Structural Design of Reactor Components in PWR and APWR

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The basis of the structural design of an advanced PWR reactor and the factors considered during the design process are summarized. These factors include the functional and design requirements for the reactor components, the design improvements of the materials and structure reflecting experience gained from existing PWR operations. With the advanced PWR reactor structure, the demand for a larger core and higher fuel burnup has resulted in a larger sized core support structure and larger diameter of the reactor vessel. We adopted a neutron reflector for higher economical efficiency and improved the reliability of the reactor by simplifying its structure and reducing the neutron irradiation. This paper describes the design objectives, features and design validation of an advanced PWR reactor structure, as well as transitions in the design of existing PWR reactor internals.

KEYWORDS: Advanced PWR, APWR, Fuel assembly, Reactor Internals, Control Rod, Neutron, Reflector, CRDM,

I. Introduction

Adopting technologies from the United States was the first step in constructing Pressured Water Reactors (PWR) for nuclear power in Japan. The Japanese government, electric power companies and manufactures cooperated in standardizing changes in third generation nuclear reactor facilities. In order to develop Advanced PWR(APWR), we have learned the most recent design technologies and experience from operating existing types of PWRs and have worked to create better safety, reliability and functions based on successful results. In addition, improvements were made in cost and reliability by using Neutron Reflector. Rod Cluster Control, the number of reactor internals and positioning were upgraded, and these upgrades were applied to the change in the structure of reactor components after operations were started.

II. Technical Standards of Structural Design for Nuclear

The technical standards that apply to the structural design of nuclear reactors in Japan are the Technical Standards of Nuclear Equipment: Ministry of Economy: Trade and Industry Notice 501 (METI Notice 501), and the regulatory guide for aseismic design of nuclear power reactor facilities and aseismic design technical guideline for nuclear power plant: JEAG4601. The technical criteria on the standardization of improving second-generation light water reactors were revised. The strength of core support internals structure and the earthquake preparedness of the design for reactor internals had to be evaluated, and a more detailed design evaluation was required to get approval for construction.

The technical standards in Japan reflect the progress of technical standards for the American Society of Mechanical Engineers (ASME) in the United States. Corresponding to the changes in standards in the United States, structural changes for the reactor components of PWR plant were made in United State of America and Japan. By using experience obtained from current operations, the structure can be strengthened and simplified and changes can be made in the materials and construction.

In addition, the standard evaluation of maintenance in operation is based on validity. Regular security checks and evaluations of long term reliability have also been carried out.
III. Functional Requirements and Structural Design

Changes in existing PWR design are demonstrated based on the structure of nuclear reactors composed of the main components indicated below. Basic requirements for function and design, and requirements for the structural design are also described.

- Fuel Assembly: FA
- Rod Cluster Control: RCC
- Reactor Internals: RI
- Reactor Vessel: RV
- Control Rod Drive Mechanism: CRDM

(1) FUEL ASSEMBLY

The reactor core consists of a specified number of fuel rods that are held in bundles by spacer grids and top and bottom nozzles. Fuel assemblies are arranged in predetermined square matrix patterns Figure 2-1. The standard fuel assembly design used in operating plants is the 17 x 17 lattice RCC fuel assembly. Previous plant designs used the 14 x 14 array in two-loop plants, or the 15 x 15 array in three-loop plants.

To begin with a description of a standard fuel assembly, it is best to mention the functional requirements of the assembly in the following:

- Provides and maintains fuel geometry and position
- Provides for handling, shipping and loading of the fuel
- Isolates fuel and fission gases from the coolant
- Allows for axial expansion of the fuel rods and the fuel assembly
- Resists hydraulic forces
- Facilitates reprocessing
- Withstands chemical and radiation material damage
  (Also can satisfy nuclear thermal and hydraulic requirements)
- Minimizes neutron loss
- Directs coolant flow
- Provides for coolant flow and heat transfer
- Provides for control of the fission process
- Optimizes drop in pressure
- Monitors the fission process and temperature

The 17 x 17 fuel assembly consists of 264 fuel rods, 24 guide thimble tubes, and 1 instrumentation thimble tube. The instrumentation thimble is positioned in the center of the fuel assembly to provide a location for an in-core neutron detector if the fuel assembly is located in an instrumented core position. Figure 2-1 shows a cross section of the fuel assembly array, and Figure 2-2 shows outline of the 17 x 17 fuel assembly.

The core must be designed to economically and reliably extract the energy from the fuel. To meet this objective, the assembly design must fill the exacting requirements of nuclear physics, thermal-hydraulic design, mechanical integrity, and correct materials utilization. Necessities often present conflicting requirements that must be delicately balanced by the designer. With this in mind, the fuel assembly is designed to minimize the non-fuel material content of the assembly, thus conserving neutrons and lowering fuel costs. This design also integrates the control rods, referred to as the Rod Cluster Control Assembly (RCCA), within the fuel assembly, thereby leveling power distribution and raising the power level of the reactor.

A fuel assembly consists of a top nozzle, bottom nozzle, guide thimbles, grids and fuel rods. The guide thimbles of the top and bottom nozzles and grids comprise the main support structure. The grids outside the thimble tubes are held rigidly in place by mechanical expansion.
There are nine grids contained in the 12-ft 17 x 17 assembly and seven and eight grids for the 14 x 14 and 15 x 15 assembly that are placed approximately equidistant along the axial plane of the fuel rods.

Only the frictional force between the rod and the grid spring and dimples holds the fuel rods. Each fuel assembly is installed vertically in the reactor vessel and stands upright on the lower core plate. After all fuel assemblies are set in place, the upper support structure is installed. Alignment pins, built into the upper core plate, engage and locate the upper ends of the fuel assemblies. The upper core plate then bears downward against the fuel assembly top nozzle via the hold-down springs to hold the fuel assemblies in place.

The guide thimbles are structural members that also provide channels for the neutron absorber rods, burnable poison rods or neutron source assemblies.

(2) CONTROL RODS

Control rods, which are located within the fuel assemblies, contain the neutron absorbing material. These rods are positioned in the control rod guide thimbles previously described. The control rods are dispersed throughout a given fuel assembly and are mechanically joined at what is called a spider. This device enables the cluster of rods within the given assembly to be moved as a unit. The resulting array of rods is known as a rod control cluster (RCC) and consists of a group of 16, 20 or 24 individual rods or fingers grouped together on a common drive shaft (see Figure 2-2 for a typical RCC assembly).

Since the RCCs enter the core from above, a trip is created by releasing the clusters from their CRDM and allowing them to fall into the core under gravitational force. The design criteria reflected in thermal-hydraulic considerations of control rods are listed below.

- The time required to drop the control rods after a reactor trip signal is actuated must be within acceptable limits as specified by an analysis of control and protection systems.
- The impact velocity of the RCC is absorbed by a spring located within the hub.
- There must be no boiling within the RCC thimble when the rod is fully inserted.
- The total thimble tube flow must be minimized. Obviously, any flow that is diverted through the guide thimble is not effective for core heat transfer and imposes a penalty on core power.

(3) REACTOR INTERNALS

Reactor internals are designed to provide the following functions.

- Support, orientation and guide the core components, that is, the fuel assemblies and control rod assemblies
- Direct the main coolant flow to and from the fuel assemblies
- Absorb control rod dynamic loads, fuel assembly loads and other loads and transmit these loads to the reactor vessel
- Support instrumentation within the reactor vessel
- Convey cooling water to the core in case of a postulated loss of coolant accident
- Provide protection for the reactor vessel against excessive irradiation exposure from the core
- Position and support reactor vessel irradiation surveillance specimens

The reactor internals consist of two major units, the lower internals assembly and the upper internals assembly.

a. Lower Internals Assembly

The lower internals assembly is shown in Figure 2.3. It consists of the core barrel, core support plate, support columns, lower core plate, core baffles, baffle formers, neutron shield elements, lower instrument guides, fuel alignment pins, lower radial support members, upper core plate guide pins, head and vessel alignment pins, head cooling flow nozzles, secondary core support, hold down spring and other complementing members.

![Figure 2-3 Lower Internals Assembly](imageurl)
conveys these vertical loads to the core barrel cylinder up to where the main support flange grounded out at the reactor vessel flange. The remaining vertical loading enters the barrel at the lower core plate support ledge. The lower core plate is equipped with a pair of precisely located pins at each fuel assembly position. These pins orient the fuel assemblies at the lower core plate and also transversely transmit any side loads arriving at the bottom of the fuel assembly through the core plate to the barrel shell. The lower core plate is perforated by holes at each fuel assembly location to evenly meter the coolant flow to the fuel assemblies across the core.

The core support plate is also perforated by holes to provide the coolant to the lower core plate. The core support plate is a thicker structural component that is designed for very minimal deflection. Thus, it has good vibration characteristics in carrying the fuel assembly weight and fuel assembly hold down spring loads transmit to the barrel shell. The radial support system at the core supports the vessel wall and consists of close fitting key to clevis arrangements. These arrangements allow radial and axial expansion of the internals relive to the vessel without binding, but eliminate any transverse movement of the internals into the vessel.

The core barrel must be equipped to accommodate the fuel assemblies which, though square in section, are arranged in a roughly circular cross sectional pattern. A former and baffle system permits the transition from the round barrel to the squared off periphery of the core. Formers, which are round on the outside, fasten against circular grooves in the barrel. The lower instrument guides are attached to the core support plate and are additionally supported and positioned at the lower core plate and tie plates. These instrument guides provide the passageways and protection from coolant flow for the neutron flux monitor thimbles that terminate in the fuel assemblies after entering the pressure vessel through penetrations in the bottom head.

The tie plates interconnect the instrument guides below the core support plate so that each guide is made more capable of withstanding the cross flow loads and associated vibration inputs imposed by the coolant. Two tie plates are required to accommodate the vessel spherical bottom head and the termination of the bottom head instrument penetrations. The tie plates also support the secondary core support assembly. The secondary core support equipment is in place in the event of a postulated downward vertical displacement of the lower internals or a portion thereof and the reactor core. The secondary core support acts as an energy absorption system.

Neutron shielding pads and radiation surveillance protect the pressure vessel from radiation by using irradiation specimens contained in holders affixed to the core barrel surface at locations of potential high neutron fluence. The neutron shielding pads limit the fluence level on the vessel wall to an acceptable value; in addition, the irradiation specimens are periodically tested to verify that the fluence level is sufficiently low. The vessel head cooling flow nozzles direct low temperature main coolant inlet water up through matching holes in the flange of the upper internals assembly and into the vessel head plenum. This flow cools the head and ultimately passes down through minor openings in the control rod guide tubes and into the main outlet flow stream.

The upper core plate alignment pins, in addition to the lower radial support keys, maintains guidance and positioning of the lower internals assembly. The upper core plate alignment pins, located in the core barrel wall, guide the upper core plate into position and maintain the critical alignment of the upper internals to the lower internals. The head and vessel alignment pins are fastened in the main flange of the core barrel in order to properly position the lower internals to the vessel at the vessel flange, the vessel head to the main vessel cylinder, and the upper internals to the lower internals.

The access holes in the core barrel flange, which provide for the removal of radiation specimens, are plugged during operation by spring-loaded devices that are held down by the upper internals. The large hold down spring is positioned on the core barrel main flange beneath the upper internals main support flange. Its function is to separate the flanges and maintain a positive force resist and prevent any uplift of the lower internals.

b. Upper Internals Assembly

The upper internals assembly consists of the upper support plate, upper core plate, upper support columns, control rod guide tubes, thermocouple columns, thermocouple cross run conduit and their support members, and other complementing components. The upper support plate is a thicker structural element consisting of a flange and a thick plate connected by a cylindrical shell.

Figure 2-4 Upper Internals Assembly

The support plate carries the uplift forces applied to the upper internals to the vessel head with only very minimal deflection so that the alignment and end slope of the guide tubes connected thereto remain acceptable. The upper support plate is the transition for the control rod drive system between the CRDM on the vessel head and the
upper internals assembly which is equipped with large, precisely located holes for positioning and fastening of the control rod guide tubes at their upper ends.

The upper core plate is the lowest major component of the upper internals assembly. It is equipped with a pair of precisely located pins at each fuel assembly position. These pins orient the fuel assemblies at the upper core plate and also transversely transmit any side loads arriving at the top of the fuel assembly through the core plate to slab sided pins in the core barrel.

The load at the slab sided pins from the upper core plate is transferred through a clevis to key arrangement where the clevis is a slot in the core plate lined with wear saddles and the key is the slab-sided pin in the core barrel. This arrangement allows radial expansion and axial movement of the core plate relative to the core barrel. A flow hole for passage of the coolant is presented at each fuel assembly location.

The upper support columns are between the upper core plate and the upper support plate and stiffen the upper internals assembly. They transfer any upward loads acting on the upper core plate directly up to the upper support where they are grounded out at the vessel head flange. The thermocouple hot junction projects below the center of the column just above the upper core plate in most plants.

During a trip, the control rods must be inserted into the core within a design time limit to provide adequate core shutdown. Since 24 rods in a 17 x 17 control rod assembly must be guided simultaneously, the sizing and positioning of the guide ways in the guide tube are critical to the trip and wear characteristics. Inadequate clearance cause binding and increase trip time. Excessive clearance result in unacceptable fretting wear. The guide tube, comprised of a lower and upper structure shield in the upper internals by spring pins at the lower end and by a bolted connection at the juncture between the lower and upper assemblies.

The guide tube assembly is specifically designed for removal and replacement, if required, at any time during the life of a plant. The lower flange, which houses the split pins, is located a small distance above the core plate to facilitate differential expansion.

The lower assembly is divided into two functional areas: the continuous region and the intermittent guidance region. The continuous region, which has controlled hydraulic loading to stabilize the rodlets, provides shrouding of the control rods against core flow. The design of the continuous region is based on hydraulic considerations and on the entry position of the control rods into the fuel assembly. During the insertion of the control rods, the assembly is continuously guided to prevent the bending of rods by cross flow. The intermittent guidance region guides the control rod through discrete guide plates spaced at the required intervals to provide rod stability.

The upper assembly provides shrouding against cross flow and flow restriction for head cooling considerations. Fatigue is the primary structural concern for the guide tube because of the flow-induced vibration, later strength in the bending mode and external and/or internal pressurization during a postulated Loss of Coolant Accident (LOCA). The guide tube is extensively tested for functional and structural integrity in response to wear.

The thermocouple columns are tubular assemblies mounted on the top of the upper support plate that penetrate the vessel head and permit hookup of the electrical connections of the thermocouples. Tubing runs within the thermocouple column convey and support the stainless steel sheathed thermocouples. They are routed into the reactor vessel and run through conduit tubing continuing above the upper support plate and within the upper support columns down to their terminal point above the upper core plate.

During reactor refueling, when thermocouple electrical terminals must be disconnected, the Conoseal joint disassembled, and then a thermocouple column protective sleeve is used to cover the upper area of the thermocouple column assembly. The reactor vessel head is removed and reinstalled.

c. Coolant Flow Paths and Loading Conditions

The main coolant flow enters the vessel through the inlet nozzles and impinges against the upper core barrel (Figure 2-6). This flow then proceeds down the annulus between the core barrel and vessel wall. In passing the neutron pads,
A small fraction of the flow enters the vessel. This flow is generally intended for vessel head cooling. The exit from the head plenum to the upper internals outlet plenum is through openings in the guide tubes above the upper support plate.

Another small fraction of the flow entering the vessel short circuits the by-pass flow directly into the main coolant outlet nozzle stream at the narrow interface gaps between the vessel outlet nozzles and the upper core barrel outlet nozzles.

During a LOCA, major cross flow and pressure differential loadings are applied to components (such as upper support columns and upper support plate) as the coolant exits toward the break.

Some of the other significant loadings not limited to coolant flow are weight loading, preloading of fuel assemblies, control rod dynamic loading and earthquake accelerations.

The weight load of the fuel and the lower internals assembly is ultimately grounded out of the vessel at the core barrel flange. The fuel assembly hold-down springs add a large force to the lower internals core support that is also grounded out at the vessel flange. The control rod dynamic loading in the dashpot at the fuel assembly lower region during the control rod trip is transferred to the lower core support and then up through the core barrel where it is absorbed at the vessel flange.

Earthquake accelerations on the fuel and other components are transmitted through the various supporting features of the internals and are grounded out (as regards the internals) at the vessel flange and the lower radial support system.

Plant load follow transients, for instance, cause changes in baffle length with respect to the core barrel and result in cyclic bending deflection of the baffle supporting bolts. The gamma-induced internal heating of barrel and baffle walls cause a temperature gradient through these elements resulting in thermal stresses.

(4) Reactor Vessel

As nuclear reaction occurs inside in the reactor vessel, the RV has to be operated under an environment of high temperature, high pressure, and high radiation. The RV is composed of a cylindrical barrel with a spherical upper and lower head as shown in Figure 2-7. The pressure vessel for Control Rod Drive Mechanism is welded to the upper RV. In addition, inlet and outlet nozzles are mounted to the barrel of the RV. At the lower barrel, a nozzle is fixed to the bottom of RV for the neutron detector.

The basic functions of the RV are to house the fuel assembly and the core and reactor internals that support fuel assembly, and to form flow paths for coolant water that cools down the core. PWR use a primary coolant for high temperatures and pressurized cooling water. Therefore, the materials and heavy plate must be resistant to high pressure. Low-alloy carbon steel is used as well as stainless steel that clothes the inside of RV for added strength.

For the basic shape of the RV, the core size is first
determined. Then, the inside diameter of the RV is arranged by the reactor internals designer. Finally, the diameter outside the RV is decided to satisfy the required thickness of the RV. The circular space inside the RV and core barrel is called the down-comer. The down-comer functions as a flow path for coolant so that its flow speed must be set up properly. The coolant effectively shield neutrons and reduces the amount of neutron irradiation in the RV, and the width of down-comer can be increased.

The walls of the pressure vessel are penetrated to allow coolant to enter and leave; the ends are penetrated to allow external mounting of the control rod drive mechanisms and to allow access for the in-core instrumentation. In addition, several spare penetrations are provided for possible future modifications.

Steel pads integral with the coolant nozzles support the vessel (Figure 2-8). The pads rest on steel base plates atop a support structure attached to the concrete foundation. Sliding surfaces between the support pads and the base plates accommodate thermal expansion and contraction of the vessel. Side stops on these plates keep the vessel centered and resist lateral loads, including all pipe loads.

The removable upper head of the vessel with a bolting flange employs studs and nuts containing O-seals against the vessel flange hydraulic tensioning of the studs permits uniform nut loading and provides the sealing force. An elongation gauge is employed to assure that uniform loading is obtained around the flange circumference.

The thermal insulation surrounding the mechanism adapters and the flange and studs on the top head are removable. The bottom of the vessel can be accessed by removing the insulation panels. A lifting device is provided for handling the vessel heater lower portion of this device has a platform for access to the control rod drive mechanisms.

A stress analysis of the reactor vessel is a complicated and important design task. The vessel is designed to last the assumed 40-year lifetime of the plant. Stresses in the walls of the vessel result from a combination of the following factors.

- The dead weight of the vessel itself
- The dead weight of the core, internals, coolant, rod drives, and insulation
- The pressure differential across the vessel wall
- Thermal stresses resulting from the temperature differential across the vessel wall
- Reaction forces induced by expansion or contraction of the piping in the reactor coolant loops
- The dynamic loads exerted by the control rods following a trip
- The dynamic loads exerted by the coolant water as it reverses direction at the bottom of the vessel
- Additional loads that might occur during accident or seismic situations

The reactor vessels are designed to be in compliance with the rules for Class 1 Nuclear Pressure Vessels in METI Notice 501 which is similar to the Section III of the ASME Boiler and Pressure Vessel Code.

The core baffle, the core barrel, and the thermal shield attenuate a large portion of the gamma radiation and some of the fast neutron flux leaving the core, thus reducing the gamma heating in the pressure vessel wall.

Gamma and neutron radiation are the principal sources of steady state heating in the walls of the vessel, therefore any reduction in the radiation intensity reduces the associated thermal stress. The passage of coolant water between the vessel wall and the thermal shield cools these structures and aids in the relief of thermal stresses.

Neutron irradiation of steel tends to affect the material in the same manner as. The steel becomes harder and more brittle. It is significant to note that only fast neutrons (un-moderated fission neutrons having an average energy of about 2 Mev) are important, since they are responsible for the cold-working effect. One measure of the embrittling effect of neutron irradiation is the reference nil-ductility transition temperature (\(RT_{NDT}\)). Above this temperature, the mode of fracture of the material is ductile; below it, brittle fracture occurs. Neutron irradiation raises this temperature. Therefore, it is important to maintain a reduced primary coolant system pressure when the vessel wall is at temperatures below the permissible \(RT_{NDT}\) to prevent brittle fractures.

The \(RT_{NDT}\) is known for a vessel when it is new, and its value should be low. However, there is a shift in the \(RT_{NDT}\)
from an initial temperature to a higher-temperature as a function of the fast neutron dose (nvt). To keep track of this important parameter, test samples of steel identical to the vessel itself are made during fabrication.

The manufacturer is constantly vigilant in making judgments and relying on experience to assure the quality of the reactor vessel. The manufacturer must establish separate procedures for specific manufacturing operations such as cold-forming, hot-forming, forging quench and tempering heat treatments, post-weld heat treatments, submerged arc welding, weld deposit cladding, and manual welding.

(5) CONTROL ROD DRIVE MECHANISMS

Control rod drive mechanisms (CRDM) are used to position the movable neutron-absorbing control rods within the core, thereby controlling core reactivity. The full-length CRDM positions a control rod assembly which, when fully inserted, provides neutron absorption uniformly throughout the fuel assembly. The reactor is shut down when the rods are fully inserted and the core reactivity is varied as the control rods are withdrawn from the core. The full-length CRDM is an electromechanical device that operates on the magnetic jack principle. The CRDM consists of four basic subassemblies: pressure housing, latch assembly, drive rod, and operating coil stack (Figure 2-8).

The pressure housing is threaded and seal-welded to the reactor vessel head and, as such, becomes an integral part of the reactor vessel. The housing contains the moving parts of the CRDM that are the latch assembly and the drive rod. No mechanical seals or electrical penetrations are required.

The latch assembly is located in the lower section of the CRDM pressure housing. It contains fixed and movable magnetic pole pieces that actuate two sets of gripper latches called stationary and movable grippers. The gripper latches are linked to the pole pieces in such a manner that movement to pole pieces causes the gripper and latches to cam in or out of the drive rod that passes through the latch assembly.

The drive rod assembly connects the control rod with the latch assembly. The drive rod has a series of circumferential grooves spaced at 5/8-inch intervals. The groove profile matches the control of the gripper latches so that the latches can carry the drive rod and attached control rod. The lower end of the drive rod attaches to the control rod by means of the collapsible coupling that is actuated from the top of the drive rod. A disconnect rod located within the drive rod, having a disconnect button at the top of the drive rod, connects the coupling at the bottom.

The operating coil stack consists of three independent coils mounted outside the pressure housing at the same elevation as the latch assembly. When the coils are energized, a magnetic flux field is generated. This flux passes through the nonmagnetic pressure housing and couples with the magnetic steel poles of the latch assembly. Force sufficient to provide vertical motion of the movable pieces is obtained by the solenoid principle.

This sequence results in raising the drive rod 5/8 of an inch. In normal operation, this sequence is accomplished in approximately 3/4 of a second. The withdrawal speed is therefore approximately 72 steps/minute. The insertion sequence is essentially a reverse of the withdrawal sequence and the rod speed is the same.

The CRDM cooling baffle functions as a cooling baffle with exhaust exiting through ducting to the CRDM fan system and into the building containment. The function of the CRDM seismic support system is to restrain the motion of the CRDM during a seismic loading. By design, the support system does not support the CRDM during normal operations.
IV. Structural Design of Advanced PWR Reactor

For the development of the advanced PWR, we tried to further improve its safety, reliability, operability, etc., based on the results of the third PWR improvement / standardization initiative, by benefiting from the operation experience and incorporating the latest technologies. The features of the improved design of the advanced PWR structure include the following.

For Reactor Internals:
* Radius enlargement to deal with the trend of larger cores
* Optimizations of coolant flow and temperature mixing to deal with larger cores

For the reactor vessel:
* Enlargement of the radius corresponding to larger core sizes
* Improved reliability by integral forging

For Fuel Assembly:
* Increased total height of the fuel assembly to achieve a higher burnup rate
* Use of zircaloy grid fuel to improve economical efficiency

For Neutron Reflector:
* Use of a neutron reflector for higher economical efficiency
* Simplified structure by using a neutron reflector

* Reduced irradiation from RV by using the neutron

In this paper, we focused to describe the design objectives for a new structure of reactor internals, that is, for Neutron Reflection, its structural design and the validation Test results.

Figure 4-1  Baffle-Former Structure in Current PWR

Figure 4-2  Neutron Reflector in APWR
In current PWRs in Japan, the lower internals consist of a core barrel, baffle former structures, neutron pads, a lower core plate, support columns, and a core support plate that is welded to the core barrel. The baffle former structure forms a cavity that contains fuel assemblies; it consists of vertical plates called baffle plates that are bolted to the horizontal plates called formers. The formers are in turn bolted to the core barrel. The structure of the baffle former is shown in Figure 4-1.

There are four corner pins between blocks that are shrunk into the lower block with clearance in the upper block. The complete neutron reflector is aligned to the core barrel with four horizontal pins and customized inserts at the circular flanges of both the bottom and top blocks; this is similar to the design currently used for aligning the upper internals to the core barrel. The stacked blocks are fastened to the lower internals with eight tie rods, and lower block is fastened to the lower internals with eight bolts.

Neutron pads that are installed in current PWRs are not necessary in the APWR because a neutron reflector sufficiently decreases the neutron exposure rate.

Table 4-1 Design Features of APWR Reactor Internals

<table>
<thead>
<tr>
<th>(1) Design Features of Reactor Internals</th>
<th>APWR</th>
<th>Current 4 Loop</th>
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<tbody>
<tr>
<td>Neutron Reflector</td>
<td>Fewer than 50 (Out of Core Region)</td>
<td>Approx. 2000</td>
</tr>
<tr>
<td>Baffle-Former</td>
<td>Approx. 4000</td>
<td></td>
</tr>
<tr>
<td>Total No. of Parts</td>
<td>Fewer than 200</td>
<td>Base</td>
</tr>
<tr>
<td>Neutron Exposure Rate on Reactor Vessel</td>
<td>1/3 of Base (Without using Neutron Pads)</td>
<td>Base</td>
</tr>
<tr>
<td>Fuel Cycle Cost</td>
<td>1% Reduction from Base</td>
<td>Base</td>
</tr>
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</table>

Reactor internals in APWR are enlarged in the radial direction due to the increase in the number of fuel assemblies. The inner diameter of the reactor vessel is approximately 5.1m while those of conventional 4-loop plants are approximately 4.4m. At the same time, the structural designs of some components are changed, based on information learned from operations, in order to improve the reliability.

One of the main features is the adoption of the neutron reflector. In PWR plants currently operating in Japan, the reactor internal structures have generally had excellent performance. However, uncertainties remain regarding the long-term behavior of the materials used in the baffle former structure because the effects of long exposure to severe radiation environment are not well known. For new plants, a longer design life and a higher capacity operating rate are required, so potential operational issues will likely increase with the longer plant life.

As a result, the neutron reflector was developed as an alternative to the baffle former structure for the APWR. The neutron reflector was designed to simplify the structure for greater reliability and to improve the neutron economy. The design features of the neutron reflector shown in Figures 4-1, 4-2 and Table 4-1.[1]-[2]

(2) Neutron Reflector Flow Test [3]

There are 1600 cooling flow holes in the neutron reflector metal blocks. A flow test was used to verify flow distribution in the cooling holes of the neutron reflector blocks. The flow deviation from the average flow through flow holes should be small in order to uniformly decrease the metal temperature. A 90% flow deviation was set as the design criterion.

To confirm uniformity of the coolant flow deviation in the cooling holes, the flow test was conducted using a full-scale, 1/8-sector test model representing the lower core plate and the lower structure of the neutron reflector. Figure 4-3 shows the test section. Orifice flow meters were installed each of about 200 cooling holes to measure the

Figure 4-3 Neutron Reflector Flow Test Model

Figure 4-4 Neutron Reflector Cooling Hole Flow Distribution
The results of measurements confirmed that the minimum flow was 95% and thus the design criterion of 90% was met (see Figure 4-4).

(3) Reactor Internal Structures Integrated Flow Test

To verify the design validity of APWR reactor internals, a 1/5-scale model test simulation of the entire core structure was performed. The test facility consist of a test section and an associated loop containing piping, a pump and a water pool. The water fed to the main piping by the pump and branched into four pipes equipped with a flow control valve and an orifice flow meter.

Figure 4-5 shows the test section of reactor internals, fuel assemblies, a reactor vessel, and inlet/outlet pipes. Each structure is a 1/5-scale model precisely representing the structure used in an actual plant. However, difficult to fabricate details such as the cooling holes of the neutron reflector and the fuel assembly were represented by using computer simulation.

Some components were installed with strain gauges, accelerometers or pressure taps so that data could be acquired. Measuring points and sensor types were selected based on technically significant factors and predicted vibration characteristics or hydraulic performance. In this way, data could be obtained for verifying the integrity of the entire reactor structure associated with its scale-up and for evaluating the design of new structures that include the neutron reflector. Tests were conducted at ambient temperature and pressure. The flow was set to be similar to flow-induced vibration characteristics and hydraulic performance in an actual plant.

a. Measurement of Vibration Characteristics

Measurements of vibration characteristics were performed to check a vibration analysis model of reactor internals, and the results were compared with analytical results.

![Figure 4-5 Setup of Reactor Internal Structures Integrated Flow](image)

![Figure 4-6 Acceleration Response of Core Barrel](image)

![Figure 4-7 Acceleration Response of Neutron Reflector](image)
C. Seismic Test and Evaluation

Thus, the structural integrity of the reactor internals was fluid force remained within the design limit and the stresses were evaluated. Consequently, it was confirmed that the strain response, and design margins and structural strength were calculated based on the result of the component’s fluid forces and varying stresses on each component. Velocity, which indicated that the vibration responses were random and no abnormal vibrations occurred.

Fluid forces and varying stresses on each component were calculated based on the result of the component’s strain response, and design margins and structural strength were evaluated. Consequently, it was confirmed that the fluid force remained within the design limit and the stresses were sufficiently small compared to the allowable stress. Thus, the structural integrity of the reactor internals was found to be free of problems.

C. Seismic Test and Evaluation [4]-[6]

NR is placed in a narrow gap between the NR and a Core Barrel, the added fluid mass and the damping become large number, compared with ones in air. A new method to estimate the added mass and the fluid damping for this case has been introduced based on a narrow passage flow theory. A vibration test is performed to assess above characteristics of NR vibration with considering a fluid–Structuer Interaction (FSI) effect and an appropriateness of the method of analysis employed to the response of the NR to seismic activity.

In this test, two kinds of nonlinear effects is observed, which is the fluid-structure interaction effect, and other is impact between NR and support pin or friction effect which occurs in some positions such on the lower core plate and between the NR blocks.

This new seismic analysis method and model are verified by comparing these analysis results with the test results. This study aimed to verify whether the method of the NR seismic analysis modeling was appropriate. This was done by carrying out tests and analyses in which horizontal seismic vibration and vertical seismic vibration were dynamically given in order to evaluate the structural integrity of the NR to seismic vibration. The results of this study showed that the analysis results and test results of the test model were in good agreement. Thus, the appropriateness of the method of modeling the structural elements of the model and the FSI effect in the gap between the NR and CB was verified. Therefore, this method of seismic evaluation is appropriate for an actual NR and it was confirmed the appropriate margin of safety for the structural integrity.

V. CONCLUSION

There are 2, 3, and 4 loop types of PWRs in existing Japanese plants. The designs of current plants with actual results of commercial operations in the United States were adapted to the structure of these loop types. Technical standards were revised in 1980. Due to improvements in the reliability of components, better functions and lower cost, the Japanese government, electric power companies and manufactures cooperated in standardizing third generation light water reactors. This established the structure of existing standard types.

In this paper, we have explained the structural design of existing PWR nuclear reactors in terms of design basis and procedures. In the structural design of Advanced PWRs, successful results and the upgraded techniques are reflected in the third generation, standard improved type that is the main strength of current plants. These improvement designs created better safety, reliability, and economical efficiency. RCC and the position of core instrumentation are rationalized in order to consider the core employment of later reactors. The neutron reflector that was adopted increased the neutron and economical efficiency and simplified the structure, and neutron radiation inside the reactor vessel that provides a covering effect of neutrons and increases the reliability involved in reducing the radiation intensity. In terms of new structures, the structural design, flow vibration evaluation, seismic design evaluation, and design validity for the reliability of long-term operation were confirmed.

REFERENCES