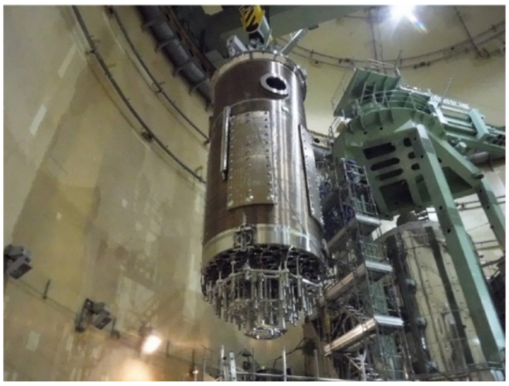
Realization of Advanced After-sales Service for Long-term Operation at Nuclear Power Plants



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The main components that compose the reactor cooling system of a PWR nuclear power plant are vital for safety, and their safety and reliability are guaranteed by implementing periodical inspections, repairs and replacement. Mitsubishi Heavy Industries, Ltd. completed the world's first all-in-one-piece extraction and replacement work of core internals in a 3-loop PWR plant in 2021. In addition, by developing an unprecedented original inspection and cleaning technology of steam generators, investigation and countermeasure of the heat transfer tube thinning event became possible. As a comprehensive nuclear plant manufacturer, we will continue to undertake initiatives to develop technologies and contribute to the further improvement of safety and reliability of plants as well as the long-term operation.

1. Introduction

With a view to extending the service life of a plant that conforms to the latest regulatory standards, Mitsubishi Heavy Industries, Ltd. (hereinafter referred to as MHI) has worked on providing after-sales service with various inspections, repairs and replacement based on our previous experiences, thereby demonstrating that safe and reliable plant operation can be continued. The main components that compose a Pressurized Water Reactor (PWR) cooling system, such as core internals (hereinafter referred to as CI) and a steam generator (hereinafter referred to as SG), are vital for safety. Therefore, we have undertaken continuous initiatives to develop innovative technologies and study work methods by combining MHI's technologies that have been cultivated so far and accomplished such challenging work for the first time in a PWR.

This report introduces two representative examples, "Accomplishment of the world's first replacement work of core internals of a 3-loop[*1] PWR plant" and "Development of SG internal inspection and high-pressure water cleaning jigs."

*1 The reactor cooling system of PWR consists mainly of SG, reactor coolant pump and pressurizer around the reactor vessel. These main components of the reactor system are connected by the reactor coolant tube and form a loop.

2. Accomplishment of world's first replacement work of 3-loop PWR core internals

2.1 Outline of efforts

CI, which are the main components of a PWR, have the functions of supporting the fuel assembly, forming a flow path of the reactor cooling water, guiding and supporting the control rods and in-core instrumentation and reducing the amount of irradiation to the reactor vessel (hereinafter referred to as RV) and are important components that consist of lower core internals (hereinafter referred to as LCI) and upper core internals (hereinafter referred to as UCI). In order to address the increased number of control rods needed for the use of high burn-up fuel and to perform preventive

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maintenance for case of damage like those reported overseas, of bolts (baffle former bolts) that make up CI (described in (a) and (b) below), MHI developed an integrated replacement method*2 for CI of a PWR plant to replace with a new CI having the design improvements described in Section 2.3. (1)

(a) A measure to address the increased number of control rods to provide a sufficient reactor shutdown margin, which is needed for the adoption of high burn-up fuel containing an increased concentration of uranium-235 (which easily fissions) for long-term use

(b) Preventive maintenance for irradiation-induced stress corrosion cracking (IASCC)*3 which has been reported overseas as a cause of damage to baffle former bolts

*2 Integrated replacement method: A method in which the old LCI and the old UCI are taken out from the RV in one piece without dividing, stored in a shielded container, and the new LCI and the new UCI are installed respectively.

*3 Irradiation Induced Stress Corrosion Cracking (IASCC): An event in which stress corrosion cracking occurs due to the action of tensile stress upon high neutron irradiation.

Since the completion of CIR by the integrated replacement method of CI in 2-loop plants for the first time in 2004, construction results have been obtained in total 4 plants (Three of the four plants were replaced in one unit, and the other one was replaced by the split replacement method (Method of taking out old LCI and old UCI individually and installing new LCI and new UCI individually).). In order to overcome the technical problems of the 3-loop plant CIR shown in **Table 1**, we developed an integrated replacement method for the 3-loop plant based on the experience of the 2-loop plant, and carried out a careful construction plan. As a result, the 3-loop plant CIR was completed at the Mihama Power Plant Unit 3 of Kansai Electric Power Co., Ltd. in 2021. (2), (3).

Section 2.2 introduces the removal of the old CI of the 3-loop plant, and Section 2.4 introduces the technology and method developed for the installation of the new CI.

Table 1 Technical problems of CIR in 3-loop plant (compared with CIR in 2-loop plant)

Problem (1)	The weight and dimensions of the shielding container for old CI, which is highly radioactive, are increased. (Weight of the shielding container including the old CI: Increased from about 450 tons to about 600 tons, External diameter: Increased from about 3.8m to about 4.5m)
Problem (2)	Increase in the size of the radial support key due to the improvement of earthquake resistance, and increase in the number of locations to be measured and adjusted for the installation of the new LCI due to the increase in the size of the 3-loop plant

2.2 Removal of old reactor inner structure

The integrated replacement method for removing old CI is a method by which old LCI and old UCI are integrally removed from the RV without being split and stored in the shielding container, allowing the reduction of radiation exposure and a shorter period of operation; this is different to the conventional method by which old LCI and old UCI are individually removed from the RV. In the old CI removal operation, the old CI was removed from the RV by the integrated replacement method, carried out from the reactor containment vessel (hereinafter referred to as CV) through the cylindrical sideways equipment hatch (hereinafter referred to as E/H) and transported to the storage warehouse in the plant. We also designed and manufactured the shielding container for storing old CI which was needed for the realization of this method. The details of the major developments are as follows:

(a) Shielding container for old CI: The shielding container for storing and transporting old CI of a 3-loop PWR, which is about 20% larger than that of a 2-loop PWR, has a structure having both a high shielding function (with the wall thickness that keeps the container's surface dose at 2 mSv/h or less) and good handleability in the narrow CV and outdoors through the adoption of a chill tank on the temporary carrying-out equipment which enables safe and efficient transportation of a superheavy load on the low floor.

(b) Carrying-out equipment: When the shielding container (600 tons) for old CI which is an unprecedentedly superheavy load is carried from the CV outdoors through the E/H, it needs to pass through a gap of at least 100 mm. Therefore, we developed dedicated temporary carrying-out equipment.

(c) Special crane in CV: The existing polar crane in the CV cannot lift and place sideways the shielding container weighing 600 tons. Therefore, we developed a special crane, which works even in the limited space in the CV, in cooperation with an overseas company (**Figure 1**).

Prior to the on-site operation, we conducted training for workers in an environment simulating the actual environment in our factory. As a result, with the improved skills of workers, the on-site operation could be efficiently promoted and the actual exposed dose (dose to which workers were actually exposed) in the whole operation could be reduced to about 50% of the planned value.

Figure 1 Shielding container for old CI overturned by special crane in CV

2.3 Design improvements of new CI

The new CI is based on the improved standard 3-loop plant. It has design improvements, mainly for preventive maintenance for damage of baffle former bolts and improvement in the seismic resistance margin, compared to the old CI, as shown in **Figure 2** and below.

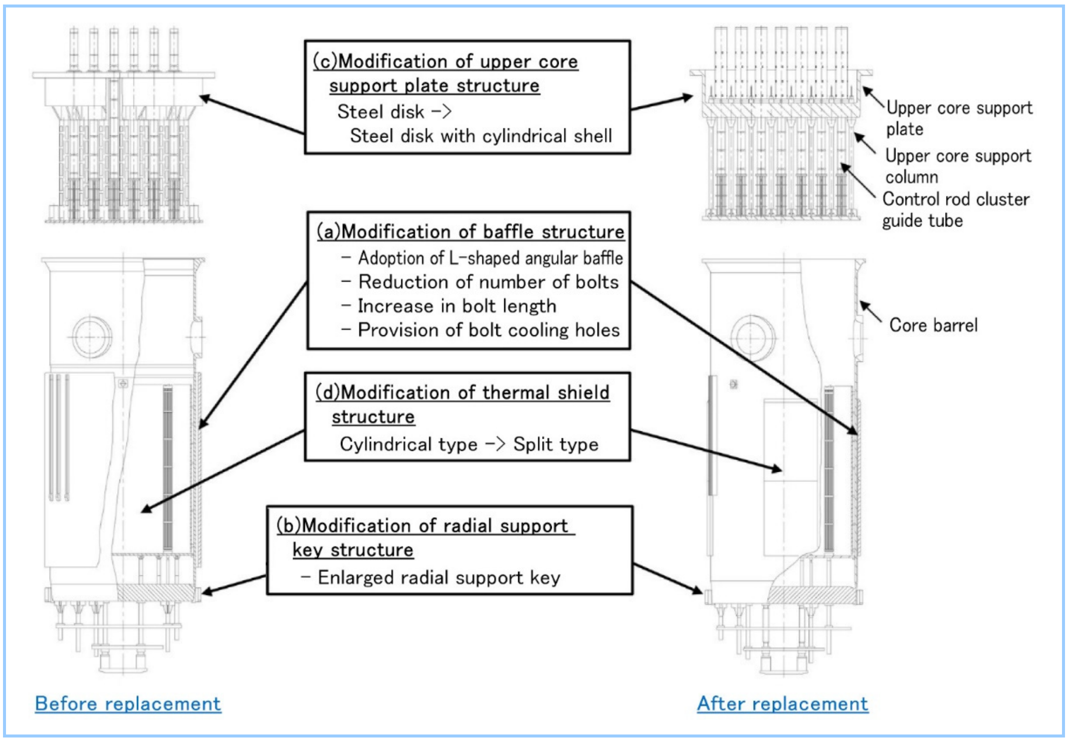

Figure 2 Major design improvements of new CI compared to old CI

(a) Modification of baffle structure
For the new CI, an L-shaped baffle (integrated type) instead of the conventional split-shaped core baffle was adopted to increase the rigidity and reduce the stress on baffle former bolts through an increase in length. In addition, bolt cooling holes were provided to reduce the environmental temperature as a preventive measure against irradiation-assisted stress corrosion cracking.

(b) Modification of radial support key structure
Compared to the conventional radial support key, the radial support key of the new CI increased in size and the number and diameter of the bolts used for installation also increased so that the seismic resistance was improved.

(c) Modification of upper core support plate structure
The steel disk was changed to a steel disk with a cylindrical shell to reduce the length of the upper core support column and control rod cluster guide tube to improve the resistance to cross-flow fluid load and seismic load.

(d) Modification of thermal shield structure
The cylindrical type shield, which coats the entire core barrel, was modified to a split type shield, which coats only the vicinity of the direction of the maximum neutron flux, to improve manufacturability.

2.4 Installation of new internal structure of the reactor

In the new CI installation, the new LCI has a structure to engage with RV by radial support keys installed in 4 directions at the lowest part, and precise clearance management (0.5 mm or less) between RV and new LCI was realized by machining the radial support keys considering as build dimensions of RV side. Specifically, since the inside of RV is a highly radioactive environment and it is difficult for workers to enter the area, the new LCI with the developed submersible position measuring device being installed was provisionally hung in the RV, the new LCI was positioned (with gap adjustment) so that a flow path of the primary cooling water can be formed, and then, the as-built dimensions on the RV side were measured. Based on the obtained measurement result, the radial support keys were processed for adjustment to the precise dimensions and then installed in the new LCI at the site. After that, the new LCI with the radial support keys installed was inserted and hung in the RV. In this way, the installation with a high level of difficulty to satisfy the strict gap requirement of 0.5 mm or less was realized (**Figure 3**).

Through the verification tests and training conducted before the on-site work at the MHI's Integrated Maintenance Training Centre shown in **Figure 4**, we developed a submersible position measuring device and established gap adjustment technology. With the improved skills of workers, we were able to efficiently promote the on-site work.

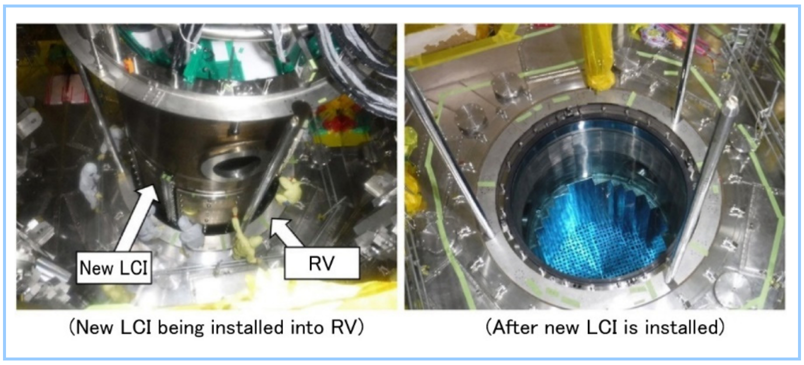

Figure 3 Installation of new LCI

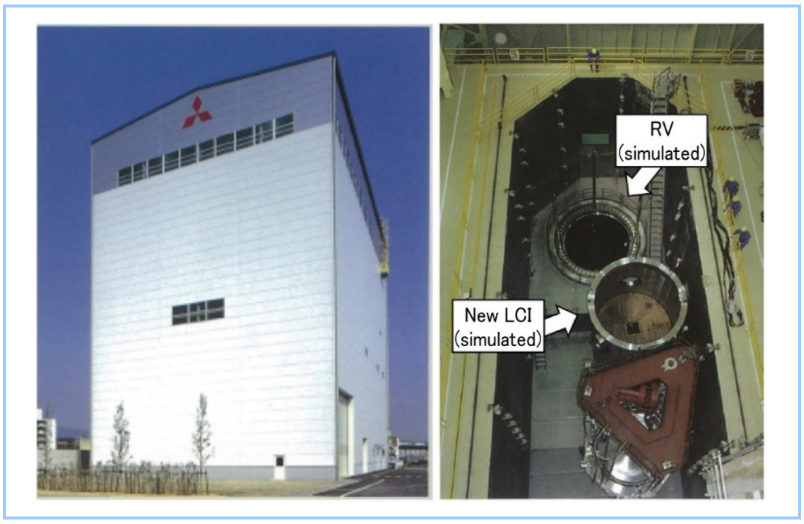

Figure 4 Integrated Maintenance Training Centre and simulation facilities in Centre

2.5 Results of efforts

CIR of Mihama Power Plant Unit 3 of Kansai Electric Power Co., Ltd. was completed in 2021 by the promotion of careful construction plan based on the development of the old CI integral replacement method and the new CI installation technology in the 3-loop plant and by the field simulation training.

We will also safely complete the CIR being planned at Takahama Nuclear Power Station Units 1 and 2 operated by Kansai Electric Power Co., Inc. by utilizing the comprehensive technological capabilities that we have developed as a nuclear power plant manufacturer, and contribute to the safety improvement and stable operation of nuclear power plants.

3. Development of steam generator internal inspection and high-pressure water cleaning jigs

3.1 Outline of efforts

An SG, which is a main component of a PWR, is a vertical U-shaped tube heat exchanger. Thermal energy generated in the reactor is transferred from the reactor cooling water (water containing radioactive materials) flowing inside the heat transfer tubes in the SG via the heat transfer tubes to the secondary cooling water (water not containing) flowing outside, and the secondary cooling water turns into steam, which is transferred to a turbine.

An SG has a large number of heat transfer tubes inside and they are inspected by the Eddy Current Testing (hereinafter referred to as ECT) for any damage such as cracking or wall thinning (partial reduction of wall thickness) of heat transfer tubes in the Periodic Operator's Inspection of a nuclear power plant (inspection of Specific Electric Facilities based on Article 55 (1) of the Electricity Business Act) which is conducted almost in every one year.

The ECT result showed a signal indicating that the heat transfer tube walls were thinned from the outside (secondary cooling water side). Therefore, we have promoted the development of an SG internal inspection jig and a high-pressure water cleaning jig and the introduction of them into the actual equipment to investigate the cause of the wall thinning of the heat transfer tubes and take appropriate measures. This chapter describes those initiatives. **Figure 5** shows the overview of the main cooling systems in a PWR, **Figure 6** shows the overview of an SG and **Figure 7** shows the overview of heat transfer tubes in the SG.

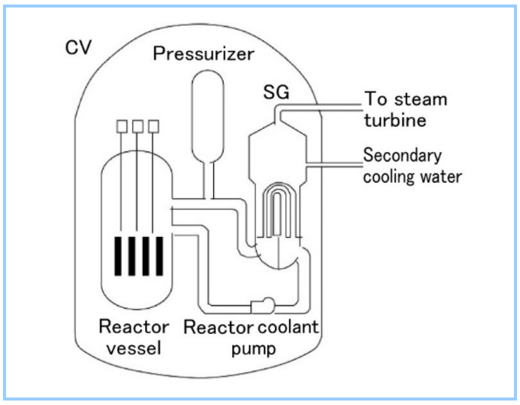

Figure 5 Overview of main cooling system of reactor cooling system of PWR

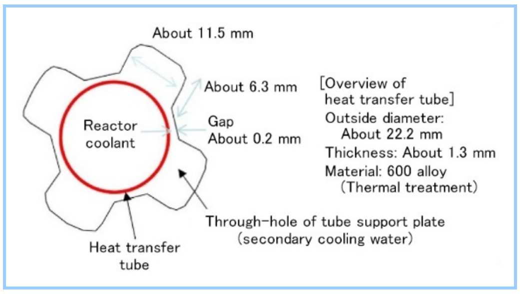

Figure 7 Overview of heat transfer tubes in steam generator

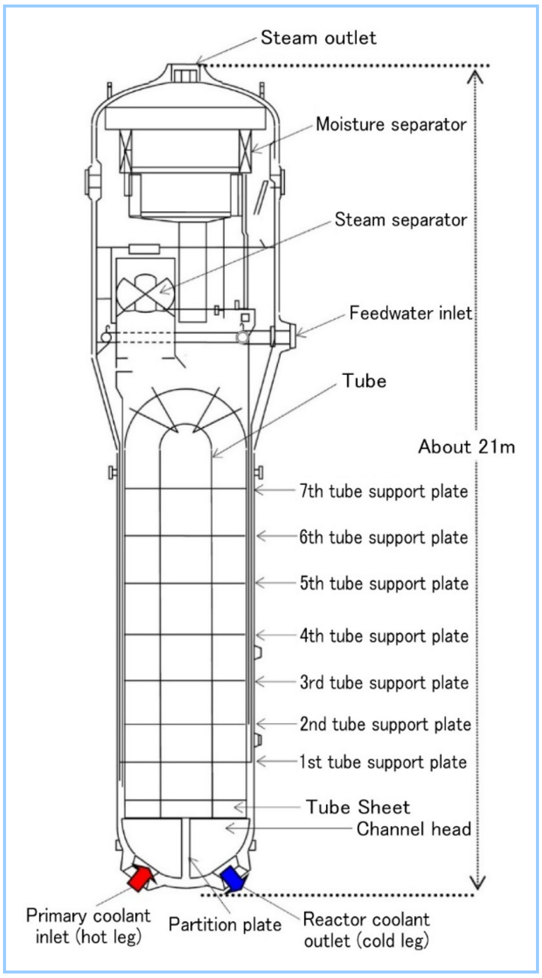

Figure 6 Overview of steam generator

3.2 Measures for wall thinning of SG heat transfer tubes

To investigate the cause of the wall thinning of the heat transfer tubes, we focused on a large amount of scale remaining inside of an SG (which was produced from fine particles of iron attached to heat transfer tube walls). We observed the state of scale remaining inside the SG by a small camera, and we also collected the scale from the inside of the SG and investigated its shape and properties. As a result, it was found that the thinning starting from the external surfaces of the heat transfer tubes was caused by abrasion as follows: dense scale (scale with low porosity in the scale produced from fine particles attached to the heat transfer tubes) that was produced on the heat transfer tube surfaces during plant operation came off and the scale was entrained in the flow of water in the SG, accumulated on the underside of the tube support plate and repeatedly came into contact with the heat transfer tubes, causing the wall thinning. **Figure 8** shows the mechanism of thinning on the underside of the tube support plate in the SG.

To prevent production of dense scale which was identified as the cause of the wall thinning, we removed scale by cleaning the tube support plates with high-pressure water using the high-pressure water cleaning jig and conducted chemical cleaning to embrittle scale. In addition, mechanical plugs (closure plugs) were inserted in the heat transfer tubes having thinned walls so that they could not be used.

Figure 9 shows the image of high-pressure water cleaning using the high-pressure water cleaning jig, **Figure 10** shows the image of chemical cleaning and **Figure 11** shows the image of the use of a closure plug.

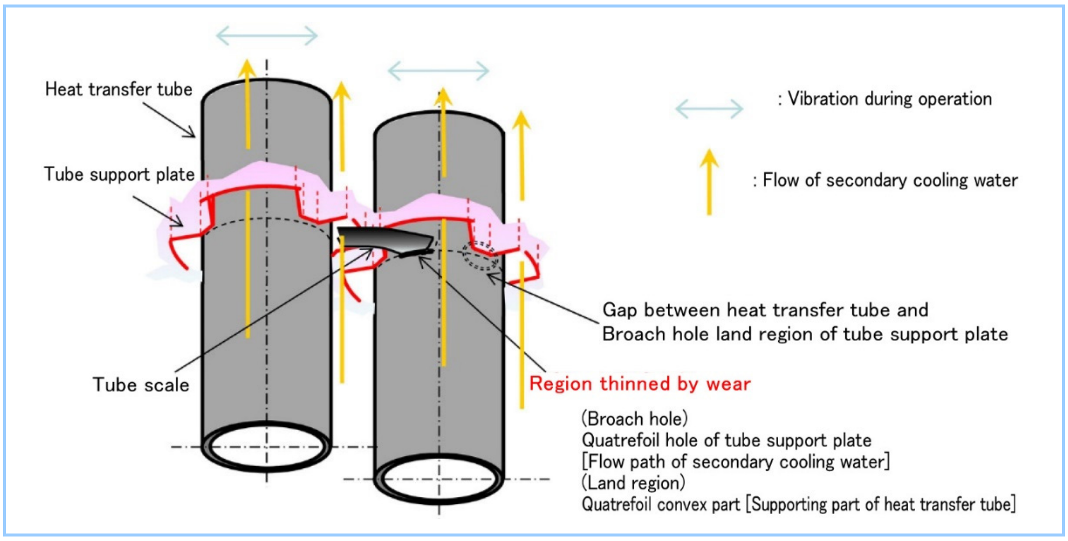

Figure 8　Mechanism of wall thinning on underside of tube support plate in SG

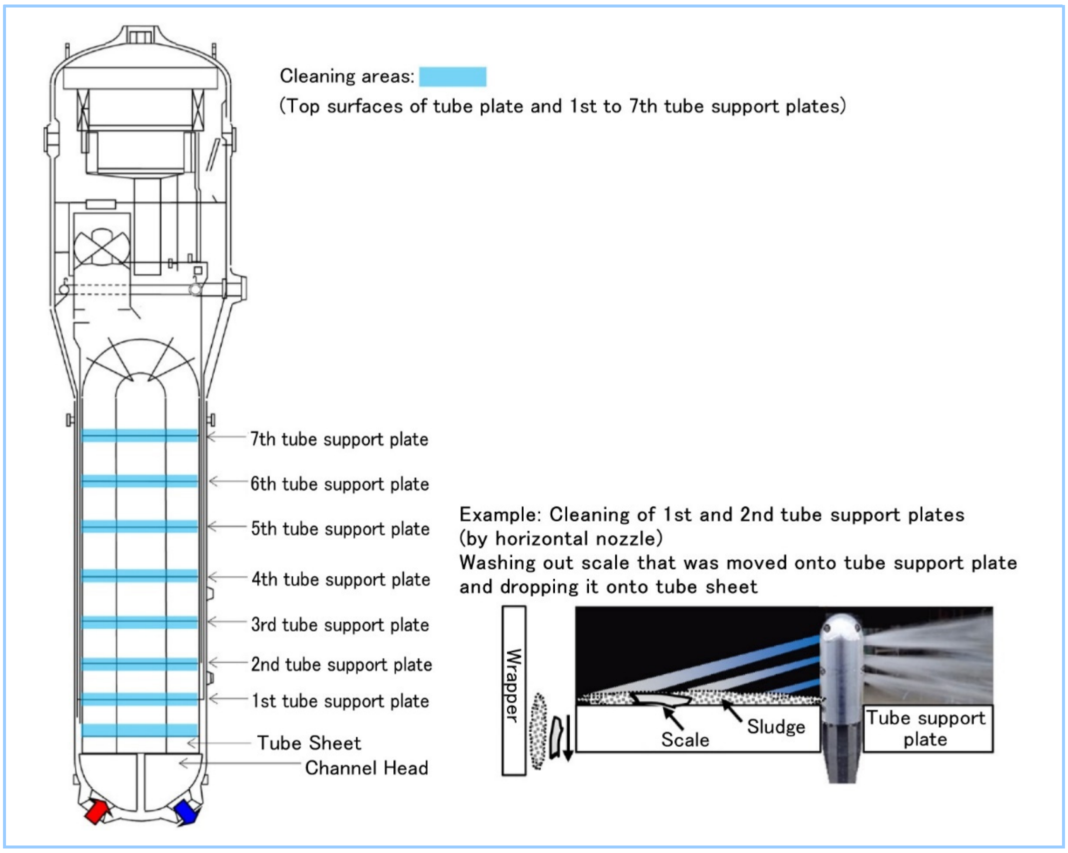

Figure 9　Image of high-pressure water cleaning using high-pressure water cleaning jig

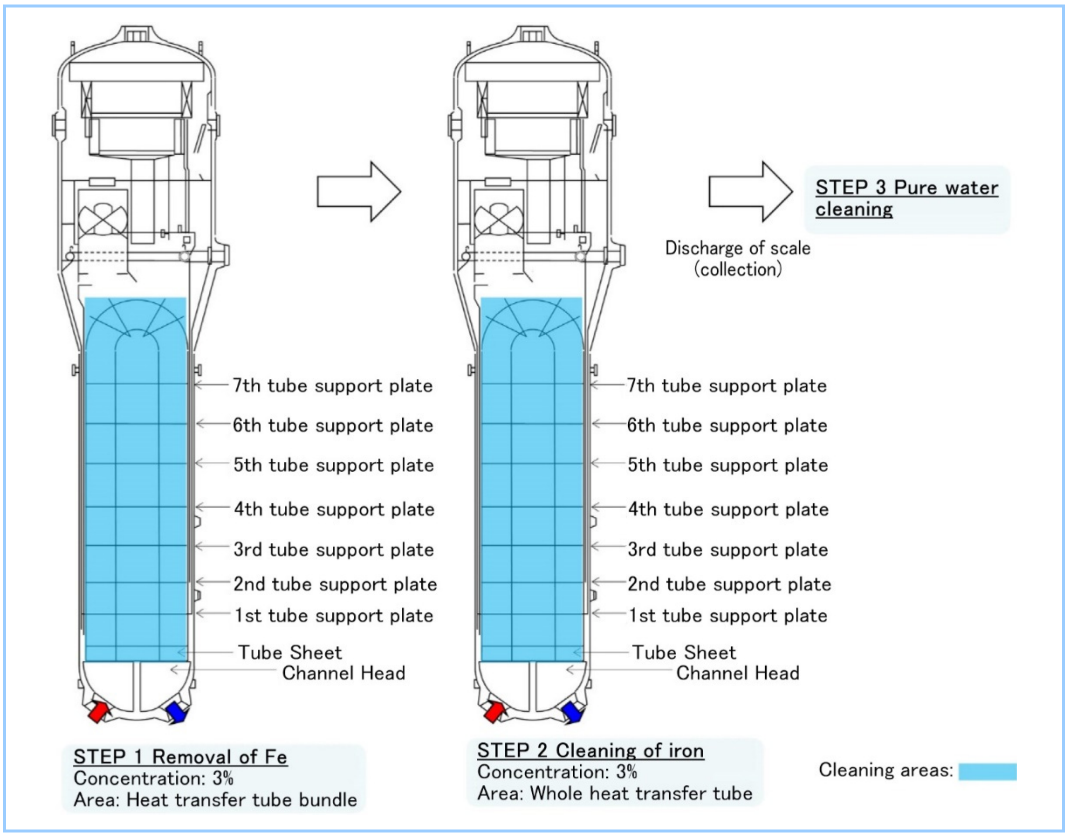

Figure 10 Image of chemical cleaning

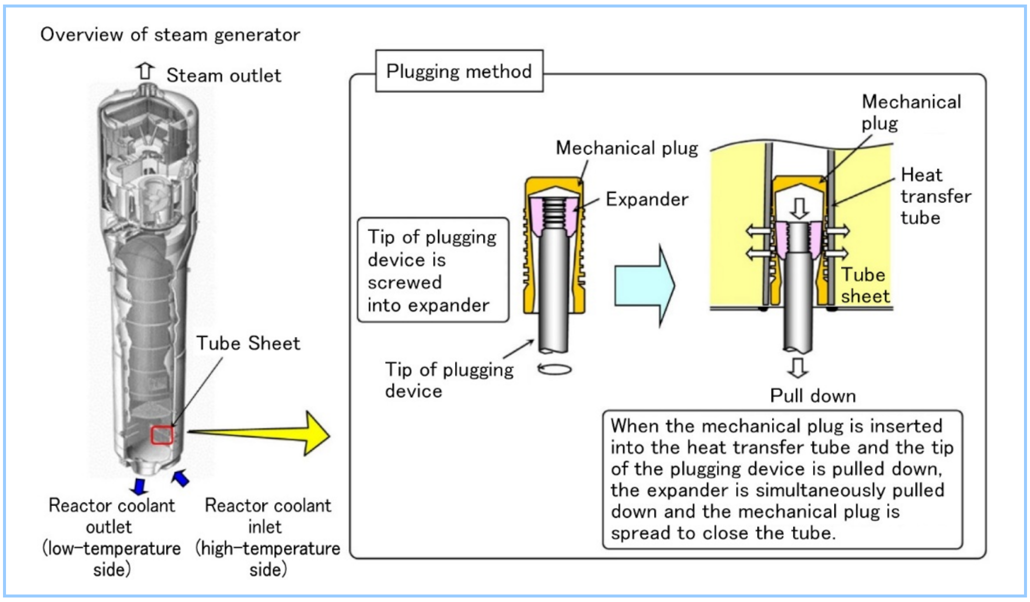

Figure 11 Image of mechanical plug (closure plug)

3.3 Inspection jig in steam generator and high-pressure water cleaning jig

We developed an SG internal inspection jig and a high-pressure water cleaning jig to investigate and prevent wall thinning of heat transfer tubes. The inside of an SG is complex in structure. Therefore, the important point of the inspection and high-pressure water cleaning jigs is that they are compact and flexible so that they can be inserted deeply into the SG. The jigs in multiple shape patterns are available so that an appropriate one can be used depending on the access route to the inspection area (SG tube plate and the 1st to 7th tube support plates), allowing inspection and high-pressure water cleaning of the whole inside area of the SG. The examples of the jigs are described below.

(a) SG internal inspection jig

The SG internal inspection jig is used to take photos of the heat transfer tube walls that are shown to have been thinned and visually check the state (distribution) of scale remaining in an SG. The SG internal inspection jig consists of a small camera (φ6 mm) for inspecting the heat transfer tubes and a guide jig composed of a flexible cable bear and a guide tray for guiding the inspection jig to the target tube support plate. One example of the SG internal inspection jig is shown in **Figure 12**. The SG internal inspection jig is inserted through the hand hole above the 7th tube support plate and moved down to the tube support plate to be inspected with the cable bear being passed through the flow slot. After that, the small camera installed in the guide jig is moved through the gap (about 10 mm) between the heat transfer tubes to check the state of scale remaining in the SG. A balloon for fixing the jig is provided at the lower end of the guide jig to prevent displacement.

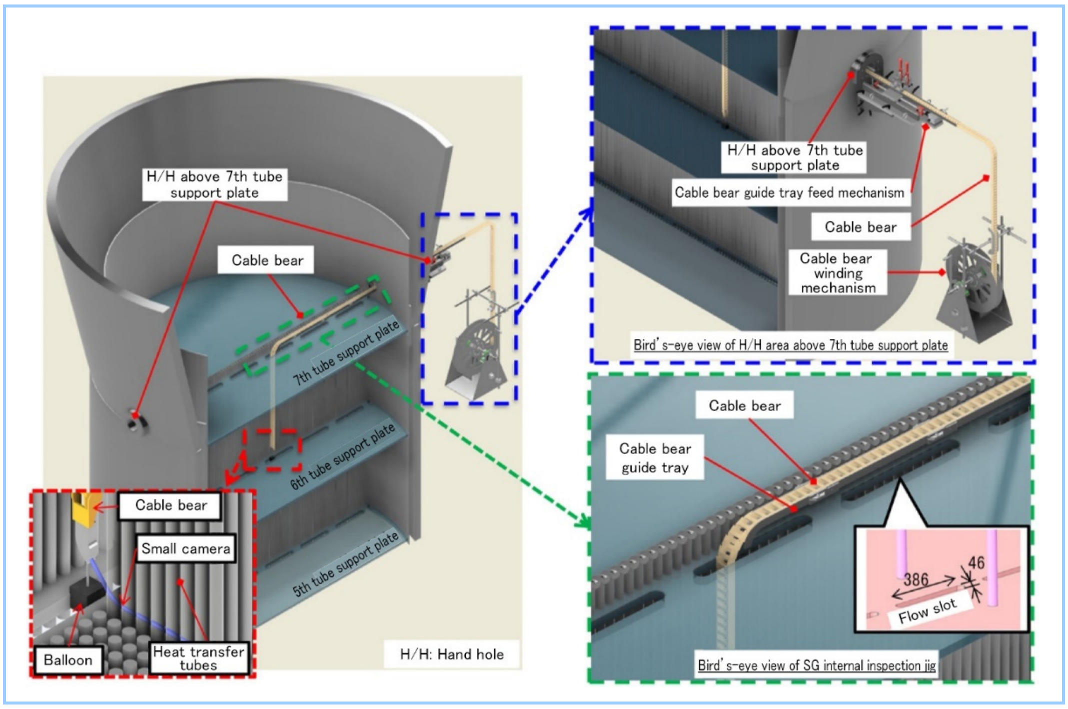

Figure 12 Image of SG internal inspection jig

(b) High-pressure water cleaning jig

The high-pressure water cleaning jig is used to remove scale remaining inside of an SG. The high-pressure water cleaning jig consists of a guide jig for guiding the cleaning jig to the target tube support plate, a cleaning nozzle for removing scale and a small camera for identifying the cleaning position. **Figure 13** shows an image of the high-pressure water cleaning jig. The target areas for high-pressure water cleaning are the top surfaces of the tube plate and 1st to 7th tube support plates. Cleaning starts from the 7th tube support plate of the upper layer and scale drops onto the tube support plates of the lower layer and the tube plate. After the top surface of the tube plate is finally cleaned, scale is collected through the hand hole above the tube plate. The cleaning jig is inserted in the SG through the hand holes of the upper part (near the 7th tube support plate) of the SG or the lower part (near the 1st tube support plate and tube plate) and the flexible guide jig is moved to the cleaning position in the same way as (a). Vertical nozzle-type and horizontal nozzle-type high-pressure water cleaning jigs are available. An appropriate jig can be used depending on the target cleaning position to improve the cleaning efficiency. The vertical nozzle-type high-pressure jets high-pressure water in the vertical direction, thereby cleaning the gaps between the tube support plates and the heat transfer tubes and dropping scale onto the tube support plates. The horizontal nozzle-type high-pressure water cleaning jig washes down the scale dropped onto the tube support plates into the gaps between the outer periphery of the tube support plates and the inner face of the outer cylinder of tubes and drops it onto the tube support plates.

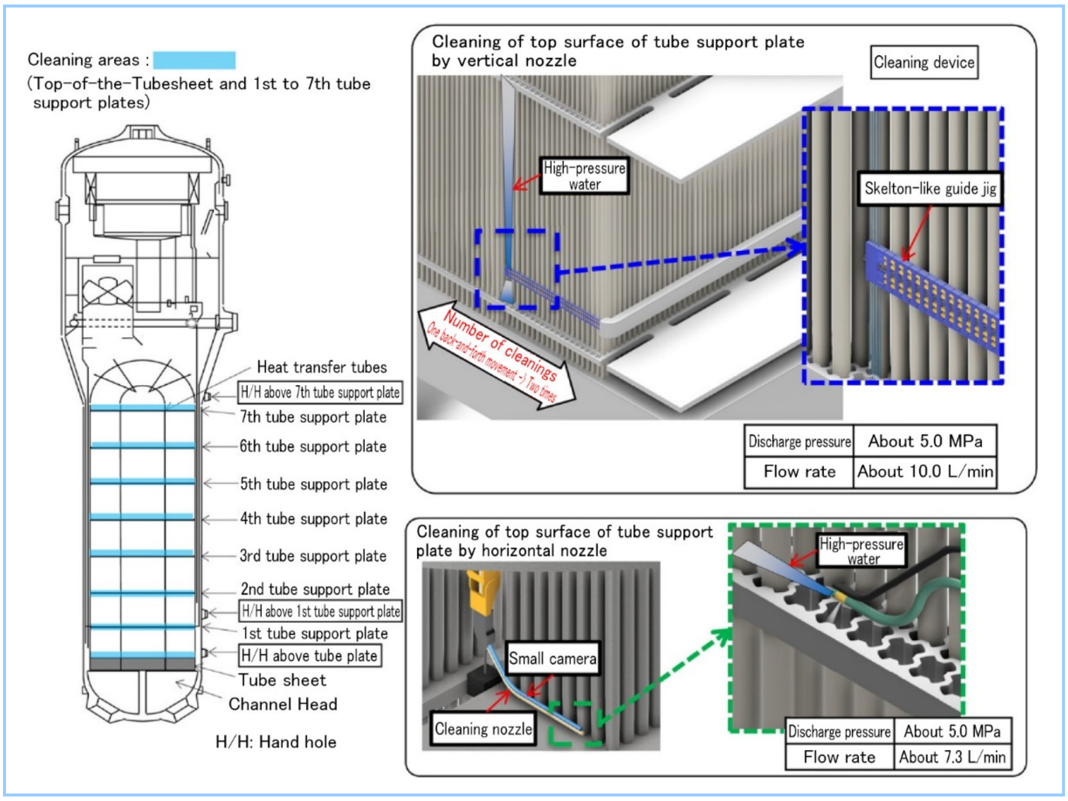

Figure 13 Image of high-pressure water cleaning jig

3.4 Results of efforts

We achieved the unprecedented development of inspection and high-pressure cleaning jigs for the inside of an SG which has a complex structure to take measures against the wall thinning of the heat transfer tubes of the SG in a PWR plant. In addition, we introduced the jigs in the actual equipment to investigate the cause of the wall thinning of heat transfer tubes and take measures against it and contributed to the improvement of safety and reliability of nuclear plants.

4. Conclusion

In the completion of the CIR of a 3-loop plant and development of the SG internal inspection and cleaning technologies that were introduced in this report, we could demonstrate their applicability in the actual nuclear power plant. As a nuclear power plant manufacturer, we have accumulated experience through continuous work on the main components of the reactor cooling system and could demonstrate our technological capabilities in after-sales service for long-term operation.

In the future, we will undertake initiatives to continue technological development to further contribute to the improvement of safety and reliability of nuclear power plants and their long-term operation.

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