DAMPING OF REACTOR INTERNALS

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SUMMARY

During normal operation, reactor internals are excited by random and deterministic forcing functions. The complex fluid elastic system formed by the internals structures and reactor coolant responds in numerous modes and depends on the modal damping coefficients. This paper presents and discusses the results of a study of internals damping using data obtained from wave analysis of PWR plant flow tests, and shaker tests.

The damping values determined in these studies were obtained from vibration data taken during the pre-operational testing of several reactor plants and some in air shaker tests. Parameters available in the data include the presence of the core, the presence and position of the control rod drive line, reactor coolant temperature, and combination of reactor coolant pumps in operation. Modal damping values for the structures reported were obtained from the decay of autocorrellograms or from the modal response half-power bandwidths of frequency spectra generated from data covering the configurations and parameter variations.

For the lower frequency core barrel-reactor vessel beam modes 2% to 5% damping values were found for minimum damping values. Significantly higher values are found in the data when, for example, intermittent contact occurs at the core barrel lever supports.

Core barrel and thermal shield shell modes having natural frequencies in the frequency range of interest for seismic response calculations exhibit damping values generally on the order of 1% to 2%. Higher frequency, very low amplitude, shell modes of these structures can have damping values of less than 1%.

Damping values for guide tubes were found to have minimum values of 2% to 5% depending on their core location. The cross flow velocity and thus the floor turbulence excited amplitude is higher for guide tubes in core locations near the outlet nozzles. Information on the damping of upper support columns which are similarly excited is also given.

The paper discusses the application of the data to component design. Damping values reported are suitable for normal operation design conditions, i.e., for oscillatory behavior and relatively small amplitudes. The extrapolation of the data to obtain realistic values for large seismic events and for loss of coolant accidents is also discussed briefly considering that for these faulted conditions, the amplitude of the response is much larger than for normal operation.

The importance of this study lies in the fact that damping values are fundamental to the determination of structural response under oscillatory loads. The damping of most practical structures is the one dynamic characteristic that is not amendable to analytical determination. This difficulty is compounded by the complexity of the PWR internals structural configurations and by the fact that they are submerged in a flowing fluid causing important fluid-elastic effects. This paper contributes to the initiation of the compilation of experimental data needed to establish the damping of reactor internals for design and faulted condition purposes.

SECTION 1

INTRODUCTION

The importance of damping in the design of structures subjected to oscillatory loads has been well recognized in the technical literature.[1,2] The loss of energy during the vibration of a structure is generally assumed to have viscous characteristics, and the loss of energy is measured by a coefficient related to the loss of energy per cycle of oscillation. This coefficient is called percent of critical damping. It has been also recognized that damping, due to the nonlinearity of the structures and materials, generally increases with amplitude of vibration[1,3] with the known exception of Coulomb damping. Damping of a structure immersed in a fluid is the resultant of material (hysteretic) losses, structural (joints, slippage, impact, etc.) losses, and hydrodynamic (friction between solid and fluid, etc.) losses. Most of these losses have a non-linear characteristic with the materials losses being practically negligible for low stress levels. The damping value of a structure is usually established by test and presented as a percentage of modal critical damping, because different modes of vibration may have different damping and most of the methods of dynamic analysis use modal superposition techniques.

For the damping of nuclear components, in general, and the reactor internals, in particular, a knowledge of damping is needed not only to establish the response under normal steady state conditions, but also for transient accident loads that most of the time are oscillatory in nature. Reactor internals are designed to withstand seismic loads which are narrowband oscillatory loads with frequencies centered on the natural frequencies of the supporting structures and soil. Seismic response is heavily dependent on the damping of the structures which, in this case, will also include losses due to impact between adjacent components. The use of realistic damping values in the analysis will certainly provide a better representation of the reactor internals behavior during an earthquake. In addition, the internals are designed to survive the effects of a loss of coolant accident which will also excite the reactor components with short duration oscillatory loads; damping is also, in this case, an input of importance for the analysis. For these studies, as well as for the less severe but higher cycle normal operating conditions, knowledge of the damping is needed to evaluate the response under given forcing functions.

Experimental damping data for reactor internals are scarce in the literature. Work has been done to measure damping of reactor coolant loops[1,2,3] and a few other piping components.[4,5] These tests provide information for specific configurations and geometries. The US Nuclear Regulatory Commission published Safety Guide 1.61[6] where damping values are suggested for several types of structures. Recognizing the limitations and conservatism of this table, the Safety Guide allows the use of different damping values if suitable data is provided.

PWR reactor vessels, internals, and core configurations are basically as described in reference 8. Structures, such as the reactor internals, that are immersed in a confined flowing fluid are not typical of the structures for which the damping values of the Safety Guide were established and certainly require experimental results for their justification.

Experimental data obtained from full size and model tests[7,8] show that during normal operation the internals are excited by flow turbulence and by excitations produced by the reactor coolant pumps. The narrowband turbulence excited modal responses of reactor internals components measured for several plants and for various test conditions and configurations were used to determine modal damping values. Recognizing that all of the data are for very small vibration levels, these damping values are reported and discussed in this paper for use in the design of reactor internals.

1-1. METHODS OF MEASURING DAMPING OF REACTOR INTERNALS

The reactor vessel-internals-core system responds to broadband flow turbulence in beam and shell modes. Regarding the component responses for which damping has been investigated, the following modes of response have been observed: guide tubes and support columns, beam modes; core barrel, shell and beam modes; reactor vessel, rocking modes. If the modes are adequately separated, as shown on figure 1, damping can be estimated from frequency spectra or autocorrelograms. Multiple mode responses can be analyzed by using filters to isolate single mode responses. The initial slope of the autocorrelation has been found to provide the most accurate value for damping of a system.[9] Comparing results of analyses with known structural properties, it was concluded that the first cycle of the autocorrelation gave the most accurate results and that use of successive cycles resulted in estimates of lower damping.[10] It was also concluded from these studies that for large ratios of lag time to sample size the autocorrelation changes from a monotonic decay to a beam form. Kitaigorodskii[11] used a computer method whereby vibrograms are successively bandpass-filtered around the center frequencies of interest, and the resulting spectral density is used to obtain a damping coefficient by a least squares fit.

For the data presented here, two techniques were used to determine damping from the response data. In the bandwidth method, the damping was found by measuring the half-power bandwidth of the modal response from frequency spectra and using the relationship:

$$\frac{C}{C_c} = \frac{\Delta f}{2 f_c}$$

where

C = damping coefficient

C_c = coefficient critical damping

Δf = bandwidth at the half-power points

f_c = center frequency of the modal response

Autocorrelograms generated from signals filtered around the frequency band of interest were also used to determine modal damping. Figure 2 is an autocorrelogram obtained from the signal from which the spectrum of figure 1 was generated.

Although both techniques yield similar damping values, one or the other may have an advantage for a particular case. For example, using the bandwidth method on responses having low damping requires that very narrow bandwidths must be used in the frequency analysis so that the response bandwidth is larger than the analyzer bandwidth. Autocorrelations by-pass this difficulty and also identify any deterministic signal content. If spectral peaks occur at frequencies that are close the center frequency of a response for which an autocorrelation is to be generated, filtering is less effective, and the envelope of the resulting autocorrelation has beats. In some cases, the filter bandwidth can be narrowed so that the effect of the nearby peak is small, and the influence of the beat on the autocorrelation can be removed from the resulting plot.

SECTION 2

DAMPING VALUES MEASURED

2-1. GUIDE TUBES

Guide tubes provide a low friction path for the control rods during movement into and out of the fuel core and shield the rod cluster control rodlets and drive shaft from cross flows, when fully withdrawn or passing through the core. They are supported at the upper end by a flange bolted to the upper support plate and at the lower end by pins which are spring loaded against the upper core plate. The guide tubes for which damping values are given in this paper are 120 to 150 inches long and have a cross section approximately 7.5 inches square, with truncated corners. The flow field surrounding the guide tubes contains both axial and cross flow velocity components.

Table 1 shows guide tube first beam mode damping values obtained from data recorded during pre-operational testing of four PWRs. All data are for all pumps in operation, with coolant temperatures varying between 135°F and 560°F, with and without the core and control rod drive lines installed (initial start-up and hot functional tests, respectively). The data show that the damping values for the guide tubes are 3.6 to 6.0 percent and are essentially independent of coolant temperature, the presence of absence of the core, and the presence or position of the control rod drive line. Slightly lower damping values have been measured for guide tubes without cross flows. The damping losses include structural losses and losses that are related to parallel flow velocities and cross flow velocities. Results of shaker tests in air on guide tubes in the assembled upper package indicate that the structural losses are smaller (less than 1 percent for similar amplitudes) than the losses due to immersion in flowing water. Guide tubes that are in low cross flow areas tend to have lower damping as indicated by the values on table 1. Data for various operating pump combinations show that damping decreases with the number of operating pumps (that is, with decreasing axial flow velocity) when similar cross flow velocity cases are considered. For the second beam mode of the guide tube, vibration amplitudes are an order of magnitude lower than the first mode, and damping varies similarly with reactor conditions. Damping is on the order of 2 percent.

2-2. UPPER SUPPORT COLUMNS

The upper support columns are in the same flow field as the guide tubes. They position the upper core plate relative to the upper support plate and transfer axial fuel assembly loads to the upper support plate. The columns for which damping values are reported here have 7-1/2 inch outside diameters and one-half inch thick slotted walls. The ends are bolted or socketed and stayed to the upper support plate and upper core plate. They are much stiffer than the guide tubes and vibrate at much lower amplitudes during reactor operation. For vibration amplitudes on the order of 0.0001 inch rms, damping values are on the order of 1 percent. Because of the extremely small amplitudes of vibration, no further damping data on these components are presented.

2-3. CORE BARREL

Damping values for the core barrel were computed from the signals of strain gages mounted near the core barrel flange. Figure 3 shows core barrel beam mode damping values for two plants computed from autocorrelograms and plotted against the number of reactor coolant pumps in operation. A zero pump damping is also shown. This value was obtained from the bandwidth of the response to ambient vibration shown in figure 4.

The IPP-2 data show that the damping increases as the number of pumps in operation increases from 2.5 percent with no pumps operating, to 8.8 to 12 percent with four pumps in operation.

Possible causes for this increase are threefold:

- Increased hydrodynamic damping due to the higher downcomer flow velocity. This is not considered to be a strong effect since scale models tested with no radial supports do not show strong changes in damping with flow rate.
- Light intermittant contact may be occurring between the core barrel and the reactor vessel at the core barrel lower radial supports. As the number of pumps in operation increases, impact damping is likely to increase, possibly from zero for some combinations without impact.
- The interface losses at the flange may increase with increases in the vibration level.

In view of these variables and the scatter in the data, no change in damping level can be attributed to the addition of the core. The strain levels corresponding to these damping values are less than 3×10^{-6} in./in. rms, and the core barrel vibration is less than 0.002 inches rms.

Damping for the barrel and thermal shield shell modes have also been investigated for the amplitudes of vibration that occur during normal operation. Values between 0.3 percent and 2.0 percent have been observed for the various shell modes. The higher damping values are associated with the higher amplitude, lower frequency modes (maximum displacements of 0.002 inches rms).

Core barrel damping values in air have been measured from scale model and prototype tests. These data show significantly lower damping in air than corresponding damping values in water.

2-4. REACTOR VESSEL

The non-axisymmetric support of the reactor vessel, the participation of the core barrel in reactor vessel-system modes, and the various possible as-assembled core barrel-reactor vessel clearances combine to cause the presence of several closely spaced frequency responses during steady state operation. The center frequencies of the most predominant responses with corresponding amplitude and damping values are shown in table 2. The data tabulated are for rocking modes in two different directions measured by accelerometers mounted on the bottom of the reactor vessel. Because of the multiplicity of responses seen in the spectra, it is difficult to insure the accuracy of the damping values. It has not been established in these studies that all of the responses are due to broadband excitation. In fact, narrowband pressure oscillations in the vicinity of these vessel frequencies have been observed in the data.

Damping values of 1.4 to 10 percent are indicated for the very small displacement levels to which they correspond. For one of the plants analyzed, it is noted that the damping with the core installed is larger than the damping measured during hot functional testing. In the other plant, this trend is not evident.

TABLE 1

GUIDE TUBE BEAM MODE DAMPING
(All Pumps Operating)

Plant	Test	Cross Flow	Temp °F	Direction	Vibr. Ampl. (RMS in. 10^{-3})	f_c (Hz)	Damping % C/C_c
Trojan 1	Hot Functional Test	Yes	541	0-180	0.54	30.9	4.2
				90-270	1.15	26.8	4.0
	(No core or drivelines)		432	0-180	0.57	30.8	4.6
				90-270	1.16	26.6	4.3
			245	0-180	0.60	30.8	4.5
				90-270	1.16	26.2	4.2
			135	0-180	0.53	30.6	4.9
				90-270	1.20	26.1	5.2
	Initial Start-up Test		558	0-180	0.38	31.1	4.1
	(Core in Place, Control			90-270	0.79	26.6	4.7
	Rod Fully Inserted)		145	0-180	0.36	30.8	4.6
				90-270	0.73	26.2	4.4
	(Control Rod Partially		554	0-180	0.40	31.2	4.8
	Withdrawn)			90-270	0.84	27.2	4.4
	(Control Rod Fully		552	0-180	0.36	30.8	4.6
	Withdrawn)			90-270	0.73	27.2	4.4
H. B. Robinson	Hot Functional Test	No	250	0-180	0.88	26.4	3.6
				90-270	0.93	27.0	3.7
		Yes	250	0-180	2.42	26.0	6.0
		No	545	0-180°	0.77	27.9	3.6
				90-270°	1.12	27.3	3.3
		Yes	545	0-180	2.69	26.7	5.4
R. E. Ginna	Hot Functional Test	No	560	90-270	0.56	29.1	3.7
		Yes	560	0-180	0.81	29.1	5.9
				90-270	0.64	31.9	5.2
Indian Point Unit 2	Hot Functional Test	No	530	90-270	1.7	19.7	3.8
		Yes	530	0-180	2.1	21.3	4.1
				90-270	3.3		4.1
	Initial Start-up Test	Yes	540	0-180	1.8	22.1	3.8
	(Control Rods Fully			90-270	3.0	20.0	3.8
	Inserted)			0-180	1.8	21.3	5.4
				90-270	3.0	19.5	4.2
		Yes	260	0-180	1.9	21.8	4.7
				90-270	3.3	19.5	3.6
		No	540	90-270	1.5	19.7	4.1

TABLE 2
DAMPING OF REACTOR VESSEL ROCKING MODES

Plant	Test	Pumps	Temp (°F)	f_c (Hz)	Damping $\%C/C_c$	Displacement (RMS in x 10^{-3})
Trojan 1	Initial Startup Test	1234	145	11.4	1.8	0.35
				12.6	2.5	0.20
		1234	327	7.9	7.9	1.17
				10.9	1.8	0.21
				12.8	2.3	0.18
		1234	558	8.8	2.6	0.24
				14.0	10.0	1.05
		12-4	555	8.8	2.1	0.31
				15.5	2.4	0.06
		123-	556	8.8	2.2	0.28
				14.6	1.9	0.10
	Hot Functional Test	-234	541	7.0	1.8	0.45
				8.3	3.3	0.16
		1234	524	11.6	2.8	0.40
				12.5	6.0	0.28
Indian Point Unit 2	Initial Startup Test	1234	534	14.2	3.9	0.13
				14.0	5.4	0.12
		12-4	546	16.0	3.7	0.16
				13.9	3.2	0.01
	Hot Functional Test	1234	530	16.3	1.7	0.23
				17.7	1.4	0.14
		12-4	529	16.3	2.2	0.14
				17.7	3.1	0.20

SECTION 3

CONCLUSIONS

Data obtained during pre-operational testing of reactor internals show that the losses due to the fluid-solid interaction are a major contributor to the damping of the major internals vibration modes. Because of the limited range of the vibration amplitudes that existed during the tests, no extrapolations have been made for damping values at larger displacement magnitudes. Furthermore, the data reported represents the most reliable measurements; other available measurements, with less well defined modal response, indicate higher damping values.

Internals damping levels with the core in place were found to be the same as damping levels without the core. It was also found that damping is unchanged over the temperature range of the measurements and that the presence or position of the control rod drive lines does not affect guide tube damping values.

The barrel beam mode which is excited during seismic (OBE & SEE) and LOCA events shows values between 4 percent and 12 percent for the small oscillations measured during these tests which suggests that damping of at least these levels can be used for analysis of these transients for the corresponding mode.

Vessel rocking modes have not been determined with the same precision as the internals, and damping values between 1.4 and 10.0 percent have been found.

It is recognized that this study covers a limited number of plants and internal components, and it is expected that more general damping correlations will be possible when a larger catalog of data is available.

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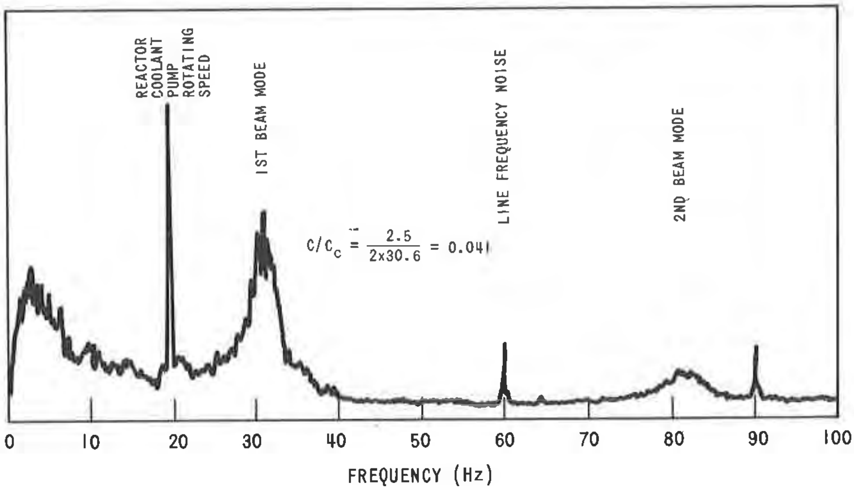

Figure 1. 0-100H_z Frequency Spectrum of a Guide Tube Response During Initial Startup Testing

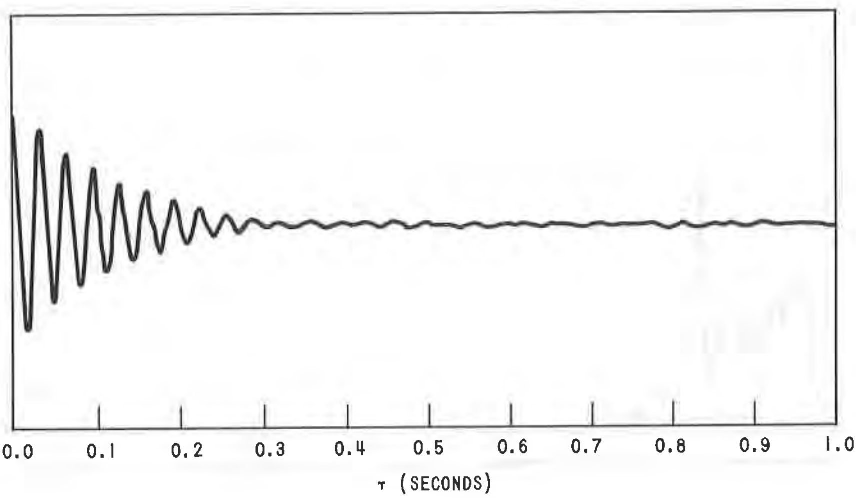

Figure 2. Autocorrelogram of the First Beam Mode Response of a Guide Tube

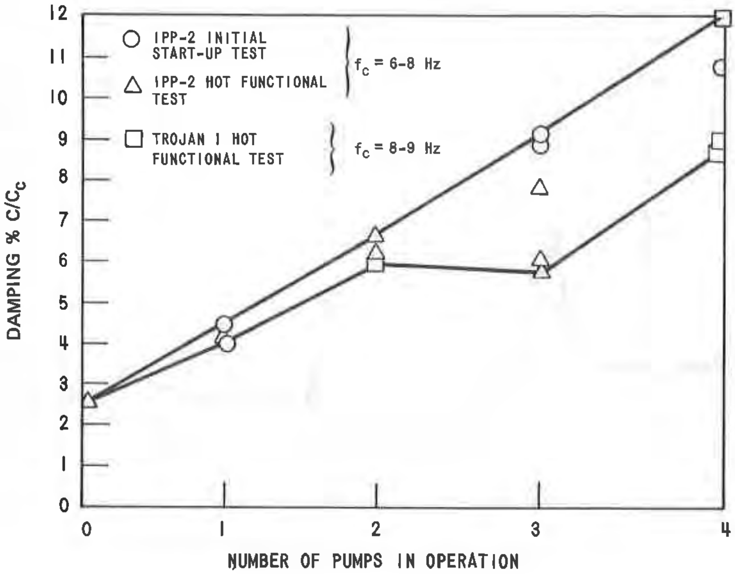

Figure 3. Damping of Core Barrel Beam Mode

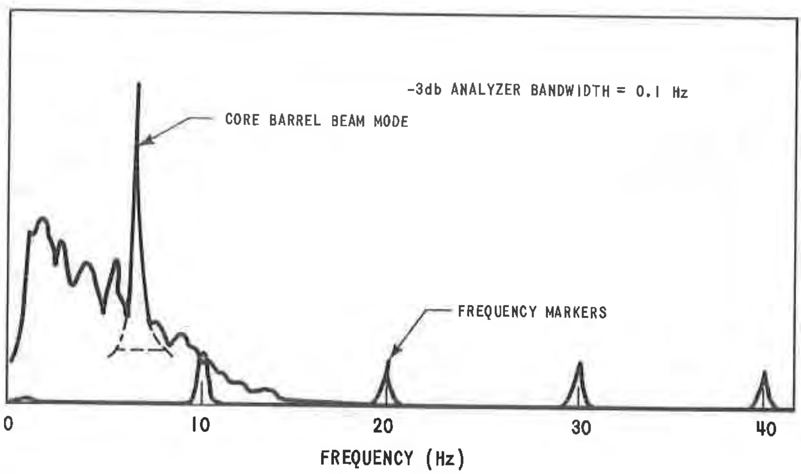

Figure 4. Frequency Spectrum of Core Barrel Flange Strain Gage at 520°F, Hot Functional Test