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FIELD VIBRATION TEST OF PWR NUCLEAR POWER PLANT REACTOR COOLANT LOOP

Muneaki KATO[1], Rokuro ENDO[2]
Katsuhisa FUJITA[3] and Masaaki TABUCHI[4]

[1,2]Construction Department, Japan Atomic Power Company,
Chiyoda-ku, Tokyo, Japan
[3]Vibration and Noise Control Laboratory, Mitsubishi Heavy
Industries, Takasago City, Hyogo, Japan
[4]Reactor Components Engineering Department, Mitsubishi
Atomic Power Industries, Minato-ku, Tokyo, Japan

SUMMARY

Presented in this paper is a method for identifying the critical damping value of Reactor Coolant Loop(RCL) system excluding the hydraulic snubber effect in actual PWR nuclear power plant due to large exciting force. In Japan, whole damping value regarding to the specified system had been already investigated through previous large exciting test, so in this work it has been expected to clarify the system damping value without hydraulic snubber effect because the snubber number and capacity shall be variable depending on the site-seismic condition. This test therefore was planned, conducted and brought the fruitful results.

INTRODUCTION

In seismic design of nuclear power piping system, the conservative damping values had been conventionally used as a design criteria in Japan, however it was sometimes pointed out that they might have too margin. To re-evaluate the damping values of piping systems, the systematic study had been conducted and it was concluded that new realistic values can be applied for the piping seismic design.(Ref. 1) On the other hand, 1% of critical damping ratio had been used for the RCL system seismic design because we have no enough experimental test data to prove the realistic value with large response. On Japanese PWR nuclear power plants, many data concerning to the RCL were collected as the result of those pre-operational tests on each site to confirm the vibrational characteristics. However the amplitude levels of deflection in those tests were very low compared with design condition. But in this stage it was important to investigate the realistic damping value of the RCL as well as auxiliary piping system to provide the rational seismic design. In these circumstances, the large force exciting test was conducted on the actual plant for estimation of reasonable design basis structural damping value of the RCL, excluding additional hydraulic snubber damping effect.

TEST PROGRAM

<u>Test article</u> The test was conducted on Tsuruga No. 2 unit owned by the Japan Atomic Power Company, 116 MWe capacity Japanese standardized 4 loop PWR nuclear power station, in 1985. The outline of the RCL and arrangement of typical Reactor Coolant Loop are shown in Figs. 1 and 2 respectively, the support

structures for the steam generator and the reactor coolant pump are appeared in Figs. 3 and 4. In Japanese general design, 3 horizontal and 2 horizontal level supports are provided for steam generator and reactor coolant pump respectively because of the high seismic condition. The upper and the intermediate horizontal supports for the steam generator are composed of ring type structure just winded around the component shell itself and several hydraulic snubbers. Lower support is consisted of steel frame work and acts as guide structure for steam generator thermal movement and/or seismic response. And pin-ended 4 vertical columns are provided at the bottom of steam generator to withstand the component dead weight. The reactor coolant pump supports have also just same concept as steam generator.

Test procedure All hydraulic snubbers provided on steam generator and reactor coolant pump supports for exciting loop were locked up by extra devices such as rigid and no damping supports and 5 tons capacity hydraulic exciter was mounted directly at the enough higher location of steam generator. Fig. 5 shows the hydraulic exciter arrangement and accelerometers provided on the RCL system are shown in Fig. 6. The test article i.e., reactor vessel, steam generator, reactor coolant pump and pipes were filled with room temperature-water up to plant normal operational water level but no pressurization. 2 to 25 Hz sine wave sweep tests were conducted and three levels of dynamic loads (1000 kgf, 3000 kgf and 4500 kgf) were applied for each test in order to survey the effect of vibrational characteristics depended on response levels. Critical damping values were derived from response spectra (i.e., half power method) and free vibration curves (i.e., natural logarithm produced method).

TEST RESULTS

Natural frequency Natural frequencies depended on exciting forces are shown in Table 1 and that values as a function of steam generator top deflection are also shown in Fig. 7.

Damping value Damping values depended on exciting forces and/or the deflection at the top of the steam generator are shown in Table 2 and Fig. 8.

DISCUSSION

Total damping value for the whole RCL system is depended on hydraulic snubber effect and other mechanical one. Here, this task was to define the RCL damping value without snubber influence as described above. However it is considered that small energy dissipation might be occurred at the snubber components because of remained small mechanical gaps. Therefore it is requested that snubber effect should be subtracted from appeared total system damping value. Fig. 9 shows one of the typical resurgent wave for the locked-up snubber and dissipation energy as a function of snubber load is shown in Fig. 10. The damping values of steam generator depended on the locked-up snubber effect are obtained by following equation.

$$hi = \frac{1}{4\pi Ei} \Sigma \Delta Eij \qquad (1)$$

Where, hi: damping ratio for i-th mode

Ei: total strain energy for i-th mode

$Ei = (1/2)KX^2$

K: stiffness matrix of the system

X: displacement vector

ΔEij: dissipation energy of the locked-up snubber component j for the i-th mode

Calculated damping values depended on the locked-up snubber influence are shown in Table 3. Subtracting the Table 3 values from Table 2 damping ratios, the RCL damping values without snubber effect shall be obtained. The results of main purpose through this task are summarized in Fig. 11.

CONCLUDING REMARKS

Estimated damping values without snubber influence are 3 to 5% of critical. The tests demonstrated the slightly increasing of damping value to respond to increasing the excitation levels, i.e., increasing of steam generator deflections. Through this effort and coupled with other domestic PWR nuclear power plant test data, it is concluded that the design basis damping value of Japanese standardized PWR RCL system is reasonable to be 3% at the minimum. And these data and/or conclusions had been used to support the amendment of Japanese seismic design guideline.(Ref. 5)

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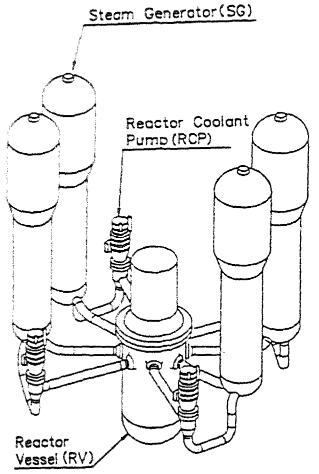

Fig. 1 Outline of Reactor Coolant Loop System

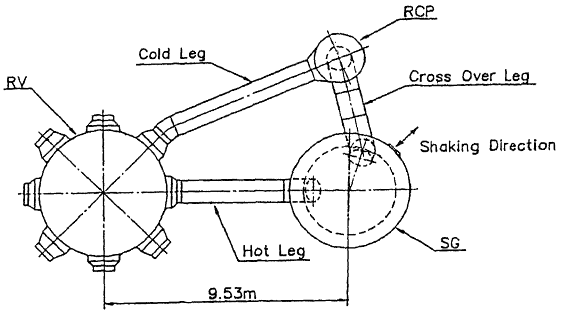

Fig. 2 Arrangement of Reactor Coolant Loop

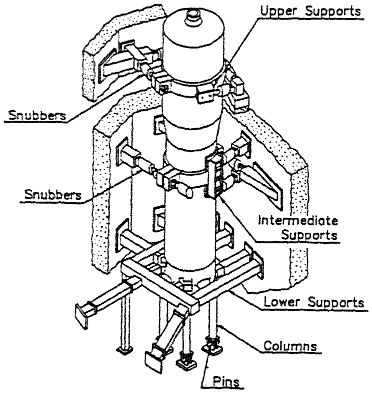

Fig. 3 Support Structures of Steam Generator

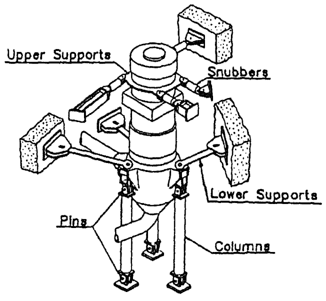

Fig. 4 Support Structures of Reactor Coolant Pump

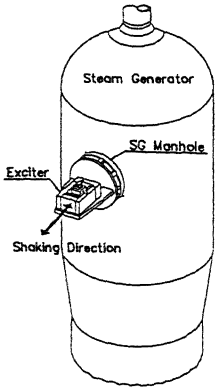

Fig. 5 Hydraulic Exciter Arrangement

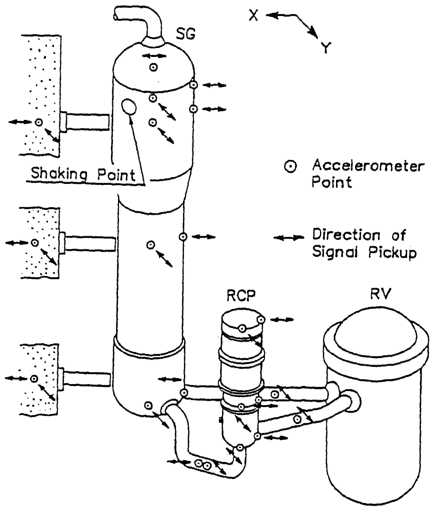

Fig. 6 Accelerometer Arrangement

Table 1 Natural Frequency (X-direction)

exciting force	1000 kgf	3000 kgf	4500 kgf
natural frequency (Hz)	16.3	16.0	16.0

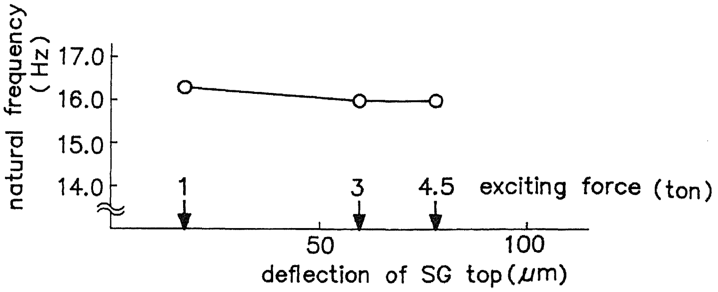

Fig. 7 Natural Frequency (X-direction)

Table 2 Damping Value

exciting force (kgf)	direction	SG top deflection (μm)	damping value (%)		
			half power	natural log.	polar diagram
1000	X	17.4	4.5	4.1	4.8
	Y	27.5	4.4	5.3	4.4
3000	X	60.0	4.3	5.1	4.2
	Y	90.0	5.3	5.3	4.3
4500	X	78.5	5.8	5.4	4.7
	Y	144.0	4.4	5.4	4.2

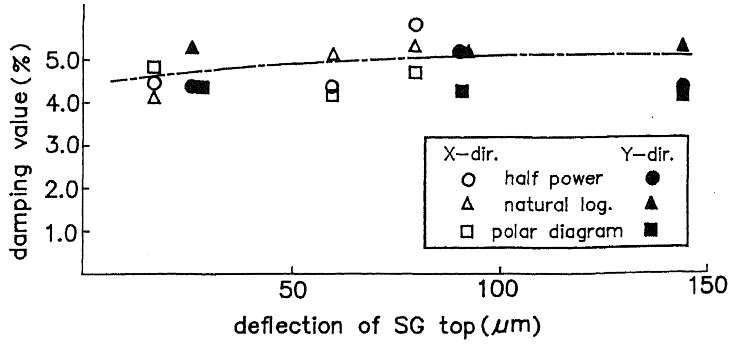

Fig. 8 Damping Value

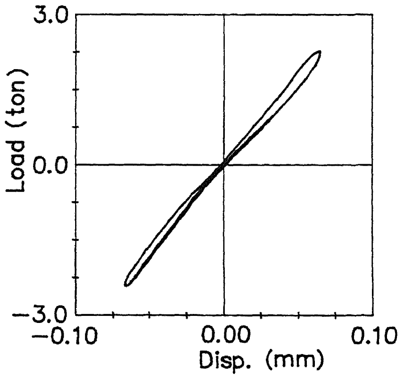

Fig. 9 Typical Resurgent wave

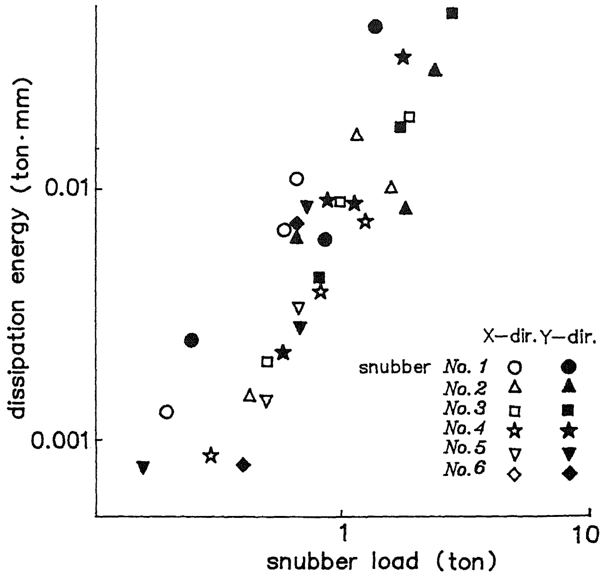

Fig. 10 Dissipation Energy

Table 3 Damping Value depended on Locked up Snubber

exciting force (kgf)	direction	SG top deflection (μm)	dissipation energy of locked up snubber: $\Sigma\Delta E$ (ton·mm)	total strain energy of RCL system: $4\pi E$ (ton·mm)	damping value of snubber element: h_s (%)
1000	X	17.4	0.505×10^{-2}	0.412	1.2
	Y	27.5	0.930×10^{-2}	1.277	0.7
3000	X	60.0	0.328×10^{-1}	4.900	0.7
	Y	90.0	0.342×10^{-1}	13.680	0.3
4500	X	78.5	0.464×10^{-1}	8.387	0.6
	Y	144.0	1.075×10^{-1}	35.023	0.3

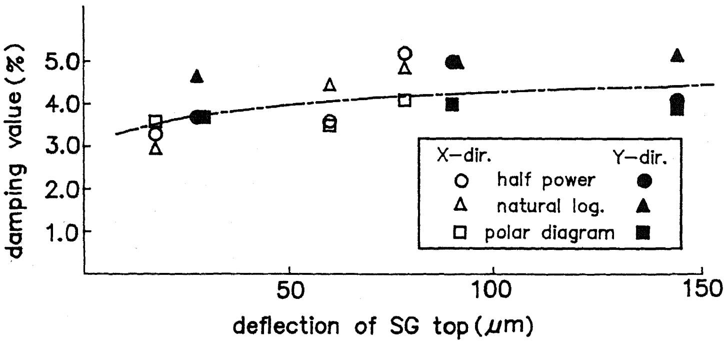

Fig. 11 Damping Values without Snubber Effect