Proving Test on the Seismic Reliability of the PWR Primary Coolant Loop System

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ABSTRACT

A series of Proving Tests on the Seismic Reliability of nuclear power plants has been carried out by the Nuclear Power Engineering Center (NUPEC), using a large-scale, high-performance vibration table at Tadotsu Engineering Laboratory, sponsored by the Ministry of International Trade and Industry (MITI) of Japan. The Seismic Proving Test of the PWR Primary Coolant Loop System was conducted as the sixth program, using a 1/2.5 scale model of an actual plant in 1988. Some of the test results of this program were reported (K. Fujita et al., 1989; H. Akiyama et al., 1990).

This paper presents a summary of the results of the above test program as well as the seismic evaluation of the 1,100 MWe class PWR standard plant in Japan.

1. INTRODUCTION

The purpose of the test is to prove seismic safety and reliability of the PWR Reactor Coolat System (RCS), and following items are to be proved.

i) To confirm the seismic reliability of the pressure boundary of the reactor coolant system and the seismic support structures.
ii) To confirm the vibrational characteristics of the system and the propriety of the seismic analysis.

In order to achieve these objectives, the test model, being as similar as possible to the actual plant configuration, material, scale and so on, is tested under design earthquake conditions. The seismic safety and reliability of the model are directly confirmed by the test, and the adequacy of the design analysis method is also confirmed by the analysis of the test data. Finally the overall evaluation for the actual plant is carried out, considering the differences between the test model and actual plant condition.

Fig. 1 shows the overall evaluation flow diagram of this proving test program.

2. TEST MODEL

The essential concept for modeling is as follows;

- The model was designed to simulate the 1,100 MWe class PWR 4-Loop standard plant.
- The model is 1/2.5 scale of one typical loop of an actual Reactor Coolant System, that is composed of a Steam Generator (SG), Reactor Coolant Pump

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(RCP), and Reactor Coolant Piping (Hot Leg, Cold Leg, and Cross-over Leg).

- Compensational mass is installed on the model so that the natural frequency is reduced, in the ratio of $(0.7)^{-1}$ to that of the actual plant, to excite the model within the frequency limit of the vibration table.
- The scale and mass ratio between the test model and the actual plant is based on the modified law of similitude, so that the same stress is produced at the test.

In the modeling, the pressure boundaries of the component models (SG, RCP) and the piping are made of the same material and welding procedures as the actual plant. The system is filled with water at the pressure of 15.5 MPa at room temperature. The inner structures of the components are modeled with the compensational weights to adjust the vibrational characteristics of the system. The overall test model is composed of the scaled-down model of the PWR RCS loop and the System Support Frame to support it on the vibration table as shown in Fig. 2.

3. EXCITATION

3.1 Input seismic wave

The basic design earthquake ground motions S_1 and S_2, which had been improved and standardized by MITI for the high seismic zones, were used as the input to the reactor building analysis of the standard PWR plant to obtain the floor response waves at the operation floor and so on. Among various kinds of response waves, the waves giving the severest effects on the primary coolant loop system were adopted as the input waves of the proving test. The conditions of the input waves are shown in Table 1. Most of the tests were conducted under the two directional excitation, both horizontal and vertical, simultaneously.

3.2 Test procedures and major results

In order to achieve the objectives of the test program, the following four kinds of tests were carried out corresponding to the evaluation flow of Fig. 1. The outline of the test and some of the results have been reported (K. Fujita et al. 1989).

(1) Preliminary test

The vibrational characteristics (resonance frequency and damping ratio) and seismic responses of the test model were roughly examined by preliminary excitations with low level sinusoidal waves and low level seismic response waves.

(2) Strength proving test

The aseismic strength of the primary coolant loop system was verified by seismic wave excitation with the conditions given in Table 1. The maximum support loads and the maximum stresses of piping are shown in Table 2, compared with the analysis values (predictive analysis and design analysis) for the test model.

The predictive analysis had been performed before the test, adopting the time history modal analysis method, while the design analysis had used the response spectrum method with ±10% spectrum broadened. It was found that the predictive analysis shows good accordance with the test results and the design analysis gives conservative results.

(3) Design method confirmation test

Data to evaluate the adequacy of the seismic response analysis method of the actual plant design were obtained, by performing the sinusoidal wave sweep

vibration test and the seismic response wave vibration test.
Table 3 shows the natural frequencies obtained by the sinusoidal wave test, compared with the design analysis values.
The seismic response wave test was performed by the excitation of one direction in horizontal, or vertical, respectively, using the S_2 (2) wave of Table 1. Fig. 3 shows the comparison of accelerations of the test results and predictive analysis results.

(4) Excessive vibration test

Two types of the test were carried out for the purpose of confirming the design margin, by the excitations up to response level beyond design. One is increased S_2 wave test, and the other is resonant S_2 wave test.
An increased S_2 wave test was performed with an excitation of 2 times horizontally and 1.5 times vertically greater than the original S_2 (2) wave. The integrity of the test model was confirmed by the post-test inspection.
In the resonant S_2 wave test, a modified S_2 (2) wave was used which has the same resonant frequency as the natural frequency of the steam generator in order to load the snubbers up to double design loads. The test model including snubbers had no damage during the test.

4. SIMULATION ANALYSIS

Simulation analyses of the test results were performed to analytically follow the vibrational response phenomena of the proving test. Time history modal analysis, where the local nonlinear rigidity has been treated as psudo external forces, was adopted. The details of the method were already described in the reference (K. Fujita et al., 1989).

The following approach was introduced in the simulation analyses.

- The resonant frequency of the model excited by the earthquake was computed by analyzing the test data using the AR mehtod.
- Since the support rigidity (particularly snubber rigidity) depends on load level, the relation between the eigenvalue and support rigidity was estimated by the stochastic finite element method.
- Then the support rigidity was estimated with the condition that the calculated values for the loads and the accelerations had to match the test results.
- The one-sided rigidity supports, such as the bumpers of the steam generator intermediate support, were treated as nonlinear springs.
- The modal damping ratio was evaluated considering both effects, one from the dissipated energies of the snubbers and the other from the structural damping.
- The pitching and rolling motions of the vibration table were considered as the components of rotational excitation.

Fairly good accordance in the responses (displacement, acceleration, and support load) between the test results and the simulation were obtained.

5. SEISMIC EVALUATION OF THE RCS SYSTEM FOR A PWR 4-LOOP STANDARD PLANT

Seismic safety of the actual plant primary coolant loop system was evaluated by stress evaluation combining loads other than earthquake, as well as by seismic response analysis for the actual plant conditions.
Seismic response analysis of the actual plant, say "verification analysis" was performed to extend the simulation analysis for the test model to the actual plant dimensions. That is, the verification analysis for the PWR 4-loop standard plant took into account the structural differences between the model and the plant, to enforce the compensation by the analysis for items such as the use of a coupled model with the reactor building, and the use of the estimated modal damping ratios (SG-X 5.5%, SG-Y 4.5%, RCP 4%) for the actual plant. The estimation of the modal damping for a PWR RCS was reported in the reference (H.Akiyama et al., 1990).

The verification of the seismic design was conducted by comparing the seismic responses of the two kinds of design analysis with that of the verification analysis. Table 4 shows a comparison of conditions between the verification analysis and the design analysis methods which were applied to the actual plant evaluation.

Table 5 and 6 show response analysis results (maximum acceleration and support load) for the S_2 (2) seismic wave which is most severe in the actual plant evaluation. Comparison of the results of design analysis methods (A), (B), and of the verification analysis confirms the following.

° Both response acceleration and support load values found by the design analysis methods (A), (B) generally showed larger values than those by the verification analysis, therefore the design analysis methods have an adequate safety margin.
[(Verification analysis) < (design analysis (B)) < (design analysis (A))]

Concerning the reactor coolant piping, both stress and fatigue evaluations were carried out, for the combination of thermal expansion load, thermal transient load, dead weight and internal pressure in addition to the seismic load obtained by the verification analysis. The strength of the actual plant primary coolant pipe was sufficiently lower than the allowable value.

6. CONCLUSIONS

(1) The structural integrity of the PWR RCS was proved in the seismic design condition. The vibrational characteristics obtained from the test were in good accordance with those from the design analysis.

(2) It was also verified in excessive vibration tests that there exists an adequate margin in the seismic design. The support structures, mainly snubbers, were confirmed to maintain their function even against twice the design load.

(3) The conventional seismic design method applied to actual plants was proved to be adequate comparing the responses from the verification analysis with those of the design analysis.

(4) By the simulation analysis of the test results, the nonlinear rigidity characteristics of the snubbers and one-sided rigidity supports, and other modelling of the support structures of the primary coolant loop system were accurately evaluated in terms of the vibration analysis.

REFERENCES

K. Fujita, T. Nakamura, H. Akiyama, et al., "Proving Test on the Seismic Reliability for the PWR Primary Coolant System", ASME PVP-Vol.182, pp.303-308, 1989.

H. Akiyama, K. Koyama, K. Tai, et al., "Study of Modal Damping of the PWR Reactor Coolant System", ASME PVP-Vol.197, pp.27-31, 1990.

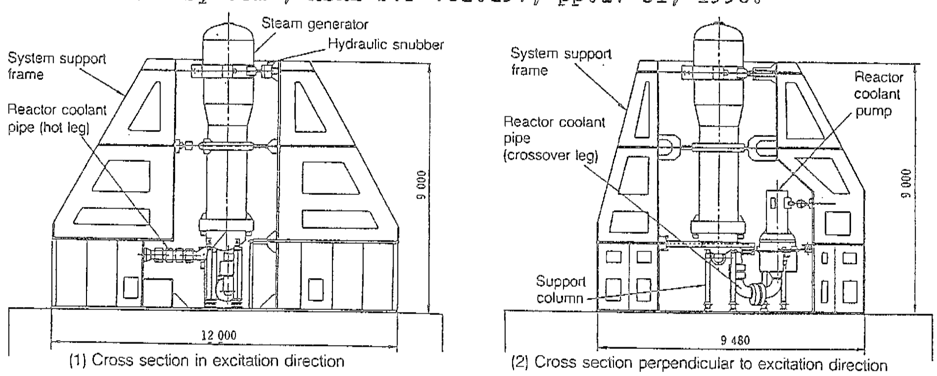

Fig. 2 Outline Drawing of PWR Primary Coolant Loop System Test Model

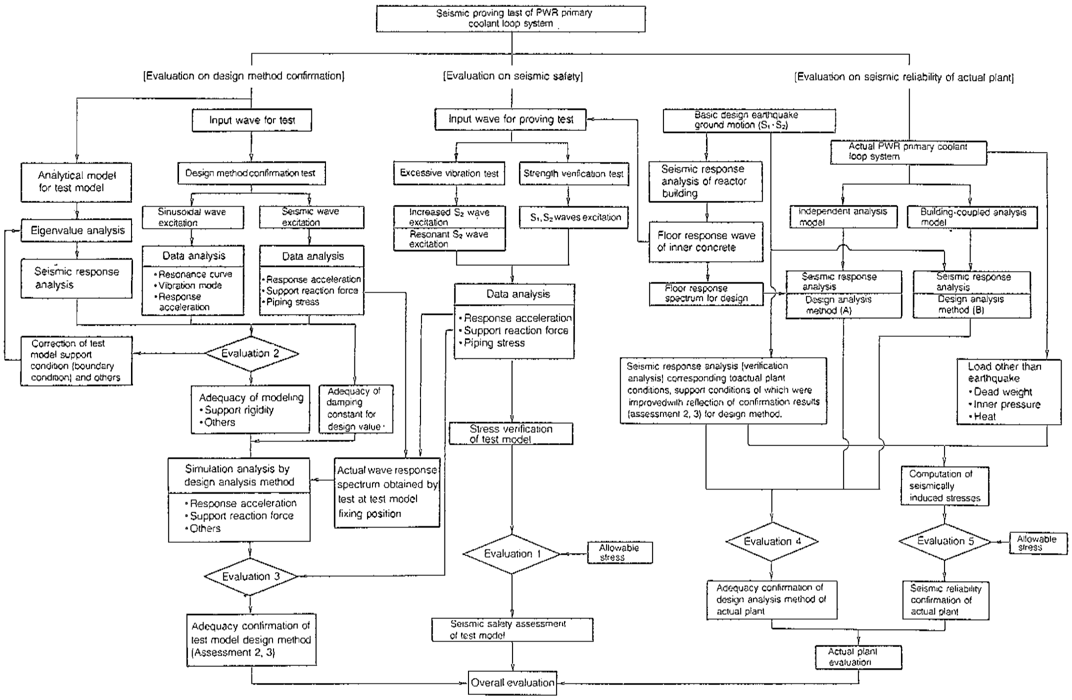

Fig. 1 Evaluation Flow of Seismic Proving Test of PWR Primary Coolant Loop System

Table 1 Input Waves for Proving Test (PWR Primary Coolant Loop System)

Input wave	Direction component	Duration** (sec)	Max. acceleration** (Gal)
S_1 floor response*	Horizontal and vertical, 2 directions	17.5	Horizontal: 433 Vertical: 98
S_2 floor response (1)*	〃	14	Horizontal: 904 Vertical: 138
S_2 floor response (2)*	〃	28	Horizontal: 1,170 Vertical: 334

Notes
* Floor response waves at the operating floor of the reactor building inner concrete (horizontal) and at the reactor vessel supporting floor (vertical), based on the basic design earthquake ground motions S_1 and S_2.

S_1 (simulated seismic wave):
Improved and standard plant for high seismic zone.
$M=7.0$, $\Delta=20$km
Phase characteristic = Eℓ Centro NS (1940)
Ground $V_1=1{,}500$m/s
S_2 (simulated seismic wave):
(1) Improved and standard plant for high seismic zone.
$M=6.5$, $\Delta=7.2$km
Phase characteristic = Cholome Shandon N40W (1966)
Ground $V_3=1{,}500$m/s
(2) Actual 4-loop PWR plant design wave (high seismic zone, distant and near earthquakes envelope seismic wave)

** The table shows values multiplied by 0.816 (1/1.225) for max. acceleration and those multiplied by 0.7 for duration, based on modified scaling law in consideration of the scale of the test model (1/2.5) and the compensational mass.

Table 2 Comparison of Support Structure Load na Pipe Stress at Strength Verification Test

(Unit: load [ton], stress [kgf/mm²])

Input earthquake / Load/stress / Direction / Components		S_1 wave			S_2 (1) wave			S_2 (2) wave		
		Test result	Predictive analysis (h=3%)	Design analysis (h=3%)	Test result	Predictive analysis (h=3%)	Design analysis (h=3%)	Test result	Predictive analysis (h=3%)	Design analysis (h=3%)
<Steam generator load>										
Upper shell support structure	X	24.9	22.0	43.5	51.8	56.6	114.2	61.0	62.2	147.1
Intermediate shell support structure	X	12.4	30.8	33.3	43.8	77.6	88.6	49.5	94.4	108.9
Lower support structure	Y	3.4	1.4	5.5	6.2	3.4	12.4	11.0	5.0	16.7
Support column	Z	19.4	17.3	21.4	32.4	36.1	41.2	68.9	58.2	61.9
<Reactor coolant pump load>										
Upper support structure	(1)	2.7	2.2	2.0	6.3	4.9	4.2	8.9	7.2	6.7
	(2)	0.7	1.0	1.7	2.9	2.6	3.7	3.1	3.4	5.8
Lower support structure	Y	3.2	1.8	3.6	5.9	4.2	7.8	7.5	7.0	11.9
Support column	Z	5.0	4.9	10.3	10.4	11.1	22.0	17.8	16.5	32.2
<Reactor coolant pipe stress>										
Hot leg		0.90	1.25	2.28	2.40	3.11	5.65	3.52	3.92	7.16
Crossover leg		0.54	1.04	2.29	1.20	2.34	4.97	2.20	2.90	6.69
Cold leg		0.64	1.15	1.54	1.48	2.48	3.40	2.16	3.74	5.12

(Notes) • Piping stress (σ) is obtained by calculation of max. strain (ε) at measuring point multiplied by Young's modulus (E). ($\sigma=\varepsilon E$)
• (1) and (2) in "direction" column mean horizontal directions (1) and (2) of upper support structures of the pump.

Table 3 Natural Frequency Obtained by the Design Method Confirmation Test

	Direction	Test result			Design analysis value
		Sinusoidal wave excitation level [Gal]	Natural frequency [Hz]	Damping ratio [%]	Natural frequency [Hz]
Steam generator	X	Horizontal 40	20.6	about 4	20.1
	Y	Horizontal 50	21.0	—	20.4
	Z	Vertical 30	20.9	1~1.5	24.5
Reactor coolant pump	X	Horizontal 40	26.3	—	27.7
	Y	Horizontal 40	27.9	—	26.5
Crossover leg piping	Y		>30		33.7

(Notes) "Damping ratio" indicates value obtained by half-power method.

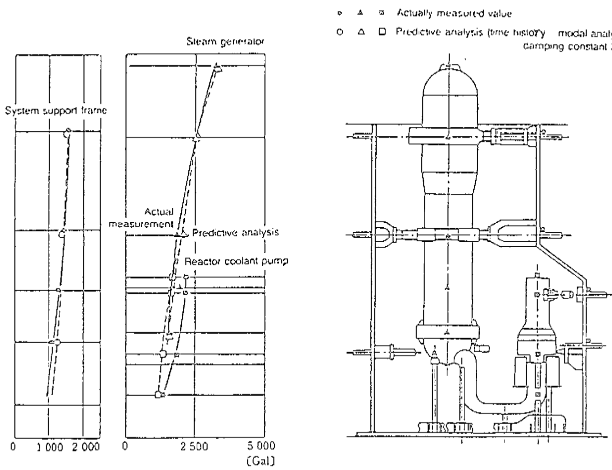

Fig. 3 Maximum Acceleration of the Test Model [S_2 (2) Horizontal Vibration]

Table 4 Analysis Method Applied to Seismic Reliability Evaluation of Actual PWR Primary Coolant Loop System

	Verification analysis	Design analysis	
		Design analysis method (A)	Design analysis method (B)
		Response spectrum	Time historical modal
Analytical model	Coupled with building	Separating equipment and piping system	Coupled with building
Rigidity of support structure	2.5 times as high as that of proving test model	Design rigidity	Design rigidity
Analysis method	Time history modal analysis considering local nonlinearity (one-sided rigidity support nonlinearity)	Floor response spectrum method	Time history modal analysis neglecting influence by one-sided rigidi-ty support nonlinearity to stay on the safe side.
Considera-tion of response fluctuation	Not considered	Floor response spectrum widened by ±10% in period direction	Consideration by ±10% change in time interval of input seismic wave
Damping constant	Modal damping SG(X) 5.5% SG(Y) 4.5% RCP(X,Y) 4%	3.0%	Modal damping 3.0%

Table 5 Max. Response Acceleration at S_2(2) Earthquake of Actual PWR Primary Coolant Loop System

(Unit: Gal)

		Verification analysis		Design analysis (A)		Design analysis (B)	
		E-W	N-S	X direction	Y direction	E-W	N-S
Steam generator top	X	3612	837	5417	122	4142	385
	Y	570	3186	294	3329	185	3828
	Z	581	528	1218	1233	669	657
Reactor coolant pump top	X	1370	653	2071	1198	1541	646
	Y	379	1498	2149	1735	420	1335
	Z	315	323	395	392	314	314
Hot leg	X	771	48	100	36	882	50
	Y	103	983	185	834	88	985
	Z	380	349	415	330	363	325
Crossover leg	X	979	394	2432	2636	1161	539
	Y	399	1171	1591	3773	438	1013
	Z	478	474	925	1001	515	534
Cold leg	X	1051	295	471	1420	1139	281
	Y	351	1078	1001	3466	349	1107
	Z	355	362	486	378	380	387

(Note) Verification analysis: Time history modal analysis (considering local nonlinearity of one-sided rigidity support).
Design analysis A: Response spectrum analysis (widening ±10% in period direction, damping 3%)
Design analysis B: Time history modal analysis (input flucutation ±10%, damping 3%)

Table 6 Max. Response Load at S_2(2) Earthquake of Actual PWR Primary Coolant Loop System

(Unit: ton)

		Verification analysis		Design analysis (A)		Design analysis (B)	
		E-W	N-S	X direction	Y direction	E-W	N-S
Steam generator upper shell snubber load	X	328.8	157.1	543.9	8.6	430	186.5
	Y	143.7	298.7	18.2	317.2	137.8	365.4
Steam generator intermediate shell snubber load		333.9	123.1	476.0	7.7	554.2	212.3
Steam generator lower support structure load		49.7	102.5	39.6	264.2	52.6	124.2
Steam generator support column load	(1)	94.3	73.5	130.3	133.2	114.5	88.7
	(2)	88.9	83.3	134.3	133.0	108.2	104.5
	(3)	100.8	74.1	133.5	134.4	120.7	89.3
	(4)	86.7	81.9	131.6	132.0	102.6	104.0
Reactor coolant pump upper snubber load	RR1	25.6	11.3	35.4	22.3	34.5	13.0
	RR2	8.7	23.8	33.9	23.7	9.5	31.5
Reactor coolant pump lower structure	RR3	53.7	57.0	49.5	162.6	68.4	71.9
Reactor coolant pump support column load	RR6	30.5	21.8	45.1	81.5	37.6	29.6
	RR7	33.1	35.7	49.9	31.4	41.7	39.0
	RR8	20.6	23.6	37.8	51.7	24.8	31.3

(Note) Verification analysis: Time history modal analysis (considering local nonlinearity of one-sided rigidity support).
Design analysis A: Response spectrum analysis (widening ±10% in period direction, damping 3%)
Design analysis B: Time history modal analysis (input flucutation ±10%, damping 3%)