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NTD-NRC-95-4460
DCP/NRC0325
Docket No.: STN-52-003

May 10, 1995

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

ATTENTION: T. R. QUAY

SUBJECT: PRESENTATION MATERIALS FROM THE MAY 10, 1995 MEETING ON AP600 REACTOR VESSEL ISSUES

Dear Mr. Quay:

Enclosed are presentation materials used during a meeting at the Westinghouse offices in Monroeville, PA on May 10, 1995 to discuss AP600 reactor vessel issues. Also enclosed are presentation materials used during the meeting on the afternoon of April 11, 1995 to discuss AP600 PCS Test and Analysis Program activities. The following materials are enclosed:

Enclosure 1 Westinghouse presentation materials, May 10, 1995 Reactor Vessel Issues (proprietary)

Enclosure 2 Westinghouse presentation materials, May 10, 1995 Reactor Vessel Issues (nonproprietary)

The Westinghouse Electric Corporation copyright notice, proprietary information notice, application for withholding and affidavit are attached.

This submittal contains Westinghouse proprietary information consisting of trade secrets, commercial information or financial information which we consider privileged or confidential pursuant to 10CFR9.17(a)(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handles on a confidential basis and be withheld from public disclosures.

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Correspondence with respect to the application for withholding should reference AW-95-822, and should be addressed to N. J. Liparulo, Manager of Nuclear Safety Regulatory and Licensing Activities, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania, 15230-0355.

Please contact Brian A. McIntyre on (412) 374-4334 if you have any questions concerning this transmittal.

JM Brennan For

Nicholas J. Liparulo, Manager
Nuclear Safety Regulatory and Licensing Activities

Enclosures

cc: T. Kenyon, NRC
B. A. McIntyre, Westinghouse (w/o Enclosures)

Enclosure 2 to Westinghouse Letter NTD-NRC-95-4460

[Open Issue 3.7.1-3]

On Page 3.7-2 and in Table 3.7-1 of the SSAR, the damping ratio assigned for the fuel assemblies is 20%. Provide the basis that justifies the use of this damping value.

[Response]

The damping factor used in a fuel assembly is used in determining the fuel assembly relative deflection and the grid impact force responses, as well as the load transfer among the fuel assemblies in a closely packed reactor core. Fuel assemblies are represented in two models. A reactor vessel and internals model is used to calculate stresses and loads, to evaluate the internals and vessel, and to develop seismic interface motions used in the reactor core finite element model to determine the fuel assembly responses. These models are discussed below with respect to the consideration of damping within the fuel. The basis of the damping values used for the fuel assemblies is dicussed below.

Reactor Vessel and Internals Model

The fuel assemblies are combined in the model so that all assemblies are considered to be in phase during the seismic (or LOCA) event. This is conservative for defining the interface loads and stresses within the vessel and internals due to the fuel. It is noted that the fuel dominant response is due to the fundamental mode in the horizontal directions. The vertical modes are not limiting in terms of the fuel assembly dynamic responses. Time history seismic analyses are performed. The structural fuel damping is represented by viscous damping. The damping value used is 20% of critical damping.

The reactor pressure vessel/internals/fuel system seismic analyses are performed using Modal Superposition technique of the WECAN Computer code. In performing these nonlinear system dynamic analyses through modal superposition, composite modal damping is used. For the SSE analysis, four (4) percent of the critical damping is used for the structural members while twenty (20) percent of the critical damping is used for the fuel elements.

Reactor Core Finite Element Model

This finite element model is used to obtain fuel assembly dynamic responses. The maximum number of assemblies along a principal axis of the reactor core is modeled in this finite element model. Thus, the reactor core model is an assemblage of 15 fuel assembly sub-models with grid spacer impact elements in an AP600 plant. Each individual assembly displacement is limited by the gap clearances between neighboring assemblies or the baffle plate. The reactor core model contains gaps to represent the geometric non-linearities caused by the inter-assembly gaps and the clearances between the peripheral fuel assemblies and core barrel (baffle plates). The seismic inputs, including the displacements of lower core plate, upper core plate, and upper core barrel motions, are simultaneously applied to the reactor model.

The AP600 fuel assembly is a 17x17 tubular array structure. The basic assembly structure consists of 264 fuel rods, 24 guide thimble tubes, and one centrally located instrumentation tube. The fuel rods are inserted into the skeleton assembly and retained in position by the friction of the spacer grid springs. The fuel assembly has a total of thirteen grid spacers: two Inconel end grids, seven internal low-pressure-drop Zircaloy grids, and four intermediate flow mixer grids. The fuel rods are supported by the grid springs and dimples. Like all other PWR fuel designs, the interactions between the fuel rods and supporting grid springs and dimples result in a high assembly damping characteristics. Thus, the damping data derived from other PWR fuel designs are also applicable to the AP600 design.

Damping is represented for the fuel assembly portion of the model using Rayleigh's damping. In this procedure, a combination of stiffness and mass damping coefficients is used to approximate the damping characteristics. Proportional damping coefficients for mass and stiffness matrices are defined based on the upper and lower response frequencies (fs and fr) and the critical damping coefficient at these frequencies (ζs and ζr). Values for ζs and ζr were determined from the distribution of damping data. The first mode (fundamental frequency) has a damping value of 20%; for a higher mode, the damping value of [ ]a,c is used. Therefore, ζs is equal to 20%, and ζr is equal to [ ]a,c In Figure 1, a plot of the critical damping coefficients versus the response frequency is shown. This plot is representative of the damping characteristics for the fuel assembly portion of the model.

The fundamental frequency of the fuel is approximately 3 hertz.

Basis of Fuel Damping

The fuel assembly damping values are obtained from mechanical tests conducted in both air and water environments. A typical test setup in air conditions is that the fuel assembly was positioned vertically in the test stand and restrained at the top and bottom fuel nozzles with core plate simulators typical of reactor support conditions. The fuel assembly was preloaded prior to testing. Linear voltage differential transformers (LVDTs) are placed at each grid location to monitor grid displacements. An electrodynamic shaker was attached to a grid elevation. A constant shaker force was applied normal to the fuel assembly cross section for each test. The output from LVDTs opposite the shaker location was stored on the spectrum analyzer. Fuel assembly tests in water conditions were performed in a specially constructed cylindrical test vessel. The sealed LVDTs or fiber optical transducers are used to monitor the fuel assembly movements.

The damping measurements conducted in water reveal a significant increase in damping characteristics due to the effects of the water. The measurements of various types of pressurized water reactor (PWR) fuel assembly designs exhibit a consistent trend in damping characteristics. The measurements indicate that the damping value tends to increase as a result of increasing fuel assembly vibrational amplitude. Fuel assembly damping values were also obtained for a typical PWR fuel assembly under operating flow conditions. These results provide damping values that are expected for the actual core assembly. These results also showed, that the damping increased as the vibration amplitude increased. The best-estimate damping coefficient for the fuel assembly fundamental mode was well in excess of 20% of the critical damping factor, even for small vibrational amplitudes. The best-estimate damping coefficient for the fundamental mode is used since this represents a composite damping value representative of many fuel assemblies that are combined into composite fuel assembly finite element models as described above.

The measurements of the fuel assembly damping are performed using the following testing methods:

- o Frequency sweep by increasing and decreasing the value of frequency with constant force
- o Decay method by using either shaker decay or plucking fuel assembly
- o Random vibration data (Reference 1)

Shown in Figures 2 to 4 are representative test and analysis results applicable to the AP600 design. As seen from these figures, damping values are dependent on the relative fuel assembly vibrational amplitude. The higher value occurring at large amplitude is due to the relative motion of the fuel rods within the grid cell. The high fuel assembly damping values are attributed to a combination of hydrodynamic effects, fuel rod slippage, and inter-fuel assembly rubbing in a closely packed reactor core. Under a postulated faulted transient such as an SSE, the fuel assembly deflection amplitude generally reaches the physical limit imposed by accumulated inter-fuel assembly gaps. For these a,c motions, the maximum fuel assembly deflection is in excess of [ ] inch, which is a moderate amplitude. As seen in Figure 2, which includes flow conditions, the percent critical damping is in excess of 20% critical damping at a nominal flow rate of 10.6 ft/sec for the AP600 design. The fuel assembly damping characteristics were discussed in References 2 through 4. As indicated in Reference 3, a typical fuel assembly damping value corresponding to a 10 mm amplitude in air is about 18% of critical value, 25% in still water, and 45% at 4.55 meter/sec. The published fuel assembly damping data are consistent with Westinghouse test data.

In Figure 3, the best estimate damping for the fuel assembly with in-core conditions as a function of frequency is shown, Reference 1. The fuel assembly damping values given in Figure 2 correspond to very small fuel assembly vibrational amplitudes. The fuel assembly damping value increases as the assembly vibrational amplitude increases. As seen by this curve, the best estimate damping for the fundamental mode (~3 hz) is 20 % or higher. The higher significant modes (< 7 hz, as seen in Figure 4) have best estimate damping values in excess of 10% of critical damping. The higher damping evident in the fundamental modes is attributed to a combination of hydrodynamic and mechanical damping. Based on the test results, it was concluded that in order to accurately predict the fuel assembly dynamic response during the reactor core seismic analyses, a uniform [ ] damping value is used for all modes to simulate the mechanical damping, and an additional [ ] is conservatively included for the fuel assembly fundamental mode to account for the hydrodynamic effects.

From a review of the fuel assembly damping data from mechanical tests performed in air, in still water and flowing water conditions, as well as from in-core neutron detectors, it is evident that the PWR fuel assembly is a highly damped structural system. The fuel assembly damping is a result of combined effects of inter-fuel assembly rubbing and scraping, fuel rod/support constraint relative motions and frictional forces, and fluid/structure interactions in a closely packed reactor core. The fundamental mode of a fuel assembly was identified as the predominant mode for fuel dynamic analysis. Therefore, a 20% damping value for the fuel assembly dynamic analysis is conservative.

References:

1. Bryan, W. J., "In-Core Detection of Nuclear Fuel Assembly Vibration," ASME Publication 79-DET-43, Design Engineering Technical Conference, Sept. 10-12, 1979

2. Stokes, F. E. and King, R. A., "PWR Fuel Assembly Dynamic Characteristics," International Conference on Vibration in Nuclear Power Plants, Keswick, United Kingdom, May 9-12, 1978 (BNES), pp27-46

3. Flamand J. C., et al, "Influence of Axial Coolant Flow on Fuel Assembly Damping for the Response to Horizontal Seismic Loads," SMiRT 11 Transaction Vol C, August, 1991 pp139-144.

4. Tanaka, M, et al, "Parallel Flow Induced Damping of PWR Fuel Assembly," ASMS PVP Vol. 133, 1988, pp121-125

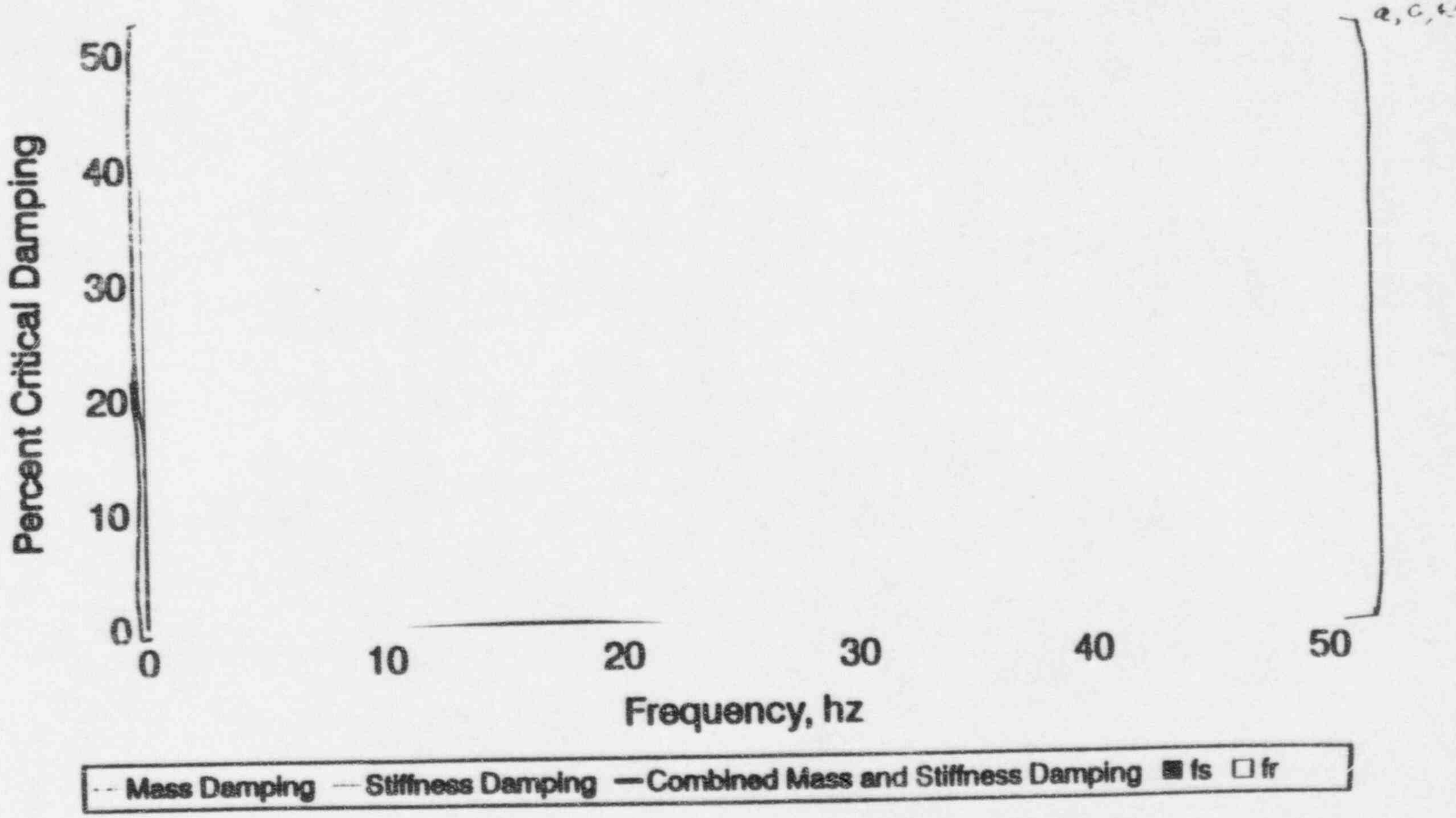

Figure 1 Combined Mass and Stiffness Damping for Fuel Assembly Model

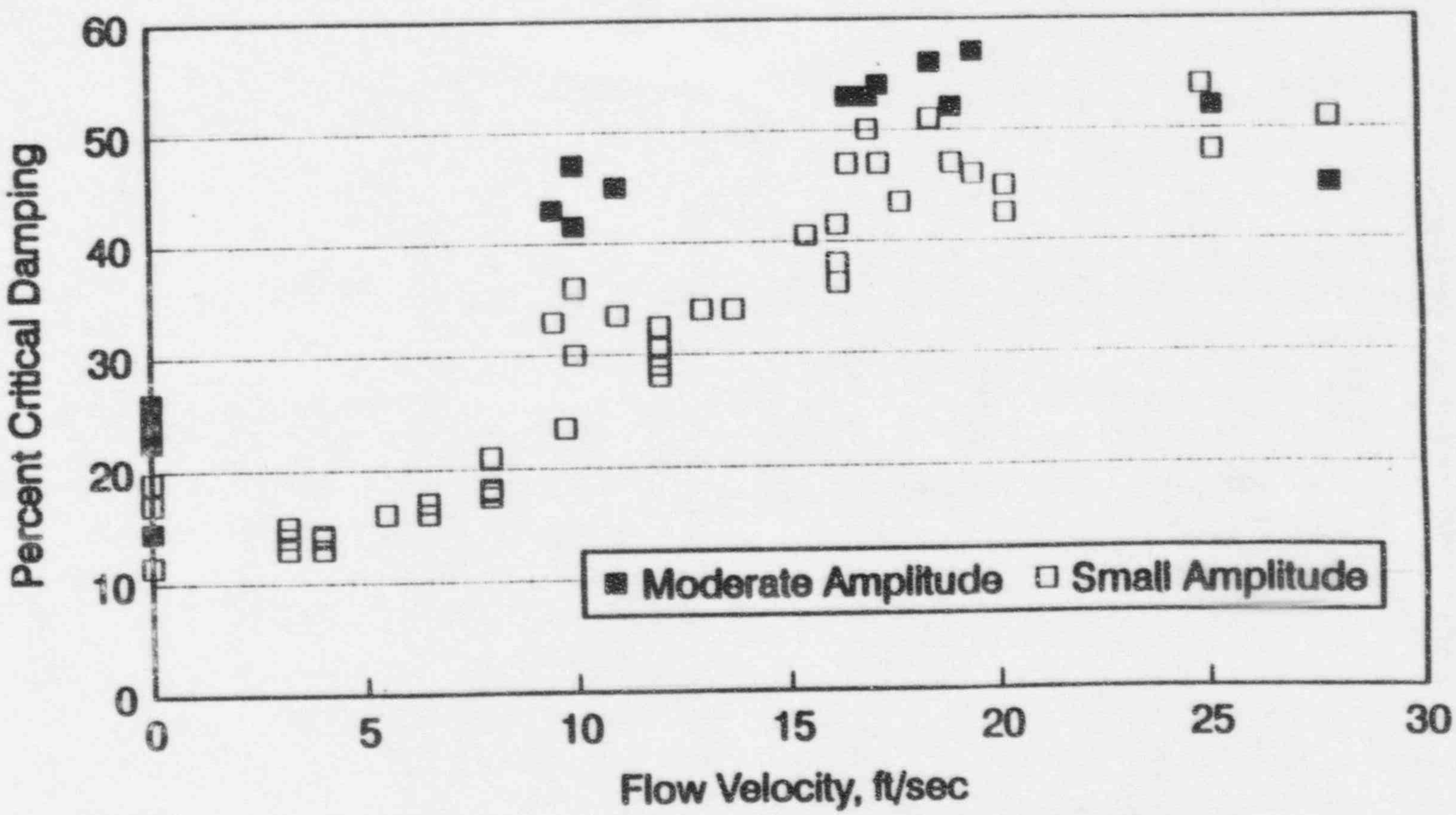

Figure 2 Fuel Assembly Damping versus Flow Velocity

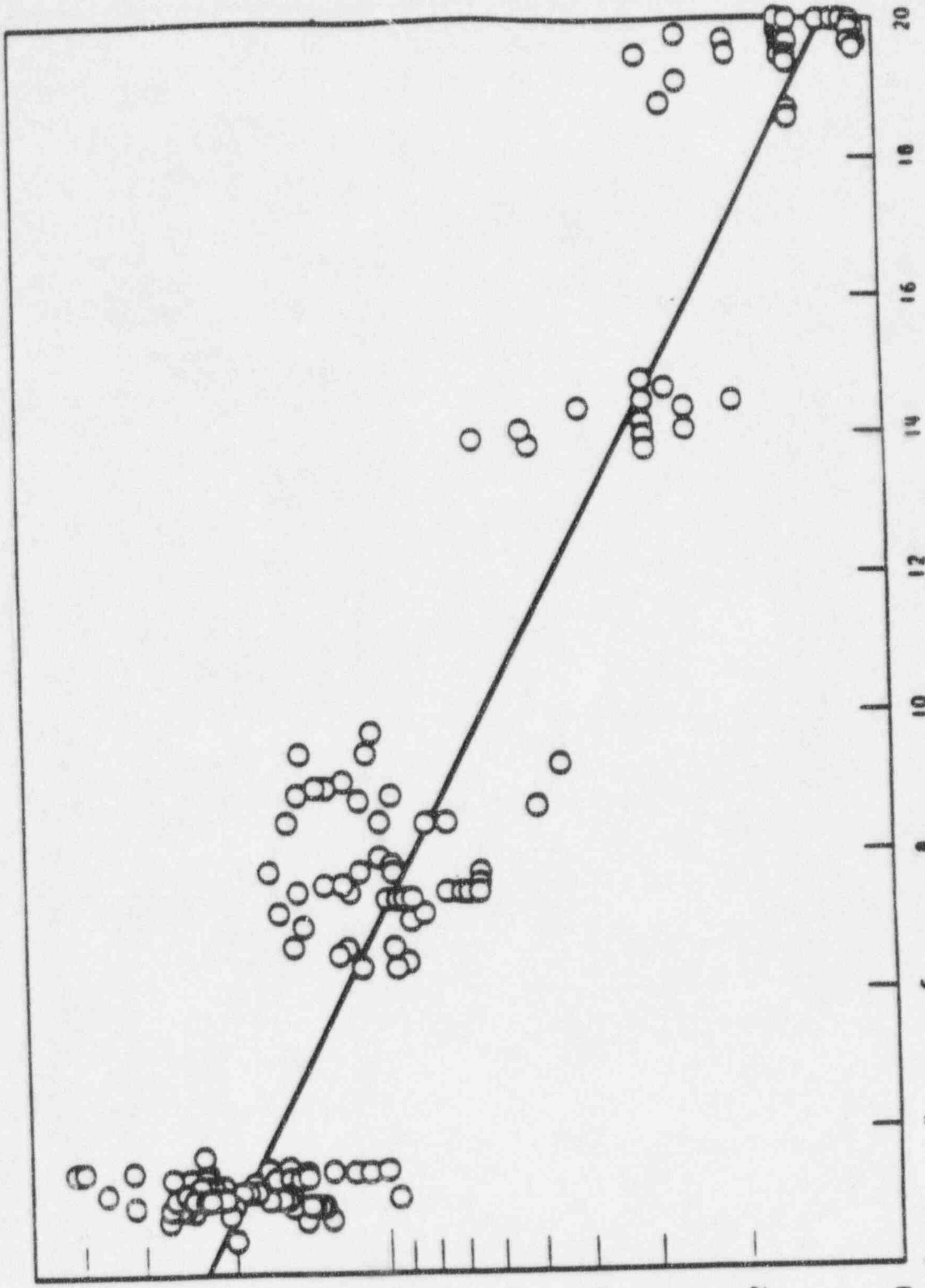

Figure 3 Best Estimate Damping Curve for the Assembly, In-Core Conditions

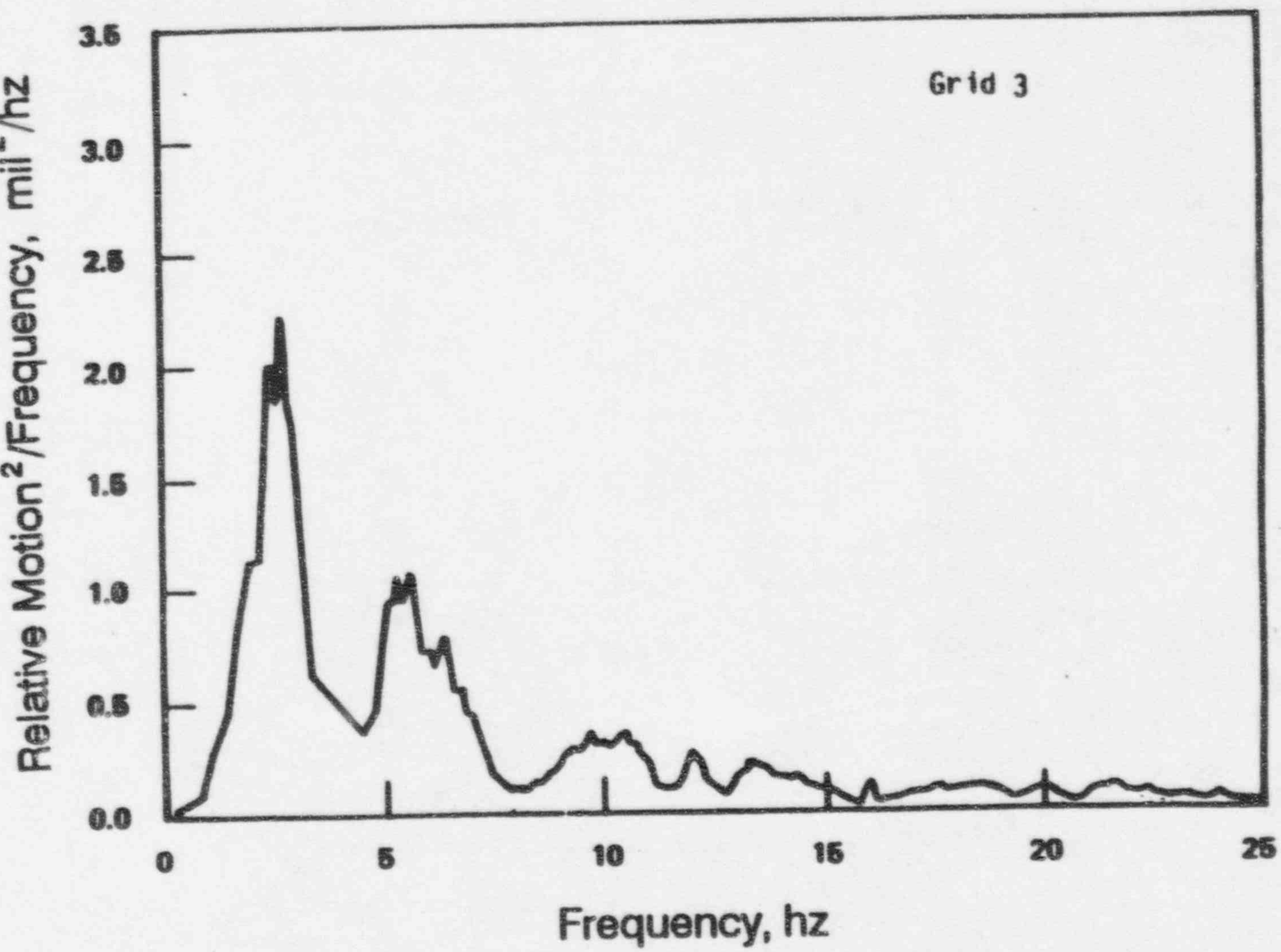

Figure 4 Typical Power Spectral Density Curve

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