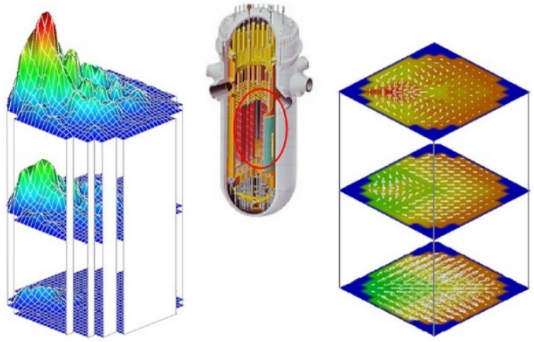
Advanced Safety Analysis Methodology - Development of New Core and Safety Analysis Codes -



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Mitsubishi Heavy Industries, Ltd. (MHI) has been working on the development of codes for core design (nuclear and thermal-hydraulic) and safety evaluation with the aim of advancing the operation of PWR (Pressurized Water Reactor) plants in terms of long cycle operations, power uprate, the adoption of high burnup fuel, etc. This report explains firstly model advancement in the nuclear and thermal-hydraulic design codes, and then explains how those advanced models have been incorporated into the non-LOCA (Loss of Coolant Accident) analysis technique. This report also describes the large break LOCA analysis method that statistically evaluates the effect of uncertainty in safety analysis inputs. Development of the CFD (Computational Fluid Dynamics) -based evaluation technology is also described, which deals with the core melt spreading behavior important in the field of severe accident, including a comparison between experiments and calculation results.

This report introduces the advanced core and safety evaluation codes of MHI which incorporate high accuracy for evaluation, in addition to their applicability to core design and safety evaluation.

1. Introduction

MHI has been working on the advancement of design analysis codes used for plant safety design and safety evaluation with the aim of advancing the operation of existing PWR plants in Japan through improvements in areas such as long cycle operation, power uprate and the adoption of high burnup fuel. Some of the advanced codes have already been employed to evaluate the ATWS (Anticipated Transient Without Scram) event upon confirmation of their applicability for analyzing such specific event; and utilized for the safety review on restarting of PWR plants after the Fukushima Daiichi Accident.

The advancement of core and safety analysis technology has been achieved by means of two main methods, in addition to the fundamental model refinement of each design analysis code. One of the two methods is to integrate the analyses in the relevant design fields. This method rationalizes overlapping conservative conditions and improves analysis accuracy by integrating the analyses without each conservativity in each design field. The other one is to introduce a statistical approach, which improves the evaluation accuracy by properly estimating overly conservative analysis conditions through the statistical evaluation of the effects of uncertainties in safety analysis input, etc.

This report introduces model refinement of the nuclear design code GalaxyCosmo-S® and the thermal-hydraulic design code MIDAC; nuclear-thermal coupled dynamic characteristics analysis code SPARKLE-2 that consists of COSMO-K, MIDAC and M-RELAP5 (dynamic plant characteristics analysis) for Non-LOCA analysis; and the statistical safety evaluation method using MCOBRA/RELAP5-GOTHIC developed for large break LOCA analysis. This report also

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presents development of the CFD-based evaluation technique that handles the core melt spreading behavior, including a comparison between experiments and calculation results.

2. Advancement of nuclear design technology

This chapter describes an overview of the newly developed nuclear design code, the advancement of analysis methods in the new code and the efforts for the verification and validation of the new code.

2.1 Overview of GalaxyCosmo-S® nuclear design code system

MHI's new nuclear design code system, GalaxyCosmo-S®(*) is developed to be applied to the nuclear design and safety analysis of PWRs. It consists of the GALAXY code, which generates the nuclear constants for each fuel assembly and the COSMO-S code, which generates the core characteristics of the three-dimensional core by using the nuclear constants and etc. **Figure 1** gives an overview of this code system.

The GALAXY code generates the nuclear constants of a two-dimensional assembly in the radial direction in consideration of complicated geometries such as fuel rod, cladding tube and moderator. The COSMO-S code carries out three-dimensional core calculation using the nuclear constants obtained from the GALAXY code as an input and evaluates the core nuclear characteristics such as critical boron concentration, reactivity coefficient and power distribution in consideration of the fuel assembly arrangement in the core. The COSMO-K code, which is a code with a three-dimensional kinetic calculation function (simulation function of neutron time-dependence in transient state) added to the COSMO-S code, has also been developed for nuclear-thermal coupled safety analysis code SPARKLE-2 refer to Chapter 4.

(*) GalaxyCosmo-S® is a registered trademark of Mitsubishi Heavy Industries, Ltd. in Japan.

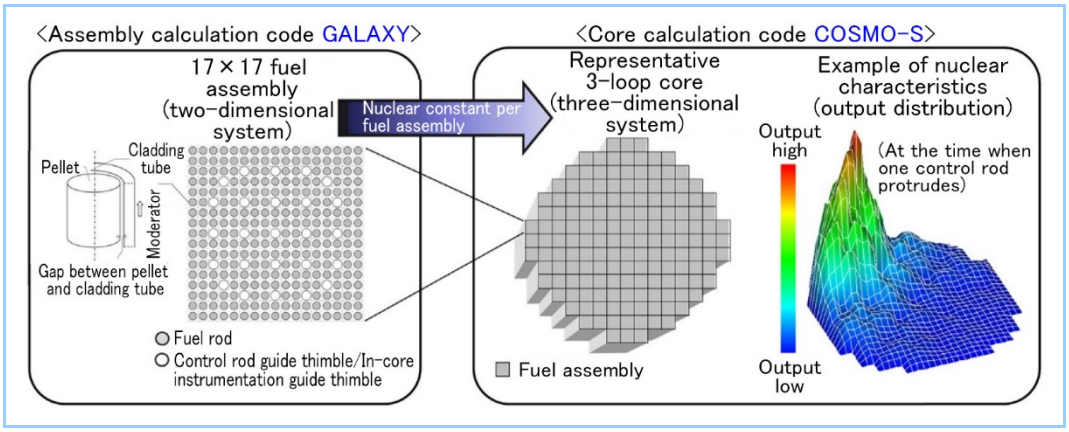

Figure 1 Overview of GalaxyCosmo-S®

2.2 Changes in requirements for core nuclear design code and advancement of analysis methods

Due to the recent incorporation of nuclear design technology and the expansion of the scope of application, GalaxyCosmo-S® can deal with not only steady-state nuclear design, but also abnormal transient and accident analyses, achieving both improved analysis reliability and convenience. As the main advancement, (1) a spectral interference effect correction model between uranium-MOX (Mixed Oxide) fuel assemblies and (2) an ultra-fine-group energy resonance calculation model are described.

(1) Spectral interference model between uranium-MOX fuel assembles

This model was developed for the objective of achieving both calculation accuracy and speed of nuclear design in a reactor core in which uranium fuel and MOX fuel are adjacent to each other. Figure 2 shows this model schematically. When uranium fuel and MOX fuel are adjacent to each other, thermal neutrons flow from the uranium fuel to the MOX fuel. However, this effect cannot be taken into account in the case of nuclear constants generation using the single fuel assembly model adopted in uranium core design. The current approved code, PHOENIX-P/ANC, considers the spectral interference effect of adjacent fuels by generating the nuclear constant in a system in which uranium fuel and MOX fuel are adjacent to each other in PHOENIX-P. Therefore, it is necessary to regenerate the nuclear constants

with PHOENIX-P when the fuel loading pattern is changed in ANC and the computational load is larger than that of uranium cores. On the other hand, the spectral interference effect correction model of GalaxyCosmo-S® does not require the regeneration of nuclear constants in a system in which uranium fuel and MOX fuel are spectral interference to each other, such as the PHOENIX-P/ANC code. This is because the COSMO-S code considers the adjacency effect of uranium fuel and MOX fuel by using the neutron flux information of the uranium fuel and MOX fuel according to the fuel loading pattern to correct the nuclear constants obtained from the single fuel assembly calculation of the GALAXY code. This technology realizes the same accuracy without a large computational load.

(2) Ultra-fine-group energy resonance calculation model

In the GALAXY code, an ultra-fine-group calculation model which is capable of precisely simulating physical phenomena relating to change of neutron energy, is adopted and accurate generation of nuclear constants was made possible regardless of moderator density. The current PHOENIX-P code adopts the resonance calculation model of 42 energy groups, and the effect of the approximation becomes large, since the neutron moderation effect by hydrogen decreases in the event of a loss of functionality of a reactor shutdown (ATWS : Anticipated Transient Without Scram) which requires applicability to moderator density conditions (about 0.4 g/cm^3) that are lower than moderator density conditions (0.6 to 0.8 g/cm^3) in normal operation. Therefore, in the GALAXY code, the nuclear constants generation by ultra-fine-group calculation of about 100,000 groups of neutron energy is adopted. The COSMO-K code, which is incorporated in the SPARKLE -2 (As described in chapter 4) code, predicts neutron flux during transient change using the nuclear constants obtained from the GALAXY code.

As described above, in the development of GalaxyCosmo-S®, from the perspective of the advancement of core nuclear design and safety analysis, the reliability of PWR are enhanced by improving the analysis model and adding functions.

2.3 Verification and validation of GalaxyCosmo-S®

Verification and validation (V&V) are important to demonstrate the applicability of GalaxyCosmo-S® to nuclear design and safety analysis for PWR cores. It has been confirmed that Galaxy Cosmo-S® is applicable to every condition from normal operation to accident analysis by combining (i) “verification” through comparison to analysis results from the continuous energy Monte Carlo code with higher-order calculation methods and (ii) “validation” using abundant actual measurement data for commercial PWRs.

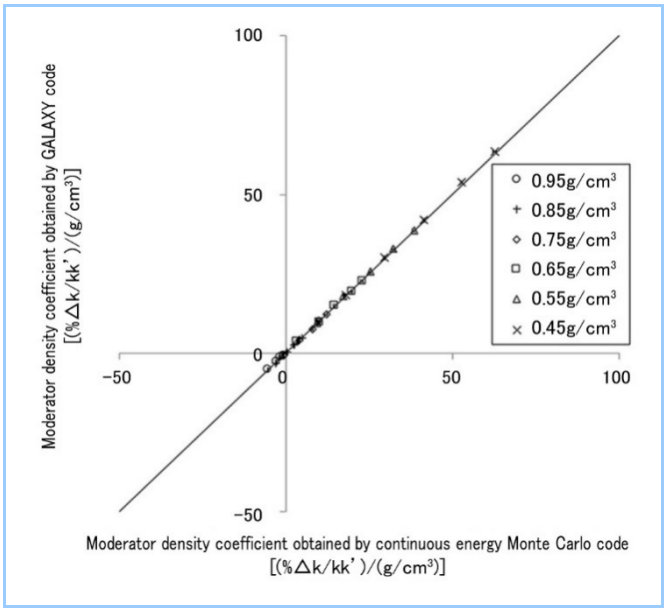

Figure 2 Moderator density coefficient verification example (comparison with MVP)

As an example of (i), the analysis results of the GALAXY code and the continuous energy Monte Carlo code (MVP) were compared for the moderator density coefficient (power reduction

effect due to decrease of the moderator density with the moderator temperature rise), which is one of the parameters indicating the self-controllability of a PWR and is important for reducing the power in the event of ATWS. As shown in **Figure 2**, the moderator density coefficient obtained from the GALAXY code is in good agreement with MVP code regardless of the moderator density condition, and the difference does not increase from that in normal operation even under the low moderator density conditions in the event of ATWS.

3. Advancement of thermal-hydraulic design technology

In thermal-hydraulic design, the DNB (departure from nucleate boiling) related to the overheating and burning of fuel rods is evaluated for the purpose of confirming safety during normal operation and abnormal transient changes. For DNB evaluation, it is necessary to simulate the coolant behavior near the fuel rods in detail, and the two technologies of the sub-channel analysis code MIDAC, which evaluates the coolant behavior and the DNB correlations (MG-S correlation and MG-NV correlation), which are prediction equations for the critical heat flux when overheat burnout occurs according to the coolant conditions. The MG-S correlation is applied to the evaluation of the operating range in normal operation and the safety analysis to confirm the effectiveness of safety functions, and the MG-NV correlation is applied to the safety analysis of low pressure/low flow conditions where the MG-S correlation is not applicable. The MIDAC code system can refine the safety margin for DNB generation conditions compared to the current code system and can respond to the diversification of core operations and etc., such as higher burn-up and longer cycle without impairing the safety of the plant.

3.1 Overview of thermal-hydraulic design code system MIDAC

We developed the MIDAC code, which can be applied to a wide range of coolant behaviors including gas-liquid two-phase flow, for simulating the coolant behavior in a core. The modeling of gas-liquid two-phase flow includes a mixture model, which treats gas-liquid two-phase flow as a homogeneous mixture and a separation flow model, which treats each phase of gas-liquid two-phase flow as a separate fluid. The MIDAC code adopted a drift flux model, which is one of the mixture models and can consider the velocity difference between the gas phase and the liquid phase, for the purpose of improving the evaluation accuracy under low-pressure conditions such as main steam line breaks. In addition, to enable the evaluation of thermal-hydraulic design and safety analysis with one code, the functions of the three codes, the THINC-III code for current transient thermal-hydraulics behavior evaluation for safety analysis, the THINC-IV code for steady-state thermal-hydraulics behavior evaluation for thermal-hydraulic design and the FACTRAN code for transient fuel behavior evaluation for safety analysis, were integrated into the MIDAC.

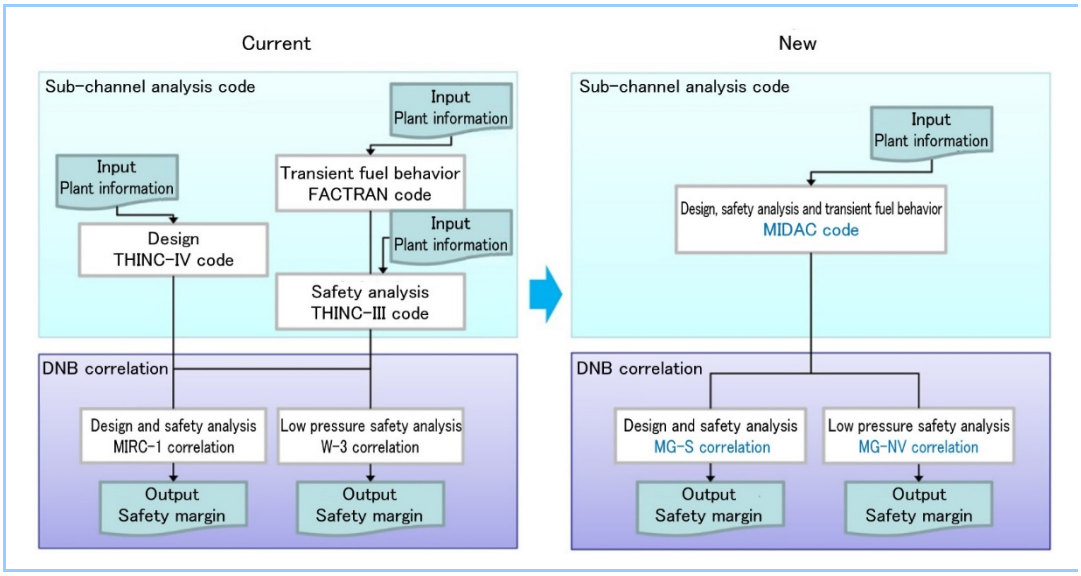

Figure 3 Current analysis code system and new analysis code system for core thermal-hydraulic design and safety analysis

The MIDAC code system integrates the functions of multiple current analysis codes to provide an efficient analysis environment.

Figure 3 illustrates the relationship between the current analysis code system applied to existing plants in Japan and the new analysis code system. In the past, input data was required for each individual code, but with this functional integration, analysis is made possible with one input data, which can lead to a reduction in the work of input data creation and output analysis. The MIDAC code also incorporates MHI's independently developed DNB correlations, "MG-S correlation" and "MG-NV correlation," for the prediction of critical heat flux[(1)]. In addition, the MG-S correlation and MG-NV correlation employ highly expandable technology so that they can flexibly respond to improvements in fuel design for the purpose of improving DNB prevention performance in the future. In particular, the range of application for low flow rate conditions was expanded and the prediction accuracy was improved, which can contribute to improving the reliability of safety analysis.

3.2 Validation of MIDAC code system

The V&V of the MIDAC code confirms the validity thereof through comparison to experimental data according to important physical phenomena occurring in the core and to other codes. This section shows the comparison results of void fraction prediction, which requires the most reliable simulation among such important physical phenomena. The test data was acquired from a NUPEC PWR Subchannel and Bundle Tests registered by the OECD and NEA (Organisation for Economic Co-operation and Development/Nuclear Energy Agency), which are international organizations[(2)]. In this test, the void fraction in the 5×5 bundle was measured using a γ-ray measuring device under the coolant conditions that envelope the safety analysis range.

Figure 4 gives the void fraction prediction results used in the current code, which uses a homogeneous flow model assuming that the velocity between the gas phase and the liquid phase is the same, and the void fraction prediction results using the drift flux model adopted by the MIDAC code. It is indicated in Figure 4 that, under high pressure conditions, the prediction results using the homogeneous flow model and that using the drift flux model are almost the same. On the other hand, under low pressure conditions, the void fraction predicted by the drift flux model is in better agreement with that of the test results. This is because the MIDAC code can accurately take into account the effect of the relatively significant density difference compared with that under high pressure conditions. This result indicates that the reliability of the simulation is improved by adopting the MIDAC code for the thermal-hydraulic design and safety analysis of PWR plants. The validation confirmation results of the MG-S correlation and the MG-NV correlation are shown in Reference (1).

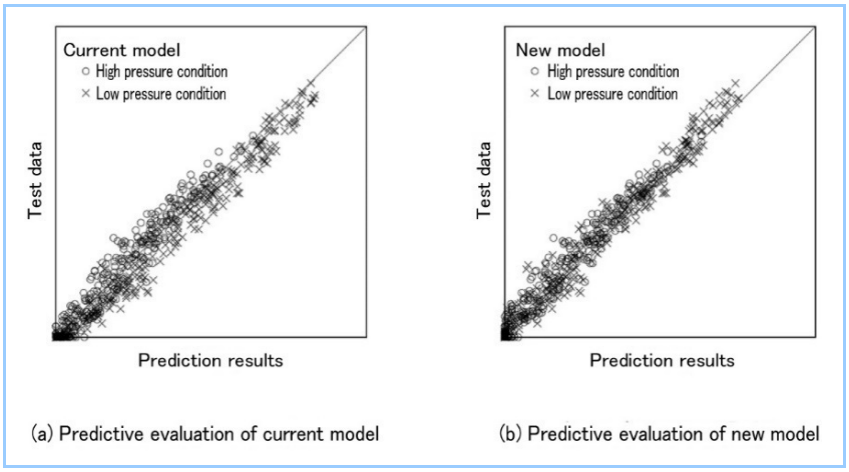

Figure 4 Prediction results of bundle void test
The prediction at the low pressure conditions by the current model is conservatively high, but the prediction results by the new model tend to match the test results.

4. Advancement of Non-LOCA evaluation technology

For nuclear plants, the design of equipment and the procedure for dealing with abnormalities are confirmed through the simulation of the plant in the case of an abnormal state. The simulation of such an abnormal state is called safety analysis, and the numerical calculation program for the simulation is the safety analysis code. Among the typical abnormal states (events)

assumed in the safety analysis of a nuclear plant, events in which the state of the plant changes transiently, excluding LOCA introduced in Chapter 5, are collectively called Non-LOCA events. This chapter describes the safety analysis code SPARKLE-2, which was developed to improve the analysis accuracy of these Non-LOCA events, and our activities to advance the evaluation technology in the advancement of PWR plant operation.

4.1 Overview of Non-LOCA analysis code SPARKLE-2

The analysis of Non-LOCA events so far consists of simulations using multiple analysis codes for each of the plant system, core, fuel rod, etc.

The SPARKLE-2 code adopts the PWR system transient analysis code M-RELAP5, COSMO-K code described in Chapter 2, which is a code with a neutron kinetics function added to the COSMO-S code, and MIDAC code described in Chapter 3 for the analysis of the thermal-hydraulic characteristics of the core and fuel. The SPARKLE-2 code is a nuclear-thermal coupled dynamic characteristic analysis code that exchanges data between various analysis codes in both directions from moment to moment. By using this code, it is possible to evaluate the thermal-hydraulic change of the primary system of the nuclear plant and the interaction between the three-dimensional nuclear characteristics and the thermal-hydraulic characteristics in the core.

Specifically, the interaction between nuclear characteristics and thermal-hydraulic in the core refers to the spatial interaction between thermal-hydraulic characteristics and nuclear characteristics as shown in **Figure 5**, When the reactor power fluctuates and voids are generated, neutrons are not sufficiently decelerated and nuclear fission is reduced, resulting in the reduction of the power near the voids. By using such coupled nuclear-thermal evaluation that incorporates the spatial interaction between thermal-hydraulic characteristics and nuclear characteristics, the evaluation accuracy is improved in comparison to the conservative conventional evaluation that ignored the spatial power distribution change due to local void generation.

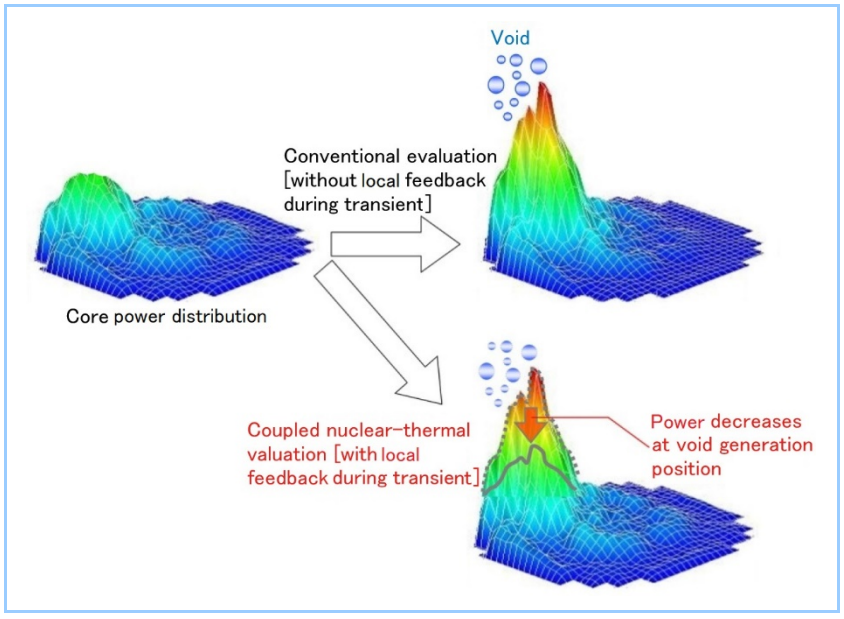

Figure 5 Coupled nuclear-thermal evaluation

When applying the SPARKLE-2 code to Non-LOCA analysis, we confirmed the reproducibility of phenomena related to thermal-hydraulic and nuclear characteristics with the SPARKLE-2 code through experimental analysis, actual PWR plant transient analysis and comparative verification with the preceding code. In confirmation of the applicability, we extracted phenomena that have a significant effect on the indicators (heat transfer between the primary and secondary systems, changes in core flow rate, etc.) for accident progress and safety confirmation (presence or absence of the pressure applied to the primary system boundary and the generation of DNB that was described in Chapter 3 Advancement of thermal-hydraulic technology) and selected an element test, a comprehensive test, etc., for the phenomenon to ensure the completeness of performance confirmation. As an example, this section indicates the analysis results with the SPARKLE-2 code of the LOFT L6-1 test that simulates loss of load,

which is one of the Non-LOCA events. The LOFT test equipment simulates a typical 4-loop PWR. In order to properly simulate the heat transfer flow behavior in the actual PWR on reduced-scale equipment, the test conditions such as the volume-to-power ratio, pressure and temperature are equivalent to those of the actual PWR. **Figure 6** presents the comparison results of the pressure as a representative system parameter. It is indicated that the analysis results of the SPARKLE-2 code show good agreement with the test data in the range of ±0.2 MPa, which is similar to the measurement error of the actual equipment.

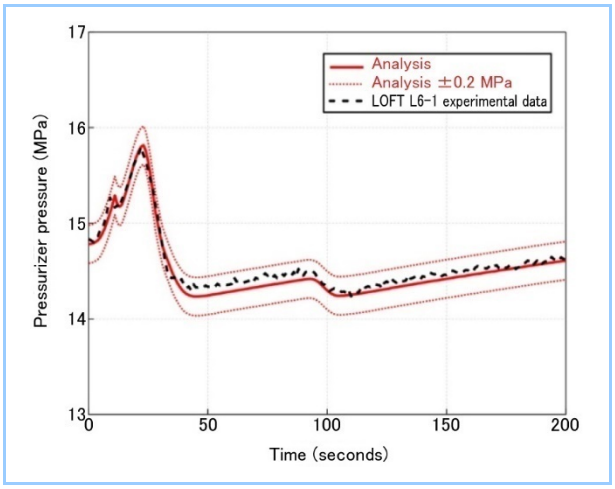

Figure 6 LOFT L6-1 test analysis

4.2 Application example of coupled nuclear-thermal hydraulics evaluation

This section describes the specific effects of the application of coupled nuclear-thermal evaluation.

The effect of coupled nuclear-thermal hydraulics evaluation appears when the power distribution is distorted or the cooling condition in the core is biased. Among the Non-LOCA events in which such a situation occurs, this section explains “main steam line break”.

A “main steam line break” is an event in which the main steam line connecting the steam generator to the turbine is ruptured and the steam emission increases rapidly to excessively cool the primary system. Due to the difference in the cooling state between the steam generator connected to the ruptured main steam line and the steam generator connected to the integrity main steam line, the flow condition in the loop and the core is biased. As a result, the horizontal power distribution of the core becomes distorted, i.e. the local power in the region where the coolant temperature is lower, becomes high. **Figure 7** compares the result of applying coupled nuclear-thermal hydraulics evaluation using the SPARKLE-2 code to a “main steam line break” accident in a domestic PWR plant with the result of the conventional evaluation. The initial condition of the accident is assumed to be a hot zero power state, in which the energy retained in the primary coolant and the fuel are low and the cooling of the primary system is larger than a hot full power state.

Excessive cooling of the primary system due to the rupture of the main steam line adds positive reactivity to the core and then causes a criticality, resulting in an increase of the reactor power (at about 25 seconds in Figure 7). The comparison with the conventional evaluation indicates that the increase in reactor power after the criticality was mitigated. This is because SPARKLE-2 can handle the reactivity feedback effect (interaction between spatial thermal hydraulic and nuclear characteristics) due to the local temperature rise of the coolant, fuel, etc., with the power distribution distorted. This accident is determined to be safe by confirming that the DNBR (DNB ratio, which is the ratio of DNB heat flux to actual heat flux) does not fall below the safety limit (see in Figure 7). As a result of the introduction of coupled nuclear-thermal hydraulics evaluation, the difference between the minimum DNBR and the safety limit, that is, the thermal margin, increased higher than that of conventional evaluation (Figure 7, right). This is the result of optimizing the margin in the analysis by the introduction of coupled nuclear thermal evaluation. The aforementioned MG-NV correlation was also adopted as the DNB correlation to improve the DNB prediction accuracy, so the safety margin in the analysis is optimized in terms

of the safety limit. In this way, the coupled nuclear-thermal evaluation can be expected to contribute to the advancement of PWR plant operations.

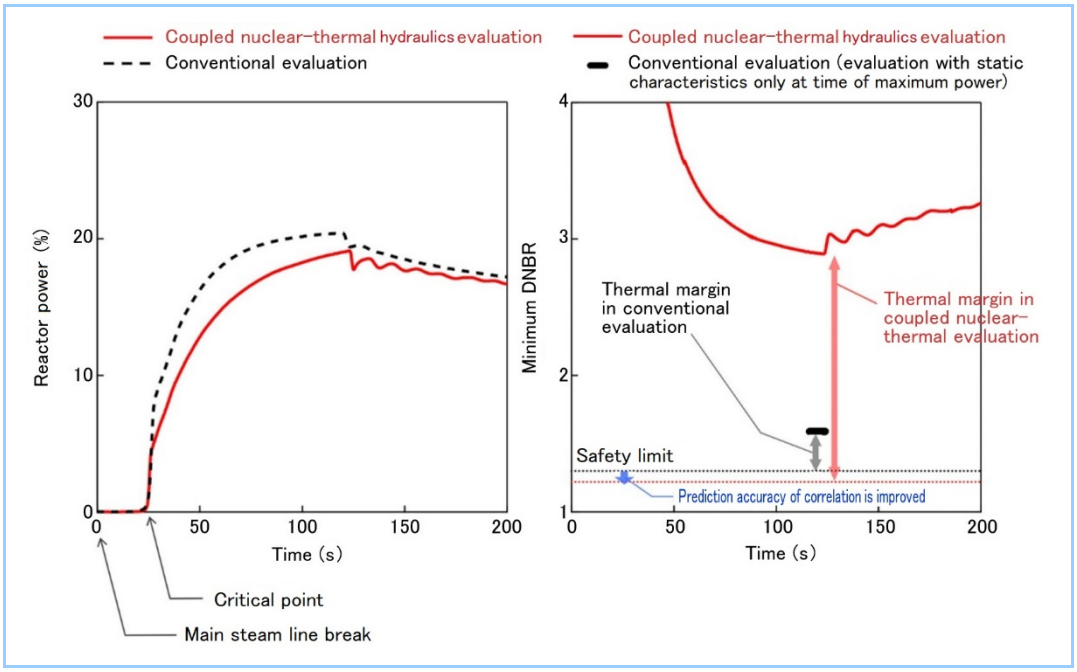

Figure 7 Accident analysis of main steam line break

5. Advancement of LOCA evaluation technology

LOCA is an event that assumes discharge of coolant from the system due to a break of the primary coolant system pipe. This accident may cause the damage of the fuel rods as the cooling capability in the core is impaired. Nuclear power plants are equipped with an emergency core cooling system that injects coolant to cool the core in the event of such an accident. It is confirmed by safety analysis that in the case of such an event, fuel integrity criteria required by Japanese regulations for the emergency core cooling system (peak cladding temperature shall be 1,200°C or less, etc.) are met and the core can be cooled. This chapter explains the statistical safety evaluation method using the new LOCA analysis code MCOBRA/RELAP5-GOTHIC, which was developed to be applied to the advancement of PWR plant operations.

5.1 Overview of LOCA analysis code MCOBRA/RELAP5-GOTHIC

The current safety analysis code for PWR LOCA events employs several simplified conservative physical models for important phenomena during a LOCA due to restrictions on the computer capability at the development of the analysis code. Furthermore, since the individual uncertainties of the physical models in the code and the analysis conditions were treated so that the evaluation results would be sufficiently conservative, the evaluation results were indeed very conservative.

In recent years, remarkable advances in computers and numerical analysis technology have enabled detailed simulations. For LOCA analysis, detailed simulation codes have been becoming mainstream globally and the analysis using the detailed simulation code have been approved by the Nuclear Regulatory Committee in the United States. For this reason, we developed the MCOBRA/RELAP5-GOTHIC code, which can more accurately simulate important physical phenomena in LOCA, with the aim of applying the detailed simulation code to LOCA analysis.

The MCOBRA/RELAP5-GOTHIC code considers the fluid in the reactor vessel to be composed of three fluid elements: liquid phase, droplet phase and vapor phase, and handles the conservation law for each, so it can simulate the thermal non-equilibrium state of water (liquid phase and droplet phase) and vapor that cannot be handled by the current LOCA safety analysis code. This code also enables detailed multidimensional analysis of the reactor vessel inside compared to the current evaluation code. These allow to realistically predict the complex thermal hydraulic behavior in the reactor vessel in the event of LOCA. When there is uncertainty in the physical model in the code that simulates an important phenomenon, by appropriately quantifying

the uncertainty, this code can be applied to the statistical safety evaluation described later.

5.2 Statistical safety evaluation method

As mentioned above, in the case of the current evaluation method, the uncertainty of each of the analysis conditions and the physical models in the analysis code is conservatively treated, so the evaluation is very conservative as a result of taking them into account for the analysis simultaneously.

On the other hand, the statistical safety evaluation method, in addition to the analysis code to be used in the method has the capability to predict the important phenomenon realistically, statistically treats various uncertainties of parameters and can result in the safety evaluation results being appropriately conservative. The statistical safety evaluation method using the MCOBRA/RELAP5-GOTHIC code was developed in accordance with "Standard Method for Safety Evaluation using Best Estimate Code Based on Uncertainty and Scaling Analyses with Statistical Approach: 2008," published by the Atomic Energy Society of Japan. The method applies an order statistic method using Latin Hypercube Sampling (LHS) and provides evaluation values that satisfy a probability of 95% or higher and a confidence level of 95% or higher. **Figure 8** illustrates the evaluation procedure using the MCOBRA/RELAP5-GOTHIC code.

Figure 9 gives an evaluation example (in the case of 153 cases) of the peak cladding temperature, which is one of the evaluation parameters, using the statistical safety evaluation method. By using the MCOBRA/RELAP5-GOTHIC code, which is a best estimate code, together with the order statistic method, the effect of parameters with uncertainty is appropriately reflected in the evaluation. The evaluation value of the peak cladding temperature obtained by the statistical method is several tens of degrees Celsius lower compared to the evaluation value obtained by the current evaluation method and the margin to the criterion is expanded. In the future, it can be expected that this statistical safety evaluation method with the MCOBRA/RELAP5-GOTHIC code supports the advancement of PWR plant operations such as long cycle operation, power uprate and the adoption of high burnup fuel.

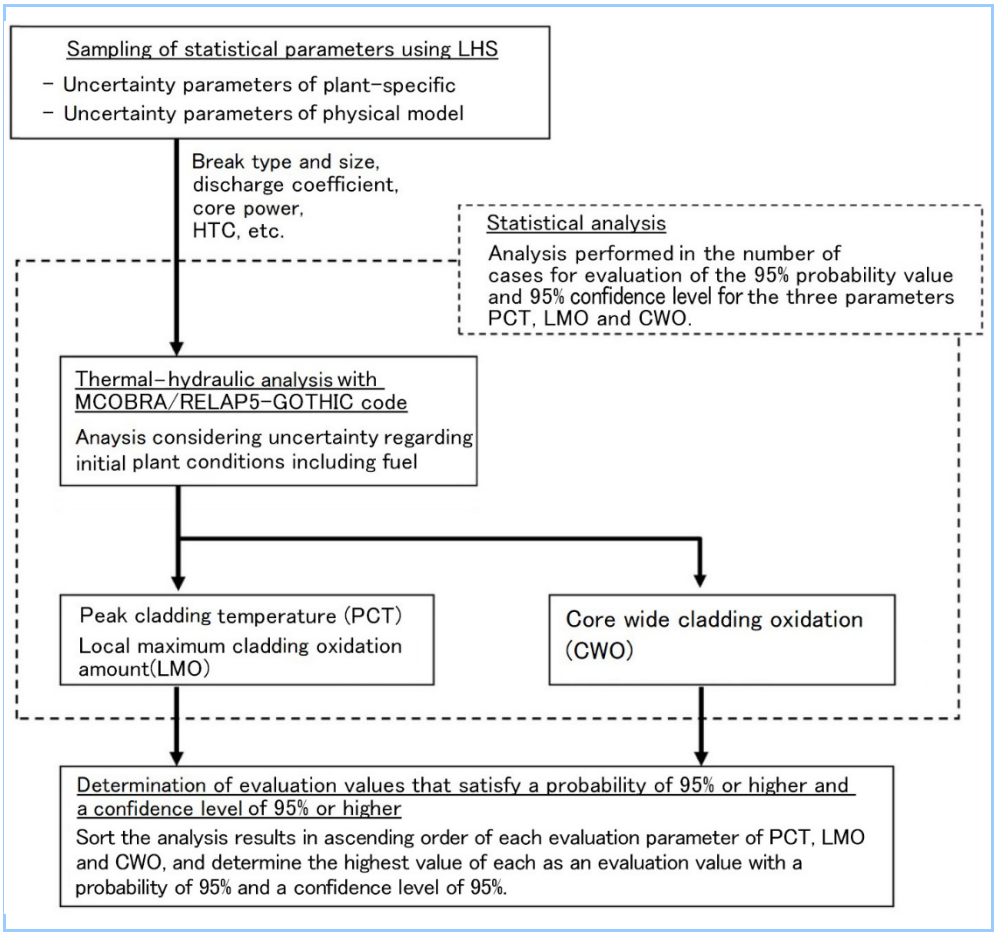

Figure 8 Statistical evaluation procedure using MCOBRA/RELAP5-GOTHIC code

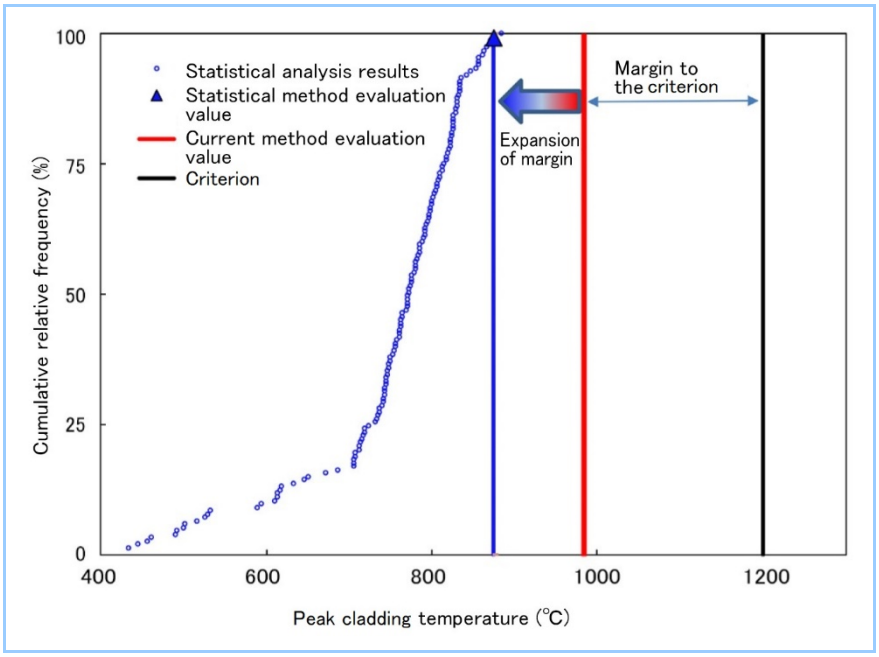

Figure 9 Analysis results by using order statistic method - Cumulative distribution of peak cladding temperature

6. Advancement of molten core evaluation technology

After the Fukushima Daiichi Accident, various safety facilities were additionally provided to existing PWR plants in Japan as a countermeasure against severe accidents. PWR plants have been restarted after confirmation of the improvement of plant safety including the effectiveness of the added facilities. Even after these restarts, we are working on the advancement of molten core evaluation technology as a research activity aimed at the further improvement of safety. As one of the technology advancements we are working on, this chapter explains an application example of the latest analysis technology using general-purpose CFD code for the spreading behavior of the molten core in a severe accident.

If the molten core is released to concrete floor of the reactor cavity, which is located at the bottom of the containment (hereinafter referred to simply as the floor surface) due to damage to the reactor vessel in a severe accident, ablation of the concrete floor material occurs because of the retained heat and decay heat of the molten core unless efficient heat removal of the molten core is possible. In addition, there is the concern that if this ablation leads to melt through of the containment basemat concrete, radioactive materials will be released into the environment. For efficient heat removal of the molten core, it is important that the molten core spreads thinly and widely without being locally deposited on the floor. In order to resolve this safety concern, we are developing technology to spread the molten core in such a manner. The following sections explains the simulation technology for predicting the spreading behavior of a molten core [(3)].

As shown in **Figure 10**, the spreading behavior of a molten core is affected by the viscosity change due to the change in temperature and composition, and the melting of floor materials. We developed a computational model that simultaneously handles typical physical phenomena of this behavior with ANSYS FLUENT code [(4)]. The FLUENT can model various physical phenomena using UDF (user defined function) in addition to the built-in physical phenomenon modeling function. In the case presented here, by simulating the change in physical properties such as viscosity, etc. due to the mixing of the molten core and floor materials, the gas generated by the reaction between the molten core and floor materials, the radiation heat from the surface of the molten core, etc., using the UDF function, the spreading behavior of the molten core is appropriately modeled.

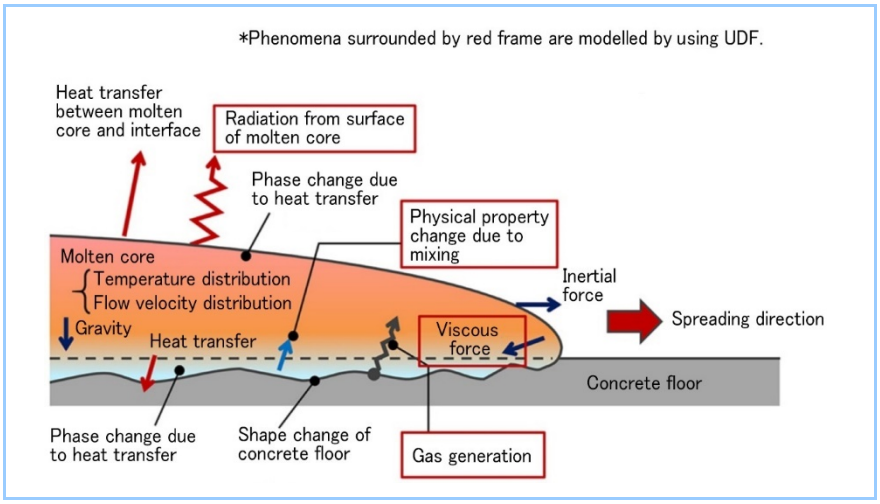

Figure 10 Typical physical phenomena related to spreading behavior of molten core

6.1 Modeling of spreading behavior of molten core

(1) Fluid physical quantity and interfacial transport

The VOF (volume of fluid) method of FLUENT was adopted for the calculation of physical quantities and interfacial transport of fluids so that the spreading behavior of the molten core can be predicted. The VOF method is a numerical calculation technique used for the calculation of the transport of free interfaces. Using the assumption that the sum of the volume fractions of each phase of multiple fluids or multiple phases in each computational mesh is equal to 1, the position of each phase (i.e. interface) is determined by obtaining the volume fraction for the calculated physical quantity in each mesh. The PLIC (piecewise linear interface calculation) solution method was employed for modelling the fluid interface, which can allow considering the interface gradient.

As for other important physical models related to the spreading behavior of the molten core, the SST-k-ω model was adopted for the turbulent flow model. Regarding the heat transfer from the surface of the molten core, only radiation heat transfer was considered since convective heat transfer on the surface of the molten core is negligibly small compared to radiation heat transfer. The heat transport between the molten core and the floor material was calculated by an energy equation that takes into account the promotion effect of turbulent heat transfer.

(2) Phenomena related to viscosity of molten core

The spreading behavior after the molten core has relocated to the floor is governed by inertial force and viscous force, it is therefore important to properly evaluate the viscosity of the molten core. The viscosity depends on the composition change and temperature change of the molten core. The viscosity change due to the composition is predicted based on the change of the chemical composition of the melt in accordance with the evaluated amount of the eroded floor concrete. For the viscosity change due to the temperature, we developed a new technique to evaluate the increase in the viscosity considering a viscosity model at temperatures above the immobilization temperature, and to model the solidification of the fluid by attenuating the momentum with a melt-solidification model at temperatures below the immobilization temperature.

(3) Phenomena related to floor surface resistance

As a molten core flows on the concrete substrate, steam ejected from the surface of the heated concrete may contribute as a resistance for the spreading behaviour. On the other hand, the generated steam between the floor surface and the molten core may have the effect of reducing the resistance acting on the molten core. Therefore, in this model, the amount of steam generated from the floor material was calculated and added as a source to the fluid contacting the floor surface. In each calculation mesh in which steam is present, the physical property values (e.g. viscosity coefficient, etc.) in the momentum equation change depending on the volume ratio of steam, so changes in wall friction can be taken into account. In addition, a phenomenon that generated steam rises in a molten core and blows into air due to

the density difference with a molten core is also modelled, so that the effect of steam existence for the flow resistance on the spreading flow of the molten core can be considered. On the other hand, the effect of reducing the frictional resistance due to the presence of steam between the floor surface and the molten core is expressed by the decrease in the viscosity coefficient due to the increase in the volume ratio of steam in the calculation mesh.

6.2 Validation calculation

To validate the newly developed spreading analysis model, reproduction calculation of the VE-U7 test of the VULCANO spreading experimental series [(4)] was performed. The VE-U7 is a test to observe the spreading behavior of a molten material simulating the prototypical corium material composition on a ceramic floor and a concrete floor in an atmospheric environment. Ceramic has a melting point higher than the temperature of the molten material used in the test and does not contain water. Therefore, no ablation of the floor material and gas generation occurs during spreading and accordingly it is considered that the influence of the interaction between the floor material and the molten material on the change in physical properties of the molten material and the floor resistance can be eliminated. On the other hand, concrete has a melting point lower than the temperature of the molten material and contains water, so it is predicted that changes in physical properties due to mixing with the ablated floor material, as well as gas generation due to interaction with the molten material. By carrying out a reproduction calculation of the VE-U7 test, it is possible to evaluate the influence on the spreading behavior of the presence or absence of the change in physical properties such as viscous force, etc. and the presence or absence of gas generation due to the mixing of the molten material and the floor material in the developed model mentioned above.

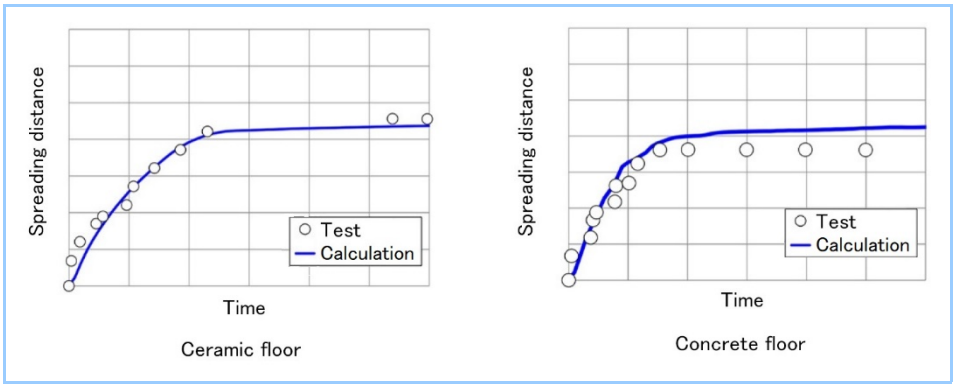

Figure 11 VULCANO VE-U7, Comparison of test results and calculation results

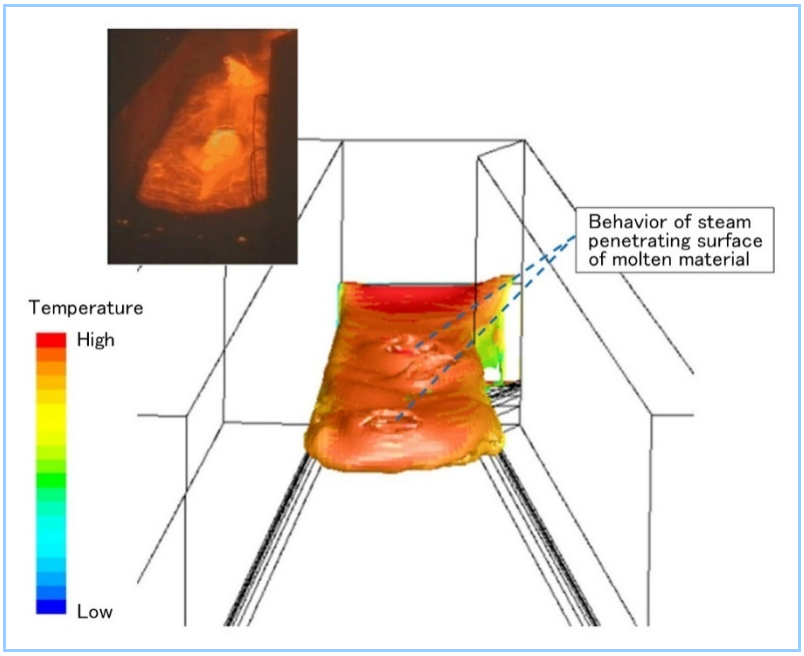

Figure 12 VULCANO VE-U7, Molten material temperature contour on concrete floor

Figure 11 compares the result of the reproduction calculation of the VE-U7 test with the test results of the spreading distance on a ceramic floor and a concrete floor. For both cases, the calculation results show good agreement with the test results with accuracy of ±20%. **Figure 12** depicts the temperature contour of the molten material at a certain time point during the spreading reproduction calculation on a concrete floor. Similar to the behavior observed in the test, the behavior of steam generated from the floor material rising in the molten material and penetrating the surface of the molten material could be reproduced. It is considered that the influence of the interaction between the molten material and concrete on the spreading behavior could be evaluated appropriately.

It is indicated by the above validation calculation results that the typical physical behavior related to the spreading of the molten core can be accurately reproduced with the CFD code.

This research was carried out with the cooperation of Heat Transfer No.2 Laboratory of the Heat Transfer Research Department, Research & Innovation Center, Mitsubishi Heavy Industries, Ltd.

7. Conclusion

This report introduced the core design (nuclear and thermal) of PWR plants, the latest technology for safety evaluation, as well as its applicability, and these technologies contribute to the accuracy improvement of design and evaluation. We will use these technologies to further improve the safety of PWR plants and for the future advancement of plant operation and core operation through improvements in areas such as longer cycle operation, increased output and the adoption of higher burn-up fuel.

In the future, the code presented in this report will be examined by the regulatory agency for these advancements of plant operation.

This report also presented the development status of the technology for molten core simulation, which is important in severe accidents. We plan to utilize this technology for future research on improving safety in existing nuclear plants and for severe accident countermeasures using next generation reactors.

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