Demonstration on Post LOCA Long-Term Core Cooling by Large-Scale Test

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There is a concern that coolant containing debris, specifically crushed pieces of thermal insulation, paint chips, and chemical precipitates, generated in a loss of coolant accident, which is one of the design basis accidents of a PWR, will be fed into the reactor vessel during recirculation operations and the debris will block the cooling flow paths at the core inlet, resulting in an inhibition of the removal of core decay heat. Regarding this concern, the flow test was carried out by using a system with two full-scale fuel assemblies and it was verified that the effect of debris on the cooling flow path blockage was negligible, and subsequently conducted a thermal-hydraulic analysis based on the test results to clarify that sufficient coolant was supplied to the core and the core decay heat was properly removed.

1. Introduction

In a loss of coolant accident (hereinafter referred to as LOCA), which is one of the design basis accidents of a Pressurized Water Reactor (PWR), it is assumed that the piping and/or other components constituting the reactor coolant pressure boundary rupture during power operation, and coolant flows out of the system through the rupture opening, resulting in a decrease in the cooling capacity of the reactor core. The high-temperature and high-pressure coolant ejected from the rupture opening damages the thermal insulation and paint inside the containment vessel, resulting in the formation of crushed pieces of thermal insulation (fiber debris or particle debris) and paint pieces (particle debris). In addition, several hours after the occurrence of a LOCA, metallic substances that have been dissolved in the coolant since immediately after the occurrence of the LOCA precipitate together with additives such as pH adjusters (i.e., chemical debris). The coolant containing the debris accumulates in a containment sump installed in the containment vessel and is supplied to the reactor vessel and containment vessel spray system for the purpose of long-term core cooling (recirculation operation). At this moment, debris contained in the coolant is captured by a screen installed in the containment sump, but some debris may pass through the screen and enter the reactor vessel. If the cooling flow paths in the core are blocked by this debris, the amount of coolant supplied to the core will be reduced, and there is concern that this will inhibit the removal of core decay heat (i.e., in-core debris effects downstream of sump screen, **Figure 1**).

In the United States, a study on the above concern has been ongoing for nearly 20 years as Generic Safety Issue (GSI)-191, which is based on the flow test using the simplified system of the actual reactor, the allowable amount of fiber debris that contributes to the blockage of the core inlet flowing into the reactor vessel is limited to several tens of grams per fuel assembly (example of the results based on Option 2a described in Chapter 2) (1) . Since this test uses a simplified system of actual machines, the obtained test results are highly conservative. On the other hand, the amount of fiber debris that may enter the reactor vessel in Japanese PWRs is conservatively estimated to be 1,600 g per fuel assembly (2), and since the test system similar to that in the United States is highly conservative, large-scale repair work to remove the fiber debris (replacement of fiber insulation material) will be required.

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Therefore, to establish a different approach from the above, a new test system that more precisely simulates the flow phenomena of this event was devised, and the effect of debris on flow path blockage was verified by using flow tests. Furthermore, based on the thermal-hydraulic analysis of the test results, it was clarified that the debris generated after LOCA does not affect long-term core cooling (2).

This paper reports on the evaluation of in-core debris effects downstream of the sump screen based on the flow test and thermal-hydraulic analysis mentioned above.

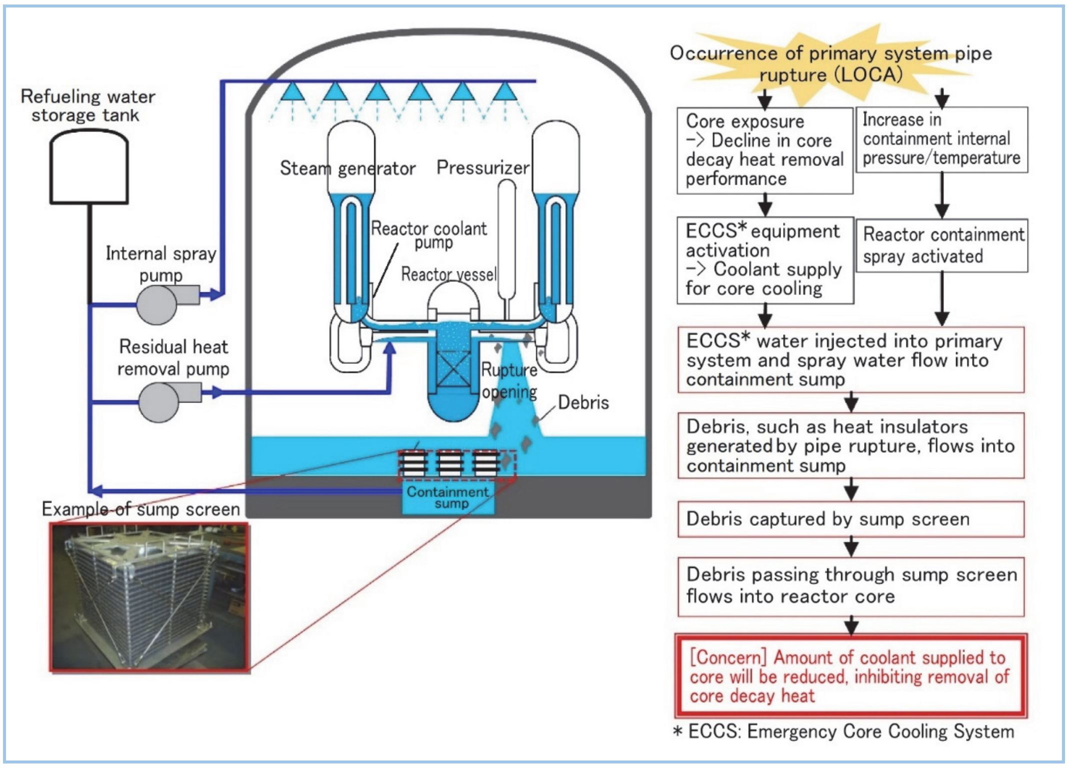


Figure 1 Outline of in-core debris effects downstream of sump screen

2. Overview of post-LOCA long-term core cooling due to debris

In the initial study in the United States, the allowable amount of fiber debris flowing into the reactor vessel was limited to 15 g per fuel assembly by the flow test assuming that fiber debris, particle debris and chemical debris flow into the reactor vessel simultaneously. However, because few PWR could satisfy this limitation, the U.S. Nuclear Regulatory Commission (NRC) proposed Option 2a to allow additional testing and analysis using plant-specific information to resolve GSI-191(3). In this test, considering the time delay until chemical debris precipitates, only fiber debris and particle debris flow into the reactor vessel until chemical debris precipitates. By this assumption, the allowable amount of fiber debris flowing into the reactor vessel was relaxed from the initial study (1). In this paper, we have also been studying based on the assumption of Option 2a, and the conceptual diagram of the core blockage event obtained from the results of the study is shown in **Figure 2**. Then, the assumptions made for the flow test and thermal-hydraulic analysis described in this conceptual diagram are summarized in the following sections.

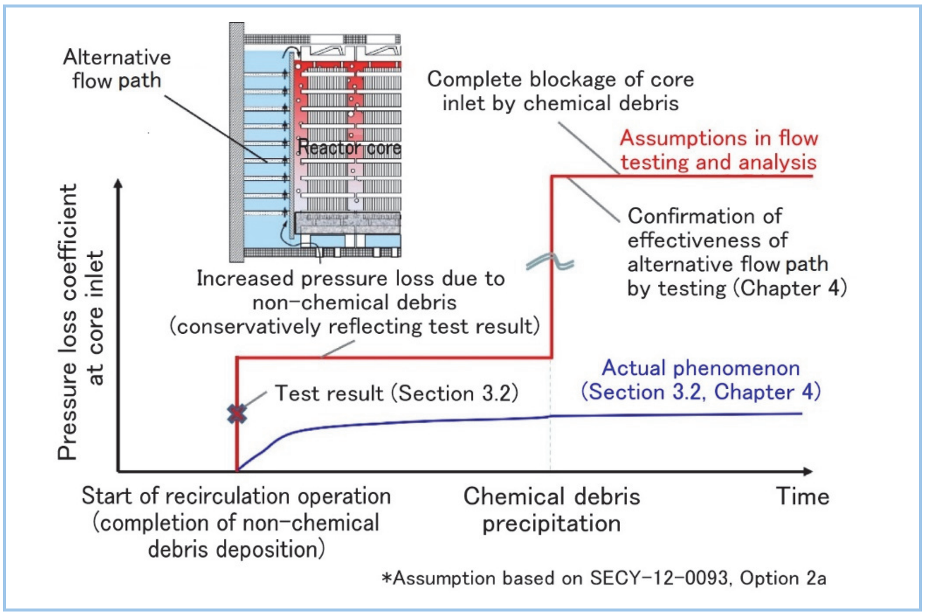


Figure 2 Conceptual diagram of post-LOCA core blockage event due to debris

2.1 Post-LOCA recirculation operation

If a LOCA occurs, the emergency core cooling system is activated and the coolant in the refueling water storage tank (or pit) is injected through the cold leg to cool the reactor core. In addition, coolant including pH adjusters (sodium hydroxide and hydrazine) is sprayed into the containment vessel by the containment vessel spray system to suppress an increase in the internal pressure in the containment vessel and to remove radioactive iodine. The sprayed water and the coolant that flows out of rupture openings accumulate in the containment sump in the reactor containment vessel, and will be injected and sprayed into the reactor vessel and containment vessel when the water level in the refueling water storage tank drops and the water intake source is switched to the containment sump. This operation, in which coolant is supplied from the containment sump to the reactor vessel and containment vessel spray system (hereinafter referred to as recirculation operation), is continued to provide post-LOCA long-term core cooling.

2.2 Blockage of core cooling flow paths by non-chemical debris

After the start of recirculation operation, non-chemical debris (i.e., fiber debris and particle debris. Hereinafter they are referred to as non-chemical debris) is injected into the reactor vessel together with coolant. When this non-chemical debris is captured at the core inlet (fuel assembly bottom nozzle), the cooling flow paths are partially blocked, and the coolant supply to the core is reduced. However, when a certain number of unblocked flow paths exist at the core inlet, the debris and coolant pass through the flow paths, so the increase in pressure loss due to blockage is limited, and coolant continues to be supplied from the core inlet. The amount of increase in pressure loss at the core inlet due to this non-chemical debris is measured by the flow test (Section 3.2).

2.3 Blockage of core cooling flow paths by chemical debris

After a LOCA, heat insulators and structural materials in contact with high-temperature coolant corrode, and the metal ions contained in these materials gradually dissolve into the coolant. These metal ions chemically react with pH adjusters in the spray water and precipitate as colloidal chemical substance (i.e., aluminum hydroxide, sodium aluminosilicate, zinc silicate, etc. Hereinafter they are referred to as chemical debris) as the solubility decreases due to the drop in water temperature. At this time, it takes time for the temperature of the coolant to drop and chemical debris to precipitate, and is accompanied by a delay of several hours from the LOCA occurrence, however we set an assumption as follows namely, if chemical debris is trapped in the non-chemical debris bed at the core inlet, the cooling channel of the core is completely cut off and coolant does not pass through. However, coolant continues to be supplied to the core via alternative flow paths that exist between the baffle barrels around the periphery of the core (Figure 2). The effectiveness of the alternative flow paths after chemical debris precipitation is confirmed by the baffle-barrel flow test (Chapter 4).

With the analytical conditions set based on the above test results, a thermal-hydraulic analysis on the long-term core cooling after a LOCA is performed. By confirming from the analysis results that no heat-up of cladding temperature occurs, it is shown that the post-LOCA effect of debris on the long-term core cooling is slight.

3. Non-chemical debris test

3.1 Basic test

To obtain basic knowledge on the blockage of the core inlet by non-chemical debris, a basic test simulating the flow path at the core inlet was conducted.

Figure 3 summarizes the test equipment and the test pieces. The 1/4-bottom nozzle system, in which the center part (1/4) of the plate of the bottom nozzle is used as the test piece, is simulated with reference to the overseas findings (1). The 2-bottom nozzle system, in which two bottom nozzles are installed in parallel, simulates gap flow paths existing around the bottom nozzle as in the actual plant.

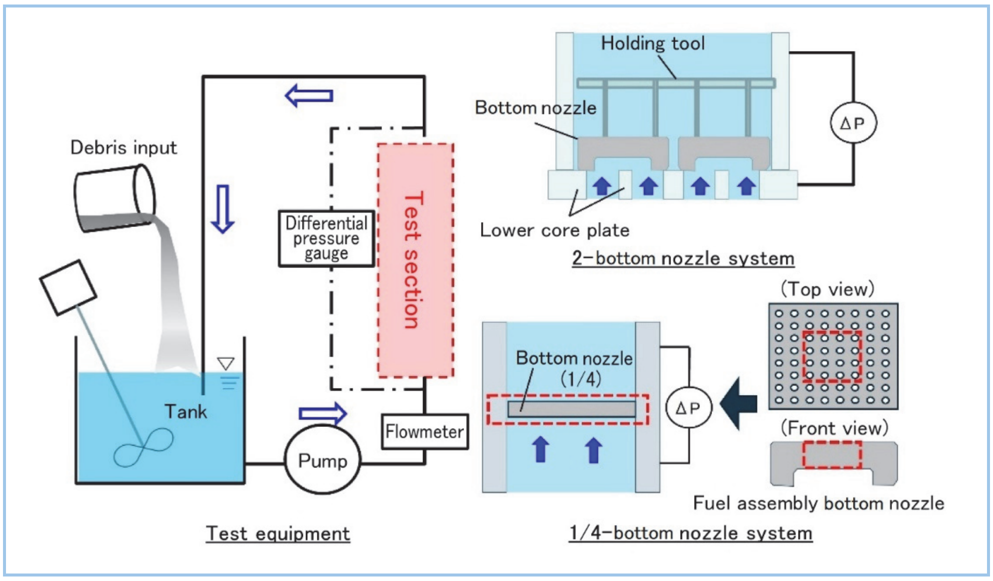


Figure 3 Summary of test equipment and test pieces (basic test)

Figure 4 shows the results of the differential pressure measurements with respect to the different test systems. In the case of the 1/4-bottom nozzle system, the differential pressure between the upstream and downstream of the test piece (hereinafter referred to as inter-test-piece differential pressure) increases rapidly with time, reaching about 200 kPa at about 4,800 seconds after the start of the test. On the other hand, in the case of the 2-bottom nozzle system, the inter-test-piece differential pressure is less than 1 kPa.

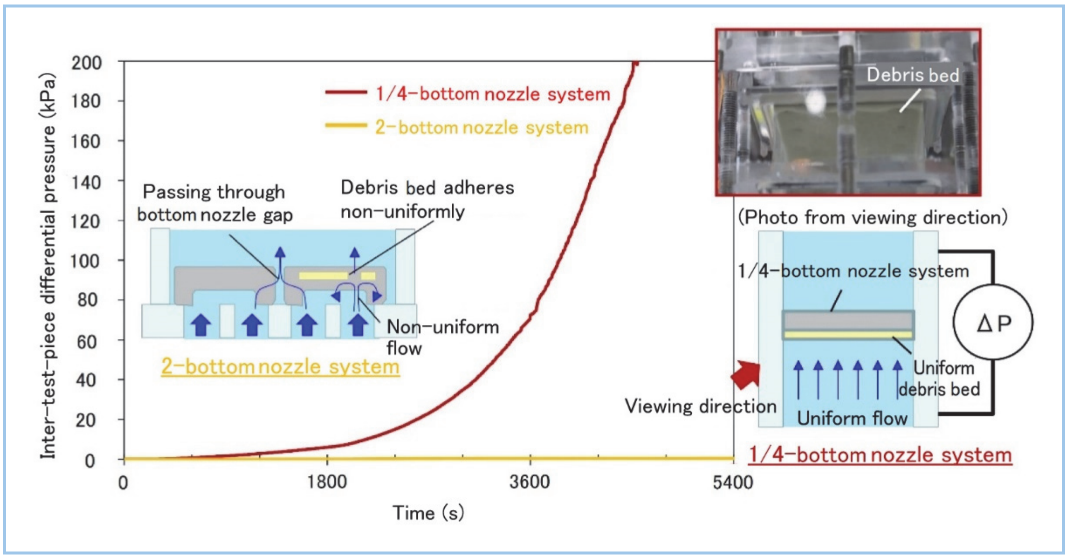


Figure 4 Result of differential pressure measurements with respect to different test systems

It is suggested that since the 1/4-bottom nozzle system has no gap between the bottom nozzles and no gap between the test sections, the flow in the test section is uniform and a uniform debris bed is formed on the lower surface of the bottom nozzle, resulting in a large increase in the inter-test-piece differential pressure. On the other hand, since the 2-bottom nozzle system simulates the gap flow paths around the lower core plate (4 holes per fuel assembly) and the bottom nozzle, the flow into the test piece becomes non-uniform, and debris in the test water is adhered and accumulated to the test piece in a non-uniform manner. This suggests that flow paths were partially secured in various parts of the test piece, resulting in a constant inter-test-piece differential pressure of less than 1 kPa.

From the above, it was found that the 1/4-bottom nozzle system has highly conservative compared to the actual plant, while the 2-bottom nozzle system simulates the gap flow paths around the lower core plate and bottom nozzle to simulate the same flow phenomena as in the actual plant, thus enabling appropriate evaluations.

In addition, sensitivity tests for other test parameters (test water temperature, flow velocity, particle debris diameter, P/F, and debris input order) were also conducted, and it was confirmed that the sensitivity of these parameters is small for the 2-bottom nozzle system (2). The P/F refers to the weight ratio of particle debris (P) to fiber debris (F).

3.2 Flow test using fuel assemblies

Based on the results of the basic test described in the previous chapter, a flow test was conducted using two full-scale fuel assemblies installed in parallel to simulate the gap flow paths around the lower core plate and the bottom nozzle. By this test, it is confirmed that the cooling channel has the water flow necessary for long-term core cooling even when non-chemical debris, which is assumed in the actual plant, flows into the reactor.

Figure 5 shows an overview of the test equipment and test section. These tests were conducted under the following conditions: Case 1, Japanese PWR representative plant conditions (approx. 1.6 kg/FA of fiber debris and approx. 4.9 kg/FA of particle debris input); Case 2, fiber debris advance input conditions (approx. 2.6 kg/FA of fiber debris and approx. 7.9 kg/FA of particle debris input); and Case 3, Japanese PWR envelope conditions (approx. 2.6 kg/FA of fiber debris and approx. 8.3 kg/FA of particle debris input). In Case 2, to check the sensitivity to the debris input method, fiber debris was input first, followed by particle debris (in Case 1 and Case 3, fiber debris and particle debris were input at the same time). The kg/FA refers to the weight of fiber debris (kg) input per fuel assembly.

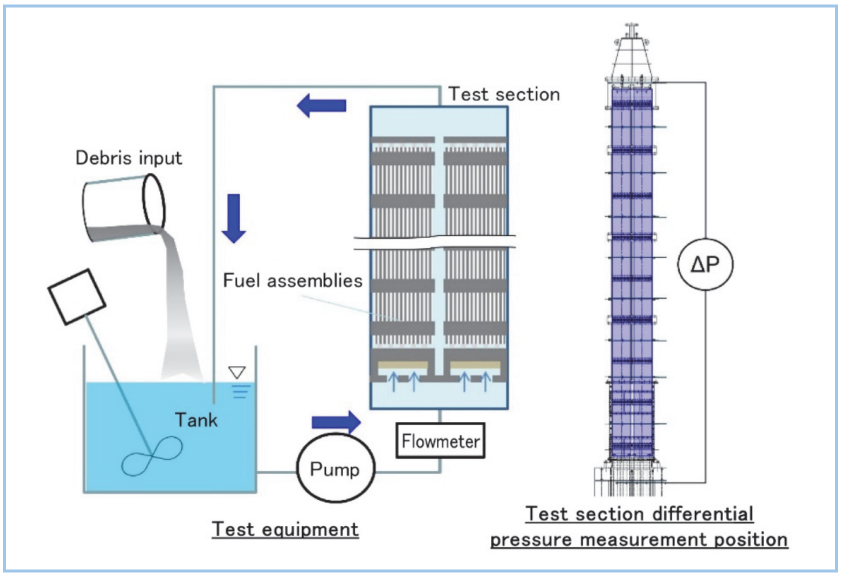


Figure 5 Schematic diagram of test equipment and test section (fuel assembly test)

Figure 6 shows the test results. The inter-test-piece differential pressure ΔP (the differential pressure between the upstream and downstream of the fuel assembly) was the highest, about 4.6 kPa, in case 3, and was sufficiently small compared to the pressure loss (reference value) of 20 kPa allowed for long-term core cooling, indicating that there was sufficient margin for the coolability of the core.

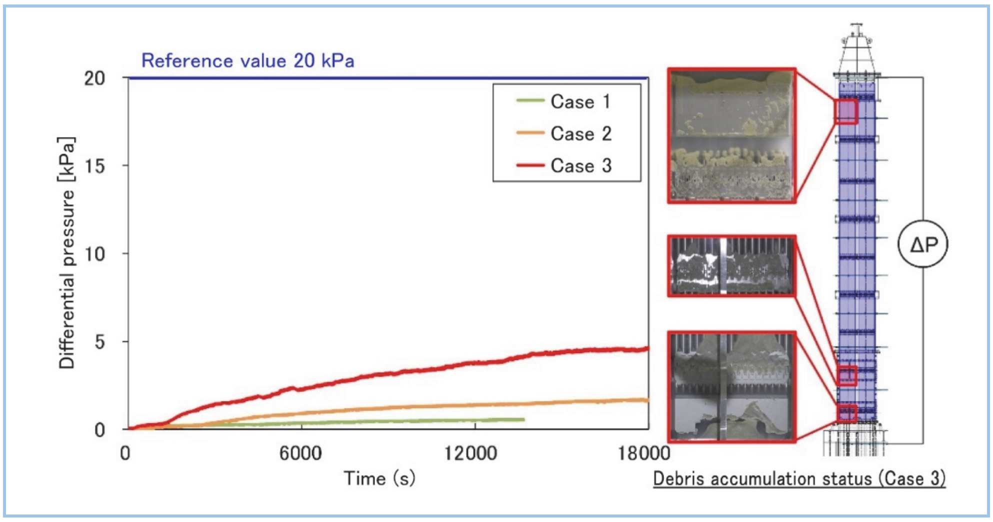


Figure 6　Measurement results in flow test using fuel assemblies

4. Chemical debris test (baffle-barrel flow test)

This test was conducted to confirm by using a full-scale fuel assembly and simulated baffle-barrel flow paths installed that coolant necessary for long-term core cooling can be supplied from alternative flow paths (baffle-barrel flow paths) even if the core inlet is completely blocked by chemical debris in addition to non-chemical debris flowing into the reactor core.

Figure 7 shows an overview of the test apparatus and test sections. These tests were conducted in two cases: Case 1, confirmation test of the baffle-barrel flow paths, and Case 2, confirmation test of the fuel assembly and the baffle-barrel flow paths. Case 1 was a conservative condition in which the flow of test water to the fuel assembly was shut off and all debris input flowed into the baffle-barrel flow paths. The test confirmed the change in differential pressure between test pieces before and after debris input using the flow rate as a parameter. In Case 2, test water was passed through the fuel assembly and baffle-barrel flow paths, and after the non-chemical debris was sufficiently captured in the fuel assembly, chemical debris was input to check the status of water supply in the fuel assembly and the baffle-barrel flow paths.

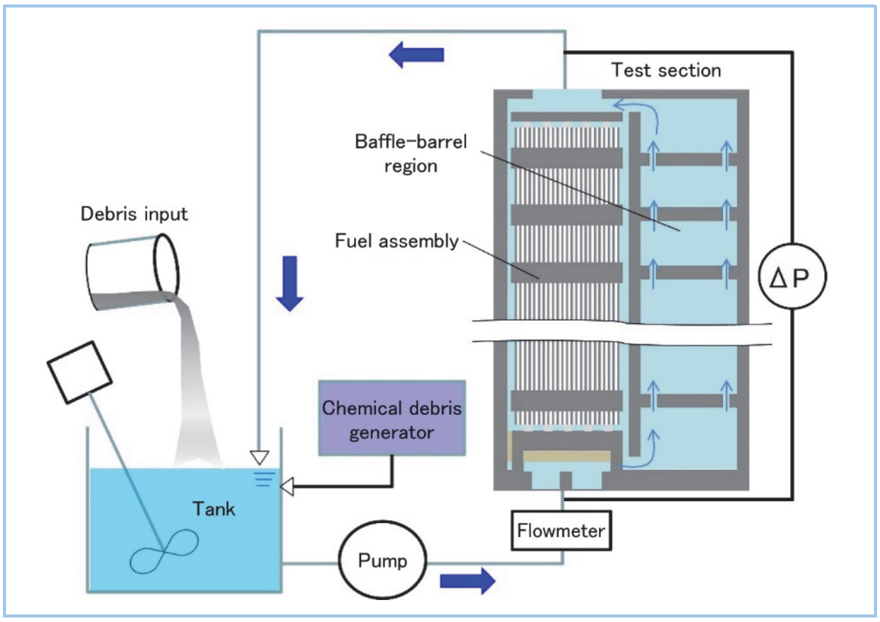
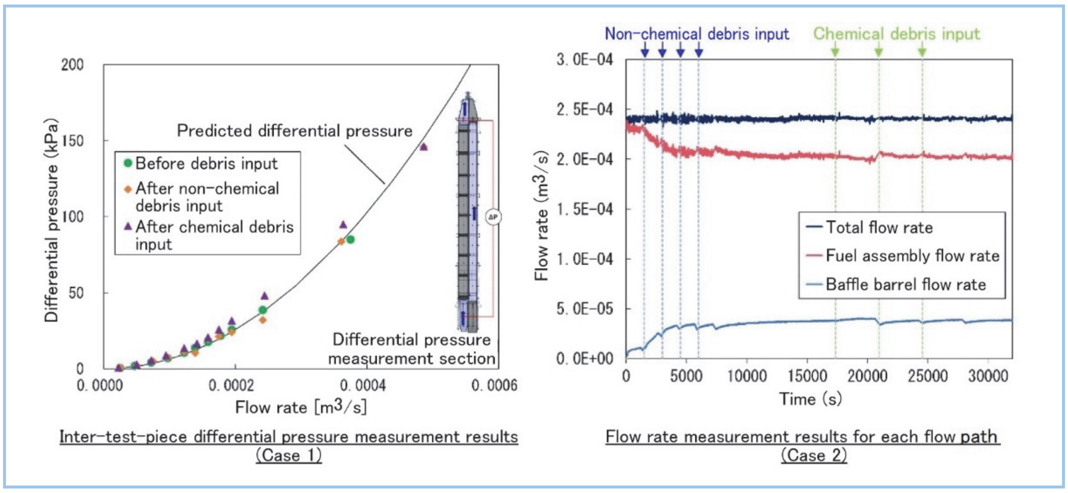


Figure 7　Schematic diagram of test equipment and test section (baffle-barrel test)

Figure 8 shows the test results. In Case 1, the change in inter-test-piece differential pressure with respect to flow rate is the same before and after debris input, and is in good agreement with the predicted differential pressure. Therefore, it was found that the baffle-barrel flow paths are not blocked by debris, and that long-term core cooling is possible with the baffle-barrel flow paths (alternative flow paths).

In Case 2, when non-chemical debris was input, the inter-test-piece differential pressure increased and the flow rate into the fuel assembly flow paths decreased slightly. However, when the chemical debris was input, the flow rate did not change and about 85% of the original flow rate flowed into the fuel assembly flow paths. This indicates that the flow path of each part of the fuel assembly (flow paths with relatively high velocity) formed by capturing and adhering non-chemical debris to the fuel assembly are still effective after chemical debris input, and that the flow paths in the fuel assembly do not become completely blocked when chemical debris is precipitated in the actual plant.



Inter-test-piece differential pressure measurement results (Case 1)

Flow rate measurement results for each flow path (Case 2)

Figure 8 Measurement results of baffled barrel test

5. Thermal-hydraulic analysis when debris blocks flow path

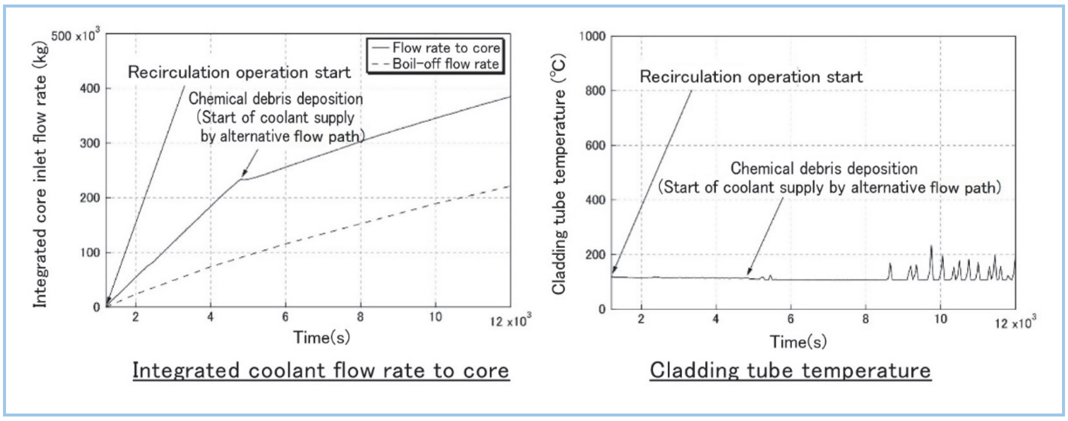
Thermal-hydraulic analysis under the conditions of increased pressure loss due to non-chemical debris and further increase in pressure loss due to chemical debris precipitation with decrease in coolant temperature was conducted. The thermal-hydraulic analysis code used was MCOBRA/RELAP5-GOTHIC, an optimal evaluation code which can evaluate thermal hydraulics in the reactor vessel in detail.

In this analysis, the increase in pressure loss due to non-chemical debris was given in a stepwise manner by twice the amount of the results of the flow test using the fuel assembly (Section 3.2), and based on the assumption of Option 2a, it was assumed conservatively that chemical debris is precipitated approximately 60 minutes after the start of recirculation operation, and that the core inlet is completely blocked after chemical debris precipitation, instantly cutting off the coolant flow from the core inlet and replacing coolant supply from the core inlet with that from the baffle-barrel flow paths. Although it is estimated that chemical debris precipitation actually starts after 24 hours of recirculation operation(1), and it is estimated that the flow path at the core inlet is still effective after chemical debris deposition(2), it was conservatively assumed in this analysis that chemical debris precipitation occurs after about 60 minutes of recirculation operation and that the core inlet is completely blocked at the same time.

The analysis was performed for a cold leg break LOCA, which is a more severe rupture location from the viewpoint of core cooling, in a standard 4-loop plant. The core decay heat was assumed for a high burnup (55 GWd/t) uranium dioxide fuel core, which is currently the most standard in Japanese PWRs.

Figure 9 shows the results of the thermal-hydraulic analysis for a standard 4-loop plant. It was confirmed that even when the increase in pressure loss at the core inlet due to non-chemical debris immediately after the start of recirculation operation (1,200 seconds after the LOCA) and the increase in pressure loss due to chemical debris precipitation 60 minutes after the start of recirculation operation (4,800 seconds after the LOCA) occurred, in terms of the average behavior of the core, the cooling flow rate supplied to the core from the alternative flow paths did not fall below the evaporation rate (boil-off flow rate) and that the fuel-cladding tube temperature did not heat up due to the increased pressure loss in the core.

The cladding tube temperature temporarily increased locally in the latter half of the event, but this is not a problem for long-term core cooling due to flow path blockage because the degree of increase was small compared to the criterion value (1,200℃) and the temperature returned to the initial temperature immediately thereafter.



Integrated coolant flow rate to core

Cladding tube temperature

Figure 9 Thermal-hydraulic analysis results for standard 4-loop plant

6. Conclusion

There was a concern that a large amount of debris generated in the event of a LOCA at Japanese PWR could flow into the reactor vessel and block the core inlet after the start of recirculation operation, resulting in an inhibition of the removal of core decay heat.

To address this, MHI devised a new test system that appropriately simulates the flow phenomena of this phenomenon, and demonstrated through flow tests that the effect of debris on the cooling flow path blockage is small, and also demonstrated through flow tests using alternative flow paths (baffle-barrel flow path) that non-chemical debris and chemical debris inflow into the alternative flow paths did not lead to flow path blockage. Furthermore, a thermal-hydraulic analysis of the long-term core cooling after a LOCA based on these test results were conducted, and as a result of the analysis, it was found that the temperature rise of the fuel-cladding tube was slight and was not a problem for long-term core cooling after LOCA.

The validity of the results of these tests and analysis was accepted by the Nuclear Regulation Authority (NRA) (4), and the regulatory authority recognized that the effect of increasing or decreasing thermal insulation materials was minor.

References

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