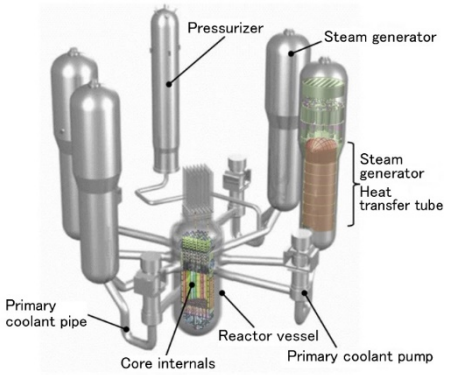
Design Improvement and Evaluation Technology Enhancement of PWR Main Components for Higher Reliability



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The main components that make up the reactor cooling system of a PWR nuclear power plant are responsible for the reactor cooling function and are vital for safety. These components are required to keep their integrity against the ground motion design specifications increased by the revision of the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities in 2006. Reliability improvement of their design, through the improvement and update of the evaluation technology enhancement, is also continuously carried out. We have accurately evaluated the seismic strength of the components using enhanced evaluation methods verified by large-scale tests, etc., and confirmed that they have sufficient reliability to withstand harsh seismic conditions. As a comprehensive nuclear plant manufacturer, we will continue to develop technologies that contribute to the further improvement of safety and reliability of nuclear power plants.

1. Introduction

The main components that make up the reactor cooling system of a PWR (Pressurized Water Reactor), such as the reactor vessel (RV), core internals (CI) and steam generator (SG), are important for the safety of the nuclear power plant. Therefore, we have been working on various design improvements and evaluation technology enhancement of the main components with the goal of improving safety and reliability of nuclear power plants. In particular, since the revision of "Regulatory Guidelines for Reviewing Seismic Design of Nuclear Power Reactor Facilities" in 2006, the design basis seismic ground motion has been increased, and further improvement in the seismic resistance is desired.

This report introduces efforts to improve the seismic resistance of replacement CI and SG heat transfer tubes using large-scale tests and enhanced analysis technology as representative examples of recent improvements in the seismic resistance margin of PWR main components.

2. Efforts to improve seismic resistance of replacement CI

2.1 Replacement of CI

Integrated replacement work of the CI was completed[(1)] in Ikata Nuclear Power Station Unit 1 operated by Shikoku Electric Power Co., Inc. for the first time in the world in 2004. This work was motivated by the needs of installing additional control rods accompanied by the adoption of high burnup fuel as well as the preventive maintenance for bolts (baffle-former bolts), whose degradation were reported overseas.

. Thereafter, integrated replacement work for CI was also carried out at Ikata Nuclear Power Station Unit 2 and Genkai Nuclear Power Station Units 1 and 2 operated by Kyushu Electric Power Co., Inc., which are two-loop plants similar to Ikata Nuclear Power Station Unit 1. Taking advantage of these experiences, in 2020, integrated CI replacement was carried out at Mihama

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Nuclear Power Station Unit 3 operated by the Kansai Electric Power Co., Inc. for a three-loop plant for the first time. In addition to preventive maintenance measures for baffle-former bolts and their reliability improvement, the replacement CI employs design improvements and enhanced seismic resistance evaluation methods to improve the margin against the recent strengthening of the design basis seismic ground motion. An overview of these design improvements and the seismic resistance evaluation is described below.

2.2 Design improvements of replacement CI

Figure 1 shows the major design improvements of the replacement CI. The replacement CI is based on the improved standard three-loop plant (Sendai Nuclear Power Station Unit 1 operated by Kyushu Electric Power Co., Inc. and later plants). It employs structural changes to the upper core support plate (changing the steel disk to a steel disk with a cylindrical shell to reduce the length of the upper core support column and control rod cluster guide tube and improve the resistance to cross-flow fluid load and seismic load). It also features modifications to the thermal shield configuration (changing the cylindrical type that covers the entire core barrel to a split type that covers only the vicinity of the direction of the maximum neutron flux to improve the manufacturability) compared to before the replacement. In addition, for the purpose of improving the reliability, the design improvements of the baffle structure and the radial support key described below have been adopted.

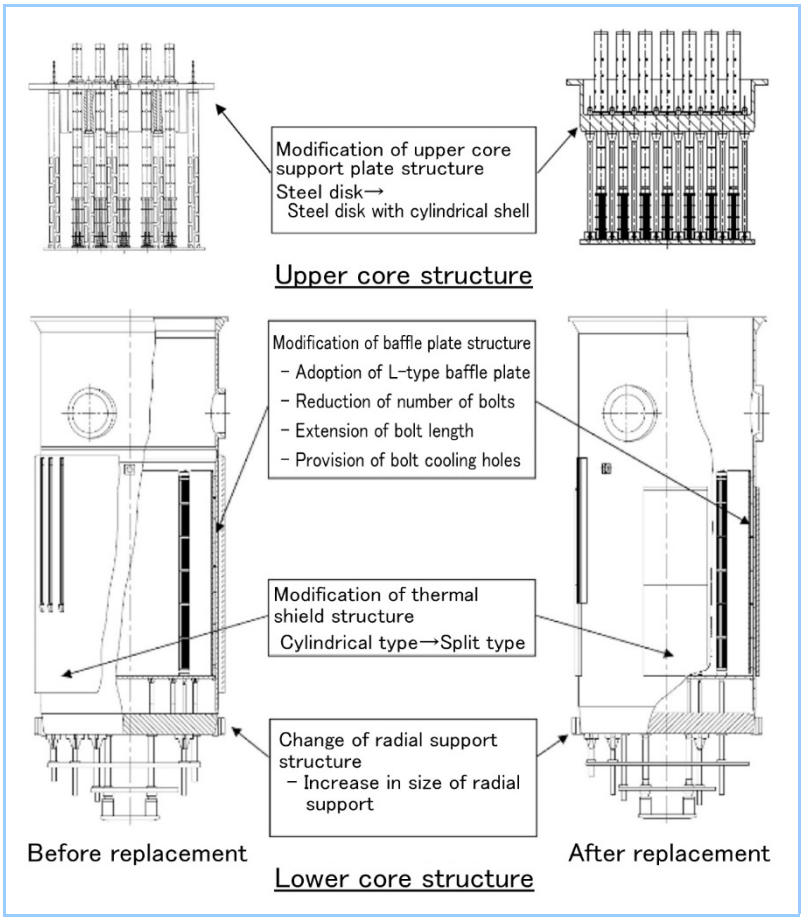


Figure 1 Major design improvements of replacement CI

(1) Baffle structure

The baffle structure is a structure that forms a flow path allocating the primary coolant in the reactor to the fuel region, and is composed of baffle plates and former plates, as well as baffle-former bolts for fastening them together. Among these components, baffle-former bolts are used under high stress due to the initial fastening force, thermal expansion of the baffle structure and bending load resulting from the deformation caused by neutron irradiation (swelling deformation). They are also subject to high neutron irradiation because they are installed adjacent to the fuel, so cases of damage due to irradiation assisted stress corrosion cracking (IASCC) have been reported at a number of overseas plants. Preventive measures in

terms of IASCC are taken for replacement CI, including the adoption of an L-type baffle (integrated type) instead of the conventional split-shaped core baffle for increasing the rigidity, the extension of the baffle-former bolt to reduce the stress thereon and the provision of bolt cooling holes to reduce the environmental temperature, as shown in **Figure 2**.

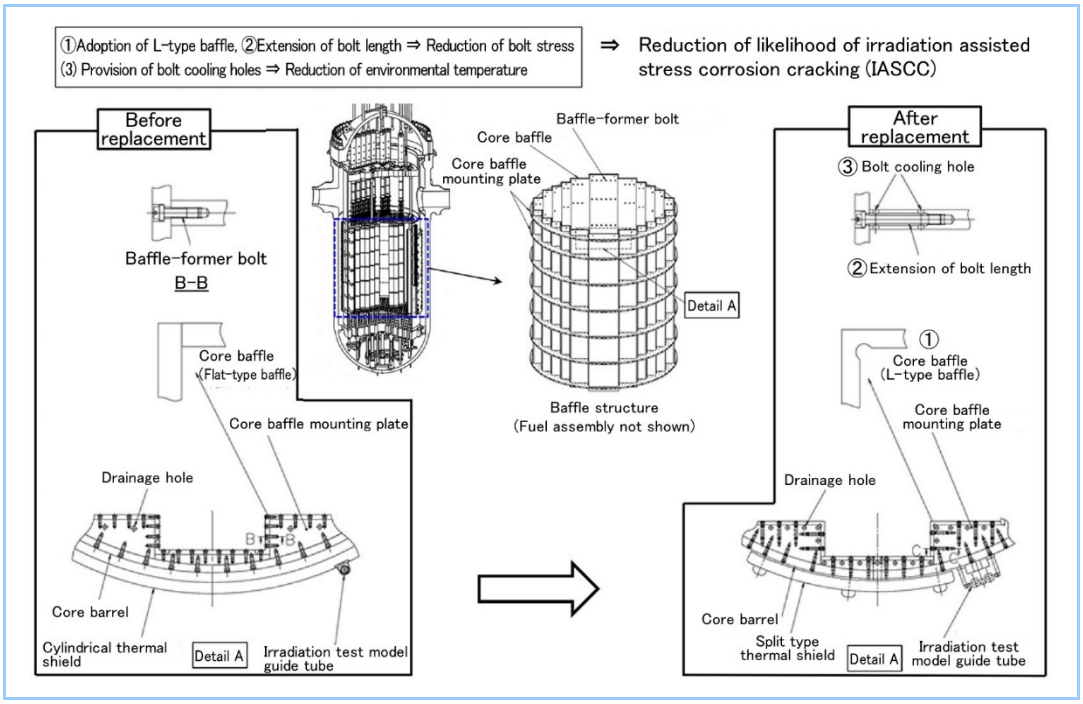


Figure 2 Design improvements of baffle structure

(2) Radial support key

The radial support key is a convex structure located at the bottom of the CI, and engages with the concave clevis insert located on the RV side in a key-groove manner to position the CI horizontally to the RV while relieving the relative elongation due to the difference in thermal expansion in the vertical direction between the CI and the RV. This is one of the important structures that support the horizontal seismic load acting on the CI when an earthquake occurs. The radial support key before the replacement was designed with sufficient margin to withstand the design basis seismic ground motion at the time of construction. However, in order to keep integrity against the recent standard ground motion, which has been increased since then, the design has been improved for the replacement to increase the seismic resistance margin.

While the design of CI for new plants can be improved with a relatively high degree of freedom including structures on the RV side, the design of replacement CI needs to be improved with the interface to the existing RV side structure maintained. Considering the interface with the RV and the difference in the thermal expansion between the RV and the CI due to start/stop and transient during operation, the radial support key of the replacement CI is enlarged as much as possible within the range where interference with the RV does not occur, and the number and diameter of the bolts are increased to improve the seismic resistance (**Figure 3**).

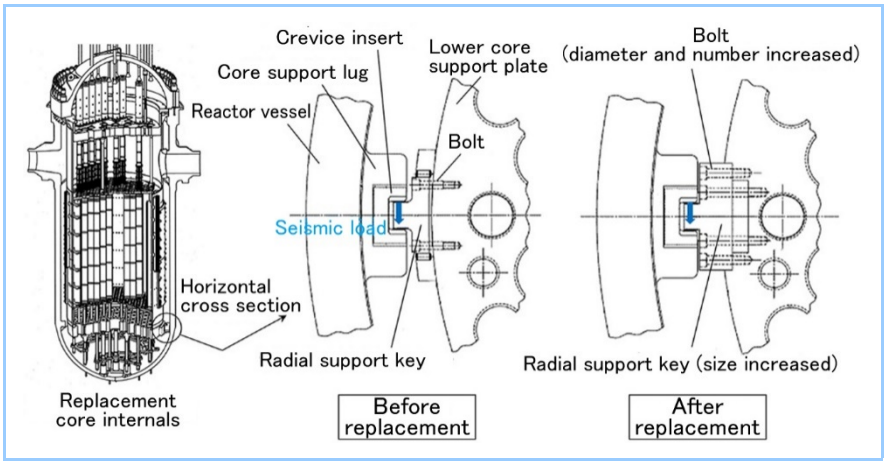


Figure 3 Increase in size of radial support key

2.3 Enhancement of seismic evaluation technology

In the past, for the seismic resistance evaluation of radial support keys, a method for verifying the integrity using the elastic stress, obtained by dividing the seismic load by the cross section, etc., has been used as an evaluation index. This method results in an overly conservative evaluation, but it is simple to use for evaluation. In order to accurately understand the margin of the radial support key under harsher seismic conditions than before, the yield strength (allowable load) of the radial support against the shear load, was obtained by a limit analysis method, and it was confirmed that the yield strength exceeds the load when an earthquake occurs. In addition, for verification of the limit analysis method, the conservativeness of the allowable load obtained by the limit analysis (meaning that the actual yield strength is higher than the allowable load obtained by limit analysis) was confirmed by a test using a scale model that simulates a radial support key.

2.3.1 Limit analysis

Limit analysis is one of the strength evaluation methods defined in the standards[(2)(3)] and is specified as an alternative method for strength evaluation using elastic stress analysis. The limit analysis assumes an ideal elastic-perfectly plastic material that does not include strain hardening and analyzes the load at which the structure yields over the entire cross section and cannot bear any more load. By using finite element (FE) analysis, it is possible to perform limit analysis of a complicated structure. The actual material is strain-hardened, therefore the collapse load obtained by the limit analysis is conservative with respect to the critical load that the actual structure can bear. **Figure 4** shows the FE analysis model used in the limit analysis of a radial support key and a graphic representation of the physical property values of the elastic-perfectly plastic material input to the model. **Figure 5** gives the results of the limit analysis. The collapse load of the radial support key resulting from the limit analysis was 1.8 times the load corresponding to the allowable stress in the case of the conventional evaluation method. It was also confirmed that the allowable load, which is obtained by multiplying the collapse load by a predetermined safety factor, exceeded the assumed seismic load of the replacement CI, which indicates that the radial support key is sound against harsher seismic conditions compared to before.

2.3.2 Validation test

The seismic resistance evaluation of radial support keys using limit analysis is a method that differs significantly from conventional seismic resistance evaluation based on elastic stress analysis. In order to confirm that this evaluation method is reasonable, a test using a scale test model of the radial support key was performed in order to determine the maximum load that the radial support key can bear (critical load), and confirm that this was sufficiently larger than the collapse load obtained by the limit analysis[(4)] (refer to Figure 5). The test was conducted using a 2000-ton class reaction wall at our Research & Innovation Center (Takasago District). **Figure 6** depicts the test model and test equipment. The scale ratio of the test model was set to 2/5 in consideration of the reproducibility of the actual equipment shape and the manufacturability of the test model. In general, in the range where the strain of the material is in the uniform elongation region, the scale law holds and the stress-strain relationship between the actual

equipment and the scale model is the same. In this test, the strain generated in the test model is expected to be basically in the uniform elongation region, it is therefore considered that the scale law holds.

From the above, it was verified that the seismic resistance evaluation of radial support keys, using the limit analysis, is sufficiently conservative and is reasonable as a seismic resistance evaluation method.

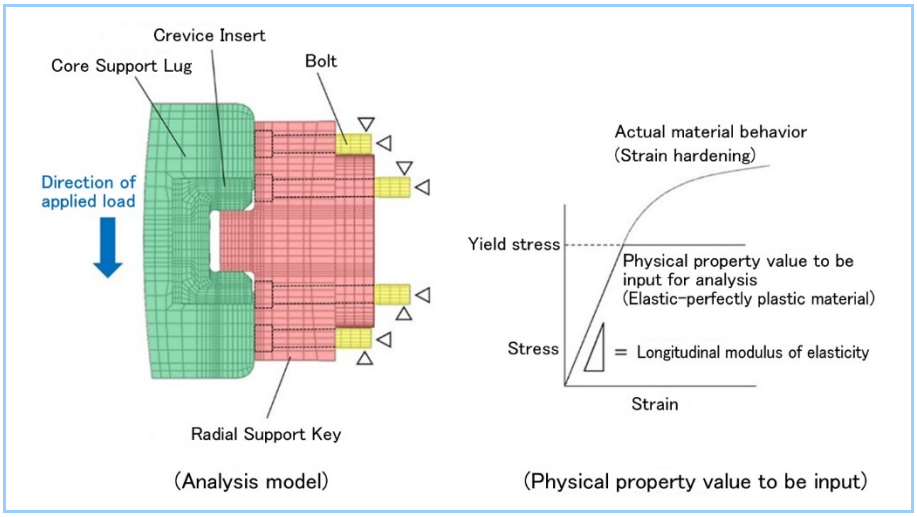


Figure 4 FEM analysis model of radial support key and input of physical property value

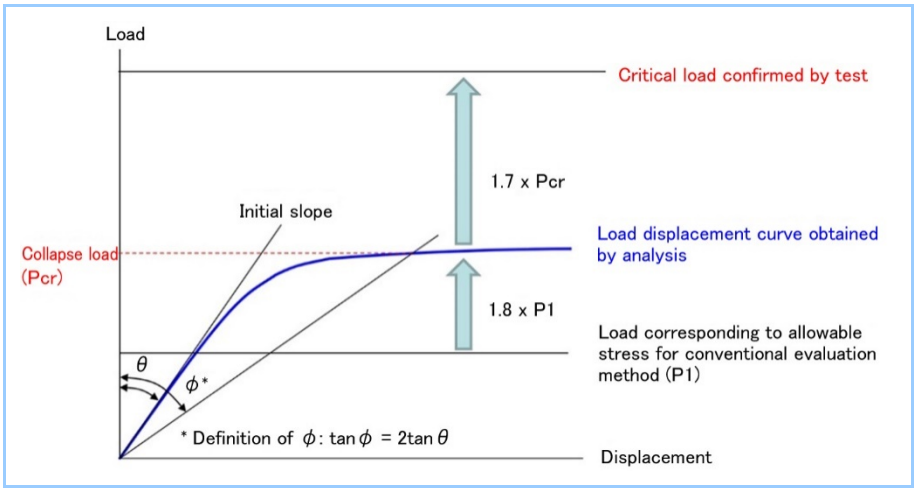


Figure 5 Result of limit analysis

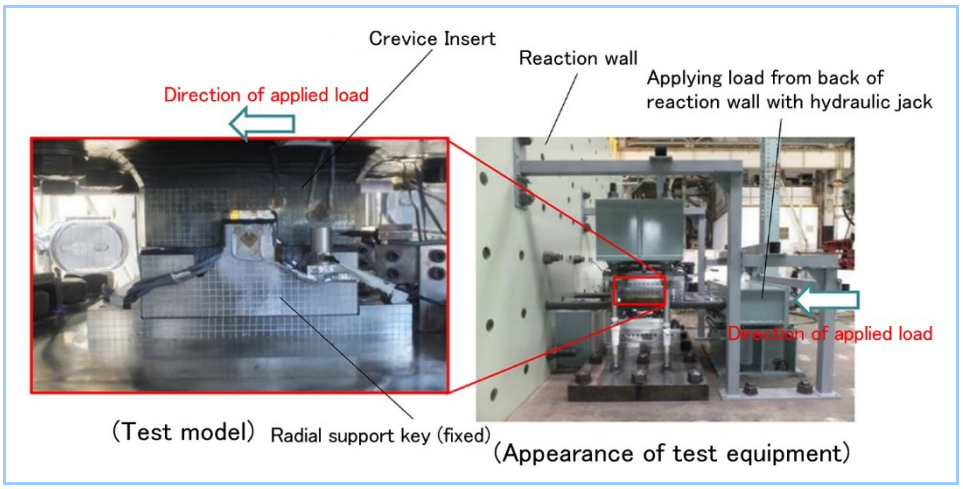


Figure 6 Test model and test equipment of radial support key

3. Efforts to enhance seismic resistance evaluation method of SG

3.1 Development of technology to enhance seismic resistance evaluation method

An SG is a heat exchanger composed of thousands of U-bend heat transfer tubes, anti-vibration bars (AVB), etc. At their straight section, the heat transfer tubes are horizontally supported by tube support baffle plates. However, at their top U-bend tube section, a large number of heat transfer tubes with different bending radii are arranged hemispherically and are not supported in the horizontal direction, as a result this U-bend tube bundle has a cantilever support structure. Therefore, it is necessary to understand the vibration behavior of the U-bend tube bundle on the moment when an earthquake occurs. In previous research[5], which was targeted at existing domestic SGs and SGs for Advanced pressurized water reactor (APWR) under development at the time (both were square arrayed heat transfer tube SGs), an excitation test in which a test model of the full-scale U-bend tube bundle shown in **Figure 7** was loaded on a large three-dimensional shaking table (6 m x 6 m) at our Research & Innovation Center (Takasago District) was performed, and data such as the natural frequency and etc. of the U-bend tube bundle were acquired. In addition, as a practical analysis method based on the comparative verification with the results of this test, a conservative stress evaluation method using linear dynamic analysis (the response spectrum analysis, etc.) with a simple model that integrates multiple heat transfer tubes was established and standardized[3].

However, as the ground motion for seismic resistance design increases, it is necessary to evaluate the margin more accurately. A measure for this includes a method considering the non-linear behavior caused by the contact condition with AVB and other components, which the U-bend tube bundle has due to its structure and the elastic-plastic characteristics of the material. In order to use analysis considering the non-linearity, it is necessary to simulate the actual structure of the U-bend tube bundle in detail.

Therefore, we developed an enhanced seismic resistance evaluation method for SG heat transfer tubes and established a method that can accurately evaluate the stress distribution of the heat transfer tubes as a part of “The Safety Improvement of Nuclear Facilities”. The use of this method made it possible to confirm that existing plant SGs have higher seismic resistance. It also made it possible to develop and design highly-seismic resistant future SGs using enhanced evaluation methods such as examining the adoption of an AVB structure that is effective in reducing stress for the maximum stress generating area by understanding the detailed stress distribution of the heat transfer tubes. An overview of the enhanced seismic evaluation method is described below.

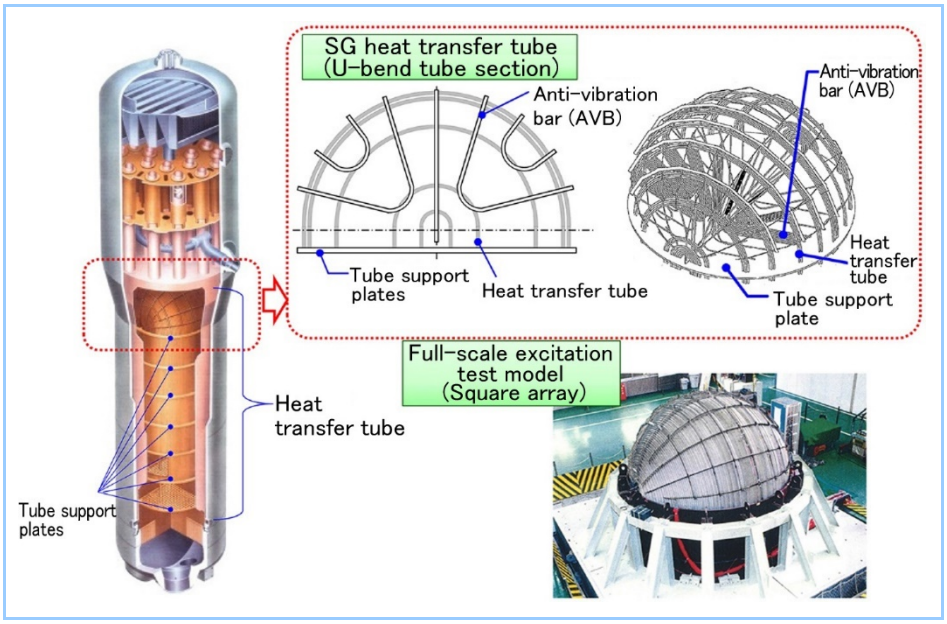


Figure 7 Structure and full-scale test model of steam generator heat transfer tube bundle

3.2 Excitation test

In the development of the enhanced seismic resistance evaluation method, using the vibration characteristics of the U-bend tube bundle acquired in the excitation test as verification data, the seismic resistance evaluation was enhanced by using an all components model of the heat transfer tubes, which modeled all the structures of the U-bend area in detail. As mentioned above, the vibration characteristics of the U-bend tube bundle had been obtained through the excitation test only for the square array. However, in order to apply the enhanced evaluation method to a wider variety of U-bend tube bundle, vibration characteristic data of the triangular array U-bend tube bundle was newly acquired through an excitation test. The test was carried out using a full-scale triangular array U-bend tube bundle test model loaded on a large three-dimensional shaking table at our Research & Innovation Center (Takasago District), as in the previous square array test.

The excitation conditions were targeted at the design basis seismic ground motion Ss, which are the seismic resistance design conditions for actual nuclear plants. An excitation wave (band excitation wave) set so that the test model resonates at the natural frequency was used in order to obtain the maximum excitation force. With this excitation wave, an excitation level, which exceeds the design basis seismic ground motion Ss of the existing nuclear power plant, was attained.

Figure 8 presents the excitation test model and the excitation test results of the triangular array heat transfer tube bundle test. In this excitation test, the damping ratio, which is an input parameter for the seismic resistance evaluation of a triangular heat transfer tube bundle, and vibration characteristic data to be used for verification of the enhanced seismic evaluation method, such as the natural frequency of the entire tube bundle and the response stress distribution of each heat transfer tube, were acquired.

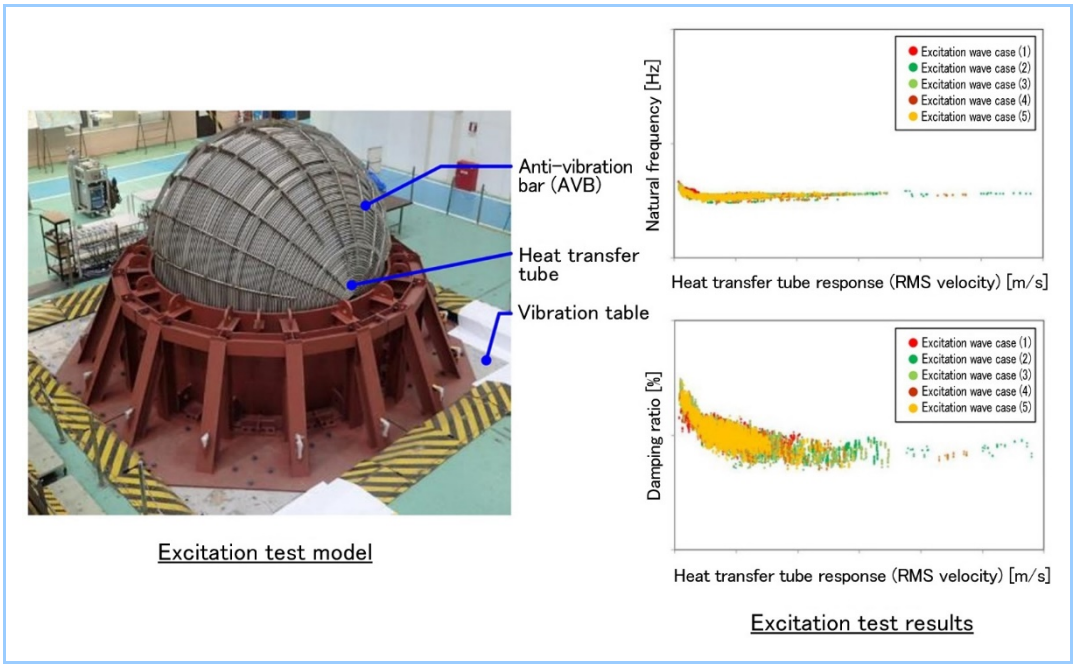


Figure 8 Triangular array U-bend tube bundle excitation test

3.3 Verification of enhanced seismic resistance evaluation method

In enhancing the seismic resistance evaluation method, for the purpose of accurately simulating the behavior of the tube bundle at the moment when an earthquake occurs and accurately simulating the stress distribution of each heat transfer tube, all structural members (several thousand to several tens of thousands) that make up a U-bend tube bundle such as the heat transfer tubes and AVBs, were modeled in detail with individual beam elements, and an all components model of the heat transfer tubes (total number of model nodes: about 1.5 million for square arrays and about 3 million for triangular arrays) that simulated non-linear elements caused by the contact conditions of AVB, etc., was constructed. Using the damping ratio obtained in the past excitation test for this model, the reproduction analysis of the excitation test was performed, and the validity of the enhanced seismic resistance evaluation method was verified by comparing the analysis results with the excitation test results.

Figure 9 and **Figure 10** give the results (stress distribution) of reproduction analyses of the excitation test using the all components model of the heat transfer tubes for the enhanced seismic resistance evaluation. As results of the reproduction analyses, the distribution tendency of the response stress of each heat transfer tube could be accurately reproduced, which verified that this method can evaluate the stress of the heat transfer tubes more accurately than the seismic resistance evaluation using conventional simple models. It was also confirmed that the natural frequency can be reproduced with a high degree of accuracy as shown in **Table 1**.

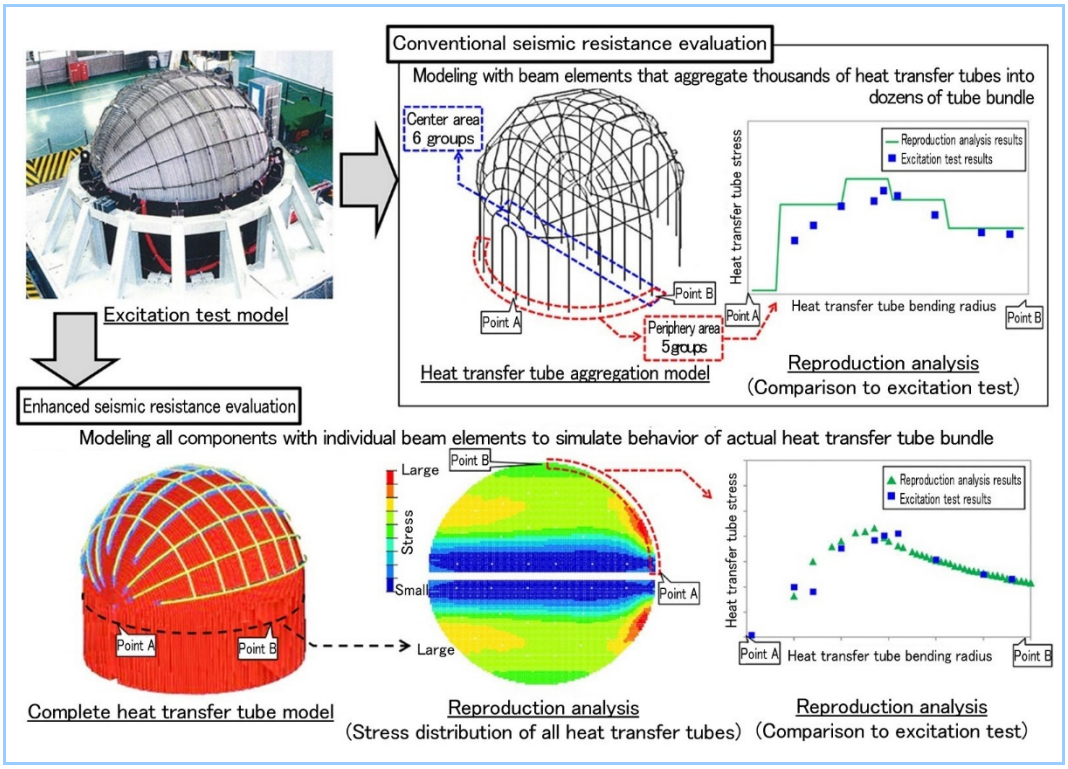


Figure 9 Reproduction results of excitation test using enhanced seismic resistance evaluation model (square array)

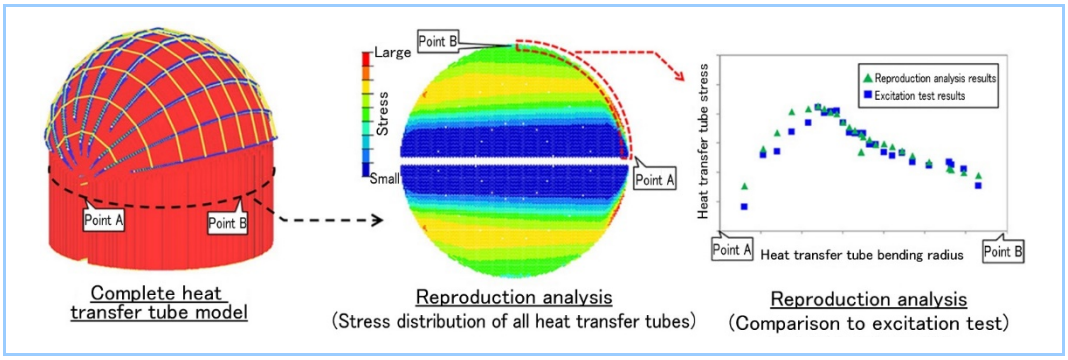
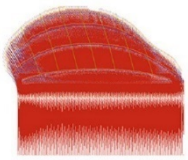
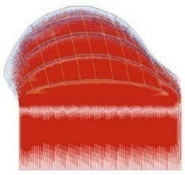


Figure 10 Reproduction result of excitation test using enhanced seismic resistance evaluation model (triangular array)

Table 1 Eigenvalue analysis results (example: primary mode)

	Square array	Triangular array
Eigenvalue analysis results (Natural frequency)	4.7Hz	3.5Hz
(Mode shape)		
Excitation test results	About 4.9Hz	About 3.5Hz

Using the enhanced seismic resistance evaluation method, due to the refinement of rigidity distribution through modeling of all members with individual elements and modeling that simulates non-linear elements, it was discovered that the tendency for the stress to increase locally in the tube bundle is caused by the mechanism where a tube with a relatively small bending radius on the periphery outer area is forcibly deformed due to the vibration displacement of the entire tube bundle. This knowledge made it possible to create a rational tube bundle design, such as reducing local stress peaks by devising a structure and arrangement of U-bend tube bundle components such as the heat transfer tubes and AVB, for example: changing the straight section length of the heat transfer tube, changing the insertion depth of the AVB, etc.

3.4 Efforts for standardization

In order to incorporate the following knowledge obtained from this technological development into the standards[3], we are currently promoting standardization in collaboration with electric power companies and academic societies.

(1) Enhanced seismic resistance evaluation method using the all components model of U-bend tube bundle

(2) Damping constant for designing triangular array heat transfer tube bundle SG

4. Conclusion

We have been increasing the reliability of PWR main components by improving the design and enhancing the seismic resistance evaluation technology. We improved the CI design from the viewpoint of preventive maintenance and improvement of the seismic resistance and realized CI replacement including even in a three-loop plant following work done in two-loop plants. We have also devised a method that can evaluate the stress distribution of a heat transfer tube bundle with a high degree of accuracy through a large-scale excitation test using a full-scale SG heat transfer tube bundle. This method made it possible to accurately evaluate the seismic resistance margin of existing plant SGs. This method is also applied to the seismic resistance evaluation of replacement SGs and new SGs and enables the development and design of SGs with high seismic resistance. We will continue to develop technologies to contribute to the further improvement of the safety and reliability of nuclear power plants.

References

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