APWR Reactor Internals
1/5 Scale Model Flow Test Report

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Abstract

This report describes summary of vibration test of the APWR reactor internals 1/5 scale model flow test.

Advanced Pressurized Water Reactor (hereafter APWR) is developed to large thermal output plant compared with the current 4 loop plant.
To increase thermal output, applied number of the fuel assemblies is 257. As for the reference, the current 4 loop plant applies 193 fuel assemblies.
Additionally, the neutron reflector, which is perforated metal ring blocks, is instead of the current baffle structure to improve the fuel cycle cost and to reduce bolts in high radiation area, The radius of the main structure of the reactor internals has expanded because it adopted 257 fuel assemblies and the neutron reflector, for example, the radius of core barrel is increased approximately 20% from current 4 loop plant.
The following two type of reactor core compositions exist as 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 is performed for J-APWR as representative type to confirm vibration characteristics and structural integrity under flow condition. Additionally hydraulic test was performed, but hydraulic test results are not including in this technical report.

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List of Acronyms

APWR	Advanced Pressurized Water Reactor
GT	Guide Tube
MHI	Mitsubishi Heavy Industries
NRC	United States Regulatory Commission
RCC	Rod Cluster Control
RCP	Reactor Coolant Pump
Re	Reynolds Number
RMS	Root Mean Square
RV	Reactor Vessel

1.0 PURPOSE

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

2.0 SCOPE

This test scope is to measure the flow-induced vibration response, to confirm the structural integrity against the flow-induced vibration load and static flow load, and to obtain the pressure fluctuation of the reactor internals for the APWR series.
This test was performed using the test model of J-APWR reactor internals as representative type of the APWR series.
The flow-induced vibration responses are used to verify the analytical models for the flow-induced vibration analysis.
This test is performed at ambient temperature and pressure.

3.0 TEST MODEL

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

- Scale of the model is 1/5. In the selection of the model scale, non-dimensional analysis has been performed. One of the requirements to be simulated the flow condition is to assure that the Reynolds number (Re) should be larger than $1x10^4$ that maintains the developed turbulent flow condition. The results of the scale low and Reynolds number with flow condition of the tests are summarized in Table 3-1.
- Numbers of the RCC guide tubes (hereafter GTs) for this test are 77 and 85.
- The fuel assembly models are simplified considering the weight and pressure loss.
- The radial supports at the bottom of the core barrel are modeled by the pushing bolts.
- The flow holes of the neutron reflector are simplified maintaining the vibratory characteristics and the weight considering scale low.

4.0 MEASUREMENT APPROACH

The measurement points are selected considering flow load and newly parts which are shown in Table 4-1 and Figure 4-1 to Figure 4-12.

- The accelerometers are attached to measure the flow-induced vibration responses.
- The strain gages are attached to evaluate stress and to obtain additional information of flow-induced vibration responses.
- The displacement sensors are attached to measure interface displacement between the reactor vessel and the core barrel, and between the core barrel and the neutron reflector.
- Two pushing bolts located at radial keys are performed to load cells to measure reacting loads at the radial keys.
- Pressure transducers are attached several poison 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

- Measure the natural frequencies, the mode shapes and the dampings by the impact excitation test and the sin wave sweep excitation. The impact excitation test is performed only air condition. The sin wave sweep excitation test is conducted both in air and water.
- The strain of the upper plenum structures and the lower plenum structures by static load are measured which are used to calibrate the test data.

5.2 Flow-Induced Vibration Test

Test conditions are shown in Table 5-1. The following parameters and the test cases were demonstrated:
- Flow rate,
- Number of the RCC guide tubes, 77 or 85,

[ ]

As for the flow rate, reasons of selection are described as follows:

[ ]

Selected[ ]cases of flow conditions: To obtain the vibration characteristic and to confirm any abnormal vibration is not observed.

Reason of various number of operation loops is to obtain effect of flow loops number.
[ ]

6.0 TEST RESULTS

Test results are copied from original test report and converted language from Japanese to English. Values and units are not converted from original test report.

6.1 Vibration Characteristic Test

The test models were verified by comparing the natural frequency between test results and analysis results. The natural frequencies and the damping ratio were obtained by the individual impact excitation tests for each part and the sin wave sweep excitation tests for assembling condition. The converted natural frequencies corresponding to actual reactor internals in water were equivalent with the analysis results. (Table 6-1)

6.2 Flow-Induced Vibration Test

- The vibration responses of both the 77 and 85 GT cases were identified as proportional to the square of flow, and it was confirmed that abnormal vibration was not identified by Karman vortex and self-excited vibration. (Figure 6-1 shows the upper reactor internals,

Figure 6-2 shows the lower plenum structure, and Figure 6-3 shows the core barrel and the neutron reflector)

- The maximum amplitude of the displacement (RMS) of the neutron reflector was about [ ] which was sufficiently low compared with the displacement level of the current PWR plant. (Tables 6-2 and 6-3)
- The fluid loads acting on each portion were lower than the estimated design load excepting the water level indication system support tube. The reason was supposed that flow came from neighboring the core barrel. (Tables 6-4 and Table 6-5) As a results of the flow load of the upper plenum structures between 77 GTs and 85 GTs, There was not observed significant difference between the 77 GTs and 85 GTs. The flow-induced vibration load at the radial key is sufficiently smaller than design value. (Table 6-6)
- The strength evaluation of each portion based on the test results led to the confirmation of all components to satisfy fully the allowable stress requirements including the RV level instrumentation support tube subjected to the load larger than the design load. The loading ratio to be shared between the components and bolts was neglected in the estimation of bolts. This estimation was made on the basis of the stress close to two times the actual stress. (Figure 6-4 and Figure 6-5, Table 6-7 through 14)
- Pressure fluctuation in the reactor vessel by the random excitation due to flow is obtained.

7.0 CONCLUSION

7.1 Vibration Characteristics Test

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

7.2 Flow Induced Vibration Test

- It had not observed any abnormal vibratory response.
- The measured stresses due to flow-induced vibration loads are sufficiently low compared with allowable stress which is equivalent with ASME Code Section III.
- Design bases to verify the for flow-induced vibration analysis model, such as pressure fluctuation data and responses, were obtained.

Table 3-1 Scale Low and Non-dimensional Value

Items		Plant condition (Base)	Test condition
Flow condition	Flow volume		
	Pressure		
	Temperature		
	Dynamic viscosity		
	Velocity at outlet nozzle		
Scale low	Length		
	Strain		
	Stress		
	Velocity		
	Acceleration		
	Load		
	Frequency		
Reynolds number	Inlet nozzle		
	Downcomer		
	Outlet nozzle		

1) Flow rate is mechanical design flow at that time.

Table 4-1 List of Number of Sensors

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					

Note: sensors installed inside reactor vessel (related to load and vibration)

Table 5-1 Test Conditions

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

Table 6-1 Evaluation of Vibration Characteristics of Actual Reactor Internals (Normal Operating Condition)

Frequencies / Components			Test results in water× 1/5 (Hz[ ])	Actual predicted by test (Hz)	Analysis (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	N				

Note)
- 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.
- M and N in core barrel and neutron reflector indicate beam mode order and shell mode order respectively.

Table 6-2 Converted Vibration Response of Displacement (RMS) Corresponding to Actual Reactor Internals (In the case of 77GTs)

(μm)

Measurement part \ Condition		Direction	Core barrel bottom end Pin supported		Core barrel bottom end Free	
			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 Response of Displacement (RMS) Corresponding to Actual Reactor Internals (In the case of 85GTs)

(μm)

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

Values inside ( ) indicate test data

Table 6-4 Evaluation Results of Flow Load (120% Flow) (77GTs)

Components	Location	Test results			Conversion to actual structure	Design condition
		Static moment (kgf-mm)	RMS moment (kgf-mm)	Total moment (kgf-mm)	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						

Table 6-5 Evaluation Results of Flow Load (120% Flow) (85 GTs)

Components	Location	Test results			Conversion to actual structure	Design condition
		Static moment (kgf-mm)	RMS moment (kgf-mm) *3	Total moment (kgf-mm)	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: Design load not established
*3: Axial component of dynamic load was also estimated (bending component conservative estimate because of larger variation)

Table 6-6 Flow Induced Vibration Load on the Radial Key (120% Flow)

Number of GT	Test data	Conversion to actual structure	Design value
77			
85			

ton= SI ton

Table 6-7 Stress Evaluation in the case of 77 GTs (120% Flow)

Evaluated components		Measured strain (μs) Static	RMS	Maximum strain Static + RMS × 3	Stress (kg/mm^2)	Allowable stress (kg/mm^2)
Core barrel						
Upper core support skirt						
RCC guide tube						
Upper support column						
Top slotted column						
RV level instrumentation support tube						17.6
Secondary core support column						
Bottom mounted instrumentation guide tube						
Bottom mounted instrumentation nozzle						25

* Young's modulus of 304 stainless steel at ambient temperature=19900kg/mm^2
Young's modulus of Alloy 690=21000kg/mm^2

Table 6-8 Fatigue Evaluation in the case of 77 GTs (100% Flow)

Evaluated components	Measured strain (μs)		Maximum stress amplitude (kg/mm^2)	Allowable stress (kg/mm^2)
	RMS	Maximum strain (RMS x 3 x 5)		
Core barrel				11.6
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				

* Young's modulus of 304 stainless steel at ambient temperature＝19900kg/mm²
Young's modulus of Alloy 690＝21000kg/mm²

Table 6-9 Stress Evaluation of the Bolts in the case of 77 GTs (120% Flow)

Evaluated components	Measured strain Static + RMS x 3 (μs)	Bending moment (kg-mm)	Bolt stress (kg/mm^2)	Allowable stress (kg/mm^2)
RCC guide tube bolt				18.1
Upper support column extension				17.6
Upper support column bolt				18.1
Top slotted column extension				17.6
Top slotted column bolt				
RV level instrumentation support tube bolt				
Secondary core support column bolt				18.1
Bottom mounted instrumentation guide tube bolt				

* Bending moment load converted to the actual reactor internals
Bolt stress estimated by the bending moment based on the dimensions of the actual reactor internals

Table 6-10 Fatigue Evaluation of the Bolts in the case of 77 GTs (100% Flow)

Evaluated components	Measured strain RMS x 3 x 4 (μs)	Bending moment (kg-mm)	Bolt stress (kg/mm²)	Allowable stress (kg/mm²)
RCC guide tube anchor bolt				35.7
Upper support column extension				24.8
Upper support column anchor bolt				35.7
Top slotted column extension				24.8
Top slotted column anchor bolt				35.7
RV level instrumentation support tube anchor bolt				
Secondary core support column anchor bolt				
Bottom mounted instrumentation guide tube anchor bolt				

* Bending moment load converted to the actual reactor internals
Bolt stress estimated by the bending moment based on the dimensions of the actual reactor internals

Table 6-11 Stress Evaluation in the case of 85 GTs (120% Flow)

Evaluated components		Measured strain (μs)			Stress (kg/mm^2)	Allowable stress (kg/mm^2)
		Static	RMS	Maximum strain Static + RMS × 3		
Core barrel						17.6
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						25

* Young's modulus of SUS304 at ambient temperature=$19900 kg/mm^2$
Young's modulus of TT690=$21000 kg/mm^2$

Table 6-12 Fatigue Evaluation in the case of 85 GTs (100% Flow)

Evaluated components	Measured strain (μs) RMS	Measured strain (μs) Maximum strain (RMS x 3 x 5)	Maximum stress amplitude (kg/mm^2)	Allowable stress (kg/mm^2)
Core barrel				11.6
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				

* Young's modulus of 304 stainless steel at ambient temperature=19900kg/mm²
Young's modulus of Alloy 690=21000kg/mm²

Table 6-13 Stress Evaluation of the Bolts in the case of 85 GTs (120% Flow)

Evaluated components	Measured strain Static + RMS x 3 (μs)	Bending moment (kg-mm)	Bolt stress (kg/mm^2)	Allowable stress (kg/mm^2)
RCC guide tube anchor bolt				18.1
Upper support column extension				17.6
Upper support column anchor bolt				18.1
Top slotted column extension				17.6
Top slotted column anchor bolt				18.1
RV level instrumentation support tube anchor bolt				
Secondary core support column anchor bolt				
Bottom mounted instrumentation guide tube anchor bolt				

* Bending moment load converted to the actual reactor internals
Bolt stress estimated by the bending moment based on the dimensions of the actual reactor internals

Table 6-14 Fatigue Evaluation of the Bolts in case the of 85 GTs (100% Flow)

Evaluated components	Measured strain RMS x 3 x 4 (μs)	Bending moment (kg-mm)	Bolt stress (kg/mm^2)	Allowable stress (kg/mm^2)
RCC guide tube anchor bolt				35.7
Upper support column extension				24.8
Upper support column anchor bolt				35.7
Top slotted column extension				24.8
Top slotted column anchor bolt				35.7
RV level instrumentation support tube anchor bolt				
Secondary core support column anchor bolt				
Bottom mounted instrumentation guide tube anchor bolt				

* Bending moment load converted to the actual reactor internals
Bolt stress 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 Point on the Upper Core Plate

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Figure 4-7 Measurement Points on the RCC Guide Tube

Figure 4-8 Measurement Address 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 Address of the Lower Plenum Structures

Figure 4-12 Pressure Fluctuation Measurement Points

Figure 6-1 Relation between Upper Reactor Internals Response Acceleration (RMS) and Flow Rate

Figure 6-2 Relation between Lower Plenum Structures Response Acceleration and Strain (RMS), and Flow

[ ]

Figure 6-3 Relation between Core barrel and Neutron Reflector Response Acceleration (RMS) and Flow Rate

Figure 6-4 Relation between Relative Displacement and Pressure Fluctuation, and Flow Rate

Figure 6-5 Fourier Transform Results of Relative Displacement and Pressure Fluctuation

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

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

Figure 6-7 Stress Evaluation Point of the Structure 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