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TO: Mr Stello

FROM: Florida Power & Light Co
Miami, Fla
R E Uhrig

DATE OF DOCUMENT: 10-15-76

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DESCRIPTION:
Ltr re our 8-20-76 ltr.....trans the follow:

PLANT NAME:
Turkey Point 3 & 4

ENCLOSURE:
Supplementary Info to their proposed tech specs change submitted 8-30-76 concerning spent fuel storage facility modifications ....(40 cys encl rec'd)

ACKNOWLEDGED

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FOR ACTION/INFORMATION — SAFETY — ENVIRO 10-22-76 ehf

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October 15, 1976
L-76-360

Office of Nuclear Reactor Regulation
Attn: Victor Stello, Jr., Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

RECEIVED OCT 20 1976 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

Dear Mr. Stello:

Re: Turkey Point Plant Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed Amendment to Facility
Operating Licenses DPR-31 and DPR-41
Supplementary Information

Mr. Lear's letter received by us August 20, 1976, requested additional information relating to our proposed spent fuel storage facility modification.

Included in this request were fourteen (14) questions requesting supplemental information to our letter dated April 30, 1976.

Enclosed are the responses to questions 2 through 14. Response to Question 1 is expected to be completed by October 24, 1976, at which time it will be promptly forward to you.

Forty (40) copies of this transmittal, including three (3) signed originals are enclosed.

Very truly yours,

for Robert E. Uhrig
Vice President

DOCKETED USNRC OCT 20 1976 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

REU/BWA/hlc
Enclosures

cc: Norman C. Moseley, Region II
Jack R. Newman, Esq.

10628

QUESTION 2 - Describe the design procedure and permissible stresses for the stainless steel welded joints.

Answer - The reactions within the fuel racks are determined for the combined normal operating and maximum seismic loadings. The welds are then sized such that the tensile and shear stresses do not exceed those allowed per Table NF-3292.1.1 (ASME, Sec. 3).

QUESTION 3 - The response to Question 21 with regard to the design of the fuel storage racks is not acceptable. It is the staff's position that the design and analysis of the new rack system be performed using either the existing input parameters including the old damping values or new parameters in accordance with Regulatory Guides 1.60 and 1.61. The use of existing input with the damping values in Reg. Guide 1.61 is not acceptable. Furthermore, seismic excitation along three orthogonal directions should be imposed similtaneously for the design of the new rack system.

Answer - The seismic analysis of the racks was performed with a damping value of 2%. This is consistent with the value listed in the table on page 5A-13 of the FSAR for welded steel framed structures.

In the previous response to question 21, page 7a, paragraph 2, should read as follows:

> Seismic analysis of the fuel storage racks are performed by the time history method for each of the two orthogonal horizontal directions, and by the response spectrum method for the vertical direction. The time histories and response spectrum utilized in these analyses represent the responses of the pool structure to the specified ground motion. The seismic analysis of the racks was performed with a damping value of 2%. This is consistent with the value listed in the table on page 5A-13 of the FSAR for welded steel structures. Rack member and support loads and displacements are obtained by combining the absolute values of the maximum vertical and maximum horizontal response for each horizontal direction.

QUESTION 4 - Section 3.2 indicates that the time history method will be used in the design of the racks. Please provide a comparison of the response spectra derived from the time history and the design response spectra for the damping values that are used in the design.

Answer - The comparison of the acceleration response spectra of the horizontal time history with the horizontal design spectra is shown on the attached two figures for 2% and 4% damping values (see figures 7 and 8).

QUESTION 5 - With regard to Question 22, only the mathematical model of the fuel pool has been provided. Please provide the information requested for the storage rack and fuel assembly system. The effects of gaps, sloshing water, and increase of effective mass and damping due to submergence in water should be quantified.

Answer - Sketches of the rack and fuel models used in the seismic analyses of the fuel storage racks are shown in Figure 1 and Figure 2. The load transfer to the pool structure from the fuel racks occurs only at the base of the racks and consists of the vertical compression loading and horizontal shear forces due to frictional restraint.

The effects of seismic excitation of the pool water (sloshing) were analyzed using the procedure in the reference. This analysis showed that the rack is below the depth where sloshing forces are effective and, therefore, should not experience excitation from this source. However, for conservatism, drag forces on the rack were calculated using the peak pool fluid velocity due to sloshing. The results show total drag forces on the entire rack structure in the pool to be sufficiently small to ignore.

The effects of gaps and submergence in water significantly affect the motion of fuel racks and must be accounted for in the seismic response analysis. The fuel, which responds to seismic excitation at its own natural frequency, will move freely through the available gap and impact the storage cavity. As the fuel moves within the rack and the rack moves relative to the pool, the water between these structures is accelerated by them. This acceleration of the water introduces hydraulic loads on these structures and results in a lowering of natural frequencies of the fuel and rack. These hydrodynamic effects are accentuated when the interacting submerged structures are in close proximity (small gaps).

The effects of gaps and submergence in water are accounted for directly in the equations of motion of the fuel rack model. To illustrate this, a simplified analog of the spent fuel rack problem (Figure 3) is considered.

The three concentric cylinders represent the pool (P), the rack (R), and the fuel (F). There is water between the fuel and the rack, and between the rack and the pool. The connection (spring K_G) between the fuel and the rack represents the gap between these structures, as well as the impact stiffness with which the fuel spacer grids interact with the rack when in contact. The connection (spring K_R) between the rack and the pool represents the manner in which the rack is supported by the pool. Nomenclature is as follows:

$\ddot{X}_p$ = seismic excitation (acceleration time-history) at spent fuel pool elevation

$\ddot{\delta}_R$ = acceleration of rack (relative to pool)

$\ddot{\delta}_F$ = acceleration of fuel (relative to pool)

δ_R = displacement of rack (relative to pool)

δ_F = displacement of fuel (relative to pool)

M_R = mass of rack

M_{R_D} = mass of water displaced by rack

M_{R_C} = mass of water contained within rack

M_F = mass of fuel

M_{F_D} = mass of water displaced by fuel

$F_{R_{IN}}$ = fluid force on inner boundary of rack

$F_{R_{OUT}}$ = fluid force on outer boundary of rack

$F_{F_{OUT}}$ = fluid force on outer boundary of fuel

$\alpha_1, \alpha_2, \beta, \gamma$ = factors describing the affect of geometric proximity on hydrodynamics

K_R, K_G = as defined above

With reference to the above nomenclature and figure 3, and neglecting damping terms for purpose of simplifying discussion, the following equations of motion can be developed:

$$M_R(\ddot{X}_p + \ddot{\delta}_R) = -K_R(\delta_R) + K_G(\delta_F - \delta_R) + F_{R_{OUT}} + F_{R_{IN}} ,$$

$$M_F(\ddot{X}_p + \ddot{\delta}_F) = -K_G(\delta_F - \delta_G) + F_{F_{OUT}}$$

The fluid forces are given by:

$$F_{R_{OUT}} = M_{B_D} (\ddot{X}_p - \alpha_1 \ddot{\delta}_R),$$

$$F_{R_{IN}} = M_{R_C} (-\ddot{X}_p + 2\beta \ddot{\delta}_F - \alpha_2 \ddot{\delta}_R),$$

$$F_{F_{OUT}} = M_{F_D} (\ddot{X}_p + 2\gamma\ddot{\delta}_R - \alpha_2 \ddot{\delta}_F).$$

Substitution of these expressions for fluid forces into the two equations of motion and simplication of terms yields the required equations corresponding to the physical problem:

$$(M_R + \alpha_1 M_{R_D} + \alpha_2 M_{R_C})\ddot{\delta}_R - (2\beta M_{R_C})\ddot{\delta}_F + (K_r + K_G)\delta_R - K_G\delta_F = -(M_R + M_{R_C} - M_{R_D})\ddot{X}_p$$

$$-(2\gamma M_{F_D})\ddot{\delta}_R + (M_F + \alpha_2 M_{F_D})\ddot{\delta}_F - K_{GR} + K_G\delta_F = -(M_F - M_{F_D})\ddot{X}_p$$

The equations account for the gap between the fuel and the rack, the hydrodynamic coupling between the submerged structures and impacting between structures. The complete equations of motion (including damping) corresponding to the physical situation are modeled and solved through the use of CESHOCK.

In contrast to the CE approach described above, the response spectrum method has also been used. In the response spectrum analysis of spent fuel racks, the lowest natural frequency of the rack structure is first calculated. The mass of the fuel is lumped together with that of the rack in this calculation; some water mass is also added. From the response spectrum curve, a "g" value (of response acceleration) is obtained corresponding to the calculated frequency. The "g" value is then used in conjunction with the weight of the rack and fuel to obtain the seismic loads. However, the response spectrum method can accommodate only a single uncoupled equation for the response of a one-degree-of-freedom system. Modifying the response spectrum method to include an approximation of the affect of water on frequency, the analogous equation of motion for the system of figure 3 that corresponds to the response spectrum method of analysis is:

$$(M + M_C + M_D)\ddot{\delta} + K\,\delta = -\,(M + M_C)\,\ddot{X}_p$$

Here the representation of the system is clearly incomplete with many approximations (of unknown effect) required to select the single values of mass, stiffness (<u>linear only</u>), etc., allowed. Comparison with the two equations above demonstrates the point that the response spectrum method does not model the real physical situation. It does not account for the gap between the fuel and the rack which causes the system to have different natural frequencies (and to respond to different frequencies of excitation) and allows fuel/rack impacting to occur. Also, it does not account for the hydrodynamic coupling between the fuel and rack with its introduction of interactive fluid forces.

A number of spent fuel rack seismic analyses have been performed by CE, covering a wide range of rack designs and seismic excitations. In order to quantify the effects of gaps and hydrodynamic coupling, the following summary of representative results from these nonlinear time history analyses (utilizing CESHOCK) is presented, together with a comparison to corresponding results from a response spectrum method analysis.

Figure 4 provides a visual description of four different seismic excitations used in obtaining the results to be described. The response spectra are shown only to illustrate the differences in the excitations corresponding to the four sites; time-histories for these sites were used in the CESHOCK analyses.

Figure 2 represents a typical CESHOCK model. The particular model shown corresponds to a CE "freestanding" design. The fuel is modeled by masses 9 through 15 and spring K_{F_1} through K_{F_6}; the rack is modeled by masses 1 through 8 and springs K_{R_1} through K_{R_7}; the hydrodynamic coupling between the rack and the fuel is represented by the coupling H_1 and H_6 and between the rack and pool wall is represented by H_7 through H_{12}; the fuel-to-rack gaps and fuel-to-rack impact characteristics are modeled by the nonlinear springs K_{G_1} through K_{G_6}; the frictional restraint between the fuel and the rack and that between the rack and the pool are represented by the friction couplings F_{F-R} and F_{R-P}, respectively.

Figure 5 displays a brief segment of typical displacement responses to the seismic excitation corresponding to Site II. Note the low-amplitude, high-frequency response of the rack portion of the model in contrast to the high-amplitude, low-frequencies response of the fuel.

Figure 6 presents a tabulation of seismic loads developed within the rack and transmitted to the pool for a number of designs and the four sites of Figure 4. The load values have been normalized. The first column identifies the site and the rack design. Four variations of design D are shown: The original version, a second version in which dynamic analysis parameters were changed by 10% (e.g., fuel stiffness), a third version with one-fourth the original fuel-to-rack gap, and a fourth version with an impact spring stiffness ten times that of the original. The second column presents the seismic loads obtained from the CESHOCK analyses. The third column presents the corresponding seismic loads obtained, for comparative purposes, by means of response spectrum method analyses. The last column gives the ratios of loads obtained by the two methods.

Comparison of results from nonlinear time-history analyses (fuel rack interaction analyses) with those from response spectrum analyses (refer to Figure 6) proves that the response spectrum method gives incorrect results.

These results demonstrate the importance of accounting for the interaction between fuel and racks. This interaction is caused by the relative motion, through the water-filled gaps between fuel and rack and impacting of fuel and rack.

QUESTION 6 - With regard to Question 23, if the racks are laterally supported from the fuel pool walls, describe how these loads were included in the re-analysis of the pool structure.

Answer: The high capacity storage racks are free-standing with no lateral restraints.

QUESTION 7 - It is not clear if the racks will be anchored to the base slab of the pool. If so, describe the design of the anchorage and the materials used.

Answer: The racks will not be anchored to the base slab of the pool. Please refer to the response to Question 19 of the August 3, 1976, submittal which states that the lateral loads from the racks will be transferred by friction between the racks and the embeds to the slab and that the transfer of loads between the embed and the new slab will be by shear.

QUESTION 8 - Section 3.2 indicates that the time history analysis is performed with the SHOCK computer code. Describe the procedure used to validate this program. Acceptable procedures for validation are delineated in paragraph II.4.3 of Standard Review Plan 3.8.1.

Answer: The procedures used to validate the SHOCK computer code include a combination of the procedures (ii) and (iii) of Paragraph II.4.e of Standard Review Plan 3.8.1.

The following is a summary of the methods employed to verify the SHOCK code.

Type of SHOCK Problem	Method of Verification
I. Nonlinear, damped pulse excited system a) Degree of nonlinearity varied (duffing oscillator) b) Type of pulse varied	Published classical solution
II. Multi-degree of freedom base excited lateral system with friction, gaps and damping a) Without hydrodynamic mass b) with hydrodynamic mass	Independent Computer Code (ANSYS)
III. Friction element with static and dynamic coefficients of friction a) Coefficient of friction velocity independent b) Coefficient of friction velocity dependent	Analytical solution
IV. Impact element with coefficient of restitution	Analytical solution
V. Hysteresis element a) Single DOF system b) Two DOF system with spring and gap elements	Independent Computer (ANSYS)

Additional validation of this code is discussed in Appendix B of CENPD-42, "Topical Report on Dynamic Analysis of Reactor Vessel Internals Under Loss-of-Coolant Accident Combinations with Application of Analyses to CE 800 Mwe Class Reactors," August 1972 (Proprietary).

QUESTION 9 - The response to Question 31(a) is inadequate. Please provide the radionuclides concentrations as specified in the question.

Answer: Measurements taken on the activity in the pool water before and after refueling indicated essentially no change in concentrations. Therefore, increasing the number of assemblies in the pool is expected to have little effect on concentrations. Measurements show that the important radionuclides are Co-58 and Co-60 with concentrations of approximately 10^{-2} uc/cc. All other radionuclide concentrations are at least 2 orders of magnitude less.

QUESTION 10 - Section 4.4 states that the spent fuel pit area is a radiation zone 1 classification with a maximum dose rate of 0.5 mrem/hr. Section 5.6 states that "the measured dose rate to personnel from the radionuclides in the spent fuel pool water is only a few mrem/hr." Please clarify this anomaly. Also, provide the dose rates that have been measured in the areas specified in question 31(b), namely the center and edge of the pool.

Answer: Section 4.4 is amended to read as listed below. The measurements show that the spent fuel pool area is slightly above the zone II limits for some locations and so the overall zone would be III. Note the dose rates in mrem/hr given on the survey maps provided (See Figures 9 and 10)

4.4 PIT CLEANUP SYSTEM

The spent fuel pit area was assigned a radiation zone classification I (normal occupancy with a maximum dose rate of 0.5 mrem/hr) as noted in FSAR Figure 11.2-2 and Table 11.2-1. A reclassification to a zone III is necessitated by radiation surveys made of the spent fuel pools.

QUESTION 11 - Section 5.6 does not respond to Questions 31(e) and (g) in an acceptable manner. Please address your estimate of the increase in the annual man-rem burden from all operations in the spent fuel pool as a result of the pool modification. Include those exposures from more frequent changing of the demineralizer resin and filter cartridges.

Answer: To date, increasing the number of assemblies in the pool has not led to significantly increased concentrations of radionuclides in the spent fuel pool water. Measurements taken on the activity in the pool water before and after refueling indicated essentially no change in concentrations. As a result, it is not expected that there will be more frequent changing of demineralizer resin or filter cartridges. Therefore, little increase in annual-man rem is expected. Measurements of radiation level from the resin beds and filter cartridges are only a few mrem/hr, so even if changes were more frequent, the increase in annual man-rem would be small.

The increase in total man-rem due to storage of a larger number of assemblies in the spent fuel pool will be negligible since the radioactivity concentrations have been measured to be relatively constant and essentially independent of the number of assemblies in the pool. Since the fuel is shielded by so much water, the direct dose rate from the fuel is insignificant. Therefore, the grouping of fuel assemblies will have a negligible effect on the dose rate or annual man-rem.

QUESTION 12 - Section 5.6 states that "the radiation levels along the sides of the pool and over the center of the pool are essentially the same. Therefore, there has been no special build-up of crud around the sides of the pool ... ." The conclusions drawn with respect to the measurements appear to be incorrect. The dose rate at the edge of the pool, if there were clean pool walls (i.e., no crud build-up), should be less than the dose rate at the pool center based on source-detector geometrical considerations (i.e., a cylindrical source at the center and half-cylinder at the edge). Please provide mathematical models and calculations to justify your conclusions with respect to crud build-up around the sides of the pool.

Answer: Refer to the survey maps provided in the response to Question 10. Examination of the surveys reveal that the spent fuel pool areas are generally uniform in radiation level with slight variations in a few locations. Measurements show that there is not an appreciable decrease in radiation level around the edges of the pool as compared to the center. The spent fuel pool is such a large source that slight changes in location have a negligible effect on the radiation level.

QUESTION 13 - Please provide the maximum and average volume of water in the spent fuel pool as requested in Question 27 (a).

Answer: At the maximum water level, i.e., just below the high water level alarm, the volume of water in the pool will be 308,000 gallons.

The average water volume maintained in the pool is 305,400 gallons, i.e., four inches below the high water alarm level.

In neither case were the volumes occupied by the fuel elements or the racks considered.

QUESTION 14 - Using realistic initial conditions that will exist for the fuel pool and the fuel assemblies within the pool, please show that the calculated neutron multiplication in the pool will be less than 0.95 under all accident conditions (including the cask drop and tip in the fuel pool accident).

Answer: The calculated multiplication factor under nominal conditions for the HI-CAP spent fuel rack is 0.888. Using the most adverse combination of mechanical tolerances and displacing the fuel assemblies into their most reactive positions (closest approach of four neighboring fuel assemblies) yields a k_∞ of 0.914. A calculational uncertainty of 0.006 k at the 95/95 confidence limit has been deduced from comparison of calculation and experiment. In addition, a calculation allowance of 0.014 k for the assumed bias of the calculation model to predict the calculated worth of the steel box walls has been assessed. The reactivity balance for the system is provided below.

Design Limit	0.950
Calculational Uncertainty	- .006
Steel Calculational Allowance	- .014
	0.930

	Nominal	Most Adverse
Multiplication Factor for Spent Fuel Storage Rack	0.888	0.914
Excess Margin (Δk_{eff})	0.042	0.016

In addition, the water in the spent fuel pit conforms to the refueling water boron concentrations of 1950 ppm minimum.

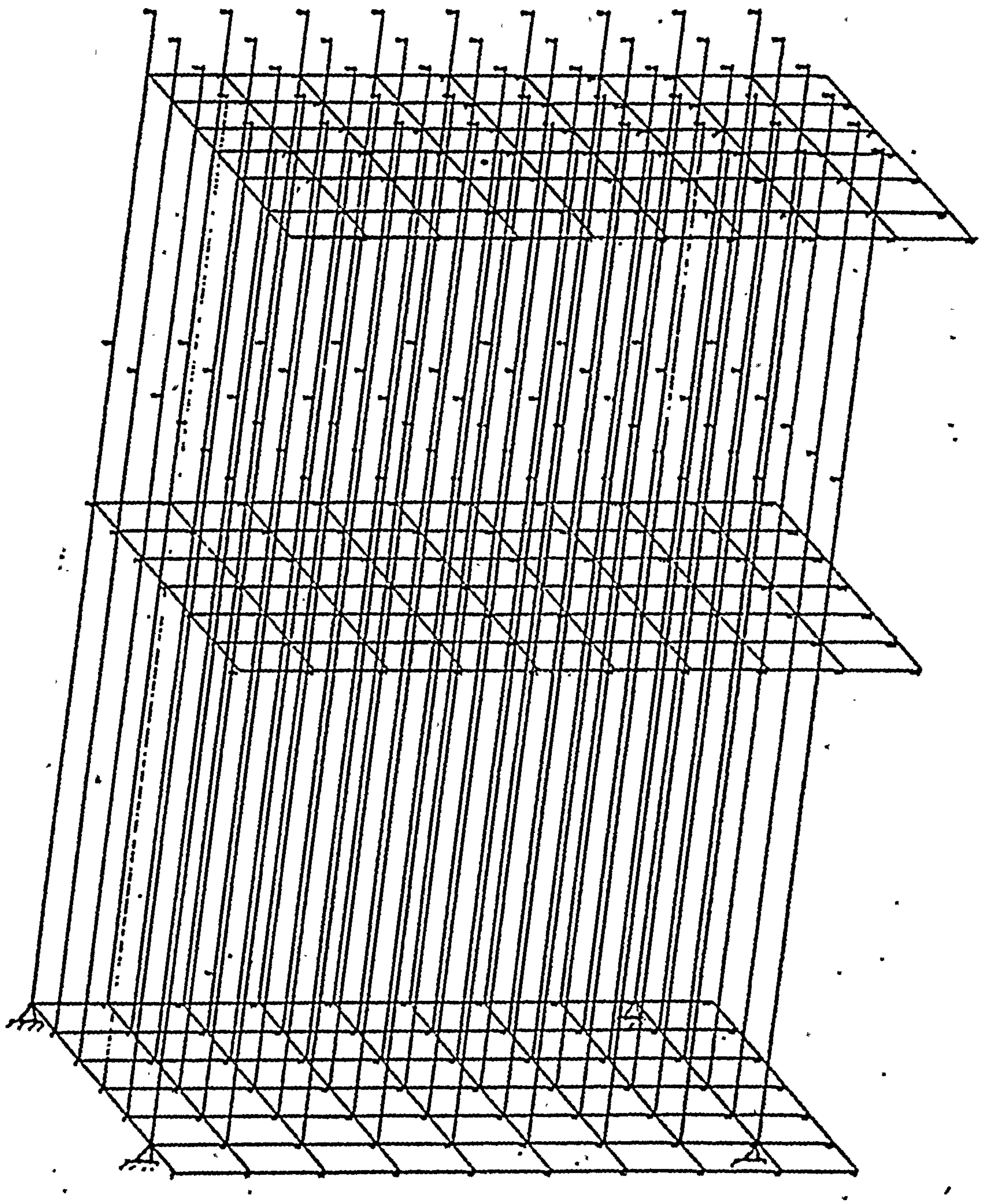


Figure 1

STORAGE RACK MODEL FOR
EIGENVALUE EXTRACTION (ST

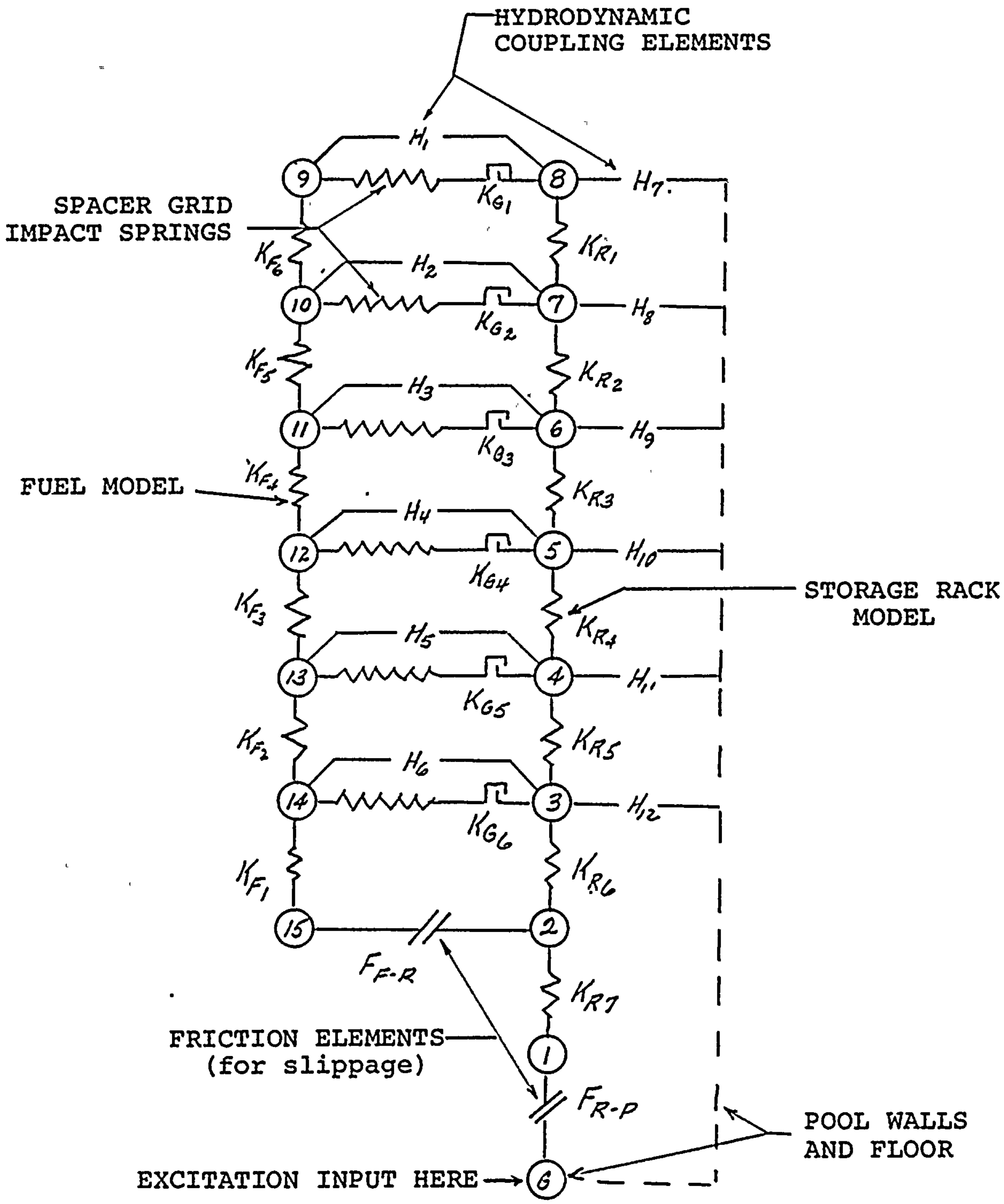


FIGURE 2

(MODEL FOR NONLINEAR [SHOCK] ANALYSIS)

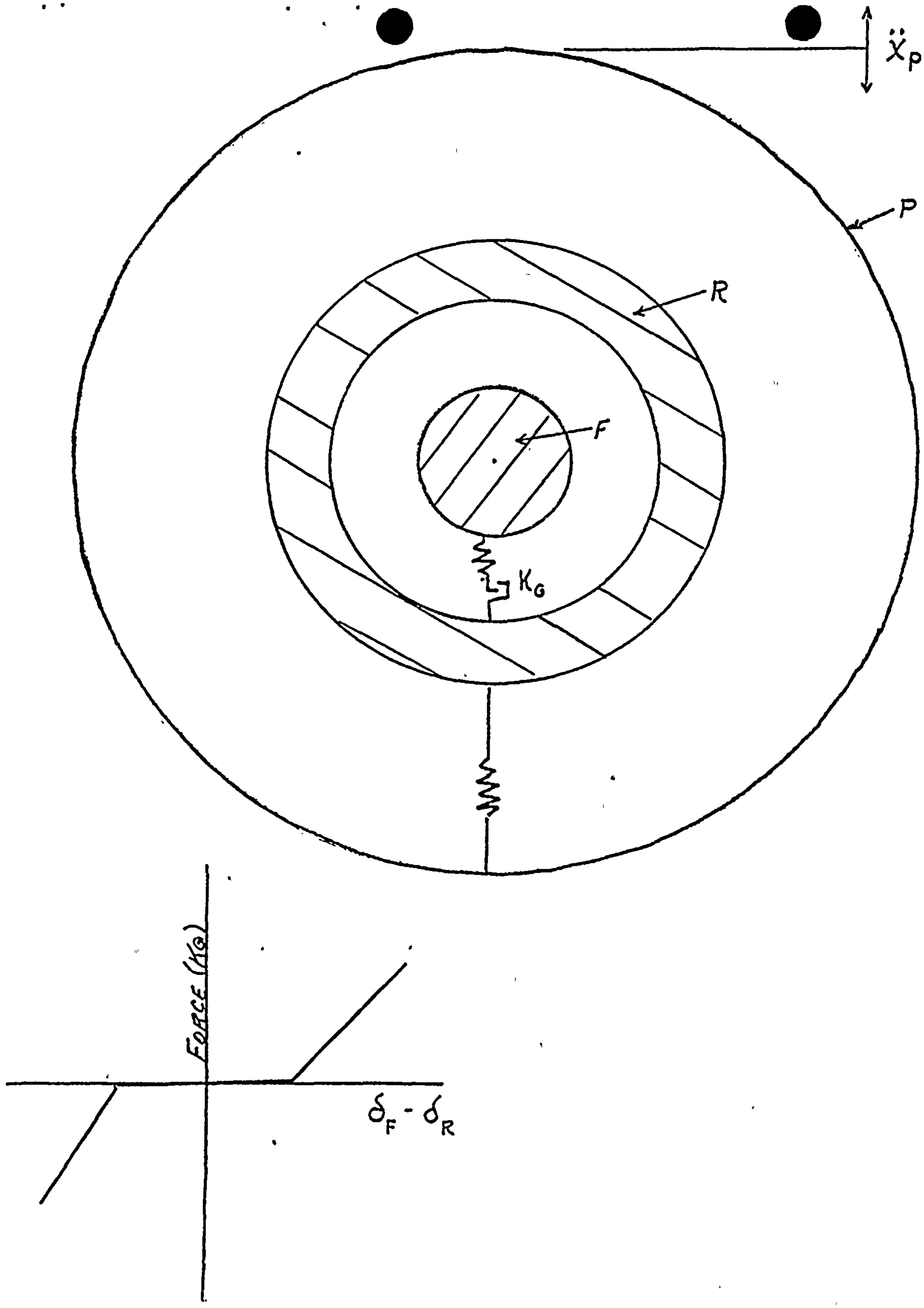


Figure 3
(SIMPLIFIED ANALOG OF PHYSICAL PROBLEM)

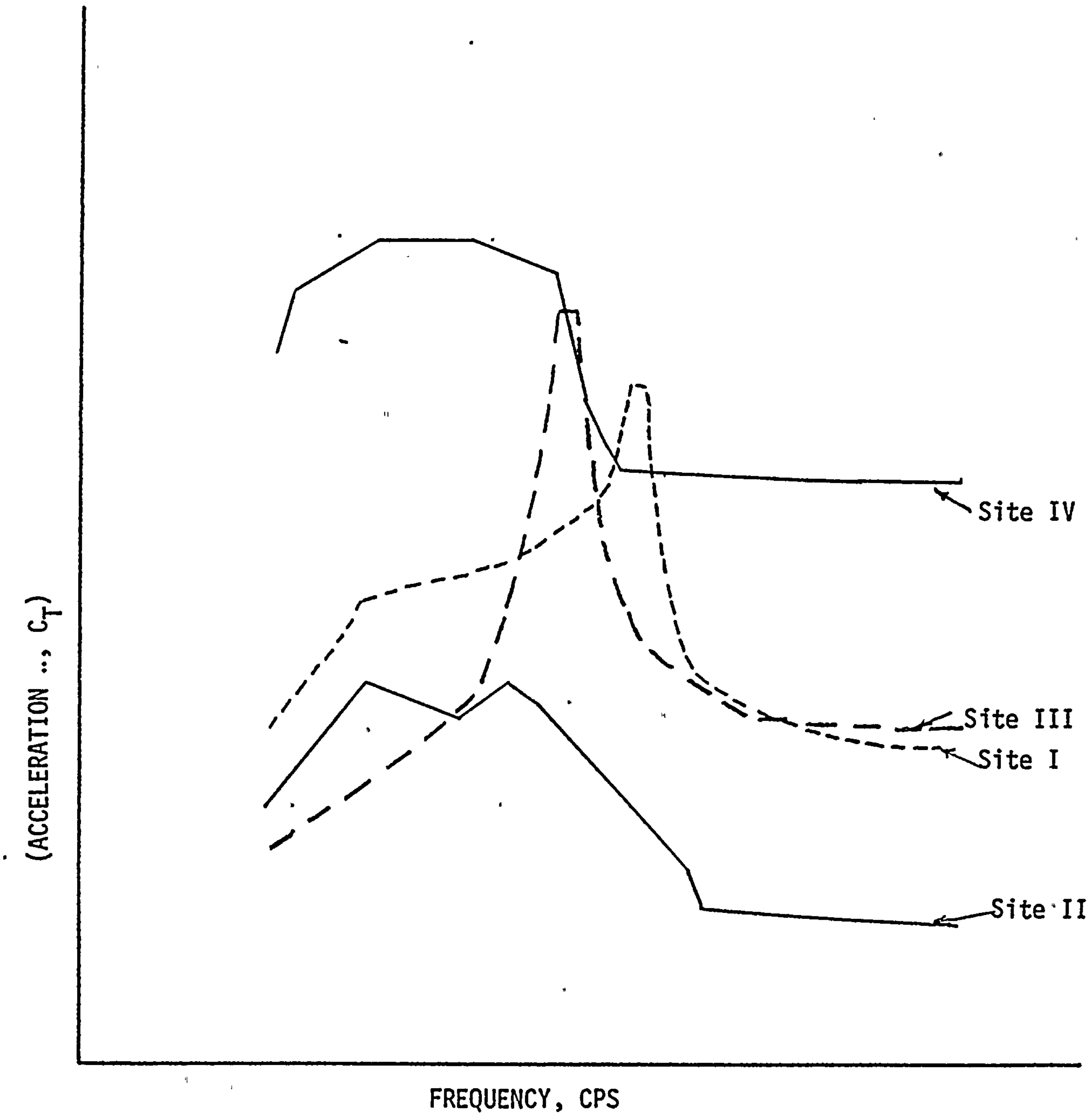


Figure 4

SPENT FUEL POOLS
SEISMIC RESPONSE SPECTRA
COMPARISON ONLY

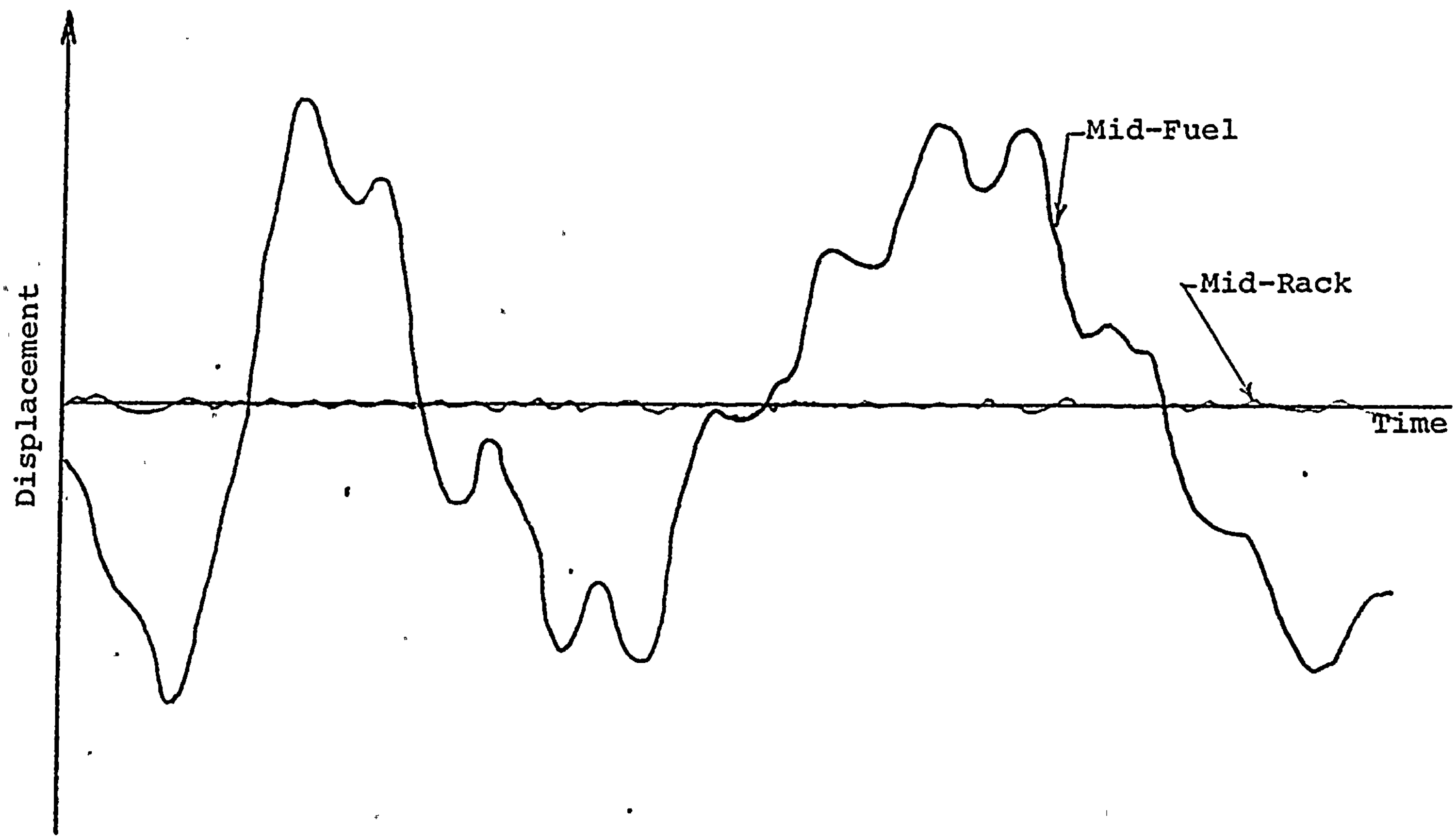


Figure 5
Typical Shock Results
Comparison Only

IDENTIFIER SITE/DESIGN	NORMALIZED REACTION LOAD PER CELL		RATIO $\frac{1}{2}$
	1 TIME-HISTORY NONLINEAR ANALYSIS	2 RESPONSE SPECTRUM METHOD	
I (DESIGN A	5.92	7.42	0.79
I (DESIGN B	8.17	15.66	0.52
I (DESIGN C	4.21	4.08	1.03
II (DESIGN A	1.99	1.79	1.11
II (DESIGN D	3.00	1.00	3.00
III (DESIGN D	2.73	2.56	1.07
IV (DESIGN D	17.08	8.74	1.95
II(DESIGN D (ORIG.	3.00	1.00	3.00
II(DESIGN D (Δ 10 %	2.83	1.00	2.83
II(DESIGN D (1/4 GAP	2.08	1.00	2.08
II(DESIGN D ($10 \times K_G$	4.27	1.00	4.27

FIGURE 6

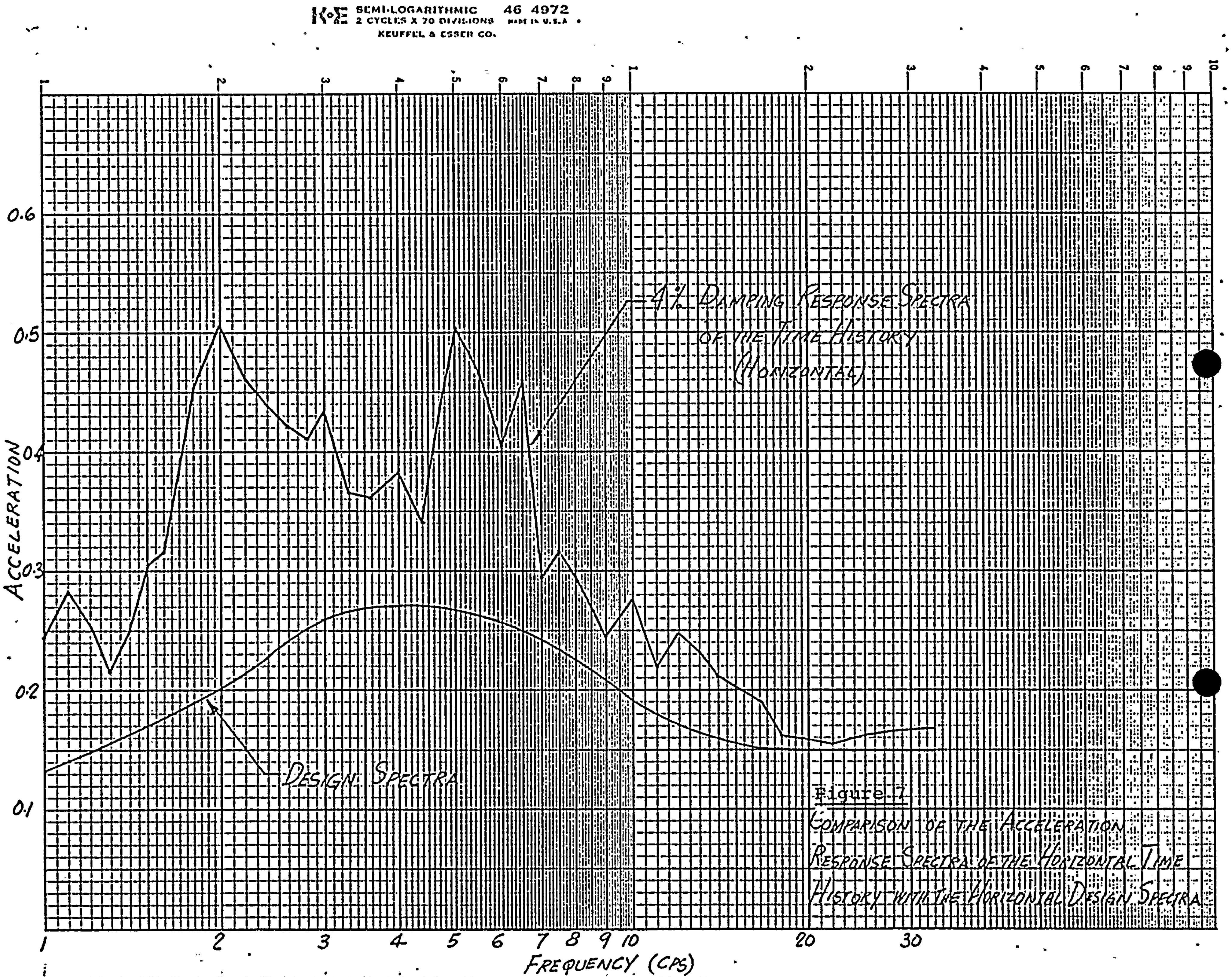


Figure 7
COMPARISON OF THE ACCELERATION RESPONSE SPECTRA OF THE HORIZONTAL TIME HISTORY WITH THE HORIZONTAL DESIGN SPECRA

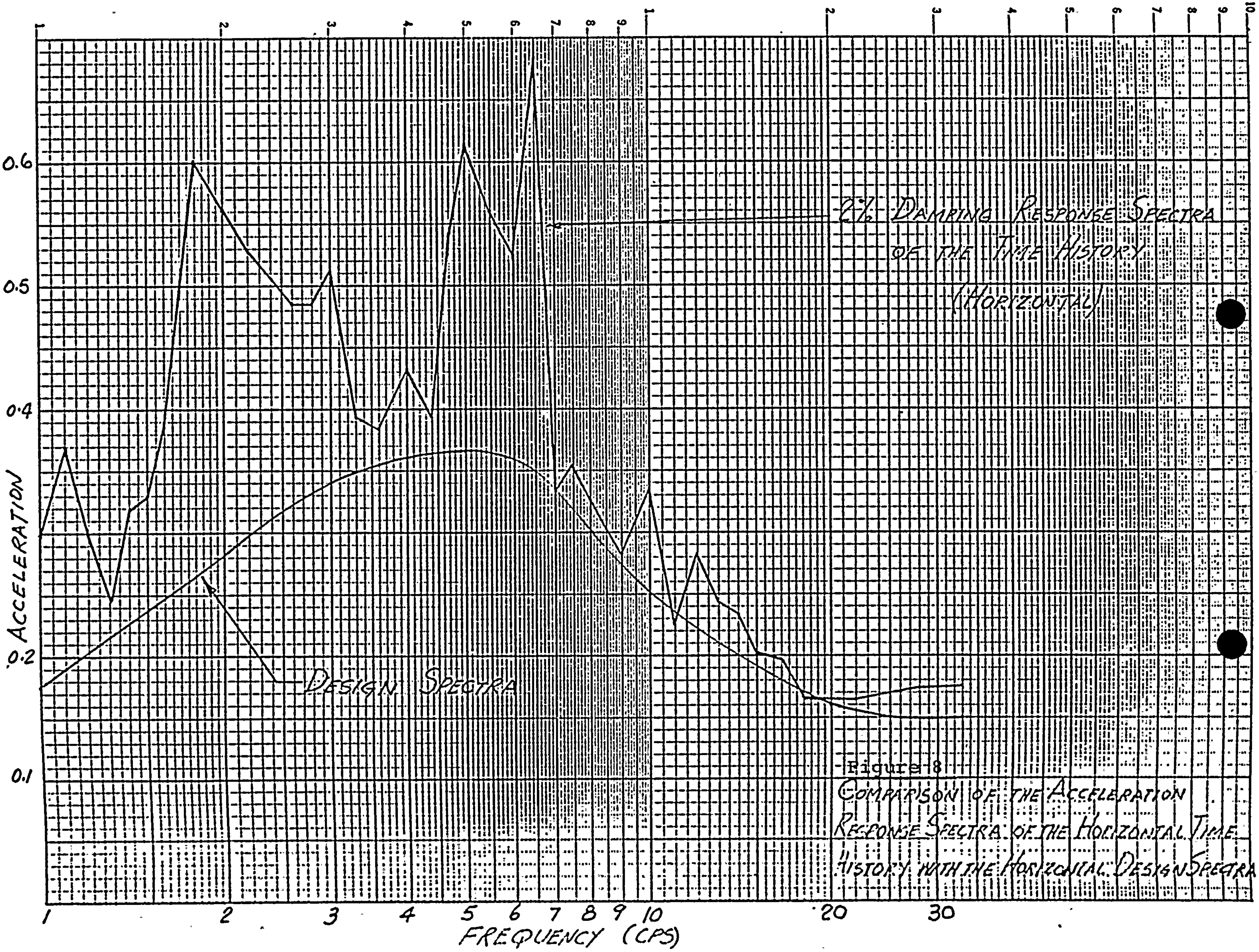


Figure 8
COMPARISON OF THE ACCELERATION RESPONSE SPECTRA OF THE HORIZONTAL TIME HISTORY WITH THE HORIZONTAL DESIGN SPECTRA

ROUTINE	RWP#	[X] SPECIAL NOTE REMARKS

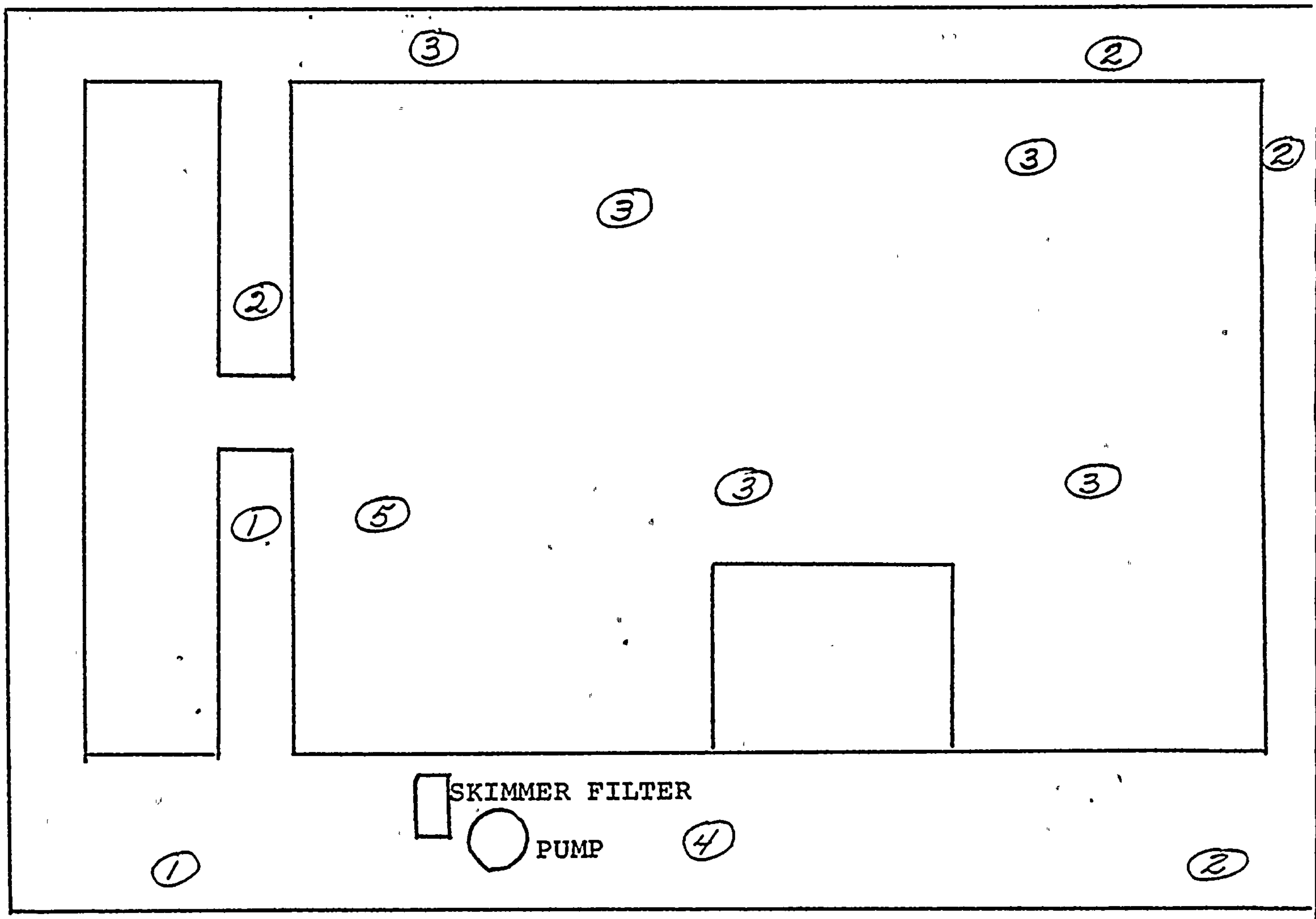
TIME: 1845 4/20/76

INSTRUMENT(S) USED & HP# Detector

REACTOR POWER: -N/A- FULL POWER

SPENT FUEL PIT 3

N



All readings taken at waist level

Figure 9

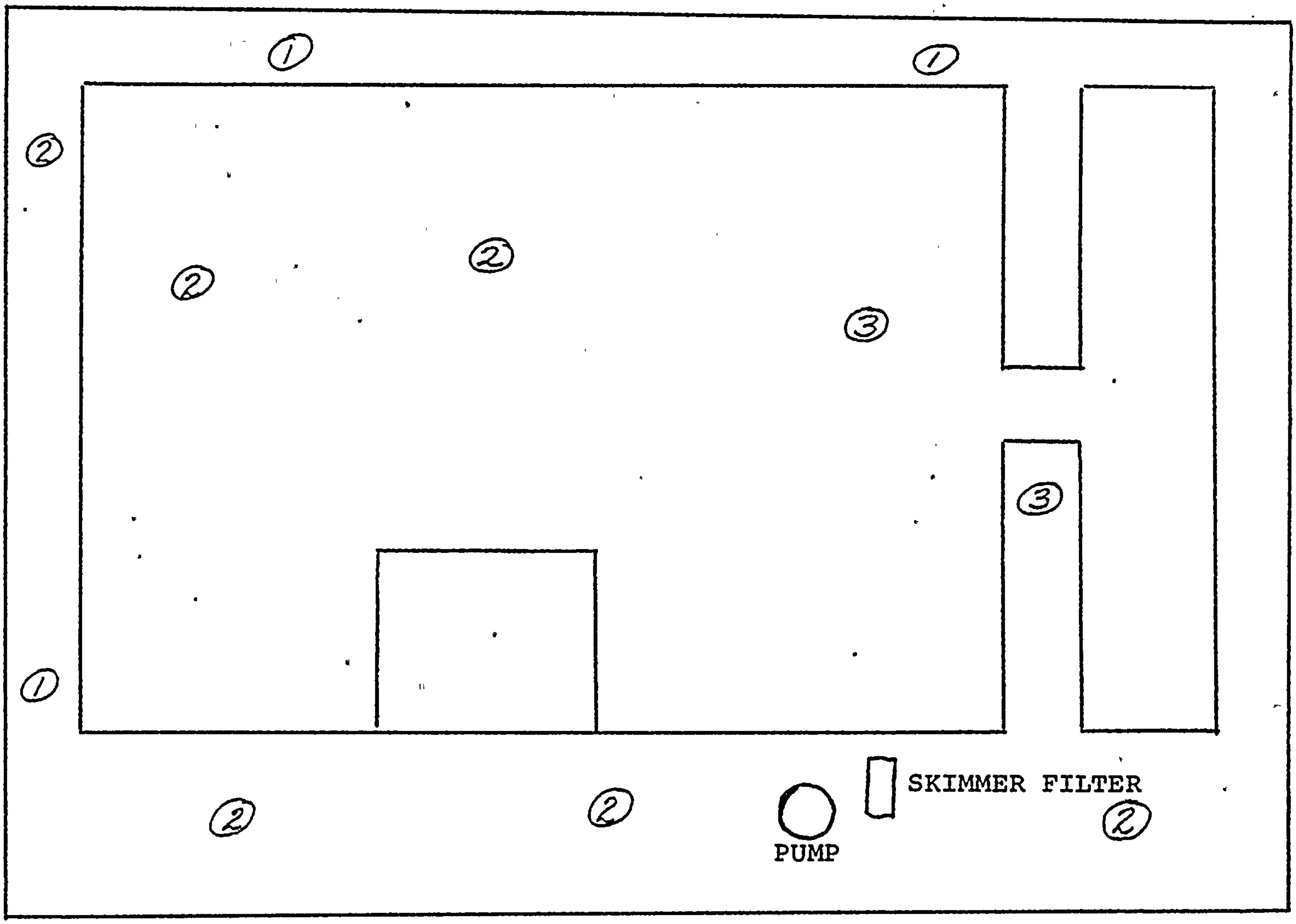
Circled numbers indicate radiation dose rate in mrem/hr.
X indicates transferable contamination in dpm/100cm^2.

SURVEY BY: Bell

☐ ROUTINE ☐ RWP # . . . ☒ SPECIAL NOTE REMARKS | TIME: 1855 4/20/76

INSTRUMENT(S) USED & HP#: T[illegible]tector | REACTOR POWER: NP SHUTDOWN REFUELING NOT STARTED

SPENT FUEL PIT 4



REMARKS:

All readings taken at waist level.

Figure 10

Circled numbers indicate radiation dose rate in mrem/hr.
X indicates transferable contamination in dpm/100 cm^2.

Bell

SURVEY BY