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FLOW-INDUCED VIBRATION FOR LIGHT-WATER REACTORS

PROGRESS REPORT
(JANUARY 1981 — MARCH 1981)

SEPTEMBER 1981

M. R. TORRES

GENERAL ELECTRIC COMPANY
NUCLEAR ENGINEERING DIVISION
175 CURTNER AVENUE
SAN JOSE, CALIFORNIA 95125

PREPARED FOR THE
U.S. DEPARTMENT OF ENERGY
ASSISTANT SECRETARY FOR NUCLEAR ENERGY
OFFICE OF LIGHT WATER REACTORS
UNDER CONTRACT DE-AC02-77ET34209

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ABSTRACT

Flow-Induced Vibration for Light Water Reactors (FIV for LWRs) is a four-year program designed to improve the FIV performance of light water reactors through the development of design criteria, analytical models for predicting behavior of components, and general scaling laws to improve the accuracy of reduced-scale tests, and through the identification of high FIV risk areas. The program is managed by the General Electric Nuclear Power Systems Engineering Department and has three major contributors: General Electric Nuclear Power Systems Engineering Department (NPSED), General Electric Corporate Research and Development (CR&D) and Argonne National Laboratory (ANL). The program commenced December 1, 1976. This progress report summarizes the accomplishments achieved during the period from January 1981 to March 1981.

INTRODUCTION

The Flow-Induced Vibration for Light Water Reactors Program (FIV for LWRs) is being conducted for the Division of Nuclear Power Development, U.S. Department of Energy (DOE). The goal of the program is to increase nuclear plant availability and to reduce radiation exposure by substantially reducing the downtime caused by FIV failure of components. It is a four-year balanced effort of fundamental studies, analyses, tests of idealized conditions, and realistic tests of reactor components, all leading to the preparation of design guides and criteria for LWRs. The program is broken into four major tasks:

- Task 1 - Fundamental Studies - General Electric Corporate Research and Development (CR&D) and Argonne National Laboratory (ANL)
- Task 2 - Model and Full-Size Tests - General Electric Nuclear Technology Department (NTD)
- Task 3 - Design Methods, Guides and Criteria (NTD)
- Task 4 - Program Administration (NTD)

A brief summary of the objectives of these tasks and a description of the technical progress achieved over the period from January 1, 1981, through March 31, 1980, are provided in the following sections. Additional background information can be obtained from other "Flow-Induced Vibration for Light Water Reactors Progress Reports" listed as follows:

1. December 1976 - May 1977, GEAP-24055, COO/4175-1.
2. June - September 1977, GEAP-24124, COO/4174-2.
3. October - December 1977, GEAP-24151, COO/4175-5.
4. January - March 1978, GEAP-24163, COO/4175-9.
5. April 1978 - December 1979, GEAP-24248, DOE/N/4175-14.

6. January - June 1980, GEAP-24926, DOE/ET/34209/16.

7. July - September 1980, GEAP-24368, DOE/ET/34209/17.

8. October - December 1980, GEAP-24369, DOE/ET/34209/18.

TASK 1 - FUNDAMENTAL STUDIES

The purpose of Task 1 is to develop a better fundamental understanding of FIV phenomena as related to light water reactors (LWRs). A combination of tests and analyses considers geometries and FIV mechanisms common to the LWR. Particular emphasis is given to characterizing the FIV forcing functions for use in establishing predictive models. The task is divided into four major areas:

Task 1.1 - External Flow over Tube Banks - Cross Flow (CR&D)

Task 1.2 - External Flow over Tube Banks - Parallel Flow (ANL)

Task 1.3 - Leakage Flow Mechanisms (CR&D)

Task 1.4 - Random Vibration Effects (ANL)

Progress has been achieved on Tasks 1.1 and 1.2 over the report period and is described on the following pages. Task 1.4 has been completed and the final results are documented in DOE report COO/4175-10, "Random High-Cycle Fatigue Analysis."

TASK 1.1 - EXTERNAL FLOW OVER TUBE BANKS - CROSS FLOW (CR&D)

Task 1.1, performed by General Electric Corporate Research and Development, is designed to develop a better understanding of the response of tube arrays to fluid cross flow. It consists of six activities:

Task 1.1a - Preliminary Array Definition Experiments

Task 1.1b - Water-Based Tests at High Reynolds Numbers

Task 1.1c - Buffeting of Cylinder Arrays in Turbulent Cross Flow

Task 1.1d - Fluidelastic Instability Boundaries

Task 1.1e - Proof Test

Task 1.1f - Design Guides

Progress achieved over the report period on these activities is described as follows:

Task 1.1a - Preliminary Buffeting Experiments (S. D. Savkar, R.M.C. So)

Task Description

The objective of this task was to develop an experimental scheme capable of directly measuring the unsteady forces induced on cylinders immersed in a turbulent cross flow. The measurement scheme will be used in several of the Task 1.1 test programs.

Work Completed This Report Period

Task 1.1a, as previously reported, was successfully completed in an earlier report period. Work on this activity has been terminated.

Task 1.1b - Water-Based Tests at High Reynolds Number (S. Savkar/T. Litzinger/ R. So)

Task Description

Task 1.1b consists of a series of experiments performed in water, using the Garfield Thomas Water Tunnel at Pennsylvania State University, to study the buffeting of a cylinder in turbulent cross flow. The primary objective is to measure and correlate the unsteady forces induced by the cross flow on the cylinder as a function of Reynolds number and turbulence. The program is designed to complement the cross flow in air activities studied in Task 1.1c.

Work Completed This Report Period

All the data obtained in this activity have been reduced. Work on a topical report summarizing the results should be issued in the next reporting period.

Task 1.1c - Buffeting of Cylinder Arrays in a Turbulent Cross Flow (S. Savkar, T. Litzinger)

Task Description

Task 1.1c consists of a series of experiments performed in air to study the buffeting of single cylinders and cylinder arrays in a turbulent cross flow. The experiments are directed toward characterizing the forces exerted on the cylinder as a function of such parameters as Reynolds number, upstream turbulence, Strouhal number, and cylinder array geometry. The program utilizes the force measurement scheme developed in Task 1.1a to characterize the cylinder forces.

Work Completed This Report Period

The following work items were carried out during this reporting period:

(1) Unsteady force data for an isolated cylinder were measured preliminary to the tube array work for the purpose of checking out the apparatus. Runs were made with both a uniform flow and a flow turbulated by

3-in. mesh and 6-in. mesh grids. Initial runs were made with a 1D active span. Since the signal-to-noise ratio was not satisfactory at low flow rates, a new test cylinder with 3D active span was fabricated. The lift coefficients and Strouhal numbers measured on the 3D span cylinder are shown in Figures 1.1-1 and 1.1-2, respectively. The measured lift coefficients were somewhat smaller than those measured in the water tunnel tests (Task 1.1.b). These differences are believed to be due to differing blockage ratios (d/h, diameter to the effective tunnel width) in the two facilities. The Penn State tunnel had an effective blockage ratio of 16 percent, while the tube array air tunnel has a blockage ratio of only 8 percent for an isolated cylinder. This argument is illustrated in Tables 1.1-1 and 1.1-2 where the Strouhal (uniform flow) and the turbulent buffeting data from both facilities are corrected to the free flow conditions, $d/h \to 0$, using the correction proposed by Richter[1] for Strouhal forces. The RMS lift coefficients corrected to the free flow condition are designated as $C'_{L\infty}$. On this basis, the two sets of data correspond very closely.

(2) Repeated attempts at devising a method of measuring the steady drag using the piezo-electric load cells were made, but the repeatability of the measurements was very poor ($\pm$ 30 percent above established values). It appears there is too much drift in the load cell output for the measurement to be useful when the load cell is operated on a long time constant. Consequently, all attempts at measuring steady drag by this means have been dropped.

(3) Unsteady lift coefficients for a row of tubes were measured. All three pitch-to-diameter ratios (1.2, 1.5, 1.71) have been tested. Only the data from the pitch-to-diameter ratio of 1.71 appear reasonable. These data are shown in Figure 1.1-3 along with isolated cylinder data for comparison. The 1.71 pitch row data appear to transition earlier than the isolated cylinder in a manner very similar to the Richter and Naudascher observation[1]. However, the amplitudes of the measured subcritical forces were about the same, unlike the data reported in Reference 1. The flow through the other two pitches appeared to flip flop (in the

repeated to insure the repeatability of results. This suggests that the tunnel wall blockage results obtained by Richter and Naudascher[1] should not simply carry over to rows of tubes. The flip flop character of the flow through the single row appears to stabilize out when the number of rows is increased to three. The latter data should be available for reporting in the next reporting period.

Task 1.1d - Fluidelastic Instability Boundaries (T. F. Balsa)

Task Description

A combined test and analysis program is directed toward establishing the nature and boundary of fluidelastic instability of arrays as a function of array geometry. A near-term objective is the numerical prediction of the forces on an oscillating single cylinder at high Reynolds numbers (10^4 to 10^7). These predictions should be useful in establishing whether the forces can produce a hydroelastic instability, or whether they simply excite cylinder motion through their fluctuating or unsteady component.

Work Completed This Report Period

Preliminary theoretical models were formulated and critical experiments were outlined in previous reporting periods. Activity on this task was terminated after the first contract suspension. No further work is planned.

References

1. A. Richter, Stromungkrafte auf starre Kreiszylinder Zwischen Parallelen Wanden, Ph.D Dissertation, University of Karlsruhe, 1973. (See also Richter and Naudascher, 1976, J. Fluid Mechanics, 78, p. 78.)

Table 1.1-1

EFFECT OF BLOCKAGE ON STROUHAL LIFT COEFFICIENTS BASED ON CORRECTIONS DUE TO RICHTER[1]

Air Tunnel			Water Tunnel		
$Re \times 10^{-4}$	C'_L	$(C'_L)_\infty$	$Re \times 10^{-4}$	C'_L	$(C'_L)_\infty$
5.23	0.69	0.48	5.11	1.05	0.54
9.92	0.61	0.43	10.3	0.95	0.50
14.8	0.52	0.38	15.4	0.78	0.44
20.9	0.43	0.33	20.5	0.39	0.26
24.4	0.38	0.30	25.2	0.23	0.18

C'_L - Data from the test tunnel

$(C'_L)_\infty$ - Data corrected to zero blockage

Table 1.1-2

EFFECT OF BLOCKAGE ON BUFFETING LIFT COEFFICIENTS BASED ON CORRECTIONS DUE TO RICHTER[1]

Air Tunnel			Water Tunnel		
$Re \times 10^{-4}$	C'_L	$(C'_L)_\infty$	$Re \times 10^{-4}$	C'_L	$(C'_L)_\infty$
3.57	0.36	0.28	3.46	0.74	0.42
7.27	0.14	0.13	6.92	0.24	0.18
10.6	0.13	0.12	10.4	0.13	0.11
14.1	0.13	0.12	13.9	0.11	0.094
17.0	0.11	0.10	17.4	0.11	0.094
21.7	0.10	0.092	20.8	0.12	0.10

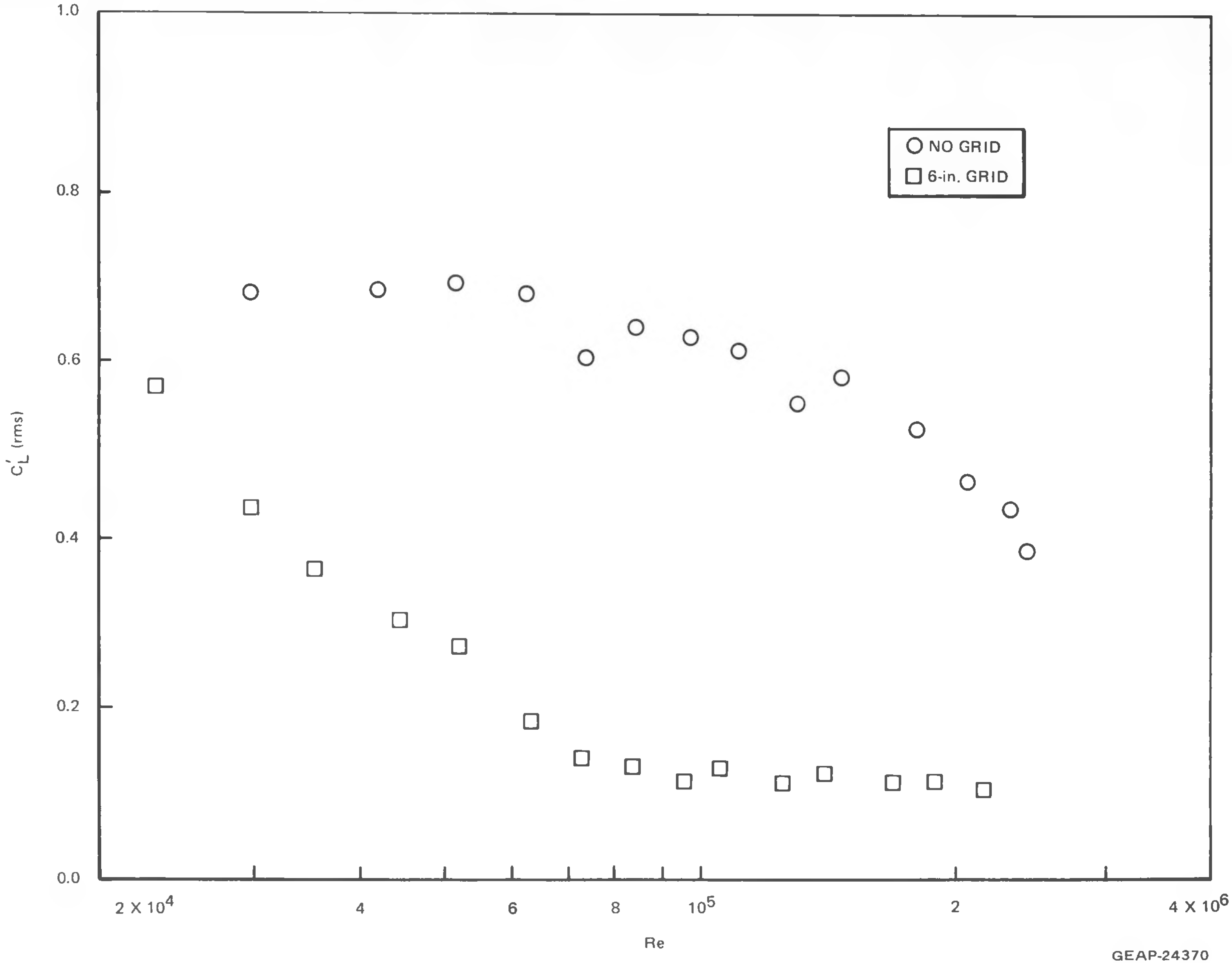

Figure 1.1-1. Fluctuating (RMS) Lift Coefficients Measured on an Isolated Cylinder in the Tube Array Apparatus

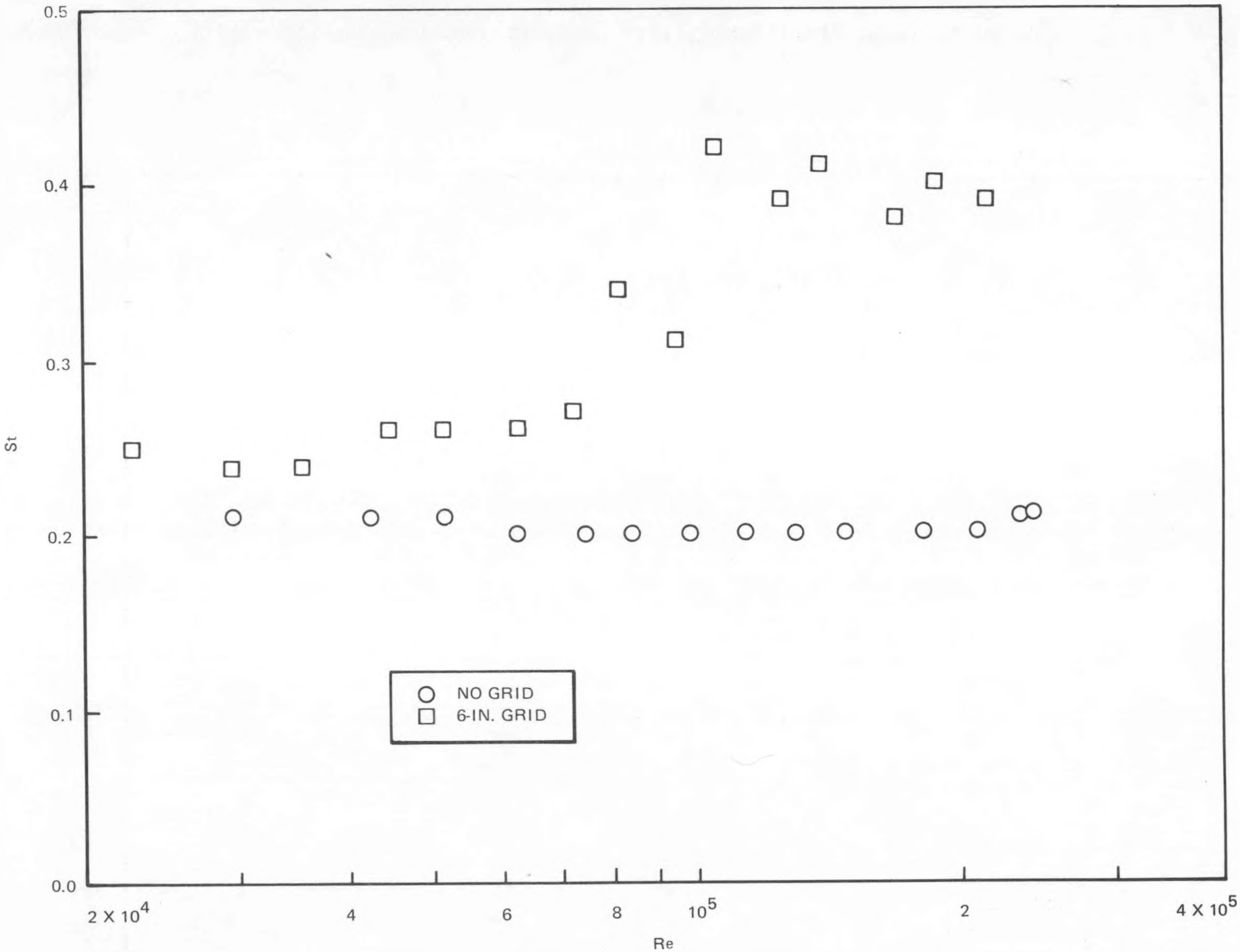

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Figure 1.1-2. Strouhal Numbers for an Isolated Cylinder in the Tube Array Apparatus

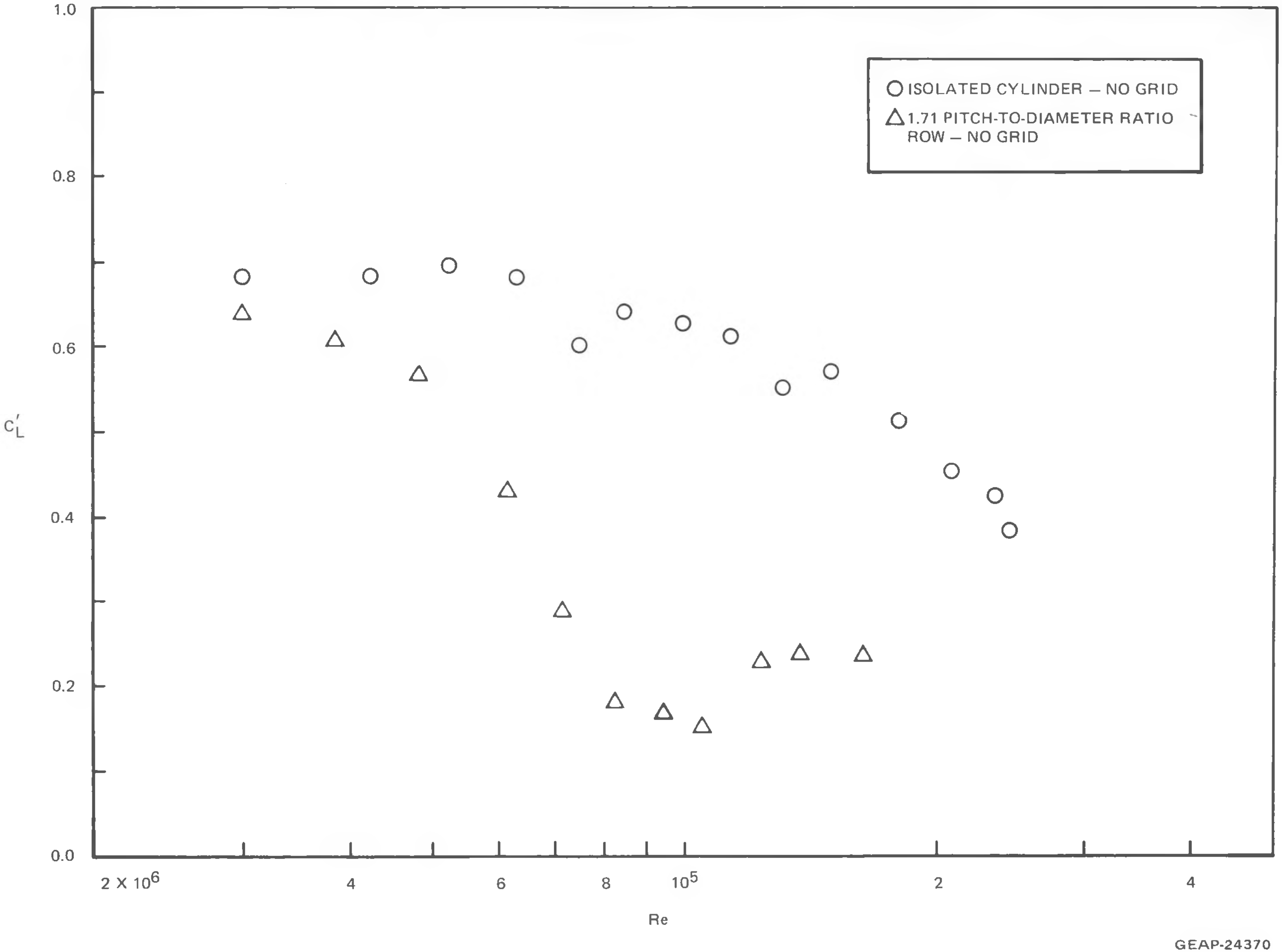


Figure 1.1-3. Comparison between Single Row of Tubes and an Isolated Cylinder

TASK 1.2 - EXTERNAL FLOW OVER TUBE BANKS - PARALLEL FLOW (ANL)

Task 1.2, performed by Argonne National Laboratory, is designed to develop a better understanding of the response of tube arrays subjected to parallel flow. The specific objectives include the following:

(1) Determine, by a systematic series of experiments, the dependence of amplitude of motion and forces on Reynolds number, upstream flow conditions, array geometry, and weak cross flow for tube arrays in essentially parallel flows;

(2) Establish weak cross-flow-induced fluidelastic stability boundaries; and

(3) Evolve design methodology to estimate the fluid forces and resulting vibration response.

The program is divided into five subtasks:

Task 1.2a - Review of Available Data from Flow-Induced Vibration Tests and Assessment of the State of the Art in Understanding/ Modeling Parallel Flow-Induced Vibration Phenomena and/or Response

Task 1.2b - Experimental Study of Single Cylinders in Nominally Parallel Flows Typically Encountered in Channels and Reactor Plenums

Task 1.2c - Experimental Study of Fluidelastic Interaction that Can Result in Coupled Response and/or Instabilities in Rod/Shell Systems, Tube Bundles, Tubes in Asymmetric Flow Channels, and Systems with Nonuniform Cross Sections

Task 1.2d - Experimental Study of Tube Arrays in Parallel Flow

Task 1.2e - Development of Mathematical Models and Participation in Design Guides Evaluation (Task 3)

The primary test fluid will be water. Incoming flow conditions, together with a weak cross flow, will be use to establish the effects of the parameters of interest. The flow will be varied over a sufficient range to clearly establish the role of Reynolds number. Besides acquiring data of significance to the parallel flow FIV of tube arrays, analytical tools that can be evolved into design guides will be devised.

A description of the progress achieved on these subtasks over the periods from January 1981 to March 1981 follows.

Task 1.2a - State-of-the-Art Assessment

Task Description

Task 1.2a consists of a day-to-day review of available parallel FIV literature applicable to light water reactors in order to maintain a current knowledge of the state of the art.

Work Completed This Report Period

Other than a continued general review of the FIV literature, no work was performed on this activity during the report period.

Task 1.2b - Single Cylinders in Nominally Parallel Flow (T. M. Mulcahy, W. H. Lin, W. P. Lawrence, B. G. Jones)

Task 1.2b consists of two complementary parallel FIV programs on isolated cylinders. Experiments are performed in water using the ANL Flow Induced Vibration Test Facility, and in air using the University of Illinois at Urbana open cycle air loop at the Thermal Hydraulics Laboratory. Work performed on these programs is described as follows.

Activity 1 - Annular Region (Water)

Task Description

Parallel FIV experiments are performed on isolated cylinders in water to determine the cylinder response, wall pressure statistics, cylinder resonances, and fluid damping as a function of the independent parameters. The ultimate goal is to develop an analytical response prediction model characterized by the test data. The independent test parameters include a wide range of Reynolds numbers, hydraulic diameters, cylinder structural boundary conditions, and, most important, upstream flow disturbances. The creation and characterization of upstream flow disturbances as a test parameter will be the unique contribution of this work.

The specific objectives are to:

(1) Provide test data to develop an integrated analytical-experimental model for predicting response of a rod located symmetrically in a cylindrical region subjected to fully developed flow with and without upstream flow disturbances; and

(2) Establish sensitivity (scaling) of wall pressures with respect to rod vibration, Reynolds number, hydraulic diameters, and upstream disturbances (turbulence).

Work Completed This Report Period

The experimental results from measurements of wall-pressure fluctuations on a circular rod concentrically located in three different circular channel flows with various upstream disturbances caused by flow spoilers are being reported in two parts. Part 1, completed and published[1], presents the results from tests with no flow spoilers. Part 2 was completed during the reporting period and submitted for publication. It includes results from tests with flow spoilers.

Activity 2 - Annular region (Air)

Task Description

Two fundamental parallel FIV studies are being conducted: in one the wall and fluid field fluctuating pressures will be measured simultaneously, and in the other the wall fluctuating pressures and the turbulent velocity field will be measured simultaneously. The first will allow determination of the stochastic structure of the pressure field and a look for correlation relationships between the fluid stochastic pressure field and that at the wall. The second study will allow determination of the turbulent velocity field and examination of its relationship to the stochastic wall pressures. These studies will be conducted in air flows in which a simply supported central rod is mounted in a circular tube to provide a concentric annular flow field.

The research program is centered around four interrelated experimental objectives to be carried out in a concentric single pin annular geometry turbulent air flow as follows:

(1) Measure turbulent velocity intensity levels and mean velocity profiles in the flow region;

(2) Measure pressure correlation in the flow region and compare with existing analytical data.

(3) Measure the correlation between the fluctuating flow field pressure and wall pressure fluctuations on the rod; and

(4) Measure the correlation between the fluctuating flow field velocity and wall pressure fluctuations on the rod.

The end result of this fundamental study should be better quantitative and qualitative understanding of the flow phenomena which generate the wall pressure field.

Work Completed This Report Period

This activity has been completed and reported.

Task 1.2c - Fluidelastic Interaction in Parallel Flow (W. H. Lin and J. A. Jendrzyczyk)

Task Description

To develop a better understanding of the fluidelastic interaction that can result in coupled response and/or instabilities in parallel flow systems, a single rod asymmetrically positioned in a circular flow channel is being investigated. Wall pressure fluctuations and rod response will be measured for a range of Reynolds numbers, hydraulic diameters, and reduced velocities, in order to provide the basis for developing a mathematical model to predict response and scaling relations.

Work Completed This Report Period

Experiments were performed in both stationary and flowing water to evaluate the effects of eccentricity and flow speed on the response of a tube in a circular flow channel. The experiments were conducted for four geometrical locations of the test element in the flow channel (Figure 1.2-1). Absolute values of the eccentricities are given in Figure 1.2-1 together with the ratios of eccentricity to hydraulic diameter. The experimental measurements include natural frequencies, damping factors, and rms displacements.

Tube motion in orthogonal directions is measured by two accelerometers mounted on a pellet that is slid into the tube and fixed at the midspan. Impact excitation was used to determine natural frequencies and damping values of the tube in air and in stationary water, with the logarithmic-decrement method used to estimate damping. In flowing water, the natural frequencies are determined from the center frequencies of response peaks shown on the PSD curves. Damping factors are also estimated from the PSD curves by use of the bandwidth method. The use of the bandwidth method to estimate the damping in flowing water is based on the assumption that the turbulent-boundary layer pressure fluctuation is a random white noise excitation so that measured PSD for the tube response is proportional to the transfer function for the tube. Measurements have shown that this assumption is reasonable.

During testing, the mean axial-flow velocity is incrementally increased. All test data are recorded on magnetic tape and processed with the aid of a Fast Fourier Transform Analyzer. The data processing includes computation of power spectra and mean-square values of displacement and acceleration, and spatial response.

The fundamental frequency of the tube in air is approximately 35.3 Hz; the corresponding frequency in stationary water is about 31.2 Hz. The average damping factor, as a percentage of critical damping, is 2.9% in air, and about 3.5% in stationary water. The above numerical values are averages from many tests. Because beating phenomena occur in some tests, determination of natural frequencies and damping factors is difficult.

The coupled natural frequencies and damping factors for the tube in flowing water (Table 1.2-1), are weakly decreasing and weakly increasing, respectively, with mean axial-flow speed when the tube is concentrically located in the circular flow channel. When the tube occupies one of its off-center positions, both the coupled natural frequencies and damping factors increase with increasing mean axial-flow speed and with increasing eccentricity. For example, see Figures 1.2-2 and 1.2-3.

Power spectral density (PSD) representations of tube displacements, obtained from the representations of tube accelerations, are shown in Figure 1.2-4 for all four cases of eccentricity. Figure 1.2-4 is intended to display the frequency content of the displacement response. As such, the scale for intensity is arbitrary. In Figure 1.2-4, the behavior of the natural frequency and damping factor can be readily observed, as presented in Table 1.2-1, Figures 1.2-2 and 1.2-3, and the previous discussion. For example, with the tube located eccentrically, the fundamental frequencies at low flow velocities (i.e., $U \leq 8$ m/s) shift to values lower than the corresponding values observed in the concentric case, while at high flow velocities ($U \geq 10$ m/s) the values shift higher than those observed in the concentric case. The magnitude of the frequency shift increases with respect to an increase of eccentricity for all ranges of mean flow velocities, but the magnitude of the shift with mean flow velocity is different for different ranges of flow velocities. For mean flow velocity above 8 m/s, the magnitude of the shift increases with increasing mean flow velocity. On the other hand, the magnitude of the shift decreases with an increase of mean flow velocity when the mean flow velocity is below 8 m/s. These effects increase with increasing eccentricity, or with proximity to the wall. In general, the smaller the gap is between the tube and the channel wall, the more pronounced the fluidelastic (or hydroelastic) interaction.

Task 1.2d - Experimental Study of Tube Arrays in Parallel Flow (W. H. Lin, T. M. Mulcahy, J. A. Jendrzejczykz)

Task Description

Task 1.2d consists of an experimental study of the wall pressures and response of rods in a seven-tube array. As in Task 1.2b, the ultimate goal is to generate data for use in developing a vibration response prediction model. The independent test parameters include a wide range of Reynolds numbers and upstream flow disturbances.

Work Completed This Report Period

Efforts during the review period focused on the analysis and interpretation of wall pressure data from tests with an instrumented rod located in both the center and a peripheral location in a seven-rod array. In particular, scaling of the wall-pressure PSDs was investigated using the approach followed in the analysis of the annular flow test data, rms pressure coefficients were computed, and rms displacement response of a rod in a peripheral location was evaluated and compared with annular flow results.

Figure 1.2-5 shows a dimensionless wall-pressure PSD plotted as a function of the wavenumber $k \equiv 2\pi f/U$; where f is frequency and U is mean axial flow speed. The parameter used for nondimensionalizing the spectral intensities is $q^2 d_h/U$; where q is dynamic head and d_h is hydraulic diameter. All spectral curves do not coalesce into a single curve, but rather into a band, because the nondimensionalizing parameter does not normalize the power spectra uniformly. The same phenomenon has been observed by other investigators on the measurements of wall-pressure fluctuations in annular flows, pipe flow, and boundary-layer flows.

As shown in Figure 1.2-5, the measured power spectra of wall-pressure fluctuations on the center rod at $x/\ell = 0.125$ normalize quite well for the entire wavenumber domain. Neglecting the low-frequency contribution of structure-borne noise, the band in which the curves lie is essentially flat over the wavenumber range $k \leq 10\ \mathrm{cm}^{-1}$.

Root-mean square (rms) values of wall-pressure fluctuations are obtained by integration of the measured spectral density of pressure-difference signals over the entire frequency domain, namely,

$$<p^2>^{1/2} = \left\{ \int_0^\infty \phi_p\ (f) df \right\}^{1/2} \quad .$$

In all computations of $<p^2>^{1/2}$ for the present study, a low-pass filter set at 10 kHz is used to include all possible contributions to pressure signals except the spurious structure-borne and acoustically induced signals. In general, the root-mean-square values of wall-pressure fluctuations are presented in dimensionless form and plotted versus the streamwise variable x/ℓ and Reynolds number Re, based on the free-stream velocity and hydraulic diameter. The dimensionless coefficients of wall pressure are defined as

$$C_p \equiv \frac{<P^2>^{1/2}}{\frac{1}{2}\rho U^2}$$

and

$$C_p \equiv \frac{<P^2>^{1/2}}{\tau_w} \quad ,$$

where ρ is the fluid density, U the free-stream velocity, and τ_w the wall-shear stress. Since wall-shear stress was not measured, Blasius' formula for smooth pipe flow (Re < 150,000) is used to obtain τ_w. A typical result is given in

Figure 1.2-6. It is seen from Figure 1.2-6 that the pressure coefficient C_p is nearly constant over the whole range of x/ℓ, regardless of incoming flow condition or rod position. The pressure coefficient for a rod in a peripheral position in the array is greater than that for the rod in the center location. Also, as might be expected, with upstream grid-generated turbulence the pressure coefficient is greater than the corresponding case without upstream disturbance.

Root-mean-square values of displacement of a peripheral rod were measured for the incoming flow without grid turbulence. The result is shown in Figure 1.2-7, together with the rms values of displacement of the same-size rod placed in annuli of d_h = 3.71 and 1.17 cm. It can be seen that the rms values of rod displacement in the large annulus are the lowest among the three cases for the same incoming flow condition. The values of rms displacement in the array channel are larger than those in the small annulus when the mean flow velocity is below 8 m/s; however, the opposite trend is observed for mean flow velocities greater than 8 m/s.

Preparation of a topical report on the measurements of wall-pressure fluctuations beneath the turbulent boundary layer on rods within a seven-rod array has been initiated. The report will also include vibration response of a rod in a peripheral location in the array.

<u>Task 1.2e - Development of Mathematical Models and Design Guides</u> (W. H. Lin, T. M. Mulcahy)

Task Description

Experimental results from subtask activities 1.2b and 1.2d will be used to evaluate and improve prediction methods for parallel flow induced vibration. The methods will be further evaluated via comparisons of predicted response with results from model and full-sized tests (Task 2). Critical flow velocities associated with the onset of fluidelastic instabilities will be expressed in terms of system parameters. Validated design guides will be developed for inclusion in an FIV Handbook.

Work Completed This Report Period

Based on the results for rod response report previously, values of the PSDs of wall-pressure fluctuations Φ_{pc} at the fundamental frequency of the rod were calculated for the case of the grid-spoiler using the ANL response prediction equation. (The equation was solved for intensity of wall pressure PSD and experimental values for rms displacement substituted.) In Table 2, calculated values are compared with measured values (Φ_{pm}) from measurement stations x/ℓ = 0.125 and 0.625. The calculated value can be considered an averaged value and the intensity of the wall pressure field, as measured by the level of the PSD curve, attenuates with axial position. Consequently, it is of interest and importance that the calculated values fall between the measured values for each of the flow rates tested. The ANL equation is based on the assumption of a spatially homogeneous pressure field. The results given in Table 1.2-2 serve to give confidence in the equation and also suggest that the model can be improved by modifying it to include the axial decay in intensity of the wall pressure PSD. This modification to the model has been initiated during the reporting period.

Reference

1. T. M. Mulcahy, M. W. Wambsganss, W. H. Lin, T. T. Yeh, W. P. Lawrence, **Measurements of Wall Pressure Fluctuations on a Cylinder in Annular Water Flow with Upstream Disturbances; Part I: No Flow Spoilers**, GEAP-24310, DOE/N/4175-15, ANL-CT-81-11, January 1981.

Table 1.2-1

EXPERIMENTAL RESULTS OF COUPLED NATURAL FREQUENCIES AND MODAL DAMPING FACTORS

		Coupled Natural Frequency, Hz		Modal Damping Factor, %	
U, m/s	e, cm	f_x	f_y	ζ_x	ζ_y
	0	31.5	30.8	3.94	4.21
4	0.32	30.2	29.8	4.25	4.80
	0.48	30.4	29.7	4.51	3.95
	0.56	28.8	28.8	5.26	5.40
	0	31.2	30.9	4.28	5.45
6	0.32	30.2	29.8	4.39	4.94
	0.48	30.4	29.7	4.37	4.62
	0.56	29.2	29.8	5.19	6.41
	0	31.2	30.8	5.20	5.38
8	0.32	30.5	29.8	5.63	4.91
	0.48	30.5	30.0	4.84	6.57
	0.56	29.8	30.6	6.83	6.84
	0	31.2	30.8	5.68	6.14
10	0.32	30.7	30.2	5.41	6.61
	0.48	30.8	30.6	5.55	6.66
	0.56	30.3	31.1	8.98	7.63
	0	31.1	30.3	6.31	6.61
12	0.32	30.8	30.2	6.35	6.94
	0.48	31.2	31.3	6.87	6.42
	0.56	30.5	31.0	9.54	11.10
	0	30.7	30.3	6.14	6.06
14	0.32	31.1	30.5	6.32	6.56
	0.48	31.8	31.5	6.87	8.44
	0.56	31.3	31.6	13.10	14.20

Table 1.2-1

EXPERIMENTAL RESULTS OF COUPLED NATURAL FREQUENCIES AND MODAL DAMPING FACTORS (Continued)

		Coupled Natural Frequency, Hz		Modal Damping Factor, %	
U, m/s	e, cm	f_x	f_y	ζ_x	ζ_y
	0	30.6	31.0	5.52	6.76
16	0.32	31.4	31.0	6.41	6.58
	0.48	32.3	31.8	9.77	11.70
	0.56	32.0	32.2	13.20	16.90
	0	30.5	30.0	6.73	7.41
18	0.32	31.5	31.2	7.97	8.14
	0.48	32.5	31.8	10.20	16.40
	0.56	32.7	31.5	14.00	22.0
	0	30.6	29.9	6.66	7.07
20	0.32	31.3	31.0	8.86	9.11
	0.48	33.0	32.3	11.80	16.80
	0.56	32.8	32.9	18.30	22.10
	0	31.0	30.3	6.59	7.27
22	0.32	31.7	31.4	9.12	10.90
	0.48	33.5	32.1	12.10	21.60
	0.56	35.0		15.60	

Table 1.2-2

CALCULATED VS. MEASURED VALUES OF INTENSITY OF WALL PRESSURE PSD* (d_h = 1.17 cm, $\Phi_p \backsim Pa^2/Hz$)

U, m/s	$\Phi_{pm}\big\vert_{\frac{x}{\ell}} = 0.125$	Φ_{pc}	$\Phi_{pm}\big\vert_{\frac{x}{\ell}} = 0.625$
6	240	136	64
8	630	420	139
10	1,400	999	233
15	16,000	6,200	418
20	18,000	14,800	768

*Φ_{pc} ≡ Calculated value

Φ_{pm} ≡ Measured value

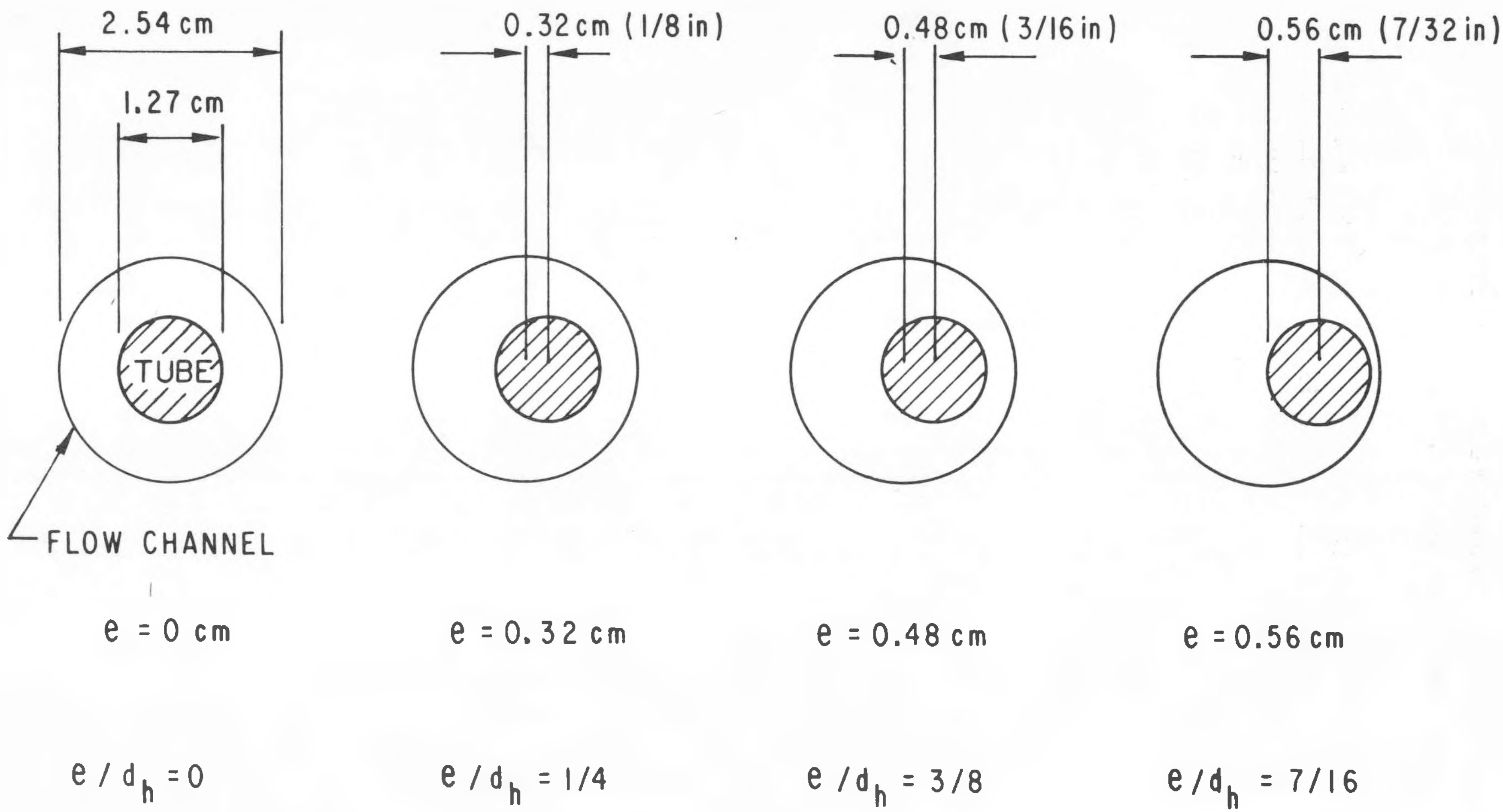


Figure 1.2-1. Positions of the Tube in the Flow Channel

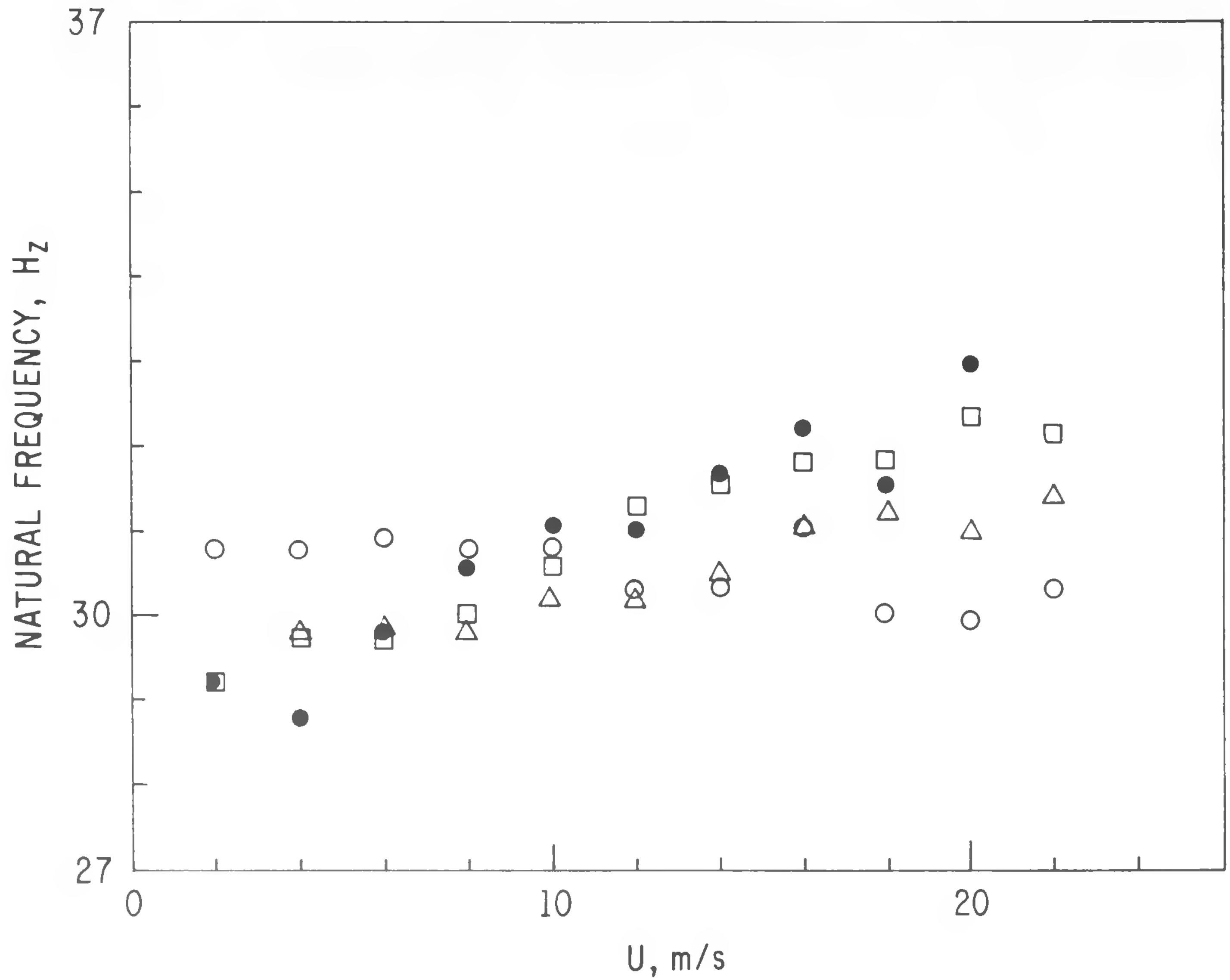


Figure 1.2-2. Fundamental Frequencies of Tube Oscillating in the y Direction;
○ - Tube Centered, △ - 1/8 in. Ecc., □ - 3/16 in. Ecc.,
● - 7/32 in. Ecc.

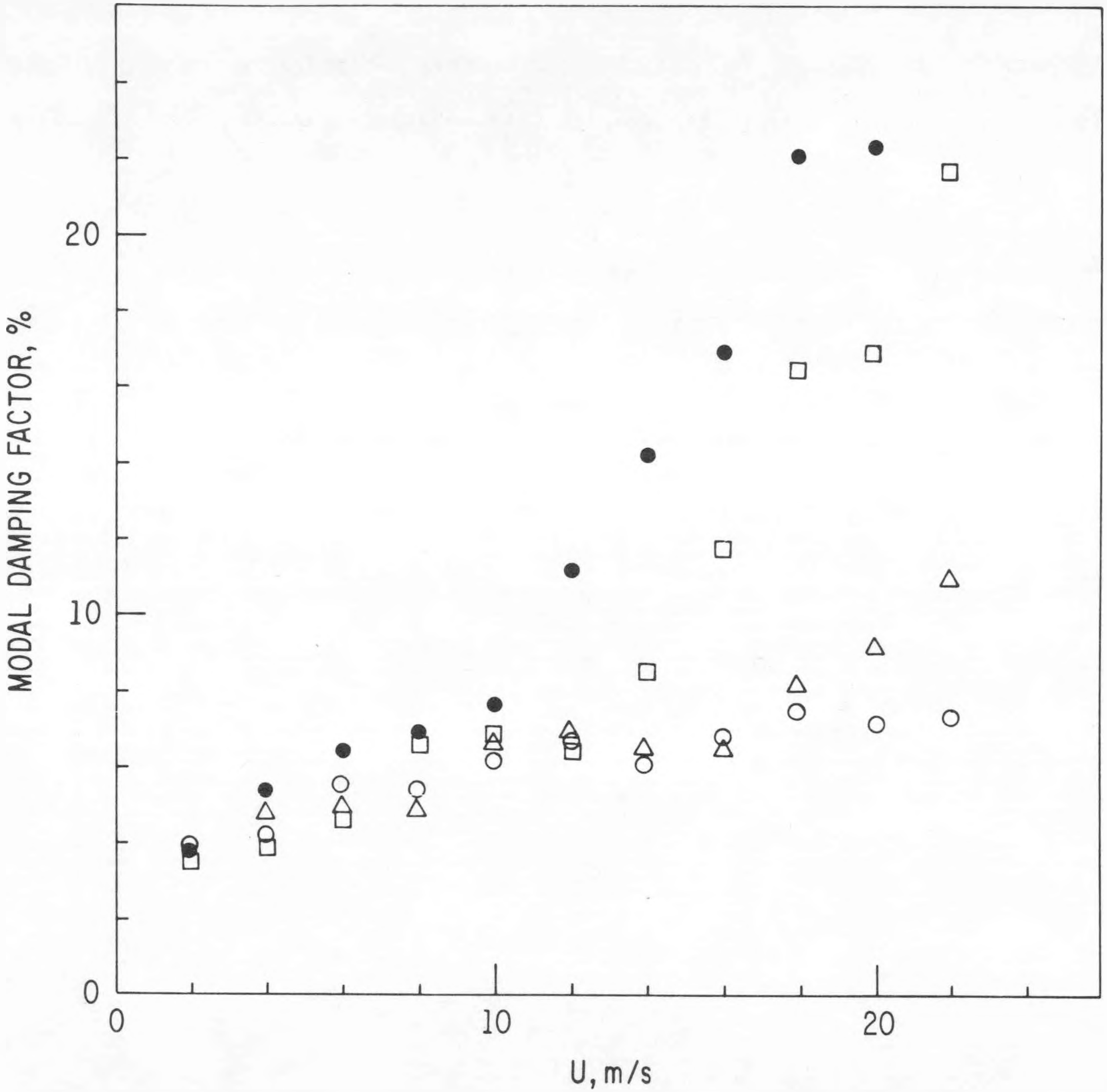


Figure 1.2-3. Modal Damping Factors of Tube in the y Direction; ○ - Tube Centered, △ - 1/8 in. Ecc., □ - 3/16 in. Ecc., ● - 7/32 in. Ecc.

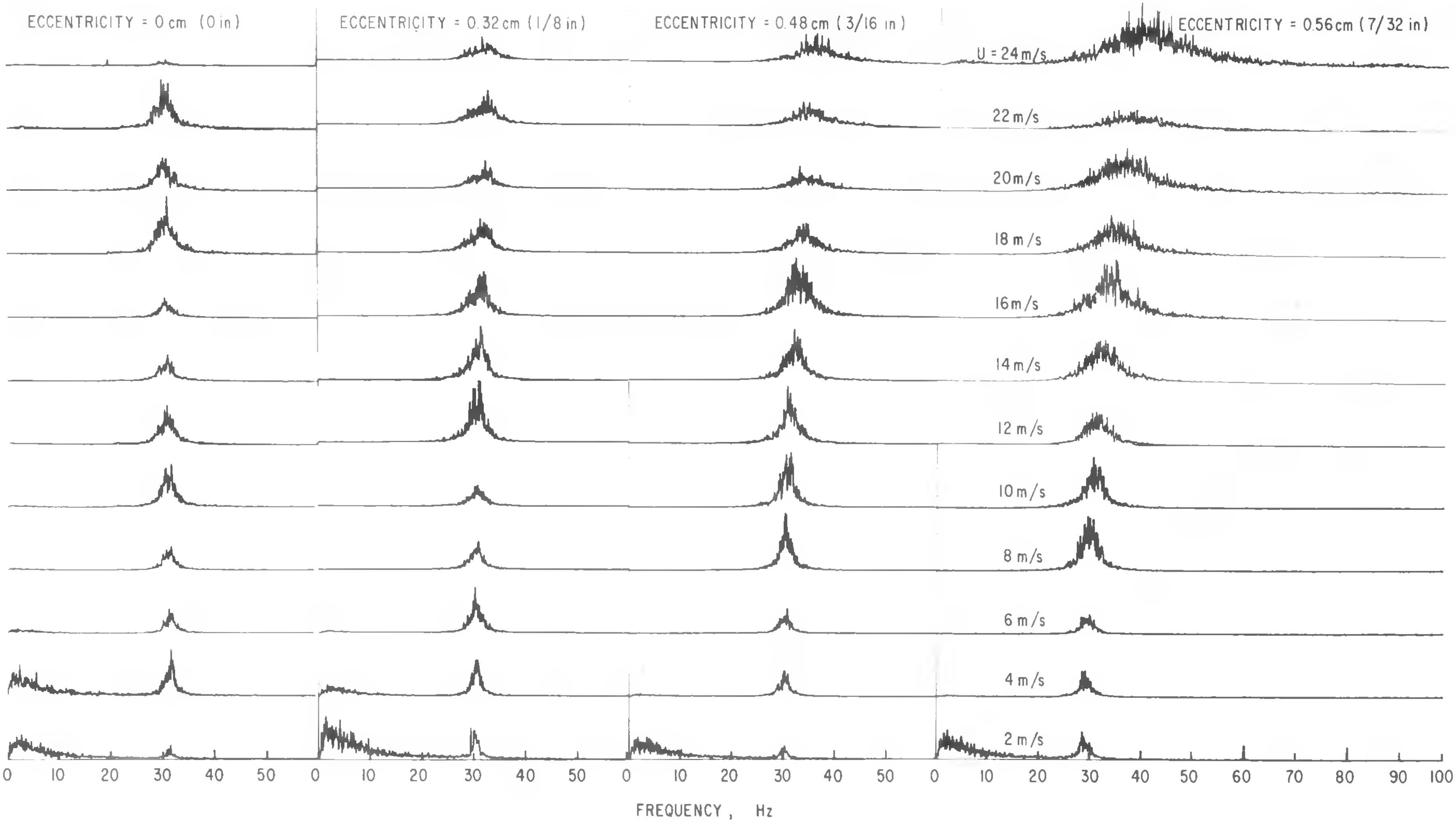


Figure 1.2-4. Acceleration PSDs as a Function of Eccentricity and Flow Speed

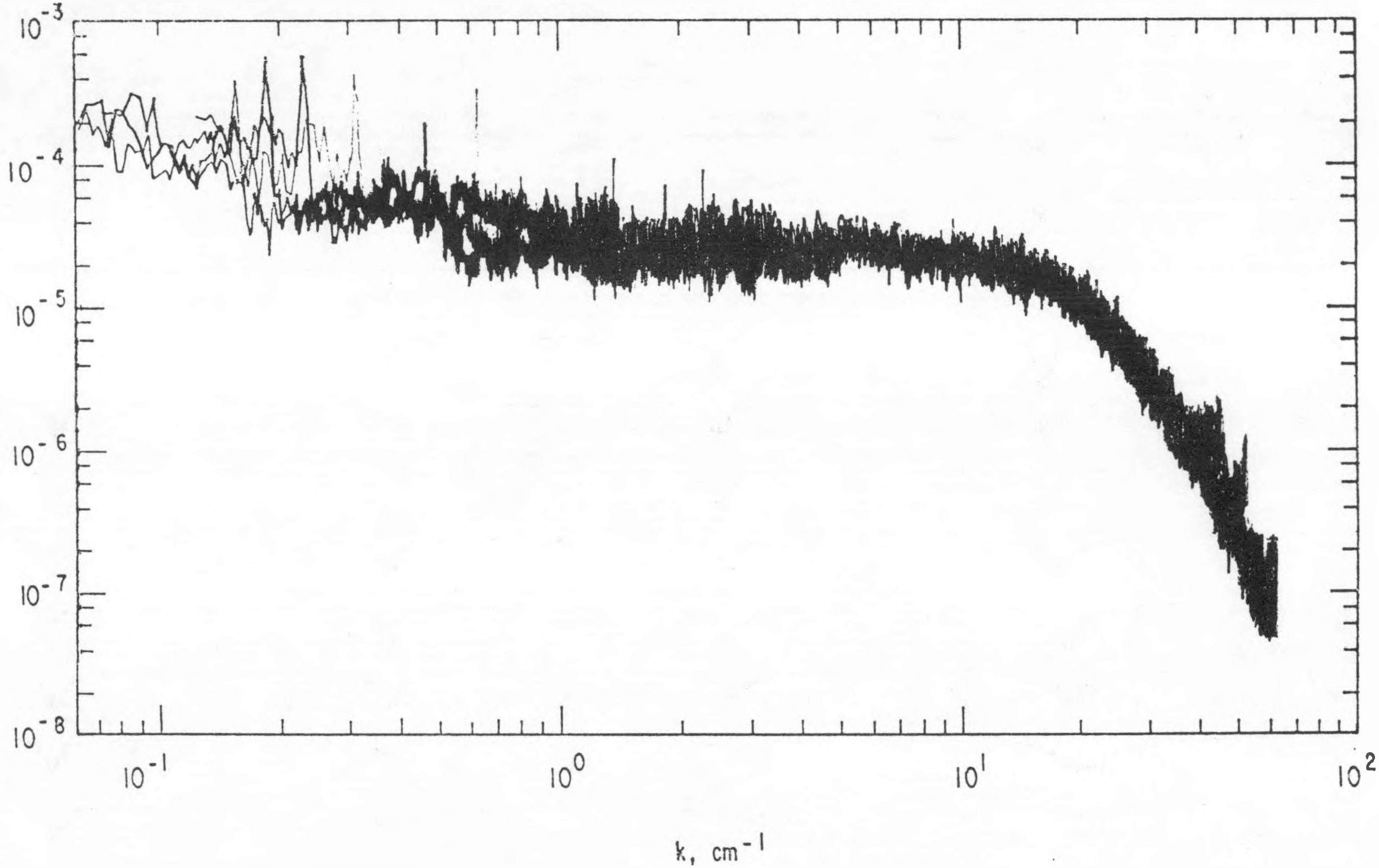


Figure 1.2-5. Dimensionless Power Spectra of Wall Pressure on the Center Rod; 0^{o}, No Grid; x/ℓ = 0.125

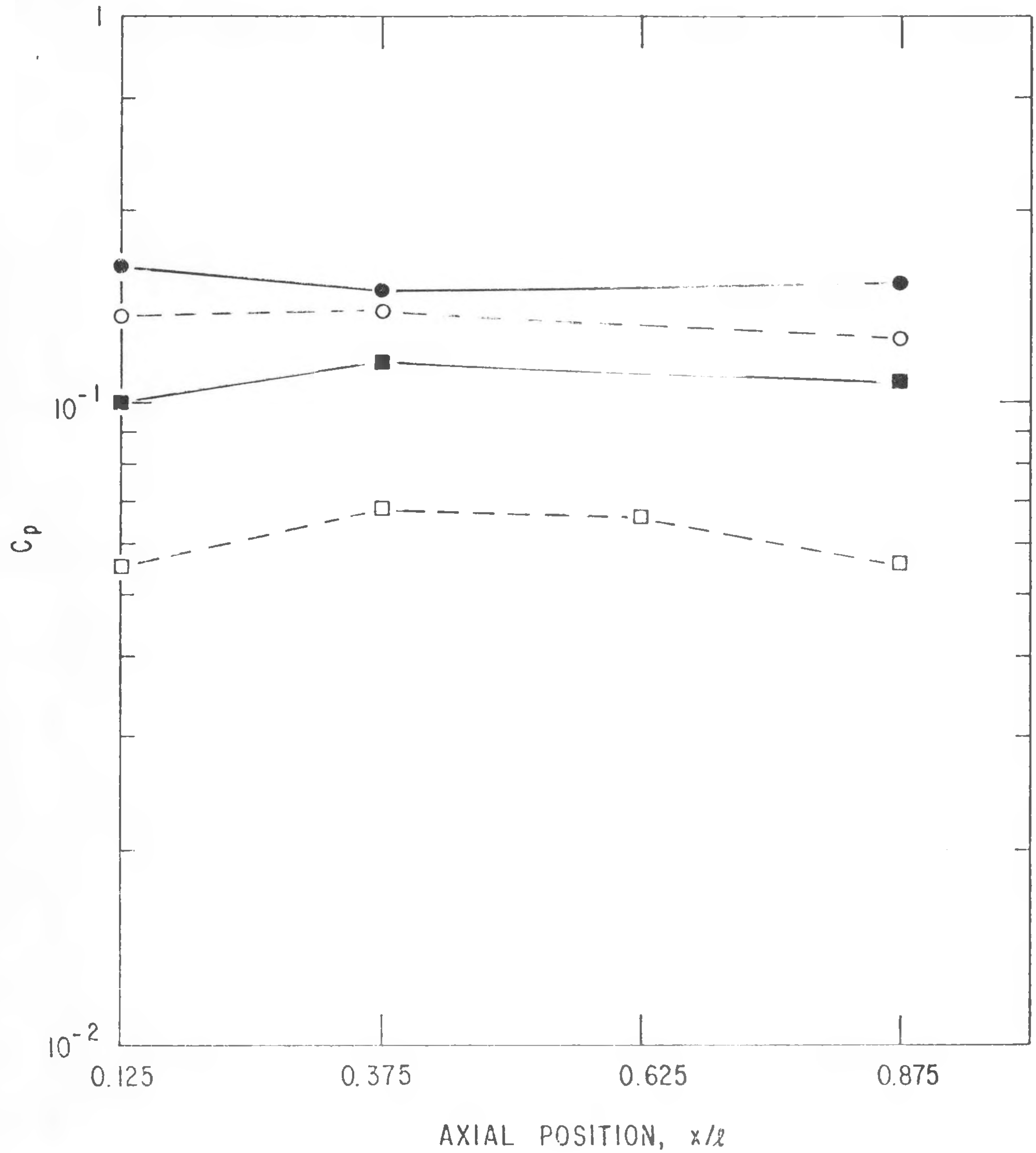


Figure 1.2-6. RMS Wall Pressures as a Function of Axial Position U = 12 m/s; □ (Center, 0°, No Grid), ■ (Center, 0°, Grid), ○ (Periphery, 0°, No Grid), ● (Periphery, 0°, Grid)

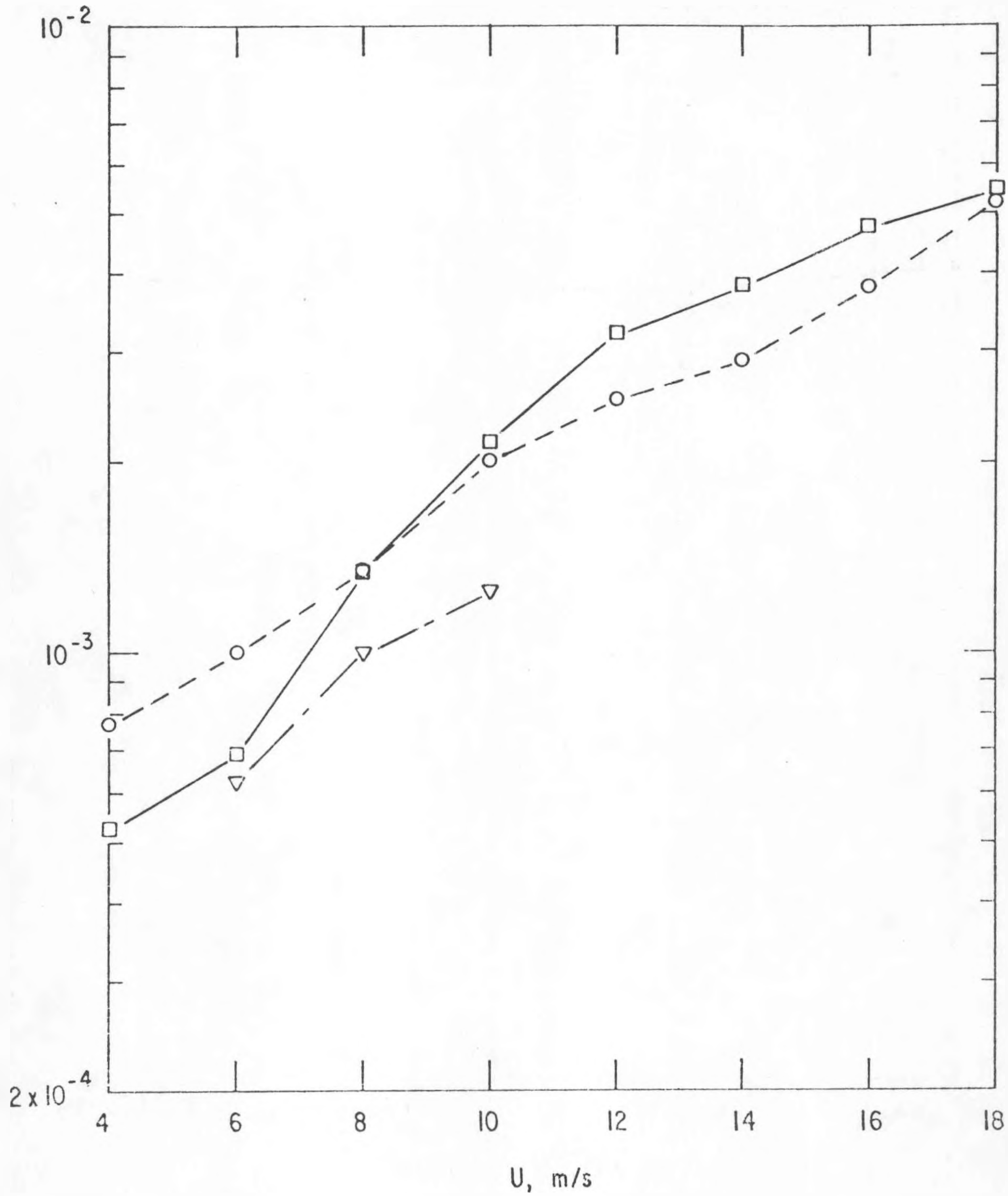


Figure 1.2-7. Rod Displacements as a Function of Mean Flow Velocity; ○ Seven-rod Array; ▽ Rod in Annulus, d_h = 3.71 cm; □ Rod in Annulus, d_h = 1.17 cm

TASK 1.3 - LEAKAGE FLOW MECHANISMS (CR&D AND NED)

Task 1.3, performed by General Electric Corporate Research and Development and Nuclear Engineering Division, is designed to establish a better understanding of leakage-type flow-induced vibrations for geometries common to light water reactors. The technical objectives include the following:

(1) Mathematically model and classify components involving leakage flows. Identify flow/geometric and structural features leading to an instability.

(2) Experimentally illustrate the general classification with critically selected experiments.

(3) Contribute input to design guides for critical reduced velocity U/f_nL as a function of flow/geometric and structural features to eliminate or reduce possibility of self-induced vibrations. Proof test the design guide by application to a selected LWR component.

Task Description

Task 1.3 is divided into four subtasks:

Task 1.3a - Modeling and Classification of Classes of Components with Leakage Flow-Induced Vibration Susceptibility

Task 1.3b - Simple Experiments to Illustrate Concepts and Governing Parameters Identified by Mathematical Models

Task 1.3c - Proof Test of Evolving Design Guides on a Selected LWR Component

Task 1.3d - Leakage Flow-Induced Vibration Design Guide

Work Performed This Report Period

Activity on this task has not been initiated as yet since it is not yet funded.

TASK 1.4 - RANDOM VIBRATION EFFECTS (ANL AND NED)

Task 1.4 consists primarily of a literature survey to review existing random vibration fatigue damage rules and experimental data related to interpretation of FIV data. Based on this review, suitable random fatigue damage criteria will be developed and recommended for application to LWR components subjected to flow-induced vibration.

Task Description

Task 1.4 is broken into three subtasks:

Task 1.4a - Survey of Literature

Task 1.4b - Derivation of Procedure/Criteria

Task 1.4c - Design Guides

Work Performed This Report Period

This task has been completed.

TASK 2 - MODEL AND FULL-SIZE COMPONENT TESTS

Task 2 is devoted to the FIV testing of realistic model and prototypical geometries, rather than the idealized geometries studied in Task 1. Model and full-scale testing of several LWR components will be performed to extend the Task 1 results and provide input for the development and verification of empirical relations, scaling laws, and predictive models generated as part of Task 3. The components tested will, in general, be from General Electric's most recent boiling water reactor product line, the BWR/6. Task 2 is divided into five component test programs:

Task 2.1 - Lower Plenum

Task 2.2 - Fuel Assembly

Task 2.3 - Jet Pump

Task 2.4 - Low Pressure Coolant Injection Line

Task 2.5 - LWR Component (undefined)

Progress achieved for each of these component tests for the report period is described on the following pages.

TASK 2.1 - LOWER PLENUM (NED)

Task 2.1 is an experimental FIV study of tubes in an array. The array is a mix of tubes with different diameters. The flow environment combines regions of cross flow and parallel flows. The results from this task will be evaluated using results from the fundamental studies (Tasks 1.1 and 1.2) in order to highlight the imporant FIV mechanisms in real LWR components. The first steps in this study were to perform FIV tests on tenth-scale and quarter-scale models of a BWR lower plenum. This provided preliminary data and the experience necessary to run a productive full-scale model test. The final step in this process is to measure vibration in an LWR before power generation startup. This will verify the more extensive full-scale model results.

Task 2.1 consists of four subtasks:

Task 2.1a - Lower Plenum One-Tenth Scale FIV Test

Task 2.1b - Lower Plenum Quarter-Scale FIV Test

Task 2.1c - Lower Plenum Full-Scale FIV Test

Task 2.1d - Lower Plenum Operational LWR Field FIV Test

Progress achieved on Task 2.1 is described on the following pages.

<u>Task 2.1a - Lower Plenum One-Tenth Scale FIV Test</u> (M. A. DeCoster, S. L. Kushman)

Task Description

A one-tenth scale model of a BWR lower plenum was constructed of clear cast epoxy to study the flow patterns in the lower plenum via flow visulization in 1974. Later, velocity probes were inserted into the flow stream to obtain quantitative measurements of the flow distribution. In 1976, flow-induced vibration tests were initiated to determine the vibrational characteristics of control rod guide tubes in the lower plenum. The tests were completed in late 1976.

Work Performed This Report Period

The one-tenth-scale lower plenum model test and analysis program was completed during a previous report period. Results were documented in GE report GEAP-24124, "Flow-Induced Vibration for Light Water Reactors Progress Report (June - September 1977)."

Task 2.1b - Lower Plenum Quarter-Scale FIV Test (M.A. DeCoster)

Task Description

A quarter-scale model of a BWR lower plenum was constructed by GE in 1974 to study flow distribution and vibration levels of lower plenum components. The model was situated at the Colorado State University Hydro-machinery Laboratory. Flow and shake tests performed in 1974, 1975, and 1976 yielded extensive flow distribution vibration data for various designs. The final tests were completed in the last quarter of 1976.

Work Performed This Report Period

The quarter-scale lower plenum model test and analysis program was completed prior to the commencement of the Flow-Induced Vibration for Light Water Reactors program. Results were documented in GEAP-24124.

Task 2.1c - Lower Plenum Full-Scale FIV Test (M.A. DeCoster)

Task Description

Task 2.1c consists of the FIV testing of prototype LWR components in the lower plenum of the 60° sector model at the High Flow Hydraulic Test Facility (HF)^2. The test components were made at the same facility and to the same standards as actual reactor components. The components are full scale but the sector model contains one-sixth the number of components in a reactor. The components in the lower plenum are cylinders of various diameters in an array pattern. Water flows around the outside of these components.

The test program consists of a series of shake tests followed by flow tests. In the shake tests, a single component is forced to vibrate by an electromagnetic shaker or impact hammer. The input force and resulting motion are measured, and the information is processed to define the structural characteristics of the component. In the subsequent flow tests, the vibration response of the components, as well as certain aspects of the FIV forces, are measured for a variety of operational and nonoperational flow conditions.

Work Performed This Report Period

All work has been completed with the exception of a topical report which is currently underway.

TASK 2.2 - FUEL ASSEMBLY (NED)

Task 2.2 is designed to fully investigate the FIV characteristics of a prototype fuel assembly. The primary objectives are to:

(1) Establish the FIV character of a typical fuel assembly including mode shapes, frequencies, character of response, and assessment of the stability margin as a function of system damping and flow;

(2) Determine the relationship between the FIV characteristics and coolant properties; i.e., coolant temperature (ambient versus LWR operational conditions) and quality (single-phase water versus two-phase water-stream); and

(3) Proof-test the concepts and forcing function estimates devised in Task 1.

Task 2.2 will be useful in verifying and extending the fundamental studies on parallel flow over tube banks of Task 1.1. It will provide input to the analytical model development of Task 3.1 and the scaling development (importance of temperature) of Task 3.2.

Task 2.2 is divided into three test programs:

Task 2.2a - Fuel Assembly Cold FIV Test

Task 2.2b - Fuel Spacer Preload FIV Test

Task 2.2c - Hot FIV Test

A description of progress achieved during this report period is given in the following pages.

Task 2.2a - Fuel Assembly Cold FIV Test

Task Description

Task 2.2a consists of the FIV testing of a prototype 8 x 8 fuel assembly over a wide range of hydraulic conditions using the General Electric Large Tank Hydraulic Loop. The fuel assembly was instrumented with accelerometers, pressure transducers, and strain gages and it measured the dynamic characteristics of the fuel assembly components as well as to describe the forcing mechanisms. The test was performed at a water temperature of approximately 75^{o}F.

The fuel assembly was made up of 64 Zircaloy rods, each approximately 16-in. long, 1/2-in. in diameter, and 32-mils thick. The rods are spaced and supported in a square (8 x 8) array by a lower and upper tie-plate. The lower tie-plate has a nosepiece which distributes coolant flow to the fuel rods. Seven spacers, evenly spaced along the rods, maintain rod-to-rod spacing. Each spacer provides a three-spring support for the fuel rod: two stiff springs contacting the rod at points azimuthally separated by 90 degrees, and a third flexible spring on the opposite side of the rod, azimuthally separated from the other two springs by 135 degrees. Three-point contact is maintained by establishing an initial preload (i.e., sizing the springs such that they are compressed when the fuel rod is first installed in the spacer). Finally, the fuel bundle was enclosed by a square channel box, causing the coolant flow to be almost exclusively parallel in nature.

Work Peformed This Report Period

The cold flow-induced vibration testing of a prototypical 8 x 8 fuel assembly was completed early in the program and was reported in previous progress reports.

Task 2.2b - Fuel Spacer Preload FIV Test

Task Description

These tests will investigate the effect of fuel spacer spring preload on the FIV characteristics of fuel rod. Specifically, the test results should indicate:

(1) At what flow-preload conditions and the fuel rod loses contact with the spacer springs and begins vibrating within the spacer.

(2) The vibration levels (rod-spacer impact acceleration levels) that result when the conditions in (1) above are met.

(3) The effect of spacer spring preload and stiffness on fuel rod FIV characteristics (frequency and amplitude).

The test program will consist of the following elements:

(1) Single-rod flow test,
(2) 3 x 3 rod bundle flow test.

The tests will be performed at the General Electric High Flow Hydraulic Facility.

Single Rod Flow Test

The test setup is shown in Figure 2.2-1. A single full-scale fuel rod is supported by segments of actual lower and upper tie plates as shown. The inside diameter of the test cylinder has been sized to approximate the hydrodynamic mass and hydraulic diameter of a single rod in an actual fuel assembly. The simulated fuel spacer location and preload adjustment fixture are shown in Figure 2.2-2. The adjustment fixture will allow for easy adjustment of spring preloads. The locations of both the preload fixture and hard supports will match the spacer locations of actual fuel assemblies.

Several different instrumented rods are used to determine fuel FIV characteristics. The fuel rods to be used with their respective instrumentation are shown in Figure 2.2-3.

Work Performed This Report Period

The single rod preload flow induced vibration tests at the High Flow Hydraulic Facility were completed during this report period. All tests were performed with cold ($\sim 70^{\circ}F$), single phase water. Fuel rods were tested with and without rod suspension. Figure 2.2-4 illustrates the difference between these two configurations. All fuel rods in the test (dummy and instrumented) were loaded with lead pellets to provide the correct fuel rod mass. All rods were within 6 percent of actual fuel weight.

Typical test results of impact accelerations measured with accelerometers in the rod are shown in Figure 2.2-5 and Figure 2.2-6.

The following preliminary conclusions have been determined from these tests.

(1) Impact levels increase appreciably with increases in fluid velocity (flow rate) in all of the experiments.

(2) For the single rod case with its dead weight removed, as preload decreases, relative impact (acceleration) levels between the rod and spacer increase by a factor of two (2) in going from 100 percent to 10 percent of initial preload.

These observations were made for initial preload values of 2.5 lbs.

3 x 3 Rod Bundle Flow Test

This test will utilize the same instrumented fuel rods described for the single rod test, surrounded by eight dummy (non-instrumented) rods (Figure 2.2-7). Actual fuel spacer equipment will be used, with the preload being adjusted by controlling the amount of spring deflection caused by rod insertion into the fuel bundle. Only the preload on the central, instrumented rod will be controlled.

The range of preloads to be tested is 0-3 pounds, which is inclusive of the preloads nominally encountered in present fuel designs.

Work Performed This Report Period

The 3 x 3 rod bundle flow tests were nearly completed in March. Preliminary data analysis shows that the amplitude of vibration (impact levels) in the 3 x 3 array is about the same as the maximum amplitude of the single rod cases.

Task 2.2c Hot FIV Test

Task Description

Flow tests will be performed on a 3 x 3 array of fuel rods under reactor temperature and pressure (550°F, 1050 psia) at the General Electric heat transfer test facility ATLAS. This is the final test in the fuel area. The primary purpose is to determine the effect of two-phase boiling conditions on fuel rod vibrations. The test will be run in four steps:

(1) Single-phase water flow at 180°F to compare to results from Task 2.2b.

(2) Single-phase hot flow at 550°F to determine any effect from temperature.

(3) Two-phase hot flow with no heat input to the fuel rod (constant quality).

(4) Two-phase hot flow with nucleate boiling from rod heating.

Work Completed This Report Period

A team of engineers and technicians from different organizations are contributing to this test. The responsible engineer is holding bi-weekly progress meetings. All test preparation work is proceeding on schedule. This includes fabrication of special test hardware, writing test procedures, assembling the instrumentation package, and preparation of computer programs specifically needed for this test. The test is scheduled for August of this year with completion before the end of August.

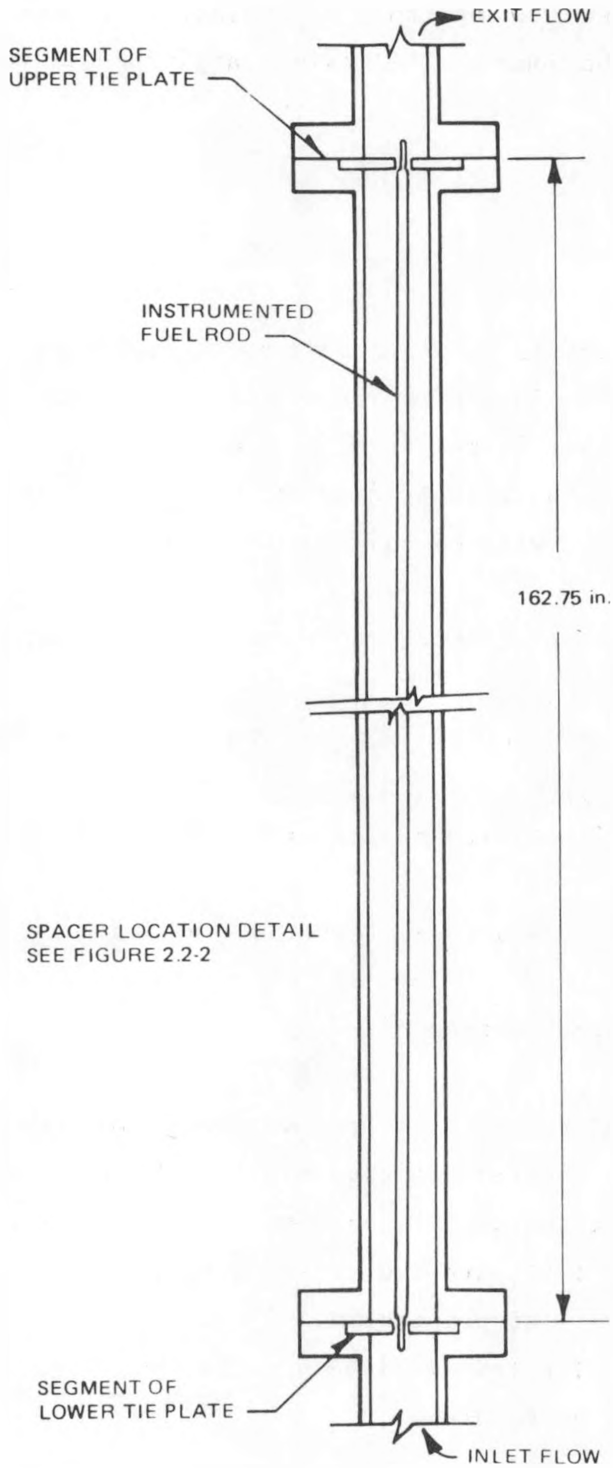


Figure 2.2-1. Test Setup (Single-Rod Test)

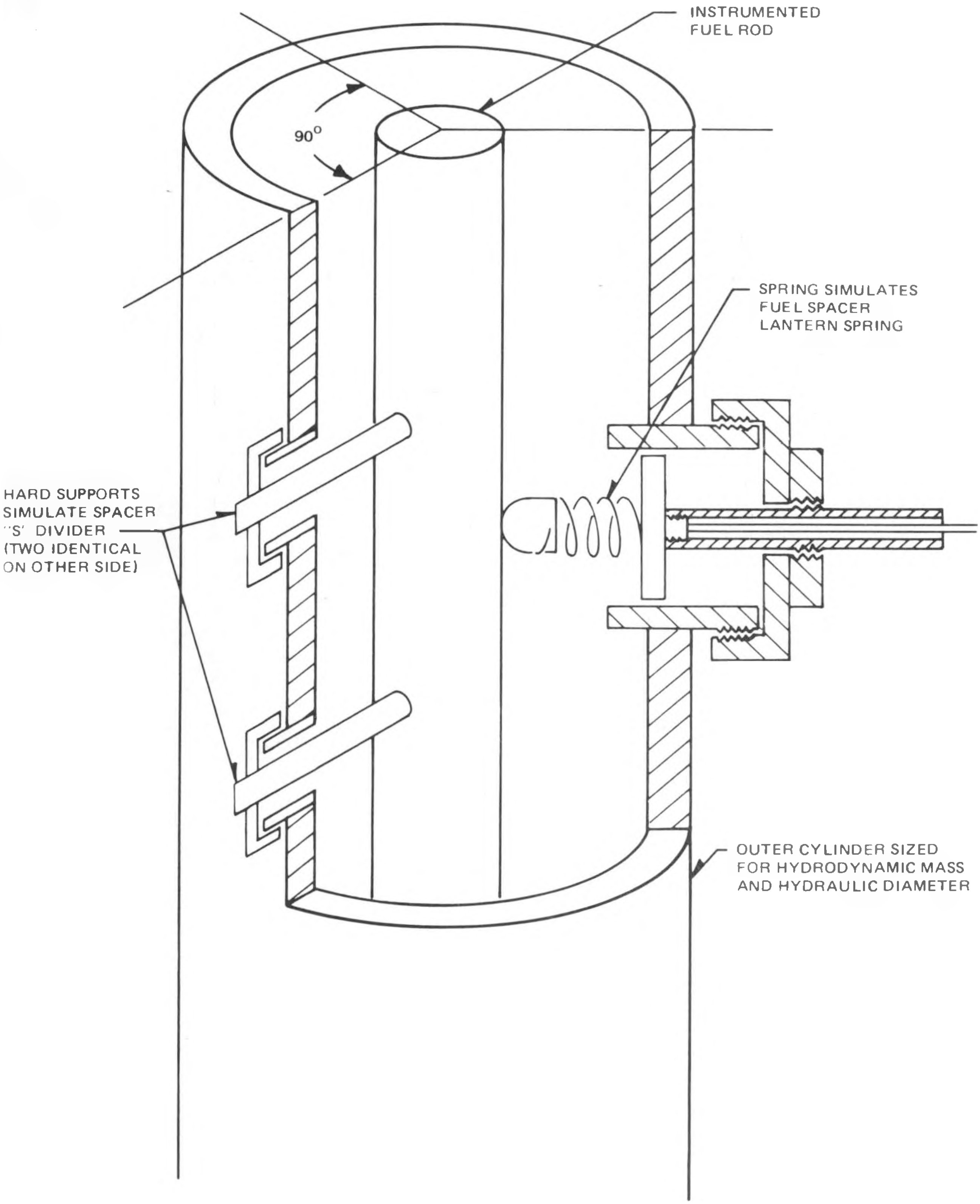


Figure 2.2-2. Spacer Location Detail

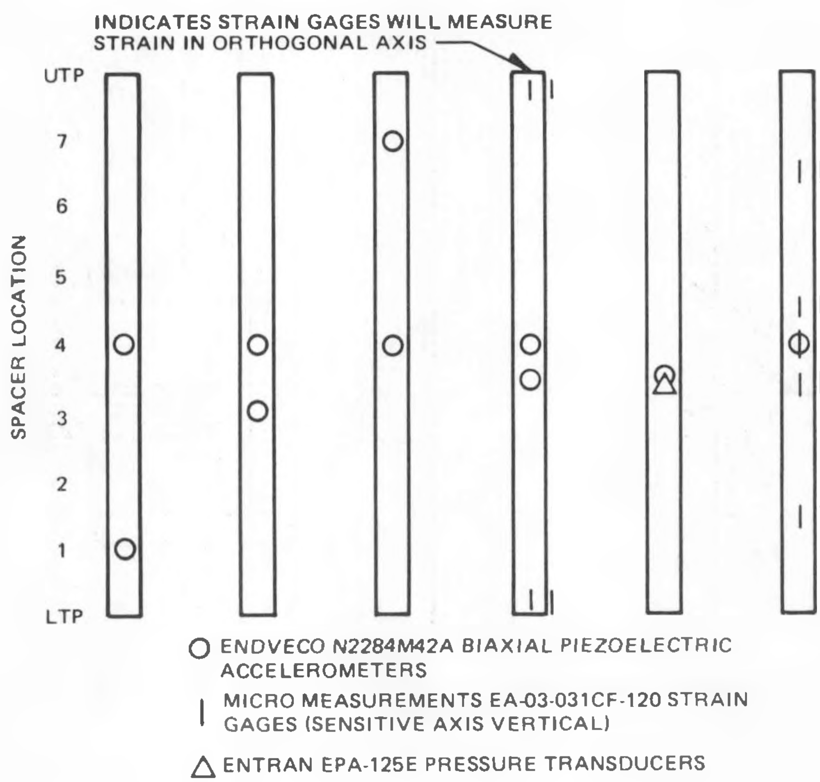


Figure 2.2-3. Instrumented Fuel Rods

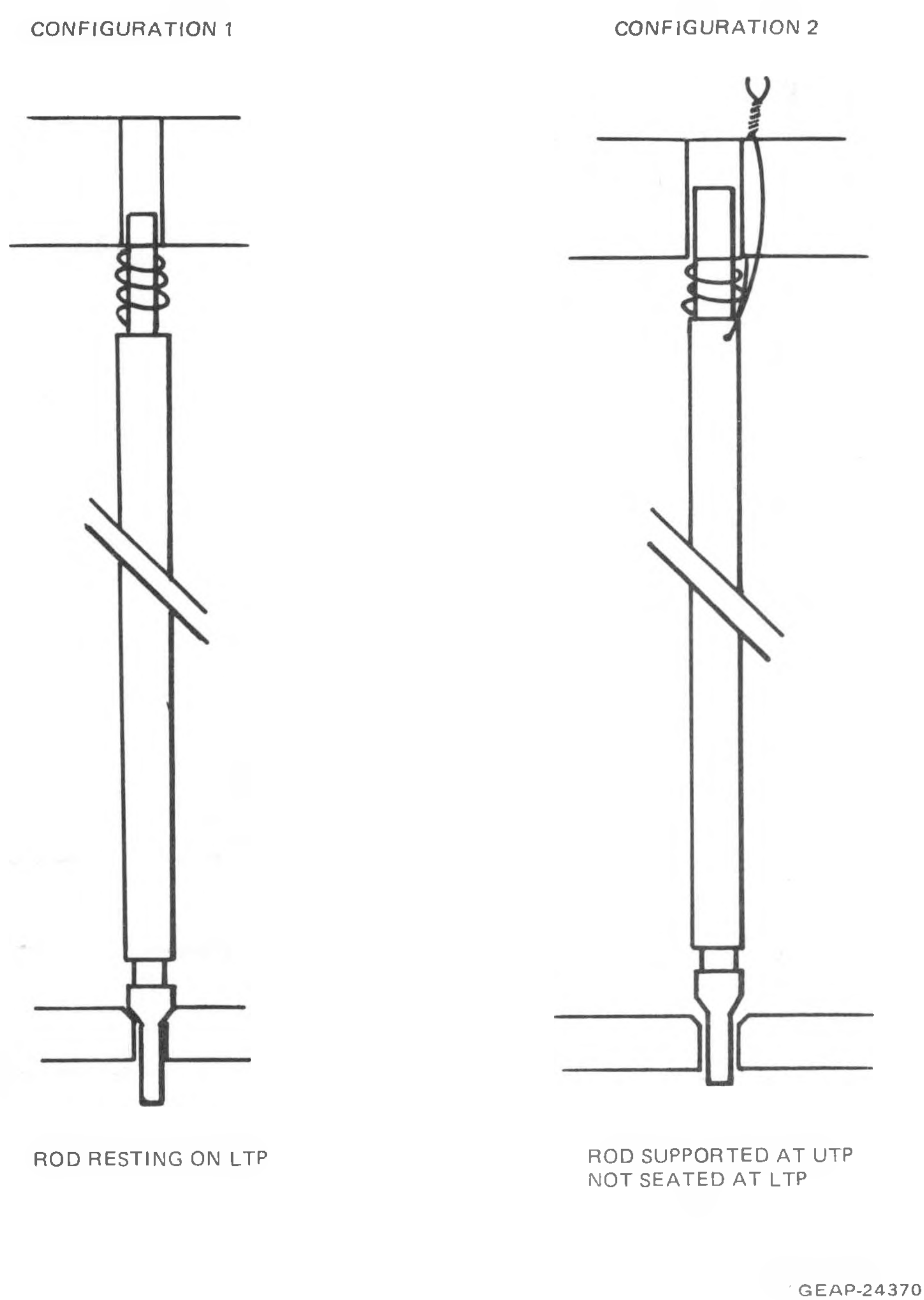


Figure 2.2-4. Test Configurations, Instrumented Rod Only

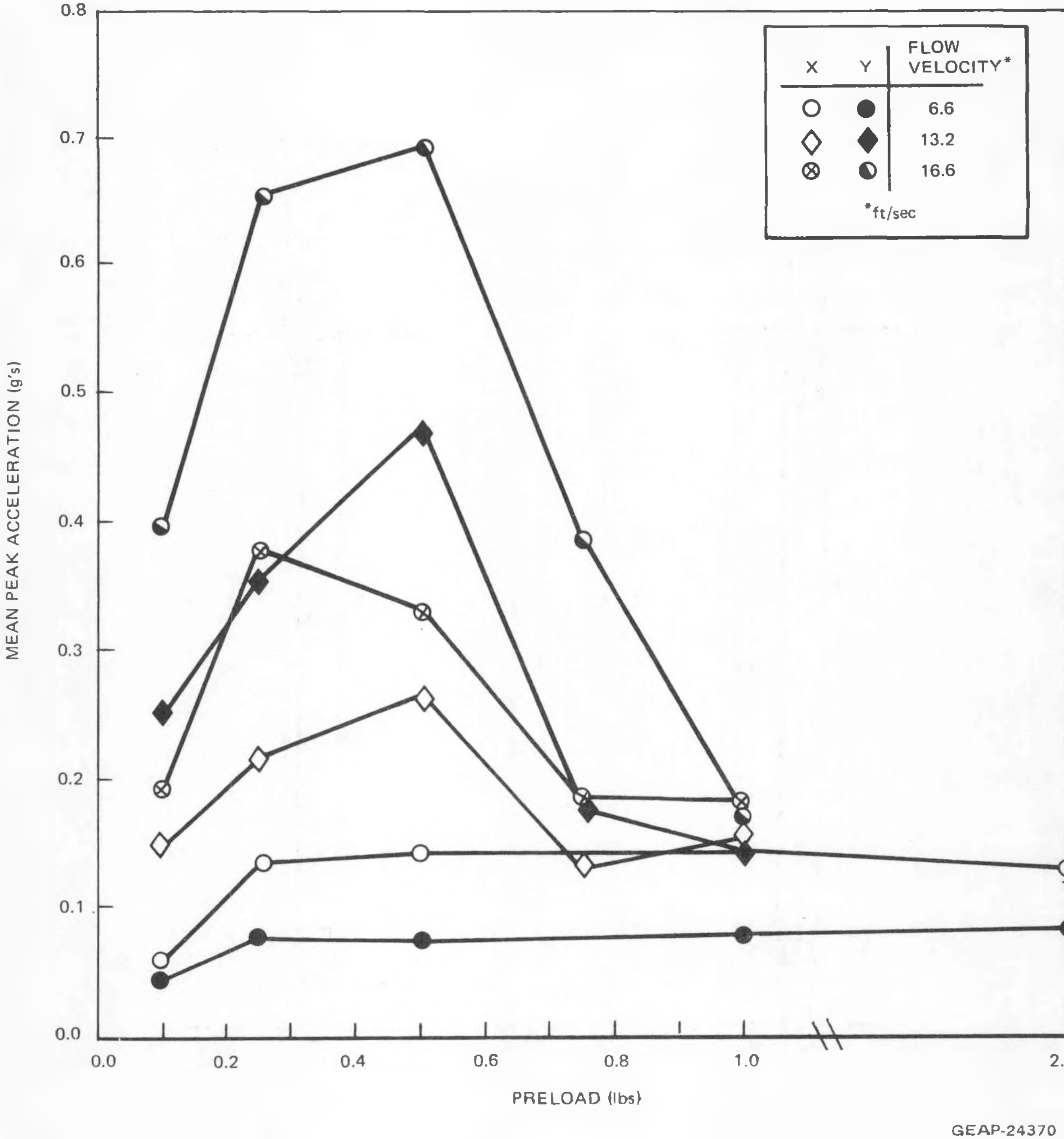


Figure 2.2-5. Fuel Spacer Preload Test, Fuel Rod Vibration at Middle Spacer

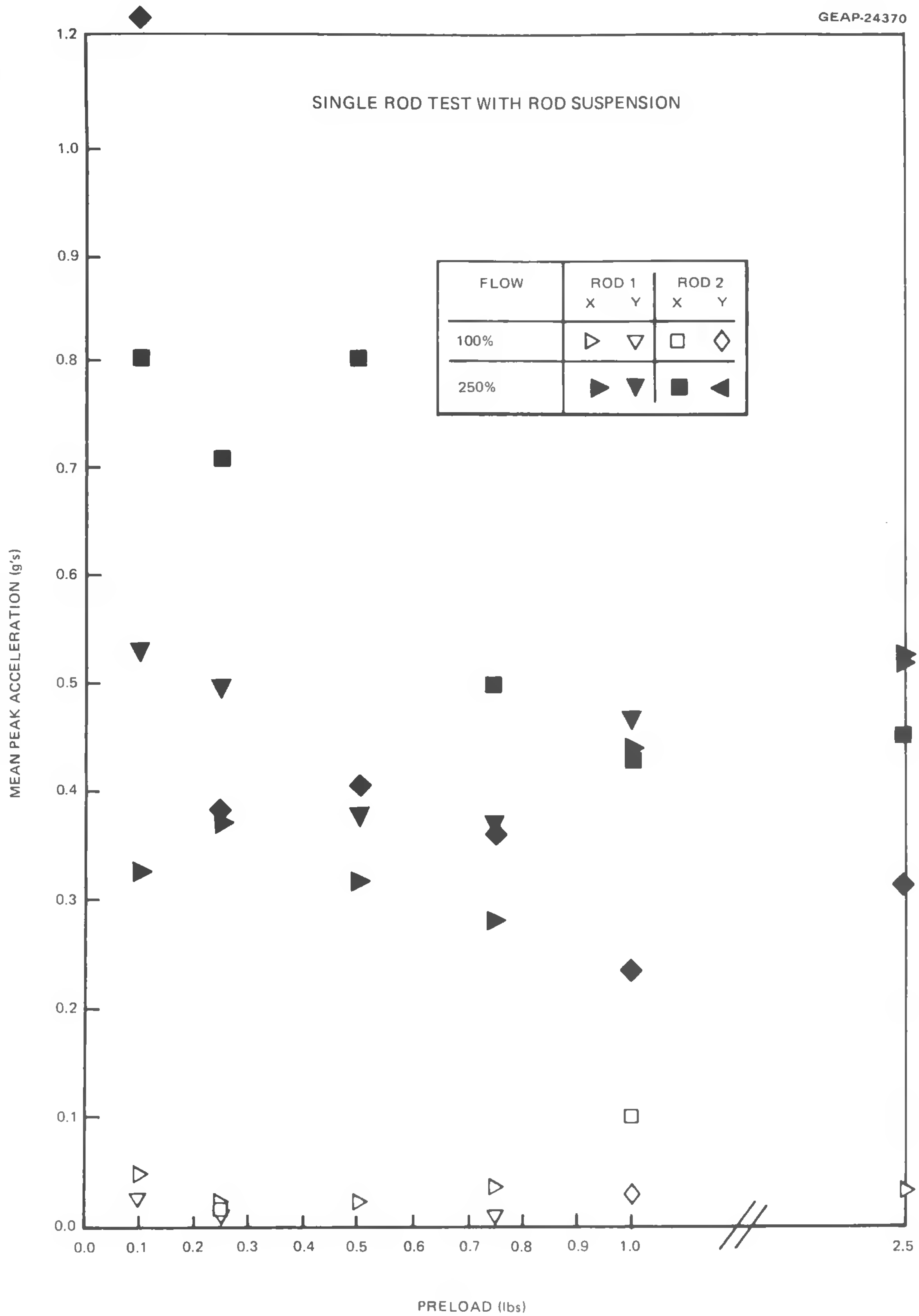


Figure 2.2-6. Fuel Rod Acceleration at Middle Spacer Elevation

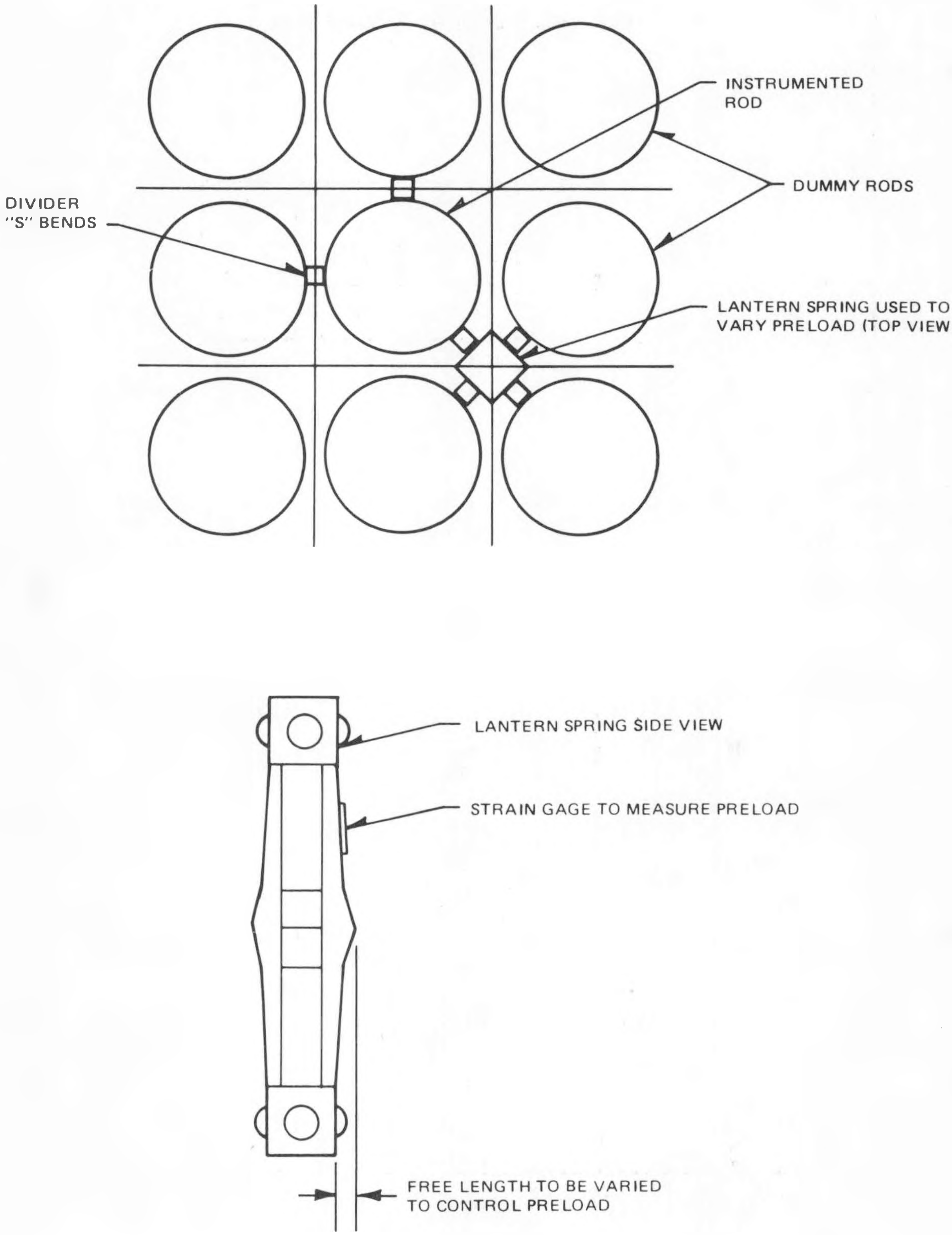


Figure 2.2-7. 3 x 3 Rod Bundle Overview

TASK 2.3 - JET PUMPS

Task 2.3 is designed to fully investigate and quantify the FIV characteristics of a jet pump assembly. A typical jet pump assembly is shown in Figure 2.3-1. There are 20 jet pumps in the current BWR design. Jet pump primary components are a mixer and diffuser which are joined together at a thermal expansion slip joint. Approximately one-third of the reactor core flow is taken from the vessel, pumped through a piping manifold system, and returned to the vessel, where it is discharged from the jet pump nozzle at high velocity ($\sim$200 ft/sec) into the initial stage of the jet pump mixer. The nozzle flow entrains the surrounding water into the jet pump throat where the two flows mix and then diffuse, to be finally discharged into the lower core plenum at velocities up to 40 ft/sec. The jet pump has many characteristics that can contribute to flow-induced vibrations (e.g., high-velocity flow in small pipes, high-velocity flow in a pipe turning 180° within a small radius, and high-velocity jets exiting into a confined region). However, the two characteristics which are judged to be the most significant contributors are, in order of magnitude:

- Leakage flow through the mixer-diffuser slip joint interface.

- Mixing of the high-velocity drive flow with the entrained suction flow in the mixer.

Jet pump tests will be performed hot (550°F) and cold (60 - 180°F), under operational and nonoperational conditions, at rated and above-rated flows. Specific goals include the following: (1) identify the vibration characteristics of a typical jet pump (natural frequencies, mode shapes, damping); (2) characterize the primary forcing functions; (3) establish a jet pump response map (i.e., response versus flow conditions); (4) relate jet pump stability to system damping, flow conditions, geometry, etc; and (5) determine the effect of fluid temperature on the FIV characteristics. This work will be directly tied to the fundamental studies performed in Task 1 (in particular, the leakage flow mechanisms work of Task 1.3) for input into the Design Guide Development work of Task 3.1. In addition, it will be utilized in developing temperature scaling relationships in Task 3.2.

Task 2.3 is divided into three subtasks:

Task 2.3a - Jet Pump Cold FIV Test

Task 2.3b - Jet Pump Hot FIV Test

Task 2.3c - Jet Pump Mechanistic Test

Task 2.3a - Jet Pump Cold FIV Test

Task Description

Four prototype BWR/6-238 jet pumps (two jet pump pairs) are flow-tested over a wide range of hydraulic conditions, using the High Flow Hydraulic Facility 60° Sector Test Stand (Figure 2.3-1). The jet pumps are heavily instrumented with accelerometers, pressure transducers, strain gages, and displacement sensors so as to determine the dynamic characteristics and vibration response, as well as to characterize the forcing function. Tests are performed at temperatures ranging from 60°F to 180°F.

Work Performed This Report Period

All testing and analysis of jet pumps has been completed. Topical reports are in progress.

Task 2.3b - Jet Pump Hot FIV Test

Task Description

Flow tests are performed on single prototypical jet pumps under reactor temperature and pressure (550°F, 1050 psia) at the Pacific Gas and Electric Moss Landing Power Station, where the GE Steam Water Test Facility is located. The test program is designed to establish a better understanding of the relationship between the vibration response of the component and the key fluid and geometric parameters, particularly in the slip joint region. The program is separated into two activities, each described below.

(1) The first activity consists of the thorough FIV testing of a BWR/5 201-in. jet pump. This jet pump is of interest to the FIV for LWR Program because its geometric and fluid characteristics (lower fundamental frequency and higher slip joint pressure drop than other designs) make it more susceptible to slip joint-type vibration stabilities. Sufficient vibration instrumentation will be utilized to characterize the natural frequencies, mode shapes, primary forcing functions, and system response for various flow conditions.

(2) The second activity consists of the very limited FIV testing of BWR 218-in. and 251-in. jet pumps. The data, consisting primarily of vibration response as a function of flow condition, are acquired through the addition of several vibration sensors to jet pumps scheduled for hydraulic performance testing at the Moss Landing facility. The additional data will provide significant information concerning the effect of geometry (diffuser length, slip joint geometry, etc.) on the dynamic response of this LWR component.

Work Performed This Report Period

All flow-induced vibration testing of single prototypical jet pumps under reactor temperature and pressure in the Pacific Gas and Electric Moss Landing Steam Water Test Facility has been completed. Much of the vibration data has been reduced and analyzed. Topical reports documenting this work will be issued.

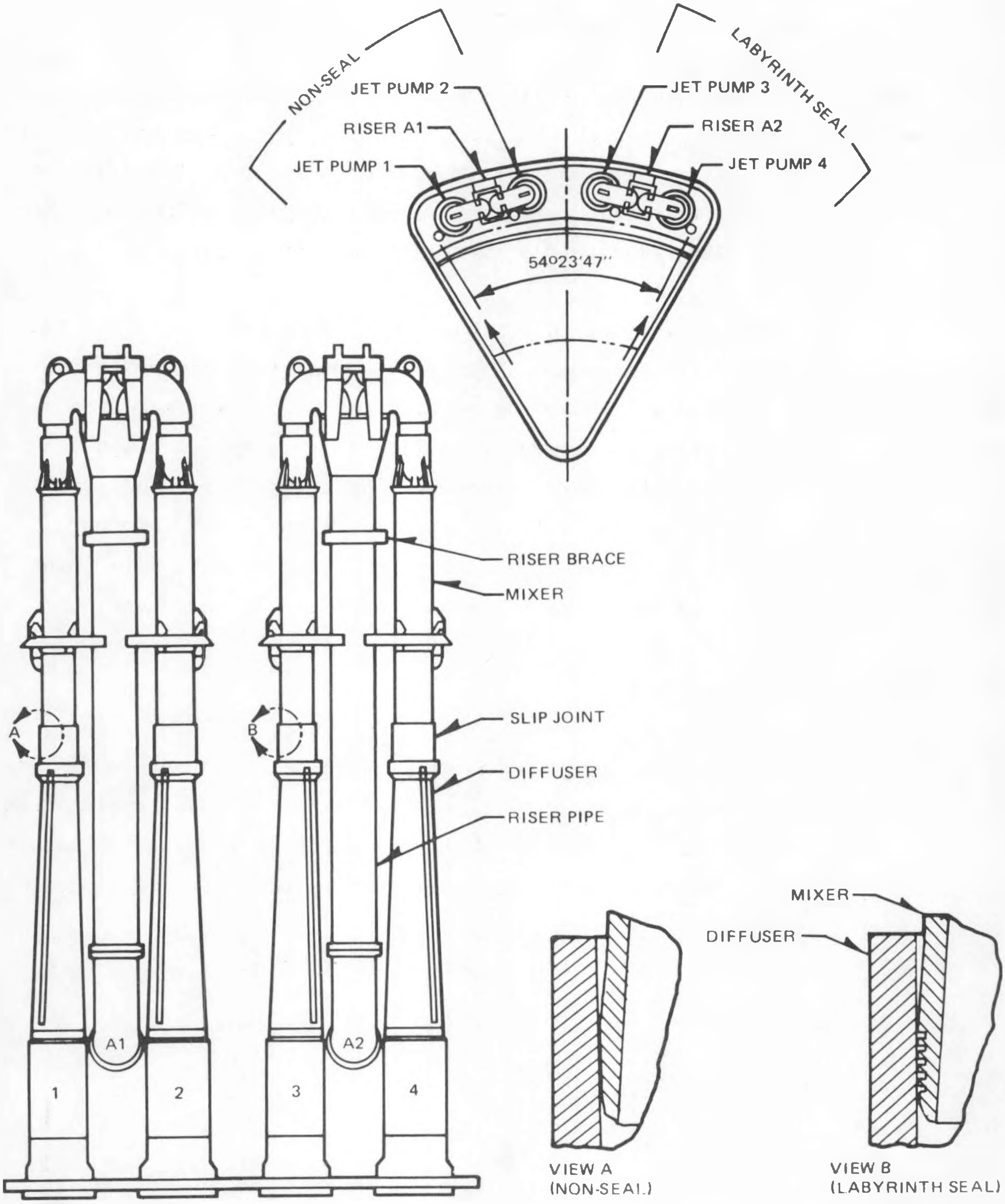


Figure 2.3-1. Jet Pump Pairs

TASK 2.4 - LOW PRESSURE COOLANT INJECTION COUPLING

The low pressure coolant injection (LPCI) system is part of the boiling water reactor emergency core cooling system. Not used during normal operation, its sole purpose is to restore and maintain the desired water level in the reactor vessel after a loss-of-coolant accident (LOCA). The overall system is shown in Figure 2.4-1; the internal coupling hardware is shown in Figure 2.4-2. The coupling hardware consists primarily of two 10-in.-ID elbows, joined together by a 26-in.-long sleeve. Collars which thread on the two elbows capture the sleeve during installation and hold it in place (to accommodate differential thermal expansion, the sleeve is not welded to the elbows). A nominal gap of 8 mils exists between the sleeve and each collar.

The LPCI is subject to FIV under both normal plant operation and emergency (LOCA) modes. Under normal operating conditions, when there is no flow through the LPCI, downcomer annulus flow buffets the LPCI coupling. In addition, a steady-state pressure drop across the collar-sleeve slip joints induces leakage flow which can result in fluid-elastic coupling. In the LOCA mode, the high flow velocities through the coupling produce turbulent buffeting of the inner walls. Also, as in the normal operating mode, slip joint differential pressures exist which can induce some leakage flow.

Task Description

Task 2.4 consists of flow-testing a prototypical BWR/6 238-in. LPCI coupling in the High Flow Hydraulic Facility's General Purpose Test Tank. The coupling is heavily instrumented with a combination of accelerometers, pressure transducers, and strain gages, as well as many thermal-hydraulic sensors to ascertain the vibration characteristics and leakage flow through the collar-sleeve slip joints as a function of the following parameters:

(1) Slip joint differential pressure,

(2) Internal (LOCA) flow,

(3) External (normal plant operation) flow,

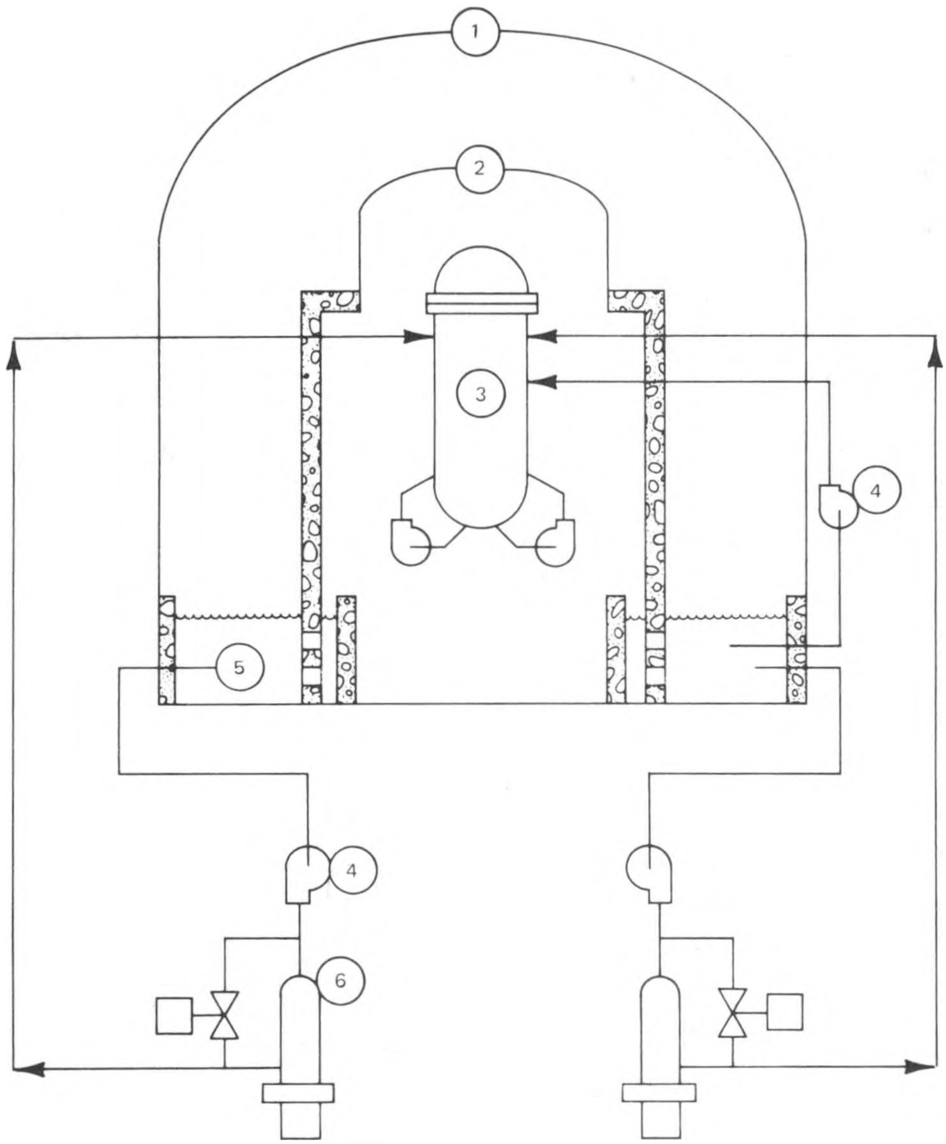
(4) LPCI coupling alignment, and

(5) Slip joints sealing (none, labyrinth seal, and piston ring).

This test program should provide some fundamental understanding of the flow-induced vibration characteristics of large piping configurations with slip joints.

Work Performed This Report Period

All test activities have been completed, and efforts are now directed toward data reduction and analysis, as well as final test documentation. Preliminary results from data analysis were reported in GEAP-24248. Data analysis is continuing and a final report is being prepared.



1 CONTAINMENT

2 DRYWELL

3 RPV

4 SYSTEM PUMP

5 SUPPRESSION POOL

6 HEAT EXCHANGERS

Figure 2.4-1. Residual Heat Removal System, Low Pressure Coolant Injection Function

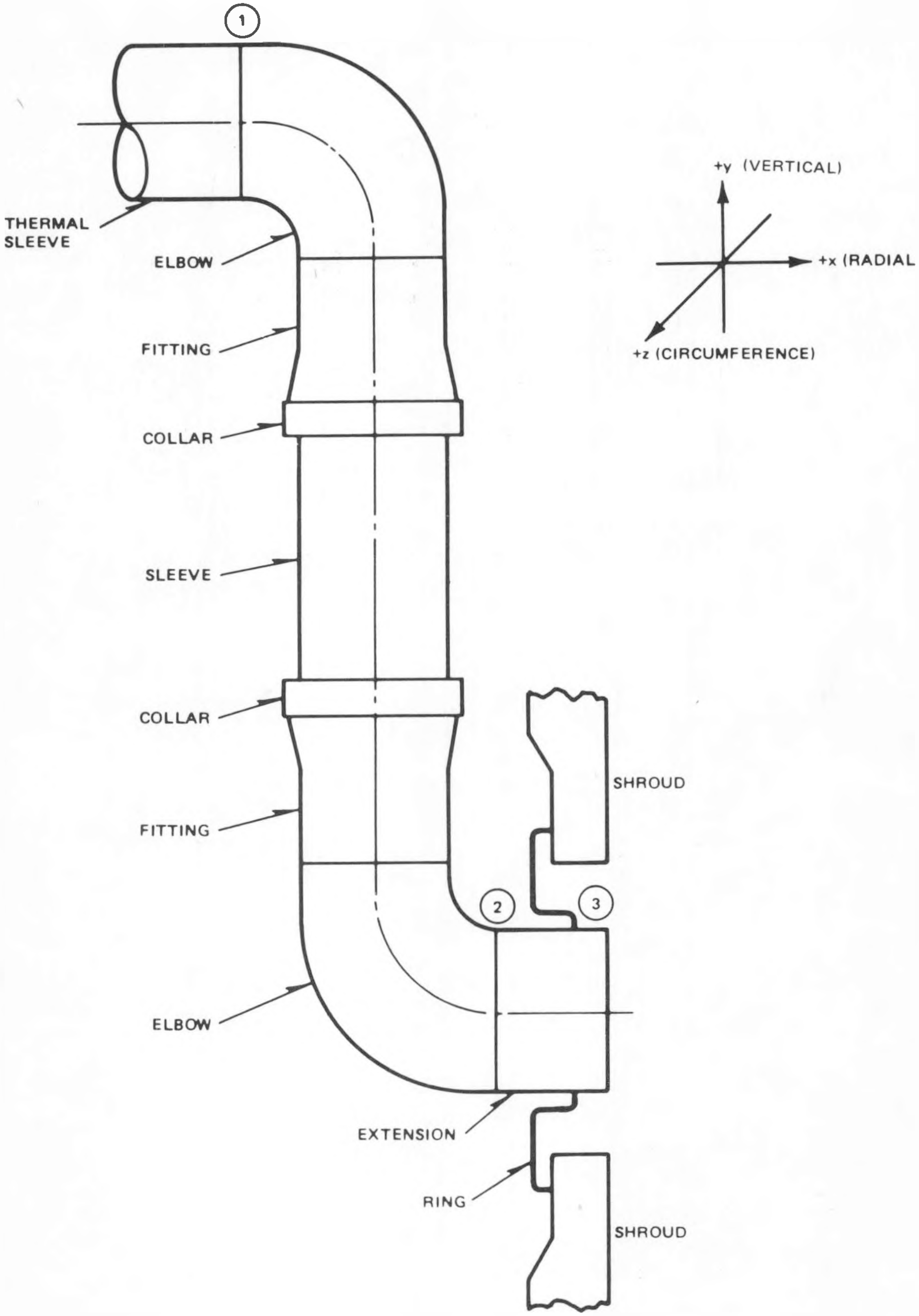


Figure 2.4-2. Low Pressure Coolant Injection System

TASK 3 - LWR FIV DESIGN METHODS, GUIDES AND CRITERIA

The primary purpose of Task 3 is the development of FIV guides and criteria which can be utilized by engineers for the design of LWR components. This is accomplished through the generation of analytical models and empirical relations, using the results of the fundamental studies, full-scale component tests, and actual plant vibration data generated in Tasks 1 and 2. Additionally, Task 3 is designed to establish scaling laws for FIV testing of reactor components, as well as to provide analytical support for the model testing of Task 2. The technical objectives are to:

(1) Develop analytical methods and computer codes for predicting the FIV response of reactor components;

(2) Develop empirical relations, scaling laws, and practical design guides for reactor design utilizing the information generated in Tasks 1 and 2; and

(3) Verify analytical methods by comparison with full-scale development tests and actual plant vibration data as available.

Task 3 is divided into three separate activities:

Task 3.1 - Predictive Model Development

Task 3.2 - Scaling

Task 3.3 - Correlation with Full-Scale Tests and Plant Data

Task 3.1 will develop the analytical methods and computer codes for predicting the FIV response of reactor components. Task 3.2 will indicate which parameters are critical (i.e., must be duplicated or accounted for) when performing scale-model testing of reactor components. Task 3.3 will verify the results of the predictive models and scaling laws, using vibration data from prototype LWR tests.

Progress achieved on these activities is described on the following pages.

TASK 3.1 - PREDICTIVE MODEL DEVELOPMENT (NED)

Task 3.1 is devoted to the generation and verification of analytical methods for predicting the FIV response of reactor component structures. The desired output from the task is one or more FIV design guides.

The basic approach used in Task 3.1 is to characterize the FIV problem into two categories, one to determine the amplitude of response, and the other to establish critical stability parameters and their range of influence. For response problems, the approach taken is to stochastically model the excitation and random response. Wherever possible, existing analytical tools, as well as special analytical techniques developed through the fundamental studies, are used for the generation of predictive FIV methodology. For the stability problem, reliance will be placed primarily on empirical results. Computer results are written to calculate the FIV response of structures, knowing structural and damping characteristics. Empirical formalisms and predictive models are verified against experimental results from Task 2. Stability curves are generated for structures to aid in safe-point operations.

Task 3.1a - Random Decrement Method (M. A. DeCoster)

Task Description

Damping is a very important parameter in evaluating the response of any vibrating structure. Unfortunately, it can be difficult to quantify, particularly for some of the complex reactor components under operational conditions. Therefore, methods are being sought to accurately measure the damping of structures in the actual reactor environment (i.e., after plant installation). The method presently under consideration is called random decrement. It is designed to identify modal parameters such as frequencies and damping for any component from a simple vibration response time history obtained during normal operations. This is accomplished by time-averaging specifically chosen blocks of data to form a randomdec signature, and then fitting one or more damped sinusoids to the signature via the modified difference Levenberg-Marquardt algorithm.

Work Performed This Report Period

This task has been completed.

Task 3.1b - Slip Joint Mechanistic Model (M. R. Torres)

Task Description

The internal flow velocities required to incur detrimental levels of flow-induced vibration in pipes are usually quite high for most practical applications. However, if submerged piping structures are coupled with annular slip joints to accommodate static load deflections or thermal growth, a practice utilized quite freely in industry, a geometry is created which is very susceptible to high-level FIV. This was demonstrated during the early stages of the FIV for LWR Program through small-scale experiments on a pair of submerged cantilever pipes joined at their free end with internal flow (documented in "FIV for LWR Progress Report, June-September 1977," Document No. COO/4175-2).

An in-depth investigation of existing technical literature has been performed which reveals extensive information exists on the dynamic stability of pipes with internal or external flow, as well as on the dynamic characteristics of submerged structures (for example, virtual mass, stiffness, frequency mode shape). However, sparse technical reporting on the subject of slip joint geometries is available, revealing that more fundamental-work is warranted, particularly as related to the development of analytical models to predict the response of reactor components with slip joints. Therefore, it is the objective of this task to generate a mathematical model which can be used to predict the level and onset of slip joint vibration as a function of piping structural parameters, details of the slip joint geometry, and hydraulic parameters.

Work Performed This Report Period

A mathematical model is under derivation. The approach taken is to solve the Navier-Stokes equation within the annular slip joint and determine the interaction force between the two cantilever pipes as a function of the motion of the pipes at the slip joint. The equations of motion of each cantilever pipe are coupled with this interaction (slip joint) force. Solution of these fluid-structure equations is underway.

TASK 4 - PROGRAM ADMINISTRATION

Task Description

The purpose of Task 4 is to provide the overall management of the program, including coordination, reports, and reviews. In addition, it is to ensure that the program's fundamental studies, model and full-size component tests, and analytical method development are responsive to the PWR as well as the BWR needs in the area of flow-induced vibrations.

Work Performed This Report Period

A progress meeting was held at a Department of Energy facility near Washington, D.C. on March 12. All contributors (Argonne National Laboratory, General Electric Corporate Research and Development, General Electric Nuclear Energy) were represented at the meeting. Technical progress presentations and discussion took place most of the day. A short discussion on the potential for a follow on flow-induced vibration program (beyond 1981) or program extension was proposed.

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