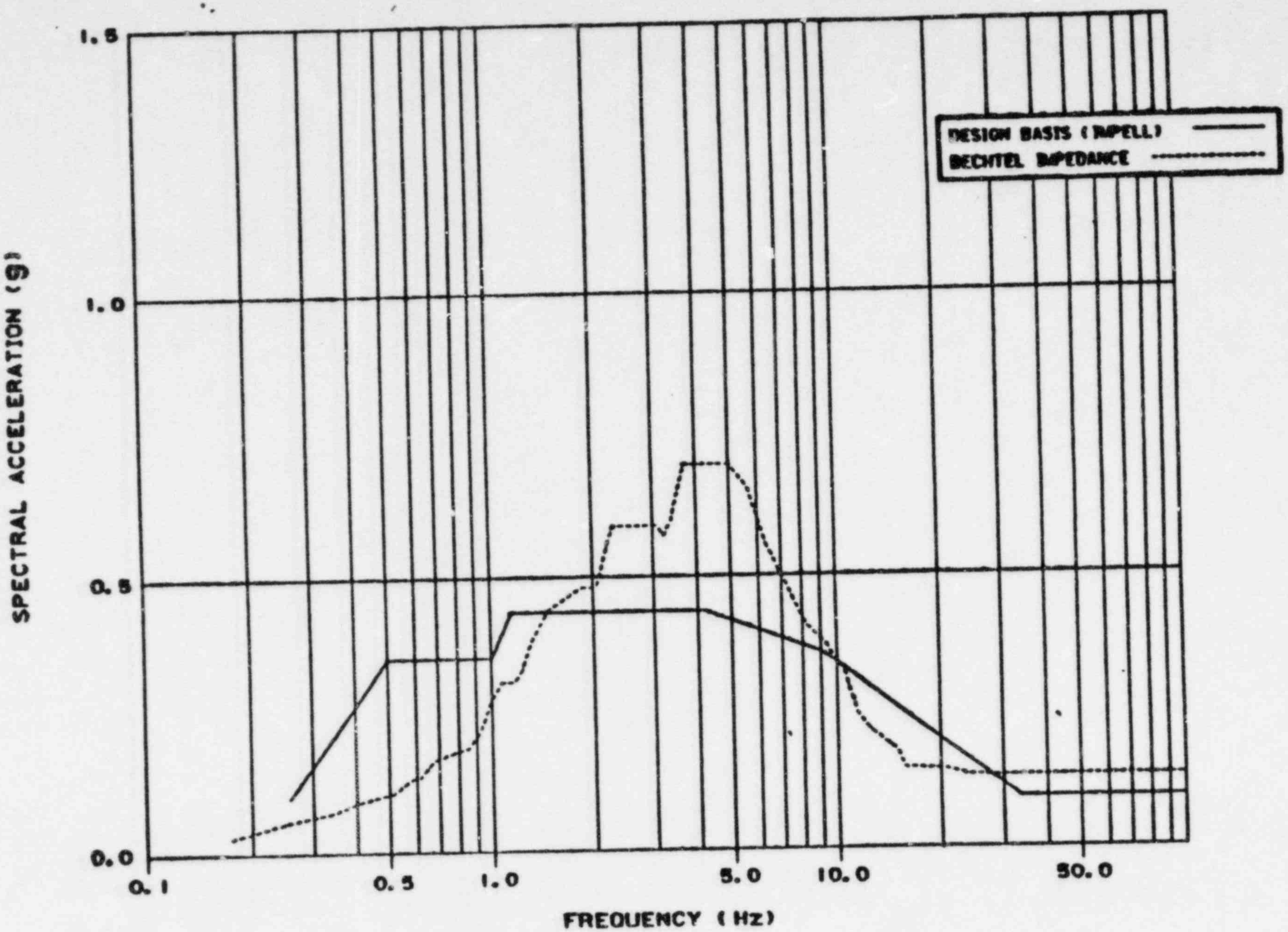

FIGURE A-16-10

DSER OPEN ITEM 66

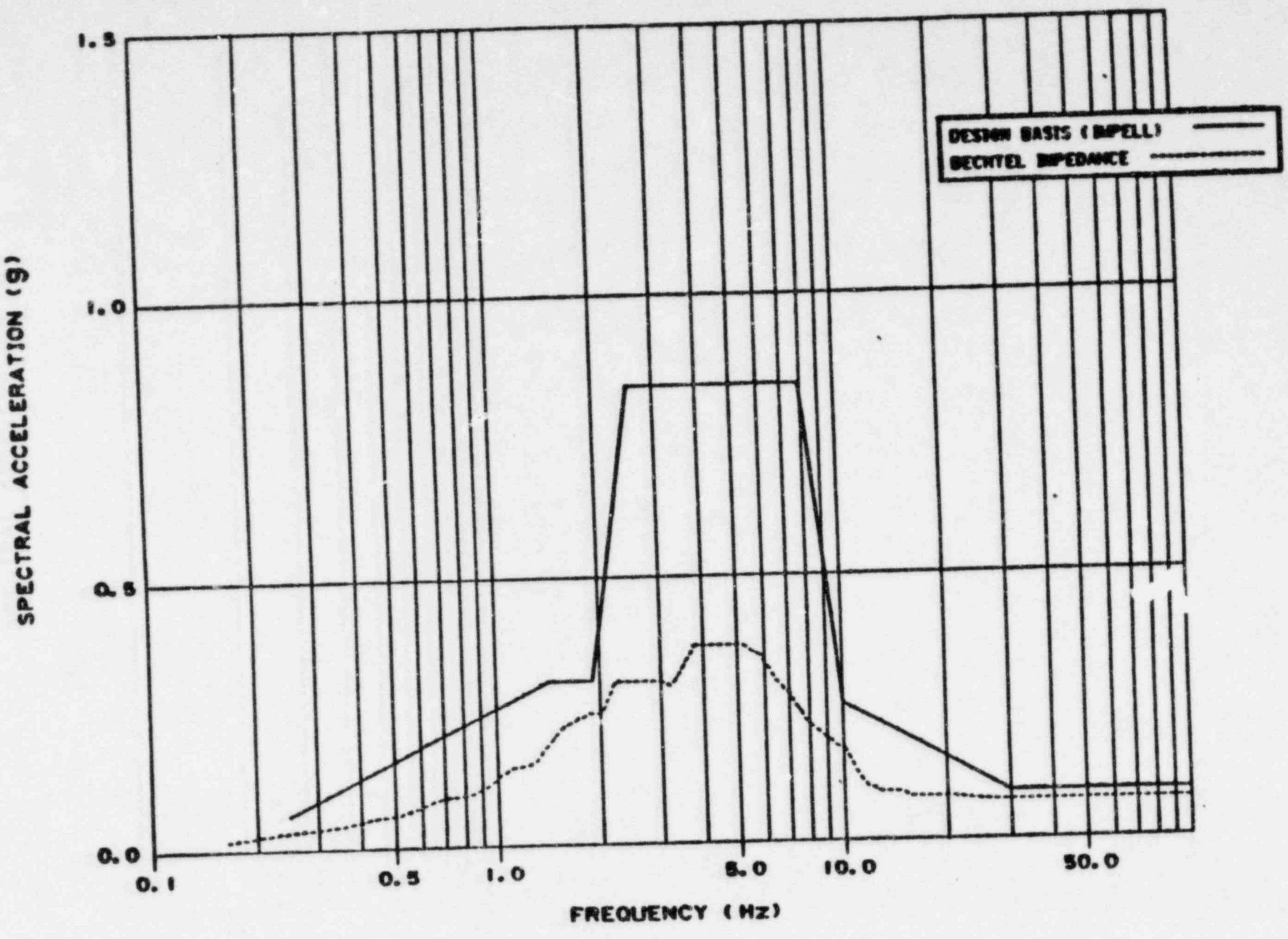

FIGURE A-16-11

DSER OPEN ITEM 66

DSER OPEN ITEM 66

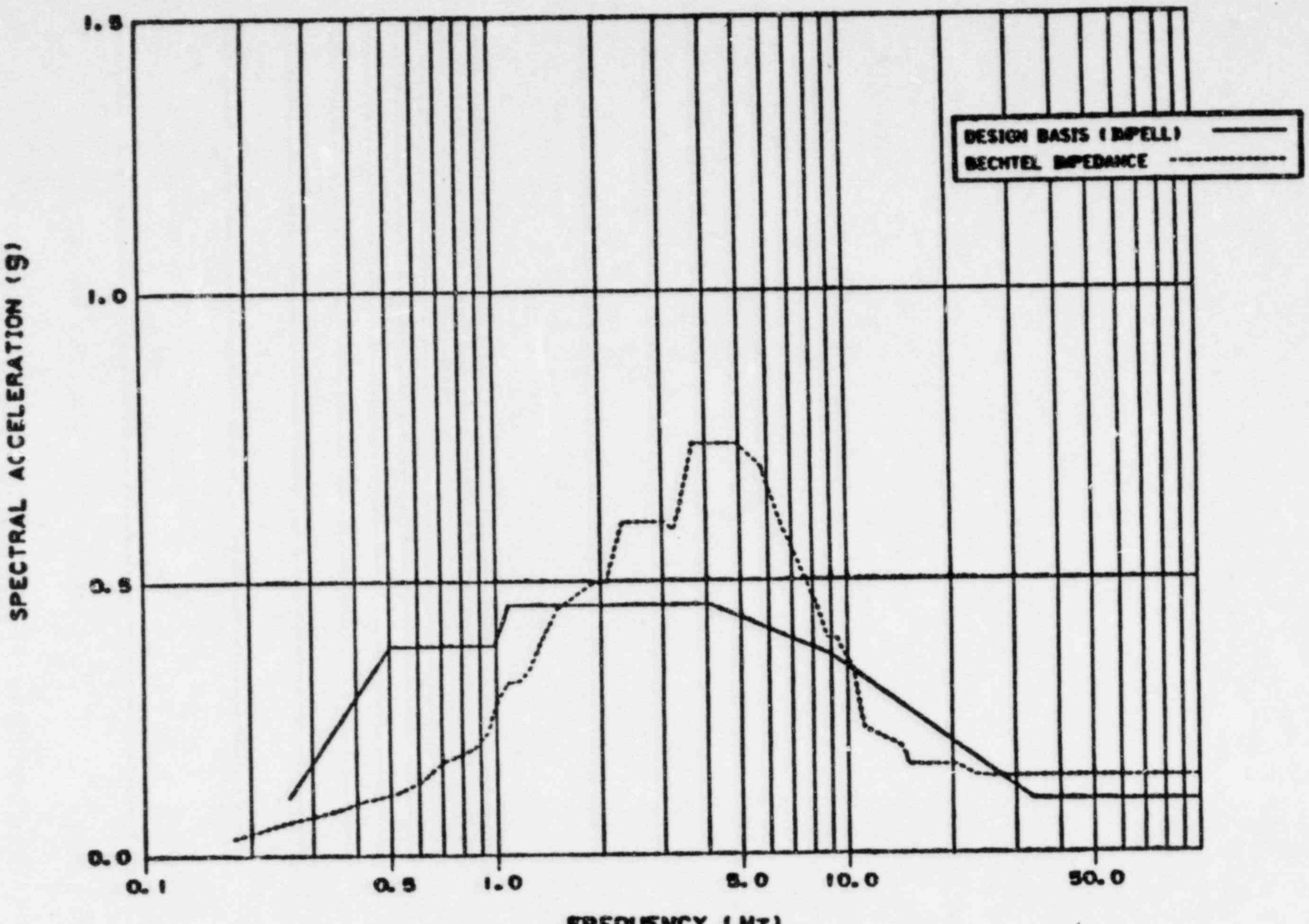

FIGURE A-16-12

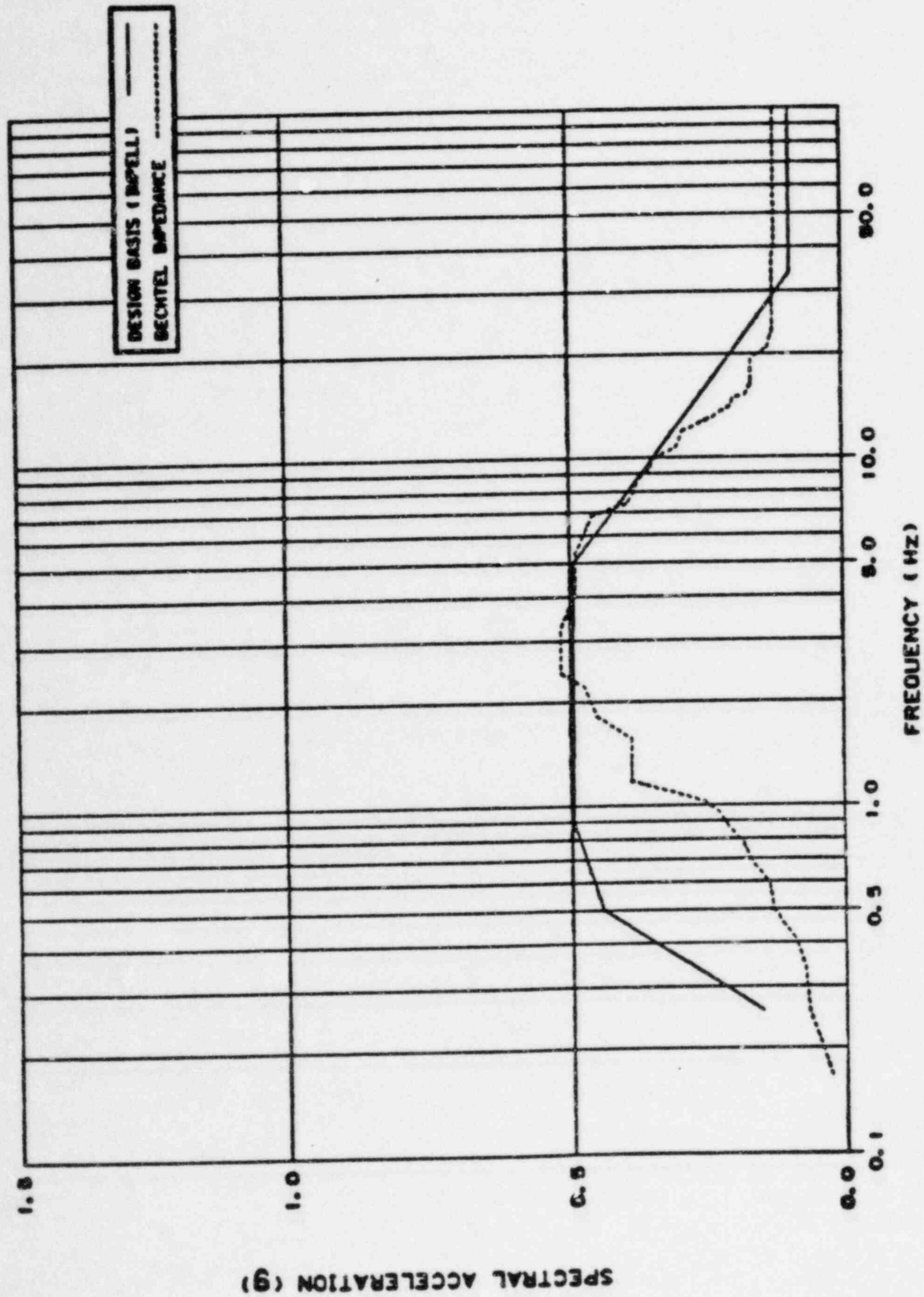

FIGURE A-16-13

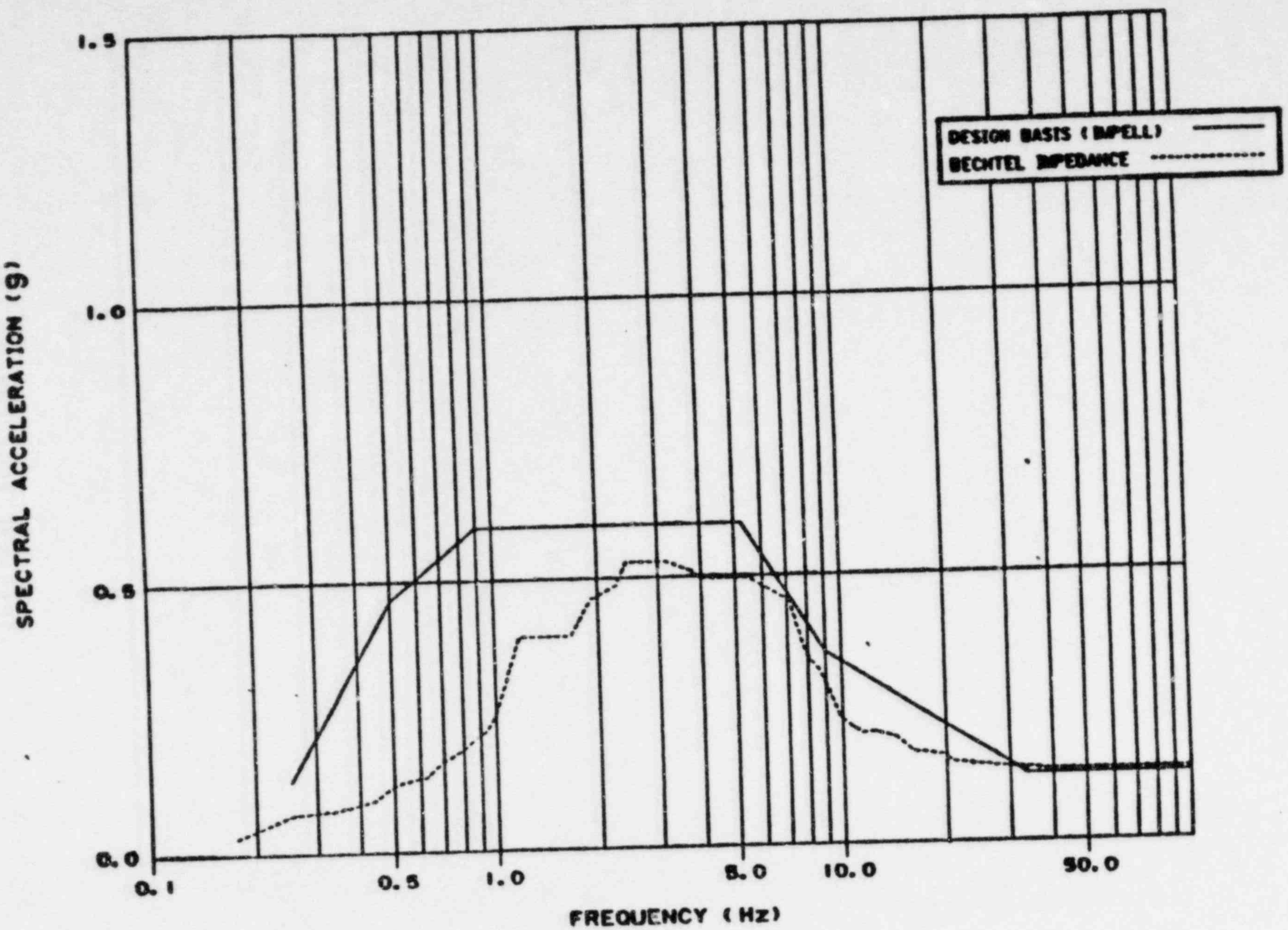

FIGURE A-16-14

DSER OPEN ITEM 66

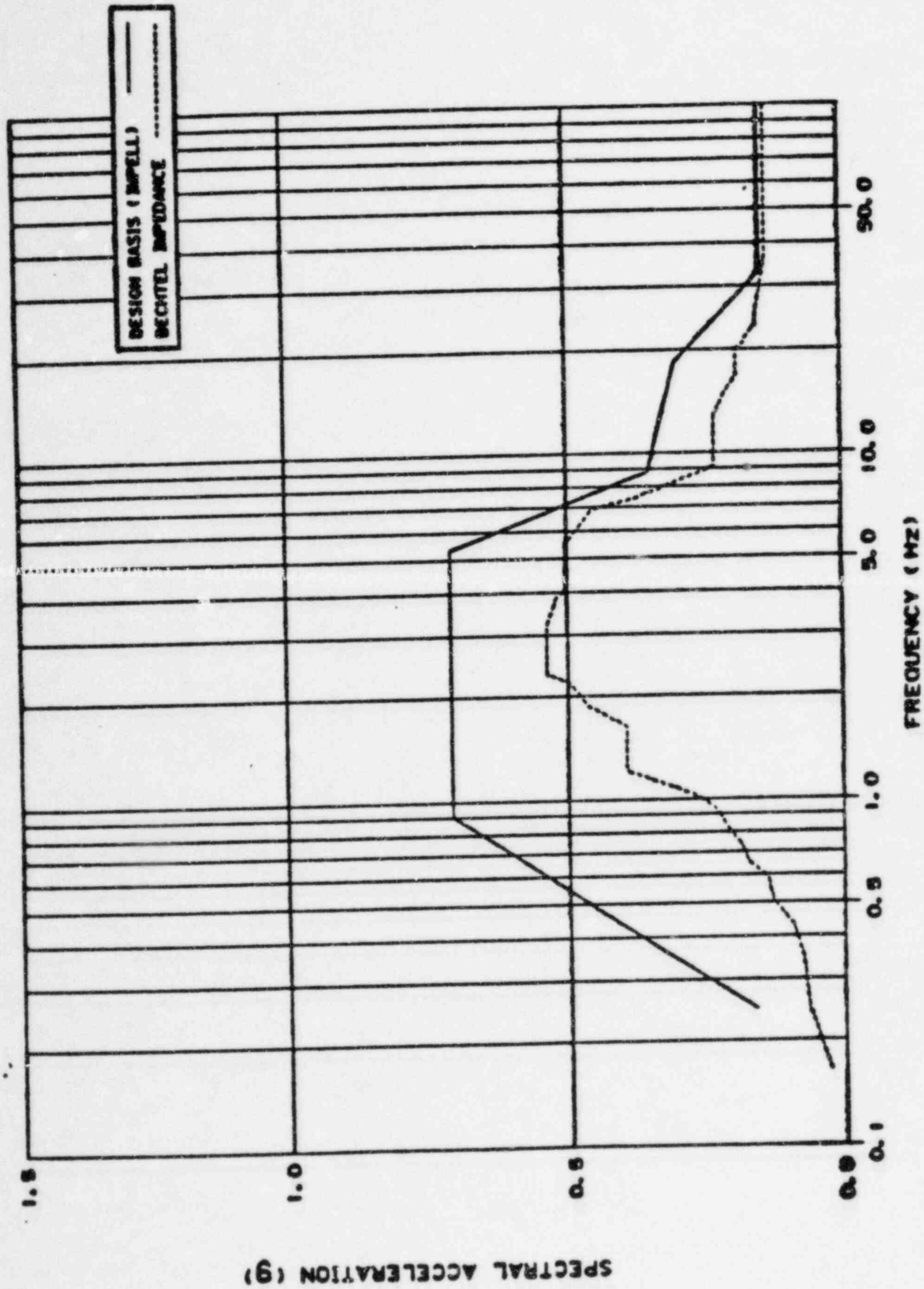

FIGURE A-16-15

DSER OPEN ITEM 66

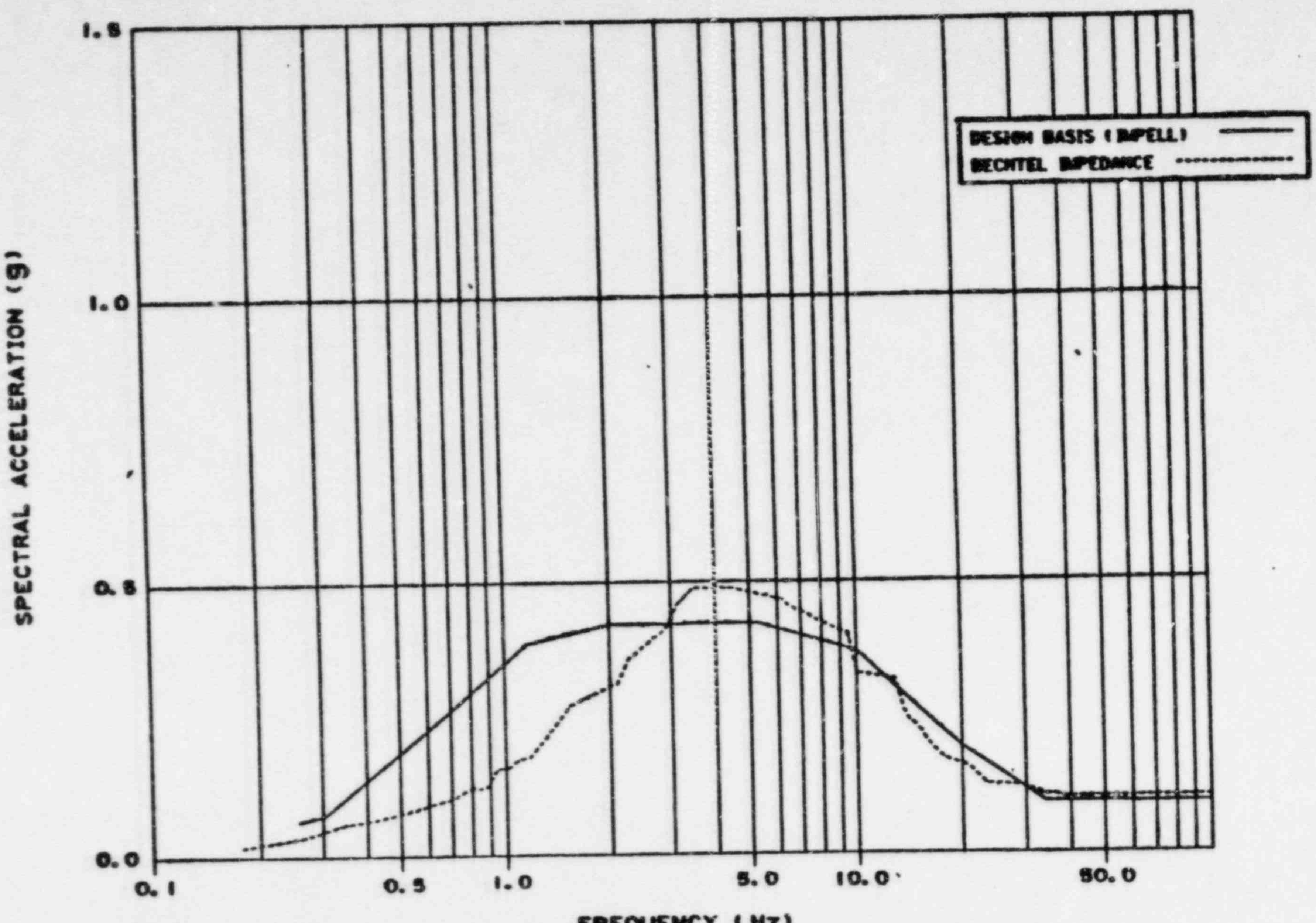

FIGURE A-16-16

DSER OPEN ITEM 66

DSER OPEN ITEM 66

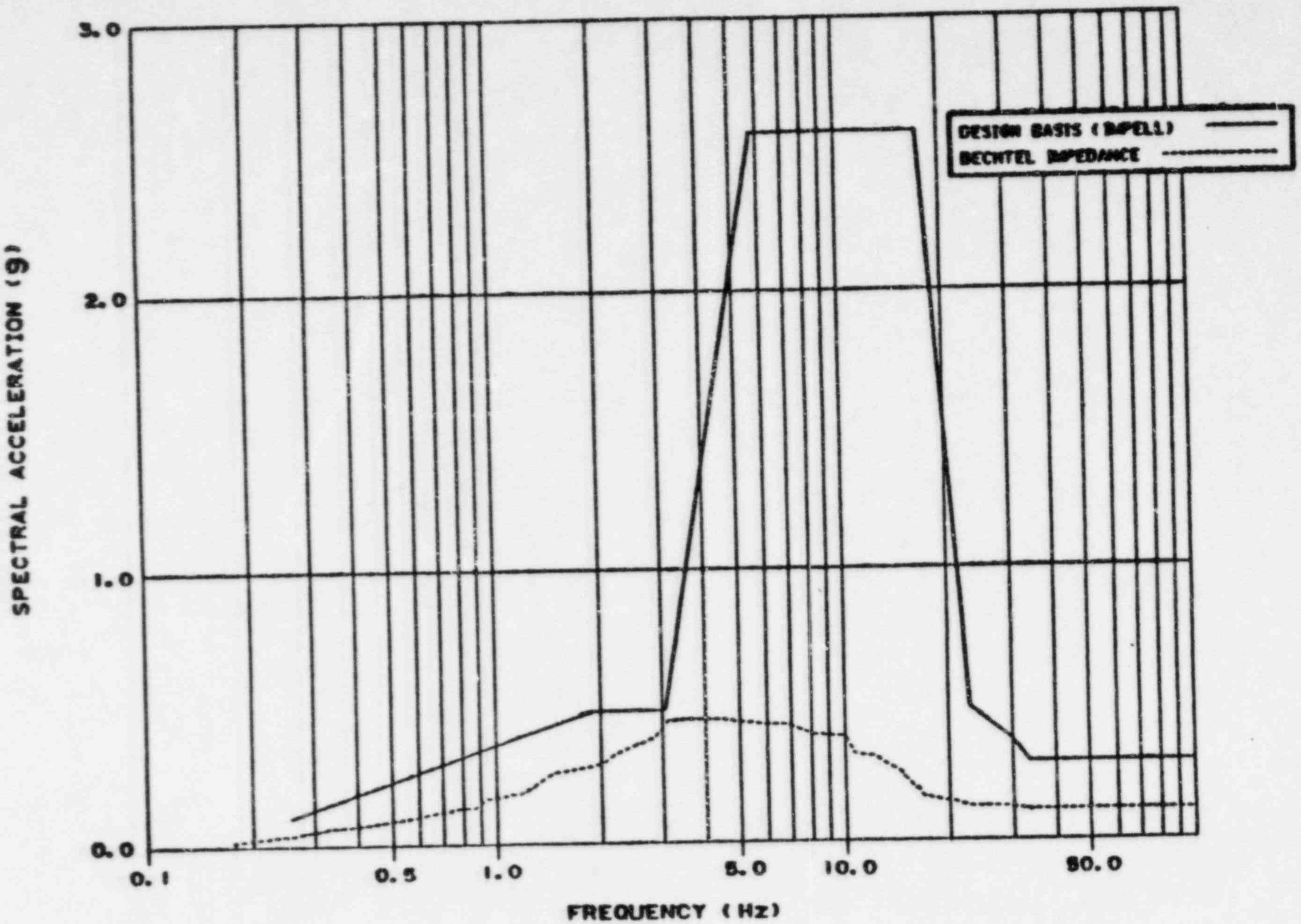

FIGURE A-16-17

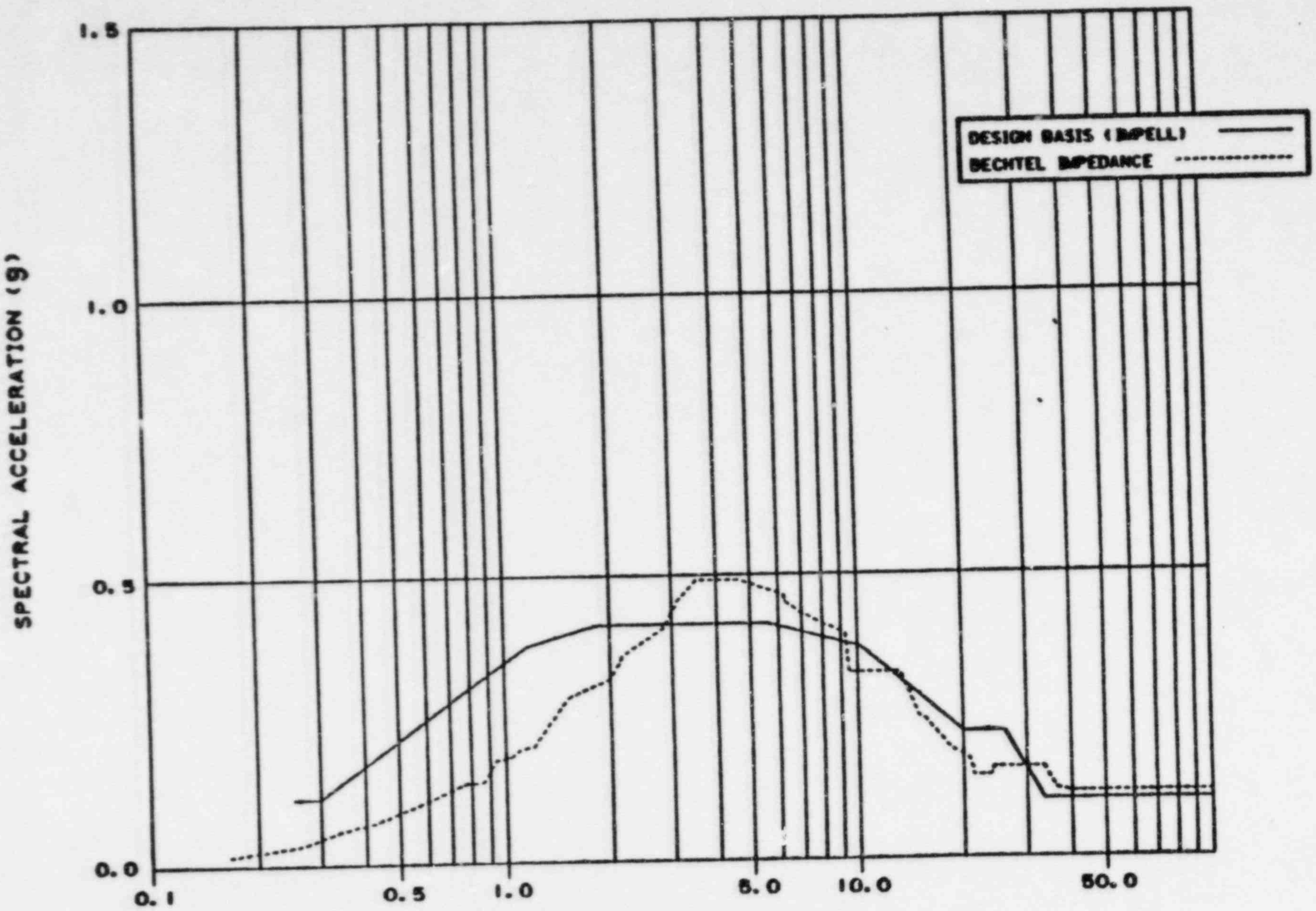

FIGURE A-16-18

RESPONSE SPECTRA COMPARISON.

DSER OPEN ITEM 66

Rev 1

DSER Open Item No. 147~~a,b,c,d~~g (DSER Section 9.3.1)

COMPRESSED AIR SYSTEMS

The service air system consists of two 100 percent capacity trains of compressors, aftercoolers, moisture separators, receivers, and associated piping and valves. Cooling is provided by the turbine auxiliary cooling system. One compressor runs automatically with the other compressor on standby. The standby compressor starts automatically on failure of the first system or failure of the first system to meet the demand for compressed air. This system maintains a constand pressure in the instrument air system. [The applicant has not provided an FSAR figure which identifies each air user, the location of each user, and all accumulators, check valves, and other appurtenances associated with safety related components, systems, and equipment, such as the ADS. The applicant has not provided readable figures in the FSAR, due to the drawing scale factor.] The service air compressor supplies air for the instrument air system by means of an intertie between the service air system and the instrument air system before the instrument air dryer package. The isolation between the two air systems is supplied air from the emergency air supply system (consisting of one compressor, filter, aftercooler, moisture separator, and receiver) for all accidents except a LOCA. Cooling is provided by the reacotr auxiliaries cooling system.

[The applicant has not identified the location of the equipment and the component classifications on the FSAR figures. Therefore, we cannot conclude that air systems satisfy the requirements of General Design Criterion 2, "Design Basis for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2, "Seismic Design Classification.]

A scheduled program of testing and inspection of the system will be provided to ensure operability of the system components and control systems. For compliance with the requirements of GDC 1, see Section 3.2 of this SER.

The service air system has no functions necessary for achieving safe reactor shutdown condition nor for accident prevention or mitigation. [The applicant has not identified and demonstrated that all instruments, controls and services required for safe shutdown of the plant such as the MSIV and ADS valves are provided with seismic Category I passive air accumulators to assure their proper function in a loss of the air system.] All other air-operated valves including the scram discharge inlet and outlet valves and other devices are designed to move to a safe position on loss of instrument air and do not require a continuous air supply under emergency or abnormal conditions.

[Additionally, the applicant has not verified that all station air system containment penetrations are provided with redundant seismic Category I, Quality Group B isolation valves. Therefore, we cannot conclude the requirements of General Design Criterion 2 and the guidelines of Regulatory Guide 1.29, Position C.2, are satisfied.]

The service instrument air systems will initially meet the requirements of American National Standards Institute (ANSI) MC11.1-1976, using non-oil-lubricated air compressors. [The applicant has not committed to perform periodic air quality testing of the air systems to assure compliance with the requirements of ANSI-MC11.1-1976.]

[Based on the above, we cannot conclude that the safety-related and non safety-related compressed air systems meet the requirements of General Design Criterion 2 regarding the protection against natural phenomena and the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2. We will report resolution of this item in a supplement to this SER. The compressed air system does not meet the applicable acceptance criteria of SRP Section 9.3.1.]

RESPONSE

The information for each air user, and all accumulators, check valves and other appurtenances associated with safety-related components, systems and equipment is provided in the response to Question 410.89

The ADS valve actuators are supplied with nitrogen (air) from the primary containment instrument gas system (see Section 9.3.6 for details of nitrogen (air) supply to ADS valves).

As described in FSAR Section 9.3.1.3 except for the containment isolation valves and penetration, whose location and classification is shown in Table 3.2-1 (Item XVII.a.3) the service air system is not safety related. Therefore, General Design Criteria 2 and Regulatory Guide 1.29, positions C.1 and C.2 do not apply.

As described in Section 6.2.4.3.2.4, "Containment Isolation System the Compressed Service Air Line" and Table 3.2-1 (Item XVII.a.3) the contaiment penetration is provided with redundant Seismic Category I, Quality Group B isolation valves.

As described in revised Section 9.3.1.~~2~~4, the quality of air supplied to the instrument air system will be periodically tested to see that it meets the requirements of ANSI MC11.1-1976 "Quality Standard for Instrument Air".

In addition, the instrument air system afterfilter is designed to remove 0.4 micrometre particles with a 98 per cent efficiency. The system is designed to permit preventive or corrective maintenance on one air dryer and after filter train without effecting system operability. Therefore, quarterly inspection of ~~the~~ afterfilter assures that the maximum particle size in the air stream at the instrument is 3.0 micrometres. This satisfies requirement 4.2 of ANSI MC 11.1-1975.

Category I and ASME B&PV Code, Section III, Class 2, requirements as defined in Sections 3.7 and 6.2.

9.3.1.4 Tests and Inspections

The containment penetration portions of the compressed air systems are preoperationally tested in accordance wth the requirements of Chapter 14. The instrument air system is tested in accordance with Regulatory Guide 1.68.3, Preoperational Testing of Instrument Air Systems. Compressors and dryers shall be tested in accordance with ASME and manufacturers' test procedures.

INSERT A →

9.3.1.5 Instrumentation Application

Instrumentation is provided for each instrument air and service air compressor train to monitor and automatically control each compressor's operation.

The compressors are tripped on the following signals: low oil pressure, high oil temperature, high cooling water discharge temperature, high air pressure in the receiver, high outlet air temperature, and high vibration. Most of these signals are annunciated in the main control room by common trouble alarms. High air temperature in the aftercooler and moisture separators, low pressure in the air receivers, and high intake filter differential pressure are also alarmed on a local control panel and the main control room by a common trouble alarm. Instrumentation is also provided locally for each instrument air dryer package train to monitor the packages operation.

Service air compressor and emergency instrument air compressor trouble are individually annunciated and alarmed on the local common service air compressor control panel. These alarms also indicate on the main control room computer, along with the air dryer trouble alarms.

9.3.2 PROCESS AND POST-ACCIDENT SAMPLING SYSTEMS

The process sampling system (PSS) is designed to monitor and provide grab samples of both radioactive and nonradioactive fluids used in the normal operation of Hope Creek Generating Station (HCGS).

DSER OPEN ITEM NO. 147

INSERT "A"

The instrument air dew point will be tested in accordance with ANSI MC11.1-1975, Quality Standard For Instrument Air, at a frequency of once per quarter as specified in the air dryer technical manual.

The Instrument Air System afterfilter is designed to remove .04 micrometre particles with a 98% efficiency. The system is designed to permit preventive or corrective maintenance on one dryer and afterfilter train without affecting system operability. Therefore quarterly inspection of the afterfilter assures that the maximum particlea size in the air stream at the instrument is 3.0 micrometres. This satisfies requirement 4.2 of ANSI MC 11.1-1975.

DSER Open Item No. 150 (DSER Section 9.3)

PRIMARY CONTAINMENT INSTRUMENT GAS SYSTEM

The applicant has committed to have the PCIG and instrument air systems meet the requirements of ANSI MC11.1-1976, using non-oil-lubricated air compressors as part of the preoperational startup tests. The applicant has not committed to perform air quality testing in accordance with ANSI MC11.1-1976. On failure to meet acceptable air quality, branch lines are to be tested to determine the extent of problems and corrective action needed.

The safety-related portions of the PCIG and instrument air systems are tested in accordance with the guidelines in RG 1.68.3, "Preoperational Testing of Instrument Air Systems" (refer to Section 14 of this SER).

We cannot conclude that the design conforms to the guidelines of ANSI MC 11.1-1976. We will report resolution of this item in a supplement to this SER. The compressed air system does not meet the applicable acceptance criteria of SRP Section 9.3.1.

RESPONSE

Testing of the PCIGS gas quality can be performed in accordance with ANSI MC 11.1-1975 by taking samples through the various vents, drains or test connections downstream of the PCIGS receivers. FSAR section 9.3.6.4 has been revised to provide the requested information

In addition, the PCIGS outlet filter removes 0.3 micrometres particles with a 98 per cent efficiency. The system is designed to permit preventive or corrective maintenance on one compressor, dryer and filter train without effecting system operability. Therefore, quarterly inspection of the filter assures that the maximum particle size in the air stream at the instrument is 3.0 micrometre. This satisfies requirement 4.2 of ANSI-MC 11.1-1975.

DSER OPEN ITEM NO. 150

preclude damage from missiles generated by the other compressing train.

d. Protection against dynamic effects associated with pipe ruptures - Section 3.6.

e. Environmental design considerations are discussed in Section 3.11.

Failure of a single component will not interrupt the operation of the PCIGS because of the redundant trains provided with separate sources of electric power fed from independent Class 1E sources.

9.3.6.4 Tests and Inspections

The PCIGS components are tested and inspected before leaving the supplier's shop to ensure that the system will meet the design criteria. The system is preoperationally tested in accordance with the requirements of Chapter 14.

Operability of the system is demonstrated by actual use during normal operation.

INSERT "A"

9.3.6.5 Instrumentation Applications

Instrumentation is provided for each train of the PCIGS to monitor and automatically control the system's operation. Further information on the system control and logic is discussed in Section 7.3.

The compressor is instrumented to shut down under the following conditions:

a. Low lubricating oil pressure

b. High lubricating oil temperature

c. High discharge gas temperature

DSER OPEN ITEM 150

DSER OPEN ITEM NO. 150

INSERT "A"

PCIGS dew point will be tested in accordance with ANSI MC11.1-1975, Quality Standard For Instrument Air, at a frequency of once per quarter as specified in the air dryer technical manual.

The PCIGS outlet filter removes .3 micrometre particles with a 98% efficiency. The system is ~~also~~ designed to permit preventive or corrective maintenance on one compressor, dryer and filter train without affecting system operability. Therefore quarterly inspection of the filter assures that the maximum particle size in the air stream at the instrument is 3.0 micrometres. This satisfies requirement 4.2 of ANSI MC11.1-1975.

DSER Open Item No. 186 (DSER Section 7.2.2.3)

TESTABILITY OF PLANT PROTECTION SYSTEM POWER

We will require that the applicant demonstrate the capability of the design for on-line testing of each instrumentation channel, logic, actuation device and actuated equipment in the ECCS and BOP ESF systems. All actuated contacts and devices should be considered and those which cannot be tested on-line should be identified and justification provided.

RESPONSE

The response to Question 421.22 will provide the requested information concerning on-line testability (at the contact level). This information will be provided by July 1984.

QUESTION 421.22 (SECTIONS 7.2, 7.3, 7.4, 7.5, 7.6, & 7.7)

The design of the instrumentation channels, logic and actuation devices of nuclear plant safety systems should include provisions for surveillance testing. Guidance is included in Reg. Guide 1.118 and IEEE Standard 338 for implementing the requirements of IEEE Standard 279, which requires in part that systems be designed to permit periodic testing during reactor operation.

Section 3.1.2.3.2 and 7.2.2.3.2 includes a brief description of the at-power testing capability of the reactor protection system. However, sufficient information has not been provided to determine the acceptability of the at-power testing capabilities provided in the Hope Creek design. Provide a detailed discussion with illustrations from applicable drawings on the at-power testing capability of the reactor trip system, engineered safety features actuation system and auxiliary supporting features, the actuation instrumentation for the reactor core isolation cooling system, and the instrumentation and controls that function to prevent accidents (i.e., high pressure/low pressure interlocks) or terminate transients (i.e., level 8 - turbine trip). This discussion should include the sensors, signal conditioning circuitry, voting logic, actuation devices and actuated components. Include in the discussion those design features that will initiate protection systems automatically, if required during testing, upon receipt of a valid initiation signal.

RESPONSE

As required by IEEE Standard 279, capability for at-power testing has been provided in the design of the HCGS safety systems. Conformance to the guidance specified in Regulatory Guide 1.118 and correspondingly, IEEE Standard 338, is as stated in Section 1.8.1.118.

The analysis portions of the various system descriptions in Chapter 7 for the safety-related systems referenced in the question describe the methods by which the safety system designs satisfy the testability requirements of IEEE Standard 279. The specific sections covering the testability of these systems are listed below:

RPS -	7.2.1.2
ECCS - HPCI	7.3.1.1.1.1(c)
- ADS	7.2.1.1.1.2(c)
- CORE SPRAY	7.3.1.1.1.3(c)
- RHR-LPCI	7.3.1.1.1.4(c)
PCRVICS	7.3.1.1.2(d)
RHR-CSCM	7.3.1.1.3(c)
RHR-SPCM	7.3.1.1.4(c)

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PCIS	7.3.1.1.5(j)
CACS - Supp. Chamber to Drywell Press. Relief	7.3.1.1.6.1(c)
- RB to Supp. Chamber Press. Relief Sys.	7.3.1.1.6.2(c)
- HOAS	7.3.1.1.6.3(c)
- CHRS	7.3.1.1.6.4(c)
MCRHIS	7.3.1.1.7(j)
MSIVSS	7.3.1.1.8(c)
FRVS	7.3.1.1.9
RBVIS	7.3.1.1.10(h)
EAS - SSWS	7.3.1.1.11.1(c)
- SACS	7.3.1.1.11.2(c)
PCIGS	7.3.1.1.1.11.4(c)
CACWS	7.3.1.1.1.11.5(c)
EACS - RBEAC	7.3.1.1.11.6.1(c)
- ABDA	7.3.1.1.11.6.2(c)
- ABCA	7.3.1.1.11.6.3(c)
- SWIS	7.3.1.1.11.6.4(c)
RCIC	7.4.1.1.3
SLC	7.4.1.2.3
RRCS	7.6.2.7.2(b)
	7.6.2.7.2(n)
	7.6.2.7.4.1

Design drawings in the form of elementary diagrams, P&IDs, logic diagrams, instrument location drawings, and electrical drawings that describe this capability are listed in Tables 1.7-1, 1.7-2, and 1.7-3.

In response to the NRC's request for additional information during the meeting of January 11, 1984, review of the systems identified above, with the exception of the reactor protection system (RPS), reactor core isolation cooling (RCIC) system, standby liquid control (SLC) system, and redundant reactivity control system (RRCS) ~~will be~~ was performed. The review ~~will~~ examined ~~determine~~ the capability for the at-power testing of all circuits and sensors used in these systems. All actuated contacts and devices were ~~will be~~ considered. ~~Any system, subsystem, or component that lacks the capability for at-power testing will be identified and a justification will be provided. The results will be documented in a revision to this response to be submitted by July 1984.~~

INSERT B

The review did not identify any device- or circuit-bypassing methods, other than those specifically permitted by position C6 of Regulatory Guide 1.118, ~~needed~~ for ESF at-power testing. Built-in test jacks, which provide connections for plug-in test switches, built-in test switches, and normal operational equipment, provide this testing capability as shown on the system elementary diagrams. During testing, redundant channels or systems are available to provide the safety function.

INSERT A →

PSE&G plans to conduct the at-power surveillance testing prescribed by the BWR 4 version of the NRC's Standard Technical Specifications.

— INSERT A —

During the review, the at-power testability of an item was established if an affirmative response could be verified for the following three questions:

a. Is the item sufficiently accessible to conduct the test during normal operation?

b. Is the item sufficiently isolatable to permit its safety-related function to be verified or is a safety-related system or subsystem encompassing the item isolatable and testable?

c. Does any bypassing method that must be used to accomplish the test conform to position C6 of Regulatory Guide 1.118?

for the NSSS safety systems

By these criteria, two items were judged to be untestable at power, the ADS SRVs, which would cause depressurization if tested, and the steam-tunnel temperature elements, which are inaccessible. The reliability and redundancy of the ADS instrumentation, logic, and actuation devices and the multiplicity of the SRVs adequately justify the lack of ADS at-power testability. Adequate element multiplicity and comparison tests of at-power output signals and electrical characteristics preclude the need for change-of-state testability of the steam-tunnel temperature elements.

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DSER OPEN ITEM 186 — INSERT B —

By these criteria, for the non-NSSS safety systems the following items were judged to be untestable at power:

a. PCIS — the LOCA signals of reactor low level (level 1), drywell high pressure, or manual initiation originating from core spray system relays K18A-D do not satisfy the criteria of question-b above. This affects actuation signals to close 43 containment isolation valves, to trip 16 MCC breakers, and to initiate control room isolation. The affected equipment is identified on Figure 7.3-26 (sheets 2-5) and Figure 7.3-27 (sheets 2 and 3). All other methods for actuation of this equipment can be verified at power; only this particular actuation signal ~~to each piece of equipment~~ can not be tested.

b. PCIS – the coincidence circuitry for the reactor building area and refueling floor area high-high radiation signals do not satisfy the criteria of question – b above. The individual high-high radiation signals can be verified up to the input buffers of the logic modules but must be tested one at a time since each signal is transmitted (through isolation devices) to all 4 channels of the PCIS simultaneously. See Figure 7.3-26 (sheets 6-9). This only affects the logic circuitry of the PCIS itself and does not inhibit the testing of the actual actuation signals from the PCIS to the individual actuated components.

DSER Open Item No. 187 (DSER Section 7.2.2.4)

LIFTING OF LEADS TO PERFORM SURVEILLANCE TESTING

To provide conformance with the recommendations of Regulatory Guide 1.118, we will require that design modifications be implemented to provide the capability to perform surveillance testing without lifting leads. Further, we will require that the applicant identify and justify the equipment that would not be tested at power. The justification should address the capability of the design to satisfy Criterion 21 of 10 CFR, Part 50. The opening of circuit breakers to perform monthly surveillance testing should also be discussed within the context of Regulatory Guide 1.22.

RESPONSE

For the information requested above, see the responses to Questions 421.4 and 421.22.

QUESTION 421.4 (SECTION 7.1)

FSAR Section 7.1.2.4 provides a discussion of design conformance to Regulatory Guide 1.118, Periodic Testing of Electrical Power and Protection Systems, June 1976 as an endorsement of IEEE-338-1977 and provides clarifications to two positions of this RG. The version of R.G. 1.118 cited is incorrect, as is the two positions discussed. To comply the staff review and the ensuing evaluation, the discussion of the justification for deviation from R.G. 1.118 will have to be corrected by referencing the 1978 version and by providing a clarification of the design deviations from the RG positions. It should be noted that the use of jury-rigged bypasses such as temporary jumpers, the removal of fuses, or removal of connectors is not an acceptable method for standard in-service testing.

RESPONSE

Although the June 1978 revision of Regulatory Guide 1.118 is not part of the design basis of the HCGS, ~~an assessment of a conformance has been prepared.~~ Section 7.1.2.4.g and Table 7.1-2 have been revised to reflect the specified date and to clarify the conformance situation.

In conjunction with the review described in the response to Question 421.22, a review ~~will be~~ was performed to determine if bypasses such as temporary jumpers, the removal of fuses, or the removal of connectors may be necessary for HCGS inservice testing. Instances where such bypasses may have to be used will be documented in a revision to this response, and discussions will be provided to justify the use of these bypasses. These justifications will be based on the exceptions authorized by position C14 of Regulatory Guide 1.118:

"a. Temporary jumper wires may be used with portable test equipment where the safety system equipment to be tested is provided with facilities specifically designed for connection of this test equipment. These facilities shall be considered part of the safety system and shall meet all the requirements of this standard, whether the portable test equipment is disconnected or remains connected to these facilities.

"b. Removal of fuses or opening a breaker is permitted only if such action causes (1) the trip of the associated protection system channel or (2) the actuation (startup and operation) of the associated Class 1E load group."

The revised reponse will be submitted by July, 1984.

This review was conducted, and the results are included in that response. Also,

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HCGS; however, equipment is qualified following the guidelines of IEEE 323-1971 as discussed in Section 3.11.2. Also refer to Section 3.11 for discussion of the environmental qualification program.

o. Assessment to Regulatory Guide 1.100, Seismic Qualifications of Electrical Equipment for Nuclear Power Plants, March 1976 - While not a design basis, the extent of conformance to Regulatory Guide 1.100 is discussed in Section 3.10.

p. Assessment to Regulatory Guide 1.105, Instrument Setpoints, November 1975 - While not a design basis, the design supplied includes the trip setpoint (instrument setpoint), allowable value (Technical Specification limit), and the analytical or design basis limit, which are all contained in Chapter 16, Technical Specifications. These parameters are all appropriately separated from each other based on instrument accuracy, calibration capabilility, and design drift (estimated) allowance data. The setpoints are within the instrument accuracy range.

The established setpoints provide margin to satisfy both safety requirements and plant availability objectives.

q. Assessment to Regulatory Guide 1.118, Periodic Testing of Electrical Power and Protection Systems, June 1978 - This regulatory guide, which endorses modified IEEE 338-1977, is not part of the design basis for HCGS. Discussion of IEEE 338 is presented on a system-by-system basis in the analysis portions of Sections 7.2, 7.3, 7.4, and 7.6. ~~with the following clarification of Position C.6: The removal of fuses and/or breakers to prevent the operation of equipment during the performance of tests could be authorized under strict administrative controls and approved procedures.~~ ~~See the response to Question 421.22 for additional information on at-power testability.~~

7.1.2.5 Independence of Safety-Related Systems

The safety-related I&C required to provide protective actions are physically arranged and separated to retain the minimum required

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QUESTION 421.22 (SECTIONS 7.2, 7.3, 7.4, 7.5, 7.6, & 7.7)

The design of the instrumentation channels, logic and actuation devices of nuclear plant safety systems should include provisions for surveillance testing. Guidance is included in Reg. Guide 1.118 and IEEE Standard 338 for implementing the requirements of IEEE Standard 279, which requires in part that systems be designed to permit periodic testing during reactor operation.

Section 3.1.2.3.2 and 7.2.2.3.2 includes a brief description of the at-power testing capability of the reactor protection system. However, sufficient information has not been provided to determine the acceptability of the at-power testing capabilities provided in the Hope Creek design. Provide a detailed discussion with illustrations from applicable drawings on the at-power testing capability of the reactor trip system, engineered safety features actuation system and auxiliary supporting features, the actuation instrumentation for the reactor core isolation cooling system, and the instrumentation and controls that function to prevent accidents (i.e., high pressure/low pressure interlocks) or terminate transients (i.e., level 8 - turbine trip). This discussion should include the sensors, signal conditioning circuitry, voting logic, actuation devices and actuated components. Include in the discussion those design features that will initiate protection systems automatically, if required during testing, upon receipt of a valid initiation signal.

RESPONSE

As required by IEEE Standard 279, capability for at-power testing has been provided in the design of the HCGS safety systems. Conformance to the guidance specified in Regulatory Guide 1.118 and correspondingly, IEEE Standard 338, is as stated in Section 1.8.1.118.

The analysis portions of the various system descriptions in Chapter 7 for the safety-related systems referenced in the question describe the methods by which the safety system designs satisfy the testability requirements of IEEE Standard 279. The specific sections covering the testability of these systems are listed below:

RPS -	7.2.1.2
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- ADS	7.2.1.1.1.2(c)
- CORE SPRAY	7.3.1.1.1.3(c)
- RHR-LPCI	7.3.1.1.1.4(c)
PCRVICS	7.3.1.1.2(d)
RHR-CSCM	7.3.1.1.3(c)
RHR-SPCM	7.3.1.1.4(c)

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PCIS	7.3.1.1.5(j)
CACS - Supp. Chamber to Drywell Press. Relief	7.3.1.1.6.1(c)
- RB to Supp. Chamber Press. Relief Sys.	7.3.1.1.6.2(c)
- HOAS	7.3.1.1.6.3(c)
- CHRS	7.3.1.1.6.4(c)
HCRHIS	7.3.1.1.7(j)
MSIVSS	7.3.1.1.8(c)
FRVS	7.3.1.1.9
RBVIS	7.3.1.1.10(h)
EAS - SSWS	7.3.1.1.11.1(c)
- SACS	7.3.1.1.11.2(c)
PCIGS	7.3.1.1.1.11.4(c)
CACWS	7.3.1.1.1.11.5(c)
EACS - RBEAC	7.3.1.1.11.6.1(c)
- ABDA	7.3.1.1.11.6.2(c)
- ABCA	7.3.1.1.11.6.3(c)
- SWIS	7.3.1.1.11.6.4(c)
RCIC	7.4.1.1.3
SLC	7.4.1.2.3
RRCS	7.6.2.7.2(b)
	7.6.2.7.2(n)
	7.6.2.7.4.1

Design drawings in the form of elementary diagrams, P&IDs, logic diagrams, instrument location drawings, and electrical drawings that describe this capability are listed in Tables 1.7-1, 1.7-2, and 1.7-3.

In response to the NRC's request for additional information during the meeting of January 11, 1984, review of the systems identified above, with the exception of the reactor protection system (RPS), reactor core isolation cooling (RCIC) system, standby liquid control (SLC) system, and redundant reactivity control system (RRCS) ~~will be~~ was performed. The review ~~will~~ examined ~~determine~~ the capability for the at-power testing of all circuits and sensors used in these systems. All actuated contacts and devices ~~will be~~ were considered. ~~Any system, subsystem, or component that lacks the capability for at-power testing will be identified and a justification will be provided. The results will be documented in a revision to this response to be submitted by July 1984.~~

¶ The review did not identify any device- or circuit-bypassing methods, other than those specifically permitted by position C6 of Regulatory Guide 1.118, needed for ESF at-power testing. Built-in test jacks, which provide connections for plug-in test switches, built-in test switches, and normal operational equipment, provide this testing capability as shown on the system elementary diagrams. During testing, redundant channels or systems are available to provide the safety function.

INSERT B

INSERT A →

¶ PSE&G plans to conduct that at-power surveillance testing prescribed by the BWR 4 version of the NRC's Standard Technical Specifications.

— INSERT A —

During the review, the at-power testability of an item was established if an affirmative response could be verified for the following three questions:

a. Is the item sufficiently accessible to conduct the test during normal operation?

b. Is the item sufficiently isolatable to permit its safety-related function to be verified or is a safety-related system or subsystem encompassing the item isolatable and testable?

c. Does any bypassing method that must be used to accomplish the test conform to position C6 of Regulatory Guide 1.118?

for the NSSS safety systems

By these criteria two items were judged to be untestable at power, the ADS SRVs, which would cause depressurization if tested, and the steam-tunnel temperature elements, which are inaccessible. The reliability and redundancy of the ADS instrumentation, logic, and actuation devices and the multiplicity of the SRVs adequately justify the lack of ADS at-power testability. Adequate element multiplicity and comparison tests of at-power output signals and electrical characteristics preclude the need for change-of-state testability of the steam-tunnel temperature elements.

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DSER OPEN ITEM 187 — INSERT B —

By these criteria, for the non-NSSS safety systems the following items were judged to be untestable at power:

a. PCIS — the LOCA signals of reactor low level (level 1), drywell high pressure, or manual initiation originating from core spray system relays K18A-D do not satisfy the criteria of question-b above. This affects actuation signals to close 43 containment isolation valves, to trip 16 MCC breakers, and to initiate control room isolation. The affected equipment is identified on Figure 7.3-26 (sheets 2-5) and Figure 7.3-27 (sheets 2 and 3). All other methods for actuation of this equipment can be verified at power; only this particular actuation signal ~~to each piece of equipment~~ can not be tested.

b. PCIS — the coincidence circuitry for the reactor building area and refueling floor area high-high radiation signals do not satisfy the criteria of question-b above. The individual high-high radiation signals can be verified up to the input buffers of the logic modules but must be tested one at a time since each signal is transmitted (through isolation devices) to all 4 channels of the PCIS simultaneously See Figure 7.3-26 (sheets 6-9). This only affects the logic circuitry of the PCIS itself and does not inhibit the testing of the actual actuation signals from the PCIS to the individual actuated components.

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DSER OPEN ITEM 201 (Section 7.4.2.3)

RCIC/HPCI INTERACTIONS

The applicant is required to confirm that HPCI operation is initiated by a LOCA signal or manual action and not by a RCIC low flow signal

Response

For the information requested above, see the response to question 421.37

DSER Open Item No. 202 (DSER Section 7.5.2.1)

LEVEL MEASUREMENT ERRORS AS A RESULT OF ENVIRONMENTAL TEMPERATURE EFFECTS ON LEVEL INSTRUMENTATION REFERENCE LEG

The applicant is required to submit the results of this evaluation for staff review and to implement any hardware and/or procedural changes that may evolve as a result of this evaluation.

RESPONSE

The response to ~~PSAR~~ e Question 421.21 will be revised by August 1984 to identify any necessary design changes to HCGS water level monitoring instrumentation.

For the information requested above, see the response to Question 421.21

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 —

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~~4/84~~ e

~~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

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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.

DSER Open Item No. 203 (DSER Section 7.5.2.2)

REGULATORY GUIDE 1.97

The staff is presently reviewing the BWROG position and the HCGS specific deviations which will be shown on revised FSAR Table 7.5-1. In addition, the applicant is required to resolve the inconsistency between FSAR Section 1.8.1.9.7 Item C, and Footnote 11 to Table 7.5-1.

RESPONSE

The response to Question 421.41 provides the HCGS specific implementation (via FSAR Table 7.5-1) of Regulatory Guide 1.97. Section 1.8.1.97 has been revised to remove the inconsistency between item C of Section 1.8.1.97 and Footnote 11 to Table 7.5-1.

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 Environs Conditions During and Following an Accident

1.8.1.97.1 General Position Statement

HCGS concurs with the intent of Regulatory Guide 1.97, Revision 2. The intent of the regulatory guide is to ensure that necessary and sufficient instrumentation exists at each nuclear power station for assessing plant and environmental conditions during and following an accident, as required by 10 CFR Part 50, Appendix A and General Design Criteria 13, 19, and 64. Regulatory Guide 1.97 requirements are being implemented except in those instances in which differences from the letter of the guide are justified technically and when they can be implemented without disrupting the general intent of the regulatory guide, or other applicable design criteria.

In assessing Regulatory Guide 1.97, HCGS has drawn upon information contained in several applicable documents, such as the BWROG Position (Reference 1.8-4) on Regulatory Guide 1.97, ANS 4.5, NUREG/CR-2100, and the BWROG Emergency Procedures Guidelines, and on data derived from other analyses and studies. HCGS has attempted to meet the intent of, as opposed to the literal compliance with the provisions of the regulatory guide, because of their specific nature. In general, HCGS intends to follow the criteria used by the NRC for establishing Category 1, 2, and 3 instruments. Where differences between the Regulatory Guide Categories exist, justification for the category chosen is provided. This approach is preferable as some Regulatory Guide 1.97 requirements call for excessive ranges or categories or both, others call for functions already available, and still others could adversely affect operator judgment under certain conditions. For example, research by S. Levy, Inc., (SLI), show that core thermocouples will provide conflicting information to BWR operators. HCGS intends to follow the criteria used by the NRC for establishing Category 1, 2, and 3 instruments.

The following HCGS compliance statement is applicable to the regulatory positions defined in Regulatory Guide 1.97, Revision 2 (the paragraph numbers cited correspond to those in Regulatory Guide 1.97).

a. Accident-Monitoring Instrumentation

Par. 1.1: HCGS concurs with this definition.

Par. 1.2: HCGS concurs with this definition.

Par. 1.3: Instruments used for accident monitoring to meet the provisions of Regulatory Guide 1.97 will have the proper sensitivity, range, transient response, and accuracy to ensure that both during and following a design basis accident the control room operator is able to perform his role in bringing the plant to, and maintaining it in, a safe shutdown condition and in assessing actual or possible releases of radioactive material.

Accident-monitoring instruments that are required to be environmentally qualified will be qualified as described in Section 3.11. The seismic qualification of instruments is described in Section 3.10.

The HCGS quality assurance program ensures that accident-monitoring instruments comply with the applicable requirements of Title 10 CFR 50, Appendix B. Table 3.2-1 identifies where these requirements have been applied.

The HCGS program for periodic checking, testing, calibrating, and calibration verification of accident-monitoring instrument channels (Regulatory Guide 1.118) is identified in Chapter 16, "Technical Specifications."

Par. 1.3.1: A third channel of instrumentation for Category 1 instruments will be provided only if:

1. a failure of one accident-monitoring channel results in information ambiguity that would lead operators to defeat or fail to accomplish a required safety function, and

2. if one of the following measures cannot provide the information:

   (a) Cross-checking with an independent channel that monitors a different variable bearing a known relationship to the variable being monitored.

   (b) Providing the operator with the capability of perturbing the measured variable to determine which channel has failed by observing the response on each instrument.

   (c) Using portable instrumentation for validation. Category 1 instrument channels, which are designated as being part of a Class 1E system, will meet the more stringent design requirements of either the system or the regulatory guide.

The requirements for physical independence of electrical systems (Regulatory Guide 1.75) are identified in Section 1.8.1.75.

Par. 1.3.2: HCGS concurs with the regulatory position for Category 2 instrumentation, except as modified by Par. 1.3 above.

Par. 1.3.3: HCGS concurs with the regulatory position for Category 3 instrumentation.

Par. 1.4: Instruments designated as Categories 1 and 2 for variable types A, B, and C should be identified in such a manner as to optimize the human factors engineering and presentation of information to the control room operator. This position is taken to clarify the intent of Regulatory Guide 1.97, which specified that these instruments be easily discerned for use during accident conditions (see Issue 1 Section 1.8.1.97.4)

Par. 1.5: HCGS concurs with the regulatory position taken in this section, except as modified by Par. 1.3 above.

Par. 1.6: It is the position of HCGS that in terms of accident monitoring at HCGS, Table 1 of Regulatory Guide 1.97 is not representative of the optimum SPT of variables required and does not necessarily represent correct variable ranges or instrumentation categories.

HCGS accident monitoring variables are identified in Table 7.5-1. The classification of instrumentation used to measure the variables as Category 1, 2, or 3 is in compliance with the intent and method used in Regulatory Guide 1.97. However, differences between the Regulatory Guide Categories and HCGS categories for each variable described in Table 1 of Regulatory Guide 1.97 is described in Section 1.8.1.97.3.

The HCGS position on the implementation of each variable described in Table 1 of Regulatory Guide 1.97 is presented in Section 1.8.1.97.3.

b. Systems Operation Monitoring and Effluent Release Monitoring Instrumentation

The HCGS position stated in Par. 1.3 above is applicable to the Type D and E variables described in Regulatory Guide 1.97.

Par. 2.1: HCGS concurs with these definitions.

Par. 2.2: HCGS concurs with this regulatory position.

Par. 2.3: HCGS concurs with this regulatory position

Par. 2.4: HCGS concurs with this regulatory position.

Par. 2.5: The HCGS position as stated in Par. 1.6 above is applicable to this regulatory position.

1.8.1.97.2 Proposed Type A Variables

Regulatory Guide 1.97, Revision 2, designates all Type A variables as Category I plant-specific, thereby defining none in particular. The regulatory guide defines Type A variables as:

- Those variables to be monitored that provide primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events.

Regulatory Guide 1.97 defines primary information as "information that is essential for the direct accomplishment of the specified safety funtions." Variables associated with contingency actions that may be identified in written procedures are excluded from this definition of primary information.

As part of their review of Regulatory Guide 1.97, the BWROG undertook the task of developing and analyzing a group of variables that were determined to be potential candidates for inclusion in Regulatory Guide 1.97 as specific Type A variables. HCGS has reviewed the generic BWROG identified variable and determined that the monitoring of the following noted safety functions for the listed operator actions are required to meet the intent of Regulatory Guide 1.97. The specific Type A variables are identified in Section 1.8.1.97.3.1:

<u>Variable A1. Oxygen or Hydrogen Concentration</u>

Safety Function: Maintain containment integrity by controlling oxygen for inerted and hydrogen for non-inerted contaminants.

Operator action: If containment atmosphere approaches the combustible limits, initiate combustible gas control systems.

<u>Variable A2. RPV Pressure</u>

Safety Function: (1) Core cooling; (2) maintain reactor coolant system integrity.

Operator action: (1) Depressurize RPV and maintain safe cooldown rate by any of several systems, such as main turbine bypass valves, HPCI, RCIC, and RWCU: (2) manually open one SRV to reduce pressure to below SRV setpoint if an SRV is cycling.

Variable A3. RPV Water Level

Safety Function: Core cooling.

Operator action: Restore and maintain RPV water level.

Variable A4. Suppression Pool Water Temperature

Safety Function: (1) Maintain containment integrity and (2) maintain reactor coolant system integrity.

Operator action: (1) Operate available suppression pool cooling system when pool temperature exceeds normal operating limits; (2) scram reactor if temperature reaches limit for scram; (3) if suppression pool temperature cannot be maintained below the heat capacity temperature limit, maintain RPV pressure below the corresponding limit; and (4) close any stuck-open relief valve.

Variable A5. Suppression Pool Water Level

Safety Function: Maintain containment integrity.

Operator action: Maintain suppression pool water level within normal operating limits: (1) transfer RCIC suction from the condensate storage tank (CST) to the suppression pool in the event of high suppression-pool level; and (2) if suppression pool water level cannot be maintained below the suppression pool load limit, maintain RPV pressure below corresponding limit.

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Variable A6. Drywell Pressure

Safety Function: (1) maintain containment integrity and (2) maintain reactor coolant system integrity.

Operator action: Control primary containment pressure by any of several systems, such as containment atmosphere control systems, suppression pool sprays, drywell sprays, etc.

1.8.1.97.3 Plant Variables For Accident Monitoring

In brief, the measurement of the following five variable types provides the noted required information to plant operators during and after an accident: (1) Type A--primary information, on the basis of which operators take planned specified manually controlled actions; (2) Type B--information about the accomplishment of plant safety functions; (3) Type C--information about the breaching of barriers to fission product release; (4) Type D--information about the operation of individual safety systems; and (5) Type E--information about the magnitude of the release of radioactive materials.

The three categories (1,2,3) of required variables define the design and qualification criteria for the instrumentation that is to be used for their measurement. Category 1 imposes the most stringent requirements; Categories 2 and 3 impose progressively less stringent requirements.

The categories are also related (per Regulatory Guide 1.97) to "key variables." Key variables are defined differently for the different variable types. For Type B and Type C variables, the key variables are those variables that most directly indicate the accomplishment of a safety function; instrumentation for these key variables is designated Category 1. Key variables that are Type D variables are defined as those variables that most directly indicate the operation of a safety function; instrumentation for these key variables is usually Category 2. And key variables that are Type E variables are defined as those variables that most directly indicate the release of radioactive material; instrumentation for these key variables is also usually Category 2. Backup variables for Type B, C, D and E variables are generally Category 3. A complete discussion of the variable types and instrumentation design criteria is presented in Regulatory Guide 1.97.

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HCGS positions on the implementation of the variables listed in Table 1 of Regulatory Guide 1.97 and on the assignment of design and qualification criteria for the instrumentation proposed for their measurement is summarized in the tabulation that follows.

The variables are listed here in the same sequence used in Table 1, Regulatory Guide 1.97; however, for convenience in cross-referencing entries and supporting data, the variables are designated by letter and number. For example, the sixth B-type variable listed in Regulatory Guide 1.97 is denoted here as variable B6.

The HCGS variable category designated ("HC") and the Regulatory Guide 1.97 category designated ("RG") are shown for each variable and for its instrumentation design criteria and category. In general, there are three positions cited by HCGS: (A) the variable and required instrumentation was implemented in accordance with the regulatory position stated in Table 1, Regulatory Guide 1.97 (B) was implemented with qualifying exceptions or revisions; and (C) was not implemented.

As necessary, the HCGS positions are justified or substantiated by the 11 "Issues" (identified in the tabulation of variables where applicable) noted in Section 1.8.1.97.4.

1.8.1.97.3.1 Type A variables (Reference Section 1.8.1.97.2)

A1. H_2 or O_2 concentration (HC Category 1, RG Category 1) Position: Monitor is plant-unique for H_2 or O_2. Needed for initiation of combustible gas controls. Implemented in accordance with NUREG-0737. See C11 and C12.

A2. Reactor pressure (HC Category 1, RG Category 1) Position: Implemented.

A3. Coolant level in reactor (HC Category 1, RG Category 1) Position: Implemented. See B4.

A4. Suppression pool water temperature (HC Category 1, RG Category 1) Position: Implemented. See D6.

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A5. Suppression pool water level (HC Category 1, RG Category 1) Position: Implemented. See C7 and D5.

A6. Drywell pressure (HC Category 1, RG Category 1) Position: Implemented. See B7, B9, C8, C10, and D4.

1.8.1.97.3.2 Type B Variables

a. Reactivity Control

B1. Neutron Flux (HC Category 2; RG Category 1) Position: Implemented, as Category 2 in accordance with data in Issue 2, Section 1.8.1.97.4.2.

B2. Control Rod Position (HC Category 3; RG Category 3) Position: Implemented.

B3. RCS Soluble Boron Concentration (sample) (HC Category 3; RG Category 3) Position: Implemented.

b. Core Cooling

B4. Coolant Level in Reactor (HC Category 1; RG Category 1) Position: Implemented. See A3.

B5. BWR Core Thermocouples (RG Category 1) Position: Not implemented. See B4, C3, and SLI-8121 (December, 1981) (Appendix A to Reference 1.8-4).

c. Maintaining Reactor Coolant System Integrity

B6. RCS Pressure (HC Category 1; RG Category 1) Position: Implemented. See A2, C4, C9, and Issue 3, Section 1.8.1.97.4.3.

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B7. Drywell Pressure (HC Category 1; RG Category 1)
Position: Implemented. See A6, B9, C8, C10, and D4.

B8. Drywell Sump Level (HC Category 3; RG Category 1)
Position: Implemented as Category 3. See C6 and Issue 4, Section 1.8.1.97.4.4.

d. Maintaining Containment Integrity

B9. Primary Containment Pressure (HC Category 1; RG Category 1)
Position: Implemented. See A6, B7, C8, C10, and D4.

B10. Primary Containment Isolation Valve Position (excluding check valves) (HC Category 1; RG Category 1)
Position: Implemented. Redundant indication is not required on each redundant isolation valve.

1.8.1.97.3.3 Type C Variables

a. Fuel Cladding

C1. Radioactivity Concentration or Radiation Level in Circulating Primary Coolant (RG Category 1)
Position: Not implemented. See Issue 5, Section 1.8.1.97.4.5.

C2. Analysis of Primary Coolant (gamma spectrum) (HC Category 3; RG Category 3)
Position: Implemented.

C3. BWR Core Thermocouples (RG Category 1)
Position: Not implemented. See B4, B5, and SLI-8121 (December, 1981) (Appendix A to Reference a.8-4).

b. Reactor Coolant Pressure Boundary

C4. RCS Pressure (HC Category 1; RG Category 1)
Position: Implemented. See A2, B6, and C9.

C5. Primary Containment Area Radiation (HC Category 1; RG Category 3)
Position: Implemented as Category 1. See E1.

C6. Drywell Drain Sumps Level (identified and unidentified leakage) (HC Category 3; RG Category 1)
Position: Implemented as Category 3. See B8 and Issue 4, Section 1.8.1.97.4.4.

C7. Suppression Pool Water Level (HC Category 1; RG Category 1)
Position: Implemented. See A5 and D5.

C8. Drywell Pressure (HC Category 1; RG Category 1)
Position: Implemented. See A6, B7, and B9, C10, and D4.

c. Containment

C9. RCS Pressure (HC Category 1; RG Category 1)
Position: Implemented. See A2, B6, and C4.

C10. Primary Containment Pressure (HC Category 1; RG Category 1)
Position: Implemented. See A6, B7, B9, C8, and D4.

C11. Containment and Drywell H_2 Concentration (HC Category 1; RG Category 1)
Position: Implemented. See A1.

C12. Containment and Drywell Oxygen Concentration (HC Category 1; RG Category 1)
Position: Implemented. See A1.

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C13. Containment Effluent Radioactivity--Noble Gases (from identified release points including Filtration, Recirculation & Ventilation System Vent) (HC Category 3; RG Category 3)
Position: Implemented.

C14. Radiation Exposure Rate (inside buildings or areas, e.g., auxiliary building, reactor building, which are in direct contact with primary containment where penetrations and hatches are located) (RG Category 2)
Position: Not implemented. See E2, E3, and Issue 6, Section 1.8.1.97.4.6.

C15. Effluent Radioactivity--Noble Gases (from buildings as indicated above) (HC Category 2; RG Category 2)
Position: Implemented.

1.8.1.97.3.4 Type D Variables

a. Condensate and Feedwater System

D1. Main Feedwater Flow (HC Category 3; RG Category 3)
Position: Implemented.

D2. Condensate Storage Tank Level (HC Category 3; RG Category 3)
Position: Implemented.

b. Primary Containment-Related System

D3. Suppression Chamber Spray Flow (HC Category 2; RG Category 2)
Position: Implemented.

D4. Drywell Pressure (HC Category 2; RG Category 2)
Position: Implemented. See A6, B7, B9, C8 and C10.

D5. Suppression Pool Water Level (HC Category 2; RG Category 2)
Position: Implemented. See A5 and C7.

D6. Suppression Pool Water Temperature (HC Category 1; RG Category 2)
Position: Implemented, but must be Category 1. Both local and bulk temperature. See A4.

D7. Drywell Atmosphere Temperature (HC Category 2; RG Category 2)
Position: Implemented.

D8. Drywell Spray Flow (HC Category 2; RG Category 2)
Position: Implemented.

c. Main Steam System

D9. Main Steamline Isolation Valves' Leakage Control System Pressure (HC Category 2; RG Category 2)
Position: Implemented. (System is identified as Main Steam Isolation Valve Sealing System at HCGS).

D10. Primary System Safety Relief Valve Position, Including ADS or Flow Through or Pressure in Valve Lines (HC Category 2; RG Category 2)
Position: Implemented.

d. Safety Systems

D11. Isolation Condenser System Shell-Side Water Level
Position: Not applicable to HCGS.

D12. Isolation Condenser System Valve Position
Position: Not applicable to HCGS.

D13. RCIC Flow (HC Category 2; RG Category 2)
Position: Implemented. See Issue 7, Section 1.8.1.97.4.7.

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D14. HPCI Flow (HC Category 2; RG Category 2)
Position: Implemented. See Issue 7, Section 1.8.1.97.4.7.

D15. Core Spray System Flow (HC Category 2; RG Category 2)
Position: Implemented. See Issue 7, Section 1.8.1.97.4.7.

D16. LPCI System Flow (HC Category 2; RG Category 2)
Position: Implemented. See Issue 7, Section 1.8.1.97.4.7.

D17. SLC System Flow (HC Category 3; RG Category 2)
Position: Implemented as Category 3. See Issue 7, Section 1.8.1.97.4.7.

D18. SLC System Storage Tank Level (HC Category 2; RG Category 2)
Position: Implemented.

e. Residual Heat Removal (RHR) Systems

D19. RHR System Flow (HC Category 2; RG Category 2)
Position: Implemented.

D20. RHR Heat Exchanger Outlet Temperature (HC Category 2; RG Category 2)
Position: Implemented.

f. Cooling Water System

D21. Cooling Water Temperature to ESF System Components (HC Category 2; RG Category 2)
Position: Interpreted as Safety Auxiliaries Cooling System (SACS) temperature and implemented.

D22. Cooling Water Flow to ESF System Components (HC Category 2; RG Category 2)
Position: Interpreted as SACS flow and implemented.

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g. Radwaste Systems

D23. High Radioactivity Liquid Tank Level
(HC Category 3; RG Category 3)
Position: Implemented.

h. Ventilation Systems

D24. Emergency Ventilation Damper Position
(HC Category 2; RG Category 2)
Position: Interpreted as meaning dampers actuated under accident conditions and whose failure could result in radioactive discharge to the environment. Control room damper position is indicated. Implemented.

i. Power Supplies

D25. Status of Standby Power and Other Energy Sources Important to Safety (hydraulic, pneumatic)
(HC Category 2; RG Category 2)
Position: Implemented; on-site sources only.

(Note: HCGS has implemented the following D-type variables as recommended by the BWROG; see Issue 8, Section 1.8.1.97.4.8.)

D26. Turbine Bypass Valve Position (HC Category 3)
Position: Implemented. See Issue 8, Section 1.8.1.97.4.8.

D27. Condenser Hotwell Level (HC Category 3)
Position: Implemented. See Issue 8, Section 1.8.1.97.4.8.

D28. Condenser Vacuum (HC Category 3)
Position: Implemented. See Issue 8, Section 1.8.1.97.4.8.

D29. Condenser Cooling Water Flow (HC Category 3)

Position: Interpreted as cooling water ΔT across the condenser and implemented. See Issue 8, Section 1.8.1.97.4.8.

D30. Primary Loop Recirculation (HC Category 3)
Position: Implemented. See Issue 8, Section 1.8.1.97.4.8.

1.8.1.97.3.5 Type E Variables

a. Containment Radiation

E1. Primary Containment Area Radiation--High Range (HC Category 1; RG Category 1)
Position: Implemented in accordance with NUREG-0737 commitment. See C5.

E2. Reactor Building or Secondary Containment Area Radiation (RG Category 2 for Mark I and II containments)
Position: Not implemented for HCGS (Mark I) containment. See C14, E3, and Issue 9, Section 1.8.1.97.4.9..

b. Area Radiation

E3. Radiation Exposure Rate (inside buildings or areas where access is required to service equipment important to safety) (HC Category 3; RG Category 2)
Position: Implemented as Category 3, using existing instrumentation. See C14, E2, and Issue 10, Section 1.8.1.97.4.10.

c. Airborne Radioactive Materials Released From Plant

E4. Noble Gases and Vent Flow Rate (HC Category 2; RG Category 2)
Position: Implemented.

E5. Particulates and Halogens (HC Category 3; RG Category 3)
Position: Implemented.

d. Environs Radiation and Radioactivity

E6. Radiation Exposure Meters (continuous indication at fixed locations)
Position: Deleted. See NRC errata of July 1981.

E7. Airborne Radiohalogens and Particulates (portable sampling with on-site analysis capability) (HC Category 3; RG Category 3)
Position: Implemented.

E8. Plant Environs Radiation (portable instrumentation) (HC Category 3; RG Category 3)
Position: Implemented (portable equipment).

E9. Plant and Environs Radioactivity (portable instrumentation) (HC Category 3; RG Category 3)
Position: Implemented (portable equipment).

e. Meteorology

E10. Wind Direction (HC Category 3; RG Category 3)
Position: Implemented.

E11. Wind Speed (HC Category 3; RG Category 3)
Position: Implemented.

E12. Estimation of Atmospheric Stability (HC Category 3; RG Category 3)
Position: Implemented.

f. Accident-Sampling Capability (Analysis Capability On-Site)

E13. Primary Coolant and Sump (HC Category 3--Primary Coolant only; RG Category 3)

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Position: Implemented Primary Coolant. (Dissolved hydrogen or Total Gas not implemented). Sump not implemented. See Issue 11, Section 1.8.1.97.4.11.

E14. Containment Air (HC Category 3; RG Category 3) Position: Implemented.

The instrumentation for monitoring and display of type A, B, C, D, and E variables at HCGS is identified on Table 7.5-1.

1.8.1.97.4 Supplementary Analyses

These supplementary analyses support positions cited in Section 1.8.1.97.1 (Issue 1) and Section 1.8.1.97.3 (Issues 2-12).

1.8.1.97.4.1 ISSUE 1 - INSTRUMENT IDENTIFICATION

Regulatory Guide 1.97 specifies, in paragraph 1.4.b, the following: "The instruments designated as Types A, B, and C and Categories 1 and 2 should be specifically identified on the control panels so that the operator can easily discern that they are intended for use under accident conditions."

The objective of this regulatory position is the achievement of good human factors engineering in the presentation of information to the control room operator. This objective is best achieved by evaluating current practices and procedures that provide for identifying instruments in a manner that aids the operator; redundant labels would tend to distract the operator and cause confusion.

Instruments designated as Categories 1 and 2 for monitoring variable types A, B, and C should be identified in such a manner as to optimize applicable human factors engineering and presentation of information to the control room operator. This position is taken to clarify the intent of Regulatory Guide 1.97, which specifies that these instruments be easily discerned for use during accident conditions. The method of identification used at HCGS will be based on the results of a human factors analysis performed on the HCGS main control room (See Chapter 18).

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1.8.1.97.4.2 ISSUE 2 - VARIABLE B1

The measurement of neutron flux is specified as the key variable in monitoring the status of reactivity. Neutron flux is classified as a Type B variable, Category 1. The specified range is 10^{-6} percent to 100 percent full power (SRM, APRM). The stated purpose is "function detection; accomplishment of mitigation."

The lower end of the specified range, 10^{-6} percent full power, is intended to allow detection of an approach to criticality by some undefined and noncontrollable mechanism after shutdown.

In attempting to analyze the performance of the neutron-flux monitoring systems, a scenario was postulated to obtain the required approach to criticality. Basically, it assumes an increase in reactivity from dilution of boron concentration in the reactor water.

The accident scenario incorporates the following factors:

a. The control rods fail (completely or partially) to insert, and the operator actuates the standby liquid control system (SLCS).

b. The SLCS shuts the reactor down.

c. A leak in the primary system results in a dilution of borated water and replacement by water that contains no boron.

d. A range of leak rates up to 20 gpm was considered (see Table 1.8-2).

Calculations were made to evaluate the rise in neutron population as a function of different leak rates. The calculations were made for a shutdown neutron level of 5 x 10^{-8} percent of full power. The choice of 5 x 10^{-8} is based on measurement at two nuclear plants. The shutdown level was assumed to have a negative reactivity of 10 dollars, an assumption that is representative of a shutdown with all rods inserted. The results of the calculations are presented in Table 1.8-2. The numbers in

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the table refer to the time in hours required to increase the flux by 1 decade. For example, with a leak of 5 gpm, it takes 100 hr to increase the power from 5×10^{-8} percent to 5×10^{-7} percent, and 10 hr to increase it from 5×10^{-7} percent to 5×10^{-6} percent.

The reactor is subcritical and the neutron level is given by

$$\text{Neutron level} = S \times M,$$

where S is the source strength and M is the multiplication which is given by

$$M = 1/(1-k).$$

For k = 0.9, M is 10; for k = 0.99, M is 100 and so forth. For criticality, the denominator approaches 0, as k approaches 1.0. Thus, the calculation model used the above equation to calculate relative neutron flux levels for a subcritical reactor until the reactor was near critical; then the critical equation of power with excess reactivity was used. Reactor power is directly proportional to neutron level.

The increase in reactivity toward criticality can be terminated by actuating the SLCS. Operating procedures provide for refilling the SLCS tank with borated water soon after its actuation. A second actuation of the SLCS would cause a decrease in reactivity because of the high concentration of boron in the injected SLCS fluid relative to that in the leaking fluid (nominally 400 ppm). The sensitivity of the detector must allow adequate time for the operator to act. Ten minutes is considered sufficient time for operator action for accident prevention and mitigation.

Table 1.8-2 shows that the detector sensitivity (i.e., lower range) requirement is a function of leak rate and therefore, of reactivity-addition rate. On the basis of a 20-gpm leak rate, Table 1.8-2 shows that a detector that is on scale within 3 decades of the shutdown power would allow 0.18 hr (10.8 min) for operator action before reactor power increased another decade. A total of 0.36 hr (21.6 min) would be available for operator action from the time the indicator comes on scale to the time reactor power reaches 0.5 percent of full power.

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The 20-gpm leak rate, which was assumed to continue for 27.75 hr, was used to define the sensitivity of the detector. It should be noted that the assumed leak rate, extended over the 27.75-hr period, would result in a loss of inventory so large that it would be detected by the operator. Moreover, detection of the reactivity addition caused by this gradual boron dilution will be noted via boron concentration sampling and measurement. Again, the conservative 20-gpm leak rate was used only to obtain a mechanistic and conservative approach for selection of instrument sensitivity.

An absolute criterion for the lower range includes consideration of the neutron source level. The use of the neutron level 100 days after shutdown is conservative. Conditions would be stable and controllable 2 days after the emergency shutdown, as the core-decay heat is at a low level and the boron monitoring system is functional. The actual neutron level will vary with fuel design, fuel history, and shutdown control strength. Measurements of shutdown neutron flux (with all rods inserted) at two BWR reactors show readings of 30 to 80 counts/sec (1000 counts/sec corresponds to 10^{-8} of full power). Measurements on other BWR reactors and for different fuel histories would show some variation, but those variations would be small compared with a criterion that is concerned with units of decades.

Regulatory Guide 1.97 classifies the instrumentation for measuring a variable as Category 1 on the basis of (1) whether it is a key variable (defined in Section 1.8.1.75.3), and (2) its importance to safety. Neutron flux is the key variable for measuring reactivity control, thus meeting the requirement of criterion (1). The degree to which this variable is important to safety is another consideration. The large number of detectors (i.e., source-range monitors and intermediate-range monitors) that are driven into the core soon after shutdown makes it highly probable that one or more of the existing NMS detectors will be inserted. On the other hand, there is little probability that there would be, simultaneously, a need for this measurement (in terms of operator action to be taken) and an accident environment in which the NMS would be rendered inoperable. Further, the operator can actuate the SLCS on loss of instrumentation.

A rigorous Category 1 requirement is not justified when the purpose and use of the measurement are analyzed as they relate to the criterion of "importance to safety." A Category 2 classification of this variable fully meets the intent of Regulatory 1.97.

A range from 5 x 10^{-5} percent of full power (within 3 decades of the neutron flux level 100 days after shutdown) to 100 percent of full power is recommended. It is concluded that a Category 2 classification is responsive to the intent of Regulatory 1.97.

As defined in this issue, instrumentation for long term monitoring of the lower end of the Regulatory Guide 1.97 specified range is only needed during an anticipated transient without scram (ATWS) event. ATWS events do not require the consideration of loss-of-coolant accident (LOCA) conditions. It is estimated that the environmental conditions existing during an ATWS event would be similar to the environmental conditions existing during normal operation, at least in the short term during operation of equipment such as the SRM and IRM drive mechanisms. Further, ATWS mitigation features have a lower importance to safety than safety systems, making a Category 2 classification for neutron flux instrumentation in lieu of Category 1 as specified in Regulatory Guide 1.97, more appropriate and more consistent with the requirements applicable to other ATWS mitigation features.

It is the HCGS position that since the HCGS neutron flux instrumentation consists of a large number of neutron monitoring channels (4 SRM, 8 IRM, and 6 APRMs plus individual LPRM channels) of proven high reliability, designed to operate in environmental conditions similar to those postulated to exist during an ATWS event, and since the ATWS mitigation features have a lower importance to safety than safety systems, a Category 2 classification for neutron flux monitoring instrumentation is justified and the existing instrumentation at HCGS is satisfactory for this monitoring function without modification.

1.8.1.97.4.3 ISSUE 3 - TREND RECORDING

The purpose of addressing Issue 3 is to determine which variables set forth in Regulatory Guide 1.97 require trend recording.

Regulatory Guide 1.97, paragraph 1.3.2f, states the general requirement for trend recording as follows: "Where direct and immediate trend or transient information is essential for operator information or action, the recording should be continuously available for dedicated recorders." Using the BWROG Emergency Procedures Guidelines (EPG's) as a basis, the only trended variables required for operator action are reactor water level and reactor vessel pressure.

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Other variables at HCGS are recorded as identified on Table 7.5-1.

1.8.1.97.4.4 ISSUE 4 - VARIABLES B8 AND C6

Regulatory Guide 1.97 requires Category 1 instrumentation to monitor drywell sump level (variable B8) and drywell drain sumps level (variable C6). These designations refer to the drywell equipment and floor-drain tank levels. Category 1 instrumentation indicates that the variable being monitored is a key variable. In Regulatory Guide 1.97, a key variable is defined as "... that single variable (or minimum number of variables) that most directly indicates the accomplishment of a safety function..." The following discussion supports the HCGS position that drywell sump level and drywell drain-sumps levels should be designated as Category 3 instrumentation requirements.

The HCGS drywell has two drain sumps. One drain is the equipment drain sump, which collects identified leakage; the other is the floor drain sump, which collects unidentified leakage.

Although the level of the drain sumps can be a direct indication of breach of the reactor coolant system pressure boundary, the indication is not unambiguous, because there can be water in those sumps during normal operation. There is other instrumentation required by Regulatory Guide 1.97 that would indicate leakage in the drywell:

a. Drywell pressure--variable B7, Category 1

b. Drywell temperature--variable D7, Category 2

c. Primary containment area radiation--variable C5, Category 1

The drywell-sump levels signal neither automatic protection control circuitry nor the operator to take safety-related actions. Both sumps have level detectors that provide only the following nonsafety indications:

a. Continuous level indication

b. Rate of rise indication

c. High-level alarm (starts first sump pump)

d. High-high-level alarm (starts second sump pump)

Regulatory Guide 1.97 requires instrumentation to function during and after an accident. The drywell sump systems are deliberately isolated at the primary containment penetration upon receipt of an accident signal to establish containment integrity. This fact renders the drywell-sump-level signal irrelevant. Therefore, by design, drywell-level instrumentation serves no useful accident-monitoring function.

The Emergency Procedure Guidelines use the RPV level and the drywell pressure as entry conditions for the Level Control Guideline. A small line break will cause the drywell pressure to increase before a noticeable increase in the sump level. Therefore, the drywell sumps will provide a "lagging" versus "early" indication of a leak.

Based on the above considerations, HCGS believes that the drywell-sump level and drywell-drain-sumps level instrumentation should be designated as Category 3, "high-quality off-the-shelf instrumentation."

1.8.1.97.4.5 ISSUE 5 - VARIABLE C1

Regulatory Guide 1.97 specifies that the status of the fuel cladding be monitored during and after an accident. The specified variable to accomplish this monitoring is variable C1--radioactivity concentration or radiation level in circulating primary coolant. The range is given as "1/2 Tech. Spec. Limit to 100 times Tech. Spec. Limit, R/hr." In Table 1 of Regulatory Guide 1.97, instrumentation for measuring variable C1 is designated as Category 1. The purpose for monitoring this variable is given as "detection of breach," referring, in this case, to breach of fuel cladding.

The usefulness of the information obtained by monitoring variable C1, in terms of helping the operator in his efforts to prevent and mitigate accidents, has not been substantiated. The particular planned operator action to be taken based on monitoring this variable is not specified in the current draft of the Emergency Procedure Guidelines (EPGs). The critical actions that must be taken to prevent and mitigate a gross breach of fuel cladding are (1) shut down the reactor and (2) maintain water level. Monitoring variable C1, as directed in Regulatory Guide 1.97, will have no influence on either of these actions. The purpose of this monitor falls in the category of "information that the barriers to release of radioactive material are being challenged" and "identification of degraded conditions and their magnitude, so the operator can take actions that are available to mitigate the consequences." Additional operator actions to mitigate the consequences of fuel barriers being challenged, other than those based on Type A and B variables, have not been identified.

Regulatory Guide 1.97 specifies measurement of the radioactivity of the circulating primary coolant as the key variable in monitoring fuel cladding status during isolation of the NSSS. The words "circulating primary coolant" are interpreted to mean coolant, or a representative sample of such coolant, that flows past the core. A basic criterion for a valid measurement of the specified variable is that the coolant being monitored is coolant that is in active contact with the fuel, that is, flowing past the failed fuel. Monitoring the active coolant (or a sample thereof) is the dominant consideration. The post-accident sampling system (PASS) provides a representative sample which can be monitored.

The subject of concern in the Regulatory Guide 1.97 requirement is assumed to be an isolated NSSS that is shutdown. This assumption is justified as current monitors in the condenser off-gas and main steam lines provide reliable and accurate information on the status of fuel cladding when the plant is not isolated. Further, the PASS will provide an accurate status of coolant radioactivity, and hence cladding status, once the PASS is activated. In the interim between NSSS isolation and operation of the PASS, monitoring of the primary containment radiation and containment hydrogen will provide information on the status of the fuel cladding. The use of a portable gross gamma monitor on the PASS sample line could likely be used to monitor primary coolant before the analytical station can be put in operation (a period of more than 2 hours).

Later in the sequence, the PASS sample can be augmented by area radiation monitors when the RHR system is being used to remove core decay heat.

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The designation of instrumentation for measuring variable C1 should be Category 3, because no planned operator actions are identified and no operator actions are anticipated based on this variable serving as the key variable. Existing Category 3 instrumentation is adequate for monitoring fuel cladding status.

1.8.1.97.4.6 ISSUE 6 - VARIABLE C14

Varable C14 is defined in Table 1 of Regulatory 1.97 as follows: "Radiation exposure rate (inside buildings or areas, e.g., auxiliary building, fuel handling building, secondary containment), which are in direct contact with primary containment where penetrations and hatches are located." The reason for monitoring variable C14 is given as "Indication of breach."

The use of local radiation exposure rate monitors to detect breach or leakage through primary containment penetrations is impractical and unnecessary. In general, radiation exposure rate in the reactor building will be largely a function of radioactivity in primary containment and in the fluids flowing in ECCS piping, which will cause direct radiation shine on the area monitors. Also, because of the amount of piping and the number of electrical penetrations and hatches and their widely scattered locations, local radiation exposure rate monitors could give ambiguous indications. The proper way to detect breach of containment is by using the plant noble gas effluent monitors.

Therefore, it is the position of HCGS that this parameter not be implemented.

1.8.1.97.4.7 ISSUE 7 - VARIABLES D13-D17

Regulatory Guide 1.97 specifies flow measurements of the following systems: reactor core isolation cooling (RCIC) (variable D13), high-pressure coolant injection (HPCI) (variable D14), core spray (variable D15), low-pressure coolant injection (LPCI) (variable D16), and standby liquid control (SLC) (variable D17). The purpose is for monitoring the operation of individual safety systems. Instrumentation for measuring these variables is designated as Category 2; the range is specified as 0 to 110 percent of design flow. These variables are related to flow into the reactor pressure vessel (RPV).

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The RCIC, HPCI, and core spray systems each have one branch line--the test line--downstream of the flow-measuring element. The test line is provided with a motor-operated valve that is normally closed (HPCI and RCIC also share a motor-operated valve that is normally open). Further, the valve in the test line automatically closes when the emergency system is actuated, thereby ensuring that indicated flow is not being diverted by the test line. Proper valve position can be verified by a direct indication of valve position on the main control board.

Although the LPCI has several branch lines located downstream of each flow-measuring element, upon initiation of the LPCI, the valves in the system automatically line up for proper operation and prevent flow diversion by branch lines. Proper valve position can be verified by the operator using main control board indication of valve position.

For all of the above systems, there are valid primary indicators other than flow measurement to verify the performance of the emergency system; for example, reactor vessel water level.

Flow-measuring devices are not provided for the SLC system. The pump-discharge header pressure, which is indicated in the control room, will indicate SLC pump operation. Besides the discharge header pressure observation, the operator can verify the proper functioning of the SLC system by monitoring the following:

a. The decrease in the level of the SLC storage tank,

b. The boron injection induced reactivity change in the reactor as measured by neutron flux

c. The main control room motor status indicating lights (or motor current),

d. Squib valve continuity indicating lights.

The use of these indications is believed to be a valid alternative to SLC system flow indication.

The flow-measurement schemes for the RCIC, HPCI, core spray, and LPCI meet the Category 2 requirements of Regulatory Guide 1.97.

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Monitoring the SLC system can be adequately done by measuring the above named Category 3 variables rather than the actual flow.

1.8.1.97.4.8 ISSUE 8 - VARIABLES D26-D30

Regulatory Guide 1.97 states that "The plant designer should select variables and information display channels required by his design to enable the control room personnel to ascertain the operating status of each individual safety system and other systems important to safety to that extent necessary to determine if each system is operating or can be placed in operation..." The purpose of this analysis was to determine whether certain other D-type variables should be added to Table 1, Regulatory Guide 1.97.

Regulatory Guide 1.97 addressed safety systems and systems important to safety to mitigate consequences of an accident. Another list of variables has been compiled for the BWR in NUREG/CR-2100 (Boiling Water Reactor Status Monitoring during Accident Conditions, April 1981). That report and a companion report, NUREG/CR-1440 (Light Water Reactor Status Monitoring during Accident Conditions, June 1980), address plant systems not important to safety, as well as systems that are important to safety. In particular, these reports consider the potential role of the turbine generator system in mitigating certain accidents. These two reports were reviewed in determining whether the listed variables (D26-D30) should be added to the Regulatory Guide 1.97 list.

The NUREG evaluations used a systematic approach to derive a variables list. The basic approach of the analysis was to focus on those accident conditions under which the operator is most likely to be confronted with "and/or" accident conditions which result in the most serious consequences should the operator fail to accomplish his required tasks. This is a probabilistic event-tree-type of study, and the reports used the sequences of the Reactor Safety Study (WASH 1400), and similar studies. The events in each sequence that involved operator action were identified; also, events were added to the event tree to include additional operator actions that could mitigate the accident. The event tree defines a series of key plant states that could evolve as the accident progresses and as the operator attempts to respond. Thus the operator's informational needs are linked to these plant states.

NUREG/CR-2100 is a BWR evaluation undertaken to address appropriate operator actions, the information needed to take those actions, and the instrumentation necessary--and sufficient-- to provide the required information.

The sequences evaluated were:

a. Anticipated transient followed by loss of decay-heat removal,

b. Anticipated transients without scram (ATWS),

c. Anticipated transient together with failure of HPCI, RCIC, and low-pressure ECCS,

d. Large loss of coolant accident (LOCA) with failure of emergency core-cooling systems,

e. Small LOCA with failure of emergency core-cooling systems.

The Regulatory Guide 1.97 list is based on accidents that result in an isolated NSSS. The NUREG documents considered accidents that could be prevented or mitigated by using water inventory and the heat sink in the turbine plant.

Five of the 15 variables identified in the NUREG, but not in Regulatory Guide 1.97, are recommended as Type D, Category 3 additions to the Regulatory Guide 1.97 list. Four of these variables are in the turbine plant: the turbine bypass valve position, condenser hotwell level, condenser vacuum, and condenser cooling water flow. These variables provide a primary measure of the status of a heat sink or water inventory in the turbine plant. The turbine-plant systems are not to be classed as "safety systems" or as systems important to safety. The addition of reactor primary-loop recirculation as a variable is also recommended.

HCGS has implemented these four variables plus reactor primary loop recirculation (Variable D26-D30) as plant specific Category 3 items in accordance with Regulatory Guide 1.97 considerations.

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Note that HCGS has implemented variable D29 (condenser cooling water flow) by monitoring the circulating water temperature rise across the condenser as a positive ΔT across the condenser coupled with no decrease in condenser vacuum is an adequate indication of condenser cooling water flow.

1.8.1.97.4.9 ISSUE 9 - VARIABLE E2

Regulatory Guide 1.97 specifies that "Reactor building or secondary containment area radiation" (variable E2) should be monitored over the range of 10^{-1} to 10^{4} R/h for Mark I and II containments, and over the range of 1 to 10^{7} R/hr for Mark III containments. The classification for Hope Creek is Category 2; for Mark III, the classification is Category 1.

As discussed in the variable C14 position statement (Issue 6), reactor building area radiation is an inappropriate parameter to use to detect or assess primary containment leakage.

Therefore, it is the position of HCGS that the specified reactor building area radiation monitors are not required for HCGS.

1.8.1.97.4.10 ISSUE 10 - VARIABLE E3

Regulatory Guide 1.97 specifies in Table 1, variable E3, that radiation exposure rate (inside buildings or areas where access is required to service equipment important to safety) be monitored over the range of 10^{-1} to 10^{4} R/hr for detection of significant releases, for release assessment, and for long-term surveillance.

In general, access is not required to any area of the reactor building in order to service safety-related equipment in a post-accident situation. When accessibility is reestablished in the long term, it will be done by a combination of portable radiation survey instruments and post-accident sampling of the reactor building atmosphere. The existing lower-range (typically 3 decades lower than the Regulatory Guide 1.97 range) area radiation monitors would be used only in those instances in which anticipated radiation levels were within measurable instrument ranges.

It is HCGS's position that this parameter was modified to allow credit for existing area radiation monitors. That is, this parameter should be reclassified as Category 3 with the ranges specified on Table 11.5-1.

1.8.1.97.4.11 ISSUE 11 - VARIABLE E13

Regulatory Guide 1.97 requires installation of the capability for obtaining grab samples (variable E13) of the containment sumps and the reactor building sumps for the purpose of release assessment, verification, and analysis.

The need for sampling a particular sump must take into account its location and the design of the plant in which it is installed. For all accidents in which radioactive material would be in the HCGS drywell sumps, these sumps will be isolated and will overflow to the suppression pool. A suppression pool sample can therefore be used as a valid alternative to a drywell sump sample.

The analysis of reactor building sumps liquid samples can be used for release assessment, as suggested in Regulatory Guide 1.97 only for those designs in which potentially radioactive water can be pumped out of a controlled area to an area such as radwaste. For designs in which sump pump-out is not allowed on a high-radiation or a LOCA signal, or in which the water is pumped to the suppression pool, a sump sample does not contribute to release assessment. The use of the subject sump samples for verification and analysis is of little value; a sample of the suppression pool and reactor water, as required by other portions of Regulatory Guide 1.97 provides a much better measurement for these purposes. The guidelines recommended by the BWR Owners Group and GE shall be followed in lieu of Total Dissolved Gas Analysis. This was agreed to in a meeting between NRC management (R. Vollmer) and GE (F. Quick) dated December 12, 1983.

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

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QUESTION 430.67 (SECTION 9.5.2)

The description of the intraplant and interplant (plant to offsite) communication systems is inadequate. Provide a detailed description for each communication system listed in Section 9.5.2.2 of the FSAR. The detailed description shall include an identification and description of each system's power source, a description of each system's components (headsets, handsets, switchboards, amplifiers, consoles, handheld radios, etc.), location of major components (power sources, consoles, etc.) and interfaces between the various systems. (SRP 9.5.2, Parts II & III)

RESPONSE

Section 9.5.2.2 has been revised to include additional description for each communication system, including offsite communications systems and power supplies.

1/5

"Merge-Isolate" capability for the plant and refueling platform PA systems is provided at the communication cabinet located in the main control room.

The telephone system of Section 9.5.2.2.2 can be patched into the PA system page channel to enable communications to be conducted between telephone and PA handset locations.

The radiation alert signal and the fire alarm signal are transmitted over the paging channel of the PA system, overriding its normal use. The PA system is fed from an uninterruptible power source, as shown on Figure 8.3-11.

DELETE

9.5.2.2.2 Telephone System

INSERT "A"

~~The automatic telephone system is furnished and maintained by the New Jersey Bell Telephone Company. The system has a capacity of approximately 300 lines. The power supply for this system consists of an independent charger and battery with a capability of operating the entire plant telephone system for a minimum of 8 hours after a loss of the normal ac supply. Direct lines, including the emergency notification system (ENS) to the Nuclear Regulatory Commission offices, are powered from a station inverter to ensure continued direct communications during loss of offsite power (LOP). Drawing Number E-1467-0 (drawing referenced in Section 1.7) illustrates the location of the components in a riser diagrammatic form.~~

9.5.2.2.3 Two-Way Radio Communications System

Two radio communication systems are provided. One System is for security personnel use and it is described in Section 13.6. The other system is for station personnel use as described herein. This radio communication system serves as an alternate communication system to the public address and the telephone systems. This system consists of three remote control consoles, a primary and a backup base repeater stations with manual switchover provision, handheld transceivers (radios) and antenna divider network with antennas and transmission lines distributed throughout the power block.

The radio system is used by the fire brigade, described in Section 9.5.1.5.2, and by other station personnel. However, during the preoperational testing phase of the plant, the radio system is used by startup personnel. The radio system also has interface capability for connection with the Salem radio system.

INSERT "A" TO PG. 9.5-65:

The telephone system at Hope Creek is a Private Automatic Branch Exchange (PABX) supplied and installed by the telephone company.. The system is equipped with the latest software package and dual processing for back-up reliability.

Hope Creek primary communication paths entering the PSE&G Network, including the EOF (Emergency Operations Facility), will be through PSE&G's private Microwave System. The lines to the corporate headquarters in Newark and the Salem EOF will be routed "first-choice" through the PSE&G Microwave System. PSE&G's microwave is equipped with its own battery chargers and emergency 8-hour batteries, and backed up with UPS (Uninterruptable Power Supply) and diesel generator.

Communication channels may also enter or exit Hope Creek Generating Station via the PSE&G Network through two additional paths, provided by the telephone company. These paths are Salem Generating Station Switch (PABX) tie lines and an alternate back-up path. The Salem path will be the primary path, entering the Salem C.O. (Central Office telephone company) through either a hardwire link or the telephone company's microwave system. The Salem switch also has direct link to the EOF and PSE&G's Central New Jersey office (Moorestown). The alternate back-up path will be a direct link to access the PSE&G Network without the use of the Salem station switch or telephone company Salem C.O. The Salem Generating Station switch (PABX) is equipped with a UPS system and diesel generator. The Hope Creek switch (PABX) will also be equipped with a UPS system and diesel generator.

Upon failure of telephone equipment or ~~in emergency situa-~~ tions, necessary telephone communications for pertinent personnel will be maintained. These communication channels will be available in the form of Newark Centrex extensions via Microwave which will be placed at strategic locations.

QUESTION 430.79 (SECTION 9.5.4)

Discuss the means for detecting or preventing growth of algae in the diesel fuel storage tank. If it were detected,describe the methods to be provided for cleaning the affected storage tank. (SRP 9.5.4, Fart III)

RESPONSE

The diesel fuel oil storage tanks are provided with manholes for inspection, as described in Section 9.5.4.4.

1/2

Insert A

Biological growth in diesel fuel oil storage tanks may occur if water is allowed to accumulate inside the tank during long term storage. Water accumulation is prevented by: ~~is removed by:~~

1. sampling fuel delivery trucks prior to loading new fuel oil in the storage tanks to prevent water intrusion
2. draining fuel oil storage tanks in accordance with the applicable recommendations provided in ~~Nureg~~ Regulatory Guide 1.137.

~~Fuel~~

Diesel fuel will be sampled during every ~~the first~~ refuelling outage ~~and every three years there after~~ to detect the presence of algae in the storage tanks. If algae is detected, a tribology professional will be consulted to provide recommendations to ~~to~~ eliminate the algae contamination.

In addition, surveillance testing to demonstrate diesel engine operability will include performance monitoring of the diesel engine fuel oil system. Fuel oil strainer and filter differential pressures will be monitored. Cleaning of the strainers and filters will be performed at the onset of an increase in differential pressure. Residue ~~&~~ will be analyzed to determine the source of contamination and the need for storage tank cleaning.

2/2

QUESTION 430.113 (SECTION 9.5.5)

Figure 9.5-23 of the FSAR shows the fuel injector cooling subsystem of the diesel engine cooling water systems. The drawing shows the flow of cooling water to the fuel injectors as going from the hot leg (inlet) of the intercooler heat exchanger through a three way thermostatic valve (refer to request 430.110 for purpose of this valve) through the fuel injectors and to the expansion tank. The line is labeled 8 gpm at 120°F. Preheating during standby conditions to enhance first try starting reliability of the emergency diesel generator is not provided for this intercooler and injector cooling water system. Insufficient data and description is given on this system (See Request 430.100) to determine the purpose and adequacy of the system. It appears from the drawing that instead of cooling the fuel injectors the purpose of the system is to preheat the diesel fuel oil prior to injection into the cylinders. Provide the following:

a. Describe the purpose of the fuel injector portion in the diesel engine cooling water system. Since the hot leg of the cooling system would normally exceed 120°F, justify the design of the system as described above or correct the design and justify why preheating is not provided to this portion of the diesel engine cooling water system during standby operations to enhance first try starting reliability.

b. Justify why preheating of the balance of the intercooler an injector diesel engine cooling water system during standby conditions to enhance first try starting reliability of the diesel generator is not provided.

(See Request 430.145 for conditions when preheating may be necessary) (SRP 9.5.5, Part III).

RESPONSE

a. The injector cooling system furnishes cooling water t the fuel injector nozzles. This cooling water functions to extend injector nozzle life by removing the heat resulting from fuel oil combustion.

The optimum water temperature for cooling the injecti nozzles is about 120°F. Hotter water from the jacket water system is mixed with cooler water from the intercooler water system in the thermostatic 3-way proportioning valve to maintain this temperature. Th mixed water is then directed through headers on the t cylinder banks to the injection nozzles on each cylinder. The water then flows into return headers each cylinder bank and is piped to the jacket water expansion tank, returning to the jacket water and intercooler water systems through the pump surge lines.

b. The purpose of the system is not to preheat the fuel oil, but to cool the injection nozzles as described in part (a). Thus, the system is not required to operate during standby operation.

INSERT B

c. The manufacturer has confirmed that the first try starting reliability of the diesel generators is unaffected by the intercooler's initial cooling water temperature, and as such, does not require cooling water preheat during standby conditions.

2/5

, which is described in response to que[stion] 43[0...]

430.113

INSERT A — The 3-way thermostatic control valve, which is described in response to que[stion] 43[0...], is located in the two systems such that if heating or cooling of the injector cooling water is necessary to attain the 120°F optimum cooling wate[r] temperature, the required amount of water is added from the jacket water system and the intercooler water systems as necessary.

Failure of the 3-way thermostatic control val[ve] in either position, causing all jacket water ~~flow~~ or all intercooler water ~~flow~~ flow to the

3/5

430.113 (cont.)

(INSERT A) cont.

injector nozzles, would not have an adverse effect on the diesel fuel oil injector nozzles. If failure of the 3-way thermostatic valve occurred in either position the cooling water to the injector nozzles could not get any hotter than the jacket water cooler outlet temperature (166.6°F) or any cooler than the intercooler heat exchanger outlet temperature (110°F). Colt Industries has confirmed that the nominal temperature spread between the jacket water cooler outlet and

4/5

430. 113 (cont.)

(INSERT A) cont.

the intercooler heat exchanger outlet is not sufficient to cause any problems in the injector cooling system

INSERT B

Colt Industries has confirmed that the injector cooling water system is for cooling the diesel fuel oil injector nozzles and is not intended for preheating of the nozzles.

QUESTION 430.123 (SECTION 9.5.6)

Diesel generators in many cases utilize air pressure or air flow devices to control diesel generator operation and/or emergency trip functions such as air operated overspeed trips. The air for these controls is normally supplied from the emergency diesel generator air starting system. Provide the following:

a. Expand your FSAR to discuss the diesel engine control functions supplied by the air starting system or any air system. The discussion should include the mode of operation for the control function (air pressure and/or flow), a failure modes and effects analysis, and the necessary P&ID's to evaluate the system.

b. Since air systems are not completely air tight, there is a potential for slight leakage from the system. The air starting system uses a non-seismic air compressor to maintain air pressure in the seismic Category I air receivers during the standby condition. In case of an accident, a seismic event, and/or loop, the air in the air receivers is used to start the diesel engine. After the engine is started, the air starting system becomes non-essential to diesel generator operation unless the air system supplies air to the engine controls. In this case the controls must rely on the air stored in the air receivers, since the air compressor may not be available to maintain system pressure and/or flow. Your air starting system is used to control engine operation, with the compressor not available, show that a sufficient quantity of air will remain in the air receivers, following a diesel engine start, to control engine operations for a minimum of seven days assuming a reasonable leakage rate.

(Refer to Request 430.64 for additional control air requirements on diesel engine restart) (SRP 9.5.6, Part III)

RESPONSE

Sections 9.5.6.2 and 9.5.6.5 have been revised to further describe the standby diesel generator air starting system. There are no control functions, supplied by the starting air systems, which affect the continued safety-related operation of the diese generator units. A failure modes used effects analysis and a system schematic are provided in Table 9.5-10 and on Figure 9.5-26 respectively. The only control function, other than starting the units (Reference Question response 430.116), i to shutdown the unit. This function is not part of the safety-related operation of the unit and is discussed in the response t

Question 430.121. Since the only control functions are the starting and stopping of the units, the seven day analysis is not applicable. It should be noted that excessive receiver leakage without compressor recharging would result in a low pressure alarm which in turn would indicate a trouble alarm in the control room. See Question 430.122 for a detailed discussion of air receiver capacities.

The standby diesel engine manufacturer has ~~confirm~~ confirmed the ~~that~~ above information by letter response as follows:

"Other than for shutting down the engine (which function is not significant to starting and operating the diesel), there are no control functions that require air from the starting air system. [There is an instrumentation function, but operation (or failure of operation) of these instruments will not impair the function of the diesel engine, once it has been started.]"

~~[Not all of the diesels we (Colt) have provided are the same. Some do include some control functions (operation of thermostatic valves, etc) that require control air. This is determined by individual spec requirements. Hope Creek HCGS does not require air for any control function.]~~ "

indent

QUESTION 430.136 (SECTION 7.5.5, 9.5.7)

You state in the FSAR that cooling to the diesel engine cooling water systems and the lube oil system is provided by the Safety Auxiliaries Cooling System (SACS). Figures 9.5-23, 9.5-24 and 9.5-27 of the FSAR show the intercooler heater exchanger, the jacket water heat exchanger and the lube oil heat exchanger connected in series with the SACS providing cooling to the intercooler heater exchanger first and the lube oil heat exchanger last. Other plants with the same type of engine design have the lube oil heat exchanger cooled by the diesel engine jacket water system. Rather than cooled by a service water system or have a separate independing connection to the service water cooling system. Justify that your design of having the lube oil heat exchanger in series with the cooling water heat exchangers, will adequately cool and maintain lube oil temperature within manufacturer's specifications during engine operation. (SRP 9.5.7, Part I, II, and III)

RESPONSE

It is the manufacturer's design to have the Intercooler Heat Exchanger, the Jacket Water Heat Exchanger, and the Lube Oil Heat Exchanger cooled by the series arrangement shown in the referred drawings. We are committed to supply inlet cooling water to these Diesel Generator Coolers in accordance with the manufacturer's requirements. these requirements are shown in Table 9.2-4.

Colt confirms that 95°F inlet temperature of cooling water is adequate for proper cooling of this unit. The series system as outlined (intercooler heat exchanger, jacket water heat exchanger, and finally, ~~L.O.~~ lube oil heat exchanger) is the manufacturer's proven standard design.

The ~~L.O.~~ lube oil heat exchanger has been sized for the expected water temperature at the outlet of the jacket water heat exchanger.

Hope Creek Generating Station SDG is not different than all of the other SDG's. ~~Hope Creek~~ HCGS uses ~~our~~ Colts "standard" design. Some others have been different as a result of either specification requirements or specific site requirements. In all cases, Colt analyzes the specific units requirements and sizes all heat exchanger equipment accordingly.

QUESTION 430.137 (SECTION 9.5.7)

You state in Section 9.5.7.1 of the FSAR under specific design criteria that the temperature of the lubricating oil is automatically maintained above a minimum value by means of an independent recirculation loop including its own pump and heater, to enhance first try starting reliability of the emergency diesel generator when in the standby condition. The rocker arm lubrication system is an independent subsystem of the diesel lube oil system which is connected to the main system by a float valve in the rocker arm oil reservoir. From the information available, it appears that the lube oil in the rocker arm lubrication system will never be preheated unless the oil level is low enough to open the float valve. If this is the case what means have you provided for preheating the rocker arm lubricating oil or justify why preheating is unnecessary. (See request 430.145 for conditions when preheating may be necessary.) (SRP 9.5.7, Parts II and III)

RESPONSE

The rocker arm lubricating oil ~~will~~ is not ~~be~~ pre-heated. This system was designed by the diesel engine manufacturer, and based upon their many years of experience, they have determined that ~~preheating of rocker arm oil is not necessary.~~ The manufacturer's recommendation is that the rocker arm prelube pump be run once a day for 5 minutes as is discussed in response to Question 430.130.

Specific heating (pre-heating) or cooling of the rocker arm lube oil system is not required.

Since the rocker arm section of the engine is insensitive to oil viscosity. The main requirement is that there be a supply of oil. The rocker arm area is heated by its proximity to the cylinder heads which are part of the jacket water system.

16

11

QUESTION 430.138 (SECTION 9.5.7)

In Sections 9.5.7.3 and 9.5.7.5 of the FSAR you discuss the level alarms associated with the lube oil system. You state that "the rocker arm lube oil reservoir level is monitored for high level and the level is maintained by a level control valve." No mention is made of a reservoir low level alarm. A failure of the level control valve to maintain lube oil level in the rocker arm reservoir could result in inadequate or no lubricating oil for the rocker arms, leading to diesel generator unavailability and/or failure. This is an unacceptable condition. Provide a low level alarm for the rocker arm lube oil reservoir. (SRP 9.5.7, Part III)

RESPONSE

The rocker arm lubrication system is also monitored by a rocker arm lube oil pressure low switch (KPLA), which would initiate an alarm in the event that insufficient pressure is available in the rocker arm lube oil system due to any of the following causes:

a. the filters are plugged,

b. the system has run low on oil level due to malfunction of the automatic level fill valve,

c. the engine driven pump (or its drive) has failed.

Upon the alarm, the motor driven rocker arm lube oil pump is also started. If the problem was caused by a or b, the operator must take appropriate action.

The function of the high level alarm switch is to alert personnel that:

a. Fluids other than oil, such as a fuel oil leak at an injector, or a water leak in the cylinder head (between the jacket water system and rocker arm lube oil drain system) have entered the rocker arm lube oil system.

b. The lube oil supply valve (float valve) has malfunctioned (open).

In either case, the operator must investigate and remedy the problem. Therefore a low level alarm for the rocker arm lube oil reservoir is not required.

1/2

Sure

Response cont'd

430.138

It is Colt's position ~~is~~ that the rocker arm low lube oil pressure alarm is sufficient to determine a problem in this system. The unit could probably run for several minutes with a "low pressure" as long as there was some pressure to maintain flow.

If the loss of pressure was caused by a failure of the float level valve to admit oil, oil could be added to the tank by hand. This is basically a closed system and the rate of oil consumption is very low.

Based on the above information, Colt does not feel their design requires a low level alarm for the rocker arm lube oil reservoir.

2/2

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

1/5

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

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%.

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).

d. A curve of the 12CR.PC2 class engine derating for ambient pressure (altitude) is attached (Figure 430.145-1). 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 depresssions.

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.

2/5

①

430.145

INSERT I

C. It is Colt's position that there, as stated in the revised response to question 430.111 and documented in Colt's letter from Mr. V.T. Stonehock to Mr Clemenson (NRC) dated September 11, is enough conservativity in the ~~direction~~ operating instructions given on Low Load (Idle) operation, ~~given~~ ~~the~~ 1975 ~~letter to the NRC~~, that the SDG ~~direction stands~~ should operate successfully 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.

3/5

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 temperature the engine can be started and idled ~~as a last step~~ to maintaining temperatures in the standby range.

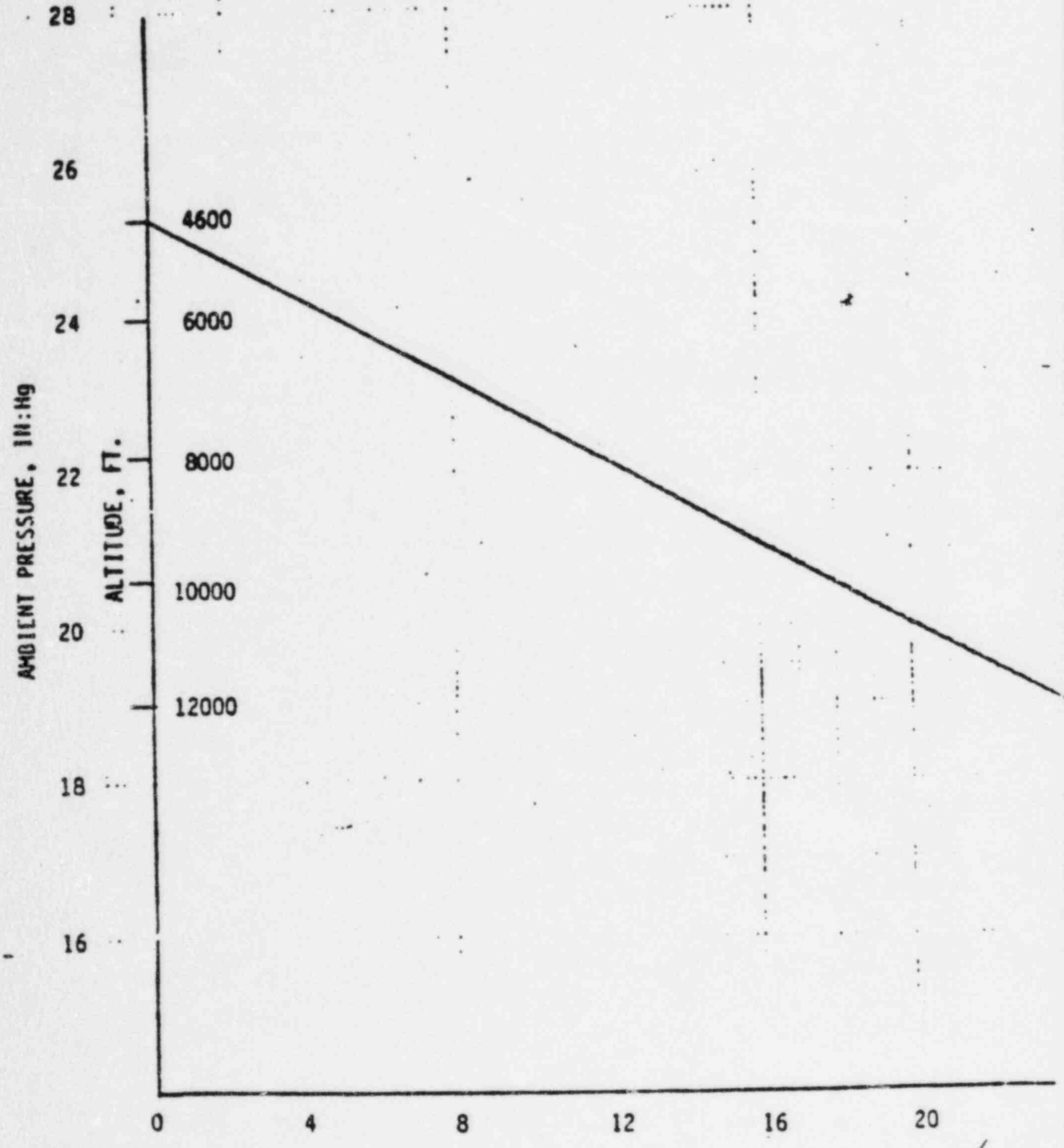
It is not anticipated that the Colt Industries supplied diesel engines would not start or operate at temperatures below the specified low temperature.

Colt Industries has supplied diesel engines which have performed successfully in much more severe climates having similar equipment.

4/5

0261889

12 CYL. PC2 (4238KW AT 514 RPM)
AMBIENT PRESSURE VS. PERCENT DERATE



5/5

10/20/83 R.C.H.

430.145 (18N)

FIGURE 430.145-1

QUESTION 430.149 (SECTION 9.5.8)

Figures 1.2-35 through 1.2-39 show the routing of the diesel engine exhaust system from the diesel generator room to the roof of the auxiliary building. The figures show that the exhaust mufflers for all the diesel generators are located in a common corridor (Elevation 102'-0") and that the exhaust stacks pass through the following areas:

1. Remote D/G control and vital switchgear areas (elevation 130'-0")

2. Vital battery control rooms (elevation 137'-0")

3. Switchgear HVAC Area (elevation 163'-0")

4. Diesel and control rooms HVAC area (elevation 178'-0")

The exhaust system is considered a high energy system by virtue of temperature. A exhaust system pipe break in any one of these areas and a single active failure in one of the other diesels or just pipe break in the exhaust system in the muffler corridor, switchgear HVAC area, or diesel and control room HVAC area could result in an inability to shut down the plant.

The figures referenced above do not clearly show or decribe the diesel engine exhaust stack enclosures. Describe the stack enclosure in each of the areas noted above and show that an exhaust stack break in any one of these areas will not result in the inability to shut down the plant or result in failure or unavailability of all diesel generators. (SRP 9.5.8, Parts II and III)

RESPONSE

As discussed in response to Question 430.82 the SDG combustion air exhaust system is not classified as high energy system. Therefore a high energy pipe break ~~in~~ is not considered.

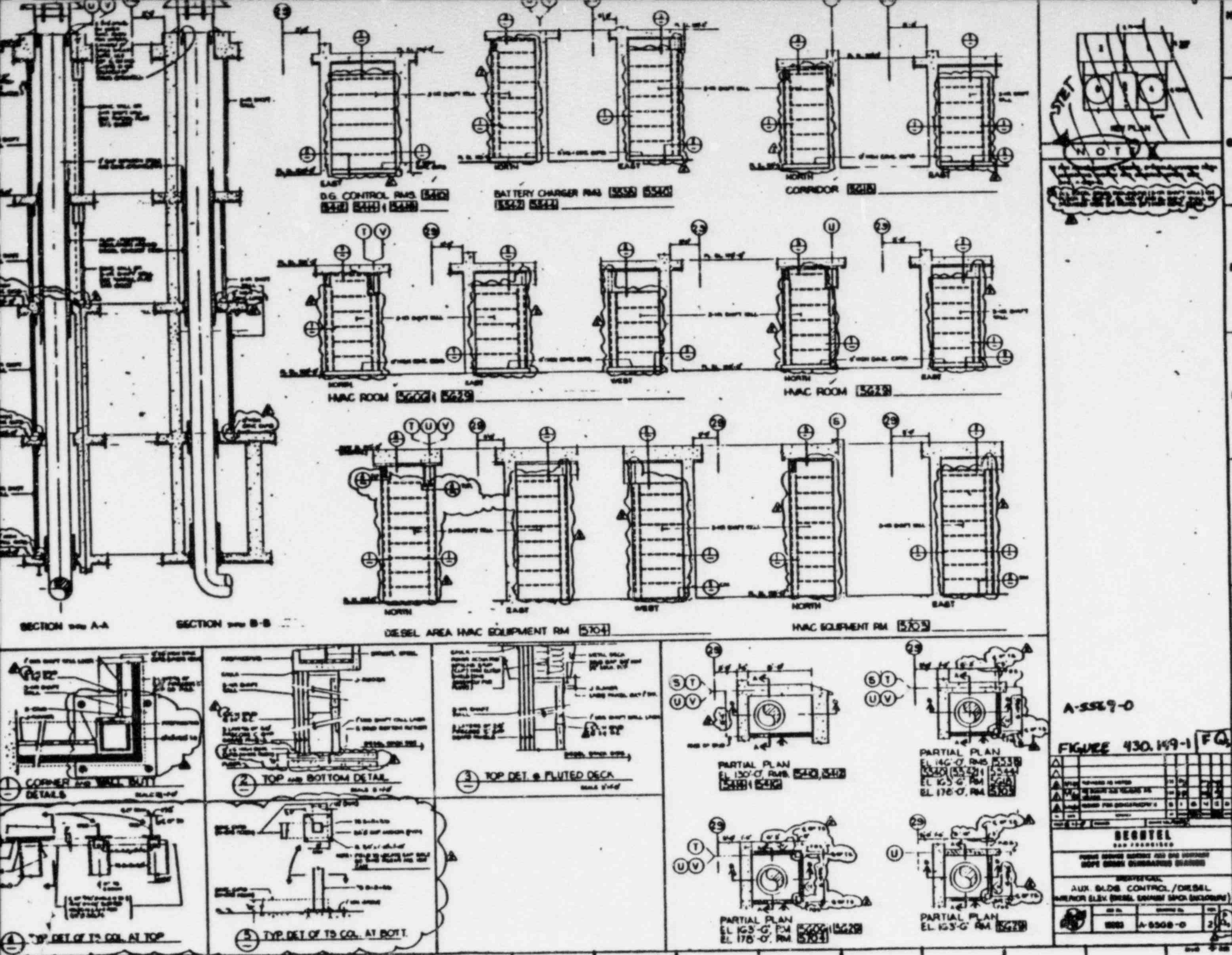
The exhaust stack which passes through the areas mentioned in the above question is designed to Seismic Category I requirements, as discussed in Section 3.7. ~~It is also provided with liner/partition panels~~ ← INSE as shown on Figures 1.2-35 through 1.2-39, c
430.149-1, to minimize heat rejection and noise in the areas through which it passes.

1/3

430.149

The standby diesel generator exhaust stack is provided with aluminium jacketed insulation, around the stack, and three hour fire barrier exhaust shaft walls ...

2/3



D.G. CONTROL RMS. 5410
BATTERY CHARGER RMS
CORRIDOR
HVAC ROOM
HVAC ROOM 5629
DIESEL AREA HVAC EQUIPMENT RM 5704
HVAC EQUIPMENT RM 5703
SECTION A-A
SECTION B-B
NORTH
EAST
WEST
NOTE
KEY PLAN
3/3
CORNER AND WALL BUTT DETAILS
TOP AND BOTTOM DETAIL
TOP DET. @ FLUTED DECK
TYP. DET OF TS COL. AT TOP
TYP. DET OF TS COL. AT BOTT.
PARTIAL PLAN
EL 130'-0" RMS. 5410, 5412
PARTIAL PLAN
EL 146'-0" RMS
EL 163'-6" RM
EL 176'-0" RM
PARTIAL PLAN
EL 163'-6" RM
EL 176'-0" RM 5704
PARTIAL PLAN
EL 163'-6" RM 5629
A-5569-0
FIGURE 430.149-1
BECHTEL
SAN FRANCISCO
AUX. BLDG. CONTROL/DIESEL
A-5508-0

Response to NRC Audit

Meeting Date: January 10, 1984

Question No.: A-3

Revised Response
Revision 1
6/30/84

QUESTION: Provide comparison between basemat response spectra and regenerated response spectra at basemat.

RESPONSE: Comparison of spectra for 2% damping was provided in the original response for both SSE and OBE cases.

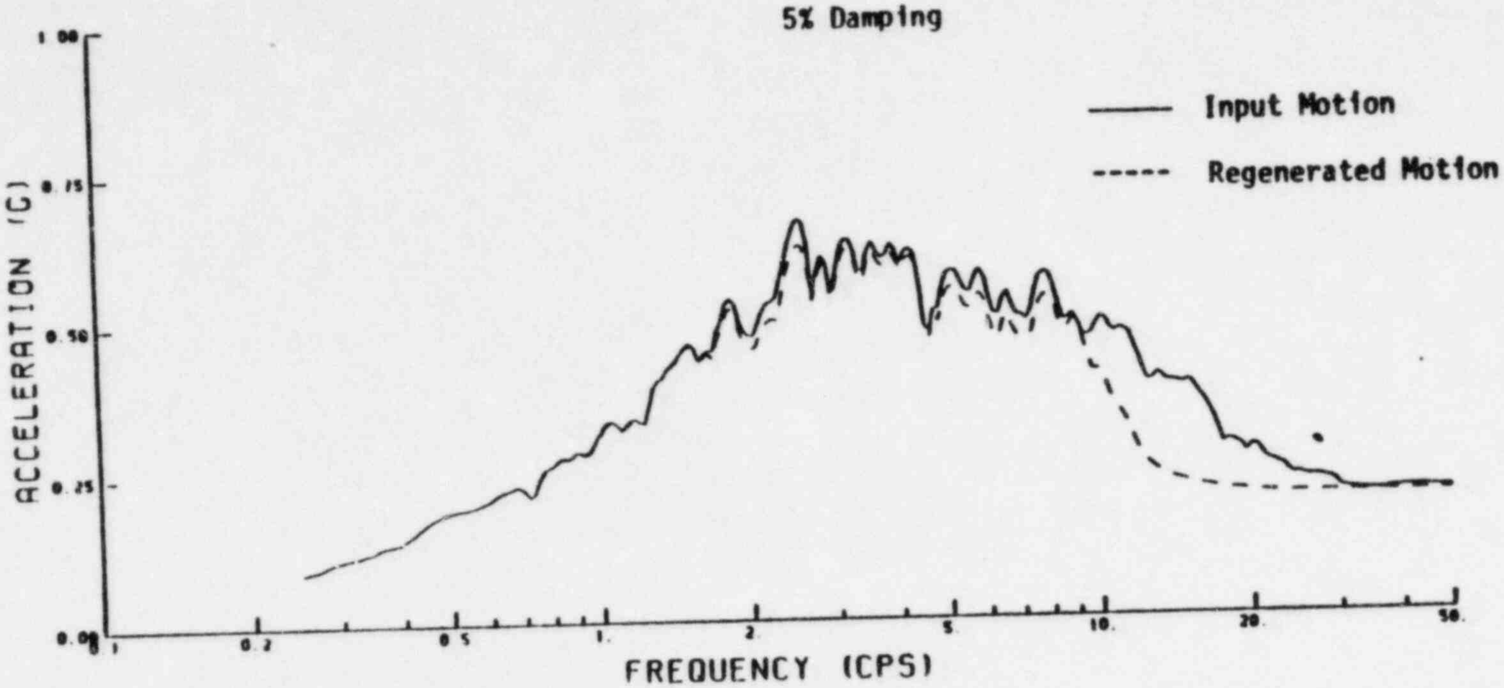
ADDITIONAL INFORMATION REQUESTED: Provide the same comparison for 5% damping value.

RESPONSE: The attached two figures provide the comparison between the response spectra of the defined input motion and regenerated response at the basemat elevation. These spectra were generated for 5% damping. Figures 1 and 2 show the comparison for the SSE and OBE events, respectively. The spectra for the input motion at the basemat level is obtained from section 3.7.1.2 of the Hope Creek FSAR.

Comparison of the response spectra for the input motion versus the R.G. 1.60 spectra is provided in response to NRC question 220.20.

A 12 Hz. cutoff frequency has been used in these analyses. As observed from the attached figures, the match between the two spectra are adequate below the 12 Hz. cutoff frequency. The adequacy of the 12 Hz. cutoff frequency is addressed in a separate response to Question A-12 from the audit meeting on January 11, 1984.

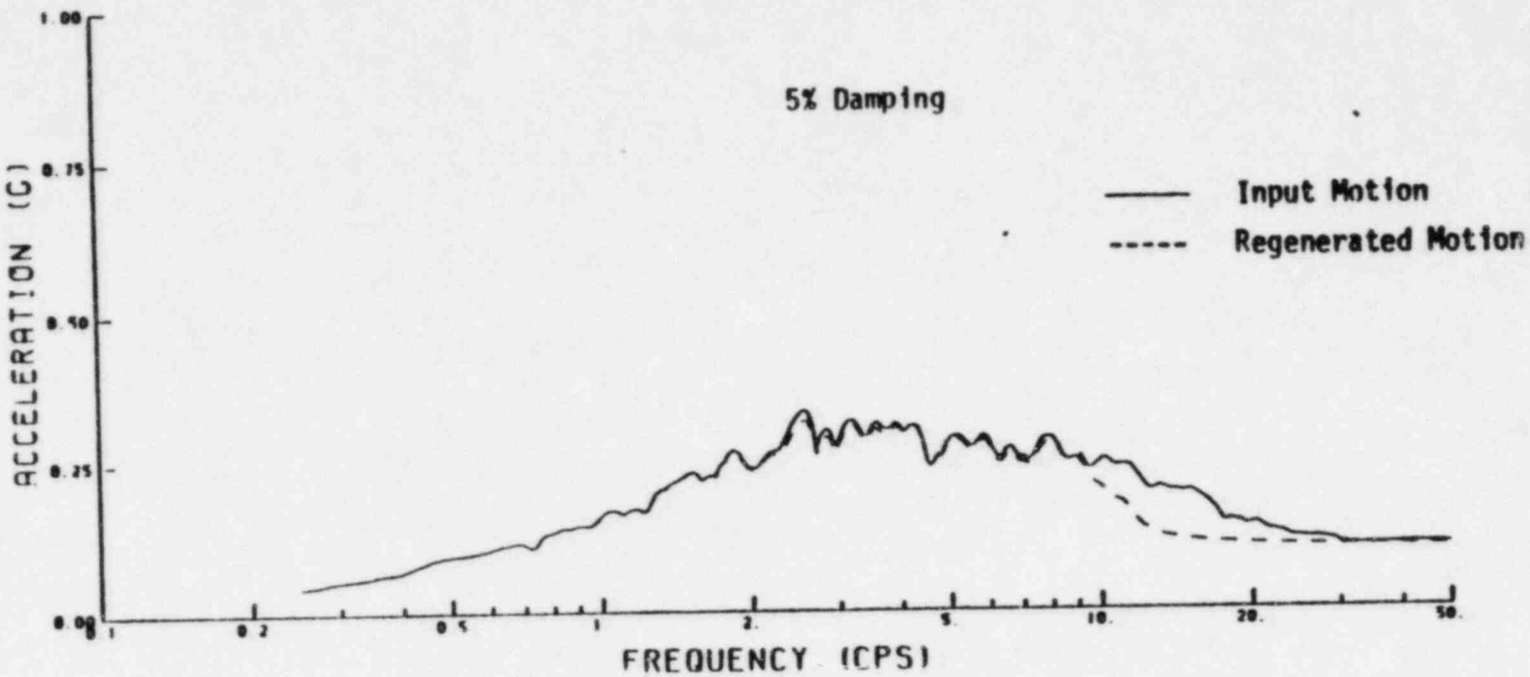
Revised Response
January 10/A-3



SSE CASE - AVERAGE SOIL PROPERTIES
COMPARISON OF RECOVERED VS. INPUT MOTIONS
ELEVATION 54.0 FT., FREE FIELD

HOPE CREEK
SOIL STRUCTURE INTERACTION

Revised Response
January 10/A-3



OBE CASE - AVERAGE SOIL PROPERTIES
COMPARISON OF RECOVERED VS. INPUT MOTIONS
ELEVATION 54.0 FT., FREE FIELD

HOPE CREEK
SOIL STRUCTURE INTERACTION

Response to NRC Audit Meeting

Date: January 10, 1984

Question no.: A-4

Revised Response
Revision 1

6/30/84

QUESTION: Describe method of establishing rocking time histories R(t).

RESPONSE: Method is described in original response to this question.

ADDITIONAL INFORMATION REQUESTED: Provide mass participation factors for the first few modes of the structure, including the dummy modes for rocking. Also provide a comparison of the response spectra for the input versus the response rocking time histories at the location of the dummy large rotational mass.

RESPONSE: Table 1 of the attachment to this response summarizes the mass participation factors for the first 10 modes of the Reactor Building, Unit 1. Note that as indicated in the original response, mode 1 (period = 200 sec.) corresponds to the dummy rocking mode about E-W axis and mode 2 (period = 150 sec.) corresponds to the dummy rocking mode about N-S axis.

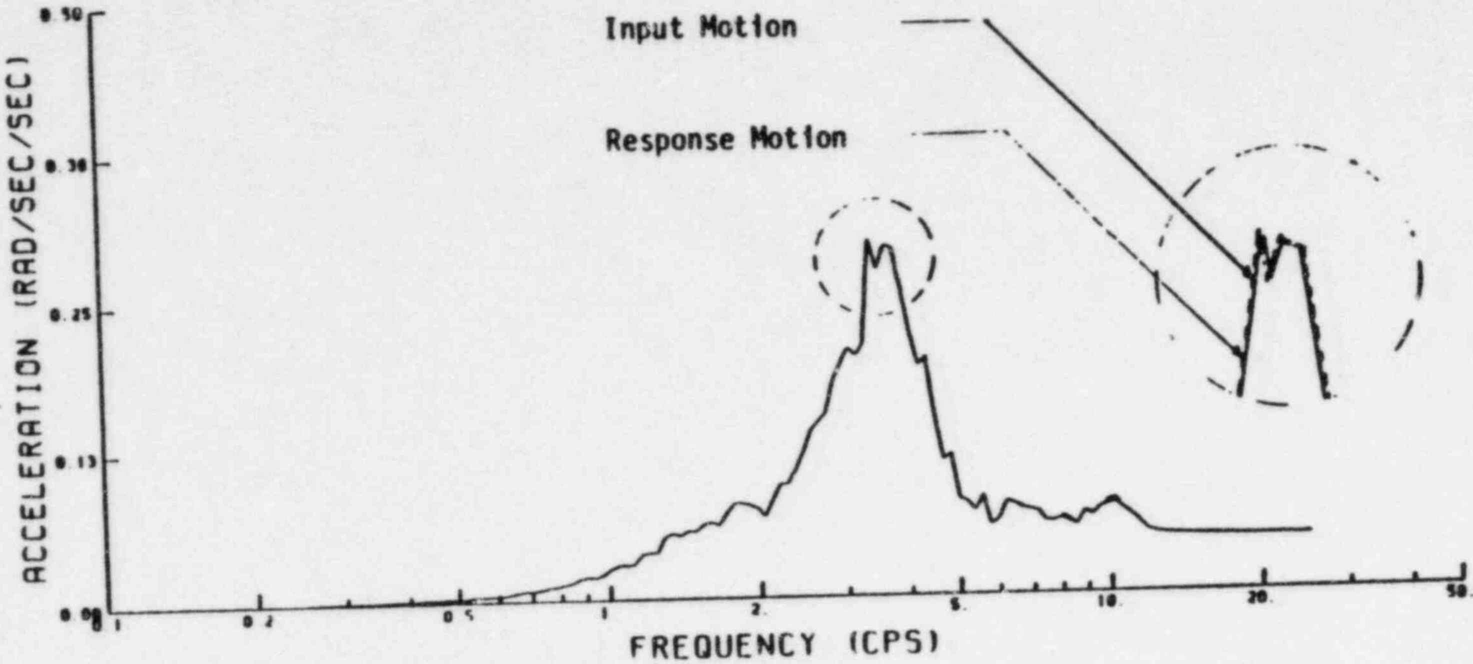
It is observed that for the x-direction earthquake (i.e., N-S translation and rocking about E-W axis) the mass participation of mode 1 is neglible compared to true structural modes (about an order of 10^{-5} lower). Similarly for the y-direction earthquake (i.e., E-W translation and rocking about N-S axis) the mass participation of mode 2 is considerably lower than other modes. Therefore, these dummy modes do not participate in the actual response of the structure.

Furthermore, to verify this point a comparison of the response spectra of input versus response time histories at the location of the dummy large rotational mass point is provided for rocking motion about the E-W axis. As the two response spectra are identical at all frequency points, it is concluded that the inclusion of these dummy modes has no influence on the response of the structure.

TABLE 1

Mode Number	Frequency (CPS)	Mass Participation Factor X	Y
1	.005	- 6.8925E-08	1.2559E-01
2	.007	2.2572E-01	- 1.8687E-07
3	2.731	- 6.7145E+00	- 1.7198E+01
4	4.100	6.5590E+01	1.3645E+01
5	4.226	- 2.2965E+00	4.4283E+01
6	4.346	- 1.9056E-01	2.2708E+00
7	4.414	1.8184E+00	- 5.3500E+01
8	7.092	- 8.7972E+00	- 2.7781E-01
9	7.103	9.2005E-01	6.3203E+00
10	8.821	1.8346E+01	- 3.2680E+01

Revised Response
January 10/A-4



BASEMAT ELEV. 54.0 FT.
ROCKING MOTION ABOUT EW AXIS
REACTOR BUILDING - SSE CASE

HOPE CREEK
SOIL STRUCTURE INTERACTION

FIGURE 1

Response to NRC Audit

Meeting Date: January 10, 1984

Revised Response
Revision 1
July 10, 1984

Question No.: A-11

QUESTION: Justify why it is acceptable to use gross concrete section.

RESPONSE:

In determination of the seismic response of Hope Creek Category I structures, gross (uncracked) concrete sections have been used when calculating the stiffnesses of the concrete structural elements. The use of gross section properties in this application is judged to be reasonable and appropriate based on the following:

1. ACI SP-60 Recommendations

   The use of gross concrete section properties is consistent with ACI recommendations. ACI publication SP-60 (Reference A-11-1), which addresses response of structures to vibratory loads, recommends neglecting cracking and basing section properties on the gross section, but neglecting the transformed area of reinforcing steel. This approach was used in the Hope Creek stiffness calculations.

2. Evaluation of Crack Potential under Seismic Loading

   Major Category I Buildings in the Hope Creek Plant are constructed with reinforced concrete shear walls. An evaluation was performed to assess whether gross cracking of Hope Creek shear walls could occur during the postulated seismic event. The lower elevations of the Reactor Building (approximate el. 54'), where the shear stresses are the maximum, were selected for evaluation. The following parameters were used in the evaluation when calculating the shear wall concrete strengths (Vc):

   - Concrete compressive strength determined in the 90 day cylinder tests was used.

   - ACI 349-76 code criteria (Equations 11.32 and 11.33) were used to establish the cracking concrete shear strength of the walls.

   The evaluation included calculation of the maximum seismic shear and flexural stresses in the shear walls. The cracking concrete shear strength was based on ACI code equation 11.33 which considers both flexural tension and shear and equation 11.32 which considers the diagonal tension shear cracking. The predicted seismically induced shear stresses were lower than the allowable concrete cracking shear strength specified by the ACI code. Thus, it is concluded that the shear walls which contribute most of the lateral stiffness are not expected to have gross cracking due to in-plane seismic loads

3. Effects of Other Loads (such as LOCA)

Concrete cracking is likely to occur in elements subject to high tensile or shear stress. Typically, this may occur in reinforced concrete containment structures required to resist high LOCA pressures, and in localized compartments outside of containment, due to local abnormal loads such as pipe breaks.

Hope Creek uses a Mark I containment system which consists of a steel containment shell which is separated from the concret shielding walls, and a steel torus suppression chamber. Sinc the steel containment is designed to resist the LOCA loads, the shielding walls will not be subjected to LOCA pressure loads.

Abnormal loads (due to pipe breaks) in local compartments may induce out-of-plane flexural stresses large enough to crack concrete. However, the cracking would be limited to the few walls and/or slabs in the vicinity of these loads. Furthermore, the localized cracking would not extend through the wal thickness. The associated reduction in stiffness in a few elements would not significantly affect the overall stiffness or response of the entire structure. Furthermore, the conser vative response spectrum broadening criteria (FSAR Section 3.7.2.5) used by Hope Creek will accommodate any minor shift in frequency resulting from local concrete cracking.

4. Effect on Global Inertia Force

The seismic responses of Category I structures including soil-structure interaction effect have been calculated using the finite element method and independently verified by those obtained from the impedance approach (half-space) analysis. The fundamental soil-structure interaction frequencies of the Category I structures computed using the finite element meth range from approximately 3.5 Hz to 6 Hz. These fundamental soil structure interaction frequencies are located within the "flat" portion of the NRC design spectra (Regulatory Guide 1.60) for horizontal (2.5 Hz to 9.0 Hz) and vertical (3.5 Hz to 9.0 Hz) earthquakes. If gross concrete cracking is postulated, the fundamental frequencies of the soil-structure system are likely to decrease even if the potential increase in structural damping due to cracking is not taken into consideration. Consequently, the global inertia forces will no change significantly. The same conclusion applies if the seismic responses of Category I structures are computed usin the impedance approach. Therefore, it is concluded that the change in inertia force due to frequency shift as a result o concrete cracking will not adversely affect the structural design.

CONCLUSION

Based on the above, it is concluded that the use of gross concrete section properties in the Hope Creek seismic analysis is i conformance with industry codes and practices. A review of the plant structures found that seismically induced cracking would b minor and localized and would not be sufficient to affect the gross seismic response of the structures. Thus the use of gros concrete sections is reasonable and appropriate.

REFERENCE: A-11-1, "Vibration of Concrete Structures", Publication SP-60, American Concrete Institute, Paper SP 60-12.

Response to NRC Audit Meeting

Date: January 10, 1984

Question No.: A-12

Revised Response
Revision 1
6/30/84

QUESTION: Pick one particular floor to define why particular modes were selected for development of vertical floor flexibility response spectra.

RESPONSE: All significant modes below 25.0 Hz were selected from the finite element model of the floor slab for development of vertical floor flexibility response spectra. In general, three to five modes were included with each mode representing a particular region of the floor slab.

For the Reactor Building floor slab at elevation 145.0 feet, five modes were used. Based on the mode shapes of these modes, these modes were determined to represent five different regions of the floor slab, as shown in Figure 1. These five modes were represented by five single-degree-of-freedom beam elements in the vertical floor flexibility model shown in Figure 2.

ADDITIONAL INFORMATION REQUESTED: Provide the frequencies and mode shapes used to develop vertical floor flexibility spectra for elevation 145 feet of the Reactor Building.

RESPONSE: Figure 1 shows the finite element model of the Reactor Building floor at elevation 145 feet. Regions where the response of each mode is dominant is also shown in Figure 1.

The frequencies corresponding to these five modes are given in Table 1. The inclusion of the above modes in the vertical floor flexibility model were considered as a sufficient representation of the dynamic characteristics of the floor slab for this elevation. Examining the mass participation factors, among these 5 modes 92% of the total mass of the floor is represented.

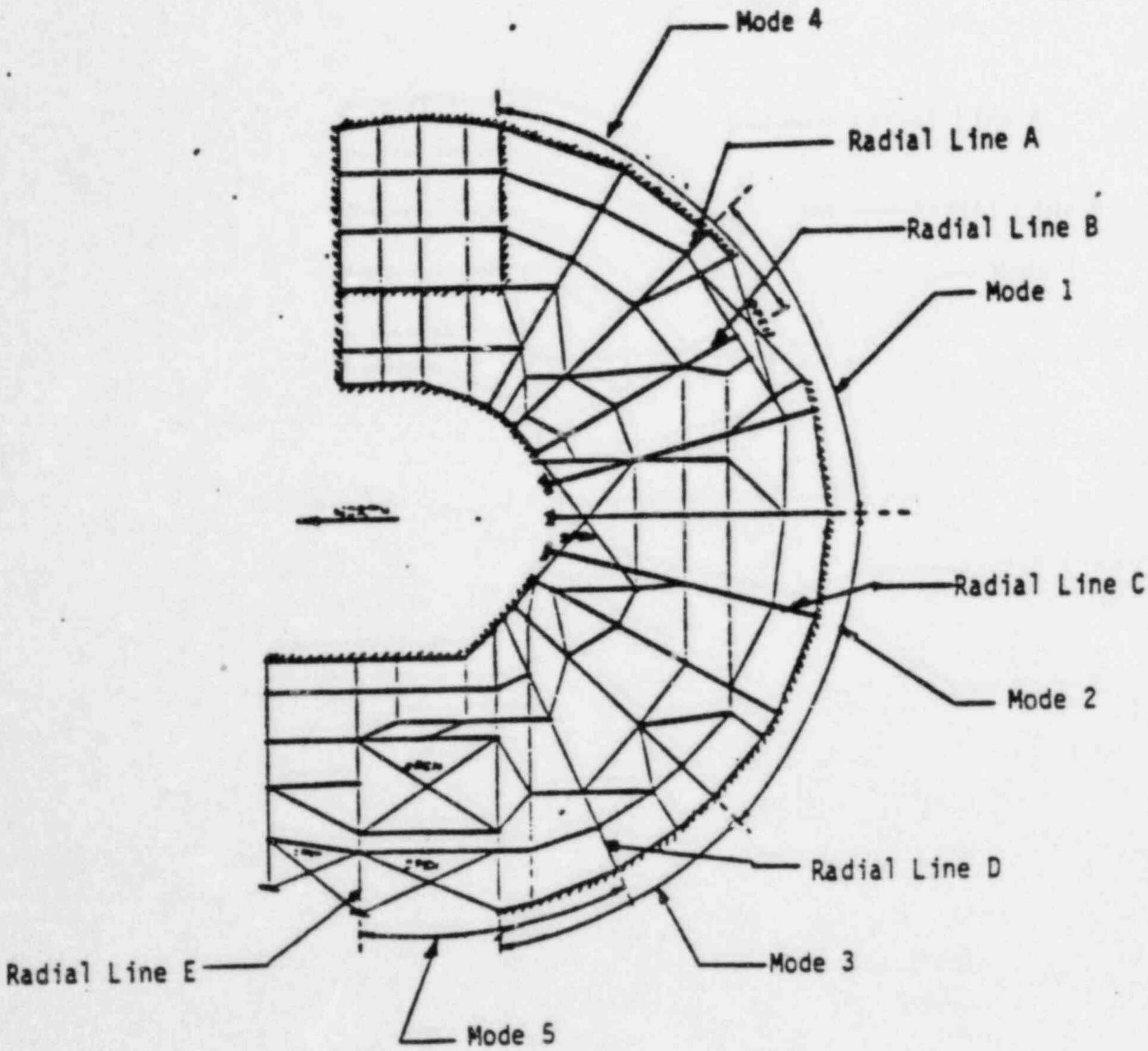
Plots of the five mode shapes along selected radial lines of the floor slab (see Figure 1) are given in Figures 3a, b, and c. Based on these mode shapes, regions 1 through 5 were identified on the floor slab.

TABLE 1

Floor Slab Frequencies and Mass Participation

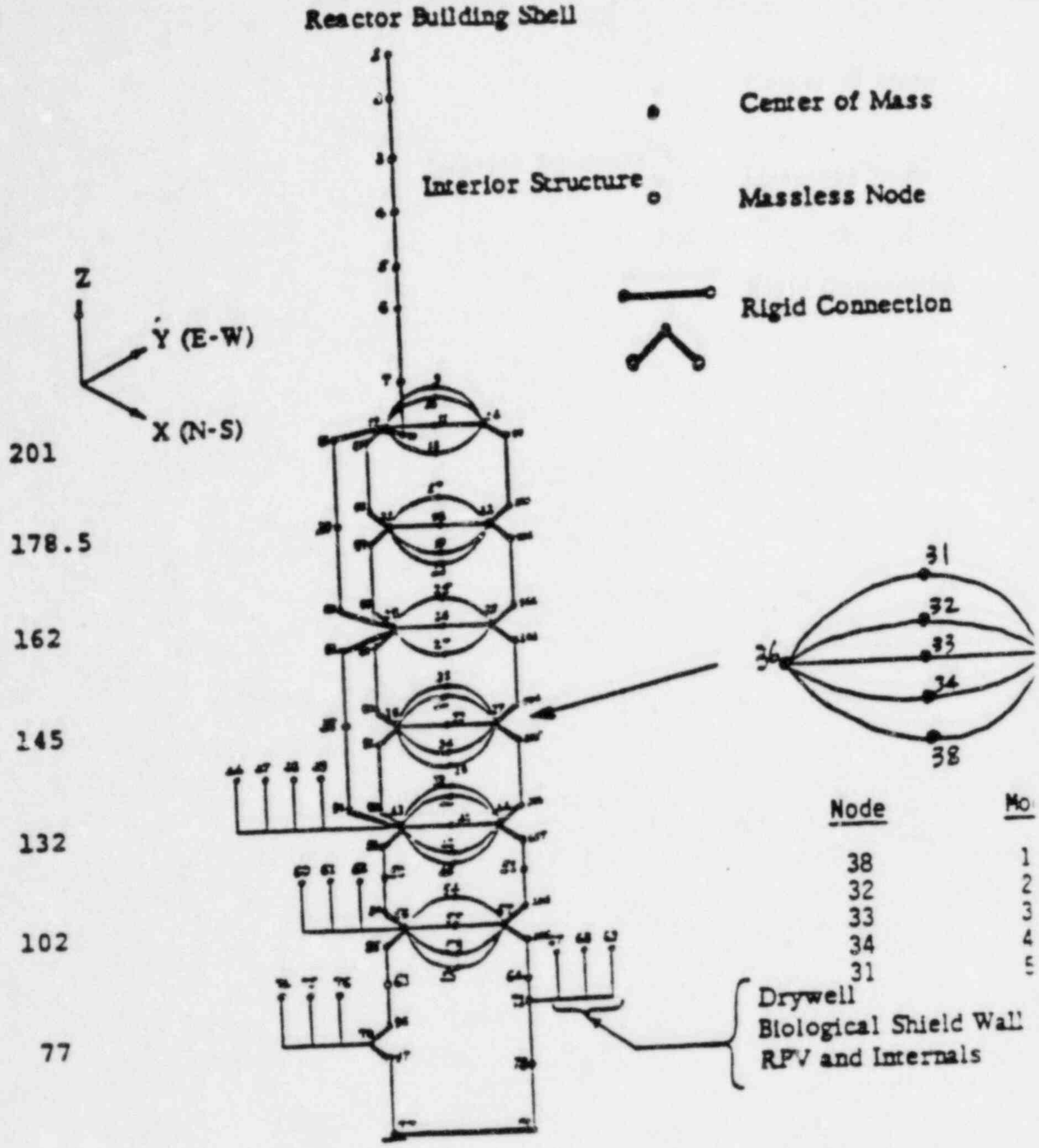
Mode No.	Frequency (Hz.)	Mass Participation Factor	% Total Mass*
1	6.31	6.98	33.9
2	7.65	7.34	37.5
3	8.53	4.01	11.2
4	11.58	2.86	5.7
5	13.27	2.42	4.1
		TOTAL	92.4%

* Total Mass = 143.68 kips $\frac{\text{-sec}^2}{\text{ft.}}$



REACTOR BUILDING VERTICAL FLEXIBILITY ANALYSIS
FINITE ELEMENT PLATE AND BEAM MODEL
FLOOR ELEVATION 145.0 FT.

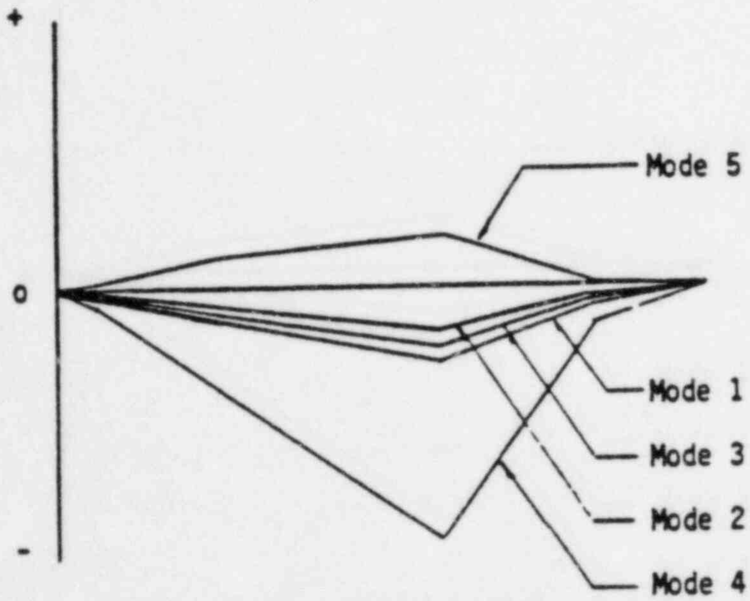
FIGURE 1



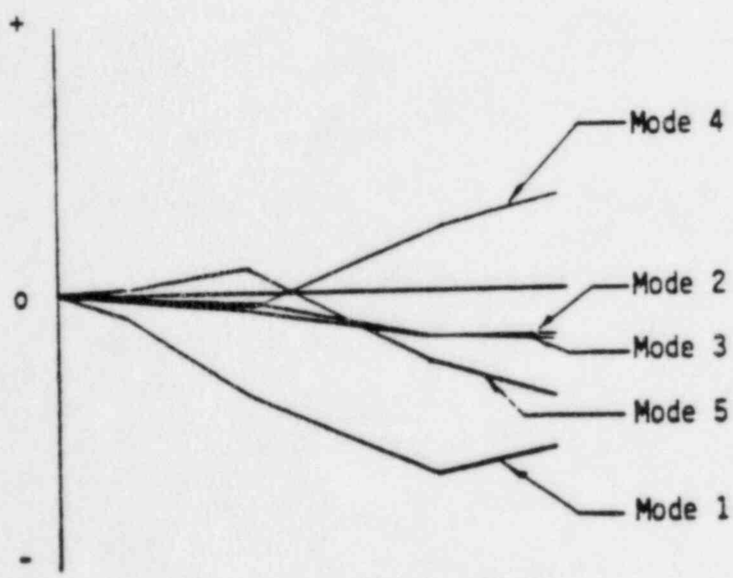
REACTOR BUILDING VERTICAL FLEXIBILITY ANALYSIS
MATHEMATICAL MODEL

FIGURE 2

Figure 3a

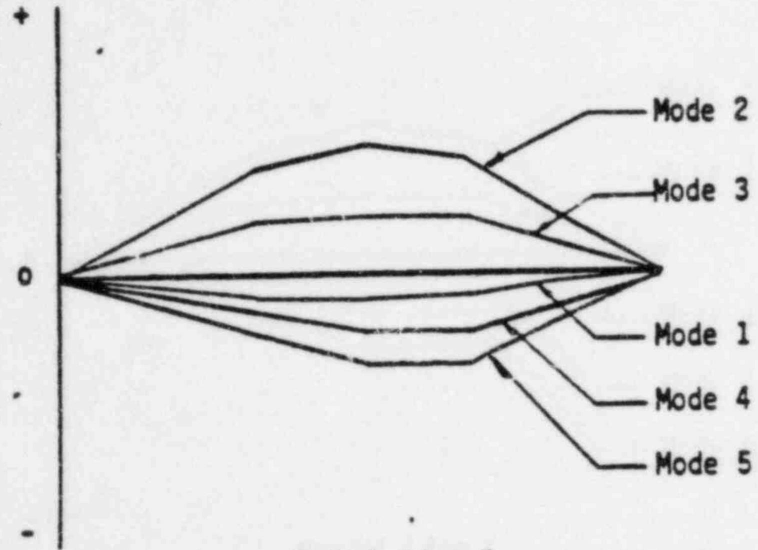


Radial Line A

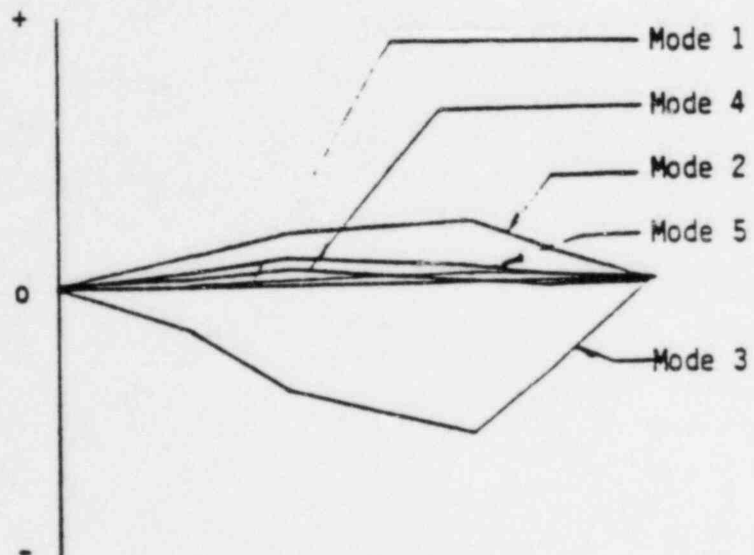


Radial Line B

Figure 3b

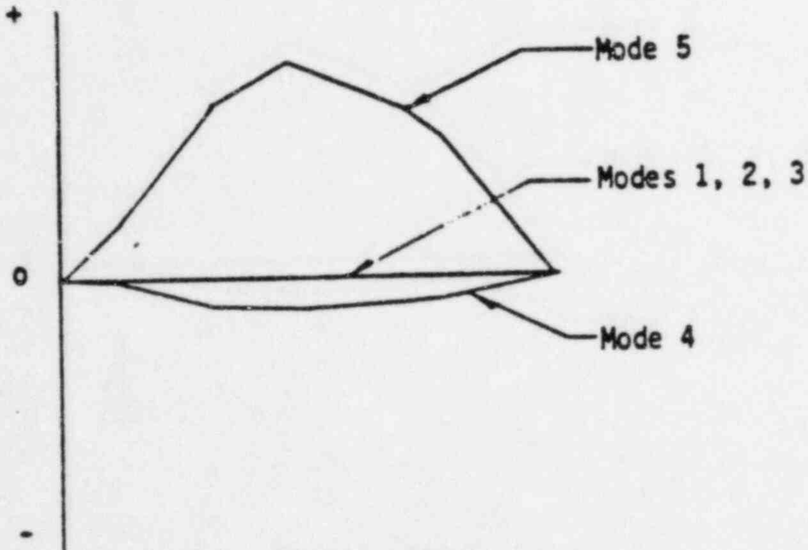


Radial Line C



Radial Line D

Figure 3c



Radial Line E

Response to NRC Audit

Meeting Date: January 10, 1984

Question No.: B-5

Revised Response
Revision 1
July 10, 1984

QUESTION: Provide example calculation for combination of N-S, E-W, and vertical responses.

RESPONSE: Example calculation was provided in the original response to this question.

ADDITIONAL INFORMATION REQUESTED: Provide summary tables showing the contributions to the in-plane response due to out-of-plane excitation for 3 orthogonal directions. Two tables to be provided for both N-S and E-W responses.

RESPONSE: Tables 1 and 2 summarize the N-S and E-W response due to N-S, E-W, and vertical base motions for Reactor Building Unit 1, SSE case. Tables 3 and 4 provide similar information for the OBE case. Individual contributions and the resultant response maxima using the SRSS procedure are listed for selected elements in the Reactor Building mathematical model. As included in the original response, the out-of-plane response maxima (shear and moment) were found to have no significant contribution to the in-plane response maxima values.

TABLE 1

REACTOR BUILDING
OUT-OF-PLANE RESPONSE
SAFE SHUTDOWN EARTHQUAKE

Revised Response
January 10/B-5

Element Number	Variable	N-S Response: N-S Base Motion (A)	E-W Base Motion (B)	Vertical Base Motion (C)	SRSS (D)	Rati (D)/(
1	Shear	9.139×10^{2}	1.746×10^{1}	2.053×10^{2}	9.368×10^{2}	1.C
	Moment	8.988×10^{3}	1.663×10^{2}	2.282×10^{3}	9.275×10^{3}	1.C
7	Shear	1.405×10^{4}	2.358×10^{2}	1.324×10^{3}	1.411×10^{4}	1.C
	Moment	9.796×10^{5}	1.646×10^{4}	1.856×10^{5}	9.972×10^{5}	1.C
11	Shear	2.180×10^{4}	3.490×10^{2}	1.714×10^{3}	2.187×10^{4}	1.C
	Moment	3.182×10^{5}	1.027×10^{5}	5.160×10^{4}	3.383×10^{5}	1.C
15	Shear	2.558×10^{4}	4.188×10^{2}	1.103×10^{3}	2.561×10^{4}	1.C
	Moment	2.653×10^{6}	4.347×10^{4}	1.070×10^{5}	2.656×10^{6}	1.C
19	Shear	4.502×10^{4}	2.635×10^{3}	2.949×10^{3}	4.519×10^{4}	1.C
	Moment	4.933×10^{6}	2.904×10^{5}	2.520×10^{5}	4.948×10^{6}	1.C
21	Shear	5.699×10^{4}	2.192×10^{3}	4.310×10^{3}	5.719×10^{4}	1.C
	Moment	6.775×10^{6}	2.881×10^{5}	3.830×10^{5}	6.792×10^{6}	1.C
33	Shear	1.523×10^{3}	5.135×10^{1}	5.328×10^{2}	1.614×10^{3}	1.(
	Moment	2.331×10^{4}	4.271×10^{3}	4.800×10^{3}	2.418×10^{4}	1.(
35	Shear	3.457×10^{3}	1.018×10^{2}	1.093×10^{3}	3.627×10^{3}	1.(
	Moment	8.488×10^{4}	1.374×10^{4}	1.830×10^{4}	8.791×10^{4}	1.(
37	Shear	9.890×10^{3}	2.031×10^{2}	1.022×10^{3}	9.945×10^{3}	1.(
	Moment	3.518×10^{5}	1.983×10^{4}	2.270×10^{4}	3.531×10^{5}	1.(
39	Shear	3.156×10^{4}	4.933×10^{2}	2.417×10^{3}	3.166×10^{4}	1.(
	Moment	1.181×10^{6}	1.984×10^{4}	7.040×10^{4}	1.183×10^{6}	1.(
42	Shear	1.280×10^{4}	1.790×10^{3}	1.153×10^{3}	1.298×10^{4}	1.(
	Moment	9.634×10^{5}	3.389×10^{4}	5.300×10^{4}	9.655×10^{5}	1.(
44	Shear	1.515×10^{4}	2.805×10^{3}	1.420×10^{3}	1.547×10^{4}	1.(
	Moment	1.471×10^{6}	5.650×10^{4}	1.020×10^{5}	1.476×10^{6}	1.(

Note: 1. Units: Kip, Ft.

TABLE 2

REACTOR BUILDING
OUT-OF-PLANE RESPONSE
SAFE SHUTDOWN EARTHQUAKE

Revised Response
January 10/B-5

Element Number	Variable	E-W Response: E-W Base Motion (A)	E-W Response: N-S Base Motion (B)	E-W Response: Vertical Base Motion (C)	SRSS (D)	Rati (D)/(
1	Shear	8.829×10^2	1.164×10^1	8.628×10^1	8.872×10^2	1.(
	Moment	8.264×10^3	1.186×10^2	8.103×10^2	8.304×10^2	1.(
7	Shear	1.323×10^4	2.203×10^2	8.034×10^2	1.326×10^4	1.(
	Moment	8.504×10^5	1.267×10^4	6.400×10^4	8.529×10^5	1.(
11	Shear	1.698×10^4	4.092×10^2	3.796×10^2	1.699×10^4	1.(
	Moment	1.583×10^6	2.653×10^5	6.930×10^4	1.607×10^6	1.(
15	Shear	4.918×10^4	5.880×10^2	9.377×10^2	4.919×10^4	1.(
	Moment	8.257×10^5	1.138×10^4	1.230×10^5	8.349×10^5	1.(
19	Shear	6.499×10^4	6.400×10^2	1.204×10^3	6.500×10^4	1.(
	Moment	3.078×10^6	4.853×10^5	1.660×10^5	3.120×10^6	1.(
21	Shear	7.055×10^4	6.283×10^2	1.440×10^3	7.057×10^4	1.(
	Moment	5.337×10^6	1.837×10^5	1.990×10^5	5.344×10^6	1.(
33	Shear	1.601×10^3	5.216×10^1	3.740×10^2	1.645×10^3	1.(
	Moment	1.593×10^4	2.022×10^3	3.370×10^3	1.641×10^4	1.(
35	Shear	3.491×10^3	8.271×10^1	7.104×10^2	3.564×10^3	1.(
	Moment	6.442×10^4	4.509×10^3	1.240×10^4	6.576×10^4	1.(
37	Shear	5.981×10^3	1.188×10^2	7.497×10^2	6.029×10^3	1.(
	Moment	1.025×10^5	9.354×10^3	2.190×10^4	1.052×10^5	1.(
39	Shear	1.482×10^4	1.707×10^2	4.034×10^2	1.483×10^4	1.(
	Moment	3.107×10^5	1.200×10^4	3.630×10^4	3.130×10^5	1.(
42	Shear	8.162×10^3	1.084×10^2	1.621×10^2	8.164×10^3	1.(
	Moment	7.449×10^4	6.000×10^3	6.160×10^3	7.498×10^4	1.(
44	Shear	1.055×10^4	1.284×10^2	2.103×10^2	1.055×10^4	1.(
	Moment	2.138×10^5	6.323×10^3	1.350×10^4	2.143×10^5	1.(

Note: 1. Units: Kip, Ft.

TABLE 3

REACTOR BUILDING
OUT-OF-PLANE RESPONSE
OPERATING BASIS EARTHQUAKE

Revised Response
January 10/B-5

Element Number	Variable	N-S Response			SRSS (D)	Ratio (D)/(A)
		N-S Base Motion (A)	E-W Base Motion (B)	Vertical Base Motion (C)		
1	Shear Moment	8.676×10^2 8.247×10^3	2.339×10^1 2.239×10^2	1.283×10^2 1.426×10^3	8.773×10^2 8.372×10^3	1.01 1.02
7	Shear Moment	1.175×10^4 8.243×10^5	3.322×10^2 2.308×10^4	8.275×10^2 1.160×10^5	1.178×10^4 8.327×10^5	1.00 1.01
11	Shear Moment	1.515×10^4 2.549×10^5	4.907×10^2 6.064×10^4	1.071×10^3 3.225×10^4	1.520×10^4 2.640×10^5	1.00 1.04
15	Shear Moment	1.306×10^4 1.830×10^6	5.542×10^2 5.553×10^4	6.894×10^2 6.688×10^4	1.309×10^4 1.832×10^6	1.00 1.00
19	Shear Moment	1.899×10^4 2.873×10^6	1.561×10^3 1.798×10^5	1.843×10^3 1.575×10^5	1.914×10^4 2.883×10^6	1.01 1.00
21	Shear Moment	2.406×10^4 3.665×10^6	1.578×10^3 2.059×10^5	2.694×10^3 2.394×10^5	2.426×10^4 3.679×10^6	1.01 1.00
33	Shear Moment	6.421×10^2 7.924×10^3	4.807×10^1 1.967×10^3	3.330×10^2 3.000×10^3	7.249×10^2 8.698×10^3	1.13 1.10
35	Shear Moment	1.468×10^3 3.015×10^4	1.040×10^2 6.522×10^3	6.831×10^2 1.144×10^4	1.622×10^3 3.290×10^4	1.10 1.09
37	Shear Moment	4.282×10^3 2.260×10^5	2.146×10^2 1.090×10^4	6.388×10^2 1.419×10^4	4.335×10^3 2.267×10^5	1.01 1.00
39	Shear Moment	1.455×10^4 7.580×10^5	6.914×10^2 2.396×10^4	1.511×10^3 4.400×10^4	1.464×10^4 7.597×10^5	1.01 1.00
42	Shear Moment	5.381×10^3 5.879×10^5	8.901×10^2 2.707×10^4	7.206×10^2 3.313×10^4	5.502×10^3 5.895×10^5	1.02 1.00
44	Shear Moment	6.382×10^3 7.841×10^5	1.315×10^3 3.660×10^4	8.875×10^2 6.375×10^4	6.576×10^3 7.875×10^5	1.03 1.00

Notes: 1. Units: Kip, Ft.
2. This is considered insignificant because the shear and moment for this beam are very small.

TABLE 4

REACTOR BUILDING
OUT-OF-PLANE RESPONSE
OPERATING BASIS EARTHQUAKE

Revised Response
January 10/B-5

Element Number	Variable	E-W Response: E-W Base Motion (A)	E-W Response: N-S Base Motion (B)	E-W Response: Vertical Base Motion (C)	SRSS (D)	Rat (D)/
1	Shear	6.049×10^2	1.604×10^1	5.393×10^1	6.075×10^2	1.
	Moment	5.660×10^3	1.571×10^2	5.064×10^2	5.685×10^3	1.
7	Shear	8.723×10^3	2.178×10^2	5.021×10^2	8.740×10^3	1.
	Moment	5.791×10^5	1.391×10^4	4.000×10^4	5.806×10^5	1.
11	Shear	9.271×10^3	6.722×10^2	2.373×10^2	9.298×10^3	1.
	Moment	9.943×10^5	1.798×10^5	4.331×10^4	1.011×10^6	1.
15	Shear	2.437×10^4	6.431×10^2	5.861×10^2	2.439×10^4	1
	Moment	5.180×10^5	1.235×10^4	7.688×10^4	5.238×10^5	1
19	Shear	3.187×10^4	7.589×10^2	7.525×10^2	3.189×10^4	1.
	Moment	1.517×10^6	3.060×10^5	1.038×10^5	1.551×10^6	1
21	Shear	3.431×10^4	8.172×10^2	9.000×10^2	3.433×10^4	1
	Moment	2.628×10^6	1.406×10^5	1.244×10^5	2.635×10^6	1
33	Shear	7.598×10^2	3.297×10^1	2.338×10^2	7.956×10^2	1
	Moment	7.889×10^3	9.159×10^2	2.106×10^3	8.217×10^3	1
35	Shear	1.679×10^3	5.966×10^1	4.440×10^2	1.738×10^3	1
	Moment	3.159×10^4	1.601×10^3	7.750×10^3	3.257×10^4	1
37	Shear	2.920×10^3	8.827×10^1	4.686×10^2	2.959×10^3	1
	Moment	6.786×10^4	1.183×10^4	1.369×10^4	7.023×10^4	1
39	Shear	7.333×10^3	1.910×10^2	2.521×10^2	7.340×10^3	1
	Moment	2.000×10^5	8.248×10^3	2.269×10^4	2.015×10^5	1
42	Shear	4.065×10^3	1.018×10^2	1.013×10^2	4.068×10^3	1
	Moment	3.724×10^4	3.039×10^3	3.850×10^3	3.756×10^4	1
44	Shear	5.111×10^3	1.129×10^2	1.314×10^2	5.117×10^3	1
	Moment	1.045×10^5	5.758×10^3	8.438×10^3	1.050×10^5	1

Note: 1. Units: Kip, Ft.

QUESTION: Provide calculation showing drywell stick model development (provide for one section only).

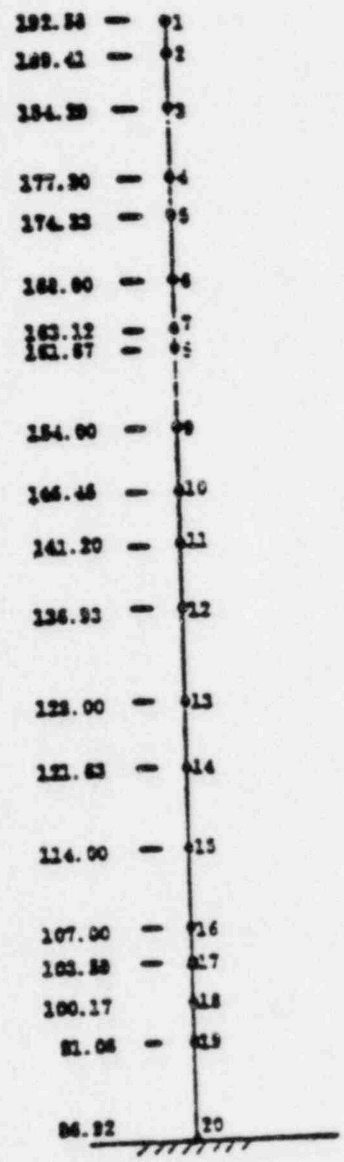
RESPONSE: Calculation is provided in the original response to this question.

ADDITIONAL INFORMATION REQUESTED: Provide clarification of the rotation value used in determining the equivalent beam properties.

RESPONSE: The rotational value ($\theta = 2.35 \times 10^{-7}$) determined in the sample calculation provided as Attachment I to this revised response is used to determine the property between elevations 86.94 and 91.06 of the equivalent beam model (Figure 1). This rotation is estimated based on the resultant displacements due to unit load applied at the shear lug location of the more detailed axisymmetric shell model of the drywell (Figure 2).

The best estimate of this rotation was arrived at based on examining the displacement pattern (Figure 3) of the shell model between node 82 (node 19 of beam model) and neighboring nodes.

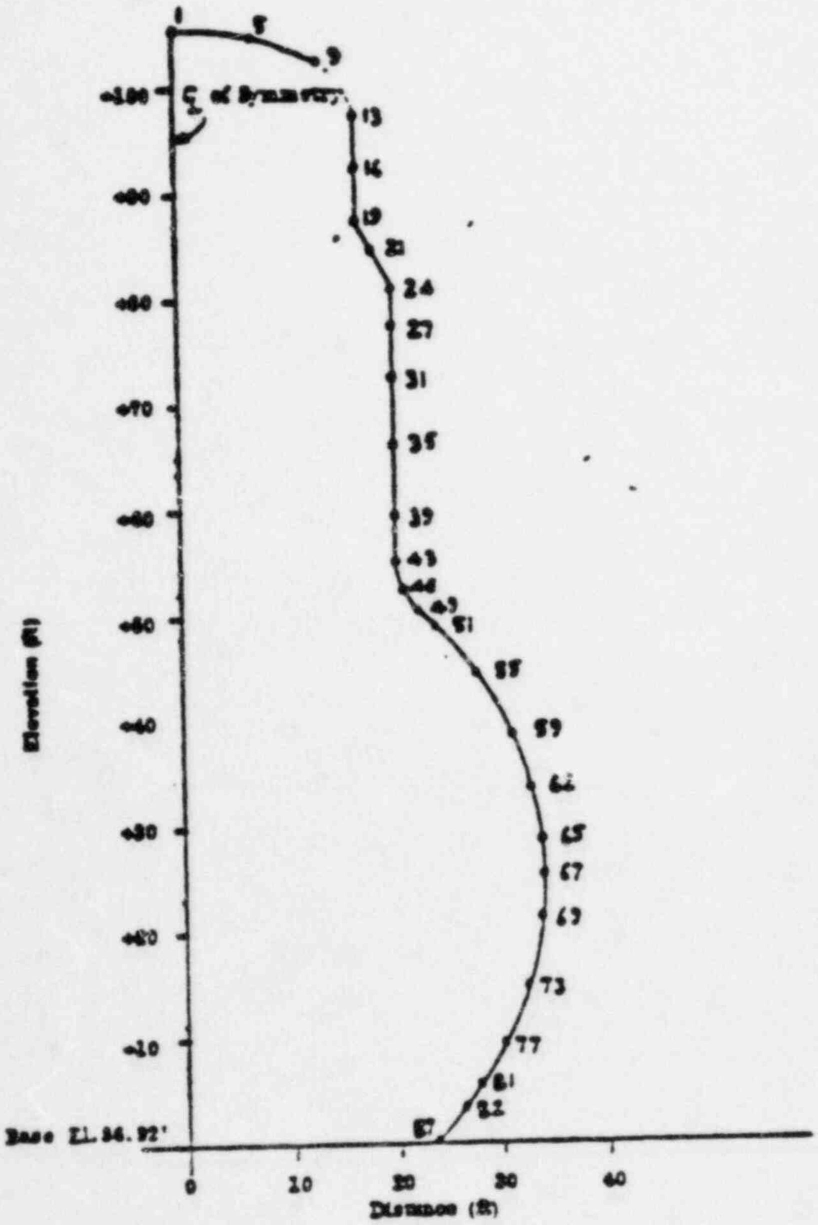
The rotational value of 2.39×10^{-7} is more representative of the rigid body rotation of the drywell stick between its base and approximately 50' above the base, as can be seen from Figure 3. However, a slightly lower value of rotation at node 82 is obtained based on calculations of rotation at this node and nodes 81 and 83 of the detailed shell model. The best estimate of this rotation is obtained as an average of these rotational values. This sample calculation is included as Attachment I to this revised response. Furthermore, the difference between the two values of rotation is less than 2% which is considered insignificant.



DRYWELL EQUIVALENT BEAM MODEL

TOTAL NODES = 20

FIGURE 1

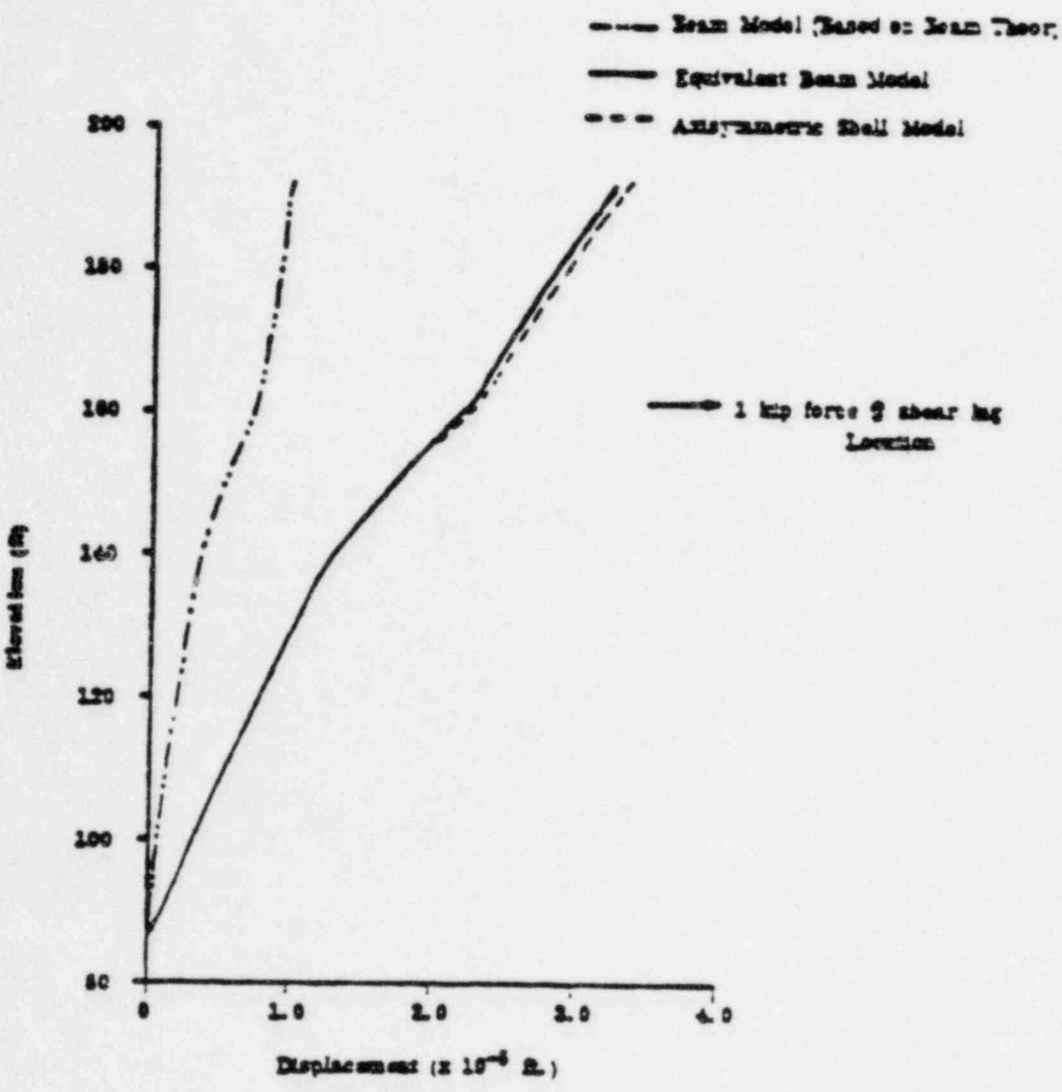


DRYWELL AXISYMMETRIC MODEL

TOTAL NODES = 87

FIGURE 2

STATIC DISPLACEMENT COMPARISON



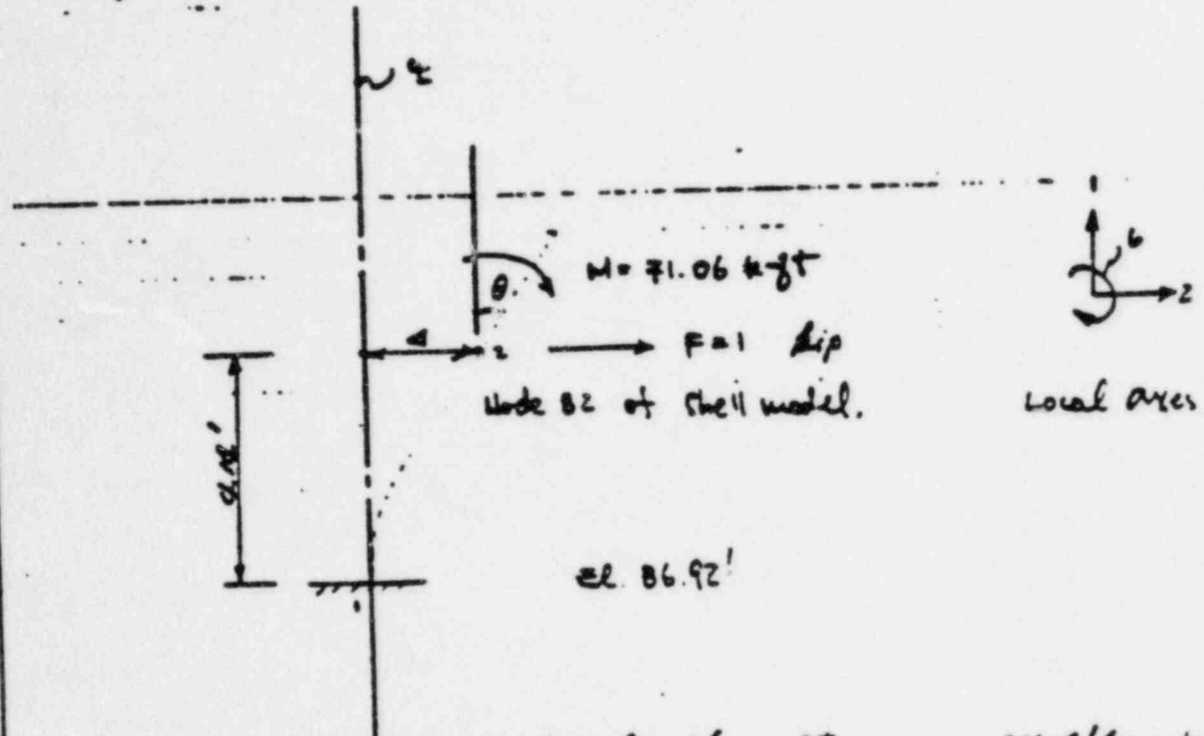
DRYWELL (FREE STANDING)

FIGURE 3

ATTACHMENT I

Equivalent Beam Model

Change node 19 coordinate to (0, 0, 91.06).



Δ @ node 82 (90°) = 0.96009 × 10⁻⁶ ft (4.16' from drywell)

Δ @ node 50 (90°) = 0.11921 × 10⁻⁴ ft (50.0' from drywell)

$$\therefore\ \theta = \frac{0.11921\times10^{-4} - 0.96009\times10^{-6}}{50-4.16} = 2.390\times10^{-7}\ *$$ ✓

Ref: J.S. Przemieniecki, "Theory of Matrix Structural Analysis", PP

$$S_8 = \frac{12EI}{\ell^3(1+\phi)}\Delta - \frac{6EI}{\ell^2(1+\phi)}\theta$$

$$S_{12} = -\frac{6EI}{\ell^2(1+\phi)}\Delta + \frac{(4+\phi)EI}{\ell(1+\phi)}\theta \qquad (1)$$

* If we use 81 & 82, $\theta = \frac{0.11345\times10^{-5} - 0.96009\times10^{-6}}{5.03-4.16} = 2.387\times10^{-7}$ ✓

82 & 83 $\theta = 2.209\times10^{-7}$

ITEM: Drywell Beam Model	CLIENT: PSEG
	PROJECT: Hope Creek
	JOB NO. 180.0012
BY: MII DATE 2/10/77 CKD: GCH DATE 3/1/77	ITEM NO. SHT 3

Use $\Delta = 0.960 \times 10^{-6}$ $\theta = 2.35 \times 10^{-7}$

$S_\delta = 1.0$ $S_\theta = 71.0$

$E = 9.0 \times 10^{6}$ $\nu = 0.3$

Then,

$I = 321.8$

$\phi = 102.1$

Now, $\phi = \dfrac{12EI}{l^2 G A_s}$ $\therefore$ $A_s = \dfrac{24(1+\nu)I}{l^2 \phi}$ ✓

$\therefore$ $A_s = 5.74$

$\therefore$ 1st try, $I = 322 \text{ ft}^4$, $A_s = 5.74 \text{ ft}^2$

Check

$\phi = \dfrac{12EI}{l^2 G A_s} = \dfrac{24(1+\nu)I}{l^2 A_s} = 102.12$ ✓

$\therefore$ $0.528(E\Delta) - 1.053(E\theta) = 1$

$-1.053(E\Delta) + 80.041(E\theta) = 71$

$\therefore E\Delta = \dfrac{80.041 + (1.053)(71)}{(0.528)(80.041) - (1.053)^2} = 3.839$

$E\theta = \dfrac{(0.528)(71) + 1.053}{(0.528)(80.041) - (1.053)^2} = 0.939$

$\therefore$ $\Delta = 9.557 \times 10^{-7}$, $\theta = 2.349 \times 10^{-7}$ ✓

ITEM: Drywell Beam Model	CLIENT: DSE&G PROJECT: Hope Creek
	JOB NO. 180.0012
BY: MH DATE: 6/10/77 CHD: GCH DATE: 6/11/77	ITEM NO. SHT 8

Meeting Date: January 11, 1984

Question No.: A.7

Revised Response
Revision 1
6/30/84

QUESTION : Provide a simplified calculation for overturning moment of reactor building foundation mat.

RESPONSE : Appendix 3H of FSAR presents the calculated factors of safety against floatation, sliding, and overturning. The factor of safety against overturning was computed using the energy balance method. This response examines the reactor building overturning stability using the conventional method.

The controlling load combination for the overturning stability check is D + H + E_s as discussed in Appendix 3H of FSAR. Stability against overturning will be ensured by the dead weight of the structures and the passive soil pressure associated with the embedded portion of the structures. The buoyant force, which tends to increase the overturning potential of the structure, has been taken into account. Overturning for the reactor building, due to North-South earthquake, has been determined to be the most critical case.

Two factors of safety have been examined in this response which are as follows:

1. Factor of safety against global overturning.
2. Factor of safety against toe contact pressure failure.

The factor of safety against global overturning failure is defined as the ratio of the resisting moment under earthquake conditions (M_r) to the overturning moments (M_o) on the foundation:

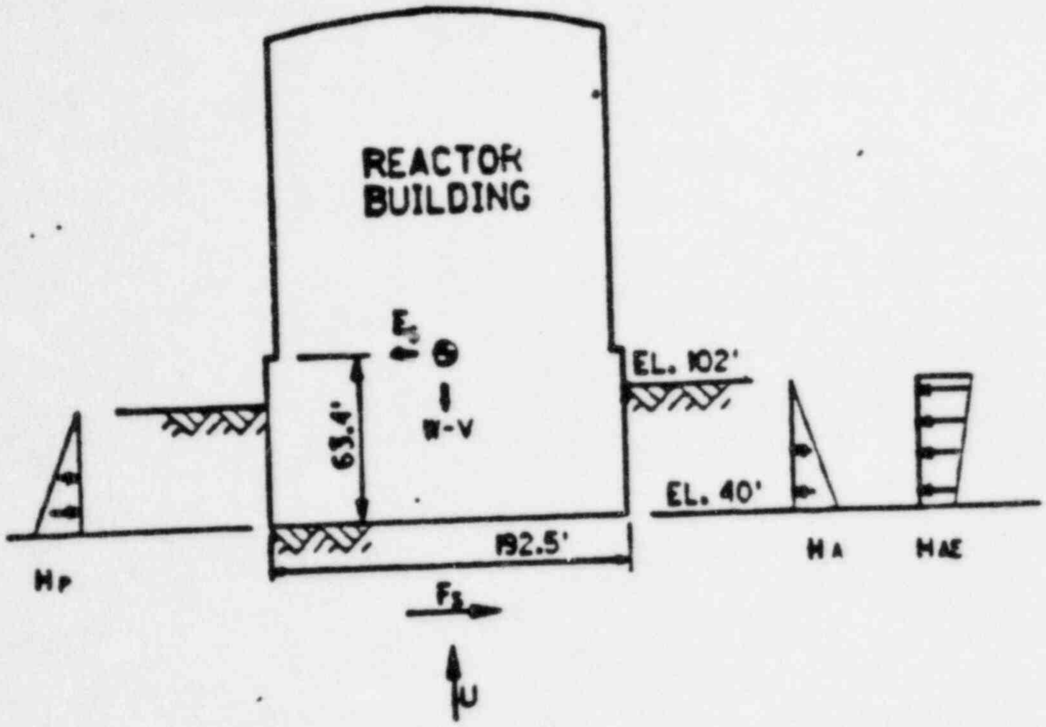
$$F.S. = M_r/M_o$$

As indicated in Figure A.7-1, the computed factor of safety against overturning is 1.61 which is greater than the required factor of safety of 1.1.

The factor of safety against toe failure is defined as the ratio of the allowable maximum soil bearing pressure to the calculated dynamic toe pressure.

$$F.S. = P\ allow/P\ actual$$

The computed factor of safety as indicated in the FSAR Appendix 3H (Section IV) is 3.60 which is greater than the required factor of safety of 1.1.



FORCE TYPE	RESISTING				DRIVING			
	NOTATION	ARM	FORCE	MOMENT	NOTATION	ARM	FORCE	MOMENT
DYNAMIC SOIL	—	NEGLECTED			H_{AE}	31.0	18,190	563,900
STATIC SOIL	H_P	20.7	210,000	4,347,000	H_A	20.7	7,675	158,900
RESULTANT VERTICAL LOAD	—	96.3	65,270	6,286,000	—	—	—	—
DYNAMIC INERTIA	—	—	—	—	E_S	63.4	92,980	5,895,000
FRICTION	F_S	0	52,870	0	—	—	—	—
TOTAL	—			10,633,000	—			6,618,000

$$(F.S.)_{OTM} = \frac{\text{RESISTING MOMENT}}{\text{DRIVING MOMENT}} = \frac{10,633,000}{6,618,000} = 1.61 > 1.1$$

NOTE:

W = DEAD WEIGHT = 464,900^K

V = VERTICAL DYNAMIC INERTIA = 37,130^K

U = UPLIFT DUE TO GROUNDWATER AND EXCESS PORE WATER PRESSURE = 36[illegible],500^K

ALL UNITS ARE GIVEN IN KIPS AND FEET

FIGURE - A.7-1

Revised Response

Revision 1
6/30/84

Response to NRC Audit

Meeting Date: January 11, 1984

Question No.: A.8

QUESTION:

Are BSAP element size limitations satisfied for the foundation mat model and the drywell shield wall model.

RESPONSE:

The BSAP program user's manual does not specify element size limitations. Element aspect ratio is a design parameter which varies with model configuration and analysis accuracy. Better accuracy is achieved with an aspect ratio close to 1.

The foundation mat and the drywell shield wall were originally analyzed using the BSAP computer program. During evaluation of Unit 2 cancellation, the drywell shield wall has been reanalyzed using the ASHSD computer program (ref.: FSAR, Appendix 3A). Brick elements were used for both the foundation mat and drywell shield wall models. For the foundation mat analysis, the element aspect ratios were approximately 5 in high stress areas. For the drywell shield wall analysis, the element aspect ratios were less than 2. Generic studies e.g., Desai, C.S., and Abel, J.F., Introduction to the Finite Element Method Van Nostrand Reinhold Co., N.Y., 1972, have shown that an aspect ratio of less than 5 or 8 gives errors of less than 10 percent or 15 percent, respectively, as compared to the bench mark solution. Sufficient design margin exists to justify this degree of error.

Response to NRC Audit

Meeting Date: January 11, 1984

Question No.: A-12

Revised Response
Revision 1
6/30/84

QUESTION: Justify the 12 Hz. cut-off frequency for SSI analysis.

RESPONSE: Two independent studies have been performed to justify the 12 Hz. cut-off frequency: a design base evaluation performed by Impell and a confirmatory evaluation by Bechtel. These studies are described separately below.

A. DESIGN BASE ANALYSIS

The selection of a cut-off frequency value was based on two primary considerations:

1. For the particular Hope Creek site, the evaluation of the highest shear wave frequency that can realistically be transmitted through the soil medium.

2. The contribution of the high frequency components of the input free-field (control) motion on the resultant structural response.

DECONVOLUTION ANALYSIS

Two cut-off frequency values were selected for consideration and study: 12 and 20 Hz. An operating basis earthquake was selected for the study, due to its lower peak acceleratic level. Because of the nonlinear characteristics of the soil, the lower excitation level will result in stiffer soil properties than for the SSE level excitation, with the soil thus capable of transmitting higher frequency waves. This case will then be more critical for establishinc a cut-off frequency value than the SSE.

A soil column, representing the Hope Creek free-field soil properties, was first constructed. The mesh refinement was selected such that a wave frequency of 20 Hz. could be transmitted without loss of numerical accuracy. A schematic representation of the soil column model is presented in Figure 1.

The free-field soil column is composed of a series of two-dimensional plane strain elements of unit width modeling the soil properties. The dimensions of the soil column extend between elevations 102.0 feet, corresponding to the elevation at finished grade for the Hope Creek site, down to elevation -300.0 feet, a depth found to be sufficiently deep to include all significant soil-structure interaction effects.

Using the above free-field soil column, a deconvolution analysis of the normalized OBE Regulatory Guide 1.60 synthetic time-history was performed for both 12 and 20 Hz. cut-off frequency values. This normalized Regulatory Guide control motion was input at elevation 40.0 feet, corresponding to the elevation of the bottom of the foundation base mats for the power block area. Deconvoluted time-history response was obtained at the base of the soil column model, corresponding to elevation -300.0 feet.

SOIL-STRUCTURE INTERACTION ANALYSIS

A simplified soil-structure model was developed for the cut-off frequency study. The model consists of a single soil colum, attached to a series of single-degree-of-freedom oscillators representing the Reactor Building structure. A sketch of the model, with the corresponding soil and structural properties, can be seen in Figures 2 and 3.

As in the case of the deconvolution analysis, the soil properties were modeled by a series of two-dimensional plane strain elements of unit width. The soil elements extend from elevation -300.0 feet to elevation 40.0 feet, corresponding to the elevation at the bottom of the foundation base mat. One additional plane strain element was placed between elevations 40.0 feet and 54.0 feet, to simulate the base mat properties. The Reactor Building dynamic proerties for the N-S modes with frequencies up to 20 Hz. were duplicated by a series of single-degree-of-freedom oscillators. The mass properties of these oscillators are drawn from the modal effective mass calculation of the detailed model.

A soil-structure interaction analysis was performed for both a 12 and 20 Hz. cut-off frequency value. The input motions obtained from the deconvolution analyses, were input at elevation -300.0 feet of the simplified interaction model. Using a system direct integration technique, a time-history analysis of the soil-structure system was performed, with time-histories of acceleration being obtained at the base mat level. An evaluation of the influence of the cut-off frequency was obtained by comparison of the derived base mat response spectra for each of the cut-off frequencies.

As demonstrated by Figure 4, the base mat response for the two cut-off frequencies are essentially identified for the frequency range below 12 Hz. For the frequency range of 12 to 20 Hz., however, the response at the base mat does diverge somewhat between the two cut-off frequencies. The 20 Hz. cut-off response exhibits a number of minor peaks, as a result of high frequency components of the bedrock motion. The effect of these minor peaks on structural response is insignicant as verified by the seismic structural analysis described in the following section. The 12 Hz. cut-off analysis, on the otherhand, exhibits a non-amplified response beyond 12 Hz., resulting in a constant spectral acceleration.

In order to verify the adequacy of the simplified model, a comparison with a detailed interaction model was made. A comparison of Figures 4 and 5 reveals that the motion at the base mat for the detailed and simplified models exhibit very similar trends, both with regard to the spectral peak and overall shape of the curves.

SEISMIC STRUCTURAL ANALYSIS

In order to identify the significance of the difference in basemat motion for the cut-off frequencies on the structural response of the Reactor Building, a seismic structural analysis of the Reactor Building was performed using the detailed three-dimensional model.

Using the basemat motions derived from the simplified interaction analyses for a 12 and 20 Hz. cut-off frequency, a modal time-history analysis of the Reactor Building was performed. Response spectra at selected elevations were computed from the resultant floor excitations, and a comparison of the results derived from the two cut-off frequencies was performed.

Figures 6 through 8 present response maxima for shear, moment and torque in the drywell of the Reactor Building, when subjected to each of the base mat excitations. The drywell was selected for comparison of cut-off frequency effects because it is a portion of the reactor pressure boundary, and the design of this structure is particularly critical. The comparison of results for the drywell is representative of other portions of the structure as well. As can be seen from these plots, the response maxima of the structure are virtually independent of the high frequency acceleration components of the base mat motion. Clearly the shear, moment and torque response values for the structure are essentially identical for the two different cut-off frequencies, indicating a dependance only on the low and mid frequency range of the base mat motions.

Response spectra plots at various elevations of the Reactor Building are presented in Figure 9 through 12. As can be seen from these figures, the spectral accelerations in the low and mid frequency range are essentially identical for the two cut-off frequencies. In the frequency range above 12 Hz., there are minor differences at the lower elevations of the Reactor Building, but almost no variation in the upper elevations. However, these minor differences are considered to have no significance on global structural response. For all elevations, the overall trend of the curves is identical, duplicating peak response values and shape of the spectral curves.

B. CONFIRMATORY ANALYSIS

Bechtel also performed a confirmatory independent analysis to evaluate the effect of cut-off frequency on the soil structure interaction analysis results. The North-South soil-structure model was analyzed for the SSE case. The soil model was discretized to have elements which are capable of transmitting frequencies of at least 18 Hz. Two soil-structure interaction analyses, with cut-off frequencies of 12 Hz and 18 Hz, were performed using computer code FLUSH. As shown in the response spectrum comparison plots (Figures 13 to 15), there is practically no effect in increasing the cut-off frequency from 12 Hz to 18 Hz on the response of the soil structure system.

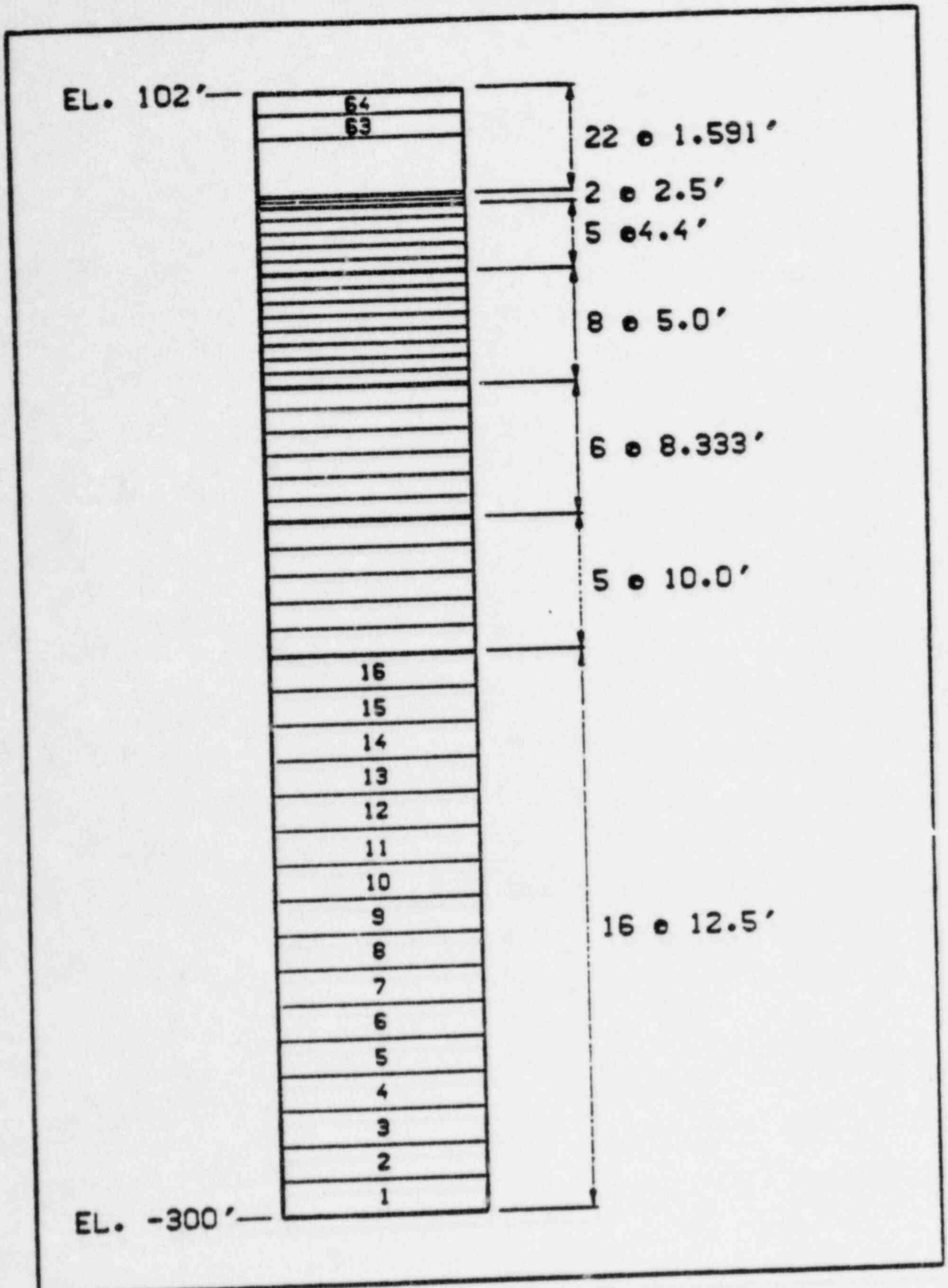
CONCLUSIONS

A study has been performed to evaluate the influence of the cut-off frequency value on the soil-structure interaction analysis subsequent seismic structural analysis for the Hope Creek site. A comparison of response results for a 12 and 20 Hz. cut-off frequency was made for all facets of the analysis.

Comparison of structural response results indicates only minor dependence on the high frequency acceleration components of the input motion, both in the generation of building response maxima and floor response spectra. It would thus be reasonable to assume that either an intermediate cut-off frequency between 12 and 20 Hz. or higher cut-off frequencies of up to 33 Hz. (cut-off for structural response evaluation) would produce only minor deviation from the response results of the 12 Hz. cut-off frequency analysis.

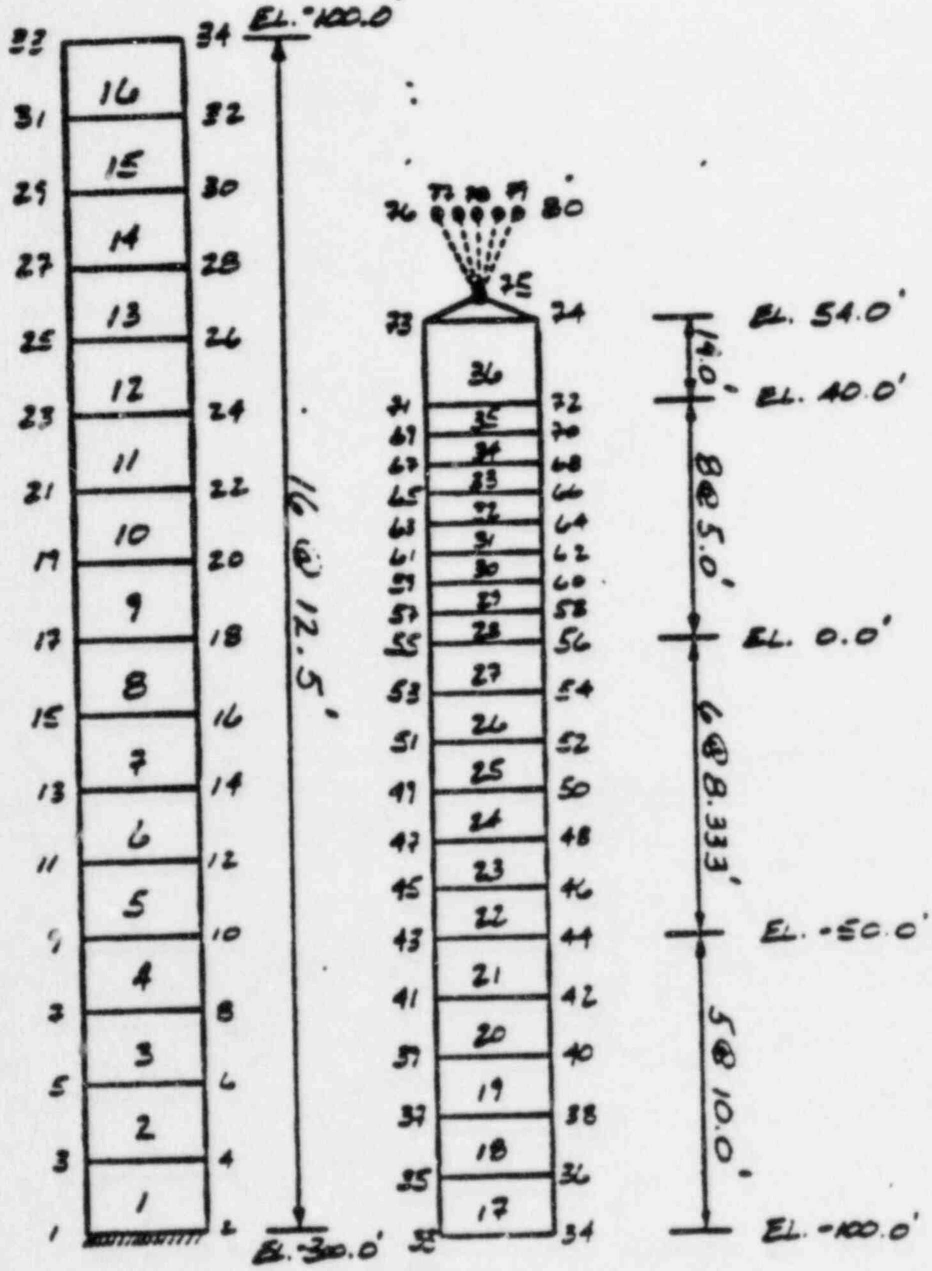
Based on the above considerations, it was found that a cut-off frequency of 12 Hz. for the soil-structure interaction analysis was both physically realistic for the Hope Creek site and, in addition, maintains adequate conservatism with regard to structural response. The use of a 12 Hz. cut-off frequency was thus selected for the Hope Creek analysis.

Furthermore, the adequacy of the 12 Hz. cut-off frequency for SSI analysis has been verified by an independent study performed by Bechtel.



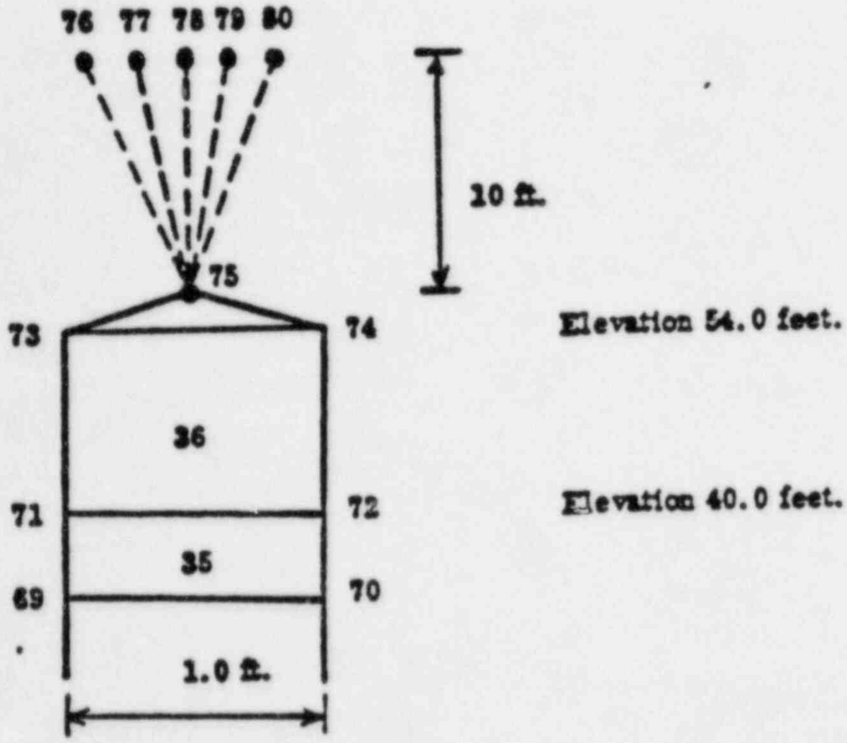
MATHEMATICAL MODEL OF FREE FIELD SOIL COLUMN
FOR DECONVOLUTION ANALYSIS

FIGURE 1



MATHEMATICAL MODEL FOR SOIL-STRUCTURE INTERACTION ANALYSIS

FIGURE 2



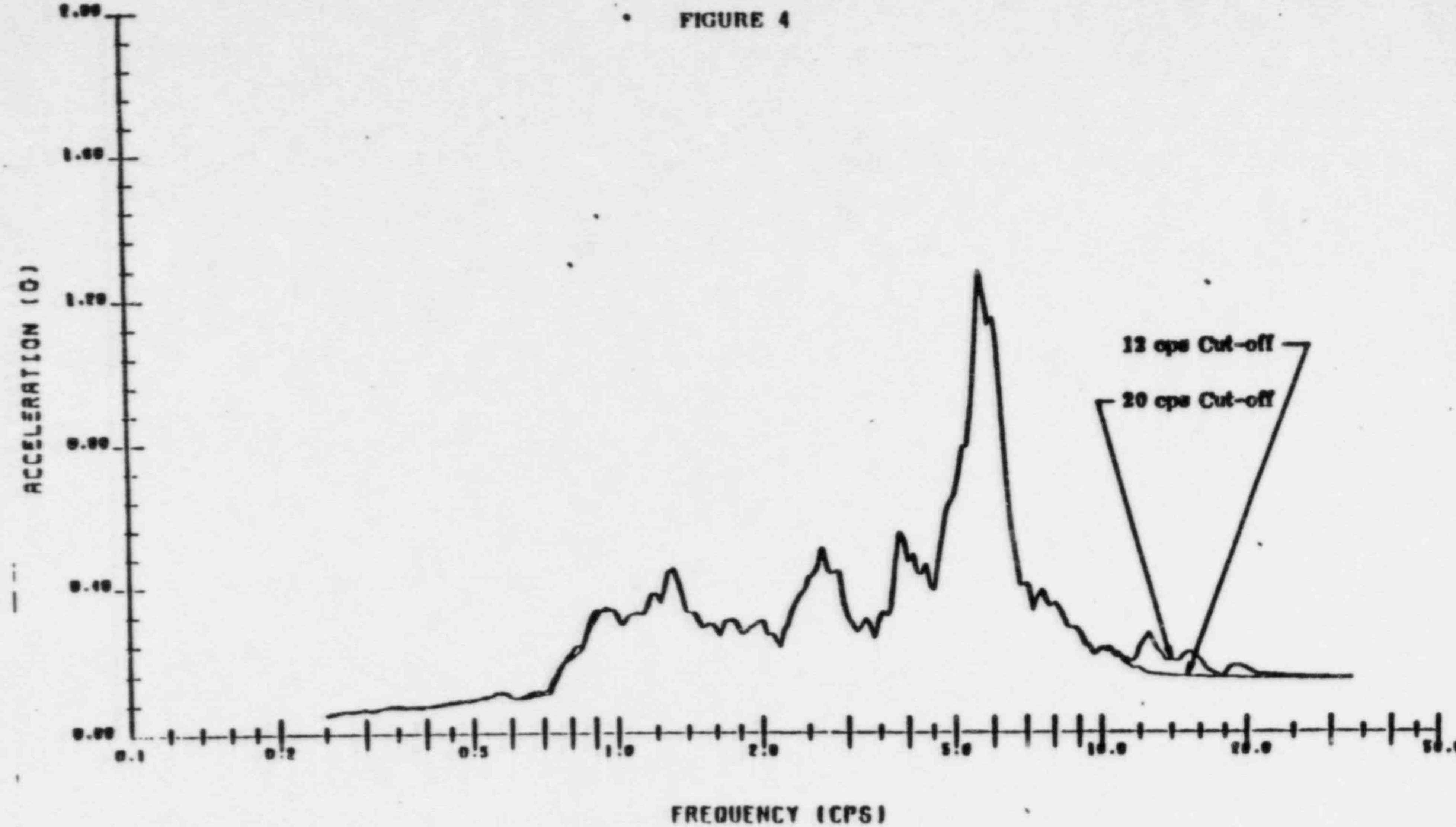
FREQUENCY	MASS PARTICIPATION FACTORS
f_1 = 4.11 Hz.	M_1 = 21.07
f_2 = 9.07 Hz.	M_2 = 10.43
f_3 = 12.08 Hz.	M_3 = 4.75
f_4 = 17.65 Hz.	M_4 = 0.65
f_5 = 19.76 Hz.	M_5 = 2.17

MATHEMATICAL REPRESENTATION OF REACTOR BUILDING
FOR SOIL-STRUCTURE INTERACTION ANALYSIS

FIGURE 3

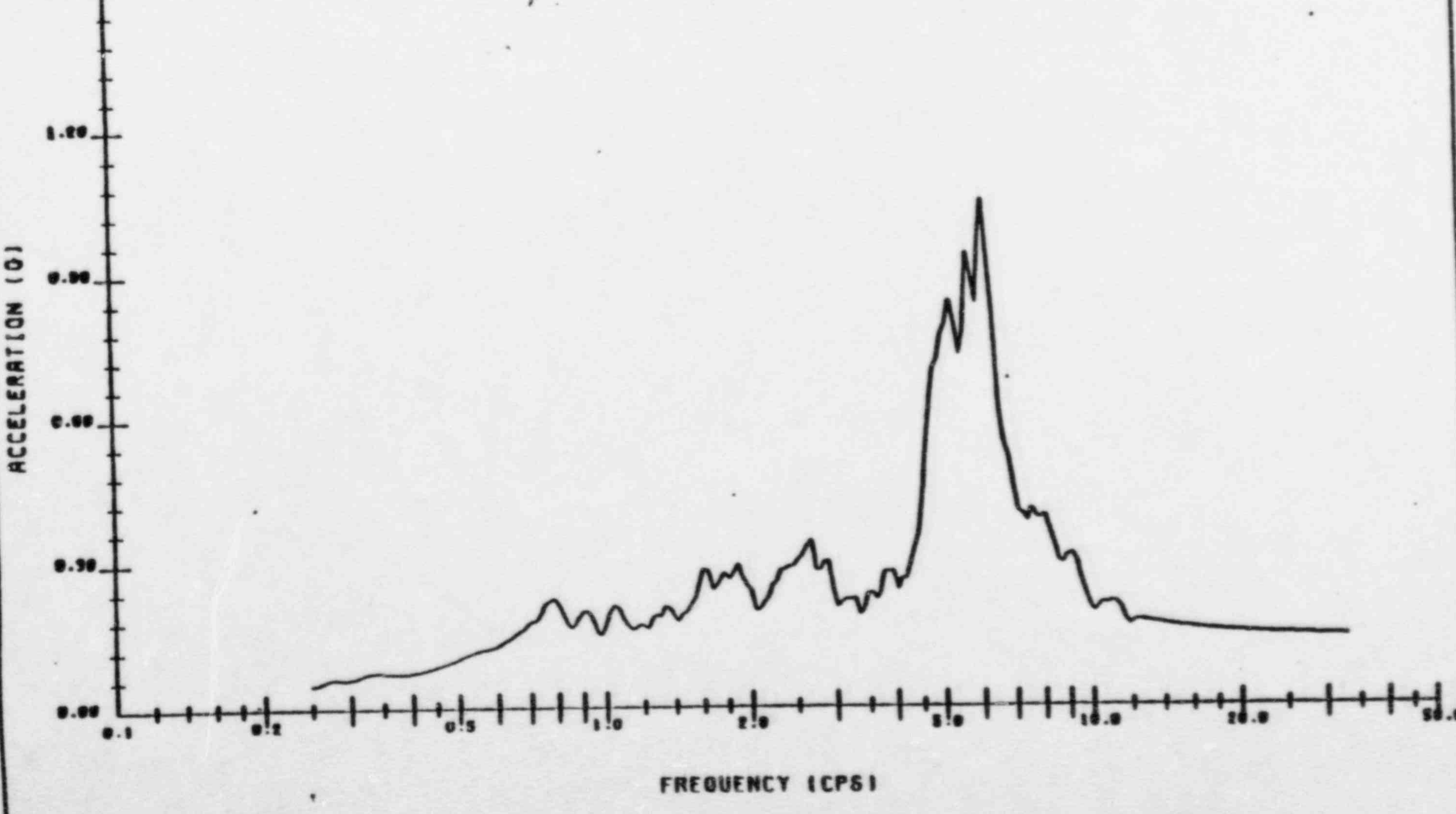
SIMPLIFIED INTERACTION ANAL.- M-6 OBE BASE MAT RESPONSE SPECTRA
(2% DAMPING)

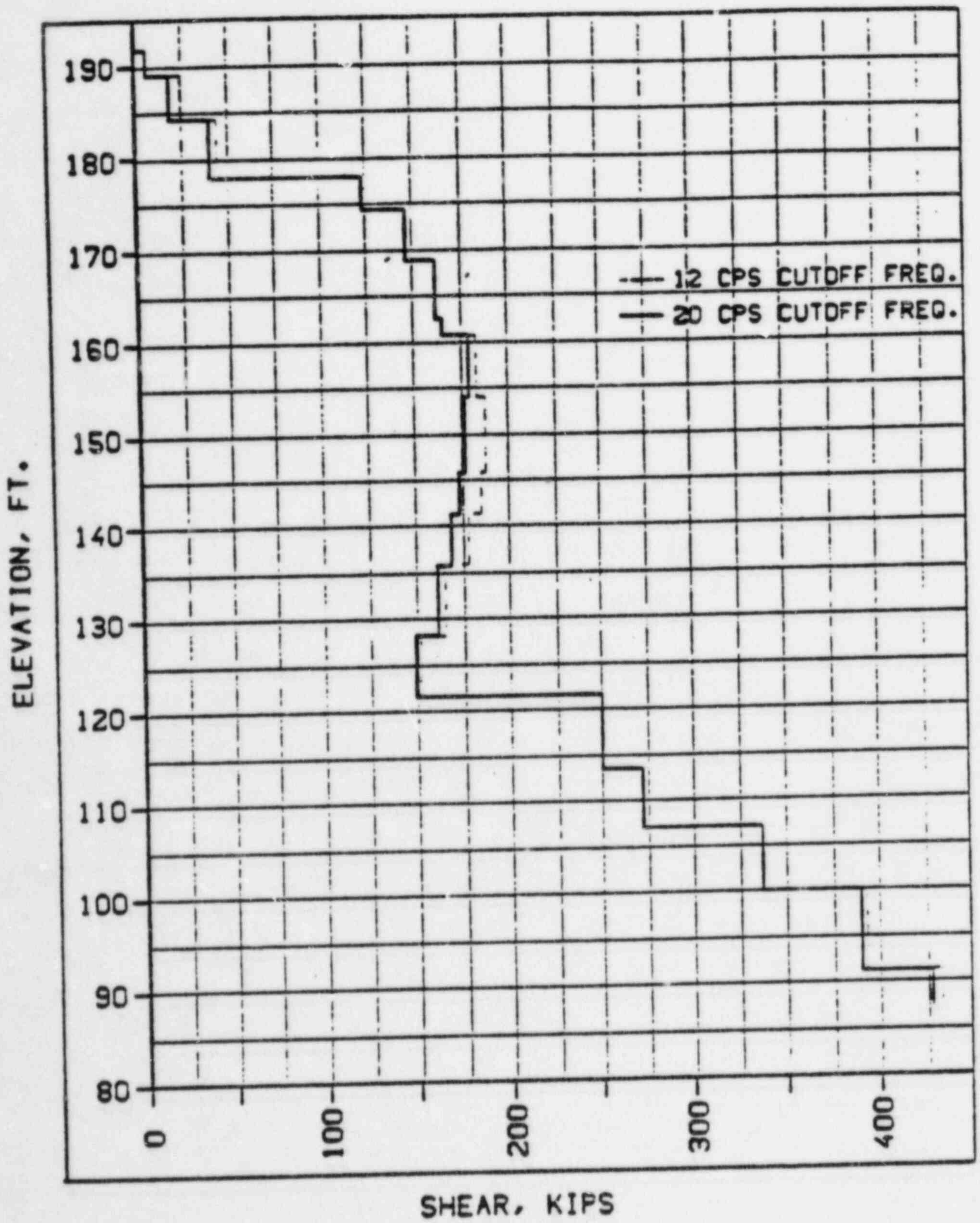
FIGURE 4



DETAILED INTERACTION ANALYSIS: N-S OBE BASE MAT RESPONSE SPECTRUM
(2% DAMPING)

FIGURE 5

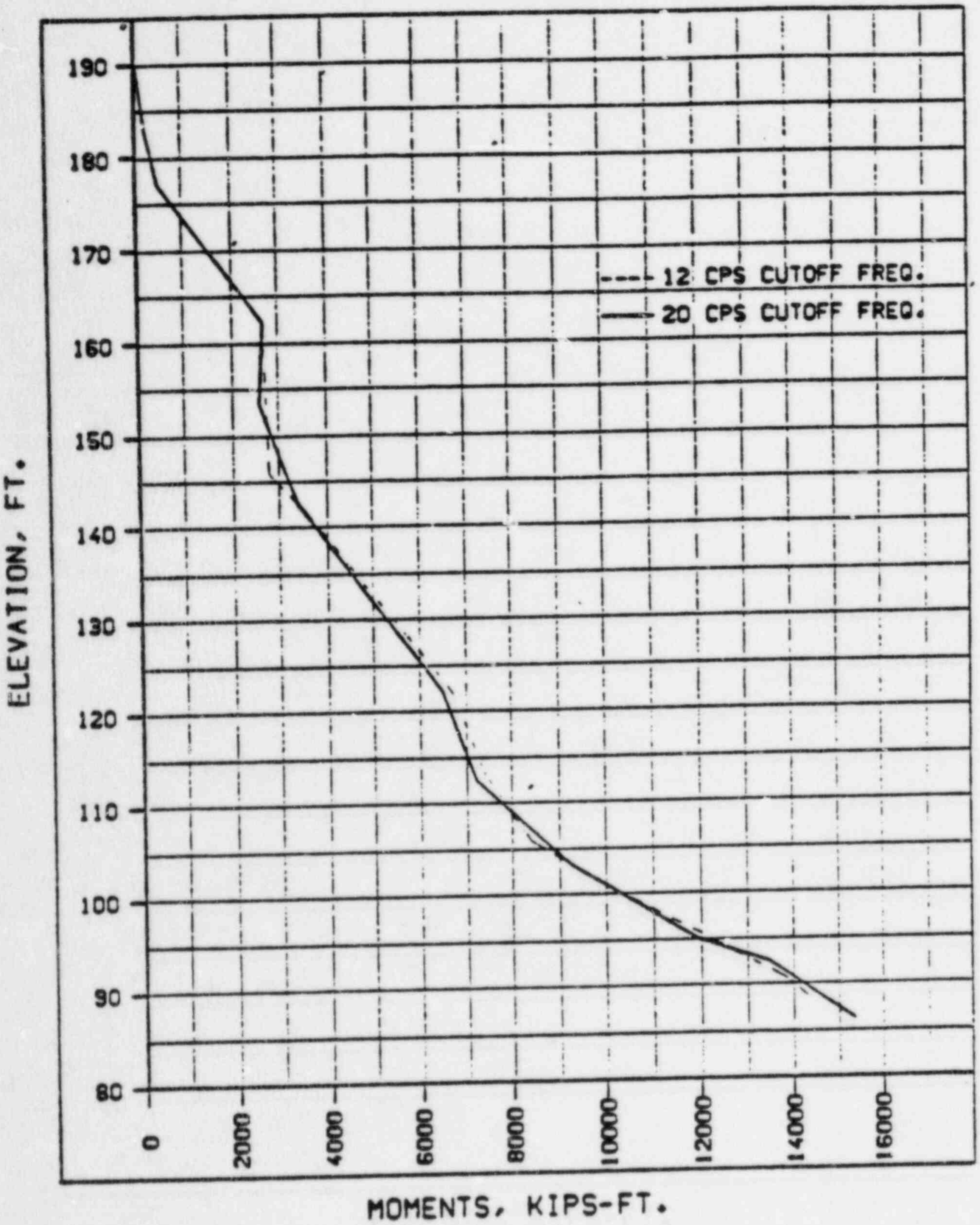




SHEAR RESPONSE MAXIMA FOR DRYWELL

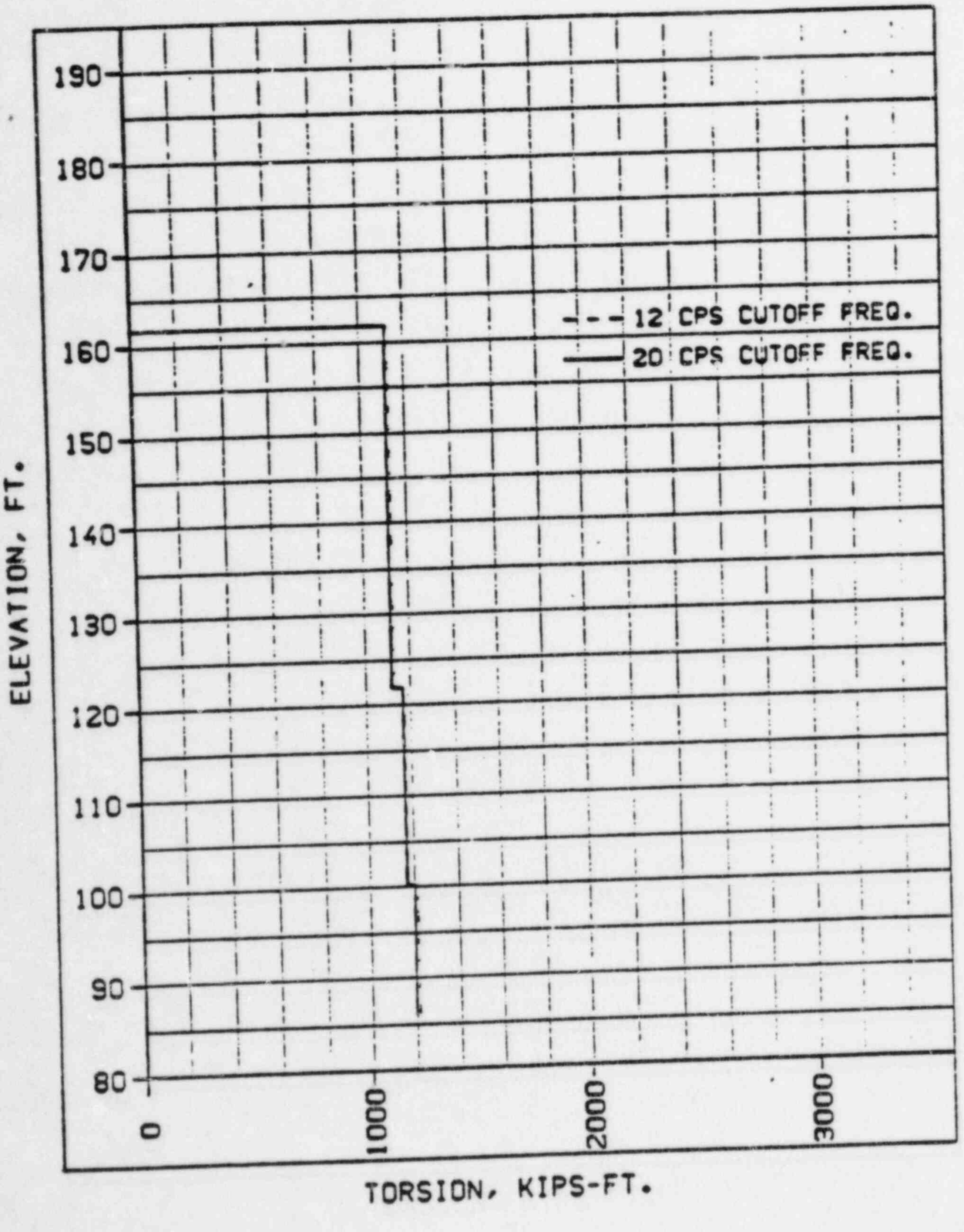
FIGURE 6

JANUARY 11/A-



MOMENT RESPONSE MAXIMA FOR DRYWELL

FIGURE 7

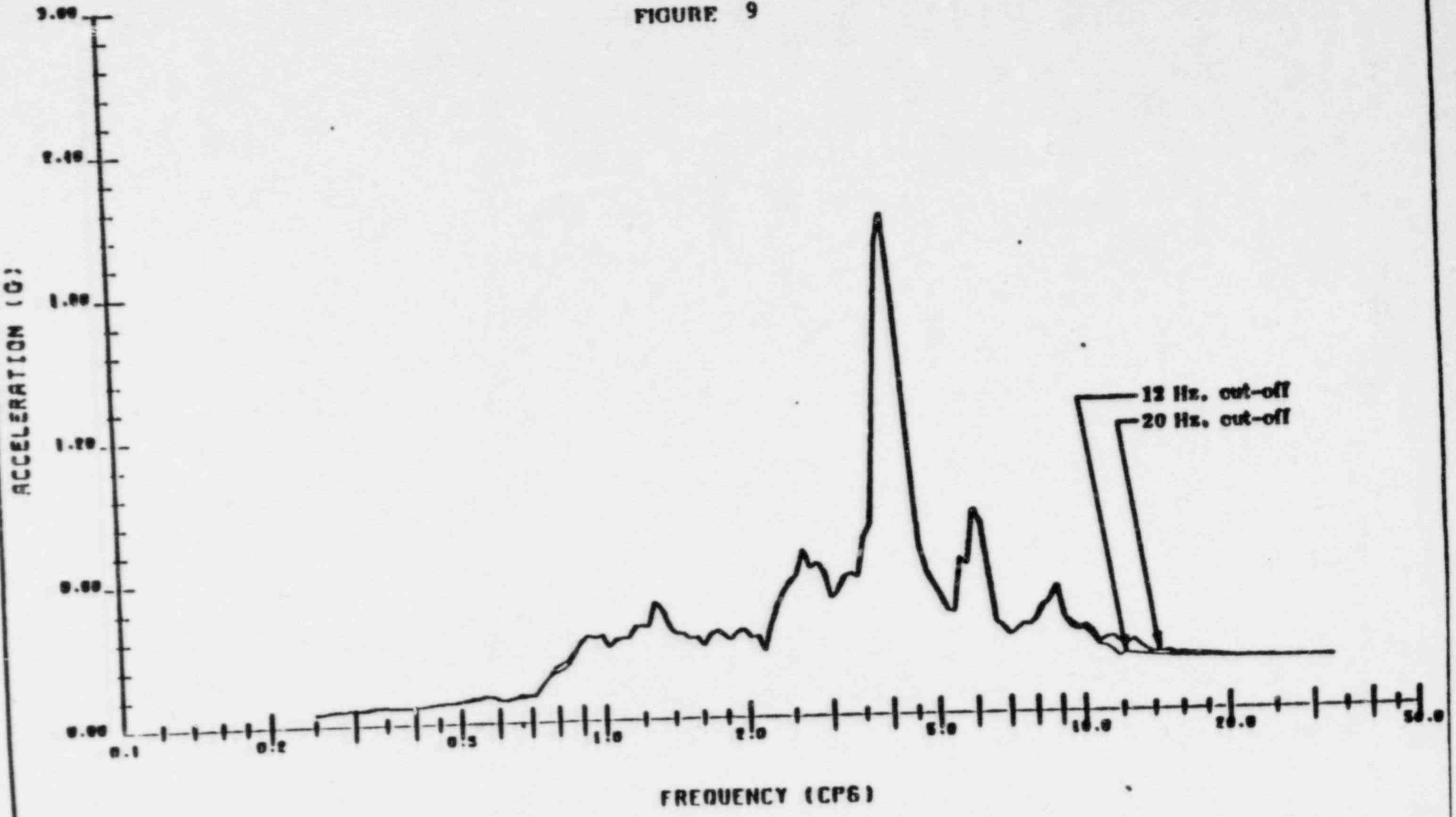


TORSION RESPONSE MAXIMA FOR DRYWELL

FIGURE 8

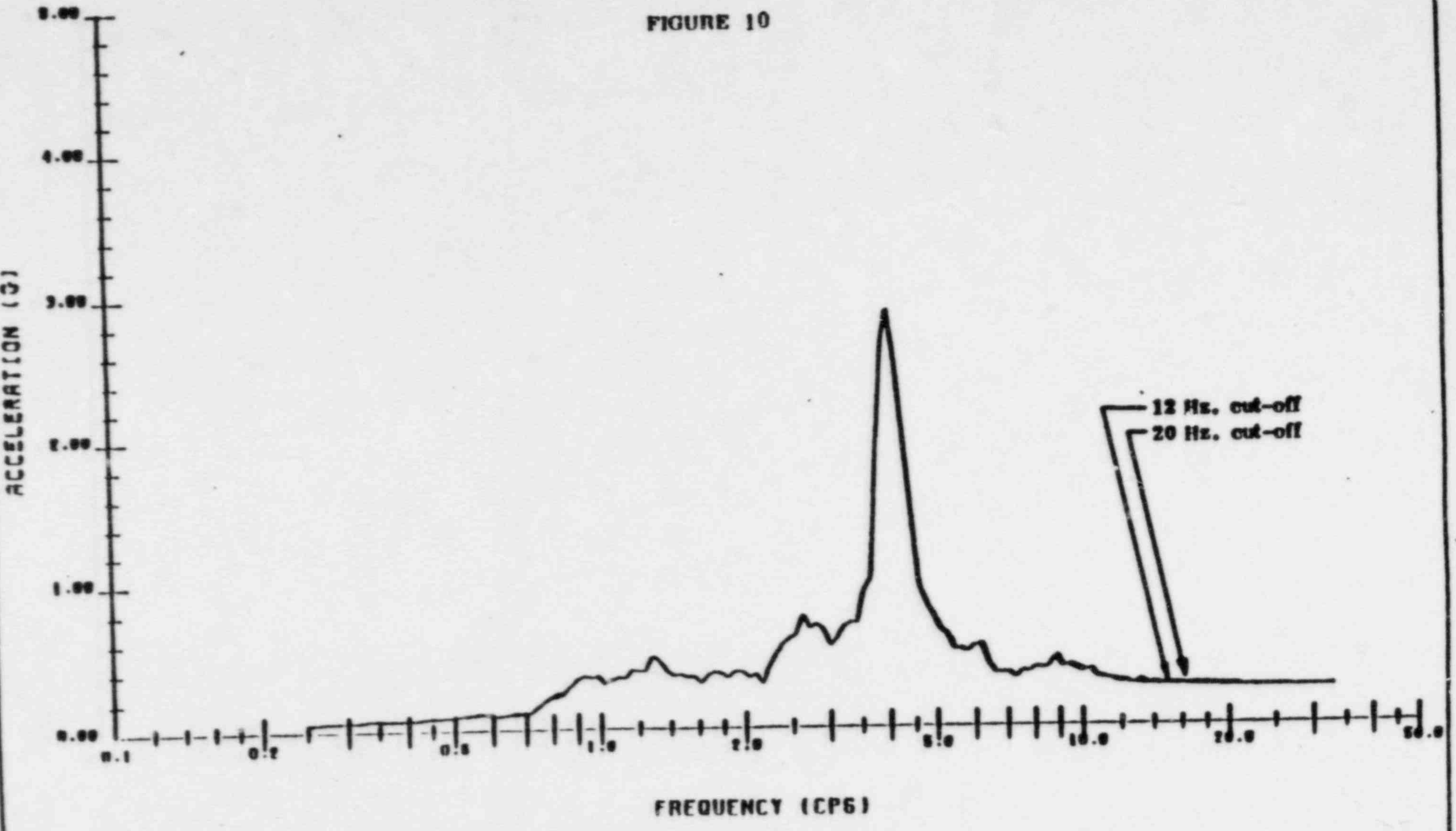
REACTOR BUILDING ELEV 145 N-S OBE RESPONSE DUE TO SIMPLIED SSIA
(2% DAMPING)

FIGURE 9



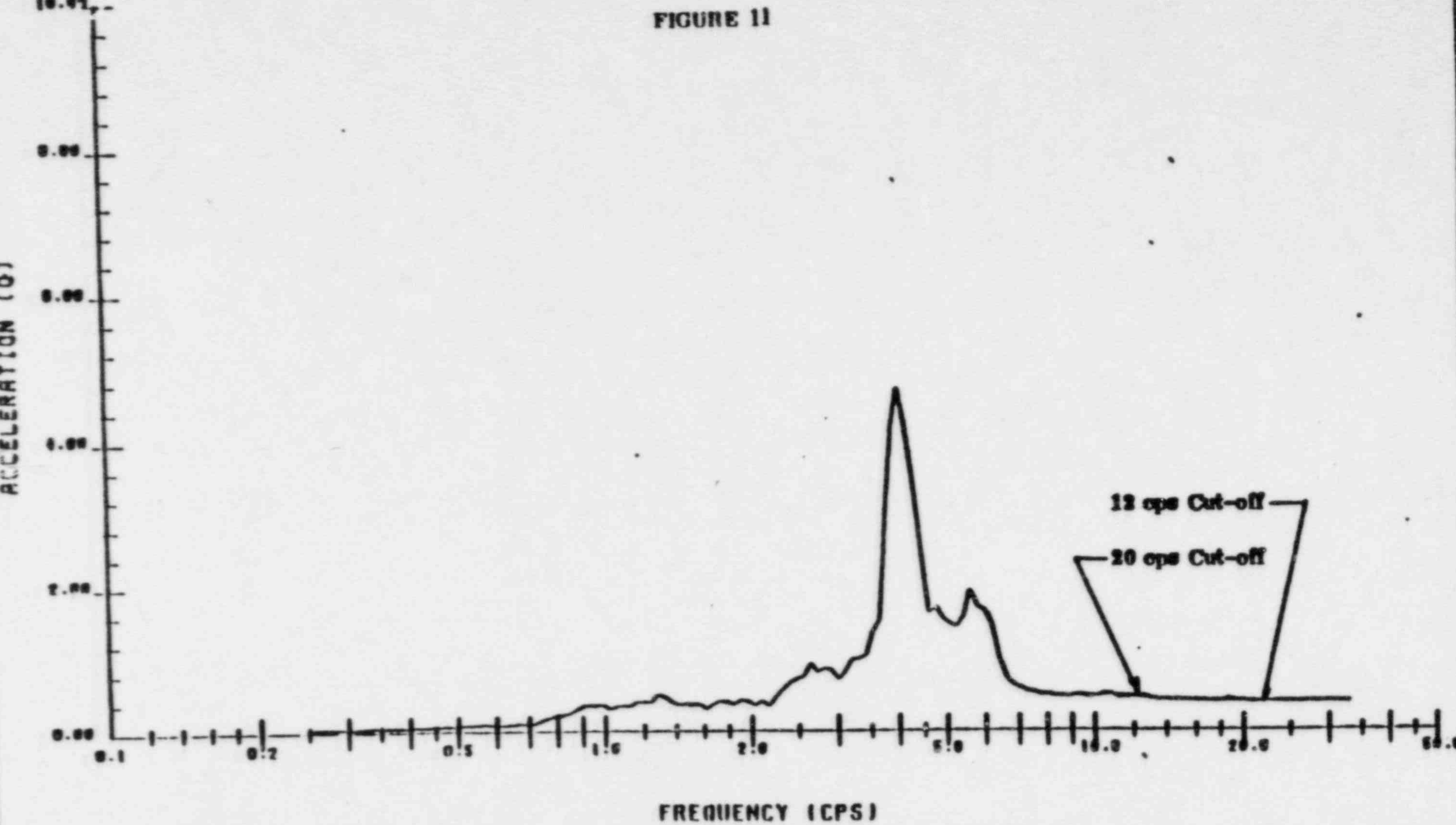
REACTOR BUILDING ELEV 162 N-S OBE RESPONSE DUE TO SIMPLIED SGIA
(2% DAMPING)

FIGURE 10



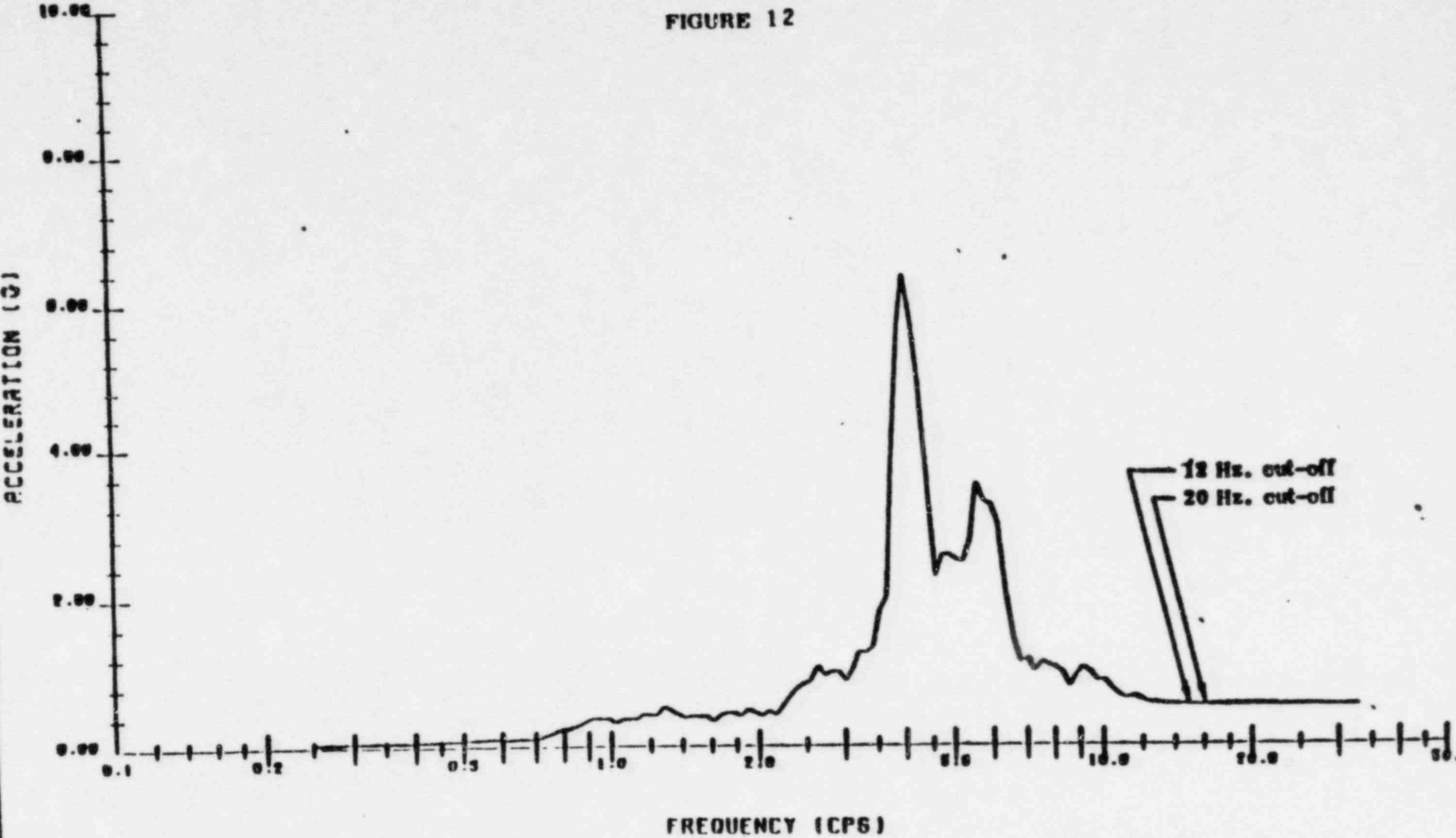
REACTOR BUILDING ELEV. 201 N-S OBE RESPONSE DUE TO SIMPLIFIED SSIR
(2% DAMPING)

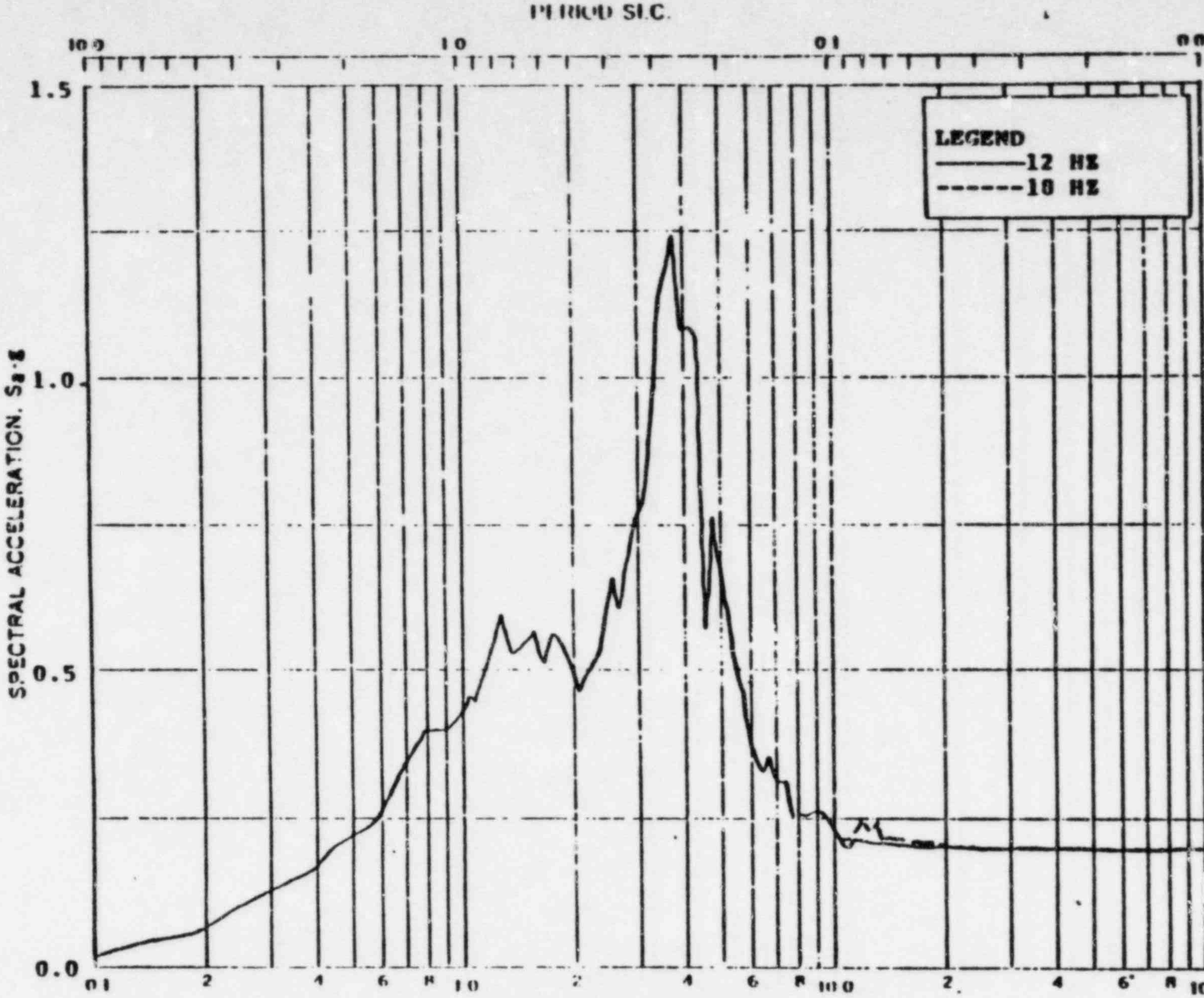
FIGURE 11



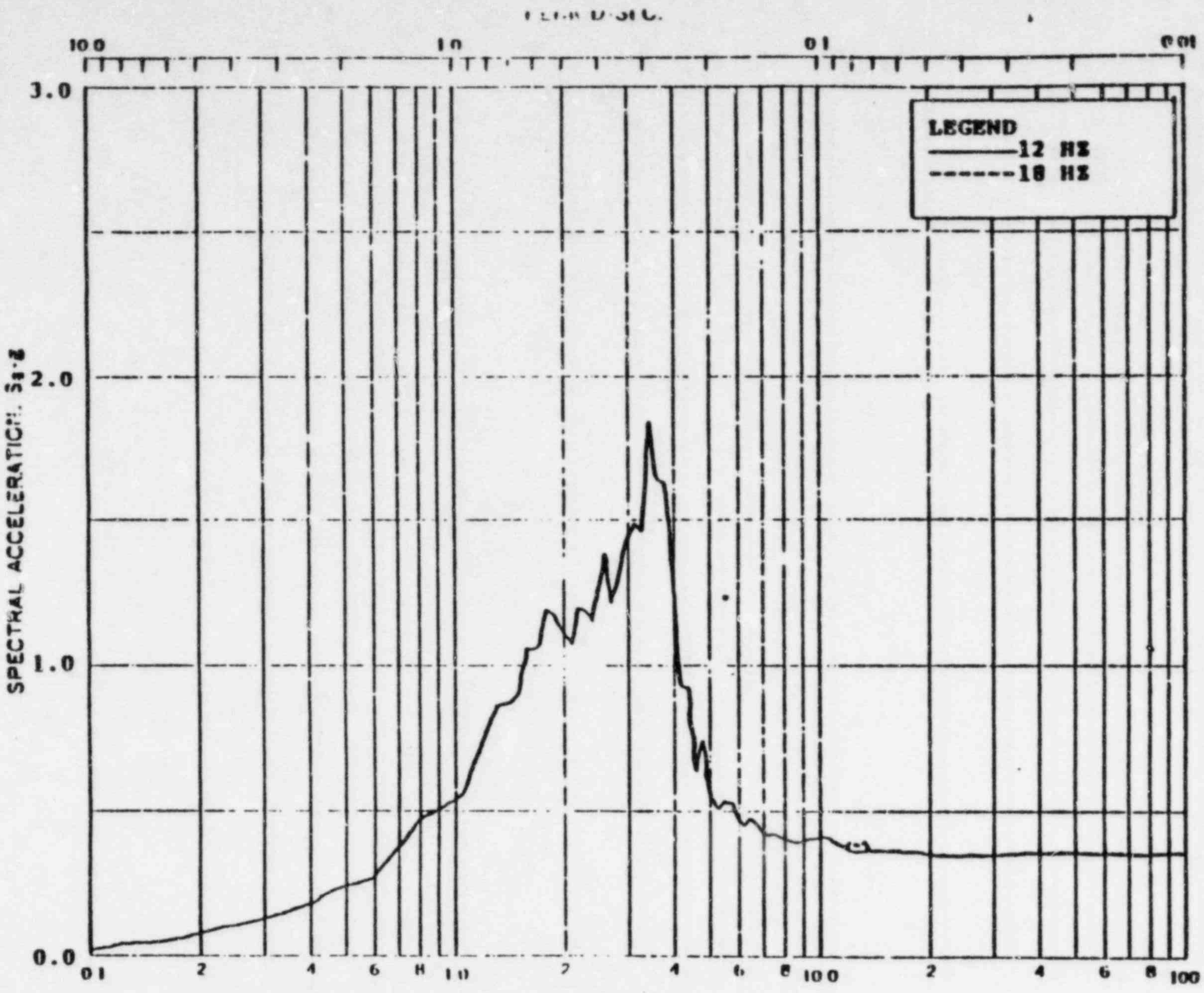
REACTOR BUILDING ELEV 252 N-S OBE RESPONSE DUE TO SIMPLIED SSIR
(2% DAMPING)

FIGURE 12

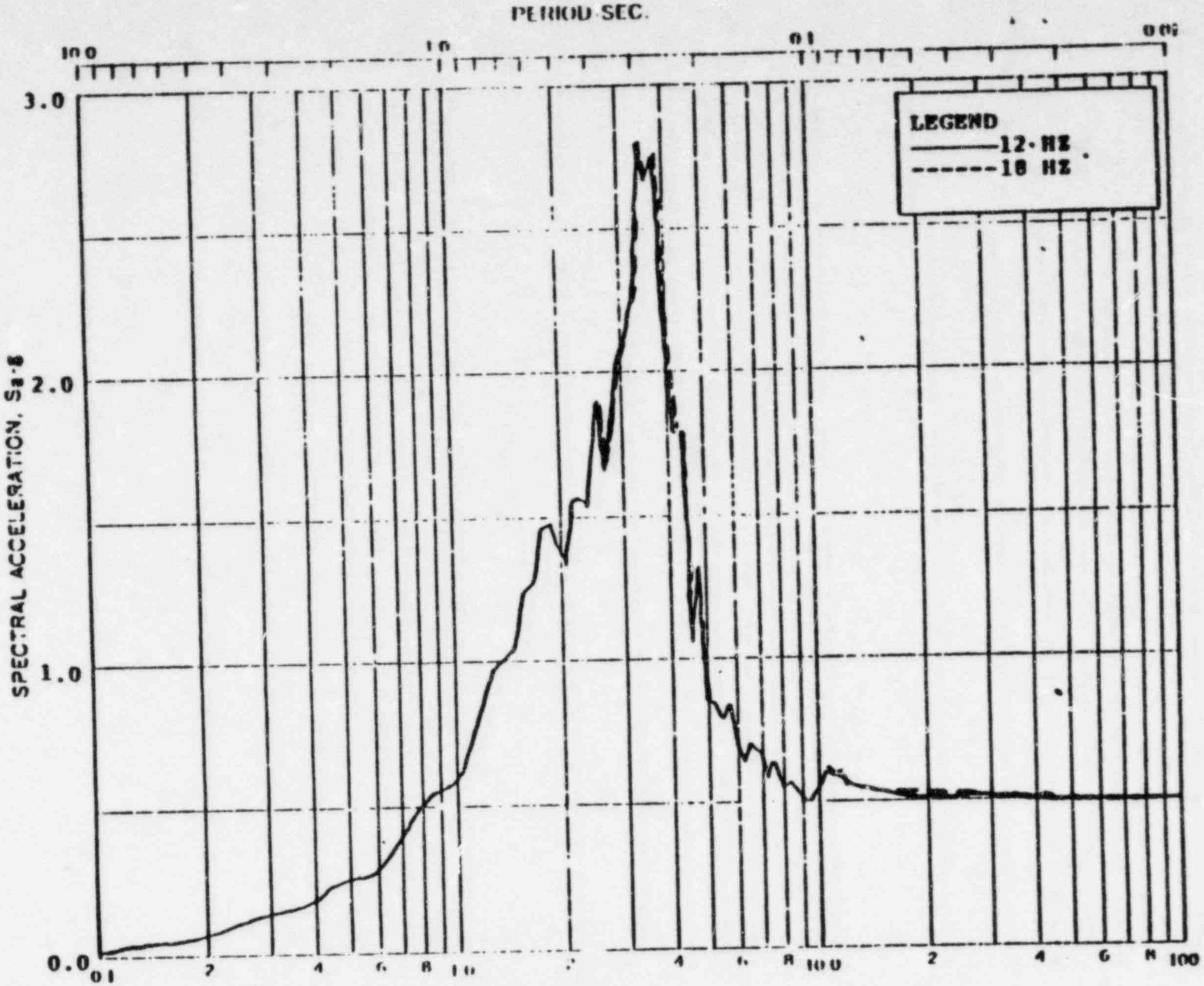




FREQUENCY CUT-OFF STUDY
12 HZ VS. 18 HZ
REACTOR BUILDING AT EL. 54'-0"
N-S, SSE, 2% DAMPING



FREQUENCY CUT-OFF STUDY
12 HZ VS. 18 HZ
REACTOR BUILDING AT EL. 201'-0"
N-S, SSE, 2% DAMPING



FREQUENCY CUT-OFF STUDY
12 HZ VS. 18 HZ
REACTOR BUILDING AT EL. 250'-0"
N-S, SSE, 2% DAMPING

January 11/A-12

Response to NRC Audit

Meeting Date: January 11, 1984

Question No.: B.12

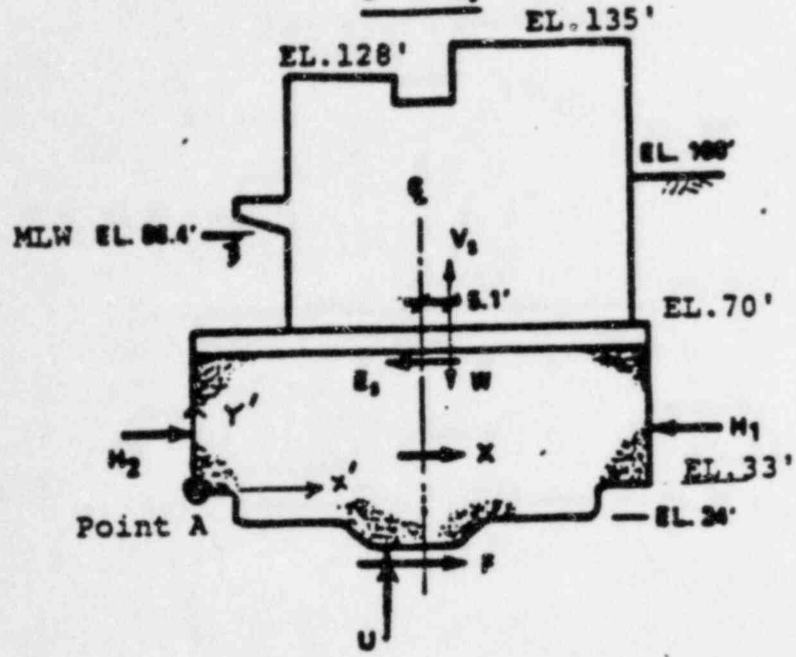
<u>QUESTION:</u> Provide static factor of safety against overturning for intake structure.

<u>RESPONSE:</u> The factor of safety against overturning for the intake structure as given in FSAR Appendix 3G was based on the energy method approach described in BC-TOP-4A (FSAR Reference 3.7-1). During discussions with the NRC, the NRC requested the factor of safety against overturning be calculated using conventional methods. The factor of safety against overturning using conventional methods is 1.12 which exceeds the minimum safety factor of 1.10 specified by SRP Section 3.8.5-II of NUREG-0800. FSAR Appendix 3G has been revised to indicate the factors of safety against overturning by both the energy method and conventional method.

Attached is a simplified calculation of the factor of safety against overturning using conventional methods.

Load Combination

D+H+E_s



OBJECTIVE :

Check Stability of Intake Structure against overturning by conventional meth

FORCES			X'*	Y'*		
	W	106,000^K	57.1	-	E_s	SAFE SHUTDOWN EARTHQUAKE LATERAL FOF
	H_1	29,000	-	16.9	F	SUBGRADE FRICTION RESISTANCE
	H_2	22,760	-	5.7	H_1	DRIVING EARTH PRESSURE
	F	14,920	-	-6.0	H_2	RESISTING EARTH PRESSURE
	U	75,570	51.0	-	U	UPLIFT DUE TO GROUNDWATER AND EXCESS PORE PRESSURE DURING SEISMIC EVENT, OR DESIGN BASIS FLOOD
	E_s	21,210	-	28.5	V_s	SAFE SHUTDOWN EARTHQUAKE VERTICAL FORCE
	V_s	8,480	57.1	-	X	TOTAL SIDE FRICTION
	X	15,860	-	4.5	W	DEAD LOADS

* X' and Y' are the coordinates of the load application point with respect to point A

OTM_A = OVERTURNING MOMENT AT 'A'

$= H_1 y' + E_s y' + U x' + V_s x'$

$= (29,000 \times 16.9) + (21,210 \times 28.5) + (75,570 \times 51.0) + (8,480 \times$

$= 5,432,900$ kft.

RM_A = RESTORING MOMENT AT 'A'

$= W x' + X y' + F y' + H_2 y'$

$= (106,000 \times 57.1) + (15,860 \times 4.5) - (14,920 \times 6.0) + (22,760 \times 5.7)$

$= 6,164,200$ kft.

F.S. = RM/OTM = 6,164,200/5,432,900 = 1.12 > 1.1 o.k.

Meeting Date: January 12, 1984

Question No: A.1

Question: Describe the procedures which assure that the post-modification seismic loads for the torus were examined and that the torus structure was found to be adequate to resist the post-modification seismic loads.

Response: The evaluation of post-modification seismic loads for the torus was separated into two parts: An evaluation for horizontal loads and an evaluation for vertical loads. The support design for the torus, i.e. pinned-pinned vertical columns and pinned lateral restraints, assures that horizontal and vertical behavior are uncoupled, thus allowing consideration of them separately. This was confirmed by the results of the seismic analysis of the unmodified structure, which also show that responses in each of the horizontal and vertical directions are dominated by one structural mode.

For horizontal loads, an evaluation was made of the effects of the torus modifications on the horizontal seismic analysis for the unmodified configuration. It was concluded that the effect of the torus modifications on the horizontal seismic response of the torus is negligible. The modifications added to the torus consist mainly of local column connection stiffening which does not significantly change the dominant horizontal torus frequency. The original analysis for horizontal loads is conservative, since the stiffening effect, though insignificant, would tend to increase the dominant frequency, resulting in lower accelerations applied to the torus because of the position of the frequency on the response spectrum curve. The evaluation described above was performed as part of the Hope Creek Plant Unique Analysis.

For vertical loads, a new analysis was performed using a finite element model of the modified torus. The results of this analysis are documented in the Hope Creek Plant Unique Analysis Report (PUAR).

Meeting Date: January 12, 1984

Question No: A.1 (Cont.)

The resulting vertical loads per support due to seismic loads are provided in PUAR Table 2-2.5-2. Combined column loads, which include seismic, hydrodynamic, and other loads, are reported in PUAR Table 2-2.5-4, and are compared to the allowable column loads. The maximum combined suppression chamber stresses, which include the effects of revised seismic loads and hydrodynamic loads, are reported in PUAR Table 2-2.5-3, and are compared to allowables therein. As can be seen by examining these tables, all column loads and component stresses are within allowable limits.

Response to NRC Audit

Revision 1
6/30/84

Meeting Date: January 12, 1984

Question No: A.3

Question: Describe how the effects of relative seismic displacements of the torus were considered for the stress evaluation of the vent system.

Response: The effects of relative seismic displacements of the torus on the vent system were considered by approximating the maximum relative support displacements using the floor response spectra. The maximum displacement for each support is predicted by $Sd = Sag/w^2$ where Sa is the spectral acceleration in g's at the high frequency end of the spectrum curve, g is the gravity constant, and w is the fundamental structural frequency of the torus in radians per second.

The resulting displacements from this calculation, 0.01 in. in the horizontal direction and 0.017 in. in the vertical direction, are small compared with those of other major vent system loadings such as SRV discharge and pool swell and have a neglible effect on the vent system. Therefore relative seismic displacements of the torus were not included in the stress evaluation of the vent system. To justify this assumption an evaluation of the effects of relative seismic displacements of the torus is provided as follows.

Imposing the torus seismic displacements on the vent system at the torus attachment points results in an increase in maximum primary membrane stress in the vent header of 0.54 ksi. The maximum increase in local primary membrane stress at the most highly stressed vent header - downcomer intersection is 0.01 ksi in the vent header and 0.01 ksi in the downcomer. Examining the maximum combined stresses shown in PUAR Table 3-2.5-3, it is apparent that these small increases in stress have a negligible effect on the adequacy of the containment.

Revised Response
Revision 1
July 10, 1984

Response to NRC Audit

Meeting Date: January 12, 1984

Question No.: A.4

QUESTION: Review the seismic design of all Seismic Category I tanks to determine whether the flexibility of the tank wall and the water mass within the tank were considered. For those tanks where these effects have not been considered, assess the impact of including these effects.

RESPONSE: All Seismic Category I tanks were reviewed to determine whether the flexibility of the tank wall and the water mass within the tank were considered. Review has indicated that in all tanks, except the diesel fuel oil storage tank, fluid mass and tank wall flexibility are addressed adequately and meet the guidelines of NUREG/CR-1161. The results of this review are shown in Table A.4-1.

In the case of the diesel fuel oil storage tank, an analysis to qualify the tank to include the effect of fluid mass and tank wall flexibility has been performed using the finite element method. The results of this revised analysis are compared with the original analysi results in Table A.4-2.

TABLE A.4-1

SUMMARY OF RESULTS FOR SEISMIC CATEGORY I TANKS
(EXCEPT DIESEL FUEL OIL STORAGE TANKS)

TANK DESCRIPTION	METHOD OF ANALYSIS	CRITICAL STRESS		
		LOCATION	STRESS (KSI)	
			CALCULATED	ALLOWABLE
FUEL OIL DAY TANKS Horizontal Length (L) = 31.5 in. Radius (R) = 18.0 in. Thickness = 3/8 in.	Finite element model was used. Fluid mass was considered in the analysis. sloshing effects were also considered. Minimum frequency is 38 Hz. L/R = 7.3	Bolt, Tension Shear	5.1 1.7 (SSE)	20.2 8.2
JACKET WATER EXPANSION TANKS Vertical Height (H) = 56.0 in. Radius (R) = 12.0 in. Thickness = 3/8 in.	Tank was qualified by similarity to fuel oil day tanks and was found to be more rigid than day tanks. Fluid mass and sloshing effects were considered. H/R = 4.7	Bolt, Tension Shear	4.6 0.3 (SSE)	20.2 8.2
LUBE OIL MAKE-UP TANKS Vertical Height (H) = 87.5 in. Radius (R) = 15.25 in. Thickness = 3/8 in.	Tank was qualified by similarity to fuel oil day tanks and was found to be stiffer than day tanks. Fluid mass and sloshing effects were considered. H/R = 5.7	Bolt, Tension Shear	3.8 0.8 (SSE)	20.2 8.2

TANK DESCRIPTION	METHOD OF ANALYSIS	CRITICAL STRESS		
		LOCATION	STRESS (KSI)	
			CALCULATED	ALLOWABLE
SACS EXPANSION TANKS Vertical Height (H) = 212.0 in. Radius (R) = 60.0 in. Thickness = 5/16 in.	Tank was analyzed by equivalent static method. Minimum frequency of 12.5 Hz was obtained using a beam model. Effect of water mas was considered. Sloshing effects were considered in the stress evaluation. H/R = 3.5	Bolt, Tension Shear	19.2 9.8 (SSE)	27.0 17.1
CONTROL AREA CHILLED WATER SYSTEM HEAD TANKS Horizontal Length (L) = 61.0 in. Radius (R) = 12.0 in. Thickness = 3/16 in.	Equivalent static analysis was performed. Fundamental frequency was found to be in the rigid range (>39Hz). Effect of water mass was considered. Sloshing effects were considered in the stress evaluation L/R = 5.0	Shell, at lugs Shell, at 2"Ø inlet Shell, at 2"Ø outlet Shell, at 1"Ø drain Mounting Tab Bolt, Tension Shear	24.9 15.0 12.8 12.4 1.1 5.8 6.7 (OBE)	25.9 15.7 15.7 15.7 17.8 20.0 10.0

JUL 10 '84 0267572

TANK DESCRIPTION	METHOD OF ANALYSIS	CRITICAL STRESS		
		LOCATION	STRESS (KSI)	
			CALCULATED	ALLOWABLE
HEAD TANKS FOR SERVICE WATER PUMP LUBRICATION SYSTEM Horizontal Length (L) = 63.0 in. Radius (R) = 21.0 in. Thickness = 3/16 in.	Minimum frequency of 31.1 Hz was obtained using beam model, including the water mass. Sloshing effects were considered in the stress evaluation. Equivalent static analysis was used. L/R = 3.0	Shell, at support Shell, at 1" Ø inlet Shell, at 2" Ø outlet Bolt, Shear Tension	25.7 8.1 13.1 5.8 10.4 (SSE)	37.7 37.7 37.7 13.3 18.1
HYDROPNEUMATIC ACCUMULATOR TANKS Vertical Height (H) = 192.0 in. Radius (R) = 60.0 in. Thickness = 3/4 in.	Tank was analyzed by finite element analysis, including the water mass. Minimum frequency was 30.6 Hz. Sloshing effects were considered in the stress evaluation. H/R = 3.2	Shell, Circum-Tensile Inlet/outlet insert plate at shell Inlet/outlet nozzle to insert plate Bolt, Tension Shear	16.8 27.6 27.3 39.2 11.2 (SSE)	17.5 28.9 28.9 52.5 21.7

TABLE A.4-2

SUMMARY OF RESULTS FOR
DIESEL FUEL OIL STORAGE TANKS

METHOD OF ANALYSIS	CRITICAL STRESS		
	LOCATION	STRESS (ksi)	
		Calculated (Revised/Original)	OBE Allowable(1)
ORIGINAL Beam model was used. Minimum frequency >34 Hz. **REVISED** Finite element quarter model was used. Fluid mass was uniformly distributed over the tank shell. Minimum frequency = 20 Hz. **HORIZONTAL** Length = 40 ft, Diameter = 11 ft. Shell thickness = 0.313 inch.	Junction of cylindrical shell and saddle	(5.4/(2)) OBE (6.5/(2)) SSE	17.5
	Saddle Support	(11.8/2.0)(3) OBE (13.8/2.6) SSE	20.6
	Head	(1.9/2.0) OBE (2.2/2.2) SSE	18.9
	Bolt, Shear (4)	(5.7/3.3) OBE (9.0/4.8) SSE	10.0
	Saddle Stiffeners	(15.4/(2)) OBE (17.4/(2)) SSE	20.6
	Base Plate	(7.7/ (2)) OBE (12.4/(2)) SSE	20.6

NOTES

1. SSE allowable stresses are not given since SSE calculated stresses are less than the OBE allowable stresses.

2. Original calculated stresses are not available.

3. Original analysis did not consider local stress evaluation.

Response to NRC Audit

Meeting Date: January 12, 1984

Question No.: B.2

QUESTION: With respect to the ultimate capacity of the containment, expand the analysis to include the ultimate capacity of the materials and eliminate seismic considerations.

RESPONSE: The ultimate capacity analysis of the containment has been expanded to include the minimum specified tensile strengths of the materials and to eliminate seismic considerations. The resulting minimum ultimate internal pressure equals 190 psi. Therefore, the safety margin against the design pressure of 62 psi is 3.06.

Appendix 3I has been added to the FSAR to describe the ultimate capacity analysis of the containment.

The above response has also been given in response to Question 220.22.