PSE&G Public Service Electric and Gas Company

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

Robert L. Mittl General Manager
Nuclear Assurance and Regulation

September 14, 1984

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20814

Attention: Mr. Albert Schwencer, Chief
Licensing Branch 2
Division of Licensing

Gentlemen:

HOPE CREEK GENERATING STATION
DOCKET NO. 50-354
FSAR COMMITMENT STATUS THROUGH AUGUST 1984

Public Service Electric and Gas Company presently does not plan to issue Amendment No. 8 to the Hope Creek Generating Station Final Safety Analysis Report before October 1984. Accordingly, this letter is provided to document the status of Hope Creek Generating Station responses to NRC requests for additional information which were forecasted to be responded to by August 1984.

Attachment I is a tabulation of the Hope Creek Generating Station Final Safety Analysis Report commitments for August 1984, and the corresponding resolution for each commitment. Attachments II through VIII provide responses to questions forecasted to be responded to in August 1984, which will be included in Amendment No. 8.

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Should you have any questions in this regard, please contact us.

Very truly yours,

R L Mittl

Attachment I - Hope Creek Generating Station - FSAR Commitment Status through August 1984
Attachment II - Response to Question 210.53
Attachment III - Response to Question 220.21
Attachment IV - Response to Question 480.11
Attachment V - Response to Question 100.6 (I.C.5)
Attachment VI - Response to Question 260.15
Attachment VII - Response to Question 471.14
Attachment VIII - Response to FSAR Table 13.1-4

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

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

MP84 123/06 1/2D

ATTACHMENT I
HOPE CREEK GENERATING STATION
FSAR COMMITMENT STATUS THROUGH AUGUST 1984

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
1. Question/Response Appendix: Question 210.1	This commitment concerns providing FSAR Section 3.6 Tables and Figure updated to final stress information. FSAR Tables 3.6-8, 9, 10, 11 have been updated to preliminary stress information and are included in Amendment 6 to the HCGS FSAR. FSAR Table 3.6-5 contains final stress information and is included in Amendment 7. FSAR Figure 3.6-34 has been partially completed and is included in Amendment 6. The preliminary tables and figure listed above will be finalized in December 1984. This revised commitment date will be included in Amendment 8 to the HCGS FSAR.
2. Question/Response Appendix: Question 210.21	This commitment concerns providing the tables and figures listed in Q210.1 and will be resolved as stated above.
3. Question/Response Appendix: Question 210.53	This commitment concerns providing criteria and analysis results for stiff pipe clamps per IE Information Notice 83-80. This information is provided in Attachment II and will be included in Amendment 8 of the HCGS FSAR.
4. Question/Response Appendix: Question 220.15	This commitment concerns providing sketches and mathematical models and method of analysis for the Spent Fuel Racks. This information is provided as response to DSER Item No. 140 in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated September 7, 1984, and will be included in Amendment 8 to the HCGS FSAR.

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
5. Question/Response Appendix: Question 220.21	This commitment concerns the results of the Soil-Structure Interaction Analysis for the reactor and auxiliary buildings and the service water intake structure. This information is provided in Attachment III and will be included in Amendment 8 to the HCGS FSAR.
6. Question/Response Appendix: Question 281.2	This commitment concerns providing Spent Fuel Rack FSAR Figures 9.1-3 and 9.1-4. This information is provided in Amendment 7 to the HCGS FSAR.
7. FSAR Section 9.1.2.2.2.2	This commitment concerns an update to the Spent Fuel Storage System description. This information is provided in Amendment 7 to the HCGS FSAR.
8. Question/Response Appendix: Question 410.38	This commitment concerns providing Spent Fuel Pool criticality information. This information is provided as response to DSER Item No. 140 in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated September 7, 1984, and will be included in Amendment 8 to the HCGS FSAR.
9. Question/Response Appendix: Question 410.39	This commitment concerns providing Spent Fuel Rack design details. This information is provided in Amendment 7 to the HCGS FSAR.
10. Question/Response Appendix: Question 421.13b	This commitment concerns testing of isolation systems. This information is provided in Amendment 7 to the HCGS FSAR.

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
11. Question/Response Appendix: Question 421.21	This commitment concerns an evaluation of the effects of high temperature on reference legs of water level measuring instruments. This information is provided as response to DSER Item No. 202 in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated August 3, 1984, and will be included in Amendment 8 to the HCGS FSAR.
12. Question/Response Appendix: Question 430.164	This commitment concerns the ISI program for the main condenser including the frequency and extent of inspection. This information is provided in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated August 1, 1984, and will be included in Amendment 8 to the HCGS FSAR.
13. Question/Response Appendix: Question 430.167	This commitment concerns the IST and ISI program for the turbine bypass system including the frequency and extent of testing and inspection. This information is provided in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated August 1, 1984, and will be included in Amendment 8 of the HCGS FSAR.
14. Question/Response Appendix: Question 440.10	This commitment concerns the trip settings for the plant leak detection system. This information will be provided in January 1985. This revised commitment date will be included in Amendment 8 to the HCGS FSAR.

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
15. Question/Response Appendix: Question 440.27	This commitment concerns which evaluation model is to be used for the ECCS analysis. This information will be provided in July 1985 as stated in response to DSER Item No. 136 in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated August 20, 1984. This revised commitment date will be included in Amendment 8 to the HCGS FSAR.
16. Question/Response Appendix: Question 480.11	This commitment concerns updating FSAR Table 3.6-5 per the results of the pressure temperature transient analyses. This information is provided in Attachment IV and will be included in Amendment 8 to the HCGS FSAR. FSAR Table 3.6-5 is provided in Amendment 7 to the HCGS FSAR.

STATUS OF AUGUST 1984 FSAR COMMITMENTS MADE IN FSAR COMMITMENT STATUS LETTER TO NRC (REFERENCE: R. L. MITTL (PSE&G) TO A. SCHWENCER, DATED AUGUST 13, 1984, ATTACHMENT I).

17. Question/Response Appendix: Question 100.6	Re: TMI Item I.C.5: This commitment concerns assuring feedback of operating personnel via procedures. This information is provided in Attachment V and will be included in Amendment 8 to the HCGS FSAR.

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
17. Question/Response Appendix: Question 100.6 (Cont'd)	Re: TMI Item II.B.3: This commitment concerns assuring compliance of the radioactive gas and liquid sampling system for shielding and source term requirements. This information is provided as response to DSER Item No. 148 (see Question 281.15) in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated September 12, 1984, and will be included in Amendment 8 to the HCGS FSAR.
18. Question/Response Appendix: Question 260.15	This commitment concerns revising FSAR Section 1.8 to reflect conformance with listed Reg. Guides which are applicable during operations phase. This information is provided in Attachment VI and will be included in Amendment 8 to the HCGS FSAR.
19. Question/Response Appendix: Question 281.14	This commitment concerns the materials monitoring program for the Spent Fuel Pool. This information is provided as response to DSER Item No. 140 in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated September 7, 1984, and will be included in Amendment 8 to the HCGS FSAR.
20. Question/Response Appendix: Question 281.15	This commitment concerns information on the Post Accident Sampling System which demonstrates compliance with NUREG-0737, Item II.B.3. This information is provided as response to DSER Item No. 148 in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated September 12, 1984, and will be included in Amendment 8 to the HCGS FSAR.

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
21. Question/Response Appendix: Question 410.91	This commitment concerns the ability of check valves in the Equipment and Floor Drain System to maintain a functional pressure boundary. This information is provided as response to DSER Item No. 149 in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated July 27, 1984, and will be included in Amendment 8 to the HCGS FSAR.
22. Question/Response Appendix: Question 410.93	This commitment concerns seismic qualifications of check valves in drainage systems. This information is provided as response to DSER Item No. 149 in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated July 27, 1984, and will be included in Amendment 8 to the HCGS FSAR.
23. Question/Response Appendix: Question 421.26	This commitment concerns reactor mode switch contact misoperations. This information is provided in response to DSER Item No. 192 in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated August 15, 1984, and will be included in Amendment 8 to the HCGS FSAR.
24. Question/Response Appendix: Question 471.14	This commitment concerns providing the resume for Senior Radiation Protection Supervisor. This information is provided in Attachment VII and will be included in Amendment 8 to the HCGS FSAR.

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
25. FSAR Table 13.1-4	This commitment concerns providing resumes for Maintenance Manager and Senior Nuclear Maintenance Supervisor. This information is provided in Attachment VIII and will be included in Amendment 8 to the HCGS FSAR.

ATTACHMENT II

QUESTION 210.53 (SECTION 3.9.3)

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

RESPONSE

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

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

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

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

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

No problems were identified by the evaluations and calculations described above. The limits of Section III of the ASME B&PV Code were not violated.

GE's Design Memos #170-107 and #170-109 and Bechtel's Stress Report SR-10855-SS-27 were submitted under separate cover (letter from R.L. Mittl, PSE&G to A. Schwencer, NRC, dated August 14, 1984.

ATTACHMENT III

QUESTION 220.21 (SECTION 3.7.2)

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

RESPONSE

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

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

DELETE

REPLACE WITH "INSERT A"

Insert A

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

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

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

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

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

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

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

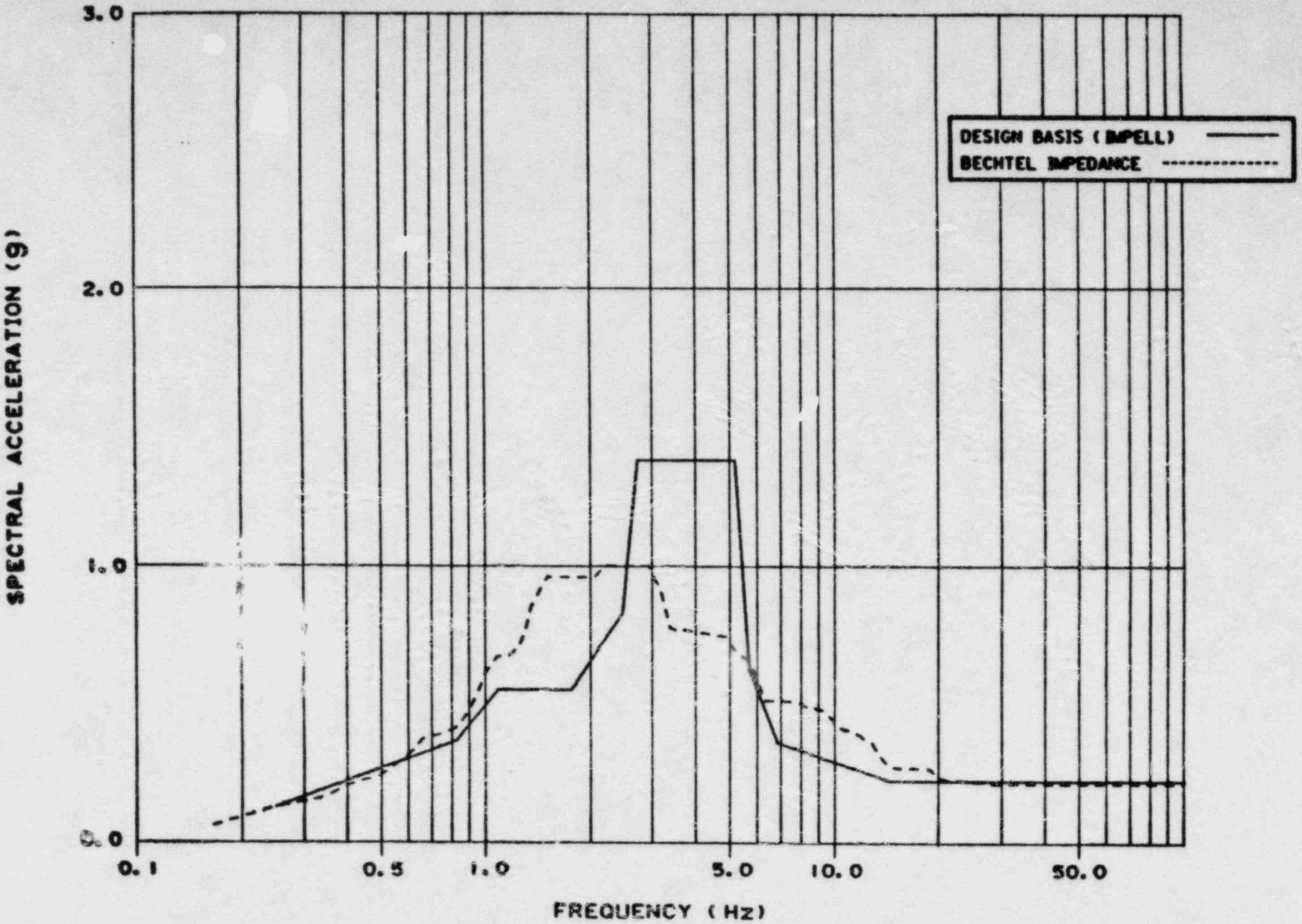

FIGURE 220.21-1

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

HOPE CREEK UNIT 1
CAD, L1A SHT 2, REV 0
07/27/84

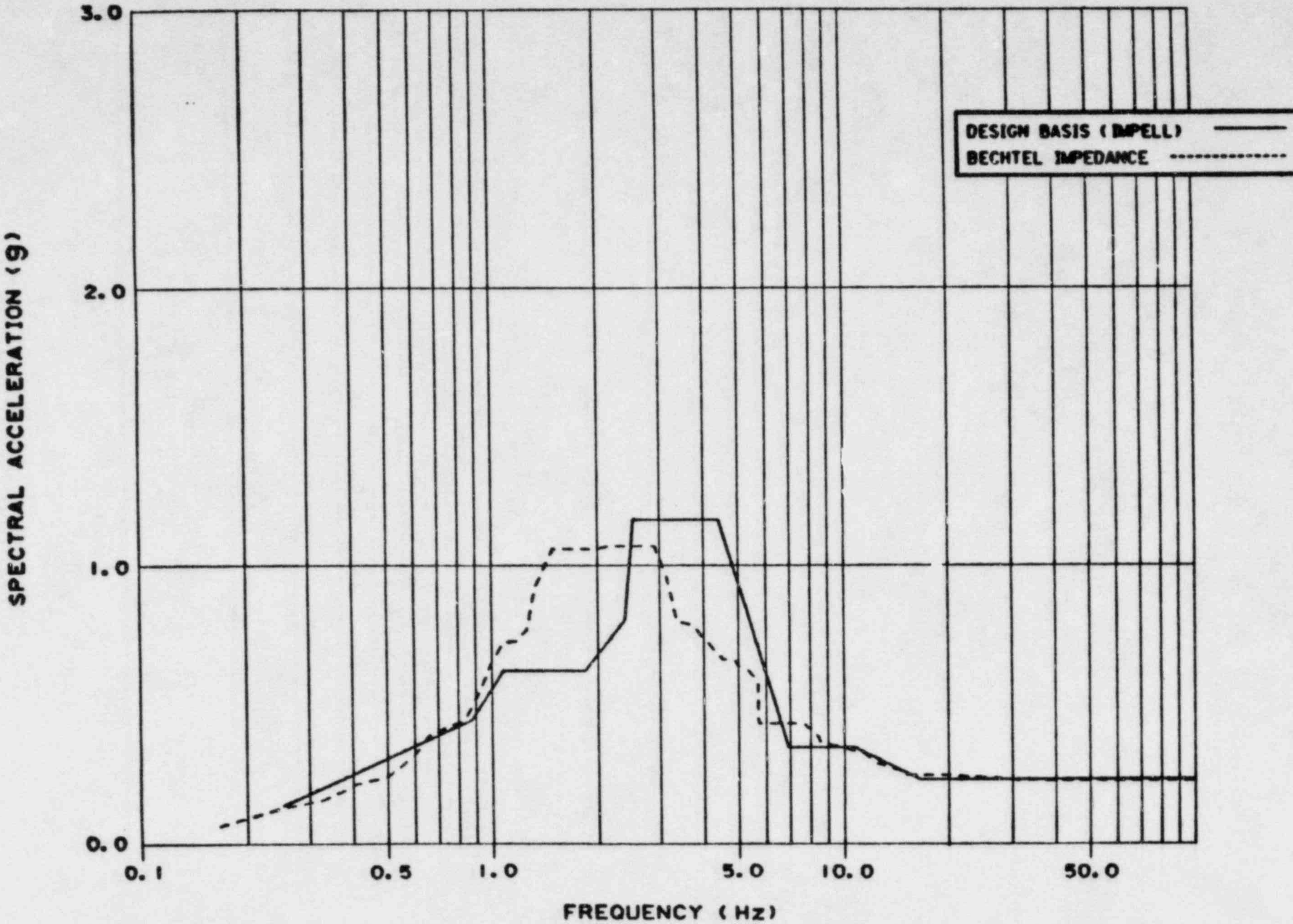

FIGURE 220.21-2

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

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

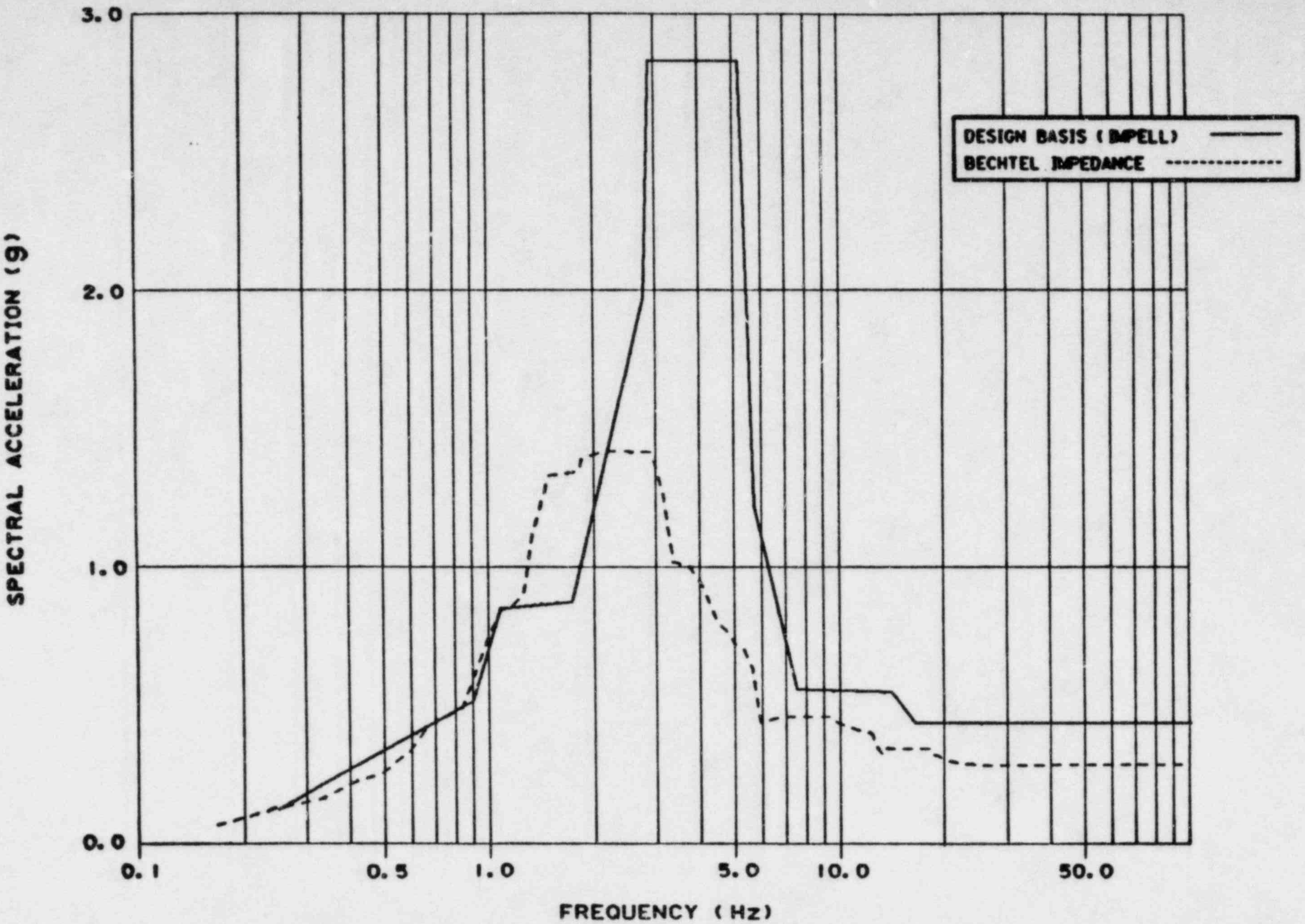

FIGURE 220.21-3

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

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

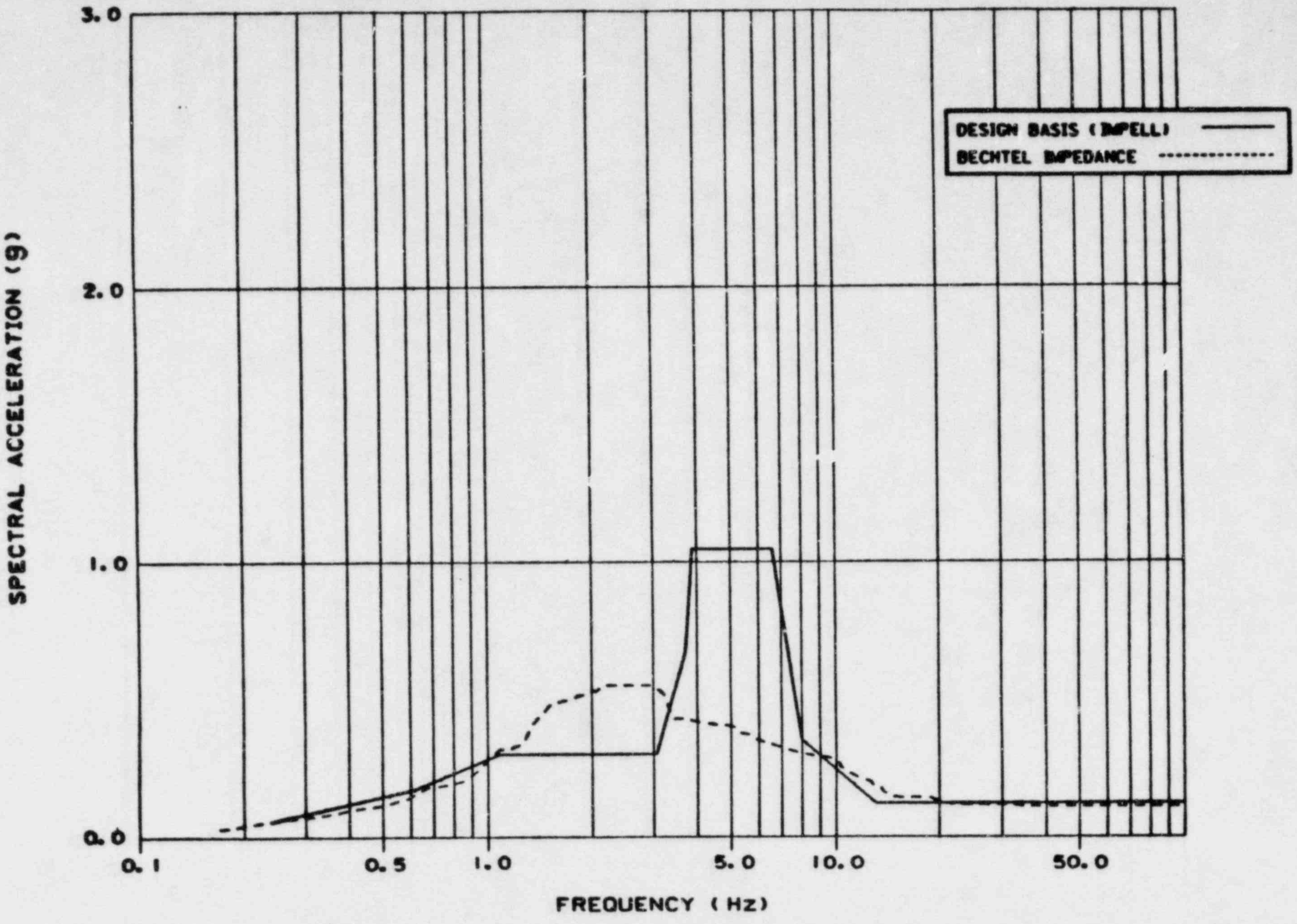

FIGURE 220.21-4

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

HOPE CREEK UNIT I
CAD. L4A SHT 2, REV 0

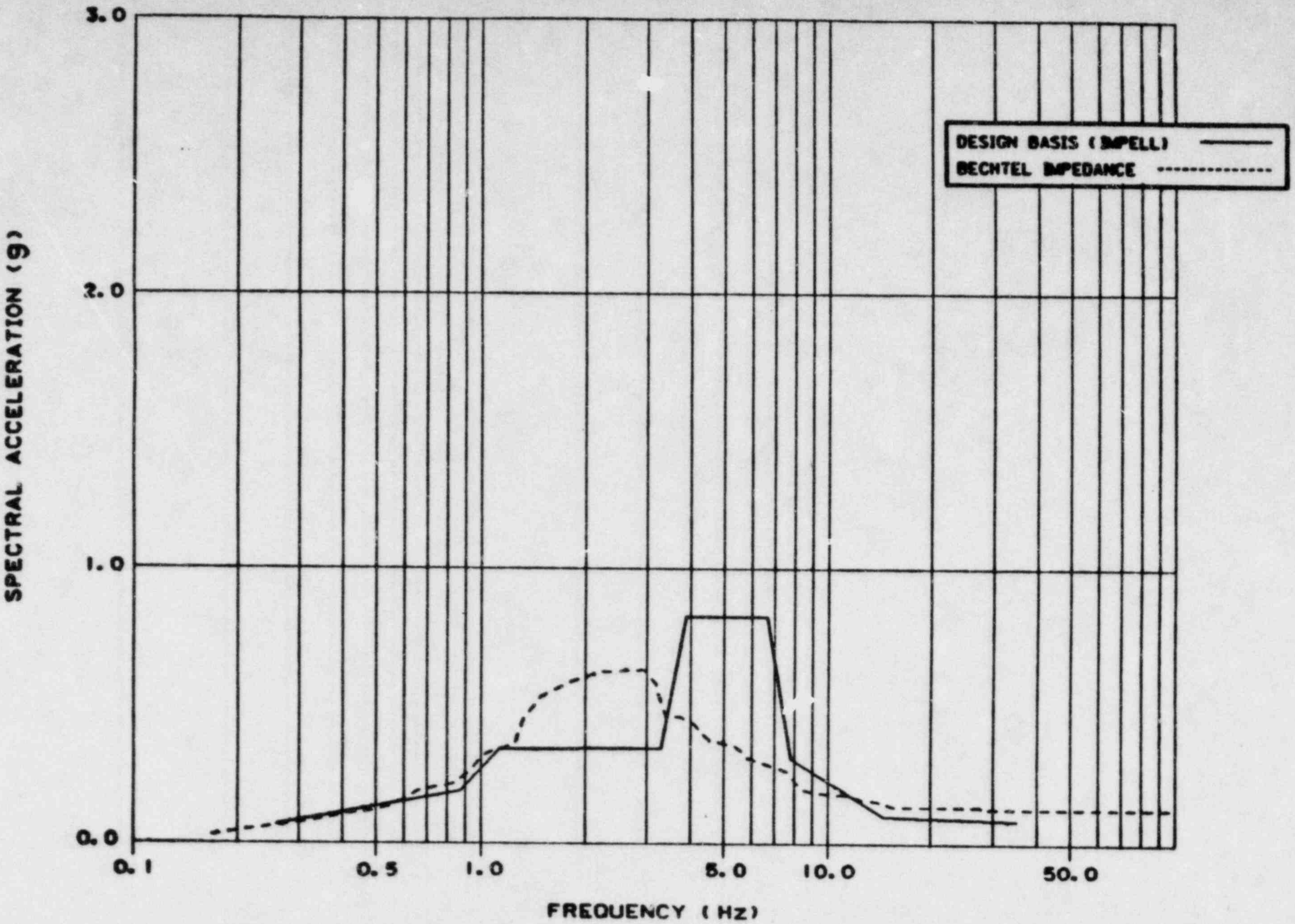

FIGURE 220.21-5

RESPONSE SPECTRA COMPARISON,
UNIT 1 REACTOR BUILDING AT ELEV. 102'-0",
N-S, OBE, 2% DAMPING.

HOPE CREEK UNIT 1
CAD, L5A SHT 2, REV 0
07/27/84

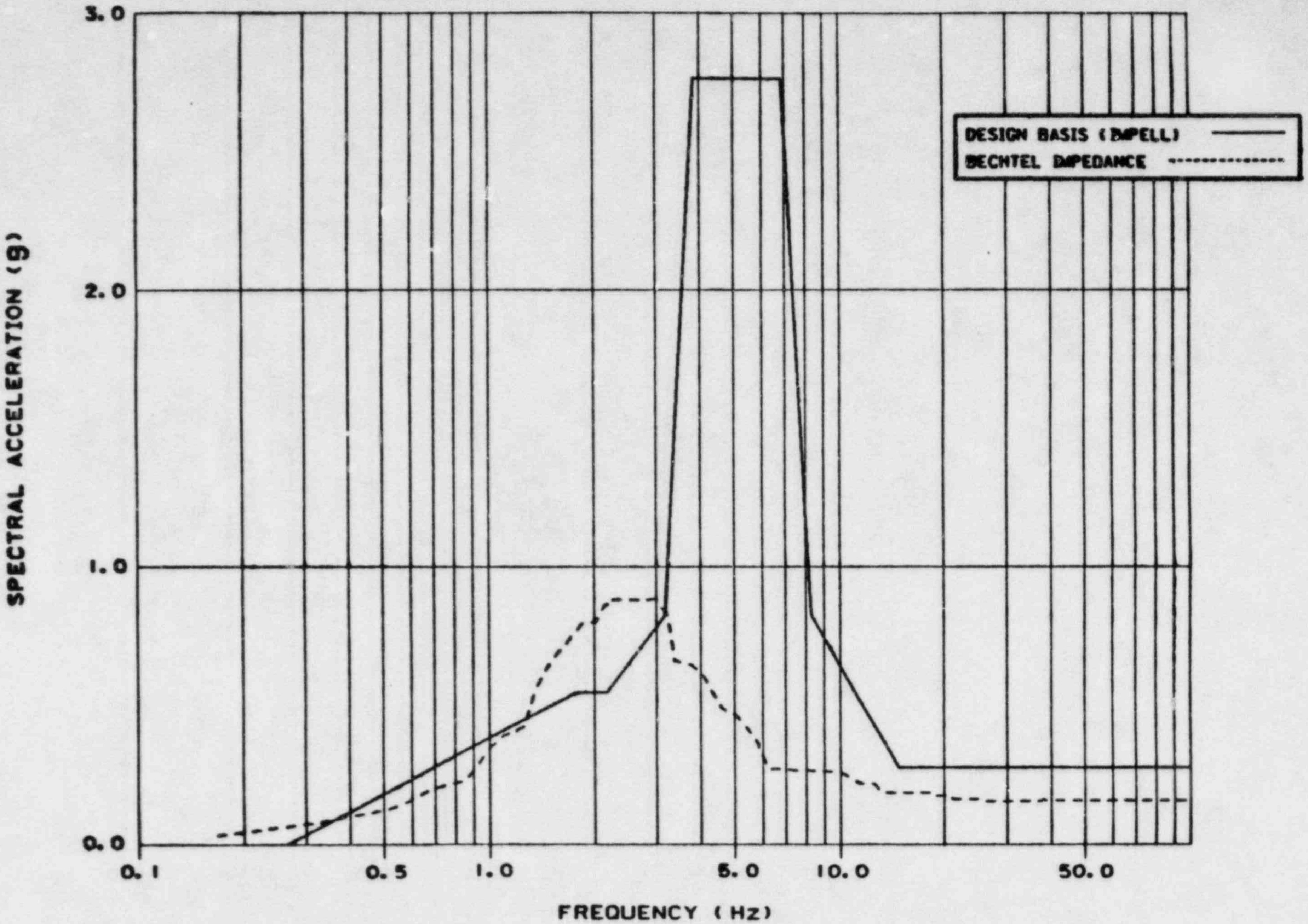

FIGURE 220.21-6

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

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

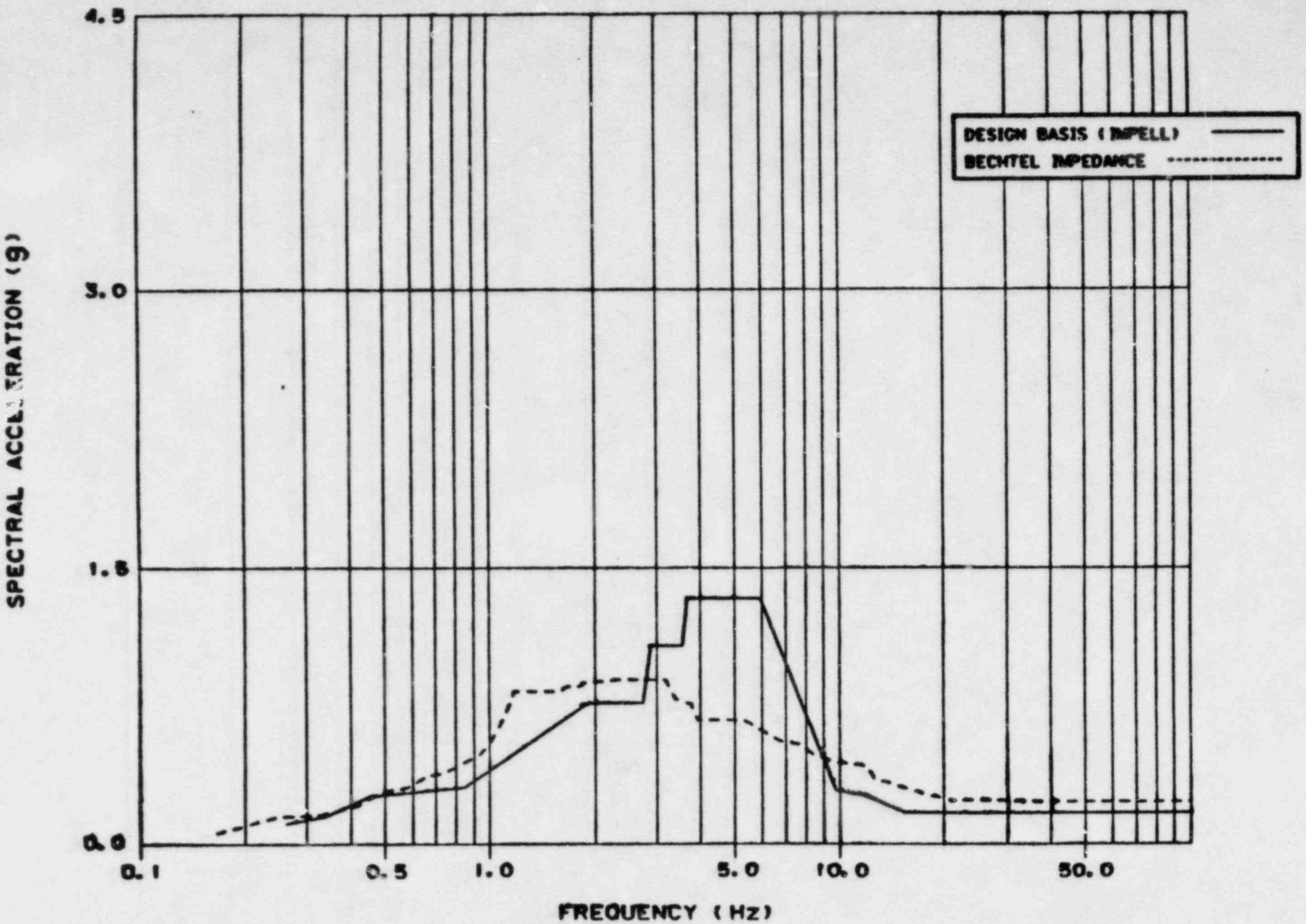

FIGURE 220.21-7

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

HOPE CREEK UNIT I
CAD.L7A SHT 2.REV 0

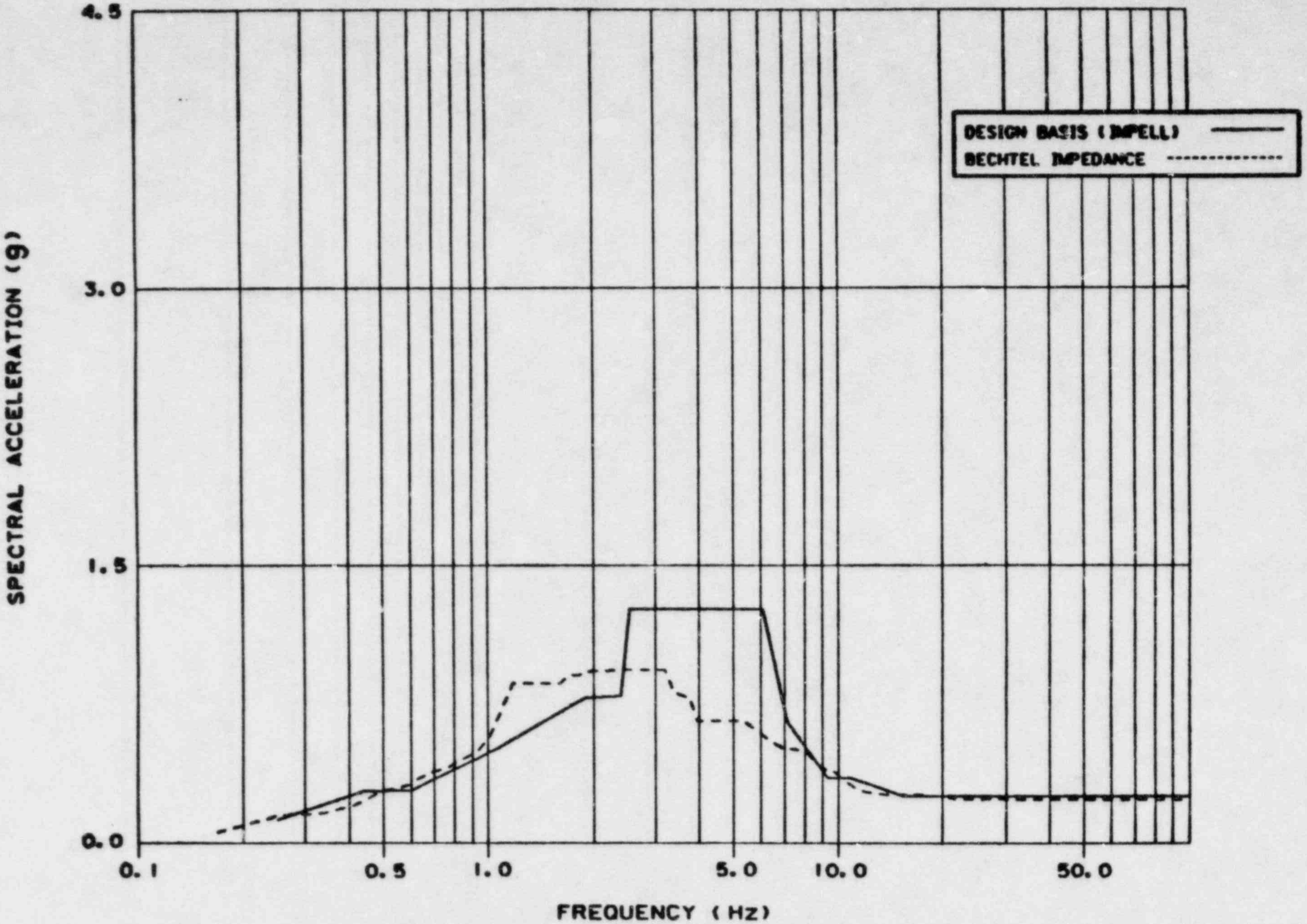

FIGURE 220.21-8

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

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

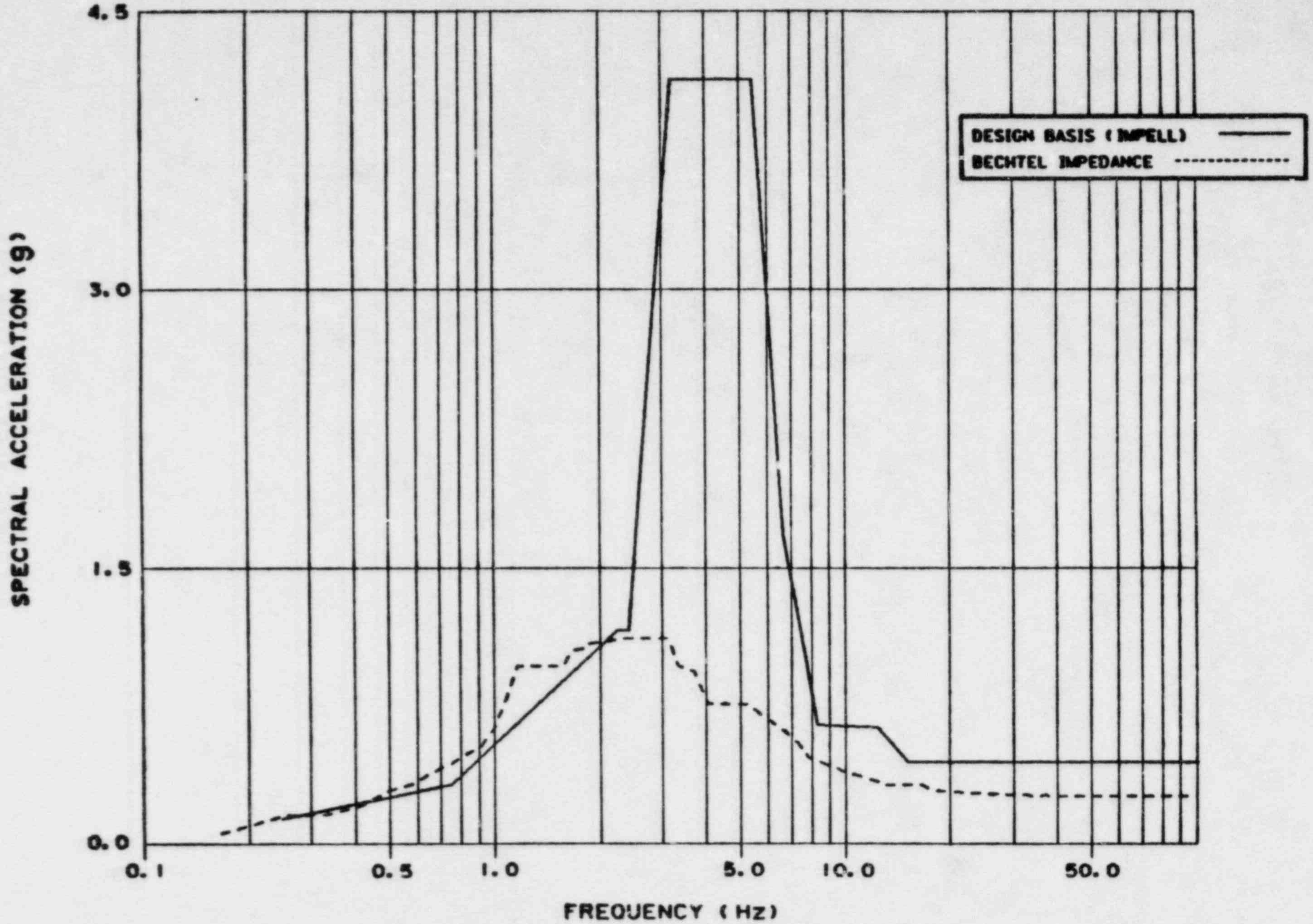

FIGURE 220.21-9

RESPONSE SPECTRA COMPARISON.
UNIT 1 REACTOR BUILDING AT ELEV. 201'-0",
E-W, SSE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L9A SHT 2, REV 0

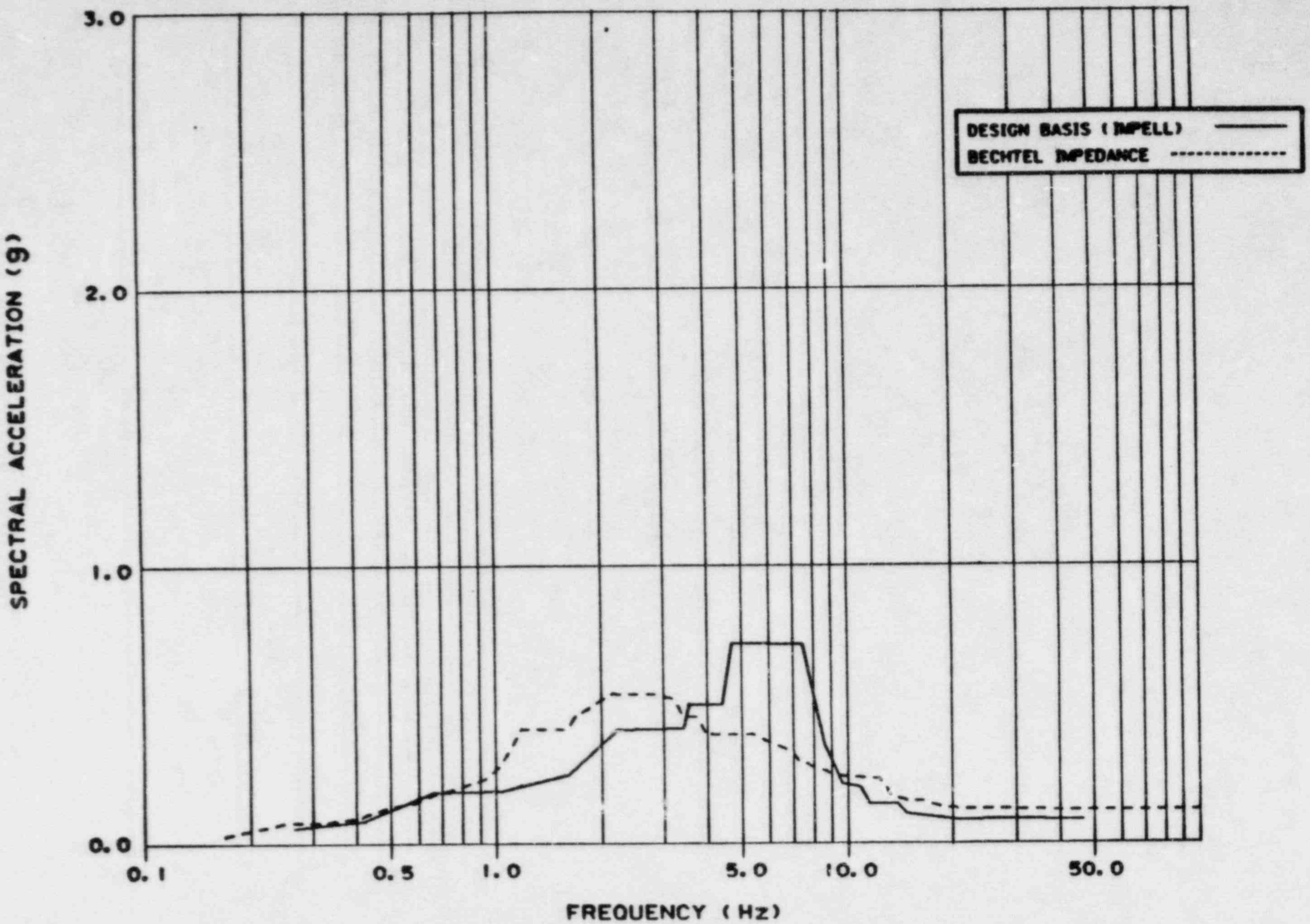

FIGURE 220.21-10

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

HOPE CREEK UNIT I
CAD, L10A SHT 2, REV 0
07/27/84

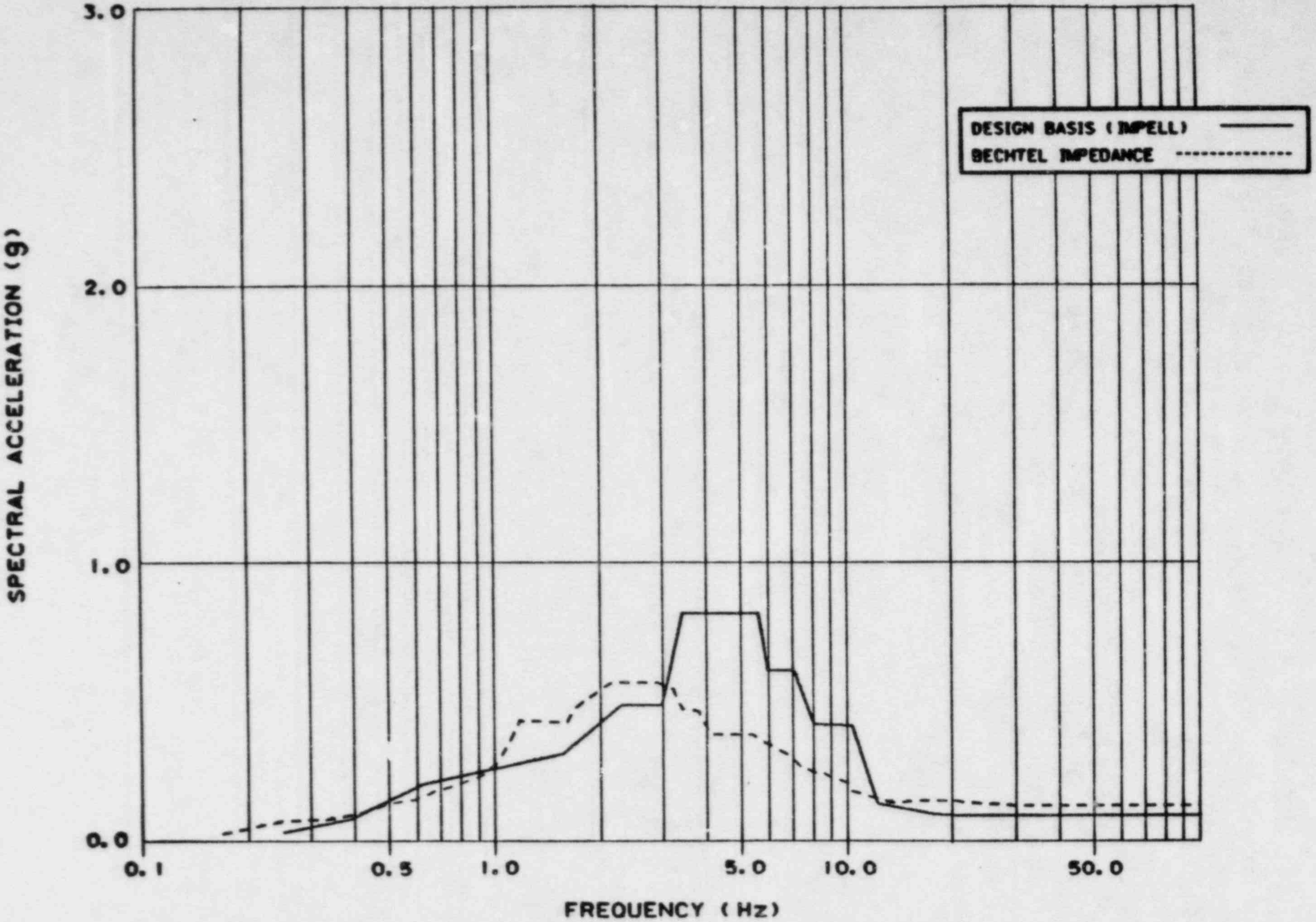

FIGURE 220.21-11

RESPONSE SPECTRA COMPARISON,
UNIT 1 REACTOR BUILDING AT ELEV. 102' -0",
E-W, OBE, 2% DAMPING

HOPE CREEK UNIT 1
CAD, L11A SHT 2, REV 0
07/27/84

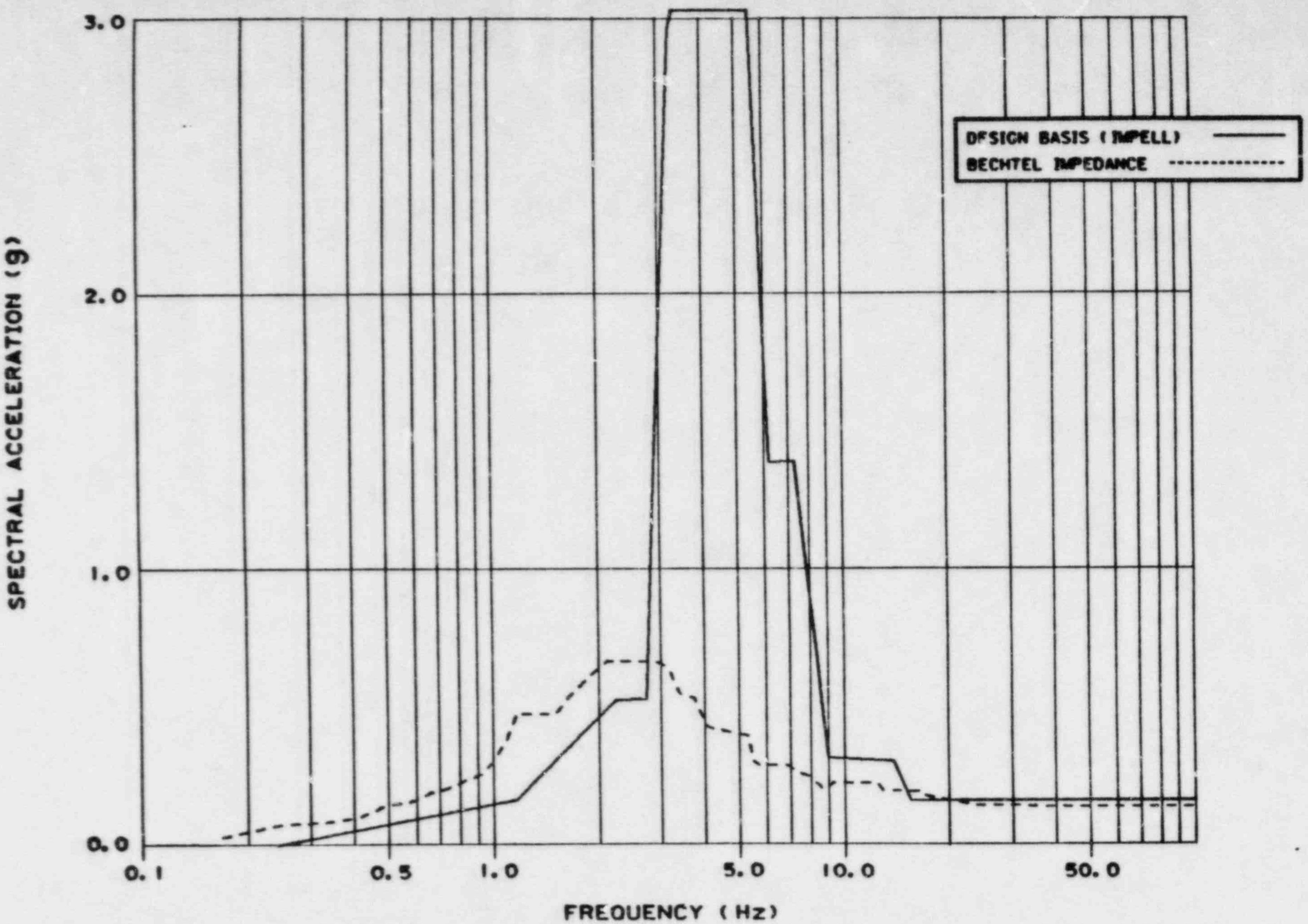

FIGURE 220.21-12

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

HOPE CREEK UNIT 1
CAD, L12A SHT 2, REV 0
07/27/84

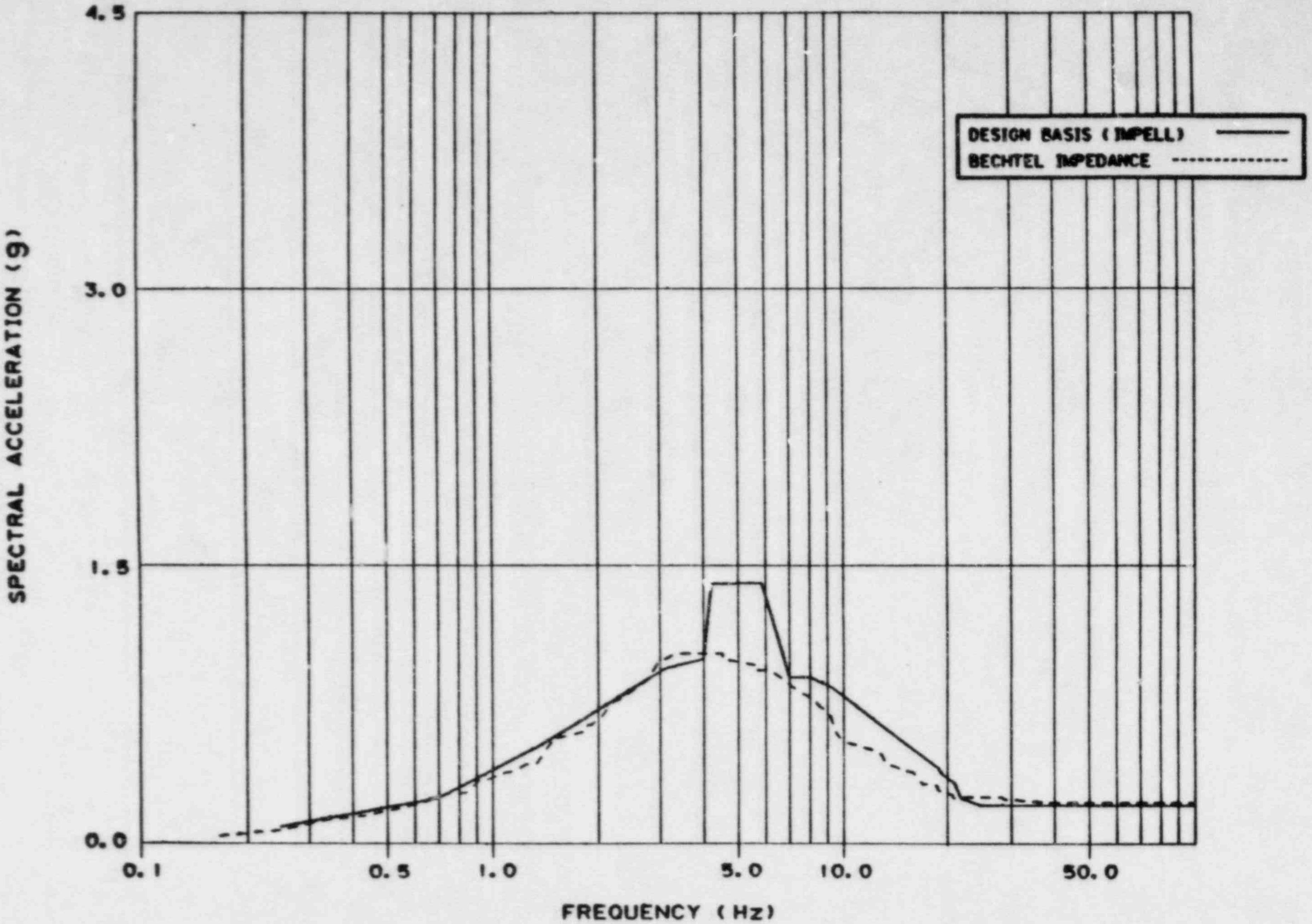

FIGURE 220.21-13

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

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

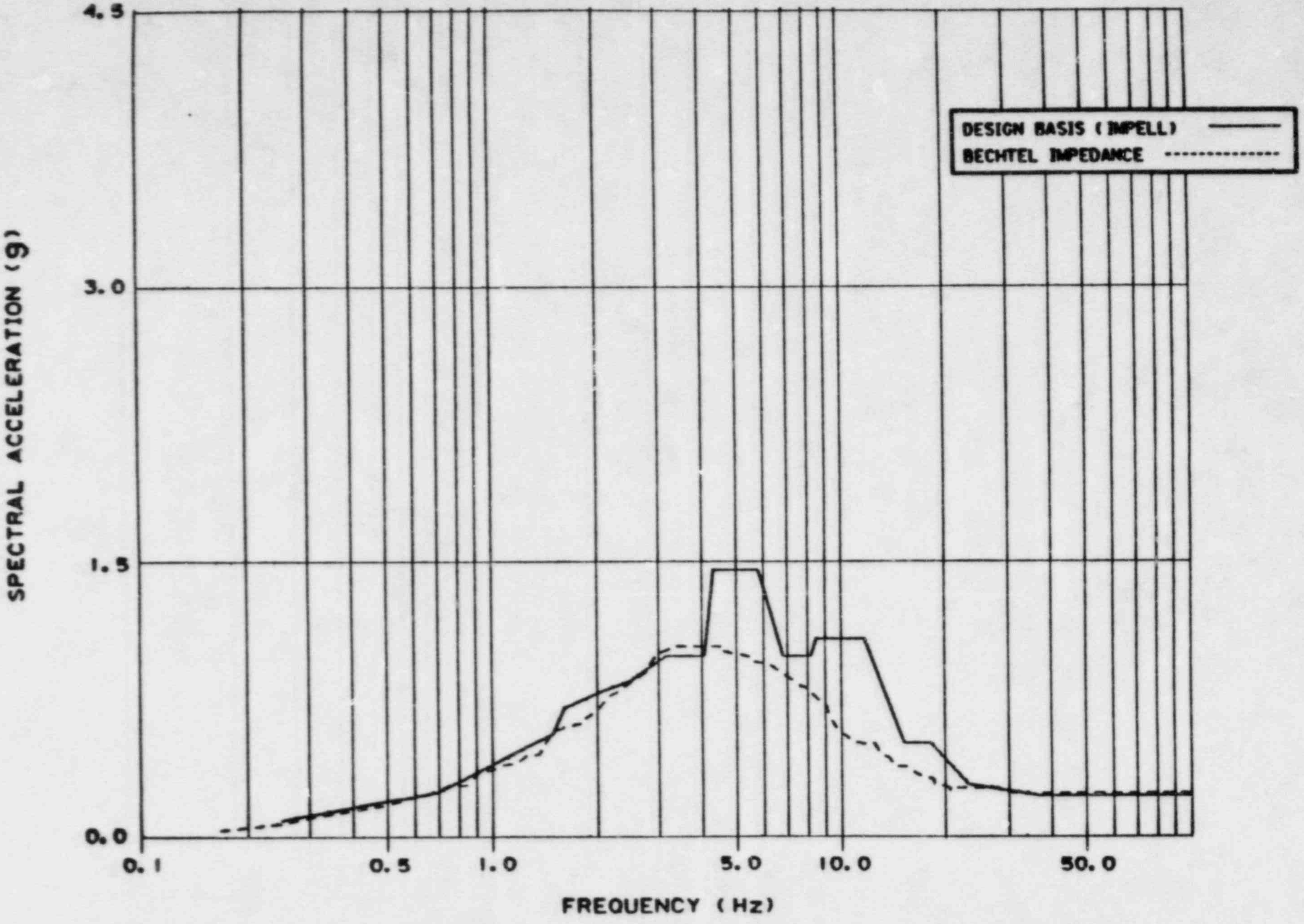

FIGURE 220.21-14

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

HOPE CREEK UNIT 1
CAD. L26A SHT 2, REV 0
07/27/84

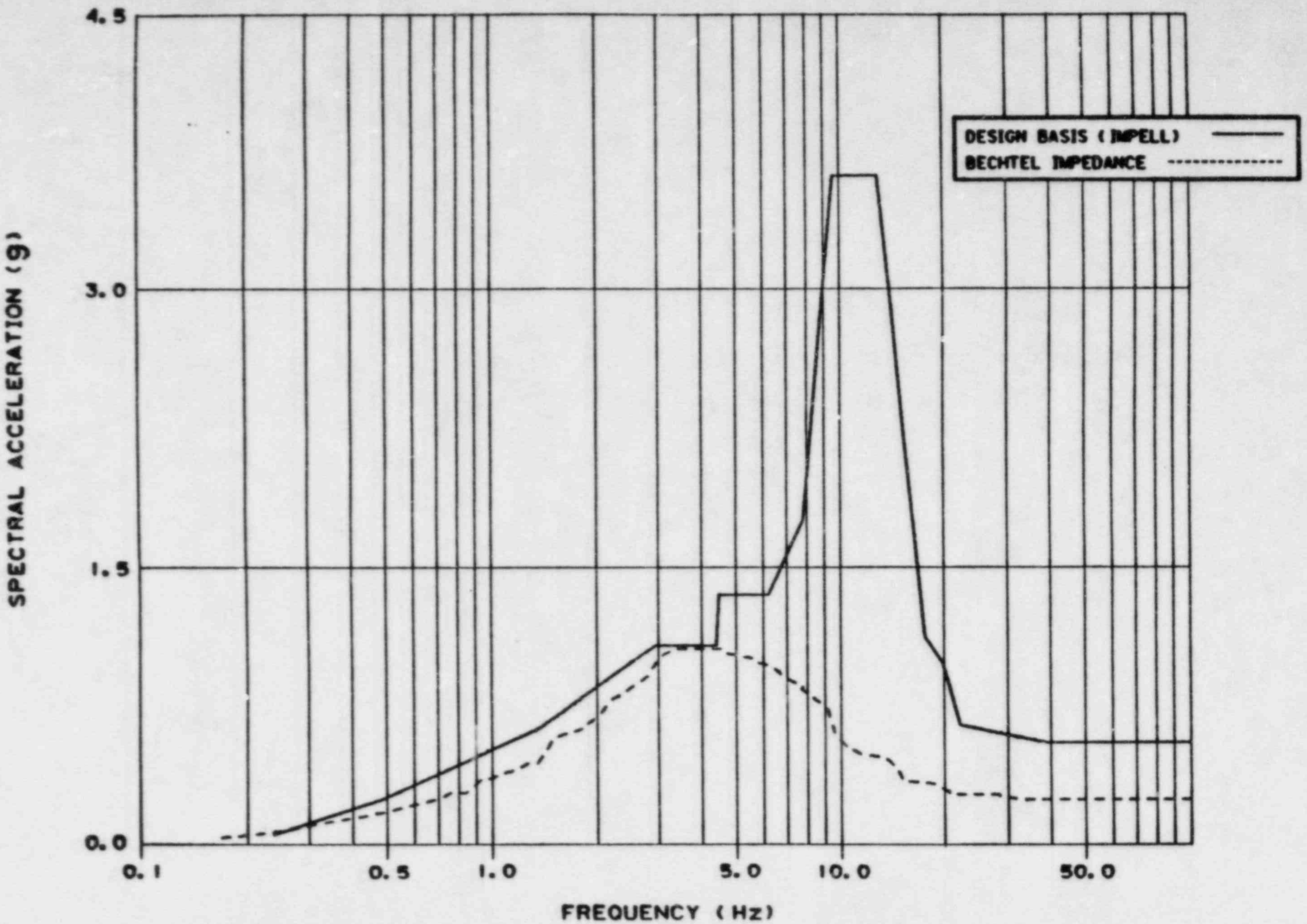

FIGURE 220.21-15

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

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

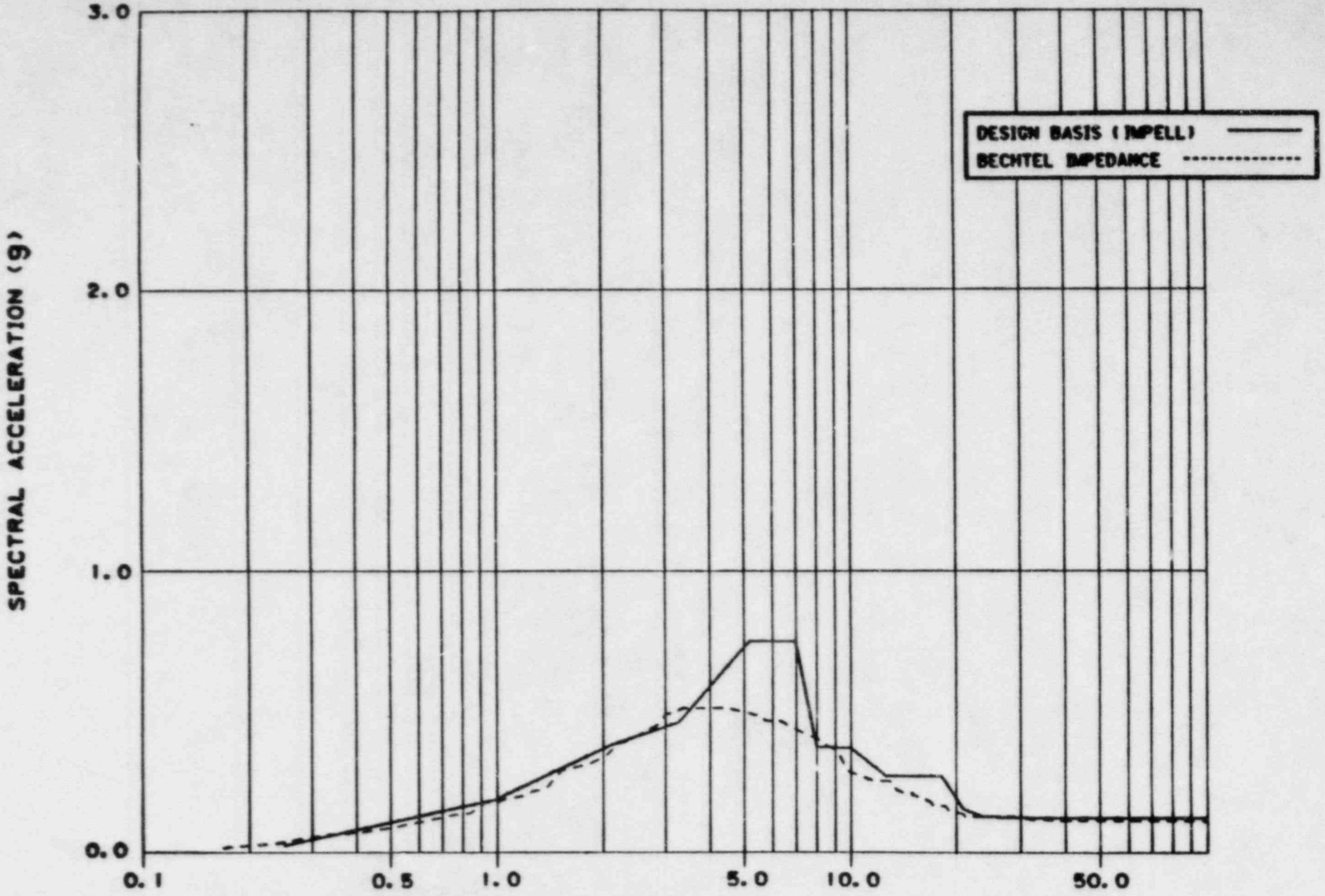

FIGURE 220.21-16

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

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

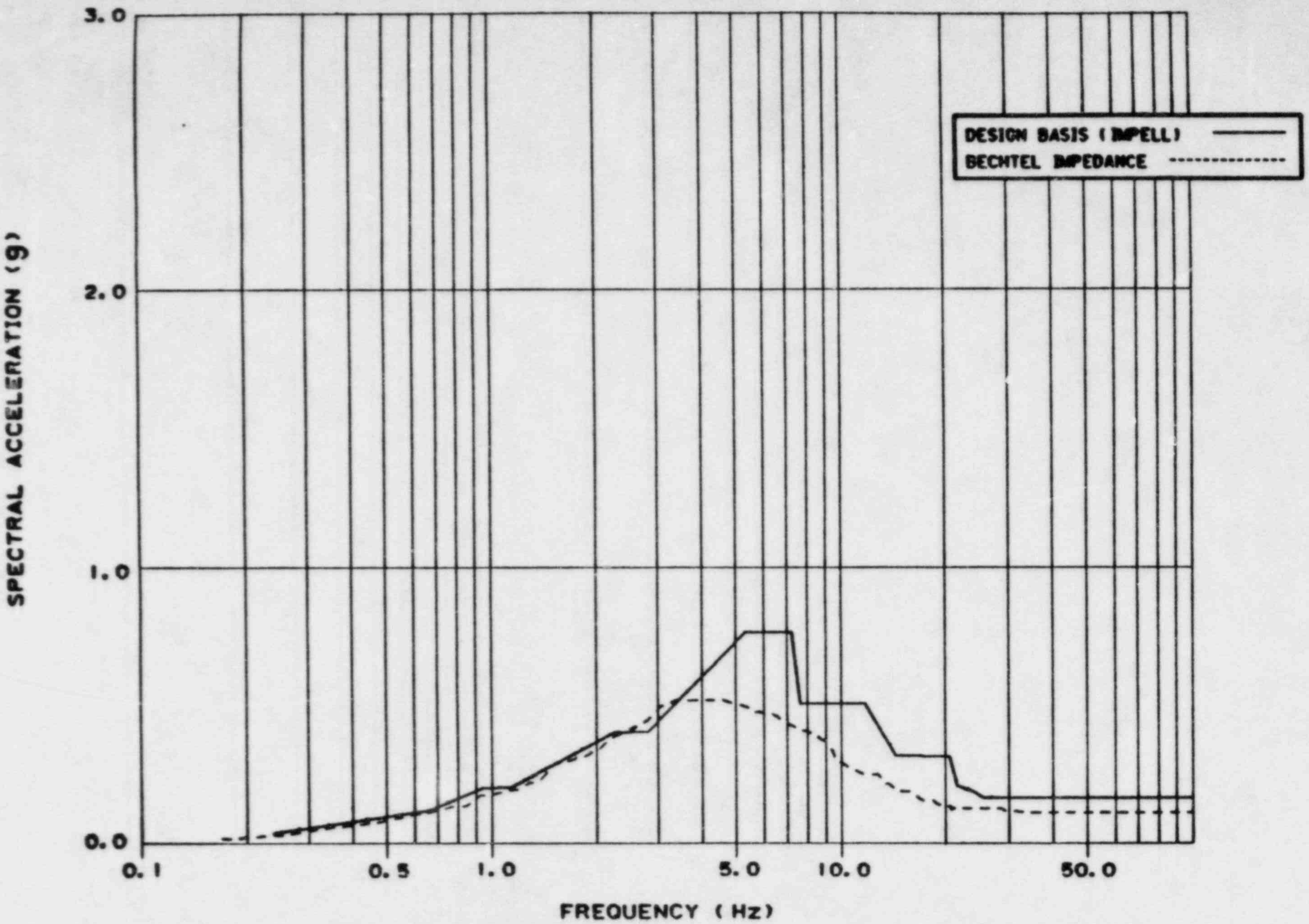

FIGURE 220.21-17

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

HOPE CREEK UNIT 1
CAD, L29A SHT 2, REV 1
07/27/84

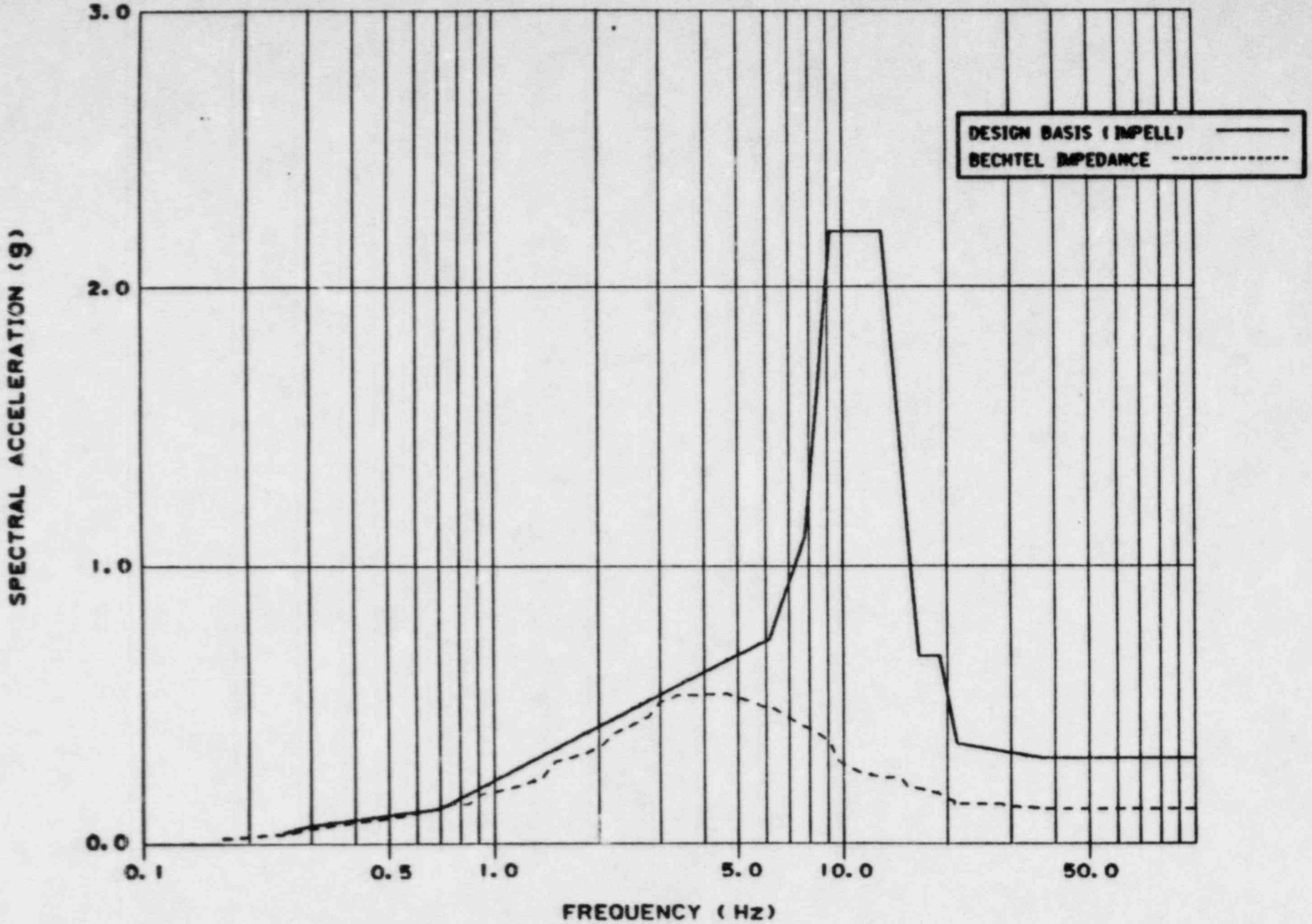

FIGURE 220.21-18

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

HOPE CREEK UNIT I
CAD, L30A SHT 2, REV I
07/27/84

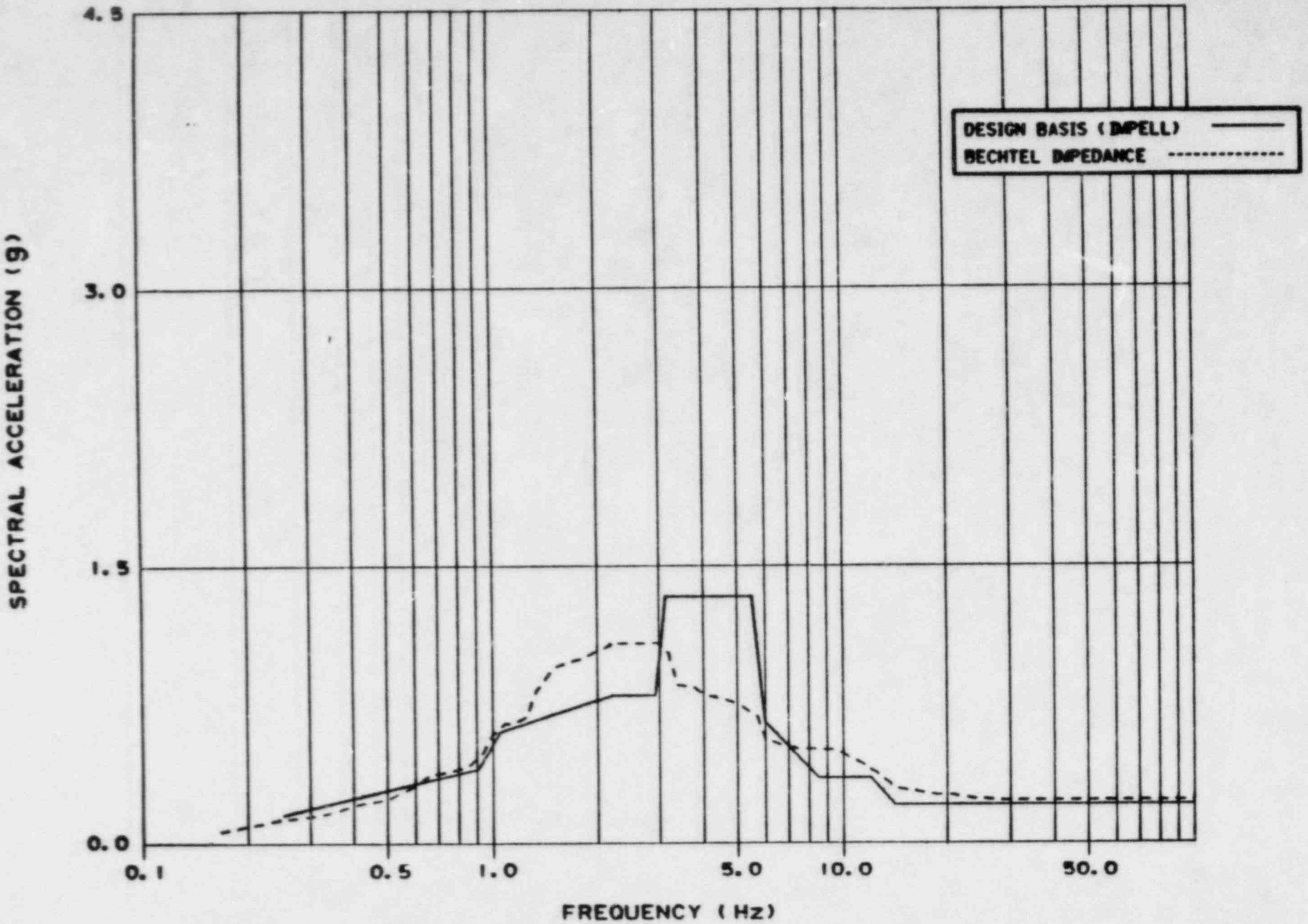

FIGURE 220.21-19

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

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

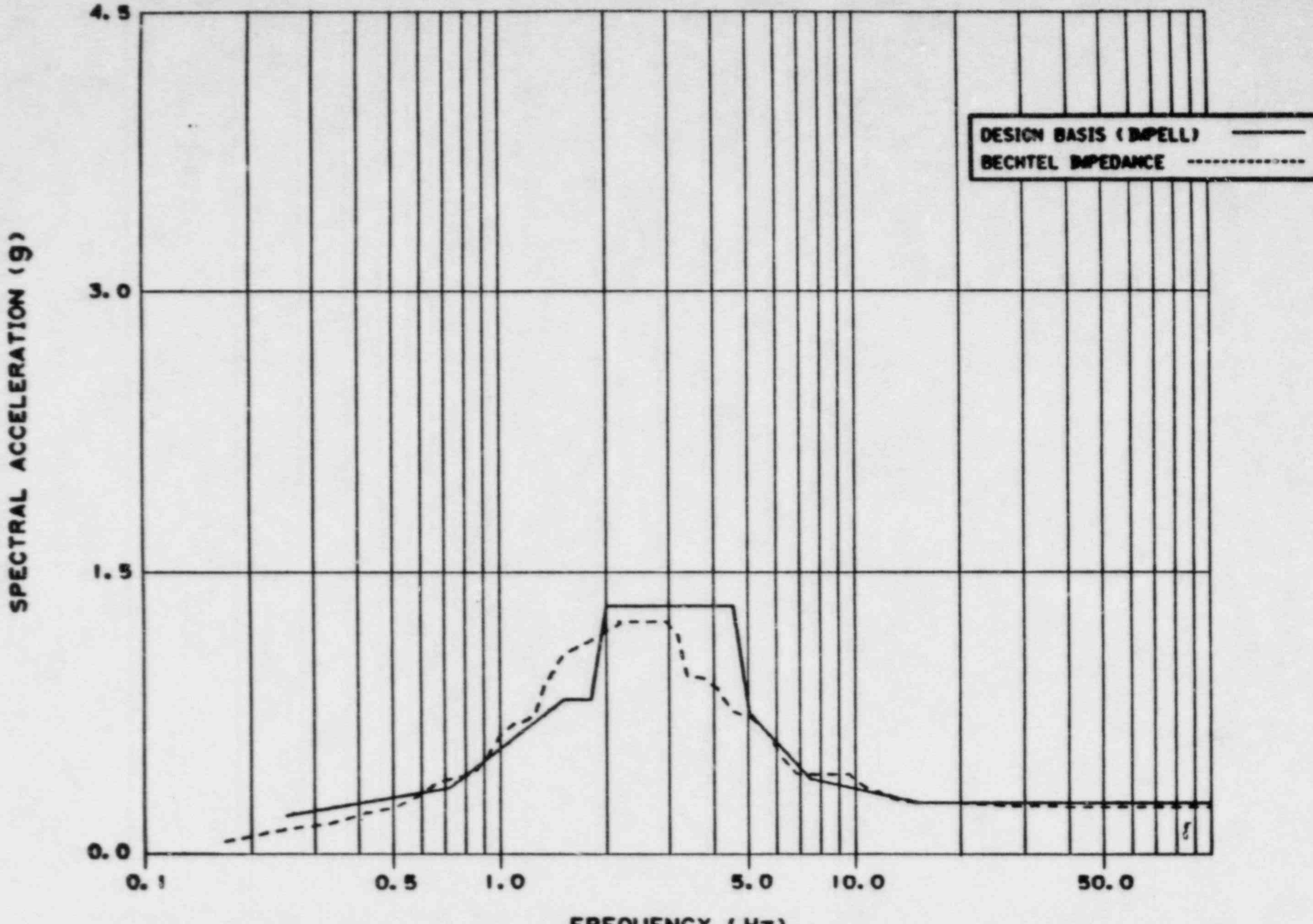

FIGURE 220.21-20

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

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

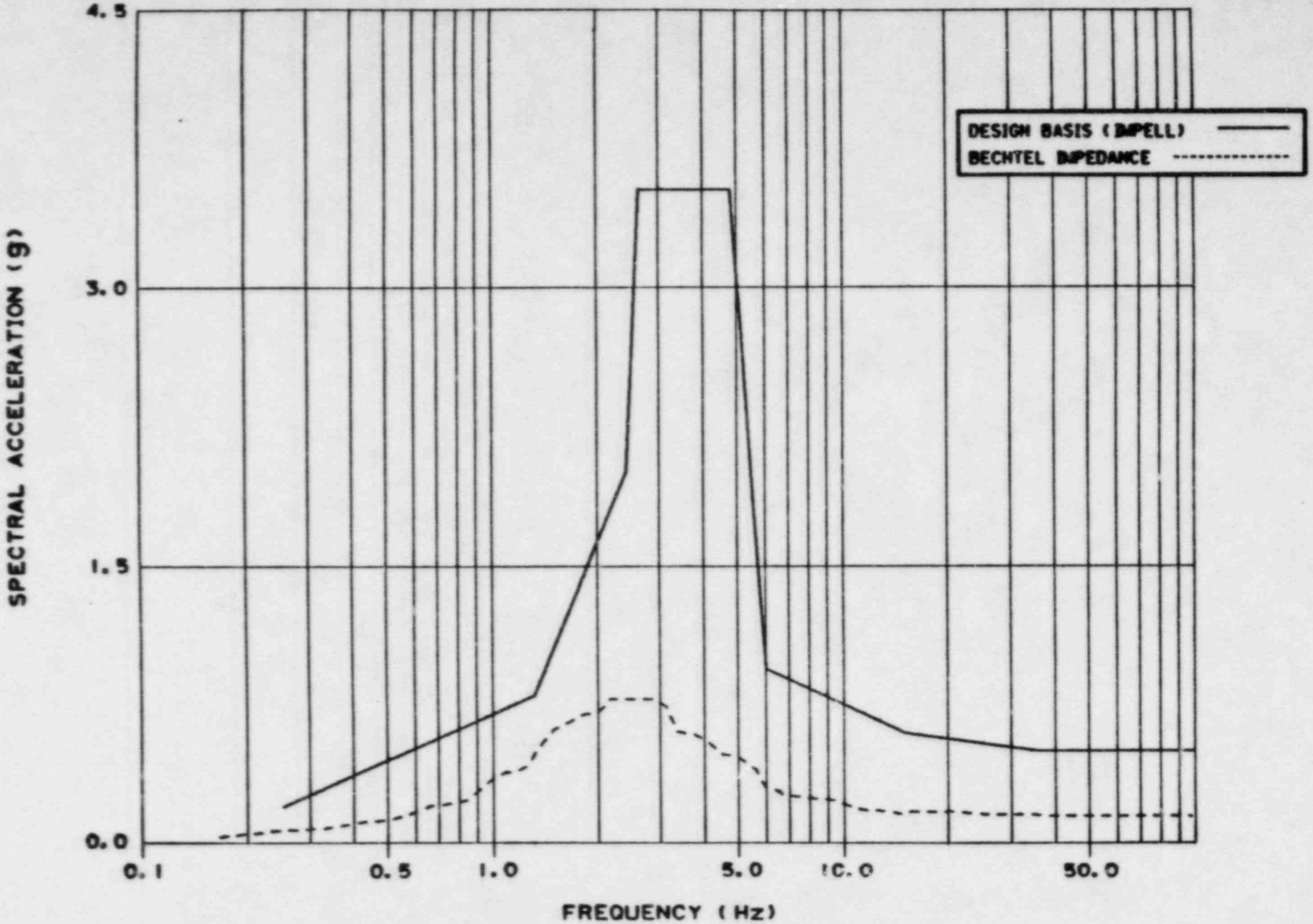

FIGURE 220.21-21

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

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

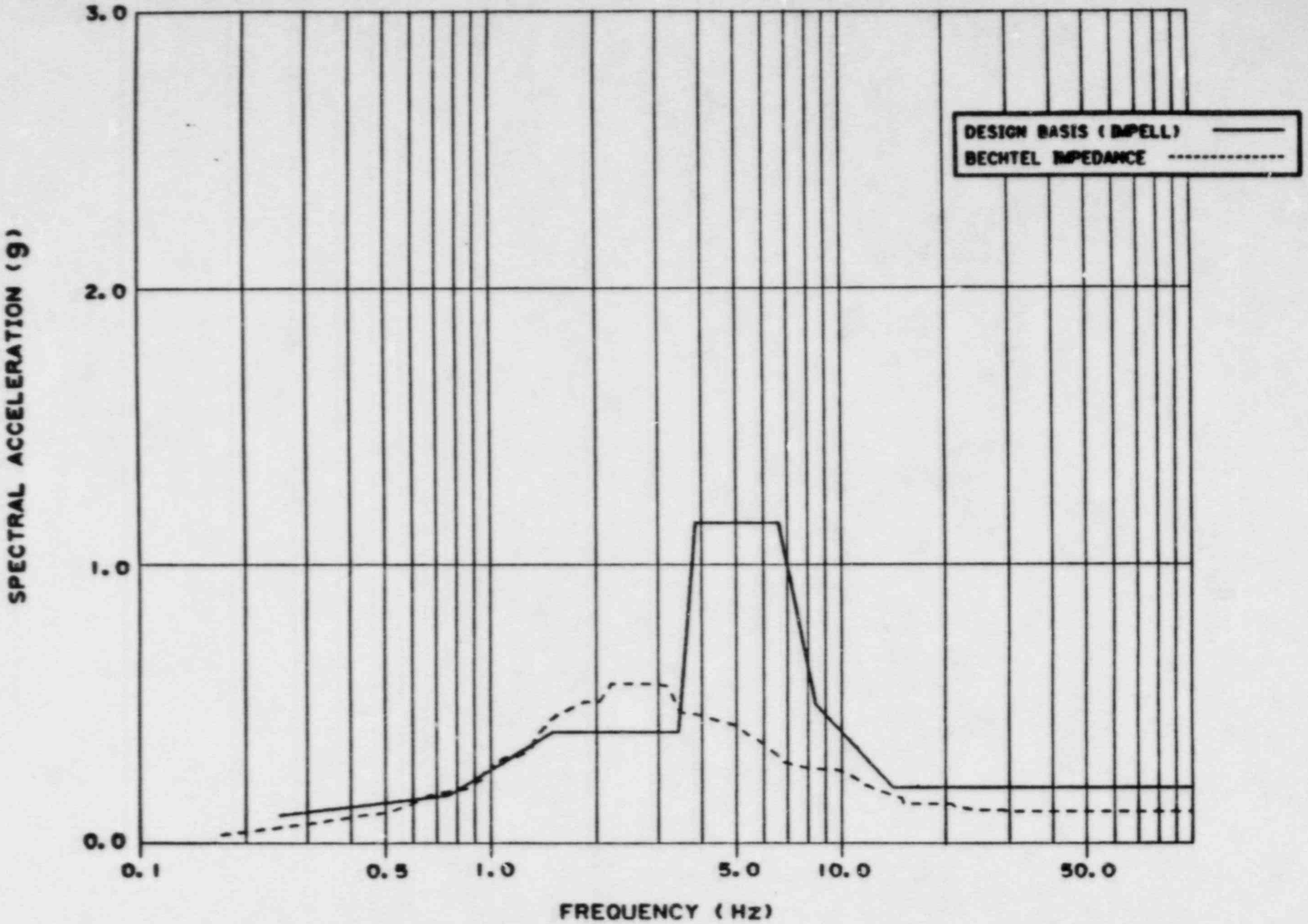

FIGURE 220.21-22

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

HOPE CREEK UNIT 1
CAD, L16A SHT 2, REV 0
07/27/84

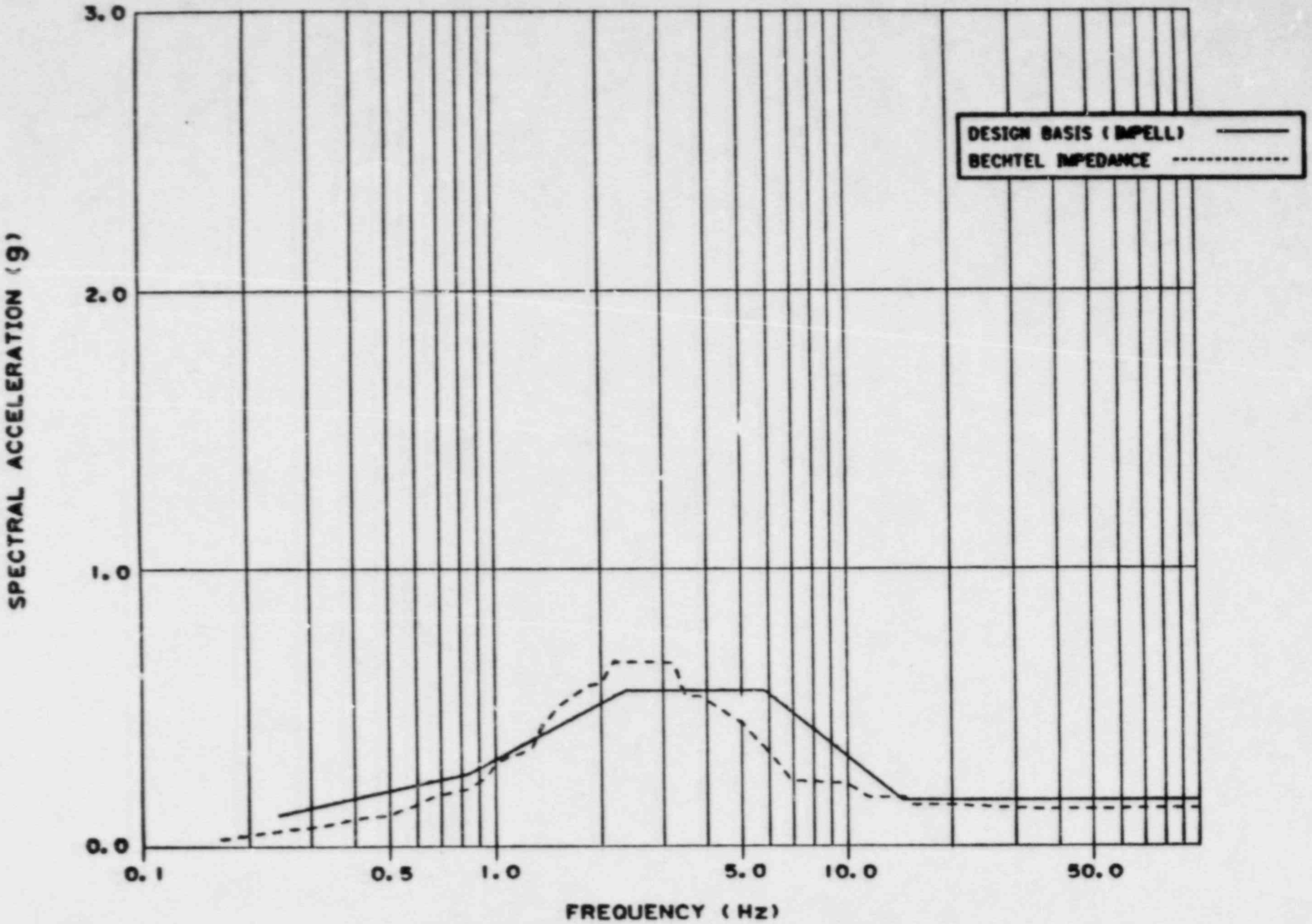

FIGURE 220.21-23

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

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

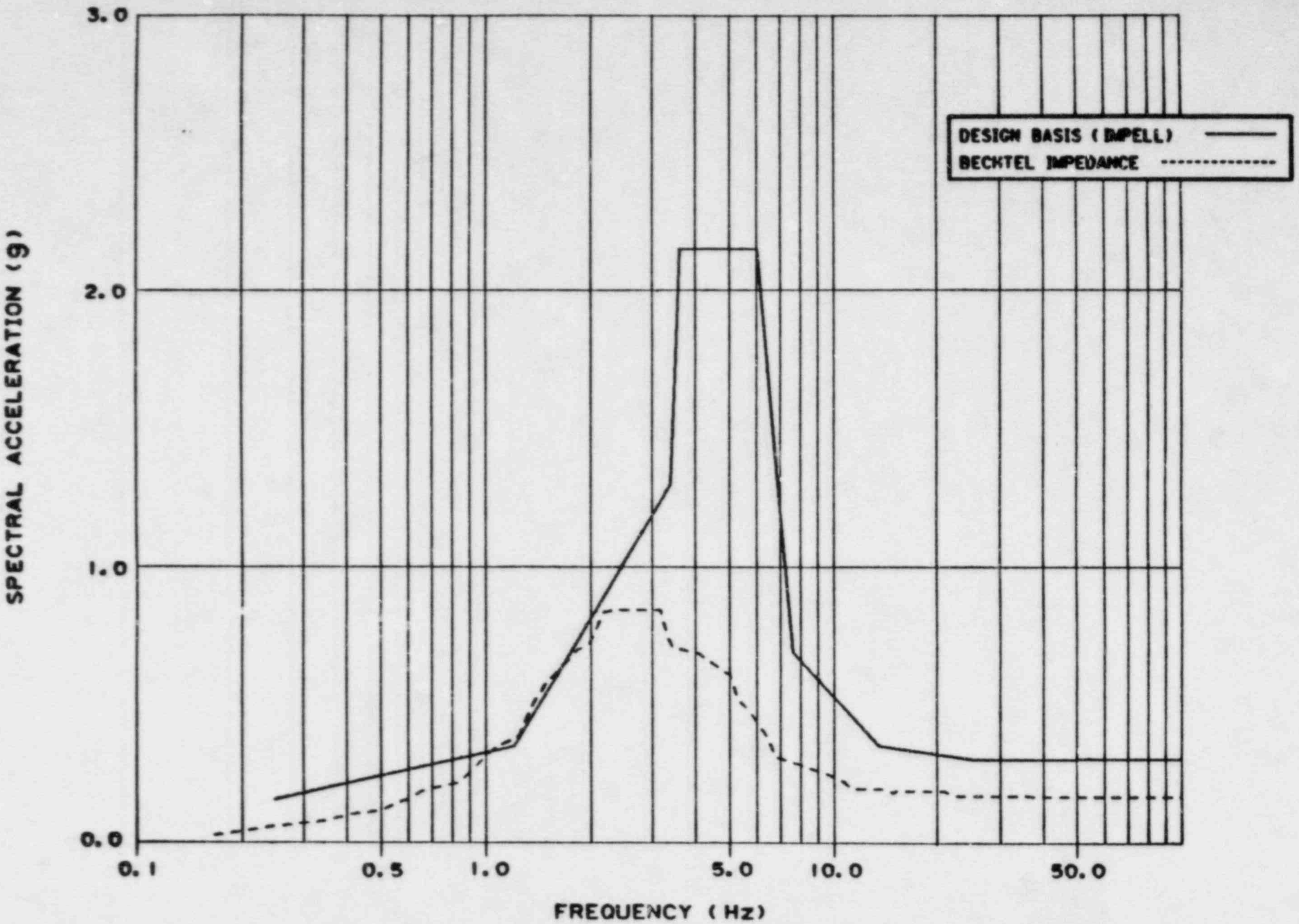

FIGURE 220.21-24

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

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

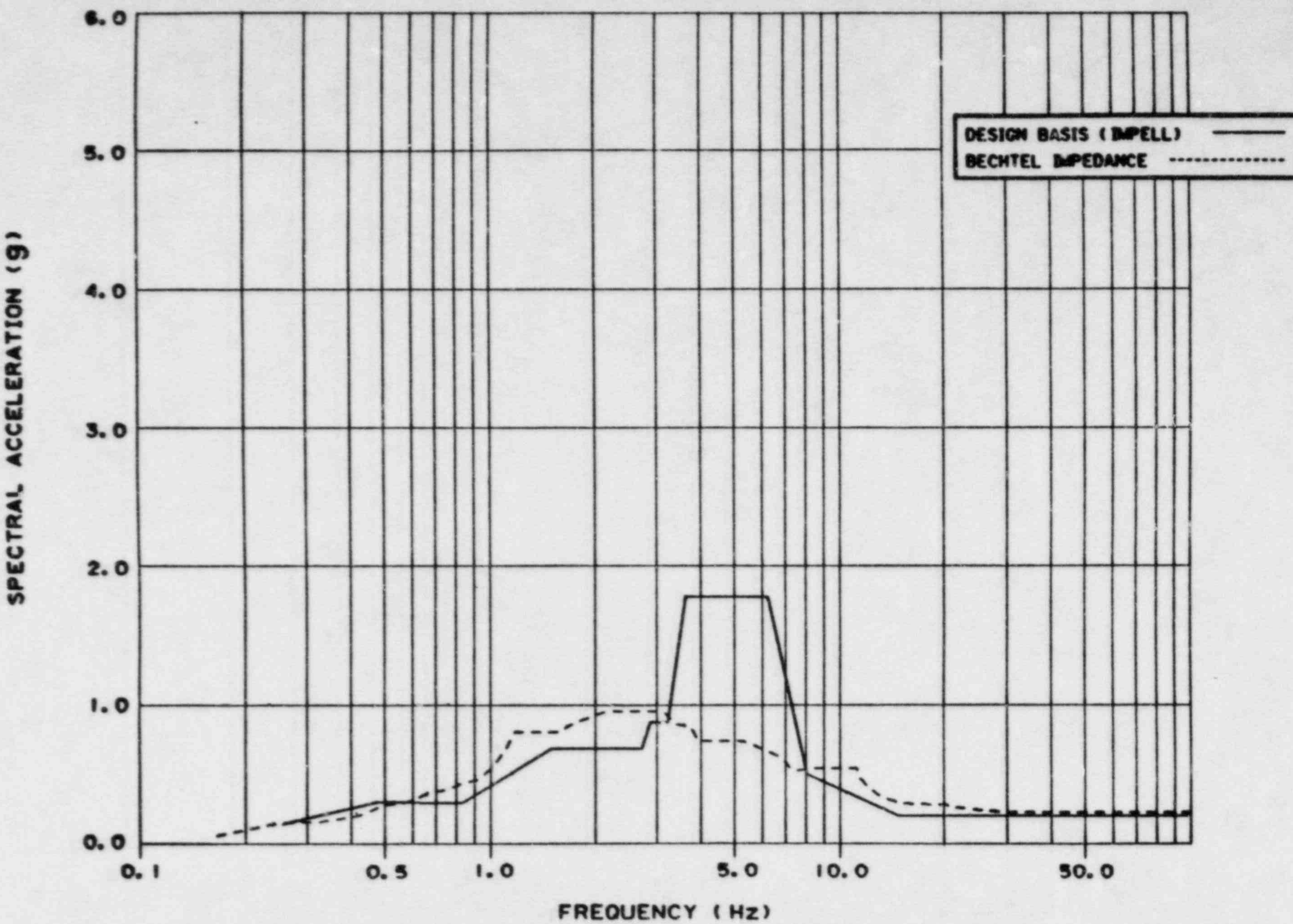

FIGURE 220.21-25

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

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

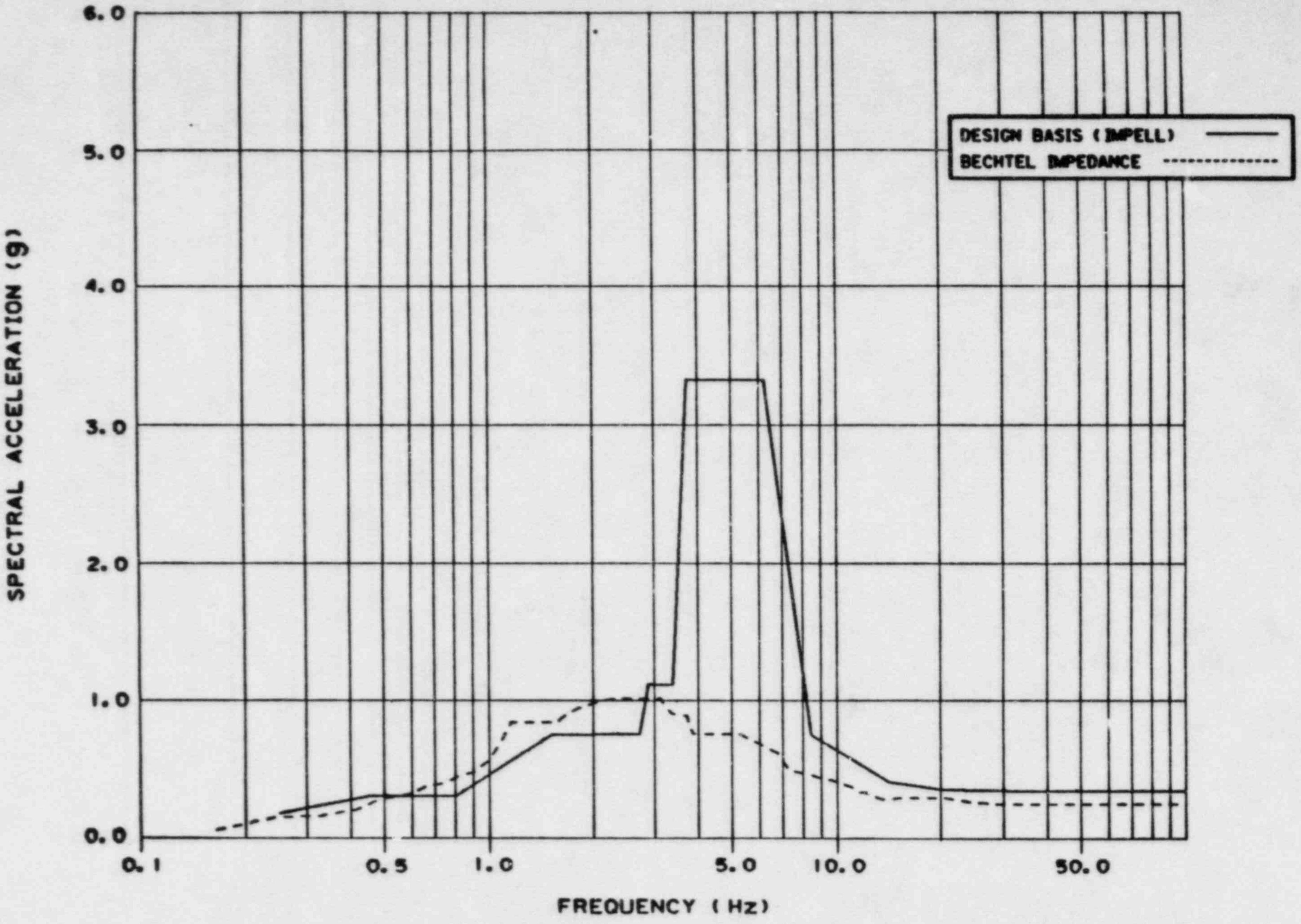

FIGURE 220.21-26

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

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

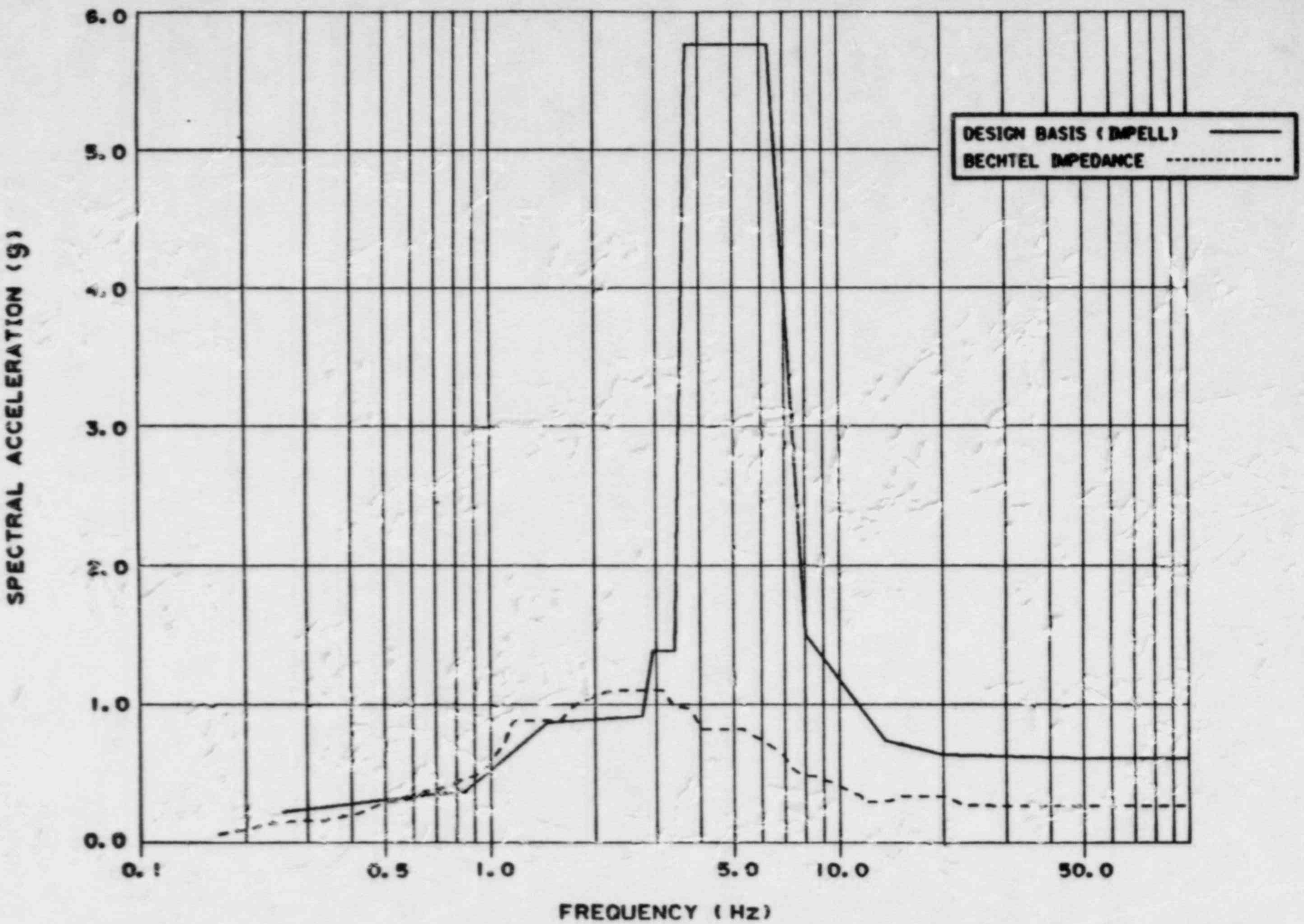

FIGURE 220.21-27

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

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

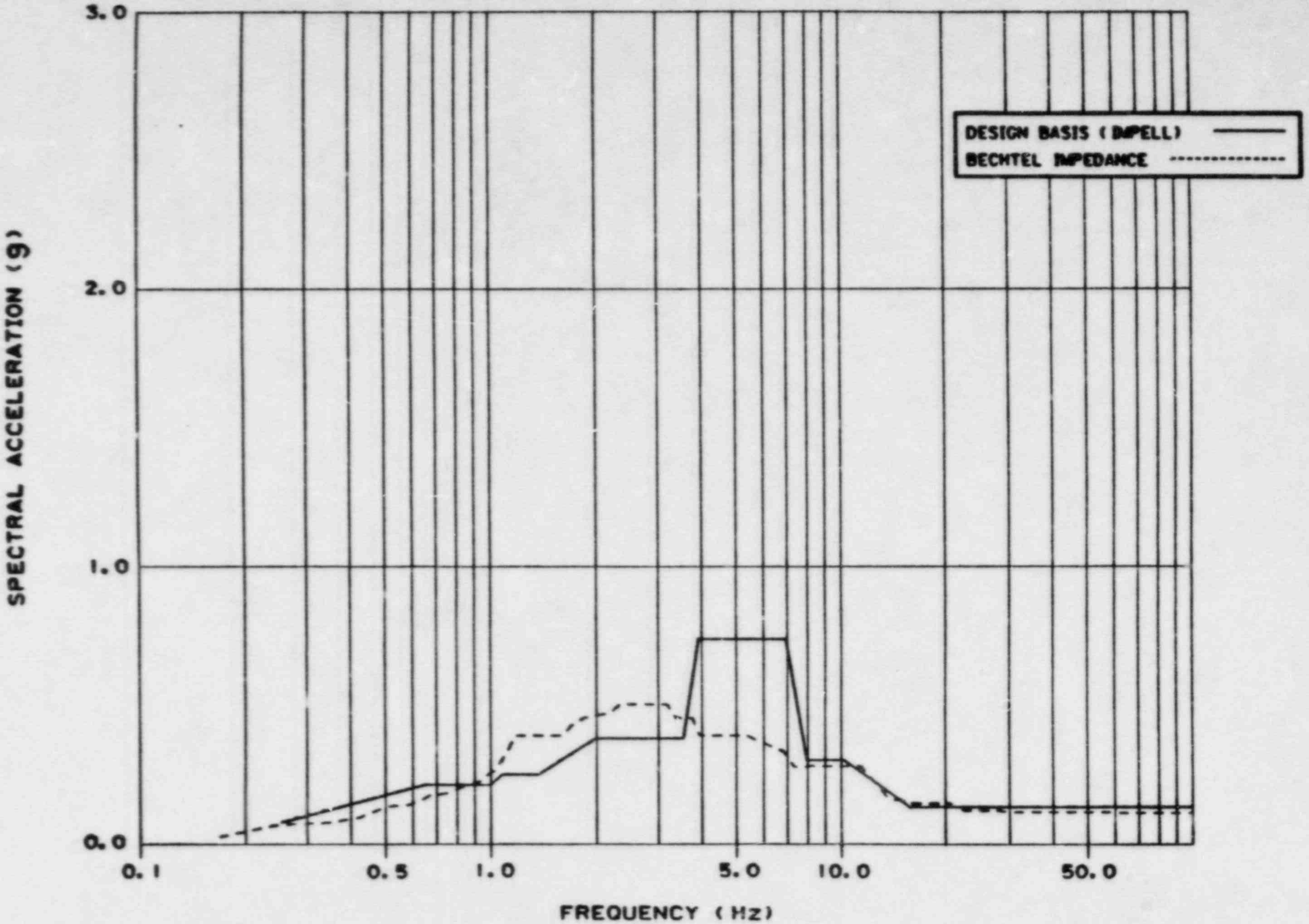

FIGURE 220.21-28

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

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

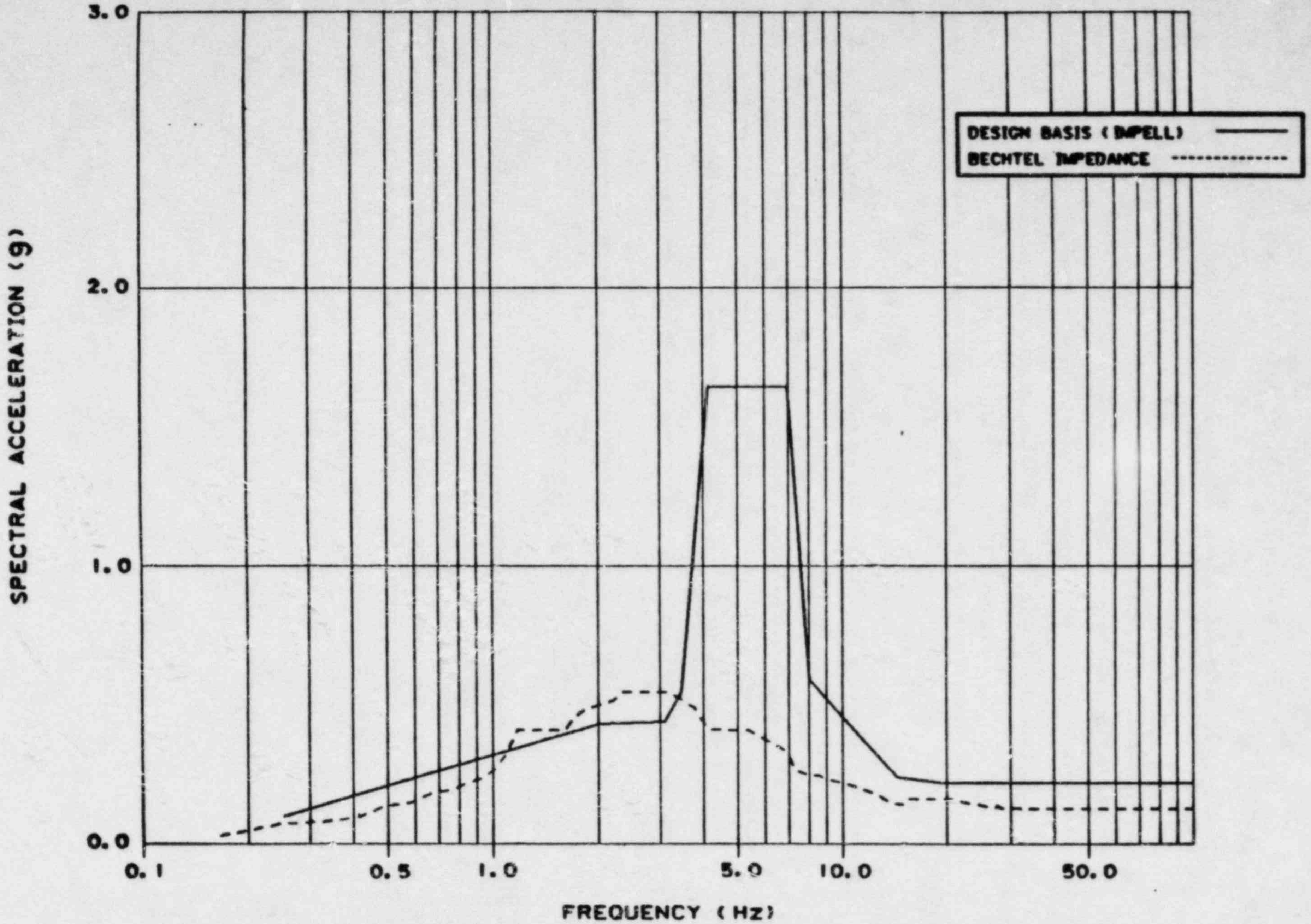

FIGURE 220.21-29

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

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

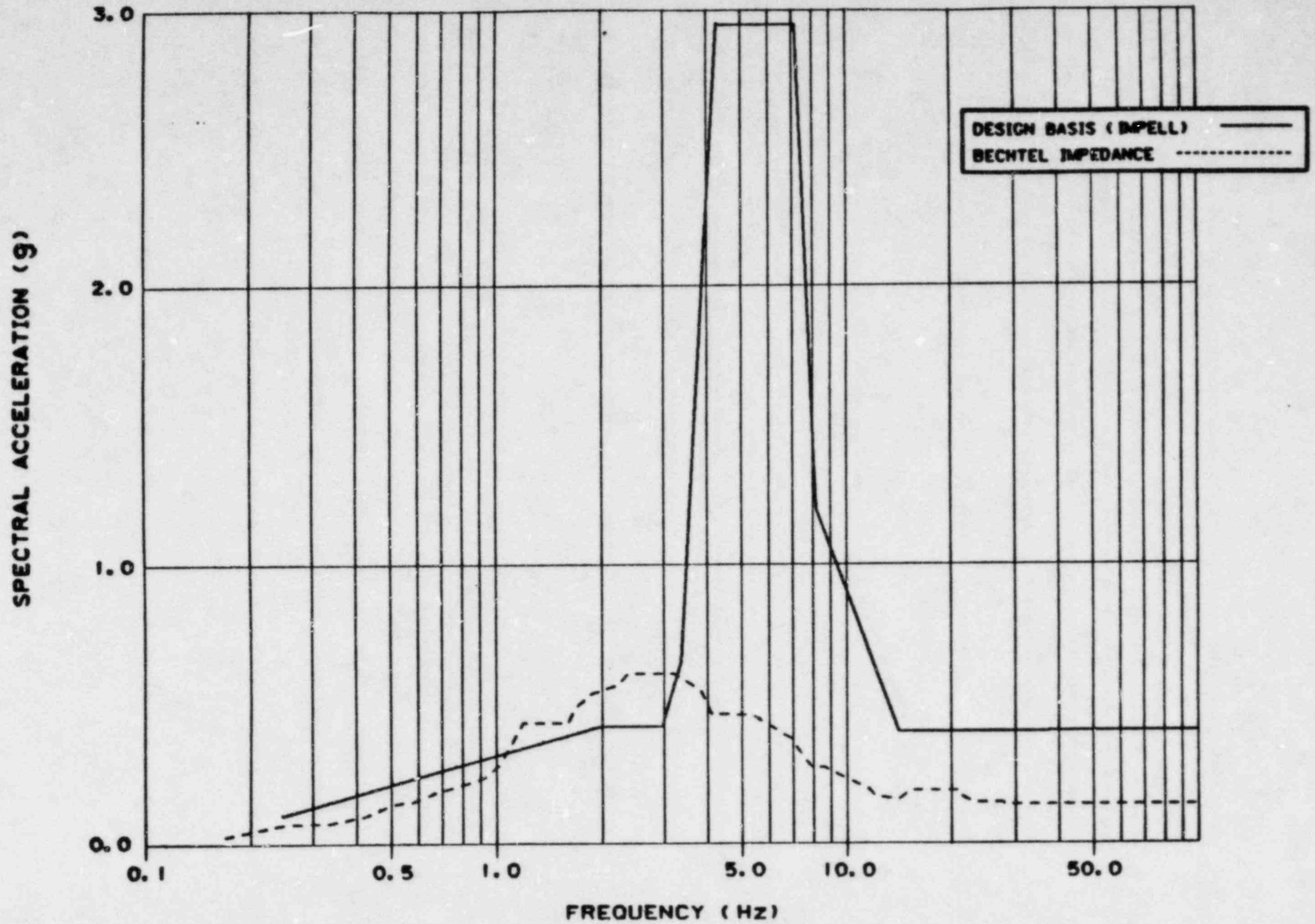

FIGURE 220.21-30

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

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

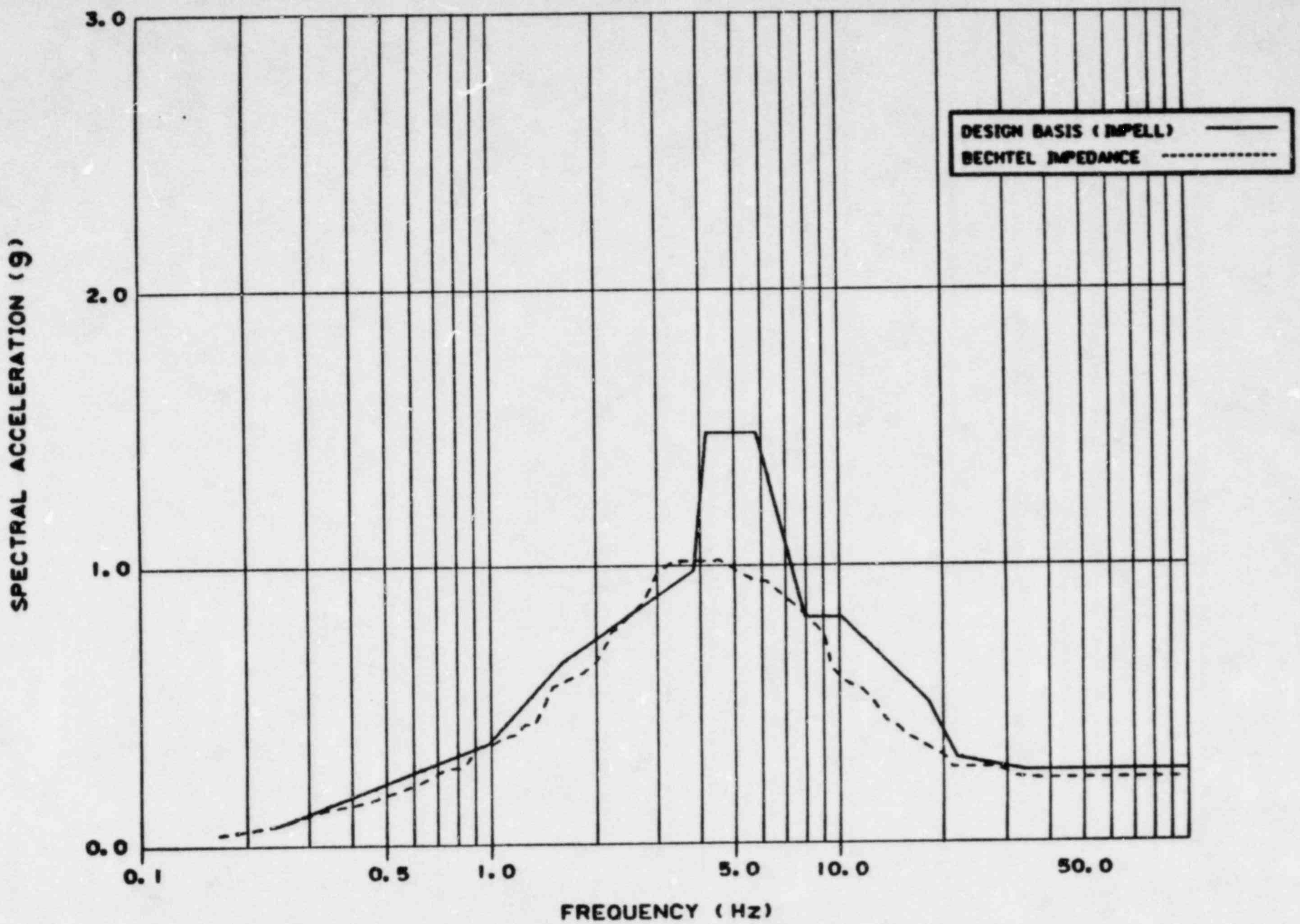

FIGURE 220.21-31

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

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

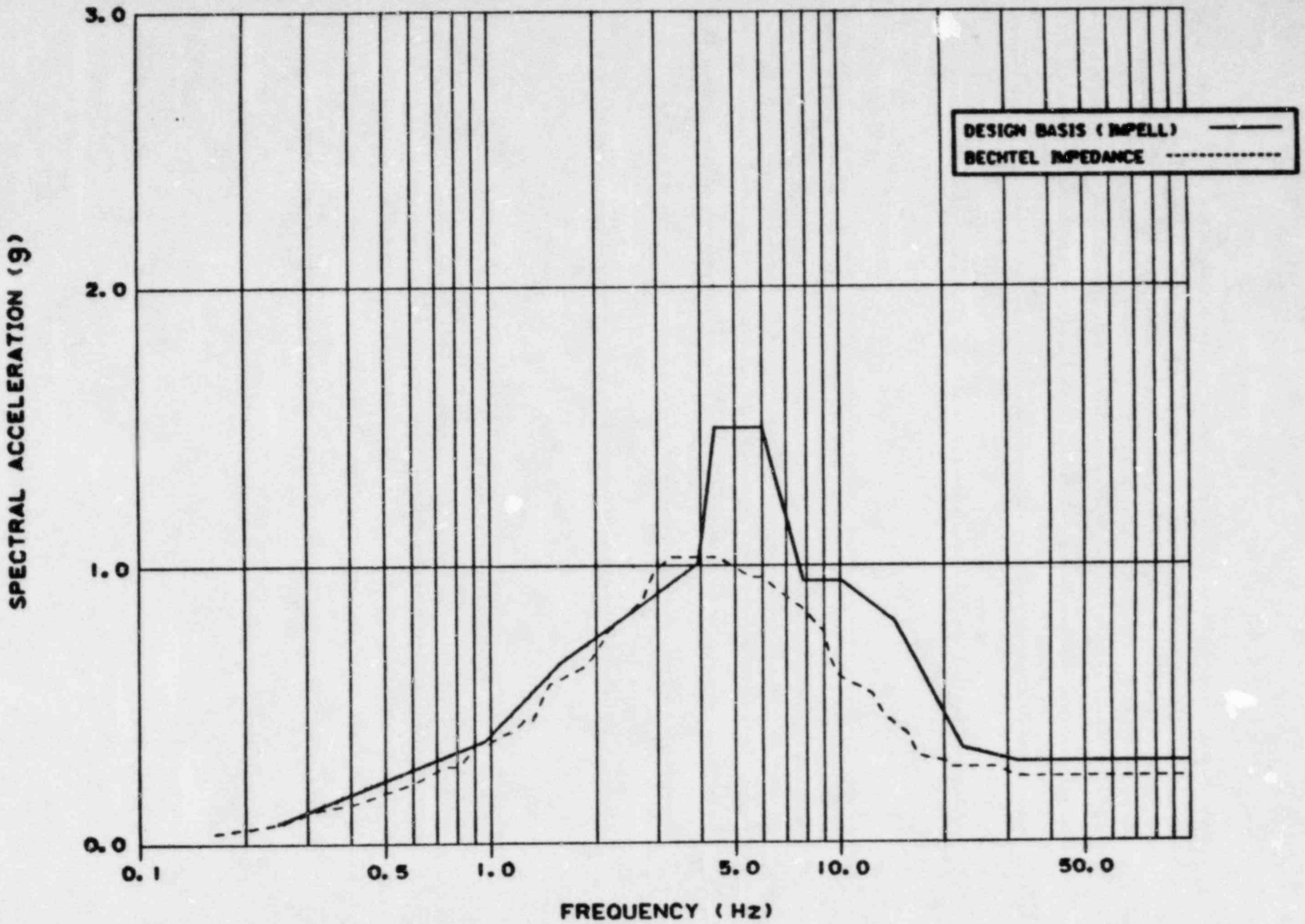

FIGURE 220.21-32

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

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

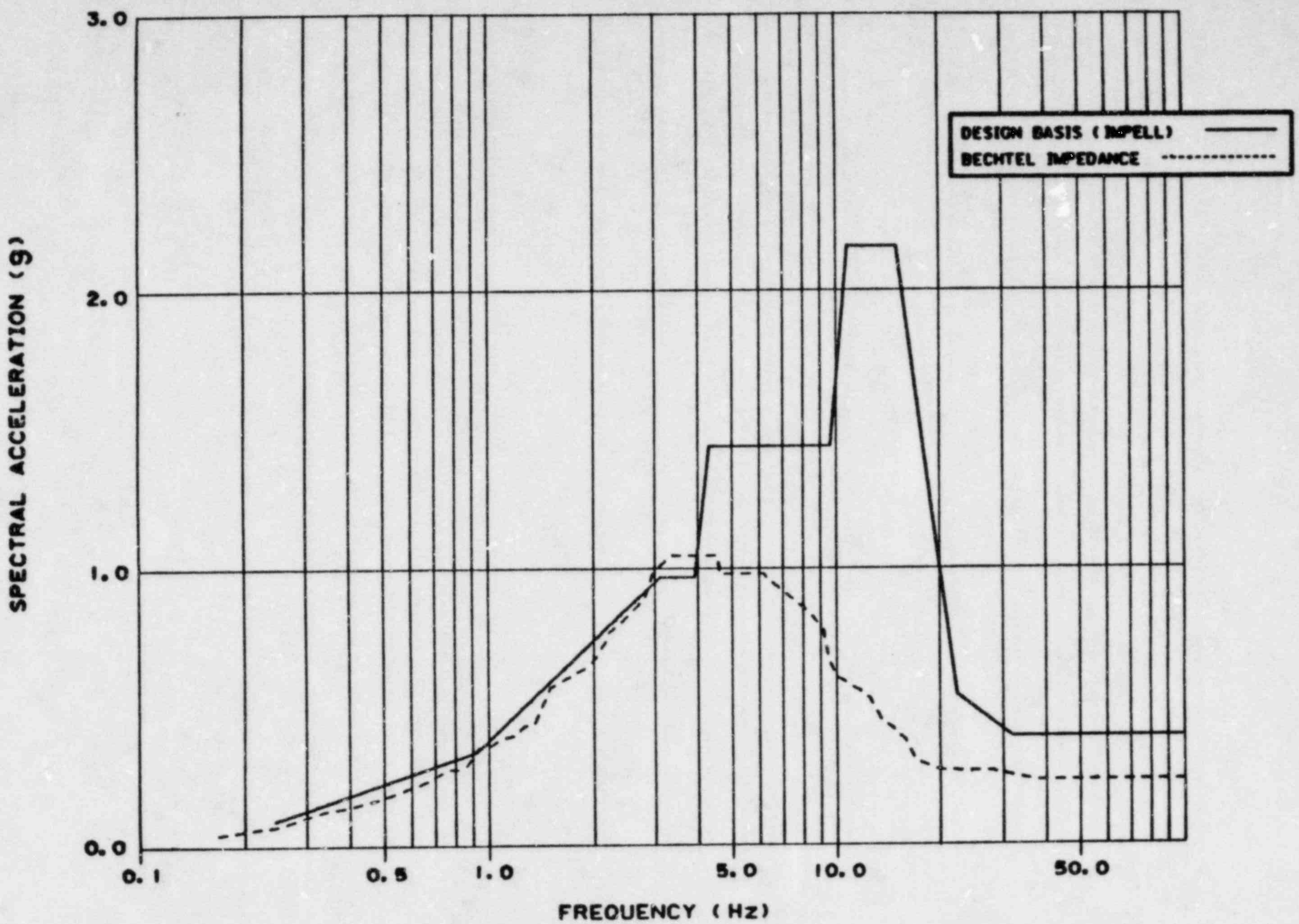

FIGURE 220.21-33

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

HOPE CREEK UNIT 1
CAD. L33A SHT 2, REV 0
07/27/84

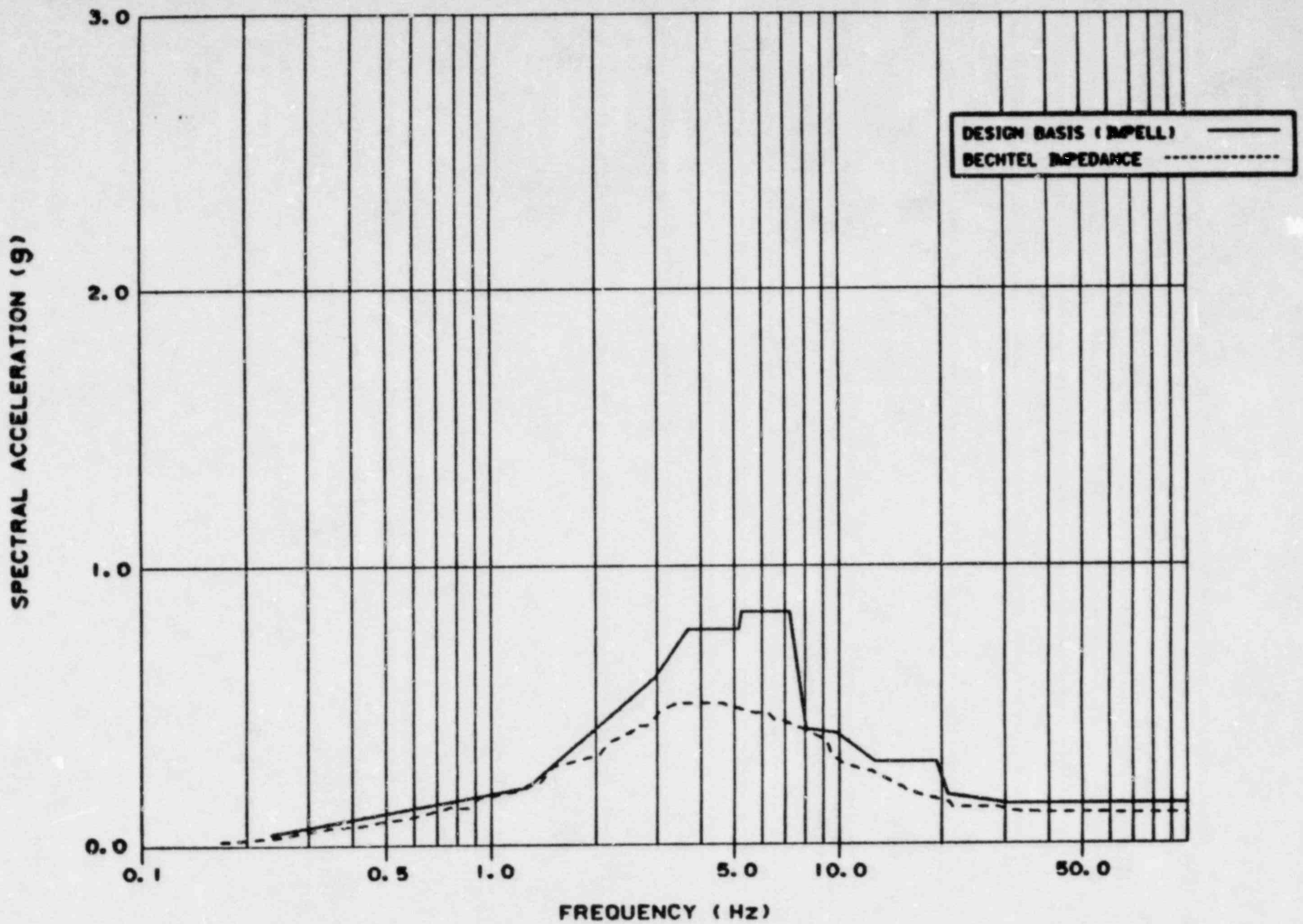

FIGURE 220.21-34

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

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

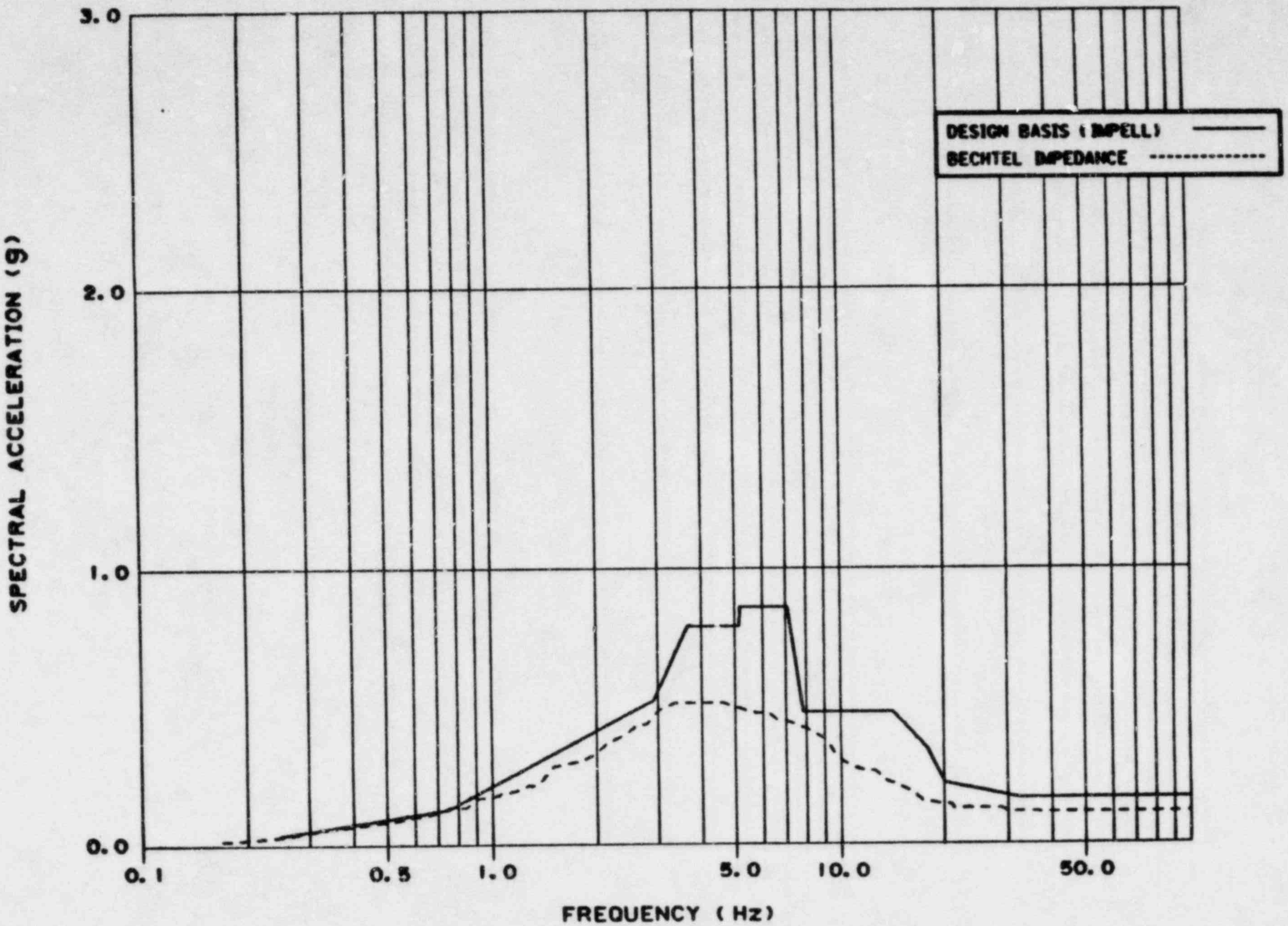

FIGURE 220.21-35

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

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

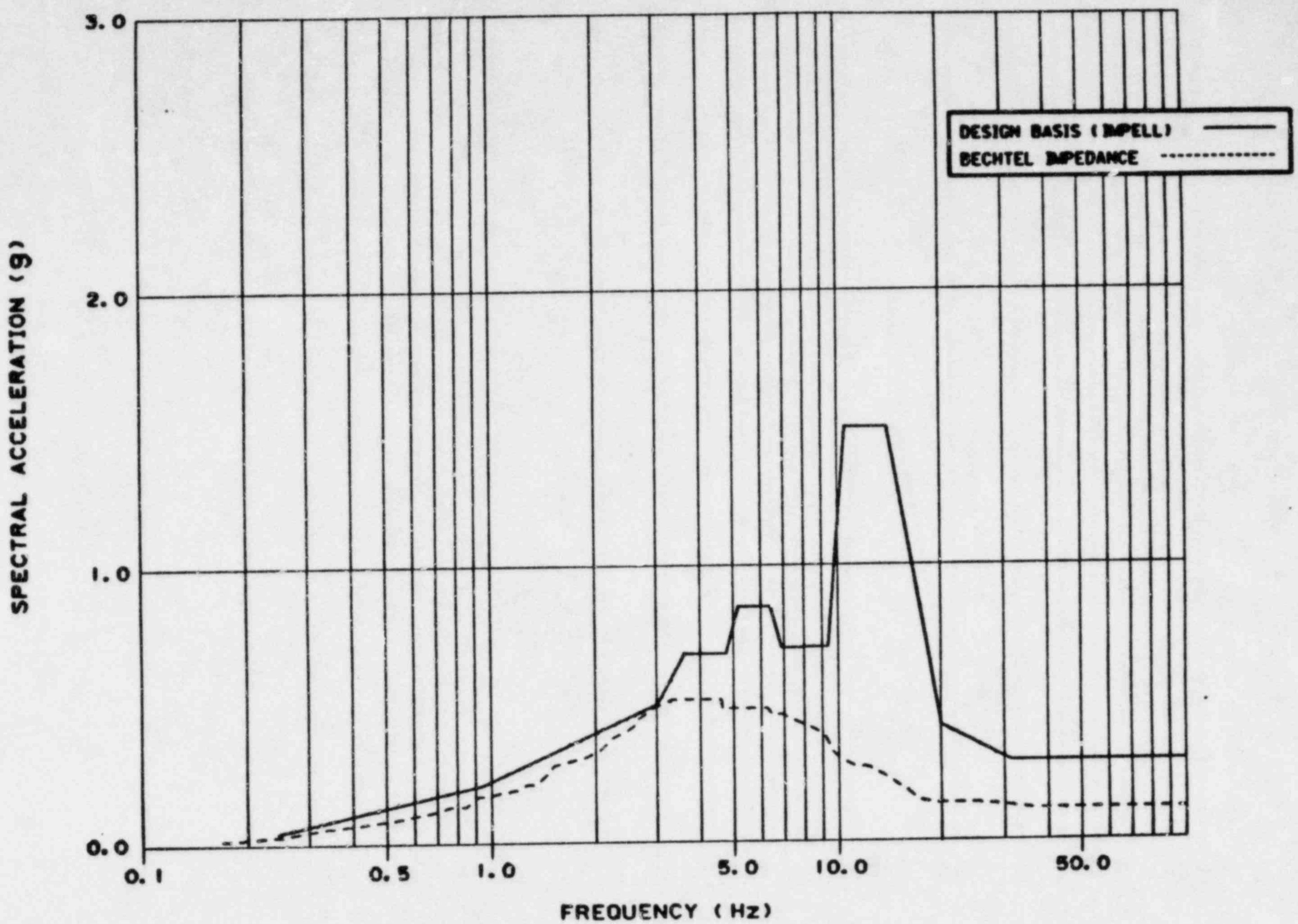

FIGURE 220.21-36

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

HOPE CREEK UNIT 1
CAD. L36A SHT 2, REV 0

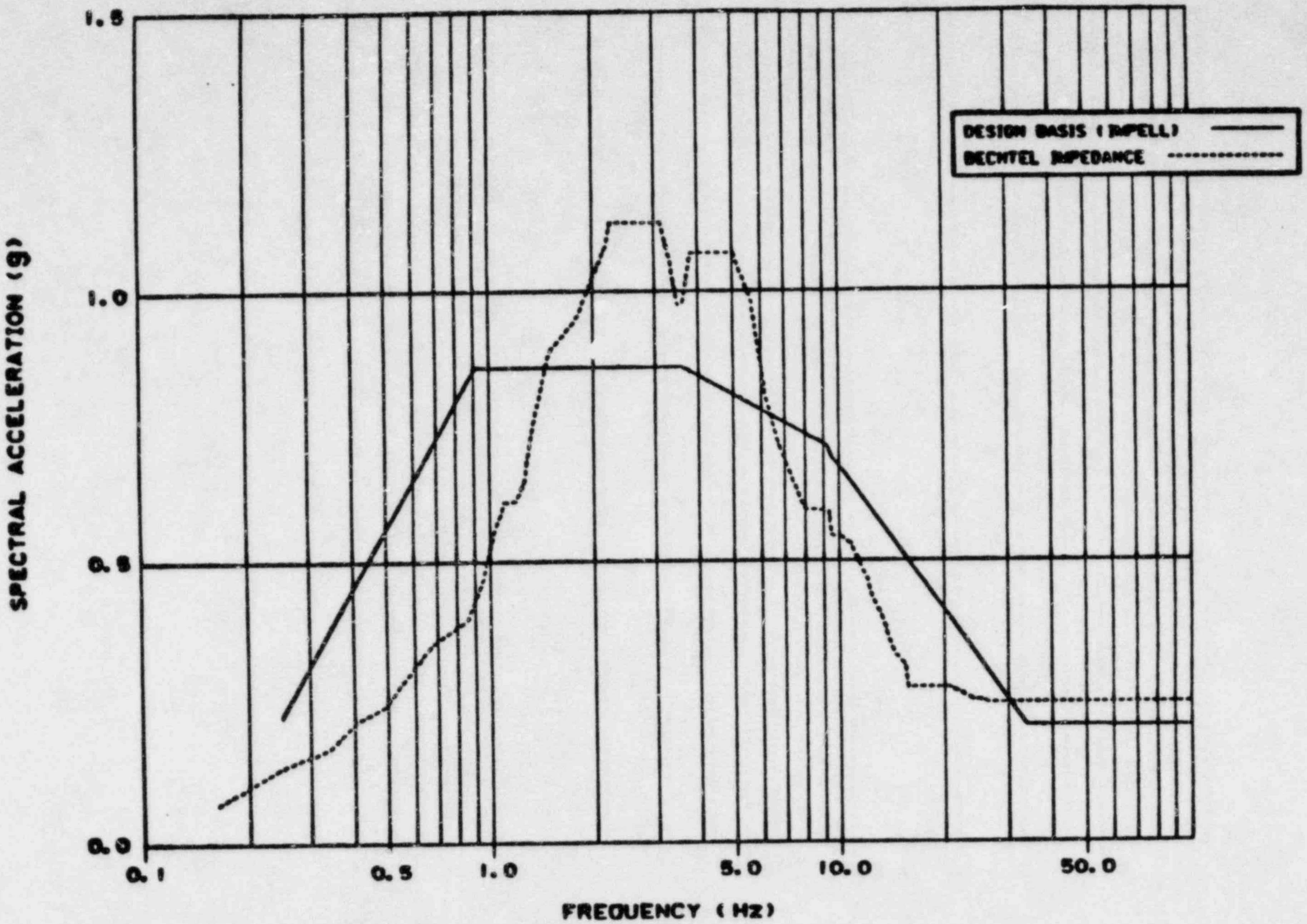

FIGURE ~~[illegible]~~ 220.21-37

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

HOPE CREEK UNIT 1
[illegible] REV 0

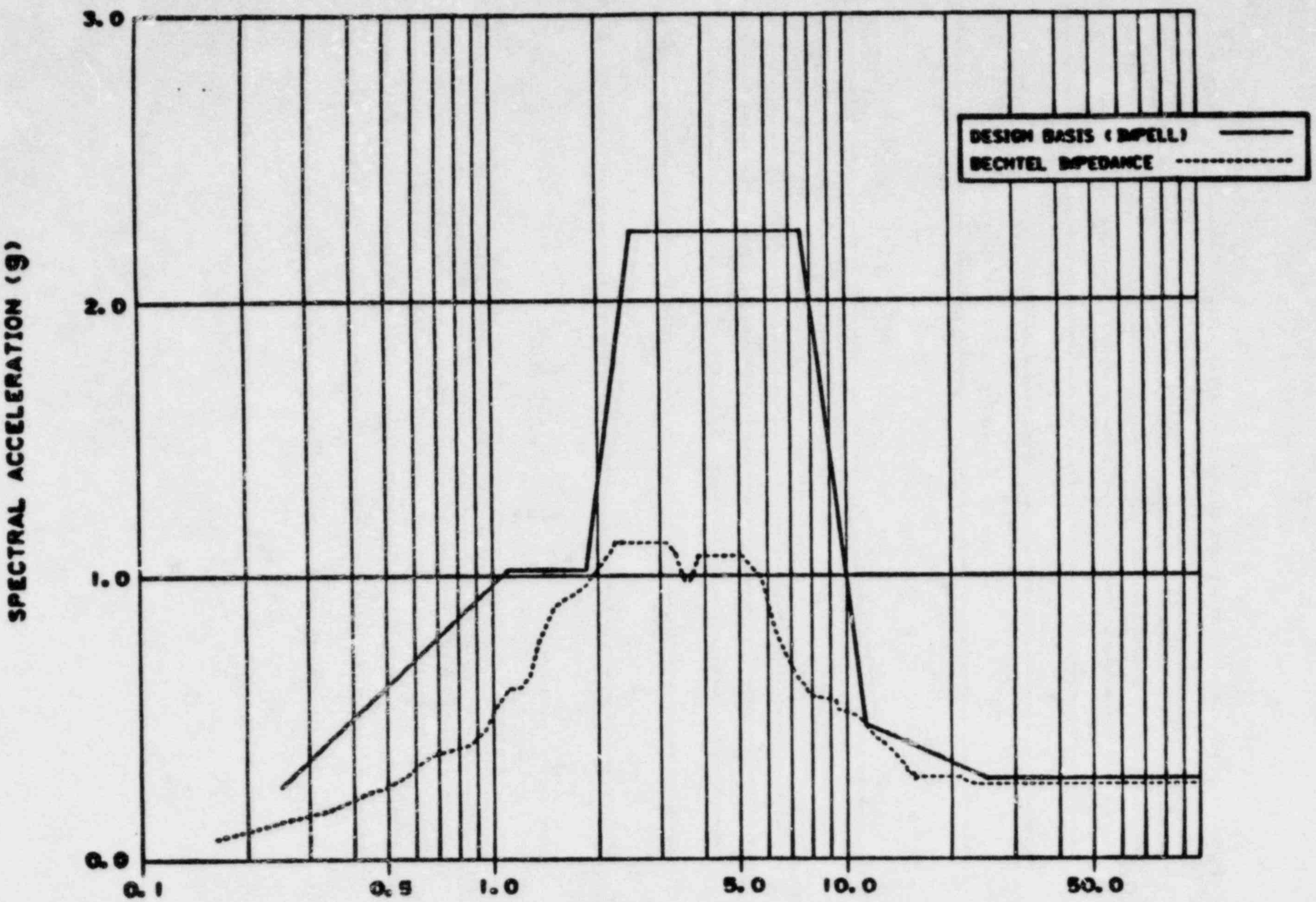

FIGURE ~~[illegible]~~ 220.21-38

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

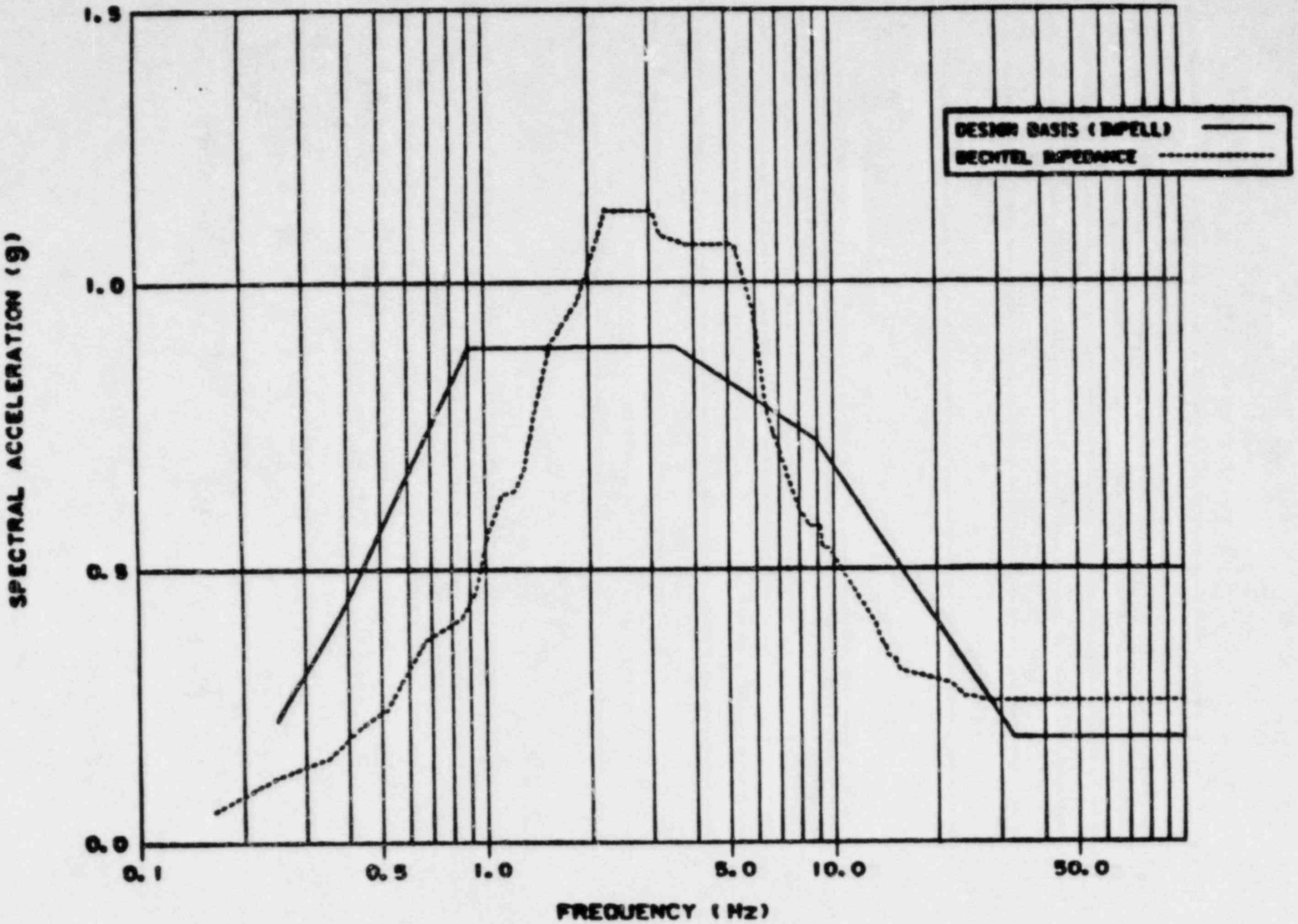

FIGURE ~~[illegible]~~ 220.21-39

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

HOPE CREEK UNIT 1
CAD, LS2, REV 0

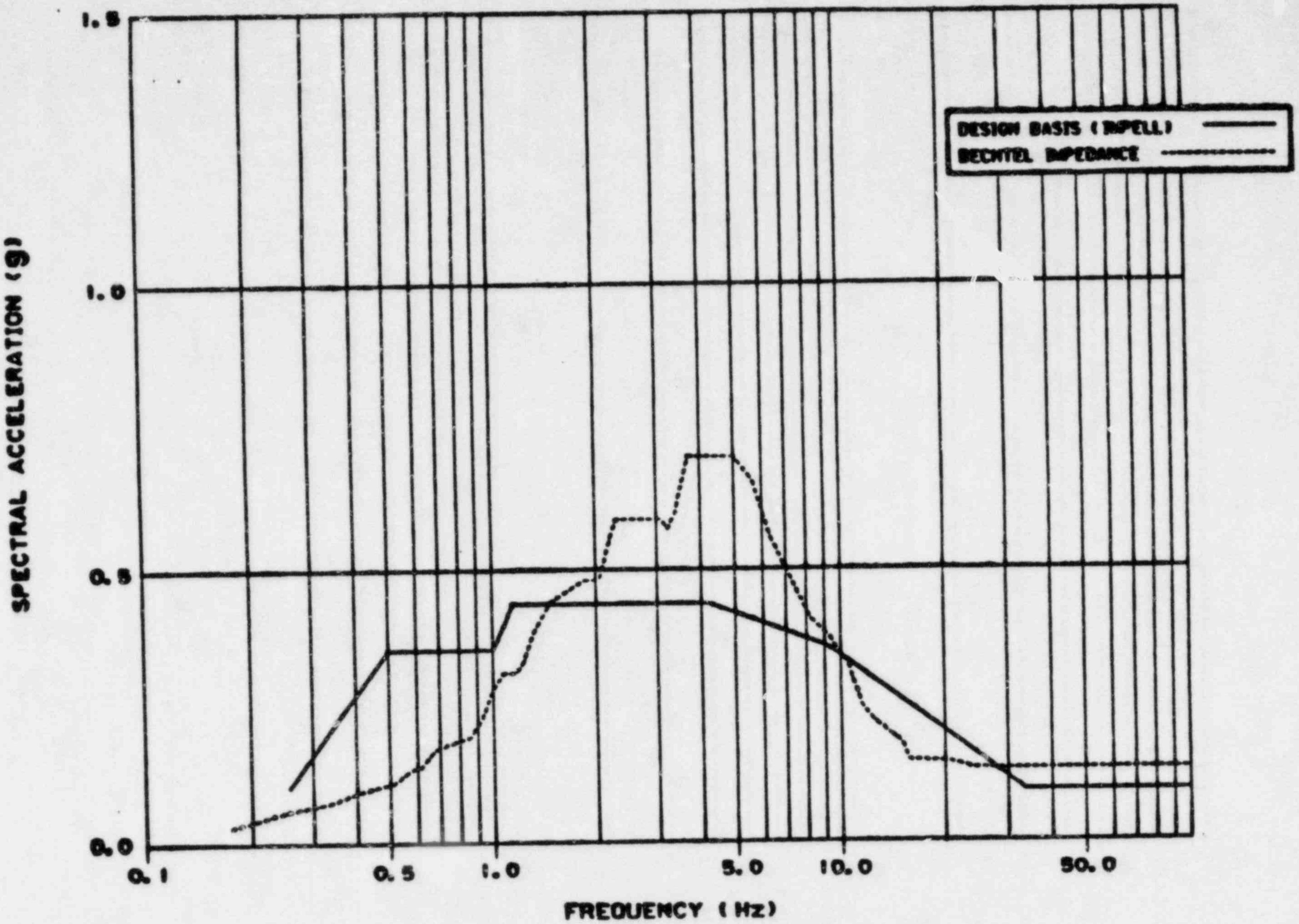

FIGURE ~~[illegible]~~ 220.21-40

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 93' -0",
N-S OBE 7% DAMPING

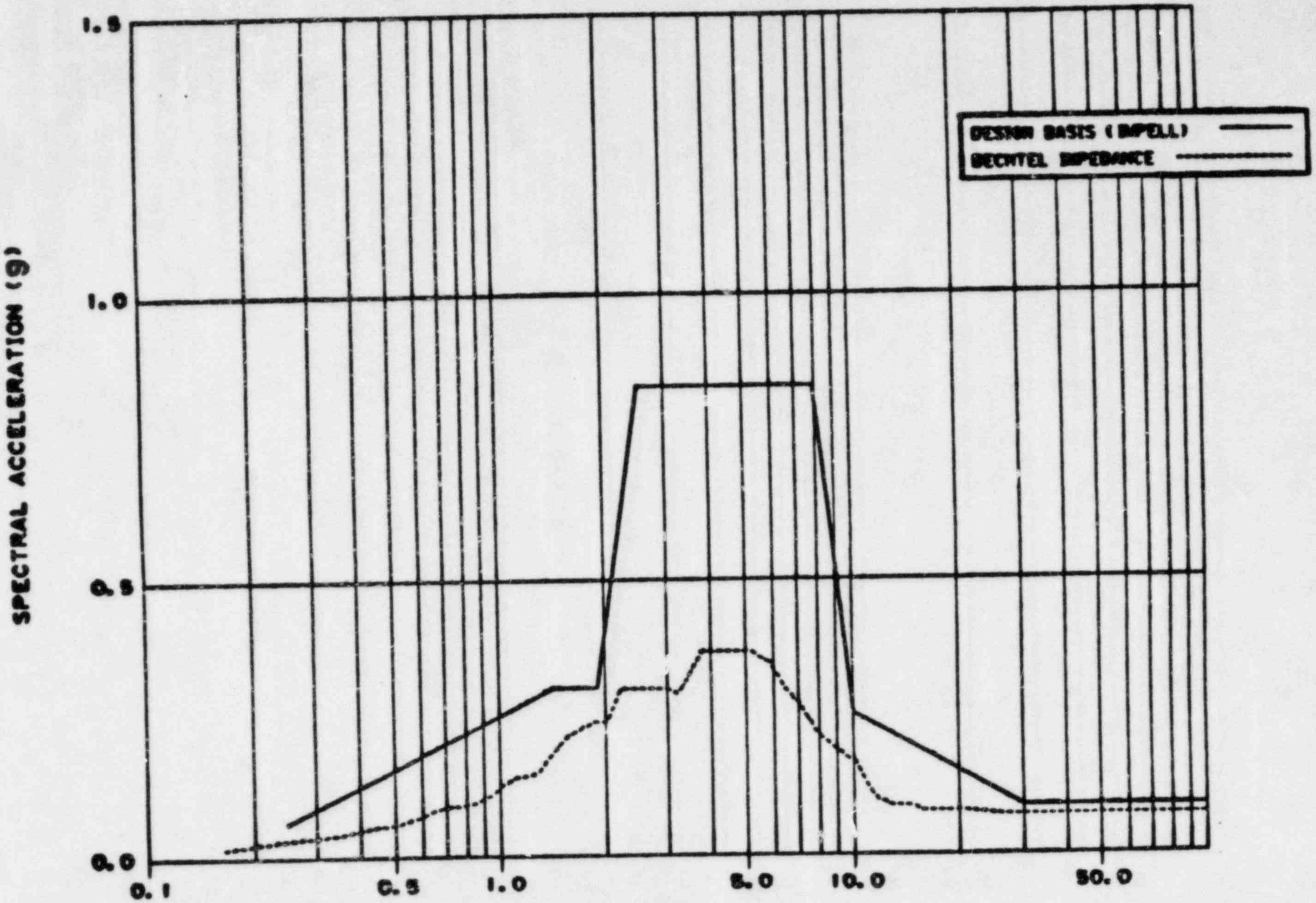

FIGURE ~~Am16-11~~ 220.21-41

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 114'-0",
N-S, OBE, 7% DAMPING

HOPE CREEK UNIT 1

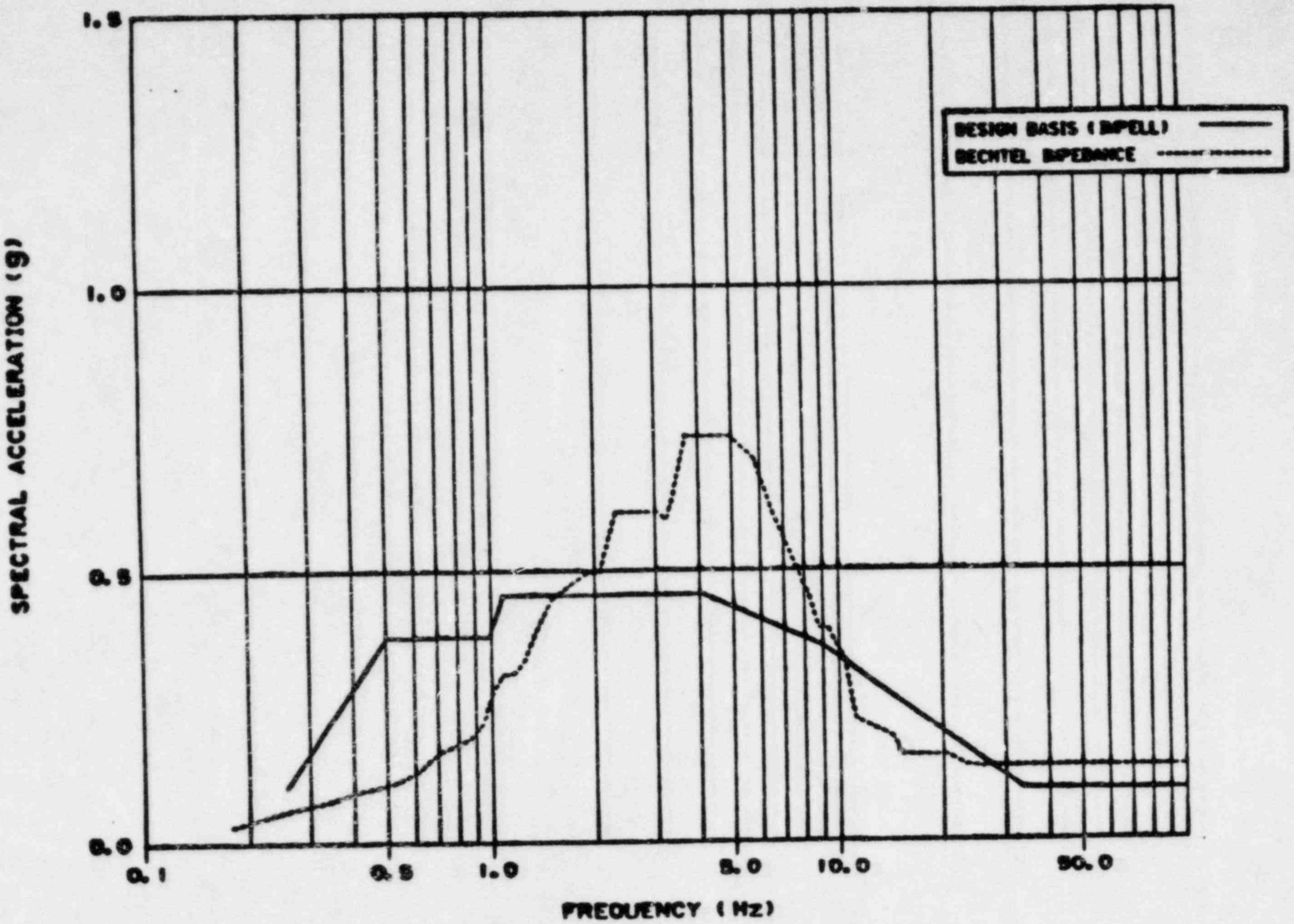

FIGURE ~~A-16-12~~ 220.21-42

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

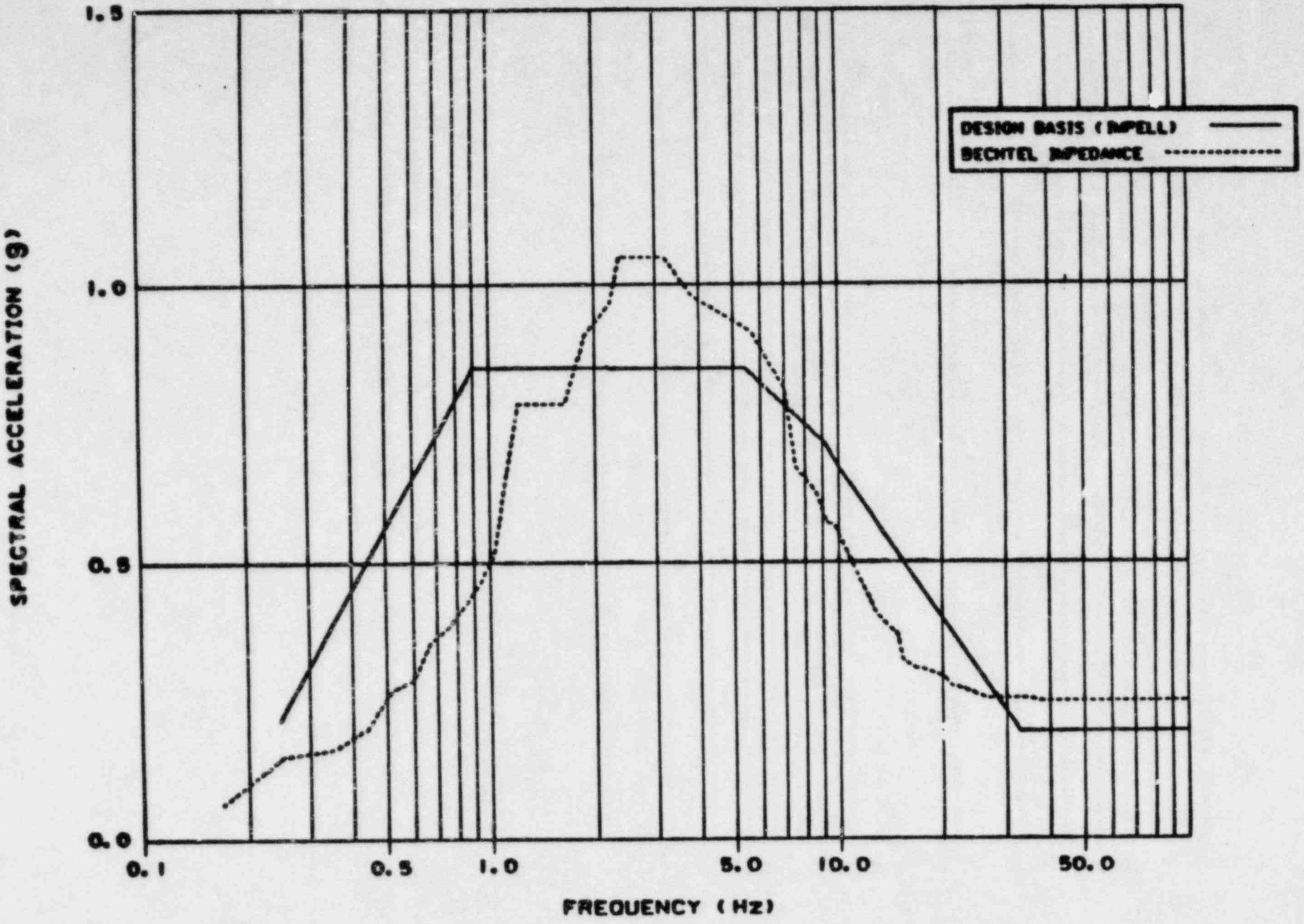

FIGURE ~~[struck-out number]~~ 220.21-43

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

HOPE CREEK UNIT 1
CAD. L37, REV 0

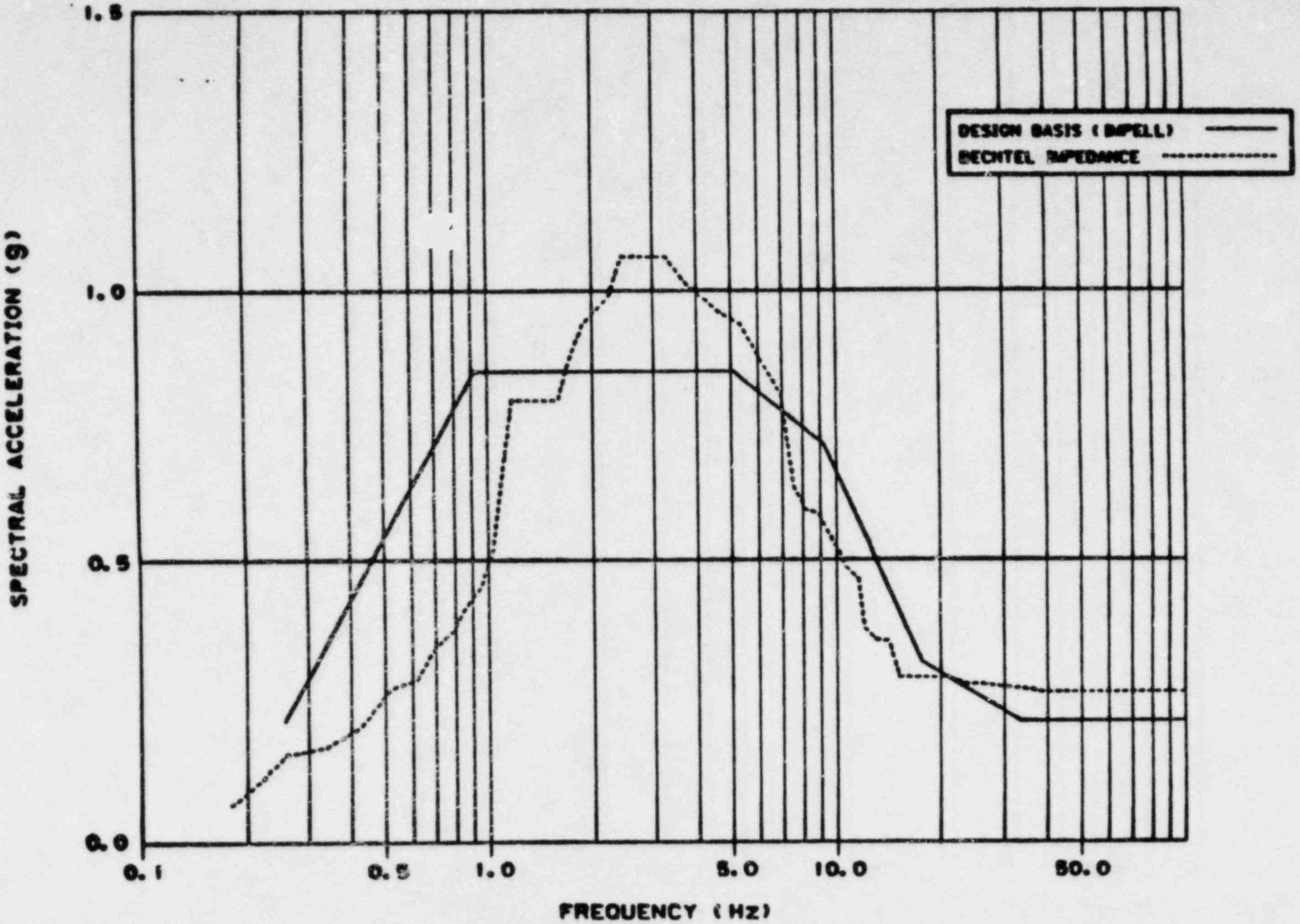

FIGURE ~~3-16-3~~ 220.21-44

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

HOPE CREEK UNIT 1

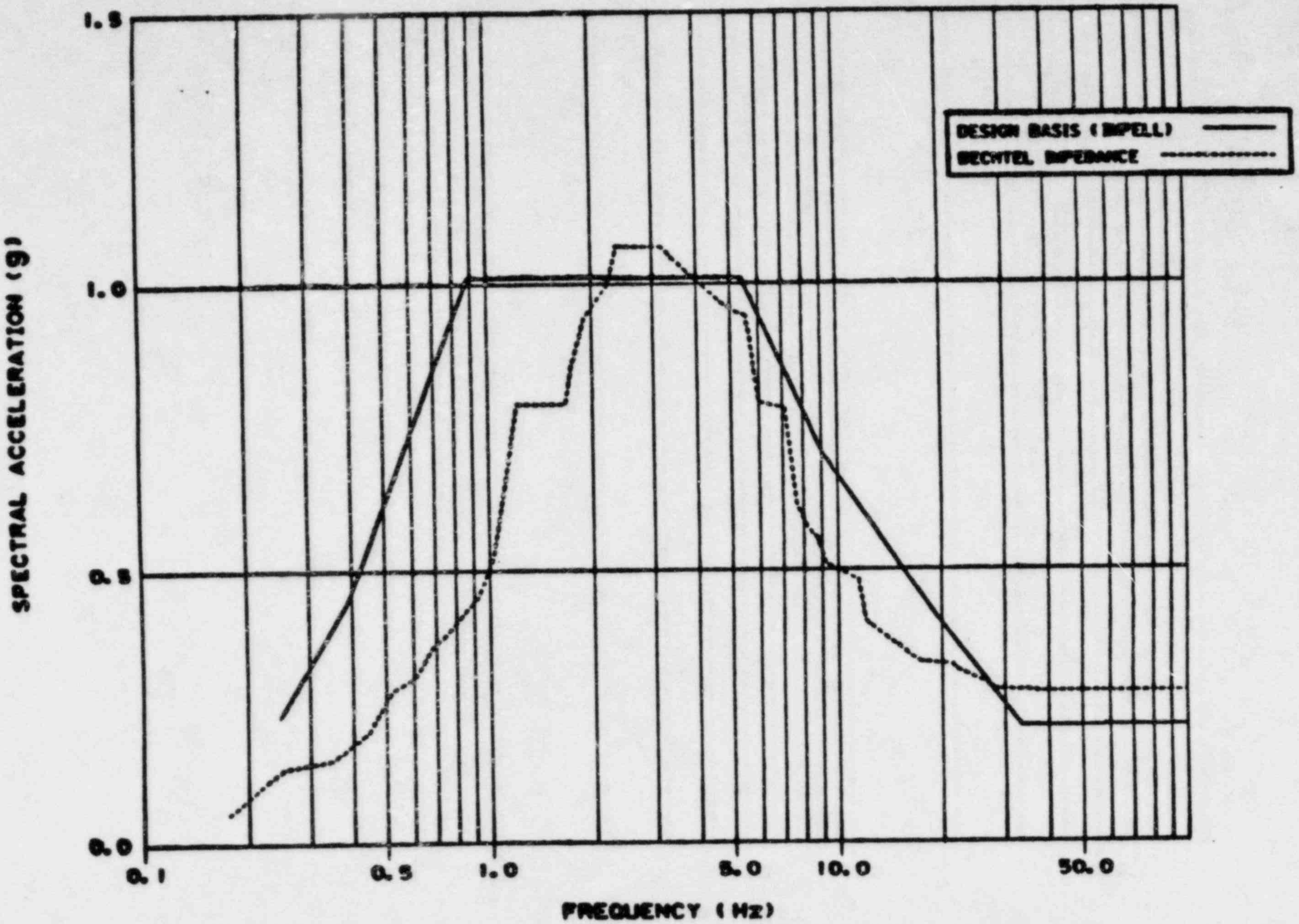

FIGURE 220.21-45

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

HOPE CREEK UNIT 1
CAD: L53, REV 0

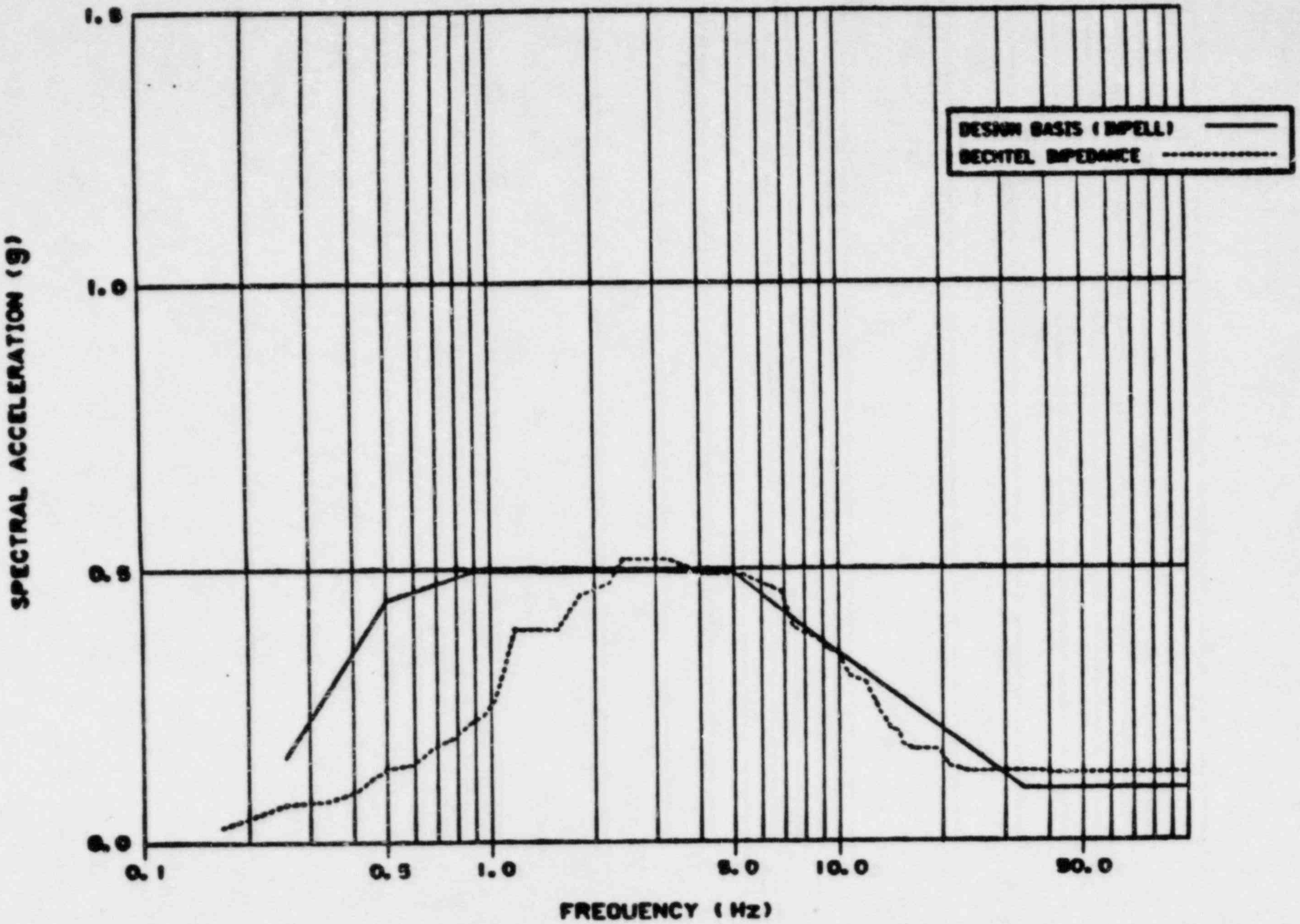

FIGURE ~~[illegible]~~ 220.21-46

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

HOPE CREEK UNIT 1
SAR-LSS-REV 0

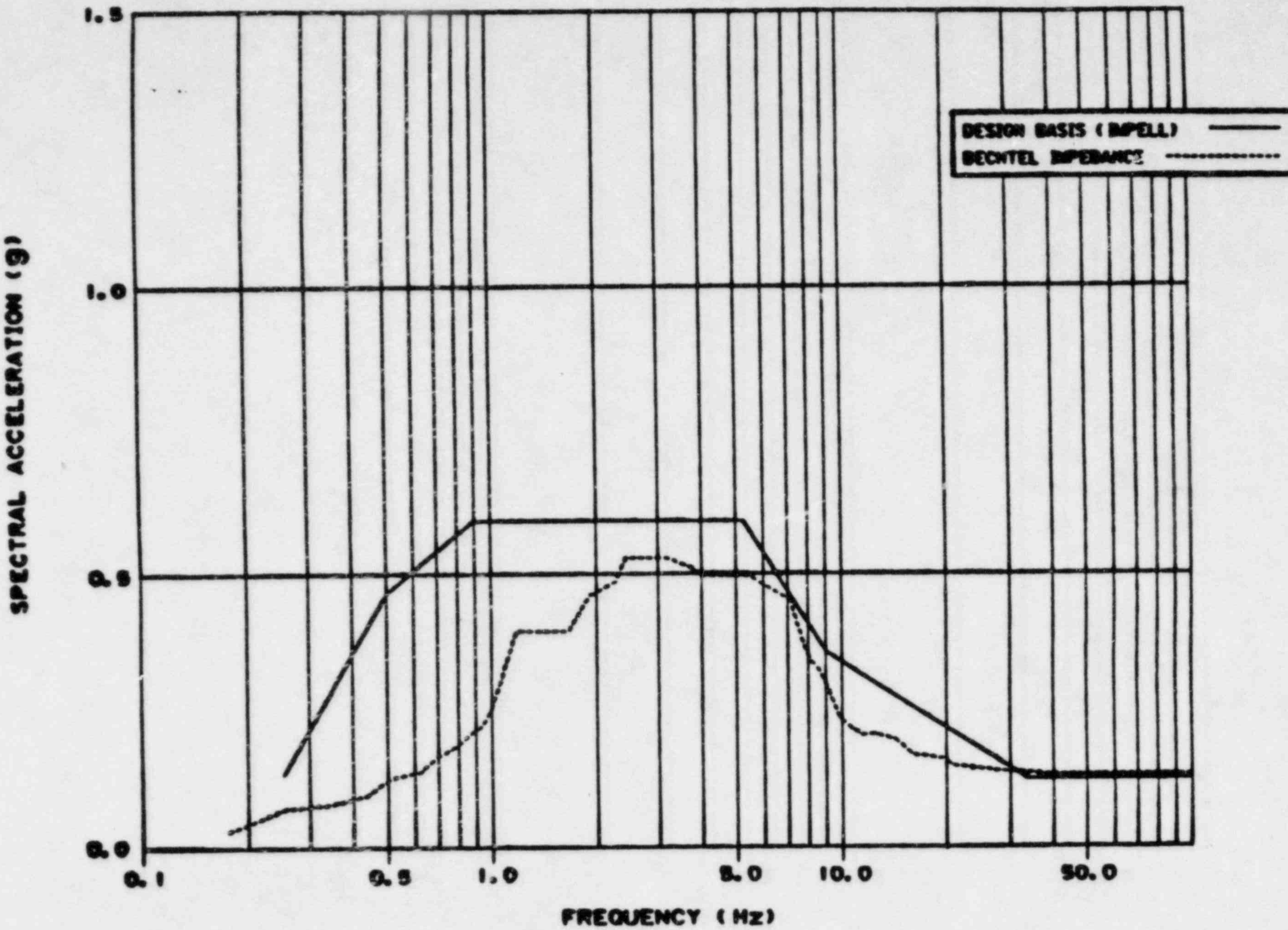

FIGURE ~~A-26-14~~ 220.21-47

RESPONSE SPECTRA COMPARISON,
INTAKE STRUCTURE AT ELEV. 114'-0",
E-W, OBE, 7% DAMPING

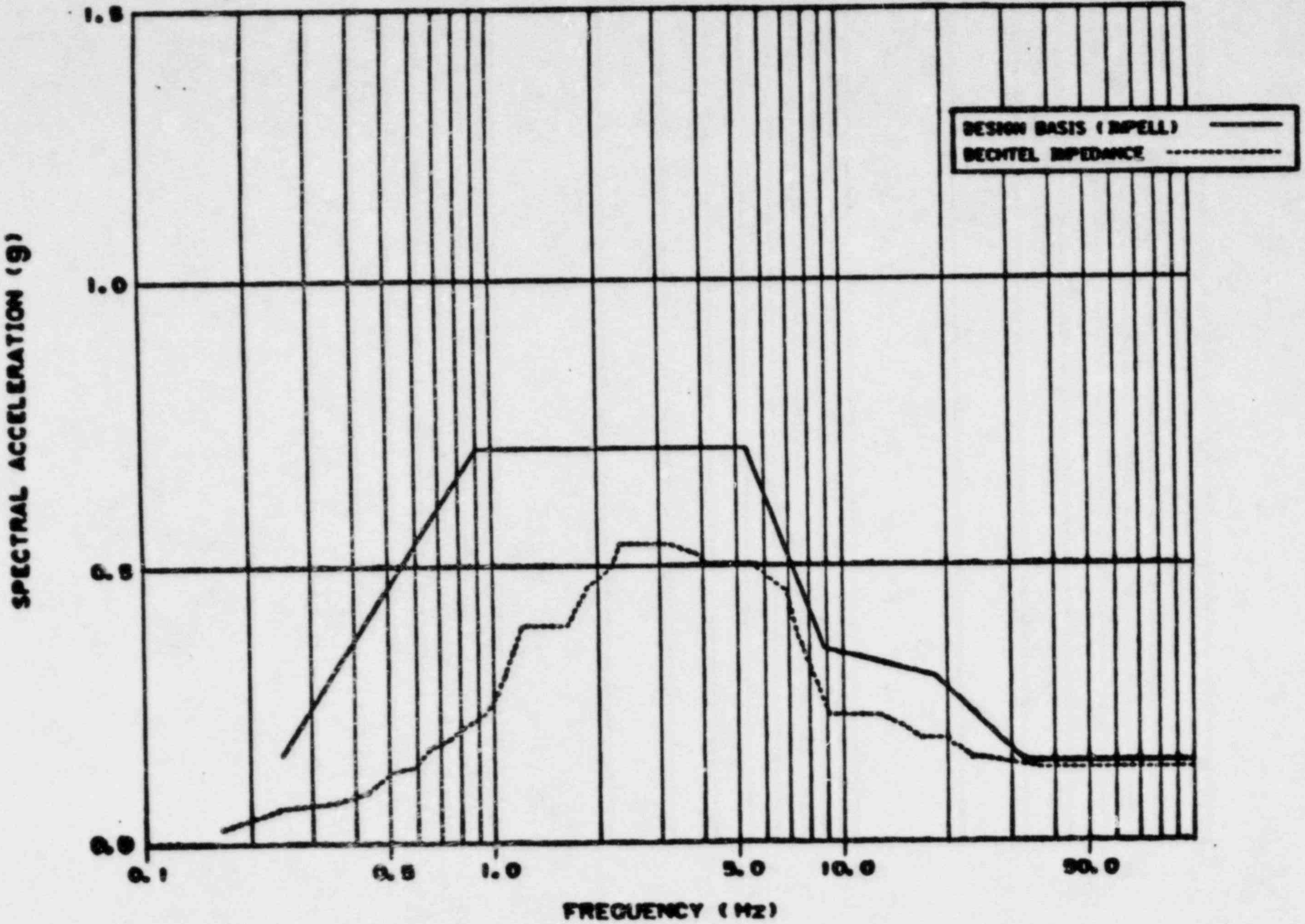

FIGURE ~~A-16-13~~ 220.21-48

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

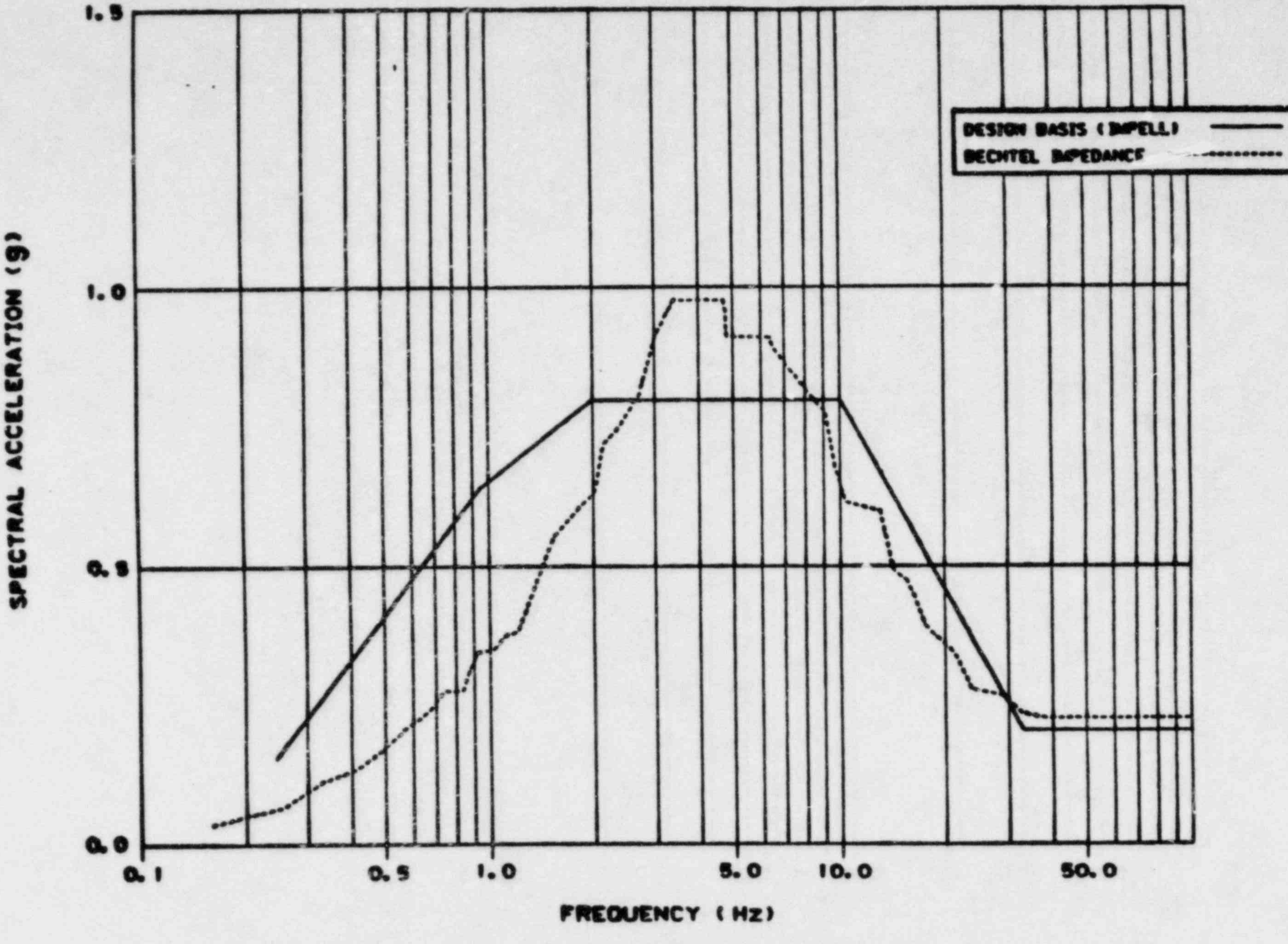

FIGURE ~~[illegible]~~ 220.21-49

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

HOPE CREEK UNIT 1
CAD. L38, REV 0

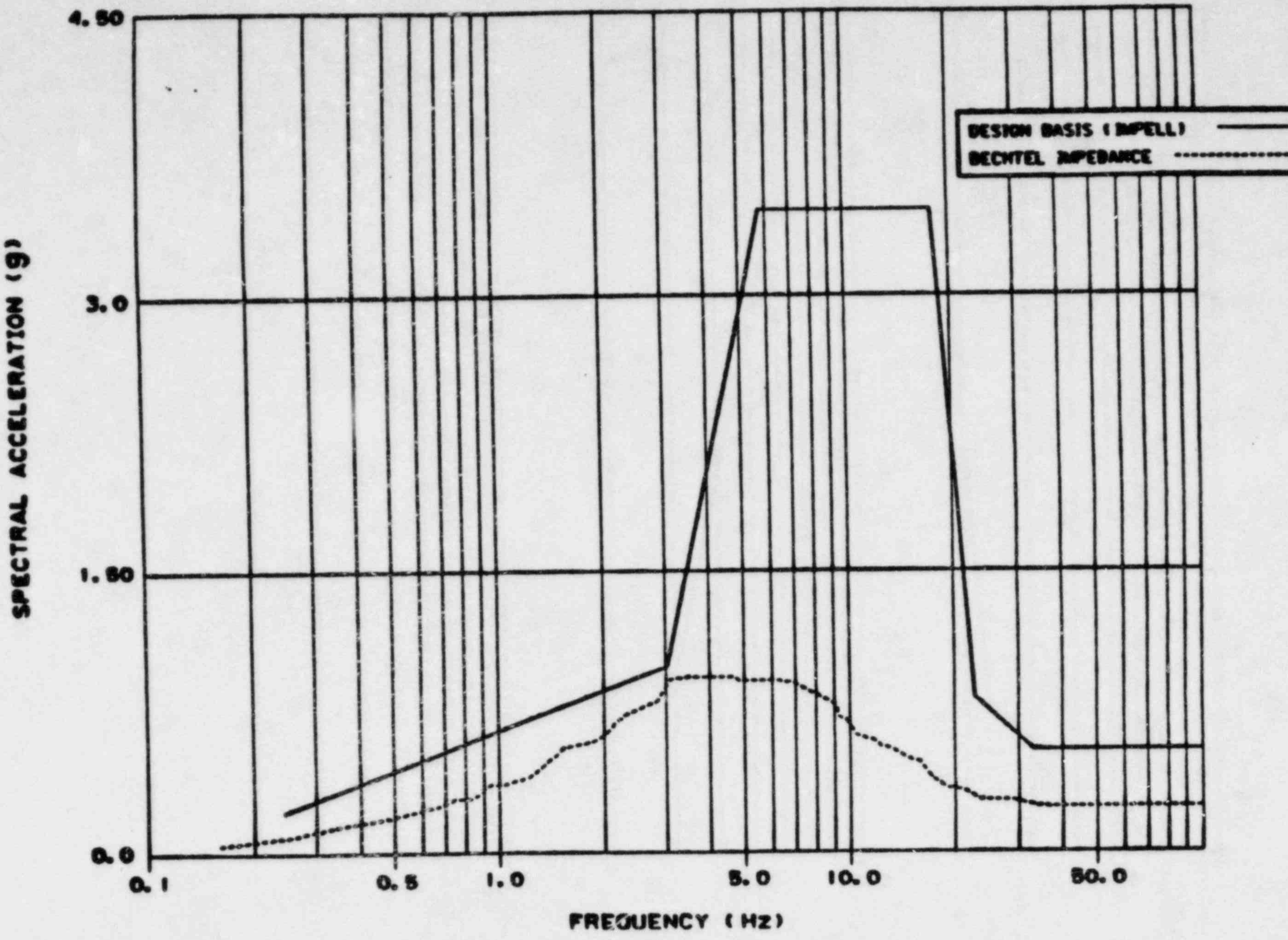

FIGURE ~~3.10-8~~ 220.21-50

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

HOPE CREEK UNIT 1
CAD. L48, REV 0

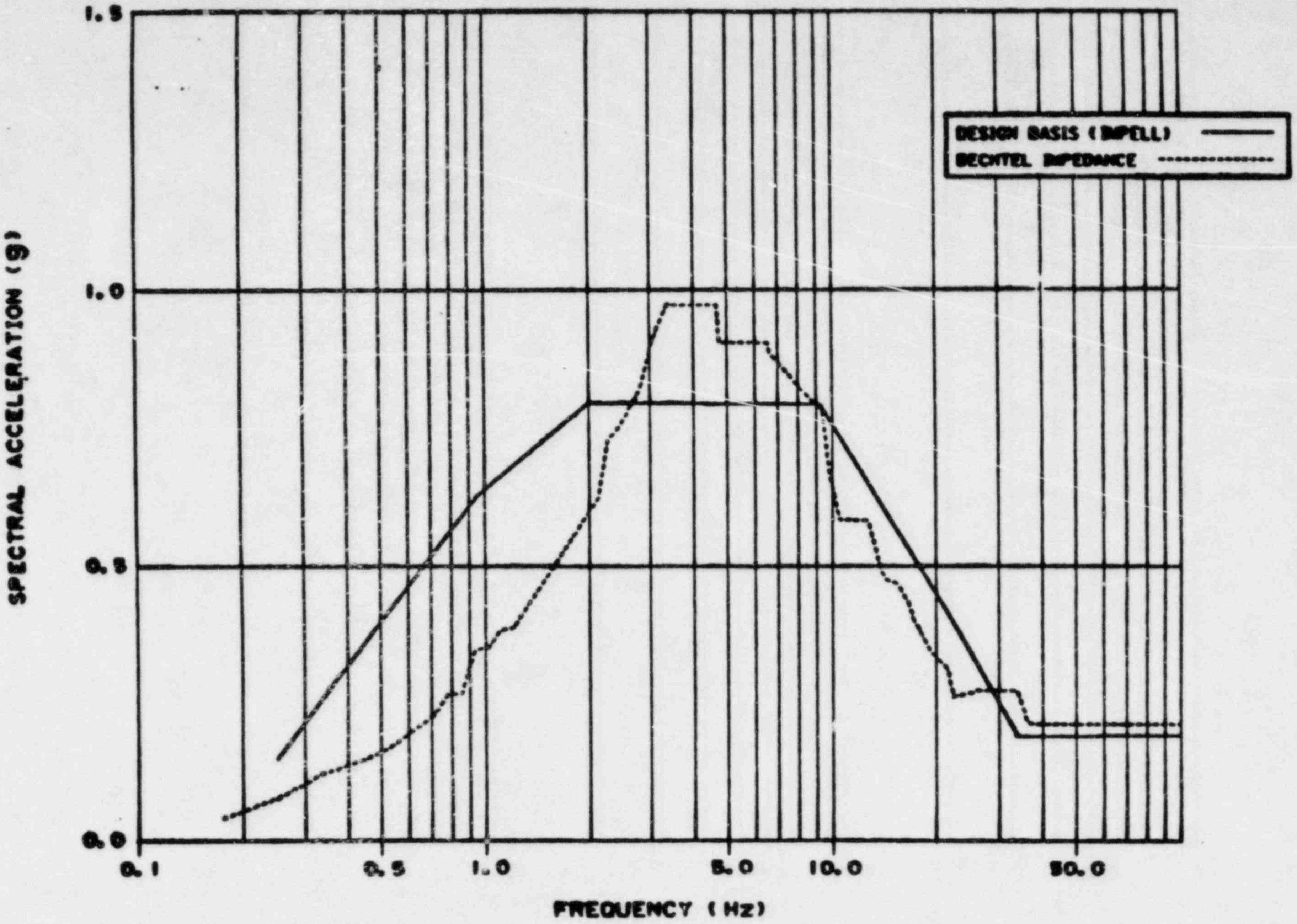

FIGURE ~~XXX~~ 220.21-51

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

HOPE CREEK UNIT 1
CAD, L54, REV 0

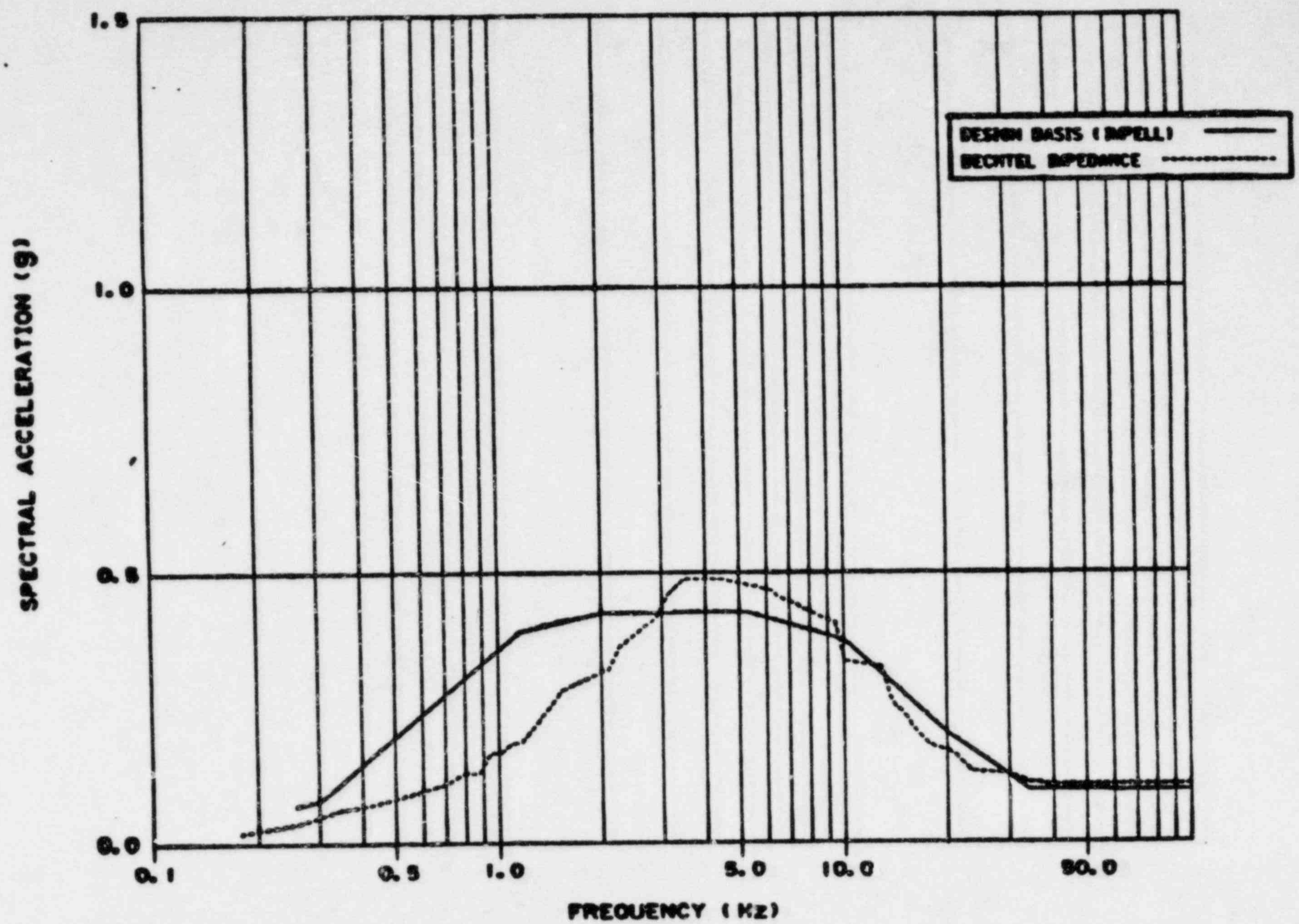

FIGURE ~~A-16-15~~ 220.21-52

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

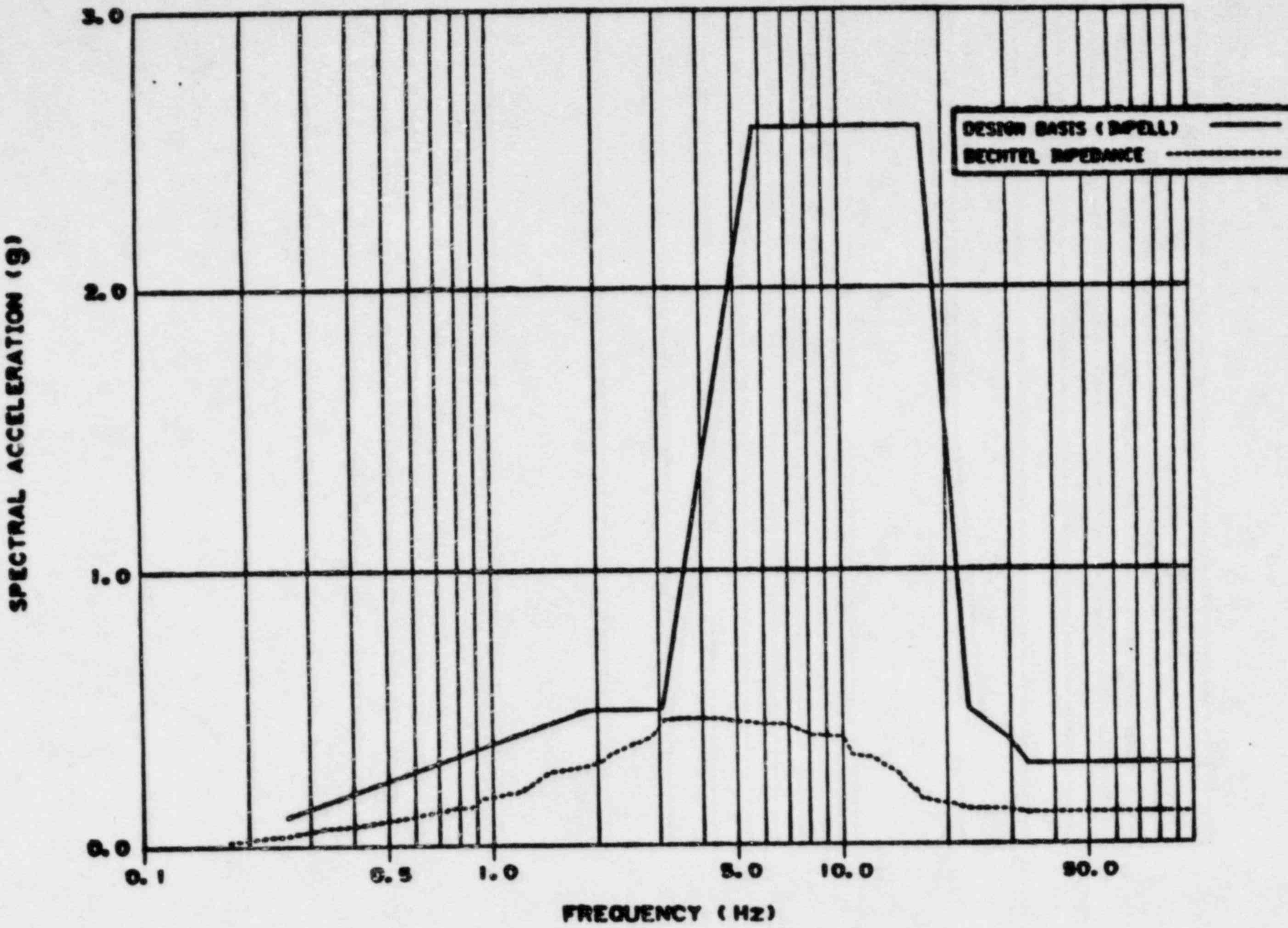

FIGURE ~~3.10-17~~ 220.21-53

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

HOPE CREEK UNIT 1
CAD. L63. REV 0

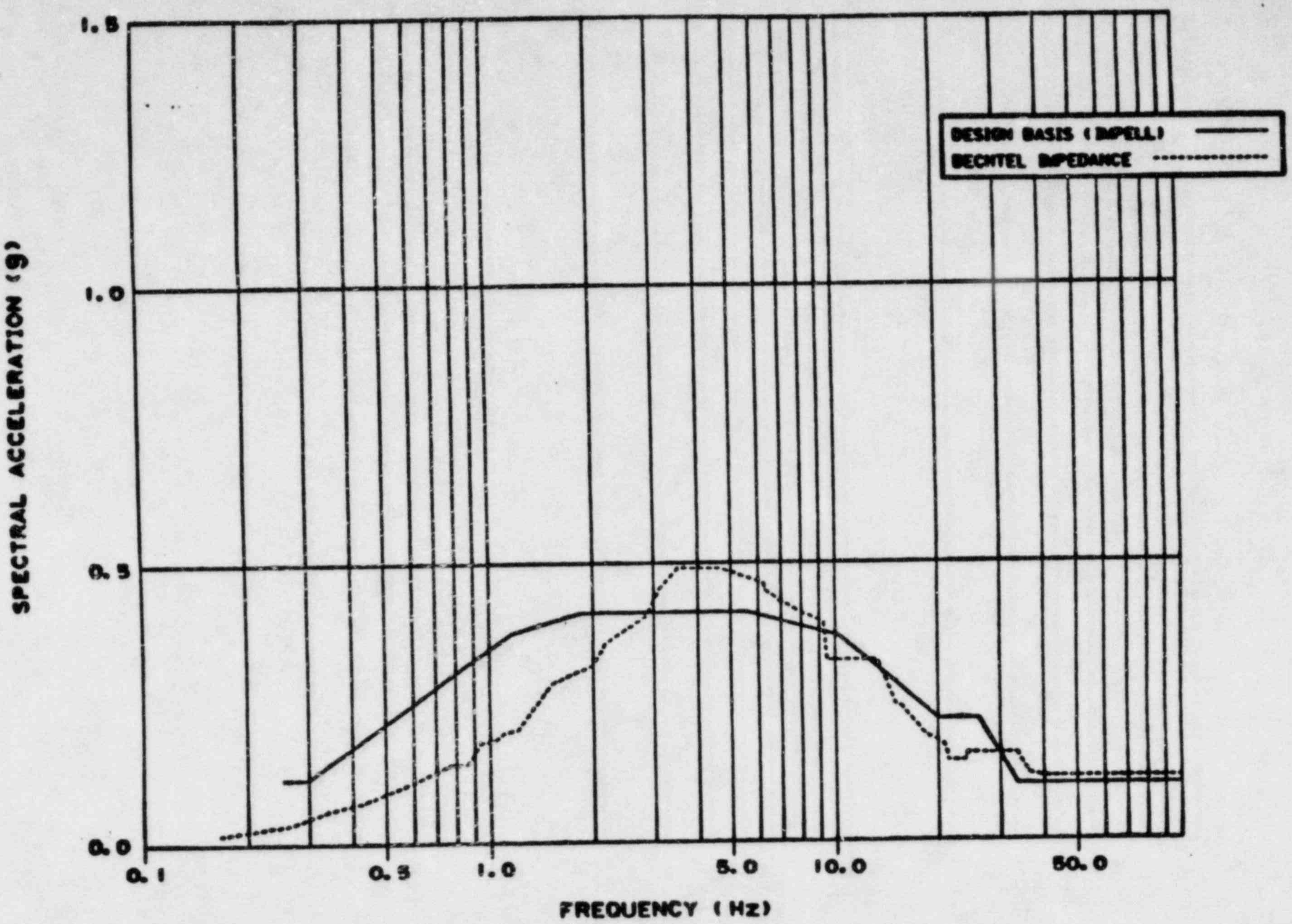

FIGURE ~~[illegible]~~ 220.21-54

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

HOPE CREEK UNIT 1
CAU, LTZ, REV 0
[illegible]

ATTACHMENT IV

QUESTION 480.11 (SECTION 6.2.3)

Provide a table listing the high energy lines which pass through the secondary containment and indicate which ones, if any, are covered with guard pipes. For those high energy lines which do not have guard pipes, provide results of analyses to demonstrate that the primary and secondary containment structure are capable of withstanding the effects of a high energy pipe rupture occurring inside the secondary containment without loss of integrity.

RESPONSE

As discussed in Section 3.6.2.4, guard pipe assemblies are not used in HCGS. Table 3.6-1 provides a listing of all high energy lines in the drywell and reactor building. The effects of high energy line rupture, including system description, pipe break location, and a verification of the reactor shutdown capability, is provided in Section 3.6.1.2.1. For all cases, the structural integrity of the reactor building (secondary containment) is maintained. Results of the pressure temperature transient analyses ~~will be~~ provided in Table 3.6-5, ~~by August, 1984.~~

has been revised

ATTACHMENT V

person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside the control room.

Response

The lines of responsibility and authority of the SSS-SRO (as promulgated in the vice president - nuclear letter of September 14, 1982), along with the authority and responsibility of the SSS-SRO (or the individual assigned the control room command function) to limit access, will be contained in Administrative Procedure SA-AP.ZZ-001, Station Organization and Responsibilities (available March 1, 1985). This item is also discussed in the response to Item I.A.1.2.

- I.C.5 FEEDBACK OF OPERATING EXPERIENCE

Position

Each licensee will review its administrative procedures to assure that operating experience from within and outside the organization is continually provided to operators and other operational personnel and is incorporated in training programs.

Response

An integrated nuclear department procedure is being prepared, and will be available by March 1, 1985. When issued, station procedures will be revised to incorporate the prescribed procedure. Operating Department Procedure OP-AP.ZZ-105(Q) will be used to disseminate information to Operating Department Personnel.

INSERT 5

~~In addition, the Nuclear Department Training Center will revise, by June 1984, the two procedures pertaining to this subject to include HCGS. The procedures are TP-306, Plant Design Review Program, and TP-307, Operational Experience Review Program.~~

INSERT 5

Industry operating experiences including events occurring within our organization are reviewed for applicability to Hope Creek by the Reliability and Assessment Department. Pertinent information is communicated to the appropriate department for their information and any actions required are tracked until they have been satisfactorily completed. In addition, information is communicated to the Manager - Nuclear Training for incorporating new material into the Training Programs. The activities of the Reliability and Assessment Department with respect to operating experiences (i.e., INPO's SEE-IN Program) are governed by procedure M3-POP-001 "Operating Experience Review Program".

Vendor technical documents describing the operation and maintenance of installed equipment and components associated with Hope Creek Generating Station shall be controlled in the following manner;

1) When vendor documents are received by disiplines within the Nuclear Department, these documents will be forwarded to the Nuclear Engineering Department for review and approval for inclusion into the Vendor Document Control System.

2) Once approved by the cognizant engineer they will be assigned a unique number and distributed to all user departments, and incorporated in procedures and training as necessary.

Information on operating experience provided by the NRC through the I & E Bulletins/Information Notices, generic letters and letters on the docket are processed by nuclear licensing and regulation department within the nuclear department. These letters are distributed to various disciplines within nuclear department for feedback of information. Response action form is utilized when a response or action is required and is monitored through the response tracking system ~~till~~ to completion.

In addition, the Nuclear Department Training Center ~~will~~ has revised ~~by June 1984~~, the two procedures pertaining to this subject to include HCGS. The procedures are TP-306 HC, Plant Design Review Program, and TP-307 HC, Operational Experience Review Program. These procedures will be revised as necessary to incorporate the integrated nuclear department procedure.

ATTACHMENT VI

QUESTION 260.15

The fourth paragraph of FSAR Section 17.2.2 refers to Section 1.8 for commitments to Regulatory Guides. Section 1.8 primarily addresses Regulatory Guide commitments during design and construction, and the staff review of the FSAR is concerned with Regulatory Guide commitments during the operations phase. With any proposed clarifications or exceptions, provide a commitment in the FSAR to the effect that: "During the operations phase of HCGS, PSE&G commits to comply with the regulatory position in ..." the appropriate issue of the Regulatory Guide listed on pages 17.1-26 and 17.1-27 (with RG 1.33 replacing RG 1.28) or NUREG-0800 (Rev. 2 - July 1981). For systems, components, and structures covered by the ASME Boiler and Pressure Vessel Code Section III (Classes 1, 2 and 3), the code QA requirements should be supplemented by the specific guidance addressed in the regulatory positions of the applicable Regulatory Guides. (2B3)

RESPONSE

Section 17.2.2 lists regulatory guidance applicable to the QA program. This list has been revised to include Regulatory Guides 1.116, 1.123, and 1.144. ~~PSE&G will revise~~ Section 1.8 ~~to~~ has been revised to reflect compliance with listed Regulatory Guides which are applicable during the operations phase, along with any clarification, modifications, etc. ~~by June 1984~~.

The code QA requirements are used for the procurement of systems, components and structures covered by the ASME Boiler and Pressure Vessel Code Section III (classes 1, 2, and 3). The standard QA program controls apply to Q-Listed code items following receipt at the station. In addition, applicable requirements of Regulatory Guide 1.38 will be applied to ASME Code procurements when necessary to assure safe shipment.

1.8.1.63 Conformance to Regulatory Guide 1.63, Revision 2, July 1978: Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants

Although Regulatory Guide 1.63 is not applicable to HCGS, per its implementation section, HCGS complies with the design, qualification, construction, installation, and testing requirements of IEEE 317-1976, as modified by Regulatory Guide 1.63, subject to the clarification in Section 8.1.4.12.

1.8.1.64 Conformance to Regulatory Guide 1.64, Revision 2, June 1976: Quality Assurance Requirements for the Design of Nuclear Power Plants

~~Although Regulatory Guide 1.64 does not apply to HCGS, per its implementation section, HCGS complies with it.~~
HCGS complies with Regulatory Guide 1.64

~~The architect-engineer indicates that their design verification procedures conform to ANSI 45.2.11-1974 and also that compliance with this standard is as modified and interpreted by Revision 1 of Regulatory Guide 1.64. However, the architect-engineer did not comply with Revision 2 in that it allowed checking of the design output documents by the originator's supervisor.~~

See Section 17.2 for further discussion of quality assurance procedures and Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

1.8.1.65 Conformance to Regulatory Guide 1.65, Revision 0, October 1973: Materials and Inspections for Reactor Vessel Closure Studs

Regulatory Guide 1.65 is not applicable.

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

1.8.1.87 <u>Conformance to Regulatory Guide 1.87, Revision 1, June 1975: Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596)</u>

Regulatory Guide 1.87 is not applicable to HCGS.

1.8.1.88 <u>Conformance to Regulatory Guide 1.88, Revision 2, October 1976: Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records</u>

During the operations phase, HCGS complies with ANSI N45.2.9-1974, as modified and interpreted by Regulatory Guide 1.88 and NUREG 0800 (Standard Review Plan) Revision 2, Section II.17.4. During the construction and startup phases, compliance is subject to the following specific changes.

The architect-engineer indicates that the original HCGS project commitment, via the Bechtel nuclear quality assurance manual (NQAM), was to ANSI N45.2.9 (Draft 11, Revision 0, January 17, 1973) rather than to ANSI N45.2.9-1974. The NQAM was revised to reference the 1974 document, as modified and interpreted by the guide, subject to the following specific changes:

a. ANSI Section 2.1, Quality Assurance Record System - Add the following sentence at the end of this section: "The procedures shall include control of records required during completion of the work activity."

b. ANSI Section 2.2.2, Nonpermanent Quality Assurance Records - Revise this section to read: "Nonpermanent records are those required to show evidence that an activity was performed in accordance with the applicable requirement but need not be retained for the life of the item and do not meet the criteria listed in Section 2.2.1."

c. ANSI Section 3.2.2, Index - Revise this section to read: "The quality assurance records shall be listed in an index. The index shall include, as a minimum, record retention times and the location of the records within the record system. The index system used by organizations for the retention of quality assurance

Although Regulatory Guide 1.122 is not applicable to HCGS, per its implementation section, HCGS complies with it.

For further discussion of seismic design, see Sections 3.7 and 3.10.

1.8.1.123 Conformance of Regulatory Guide 1.123, Revision 1, July 1977: Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants

HCGS compiles with Regulatory Guide 1.123 during construction and startup phases, subject to clarifications stated below. During the operations phase, item a clarification applies ~~only~~ with the exception that applicable requirements of Regulatory Guide 1.38 will be applied to code procurements where necessary to assure safe shipment. The architect-engineer indicates that the original HCGS project commitment was to ANSI N45.2.13 (Draft October 1973) rather than to ANSI N45.2.13-1976. The architect-engineer NQAM has been revised to reference the 1976 document, as modified by the Regulatory Guide, subject to the following specific changes:

a. Regulatory Guide Section C.2 - This section requires the application of elements of the ASME B&PV Code, Section III, Divisions 1 and 2, and Section XI; and ANSI N45.2.13-1976; specifically, those elements not covered by the ASME B&PV Code for procurement of ASME B&PV Code items and services. The architect-engineer takes exception to the requirement, and has the following alternate position:

   The application of the ASME B&PV Code requirements above to the procurement of ASME B&PV Code items and services is adequate, based on the fact that ASME B&PV Code represents the composite knowledge and experience of a large segment of the nuclear industry, that the ASME B&PV Code is constantly being reevaluted for adequacy, that addenda are issued frequently, and that, to our knowledge, historical data do not exist that would indicate that the ASME B&PV Code quality assurance requirements, relative to the procurement of ASME B&PV items and services, are inadequate.

b. Regulatory Guide Section C.2 - This section of the regulatory position appears to be inconsistent. It states that the purchase should verify the implementation of the suppliers corrective action

Positions C.1.1.2, C.2.1.2, C.3.1.2, and Table 1 of Regulatory Guide 1.143 require that all material specifications for pressure-retaining components within the radioactive process boundary conform to ASME B&PV Code, Section II. In addition, they require that piping materials conform to both the ASME and the identical ASTM specification, and they permit substitution of manufacturers' standards, instead of the ASME specification, in the case of pump materials. Although Regulatory Guide 1.143 does not explicitly address in-line process components, sight flow glasses, Y-strainers, and steam traps procured by the architect-engineer, and the orifice plates and conductivity elements in the NSSS scope of supply do not have certificates of compliance for the materials specified. Also, the records of shop inspection, required by Table 1, for the Y-strainers and the steam traps are not available from the supplier.

Nevertheless, the quality assurance measures taken provide the reasonable assurance needed to protect the health and safety of the public and that of plant operating personnel.

Position C.1.2.1 requires that the designated high-liquid-level conditions should actuate alarms both locally and in the control room. For all tanks, a high-liquid-level condition actuates an alarm in the radwaste control room only. There are no local alarms since the tank rooms are controlled areas and normally unmanned.

Position C.4.3 requires that process lines should not be less than 3/4 inch (nominal). The crystallizer concentrates and slurry waste transfer lines to the extruder/evaporators are 1/2 inch nominal, in order to maintain acceptable flow velocities to prevent settling in the lines. The fluid flowrates are on the order of one (1) GPM as shown in Table 11.4-7 and on Figure 11.4-9.

1.8.1.144 <u>Conformance to Regulatory Guide 1.144, Revision 1, September 1980: Auditing of Quality Assurance Programs for Nuclear Power Plants</u>

HCGS complies with Regulatory Guide 1.144~~.~~ during the operations phase. During the design and construction phase, the following clarifications apply:

The architect-engineer's quality program for safety-related items during the design and construction phases meets the requirements of ANSI N45.2.12-19.. as modified and interpreted by Regulatory

ATTACHMENT VII

QUESTION 471.14 (TABLE 13.1-2)

Provide the resumes for the Radiation Protection Engineer and the Senior Radiation Protection Supervisor (see "Areas of Review" 1.A.2 of SRP 12.5 (NUREG-0800)), or provide a schedule for submittal.

RESPONSE

Tables 13.1-2 and 13.1-4 have been revised to incorporate the resume for the Radiation Protection Engineer. The resume for the Senior Radiation Protection Engineer ~~will be~~ is provided ~~when this position is filled. The current plan is to staff this position by June 1984.~~ in Table 13.1-4.

SENIOR RADIATION PROTECTION SUPERVISOR

~~Will be provided by December 1984~~

Add Insert (c).

SENIOR RADIATION PROTECTION SUPERVISOR

~~Will be provided by December 1984~~

NAME: Leo J. Krajewski

EDUCATION AND TRAINING:

1962	Aquinas High School
1963	University of Wisconsin - LaCrosse pre-engineering credits
1963	U.S. Navy Machinist Mate "A" School
1964	U.S. Navy Nuclear Power School
1964	U.S. Navy Nuclear Power Training Unit - D1G prototype
1965	U.S. Navy Engineering Laboratory Technician School
1969	Dairyland Power Cooperative Radiation Protection Apprentice Program
1981	Quality Assurance Level II Training
1981	Biological Effects of Radiation & Radiation Measurements - University of Michigan
1981	Non-licensed Nuclear Supervisors System Training - SNGS
1982	49 CFR Regulatory Awareness Course
1982	ALARA Design Review Methods Course
1983	PSE&G Supervisory Training
1983	BWR Technology
1983	Health Physics Supervisory Training - University of Florida
1984	Technical Supervisory Skills Program (TSSP-1)

EXPERIENCE:

1981 - Present	Public Service Electric and Gas Company
1984 - Present	Senior Radiation Protection Supervisor - Hope Creek Staff, equipment, procedure and program development. Act as department head in absence of radiation protection engineer
1982 - 1984	Lead Engineer - Hope Creek Radiation Protection Construction review for ALARA design and utilization. Department initiation, organization and development, staffing, equipment and procedure planning.
1981 - 1982	Radiation Protection Supervisor - Salem Generating Station First line supervisor in surveillance group and radiation exposure permit workroom. Lead engineer activities for ALARA review, administrative assistant to radiation protection engineer and acting department head in absence of radiation protection engineer.
1969 - 1981	Dairyland Power Cooperative - LaCrosse Boiling Water Reactor Acting RPE - Radiation protection management during 3 years of operations and 2 refuelings Health & Safety Supervisor - First line supervision of department technicians and all program documentation in Health Physics and industrial safety for 4 years.

[redacted] Health & Safety Technician - Health physics and industrial safety technician during 8 years of operation and 3 refuelings.

1963 - 1969 U.S. Navy

Machinist Mate - Operation and maintenance of naval nuclear propulsion plant engineroom equipment

Instructor - In-plant instructor in engineroom operation to officer and enlisted personnel

ELT - Radiological and chemistry control of PWR systems

ATTACHMENT VIII

MAINTENANCE MANAGER

~~Will be provided by June 1984~~

INSERT

MAINTENANCE MANAGER

NAME: Peter J. Kudless

LICENSES AND CERTIFICATES:

Professional Engineer - New Jersey

MILITARY:

1976 - 1971	U.S. Navy (Active Duty)
1971 - present	U.S. Naval Reserve (currently Commander, CEC, USNR-R)

EDUCATION AND TRAINING:

196~~1~~ 2 - 1966	Worcester Polytechnic Institute, BS, Civil Engineering
1970 - 1971	University of Rhode Island, Masters level courses in Statistics, Math, Economics and Law
1975	PSE&G QA Orientation for Engineers
	ASME, Nuclear Power Engineering
1976	PSE&G, CPM of Scheduling
1977	PSE&G, ~~GAML~~ ASME Boiler & Pressure Vessel Code
	General Electric , BWR Design Survey Course
	PSE&G, Control Valves & Pipe Fittings
	PSE&G, Welding Inspection
	General Electric, B~~S~~WR Installation Course
	PSE&G, Non-Destructi~~on~~ve Examination
	PSE&G, QA Orientation for Construction Engineers

EDUCATION AND TRAINING: (cont.)

1978	PSE&G, Hydraulic & Friction Crane Operation, Maintenance & Safety
	AMR Internation, Project Management
1979	PSE&G, Supervisory Training Program
1980	PSE&G, Strategies of Effective Listening
	PSE&G, QAD Follow Up Training for Engineering & Construction Personnel
1981	Rutgers University, Advance Management Training

EXPERIENCE:

1971 - Present	Public Service Electric and Gas Company
6/84 - Present	Maintenance Manager - HCGS: Responsible for management, direction and control of the work of the Maintenance Department. Assure conduct of electrical and mechanical maintenance activities is in accordance with facility license, company and government regulations. Assure maintenance activities are accomplished safely and efficiently by properly trained and qualified personnel. Assure that maintenance is conducted safely and efficiently during outages to achieve maximum possible unit availability and reliability. Develop and control budgets for the Maintenance Department. Assure a cost effective spare parts inventory. Act as Vice Chairman and member of the Station Operations Review Committee.
12/80 - 6/84	Project Construction Manager, HCGS: Responsible for monitoring the field construction efforts of a 1100MW BWR power plant. Managed a staff in excess of 50 personnel (Civil, mechanical, electrical, HVAC, cost & scheduling engineers and administrative personnel) who controlled the over field construction and support work of 4000 personnel. Also responsible for site security contract.

EXPERIENCE: (cont.)

11/78 - 11/80 Principle Construction Engineer - HCGS: Supervised a staff of 20 Construction engineers (all disciplines) who monitored and controlled all the field construction effort of over 3500~~+~~ personnel.

7/76 - 11/78 Senior Construction Engineer - HCGS: Supervised a staff of 5 Construction Engineers who monitored and controlled the field construction efforts of over 1500~~+~~ personnel in the power block area of the plant.

3/75 - 7/76 Associate Field Representative - Assigned to the corporate home offices, with ~~rational~~ rotational field assignments, providing staff support (i.e., procedure, specification, drawing review) in prepa~~??~~ration for the start of the full site construction effort on the HCGS.

8/71 - 7/75 Engineer - Gas Engineering Dept. ~~PN~~ for 2 1/2 years, was senior site representative supervising a staff of 15 construction engineers and administrative personnel who monitored and controlled the site construction and support efforts for a ~~new~~ Synthetic Natural Gas (SNG) plant. For 1 1/2 years provided home office staff support for construction of the first SNG plant constructed in this country, and a peak sharing Liquified Natural Gas (LNG) storage facility.

1966 - 8/71 U.S. Navy - Served on active duty as a Civil Engineer Corps (CEC) Officer. Attended Naval OCS and CEC officers school. After commissioning, served at Newport, R.I., and Republic of Viet Nam. While at Newport, R.I. was assigned as Assistant Operation officer for 1 1/2 years and a Resident Officer in Charge of Construction (ROICC) 1 1/2 years at the Navy Public Work Center~~s~~. In the former position was responsible for directing the day to day operations of 3 divisions (Utilities, Maintenance and Transportation) of the Operations Department, which provided the services to all the Naval commands (approximately 10,000 personnel) at the Newport Naval Base. In the latter position, was responsible for the administ~~ering in excess of~~ ration of greater than

EXPERIENCE: (cont.)

10 million dollars of construction contracts including utility (i.e., boiler repair), vertical and horizontal construction projects. While serving 1 year as a ROICC in Viet Nam, I stet was assigned to two different field locations to administer construction contracts in excess of 1 billion dollars. Type of projects included power plants and utility systems, waterfront, vertical and horizontal construction.

SENIOR/Nuclear MAINTENANCE SUPERVISOR

NAME: Thomas B. Wysocki

EDUCATION:

1958	Greenport High School
1962	Wentworth Institute, Credits toward Associate Degree in Architecture
1962	Franklin Institute, Credits in Math and English
1973	Delaware Community College, Credits in Computers and Math
1975	Salem Community College, Associated Degree Construction Technology

TRAINING

1963	U.S. Navy Machinist Mate A School
1964	U.S. Navy Submarine School
1965	U.S. Navy Nuclear Power School
1966 - 1970	Various U.S. Navy Service Schools on Nuclear Submarine Equipment
1974	Quality Assurance Orientation Course
1977	Quality Assurance Visual Examiners Level I and II Training Course
1980	Quality Assurance Liquid Penetrant Method of Nondestructive Examination Course
1980	Quality Assurance Magnetic Particle Method of Nondestructive Examination Course
1981	BWR Technology Course
1983	Aberrant Behavior Identification and Leadership Skills
1983	Advance Supervisory Training Program
1984	Diesel Engine Training

EXPERIENCE

1970 - Present	Public Service Electric and Gas Company
1981 - Present	Senior maintenance supervisor for Hope Creek Generating Station: Responsible for planning assignments, assigning manpower, and establishing work priorities. Involved with the maintenance department establishment, spare parts acquisition, system and facility design reviews, procedure writing, and preparing for the startup test program.
1975 - 1981	Maintenance supervisor for Salem Generating Station: Responsible for the direct planning and execution of maintenance activities. Direct supervisor of bargaining unit work force
1972 - 1975	Startup test engineer for Salem Generating Station: Duties included developing, writing, and directing startup procedures. Prepared test reports and evaluated test results
1970 - 1972	Junior staff assistant for corporate office: Assisted the production department performance engineer in collecting data and preparing reports
1965 - 1970	Machinist mate, U.S. Navy: Served as a mechanical repairman/operator on S5W pressurized water reactor. Responsibilities included repairing, testing, and operation of the reactor and associated systems

SENIOR NUCLEAR MAINTENANCE SUPERVISOR

add (A)

~~Will be provided by June 1984~~

NUCLEAR

SENIOR MAINTENANCE SUPERVISOR

NAME: MARK SHEDLOCK

EDUCATION:

1971 - Colonia High School
1975 - Union Junior College
A.A. Degree in Engineering
1977 - Rutgers University - College of Engineering
B.S.E.E. in Electrical Engineering, Magna Cum Laude

TRAINING:

1977 - Westinghouse P.W.R. Course
1977 - QAD Course #25 for P.C.D. Personnel
1977 - Non-Destructive Examination Course
1978 - 23rd Annual Appalachian Underground Corrosion Short Course
1978 - Seminar on State of the Art Power Plant Construction
1978 - QAD Course #5 for Orientation Training for Engineers
1978 - QAD Course #27 for Reg. Guide 1.58 & ANSI N45.2.6
1979 - CCM-5 Construction Backcharges/Subcontracts Course
1979 - CCM-3 Construction Scheduling Course
1979 - CCM-2 Construction Accountability Course
1979 - CCM-1 Overall Construction Process Course
1984 - QA Training for Hope Creek Start-up and Test Personnel

EXPERIENCE:

1977 to Present - Public Service Electric & Gas Company,

1983 to Present - Senior Maintenance Supervisor for Hope Creek Generating Station: Responsible for planning assignments, assigning manpower and establishing work priorities for maintenance activities and start-up testing activities. Involved with the Maintenance Department establishment, system and facility acceptance, acquisition of department tools and test equipment & procedure review and approval.

(A)

EXPERIENCE (cont.):

1979 to 1983 - Resident Electrical Engineer for Salem Generating Station, responsible for resolution of electrical operating problems to maintain unit reliability; develop specifications for subcontract work packages; modification to system design to suit station conditions; review new design for constructability and material needs; site fire protection coordinator; interface with maintenance contractor and home office Engineering regarding new work items and NRC commitments; member - Salem Emergency Response Team.

May 1979 to Nov. 1979 - Public Service Electric and Gas Company
Hope Creek Generating Station
Construction Engineer responsbile for monitoring the progress and quality of electrical, control and instrumentation activities in the power block; procurement review for permanent plant and construction equipment and materials; subcontract administration encompassing specification review, bid invitation, contract award and progress to closeout.

July 1977 to May 1979 - Public Service Electric and Gas Company
Salem Generating Station - Unit No. 2
Acting Lead Construction Engineer - Electrical responsible for integration of discipline activities; review and approve Engineering changes; direct administration of Electrical subcontractor work packages; review and revise work package specifications; provide functional direction to Field Engineers and Designers. Identify and resolve all construction problems; certified Start-up/Test Engineer - Level II.

1975 - 1981	Instrument supervisor - Salem Generating Station: Supervised I&C technicians in all aspects of instrumentation system maintenance, testing, and calibration. Functioned as department planning coordinator for initial startup of Unit 2 and two refueling outages for Unit 1
1972 - 1975	I&C technician for Salem Generating Station: Performed maintenance, calibration, and testing activities on instrumentation and control systems
1965 - 1972	Electronics technician for U.S. Navy: Performed maintenance and testing of reactor plant instrumentation systems
	Reactor operator: Daily operations of submarine S5W reactor plant
	Engineering watch supervisor: Supervised activities of technicians, machinists, and electricians in the varied aspects of submarine reactor plant operation and maintenance