APWR Reactor Internals 1/5 Scale Model Flow Test Report

Non-proprietary Version

August 2011

Revision History

Revision	Date	Page	Description
0	December 2007	All	Original issued
1	May 2009	All	Revision 1 is revised according to the NRC request by RAI 3.9.2-33 (RAI272-1585) with following changes. 1. Clarify the position of this technical report on the vibration assessment of the US-APWR reactor internals. 2. Addition of explanation of the test results and evaluation. 3. Identify the data base for Tables and Figures, such as the direct measured data, measured data scaled to J-APWR dimensions and analysis results.
2	August 2011	 ix 1 3,49-50 5-6 6-7 9 11 17 18-25 All	Revision 2 is revised with following changes: 1. Addition of Reference (3); R.G.1.20, Revision 3. 2. Addition of the list of RAI question; the responses to these questions are included in this report. 3. Addition of Appendix-A, in which the model modification was discussed with response to RAI question 03.09.02-70 (RAI498-3782). 4. Identify the manner to convert the flow-induced dynamic responses from a scale model test at the room temperature to those under the plant operating conditions in Section 5.2.with response to RAI question 3.9.2-72 (RAI498-3782). 5. Move the equation for the High Cycle Fatigue Evaluation from Section 6.2 to Section 5.3. 6. Identify the relationship between the loading moment and the measured strain in Section 6.2 with response to RAI question 3.9.2-73 (RAI498-3782) 7. New Chapter 8 is added to identify the information for the FIV bench-mark analysis as requested by NRC with RAI 03.09.02-89 (RAI614-4853). 8. Modify the column header in Tables 6-4 and 6-5. 9. Modify the editorial errors such as allowable stress in Tables 6-7 to 6-14. 10. Modify some editorial errors such as the addition of the article.

Mitsubishi Heavy Industries, Ltd.
16-5, Konan 2-chome, Minato-ku
Tokyo 108-8215 Japan

Abstract

This report summarizes the vibration test results of the APWR Reactor Internals 1/5 Scale Model Flow Test. This report was used as the reference benchmark analysis to verify the dynamic analysis methodology which was used in the vibration assessment program of the US-APWR reactor internals.

The Advanced Pressurized Water Reactor for Japan (hereafter J-APWR) is designed to produce a larger thermal output than the current operating 4 loop plants. To increase the thermal output, the number of the fuel assemblies is increased from 193 to 257. To accommodate the increased number of fuel assemblies, the diameter of the J-APWR core barrel was increased by 20%. Additionally, a neutron reflector (NR), which is the stacked and perforated metal ring blocks, was designed to replace the current baffle structure. The NR is designed improve the fuel cycle cost and to reduce the number of bolts in the high radiation area.

The radius of the main structure of the reactor internals is increased to accommodate the 257 fuel assemblies and the neutron reflector. Specifically, the radius of the core barrel is increased by approximately 20% from the current 4-loop plant. The following two types of reactor core compositions exist in the APWR series.

- J-APWR : 12 ft, 17x17 type, 257 fuel assemblies.
- US-APWR: 14 ft, 17x17 type, 257 fuel assemblies.

This scale model flow test was performed for the J-APWR as representative to confirm the vibration characteristics and structural integrity under flow conditions. Additionally a hydraulic test, was performed but is not included in this technical report

This technical report is the reference, based on the J-APWR 1/5 scale model and test data, to verify the analysis method used in the US-APWR vibration assessment. The detail of the analysis is discussed in Section 3 of MUAP-07027-P “The Comprehensive Vibration Assessment Program for US-APWR Reactor Internals” (Reference (1)).

Revision 1 of the above report was revised in response to the NRC request for additional information in RAI question 3.9.2-33 (RAI272-1585), with the following changes:

1. Clarify the position of this technical report on the vibration assessment of the US-APWR reactor internals.

2. Addition of explanation of the test results and their evaluation.

3. Identify the data base for Tables and Figures, such as the direct measured data, measured data scaled to the J-APWR dimensions and analysis results.

Revision 2 of the above report was revised in response to the NRC request for additional information in RAI questions, with the following changes:

1. New Chapter 8 is added to identify the information for the FIV bench-mark analysis as requested by NRC with RAI question 03.09.02-89 (RAI614-4853).

2. Appendix-A is added to identify the model modification of the J-APWR 1/5 SMT with response to RAI question 03.09.02-70 (RAI498-3782).

3. The adjustments method of displacement, strain and stress are added in Section 5.2 with response to RAI question 3.9.2-72 (RAI498-3782).

4. It is added in Section 6.2 that the relationship between the loading moment and the measured strain is obtained by a series of unit loading tests with response to RAI question 3.9.2-73 (RAI498-3782).

Table of Contents

List of Tables

List of Figures

List of Acronyms

APWR	Advanced Pressurized Water Reactor
J-APWR	Japan Advanced Pressurized Water Reactor
US-APWR	United States Advanced Pressurized Water Reactor
SMT	Scale Model test
GT	Guide Tube
MHI	Mitsubishi Heavy Industries
NRC	United States Nuclear Regulatory Commission
RCC	Rod Cluster Control
RCP	Reactor Coolant Pump
Re	Reynolds Number
RMS or rms	Root Mean Square
RV	Reactor Vessel
μs	Micro-strain
μm	Micro-meter

List of References

(1) Comprehensive Vibration Assessment Program for US-APWR Reactor Internals, MUAP-07027-P, Revision 2 Mitsubishi Heavy Industries, Ltd. August 2011.

(2) “Nuclear Power Plant Components,” ASME Boiler and Pressure Vessel Code. Section III, Division 1, American Society of Mechanical Engineers. Includes: NCA, NB, NC, ND, NF, NG, Code Cases and Appendices including Appendix I, F, and N, 2001 edition thru 2003 Addenda.

(3) Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing, Regulatory Guide 1.20, Revision 3.

1.0 INTRODUCTION

This technical report is the English version of the “Summary of the APWR Reactor Internals 1/5 Scale Model Flow Test Report” for the Japanese utilities (here after J-APWR 1/5 SMT). Thus the style of report was kept and the unit of the table and figure are described without converting it into US-units.

The J-APWR reactor internals 1/5 scale model flow test was performed to confirm the vibration characteristics and the structural integrity under flow condition of the APWR series reactor internals, and the pressure loss in the reactor of the J-APWR. This technical report summarizes the models and results of the vibration test only.

This report was used for the reference for the benchmark analysis of the J-APWR 1/5 scale model, as a verification of the dynamic analysis methods used in the vibration assessment of the US-APWR reactor internals. Details of the benchmark analysis are discussed in MUAP-07027-P “The Comprehensive Vibration Assessment Program for US-APWR Reactor Internals.” Reference(1)).

The response to the following RAI question is included in Revision 1 of this report.

(1) RAI question 3.9.2-33

The responses to the following RAI questions are included in Revision 2 of this report.

(1) RAI question 03.09.02-89
(2) RAI question 03.09.02-70
(3) RAI question 3.9.2-72
(4) RAI question 3.9.2-73

Revision 2 is revised with following changes:

1. New Chapter 8 is added to identify the information for the FIV bench-mark analysis as requested by NRC with RAI question 03.09.02-89 (RAI614-4853).

2. Appendix-A is added to identify the model modification of the J-APWR 1/5 SMT with response to RAI question 03.09.02-70 (RAI498-3782).

3. The adjustments method of displacement, strain and stress are added in Section 5.2 with response to RAI question 3.9.2-72 (RAI498-3782).

4. It is added in Section 6.2 that the relationship between the loading moment and the measured strain is obtained by a series of unit loading tests with response to RAI question 3.9.2-73 (RAI498-3782).

5. Modify some editorial errors and the composition.

2.0 SCOPE

This test scope was to measure the flow-induced vibration response, to confirm the structural integrity against the flow induced vibration loads and static flow loads, and to obtain the pressure fluctuations of the reactor internals for the APWR series. This test was performed based on the J-APWR reactor internals as a representative model in the APWR series. This test was performed at ambient temperature and pressure.

It is not intended that the evaluation results of the J-APWR reactor internals in this report are directly applied to the vibration assessment of the US-APWR reactor internals. The following data from this test were referenced to the verification of the vibration analysis methodology which was used in the vibration assessment of the US-APWR reactor internals.

(1) Measured natural frequencies of the 1/5 scale and the damping ratio of the J-APWR 1/5 SMT:
Refer to the test results in water 1/5 scale in Table 6-1.

(2) Spectra of the pressure fluctuation in the downcomer:
Refer to the test results of 77 GTs arrangements at the downcomer in Figure 6-5.

(3) Root mean square (rms) values of the measured vibration response:
Refer to the measured core barrel rms displacement data at the bottom under 100% flow condition in Tables 6-2 and 6-3.
Refer to the measured rms bending moment data for the beam structure of the reactor internals under 120% flow condition in Tables 6-4 and 6-5.

3.0 TEST MODEL

The outline of the test loop is shown in Figure 3-1 the test model of the reactor vessel and the reactor internals are shown in Figure 3-2.

- The scale factor of the model is 1/5. In the selection of the model scale was based on the results of non-dimensional analysis performed. One of the requirements to simulate the flow condition was to assure that the Reynolds number (Re) should be larger than 1×10^4 in order to maintain the developed turbulent flow condition. The results of the scaling law and Reynolds number under the flow condition of the tests are summarized in Table 3-1.
- Numbers of the RCC guide tubes (hereafter GTs) for this test were 77 and 85.
- The fuel assembly models were simplified to optimize the weight and pressure loss.
- The radial supports at the bottom of the core barrel were modeled to adjust the condition of closed gaps or opened gaps by the pushing bolts.
- The flow holes of the neutron reflector were simplified while maintaining the vibratory characteristics in accordance with the scaling law.

The model modification was discussed in Appendix-A.

4.0 MEASUREMENT APPROACH

The measurement points were selected taking into consideration the flow load and the new components which are shown in Table 4-1 and Figures 4-1 through 4-12.

- The accelerometers were installed to measure the flow-induced vibration responses.
- The strain gages were installed to evaluate the stresses and to obtain additional information of the flow-induced vibration responses.
- The displacement transducers were installed to measure the relative displacement between the reactor vessel and the core barrel, and between the core barrel and the neutron reflector
- Two pushing bolts located at the radial keys were connected to the load cells to measure the reaction loads at the radial keys.
- Pressure transducers were installed at several locations from the inlet nozzle to the outlet nozzles of the reactor vessel to measure the pressure fluctuation.

5.0 TEST PROCEDURE

5.1 Vibration Characteristic Test

- The natural frequencies, the mode shapes and the damping ratios (relative to the critical damping) were measured by both the impact excitation and the sine wave sweep excitation methods. The impact excitation test was performed only in-air. The sine wave sweep excitation test was conducted both in-air and in-water.
- The strains in the upper plenum structures and in the lower plenum structures due to the static loads were measured. These were used to calibrate the test data.

5.2 Flow Induced Vibration Test

Test conditions are shown in Table 5-1. The following test cases were performed with the specific parameters given:

- Flow rate,
- Number of the RCC guide tubes, 77 or 85 considering the variation of the core pattern,

[ ]

The following flow rates were selected for the reasons described below:

[ ]

These [three] flow conditions were chosen to obtain the vibration characteristic and to ensure that abnormal vibration was not observed.

The reason of performing the test with the various number of operation loops was to obtain the effect of number of the flow loops to the flow-induced vibration.
[ ]

The adjustments due to the differences in the geometric scaling, the fluid mass densities and Young's moduli are needed to convert the flow-induced dynamic responses from a scale model test at the room temperature to those in the full-size reactor under the plant operating conditions as following manner.

$$D_P = \frac{D_T}{Sc}\frac{\rho_P}{\rho_T}\frac{E_T}{E_P}$$

$$\varepsilon_P = \varepsilon_T \frac{\rho_P}{\rho_T}\frac{E_T}{E_P}$$

$$\sigma_P = \sigma_T \frac{\rho_P}{\rho_T} = \varepsilon_T E_T \frac{\rho_P}{\rho_T}$$

Where,

D :displacement (mm)
ε :strain (mm/mm)
σ :stress (kgf/mm^2)
ρ :fluid density (kg/mm^3)
E :Young's modulus (kgf/mm^2)

suffix P: in plant operating conditions
T: in Test conditions

Sc : geometric scale ratio (= 1/5 for J-APWR SMT)

The dynamic pressures in the room temperature test is 30-40 % higher than those under the plant operating conditions as determined by the ratio of fluid mass densities (1000 kg/m^3 at room temperature and 660-750 kg/m^3 under the plant operating conditions). On the other hand the stiffness of the structure under the plant operating conditions is reduced by 10 % from that at the room temperature in accordance with the difference in the Young's moduli.

In the data reduction process of the J-APWR 1/5 SMT, the difference in the fluid mass densities and Young's moduli due to the temperature difference were intentionally ignored for a conservative bias.

Due to a combination of the dynamic pressure and stiffness as discussed above, the dynamic responses such as the displacement, acceleration, stress and moment at room temperature are 20-30 % higher than those under the plant operating conditions because of the effect of temperature difference.

5.3 High Cycle Fatigue Evaluation

High cycle fatigue evaluations of the structures in the lower plenum were performed. The amplitudes of the alternating stresses in the test model were derived by multiplying the measured rms strains with the ratio of peak to rms amplitudes (peak factor) and the Young's modulus at the test temperature. This amplitude at the measurement point was translated into the amplitude of peak stress under the plant operating conditions by adjusting for the differences in the fluid mass densities, the section moduli, the stress concentration factors and the ratio of Young's moduli at the test and at the actual plant operating temperatures. The amplitudes of the alternating stress intensity were derived from the following equation.

$$S = \varepsilon_{rms} \times K_{rms} \times E_T \times \frac{\rho_P}{\rho_T} \times \frac{Z'}{Z} \times K_{ST} \times \frac{E_T}{E_P} < S_a$$

S : amplitude of the alternating stress intensity (kgf/mm^2)
ε_{rms} : rms amplitude of the measured strain

K_{rms} : assumed ratio of the peak amplitude to the rms response; [ ] (This peak factor was used in the J-APWR 1/5 SMT)

E_T : Young's modulus of the austenitic stainless steel at the room temperature; 19900kg/mm²

ρ_P/ρ_T : ratio of the fluid densities under plant operating condition and at the room temperature

Z'/Z : ratio of the section moduli at the measurement point to the stress evaluation point

K_{ST} : stress concentration factor for the discontinuous structures ; 5 (Reference (2)

E_P : Young's modulus of the austenitic stainless steel under the plant operating condition

S_a : allowable amplitude of the alternating stress intensities for 10^{11} cycles

6.0 TEST RESULTS

Note: The following test results were copied from the original test report for the J-APWR and translated from Japanese to English. The following abbreviations are used in the tables and figures where the results are summarized.

a. Measured data scaled to the actual dimensions: Measured values were converted by the scaling laws as shown in Table 3-1. For example, a 1/5 scaling factor for the dimension was used to scale the J-APWR design dimensions.
b. Application to the actual reactor internals or conversion to the actual structure or the actual plant: Measured values were converted by the scaling laws as shown in Table 3-1 and compensated for the difference in the material properties due to the difference in the temperature. For example, a multiplication factor of 1/5 for the dimension and another for the ratio of the Young's moduli at the two temperatures were used.
c. Test results: No conversion was used for measured strain.
d. Design loads: The design loads were used for the sizing or analysis of the components of the reactor internals.
e. Maximum strain or stress: These are the design values for the J-APWR reactor internals that are adjusted for the measurement results. For example, in case of the J-APWR reactor internals, the measured zero to peak rms value was adjusted by a factor of [ ] and multiplied by any structural stress concentration effect.

6.1 Vibration Characteristic Test

The natural frequencies and the damping ratio were obtained by the individual impact excitation tests for each component and the sine wave sweep excitation tests for the assemblies of the test model. These natural frequencies, after scaling up to the J-APWR reactor internals in water were shown in Table 6-1.

6.2 Flow Induced Vibration Test

a. Adverse flow effect

From the time histories of the strain data, the rms vibration responses were derived under each flow rate condition. The relationships between the rms responses and the flow rates are shown in Figures 6-1 through 6-4. In these figures, the measured rms responses are compared with the proportionality line to the square of the flow rate. It confirms that the vibration responses show good agreements with the slope of the proportionality lines. And there were no sudden changes under all flow conditions. Based on the test results, it is concluded that adverse effects such as the fluid-elastic vibration or the vortex shedding lock-in did not occur (Reference (2)).

b. Test data for FIV analysis

The power spectral densities data and the time history data were shown in Figures 6-5 and 6-6, which data were used for the FIV analysis.

c. Hydraulic loads

The hydraulic loads acting on each component were lower than the predicted design loads based on the potential flow theory except on the RV level instrumentation support tube. The reason was that the flow load on the RV level instrumentation support tube in the model test was caused by the flow from the neighboring the core barrel (shown in Tables 6-4 and 6-5). The results of the flow loads in the upper plenum structures show that there was no significant difference between the loads in the 77-GT and 85-GT configuration. The flow induced vibration load at the radial key was also smaller than the design value (shown in Table 6-6).

The relationship between the loading moment and the measured strain was obtained by a series of unit loading tests for the column structures, which have been performed as a part of the flow test. The moment at the end of the column was derived from the measured strain and correlation factor determined by the unit loading test.

d. Stress and high cycle fatigue

The stress and fatigue of each component based on the test results led to the confirmation that all components satisfy fully the allowable stress requirements including the RV level instrumentation support tube subjected to a load larger than the design load. The ratio of secondary loads shared between the components and the connecting bolts was ignored in the estimate of stresses in the bolts. This estimate was made on the basis that the stress was close to two times the actual stress, and so there is sufficient margin for ignoring the secondary loads on the bolts. (Figures 6-7 and 6-8, Tables 6-7 through 6-14)

7.0 CONCLUSION

7.1 Vibration Characteristics Test

- The test models were verified by comparing the natural frequencies between the test results and the J-APWR Pre-analysis results.
- The damping ratios of the each component were obtained.

7.2 Flow Induced Vibration Test for the J-APWR

- No adverse flow effects such as the lock-in with the vortex shedding or the fluid elastic instability was not observed.
- The measured stresses due to the flow induced vibration loads were sufficiently low compared with the allowable stresses in ASME Boiler Code Section III.

Although the above conclusions were not directly used as the design basis for the US-APWR, because of the similarities between the US-APWR and the J-APWR, this information added confidence to the vibration analysis results of the US-APWR, using the methodology and modeling technique verified by benchmarking the computed vibration results based on this scale model with the test data as described in 8.0.

8.0 INFORMATION FOR FIV BENCH-MARK ANALYSIS

Followings data are applied for the bench-mark analysis as the reference to verify the analysis methodology including the FE modeling and forcing functions. Same methodology is applied for the US-APWR design analysis.

- The natural frequencies of the 1/5 scale J-APWR reactor internals as shown in Table 6-1. Note that these data have been scaled up to the actual plant size by the scaling law as shown in Table 3-1.
- The damping ratio in water as shown in Table 6-1 is directly applied for the bench mark analysis as the best estimate value. (For the US-APWR, the 1% damping ratio is applied as the design value based on R.G. 1.20 (Reference (3))).
- The measured vibration rms responses under the nominal flow rate as shown in Table 6-2 and 6-3. Note that these data has been scaled up to the actual plant size by the scaling law in Table 3-1.

The bench-mark analysis is performed with the 1/5 scale model of J-APWR configuration. The material properties are determined in the room temperature as the test conditions. The flow velocities are also determined by accordance with the nominal test flow conditions. To comparison with the test results, the analysis results of the natural frequencies and the vibration responses are scaled to the actual plant size by same manner with the test data as shown in Table 3-1.

For the FIV analysis reported as MUAP-07027-P (Rev.0) in December 2007, the pressure fluctuation data measured in the down-comer of the 1/5 scale J-APWR model were applied as the input both for the bench-mark analysis and the US-APWR analysis. In the revision 1 analysis reported in May 2009, the down-comer forcing function have been replaced with the US-APWR 1/7 scale lower plenum test data which was completed in March 2008. In the revision 2 analysis, the US-APWR 1/7 scale lower plenum test data has been used (Reference (1)).

Table 3-1 Scaling Law and Flow Condition Comparison with J-APWR and Test Condition

Items		J-APWR	Test condition
Flow condition	Flow volume		
	Pressure		
	Temperature		
	Dynamic viscosity		
	Velocity at outlet nozzle		
Scaling law	Length		
	Strain		
	Stress		
	Velocity		
	Acceleration		
	Load		
	Frequency		
Reynolds number	Inlet nozzle		
	Downcomer		
	Outlet nozzle		

(1): Flow rate is mechanical design flow at that time.
(2): Add the unit of m^2/s.

Table 4-1 List of Transducers

Components	Accelerometer	Strain gauge	Displacement	Load cell	Pressure fluctuation
Hot-leg					
Cold-leg					
Reactor vessel					
Core barrel					
Neutron reflector					
Upper core support plate					
Upper core plate					
Lower core support plate					
RCC guide tube					
Upper support column					
Bottom mounted instrumentation					
Tie plate					
Mixing device					
Top slotted column					
RV level instrumentation support tube					
Radial key					
Total					

transducers installed inside reactor vessel (related to load and vibration)

Table 5-1 Test Conditions of J-APWR SMT

Operation loops	Flow rate (1)			Number of GTs (2)	Support condition at the radial key portion
				77 / 85	
				77 / 85	

(1): Flow rate was varied to confirm the stability of the structure.
(2): There are different numbers of GTs in the APWR series. In this test, two sets of GT arrangement, 77GT and 85GT, were used to study the effect of core loading pattern.

Table 6-1 Comparison of Vibration Characteristics of Test Condition and J-APWR Reactor Internals (77 GTs)

Components(1) \ Frequencies			Test results in water × 1/5 (Hz) (3)	J-APWR predicted by test (Hz)	Damping ratio (%)
Upper RCC guide tube					
Lower RCC guide tube					
Upper support column					
Top slotted column					
RV level instrumentation support tube					
Upper tie plate					
Lower tie plate					
Core barrel and neutron reflector	M (2)	N(2)			

(1): Mixing device (MD) was not included in evaluation as its natural frequency is extremely high, calculated as 300 Hz. Accordingly MD is regarded as a completely rigid body.
(2): M and N in core barrel and neutron reflector indicate beam mode order and shell mode order respectively.
(3): Test results in water× 1/5: Measured data is scaled to the J-APWR scale.
(4): The mode in phase and that of out of phase are identified.

Table 6-2 Converted Vibration Test Displacements (rms) to J-APWR Reactor Internals Displacements (77 GTs)

Condition / Measurement part		Direction	Core barrel bottom end Pin supported (μm)		Core barrel bottom end Free (μm)	
			Flow 100%	Flow 120%	Flow 100%	Flow 120%
Reactor vessel – core barrel						
Core barrel – neutron reflector						

Values inside ( ) indicate test data

Table 6-3 Converted Vibration Test Displacements (rms) to J-APWR Reactor Internals Displacements (85 GTs)

Condition / Measurement part		Direction	Core barrel bottom end Pin supported (μm)		Core barrel bottom end Free (μm)	
			Flow 100%	Flow 120%	Flow 100%	Flow 120%
Reactor vessel – core barrel						
Core barrel – neutron reflector						

Values inside ( ) indicate test data

Table 6-4 Conversion of Test Flow Loads to J-APWR Reactor Internals Flow Loads (77 GTs, 120% Flow)

Components	Location	measured moment(kgf-mm)			Conversion to J-APWR	Design loads
		Static	rms× [ ](3)	Total	Moment (kgf-mm)	Moment (kgf-mm)
RCC guide tube						
Upper support column						
Top slotted column						
Mixing device						
RV level instrumentation support tube						
Secondary core support column						
Base of bottom mounted instrumentation guide tube						
Bottom mounted instrumentation nozzle						

(1): Impossible to evaluate because of very minor strain below measurement limit
(2): The design loads for the bottom mounted instrumentation nozzle are not defined since this load value is negligible.
(3): Axial component of dynamic load was also estimated (bending component conservative estimate because of larger variation)

Table 6-5 Conversion of Test Flow Loads to J-APWR Reactor Internals Flow Loads (85 GTs, 120% Flow)

Components	Location	measured moment(kgf-mm)			Conversion to J-APWR	Design loads
		Static	rms× [ ] (3)	Total	Moment (kgf-mm)	Moment (kgf-mm)
RCC guide tube						
Upper support column						
Top slotted column						
Mixing device						
RV level instrumentation support tube						
Secondary core support column						
Base of bottom mounted instrumentation guide tube						
Bottom mounted instrumentation nozzle						

(1): Impossible to evaluate because of very minor strain below measurement limit
(2): The design loads for the bottom mounted instrumentation nozzle are not defined since this load value is negligible.
(3): Axial component of dynamic load was also estimated (bending component conservative estimate because of larger variation)

Table 6-6 Conversion of Test Vibration Loads to J-APWR Flow Induced Vibration Load on the Radial Key (120% Flow)

Number of GT	Test load	Conversion to J-APWR Radial Key Load (1)	Design loads
77			
85			

(1):Conversion to J-APWR Radial Key Load = Test Load x [ ] x 25
(2): ton= SI ton

Table 6-7 Conversion of Test Strains to J-APWR Stresses for Evaluation of the Reactor Internals (77 GTs, 120% Flow)

Evaluated components(1)		Measured strain (μs)			Conversion to J-APWR Stresses	
		Static	rms	Maximum strain(1) Static + rms × [ ]	Stress (2) (kgf/mm^2)	Allowable stress (kgf/mm^2)
Core barrel						17.5
Upper core support skirt						17.2
RCC guide tube						
Upper support column						
Top slotted column						
RV level instrumentation support tube						
Secondary core support column						17.7
Bottom mounted instrumentation guide tube						
Bottom mounted instrumentation nozzle						25.0

(1): In case of the J-APWR evaluation, factor (ratio of rms to peak) [ ] was used to calculate the maximum strain
(2): Young's modulus of 304 stainless steel at ambient temperature=19900kgf/mm²
Young's modulus of Alloy 690=21000kgf/mm²

Table 6-8 Converted Test Strains to J-APWR Stresses for Fatigue Evaluation of the Reactor Internals (77 GTs, 100% Flow)

Evaluated components(1)		Measured strain (μs)		Conversion to J-APWR stresses	
		rms	Maximum strain (1) (rms x [ ] x 5)	Maximum stress amplitude (kgf/mm²)	Allowable stress (kgf/mm²)
Core barrel					9.5
Upper core support skirt					
RCC guide tube					
Upper support column					
Top slotted column					
RV level instrumentation support tube					
Secondary core support column					
Bottom mounted instrumentation guide tube					
Bottom mounted instrumentation nozzle					

(1): In case of the J-APWR evaluation, factor (ratio of rms to peak) [ ] was used to calculate the maximum strain
(2): Young's modulus of 304 stainless steel at ambient temperature=19900kgf/mm²
Young's modulus of Alloy 690=21000kgf/mm²

Table 6-9 Converted Test Strains to J-APWR Stresses for Evaluation of the TSFs and Bolts(77 GTs, 120% Flow)

Evaluated components	Measured strain(1) Static +rms x [ ] (μs)	Conversion to J-APWR stresses: Bending moment(2) (kgf-mm)	TSF & Bolt stress(3) (kgf/mm^2)	Allowable stress (kgf/mm^2)
RCC guide tube bolt				18.1
Upper support column extension TSF				11.5
Upper support column TSF				18.1
Top slotted column extension TSF				11.5
Top slotted column bolt				18.1
RV level instrumentation support tube bolt				
Secondary core support column bolt				
Bottom mounted instrumentation guide tube bolt				

(1): In case of the J-APWR evaluation, factor (ratio of rms to peak) [ ] was used to calculate the maximum strain
(2): Bending moment load converted to the actual reactor internals
(3): TSF and Bolt stresses estimated by the bending moment based on the dimensions of the actual reactor internals

Table 6-10 Converted Test Strains to J-APWR Stresses for Fatigue Evaluation of the TSFs and Bolts (77 GTs, 100% Flow)

Evaluated components	Measured strain (1) rms x [ ] x 4 (μs)	Conversion to J-APWR stresses: Bending Moment(2) (kgf-mm)	TSF & Bolt Stress(3) (kgf/mm^2)	Allowable stress (kgf/mm^2)
RCC guide tube bolt				9.5
Upper support column extension TSFs				
Upper support column TSF				
Top slotted column extension TSFs				
Top slotted column TSFs				
RV level instrumentation support tube bolt				
Secondary core support column anchor bolt				
Bottom mounted instrumentation guide tube				

(1): In case of the J-APWR evaluation, factor (ratio of rms to peak) [ ] was used to calculate the maximum strain
(2): Bending moment load converted to the actual reactor internals
(3): TSF and Bolt stresses estimated by the bending moment based on the dimensions of the actual reactor internals

Table 6-11 Converted Test Strains to J-APWR Stresses for Evaluation of the Reactor Internals (85 GTs, 120% Flow)

Evaluated components(2)		Measured strain (1) (μs)			Conversion to J-APWR stresses	
		Static	rms	Maximum strain Static + rms × [ ]	Stress (kgf/mm^2)	Allowable stress (kgf/mm^2)
Core barrel	[					17.5
Upper core support skirt						17.2
RCC guide tube						
Upper support column						
Top slotted column						
RV level instrumentation support tube						
Secondary core support column						17.7
Bottom mounted instrumentation guide tube						
Bottom mounted instrumentation nozzle					]	25.0

(1): In case of the J-APWR evaluation, factor (ratio of rms to peak) [ ] was used to calculate the maximum strain

(2): Young's modulus of 304 stainless steel at ambient temperature=19900kgf/mm²
Young's modulus of Alloy 690=21000kgf/mm²

Table 6-12 Converted Test Strains to J-APWR Stresses for Fatigue Evaluation of Reactor Internals (85 GTs, 100% Flow)

Evaluated components(2)		Measured strain (1) (μs) rms	Measured strain (1) (μs) Maximum strain (rms x [ ] x 5)	Conversion to J-APWR Maximum stress amplitude (kgf/mm^2)	Conversion to J-APWR Allowable stress (kgf/mm^2)
Core barrel					
Upper core support skirt					
RCC guide tube					
Upper support column					
Top slotted column					
RV level instrumentation support tube					9.5
Secondary core support column					
Bottom mounted instrumentation guide tube					
Bottom mounted instrumentation nozzle					

(1): In case of the J-APWR evaluation, factor (ratio of rms to peak) [ ] was used to calculate the maximum strain
(2): Young's modulus of 304 stainless steel at ambient temperature=19900kgf/mm²
Young's modulus of Alloy 690=21000kgf/mm²

Table 6-13 Converted Test Strains to J-APWR Stresses for Evaluation of the TSFs and Bolts (85 GTs, 120% Flow)

Evaluated components	Measured strain (1) Static + rms x [ ] (μs)	Conversion to J-APWR stresses: Bending moment(2) (kgf-mm)	TSF or Bolt stress(3) (kgf/mm^2)	Allowable stress (kgf/mm^2)
RCC guide tube bolt				18.1
Upper support column extension TSF				11.5
Upper support column TSF				18.1
Top slotted column extension TSF				11.5
Top slotted column TSF				18.1
RV level instrumentation support tube bolt				
Secondary core support column bolt				
Bottom mounted instrumentation guide tube bolt				

(1): In case of the J-APWR evaluation, factor (ratio of rms to peak) [ ] was used to calculate the maximum strain
(2): Bending moment load converted to the actual reactor internals
(3): TSF and Bolt stresses estimated by the bending moment based on the dimensions of the actual reactor internals

Table 6-14 Converted Test Strains to J-APWR Stresses for Fatigue Evaluation of the TSFs and Bolts (85 GTs, 100% Flow)

Evaluated components		Measured strain(1) rms x [ ] x 4 (μs)	Conversion to J-APWR stresses		
			Bending moment(2) (kgf-mm)	TSF & Bolt stress(3) (kgf/mm²)	Allowable stress (kgf/mm²)
RCC guide tube bolt					9.5
Upper support column extension TSF					
Upper support column TSF					
Top slotted column extension TSF					
Top slotted column TSF					
RV level instrumentation support tube bolt					
Secondary core support column bolt					
Bottom mounted instrumentation guide tube bolt					

(1): In case of the J-APWR evaluation, factor (ratio of rms to peak) [ ] was used to calculate the maximum strain
(2): Bending moment load converted to the actual reactor internals
(3): TSF and Bolt stresses estimated by the bending moment based on the dimensions of the actual reactor internals

Figure 3-1 Outline of the Test Facilities

Figure 3-2 Overview of the Test Model

Figure 4-1 Measurement Points on the Reactor Vessel

Figure 4-2 Measurement Points on the Core Barrel

Figure 4-3 Measurement Points on the Neutron Reflector

Figure 4-4 Measurement Points on the Upper Core Support

Figure 4-5 Measurement Points on the Upper Core Plate

Figure 4-6 Measurement Points on the Lower Core Support Plate

Figure 4-7 Measurement Points on the RCC Guide Tube

Figure 4-8 Measurement Addresses of the RCC Guide Tubes

Figure 4-9 Measurement Points on the Upper Support Column, the Top Slotted Column, and the RV Level Instrumentation Support Tube

Figure 4-10 Measurement Addresses of the Upper Support Column, the Top Slotted Column, and the RV Level Instrumentation Support Tube

Figure 4-11 Measurement points and Addresses of the Lower Plenum Structures

Figure 4-12 Dynamic Pressure Measurement Points

Figure 6-1 Relationship between Upper Reactor Internals rms Acceleration and Flow Rate

Figure 6-2 Relationship between Lower Plenum Structures rms Acceleration and Strain, and Flow Rate

Figure 6-3 Relationship between Core Barrel and Neutron Reflector rms Acceleration and Flow Rate

Figure 6-4 Relationship between Relative Displacement rms Fluctuation and Flow Rate

Figure 6-5 Power Spectral Densities of Relative Displacement and Pressure Fluctuation

Figure 6-6 Time History Data of Relative Displacement and Pressure Fluctuation

Figure 6-7 Stress Evaluation Points of the Structures and the Bolts (120% Flow) 1/2

Figure 6-7 Stress Evaluation Points of the Structures and the Bolts (120% Flow) 2/2

Figure 6-8 Fatigue Evaluation Points of the Structures and the Bolts (100% Flow) 1/2

Figure 6-8 Fatigue Evaluation Points of the Structures and the Bolts (100% Flow) 2/2

Appendix-A Model modification of the J-APWR 1/5 SMT

The descriptions on the three kinds of the model modifications in the J-APWR 1/5 SMT and their effects on the test results are summarized in Table A-1.

Because the vibration of the fuel assembly was verified with a mock-up test, the fuel assembly in the 1/5 scale model was simplified. The numbers of rods and grids were reduced although the scaled mass and pressure drop were still simulated. The natural frequency of the fuel was not simulated. But its impact on the vibration responses of the reactor internals was small because their natural frequencies are well separated.

For the neutron reflector, the numbers and diameter of the flow-holes were modified so that the total section area of flow holes was properly scaled (1/25 of that in the actual plant). This modification had no impact on the shell mode stiffness and the natural frequencies as confirmed by the FE analysis.

The modification of the radial key was not for simplification but to control the test conditions. The shapes and locations of the radial key were modeled to simulate the flow around the radial key. Because the support condition of the core barrel bottom was controlled by the additional push bolts, the clearance of the radial key were extend to assure the no contact condition.

Table A-1 J-APWR 1/5 Scale Model Test Model Modifications

	Modification	Simulated properties	Effects on test results
Fuel Assemblies			
Clearance between radial support key and core barrel to support of core barrel			
Neutron Reflector			