HOT CHANNEL ANALYSIS CAPABILITY OF THE BEST-ESTIMATE MULTI-DIMENSIONAL SYSTEM CODE, MARS 3.0

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The subchannel analysis capability of MARS, a multi-dimensional thermal-hydraulic system code, has been enhanced. In particular, the turbulent mixing and void drift models for the flow-mixing phenomena in rod bundles were improved. Then, the subchannel analysis feature was combined with the existing coupled system thermal-hydraulics (T/H) and 3D reactor kinetics calculation capability of MARS. These features allow for more realistic simulations of both the hot channel behavior and the global system T/H behavior. Using the coupled features of MARS, a coupled analysis of a main steam line break (MSLB) is carried out for demonstration purposes. The results of the calculations are very reasonable and promising.

KEYWORDS : MARS, MASTER, Coupled Calculation, System T/H Analysis, Hot Channel Analysis, 3D Reactor Kinetics

1. INTRODUCTION

A best-estimate multi-dimensional system code, MARS, has been developed at Korea Atomic Energy Research Institute (KAERI) [1]. In the MARS code, the RELAP5/MOD3 [2] and COBRA-TF [3] codes had been adapted as the one-dimensional (1D) and three-dimensional (3D) T/H modules, respectively. The MARS 3D module (COBRA-TF) uses a two-fluid, three-field model for two-phase flows on rectangular Cartesian coordinates or "subchannel" coordinates. This permits an extremely flexible noding, including subchannel meshes for the flow in fuel rod bundles. Recently, the models for critical heat flux (CHF) and flow mixing phenomena in rod bundles, such as turbulent mixing and void drift, have been improved to allow the MARS 3D module to be used for subchannel analysis [4].

The three-dimensional reactor kinetics code, MASTER [5], was also coupled to MARS to attain more accurate predictions for system transients analyses that involve strong interactions between neutronic and T/H phenomena [6, 7]. This feature of MARS was again combined with the above-mentioned subchannel analysis capability. As a result, MARS now has the coupled calculation capability of system T/H, hot channel analysis, and 3D reactor kinetics.

In recent years, numerous coupled system T/H and 3D reactor kinetics codes have been developed, such as RELAP5/PARCS, TRAC-PF1/NEM, RELAP5/PANBOX, and MARS/MASTER. The performance of these coupled codes has been assessed through the OECD main steam line break (MSLB) benchmark program [8]. Among these codes, MARS has unique features. In general, transient T/H conditions of a reactor core are calculated using one-dimensional system codes, and therefore the T/H mesh size for the reactor core is very coarse. Using the MARS code, however, the COBRA-III/CP module in MASTER can be used for a more accurate core T/H simulation by utilizing a refined core T/H mesh [7]. Another unique feature of MARS is its subchannel analysis capability, which no other coupled codes have, at present.

In this study, these unique features of MARS were used to carry out a coupled analysis of the main steam line break (MSLB), focusing on the hot channel DNBR (Departure from Nucleate Boiling Ratio) behavior. Because DNB is a local phenomenon, the accuracy of the calculated DNBR depends on the accuracy of the local power distribution, as well as on the global system behavior. Taking this into consideration, a coupled calculation of system T/H and 3D reactor kinetics was performed first. Next, to obtain a more accurate DNBR behavior, another calculation with the subchannel analysis capability was performed in combination with the 3D kinetics calculation. In this calculation, the T/H simulation was limited to the reactor core and the results of the previous calculation were utilized

as T/H boundary conditions.

In Section 2, the coupled 'system T/H – 3D reactor kinetics' calculation feature of MARS is described. The coupled hot channel analysis capability of MARS is presented in Section 3. In Section 4, the coupled calculations of an MSLB accident at Yongggwang Nuclear Unit 3 (YGN-3) are described. Our conclusions are presented in section 5.

2. THE COUPLED 'SYSTEM T/H – 3D REACTOR KINETICS' CALCULATION FEATURE

Figure 1 shows the MARS code structure. The MARS code was developed as a system analysis code. The main structures of MARS are the RELAP5/MOD3 and COBRA-TF codes, which were consolidated into a single code and adapted as the 1D and 3D T/H modules of MARS, respectively. The consolidated code MARS has been further improved, in terms of both the physical models and the code structure [1]. In addition, MARS has been coupled with the 3D reactor kinetics code, MASTER, and the containment T/H codes, CONTAIN and CONTEMPT. Furthermore, MARS has been indirectly coupled with the subchannel analysis module, COBRA-III/CP, in MASTER, which is used for reactor core T/H and DNBR analyses.

As shown in Fig. 1, there are two coupling schemes between MARS and MASTER [7]. In the single coupling scheme, the MARS code is connected only to the power calculation module of MASTER. The transient core T/H conditions, such as coolant density and effective fuel temperature distributions, are determined at each time step by MARS and transferred to MASTER to update the group constants. The power distribution newly obtained by MASTER is then sent back to MARS for the T/H calculation at the next time step.

For a more accurate incorporation of local T/H feedback effects, it is necessary to enhance the resolution of the core T/H solution. The use of refined meshes in the MARS 3D module is possible, but it is not computationally efficient. The COBRA-III/CP module in MASTER is an appealing alternative, since it is capable of employing fuel assembly-sized nodes or even a quadrant of an assembly with a negligible computational burden. In the COBRA-III/CP module, the continuity, energy, axial momentum, and transverse momentum equations are solved as a boundary-value problem by employing the homogeneous equilibrium model. The boundary conditions to be specified are the inlet enthalpy, inlet axial flow, and exit pressure. With these considerations in mind, the double-coupling scheme was developed. In this scheme, the MARS 3D module provides COBRA-III/CP with lumped channel core inlet flow conditions and core exit pressures at each time step. Using these as boundary conditions, the COBRA-III/CP module performs its core T/H calculations with refined thermal-hydraulic nodes. The resulting local fuel temperatures and coolant densities are transferred to the MASTER 3D kinetics module. Finally, the local powers are sent back to the MARS 3D module by MASTER.

A similar coupling scheme was reported for the RELAP /PANBOX/COBRA (R/P/C) coupled code [8]. The primary difference between the R/P/C and MARS codes lies in the flow representation scheme for the reactor vessel. MARS employs the COBRA-TF module for a full 3D flow representation, whereas the R/P/C code uses the one-dimensional system code, RELAP5.

The results of the coupled calculations with and without the refined core T/H mesh are slightly different in terms

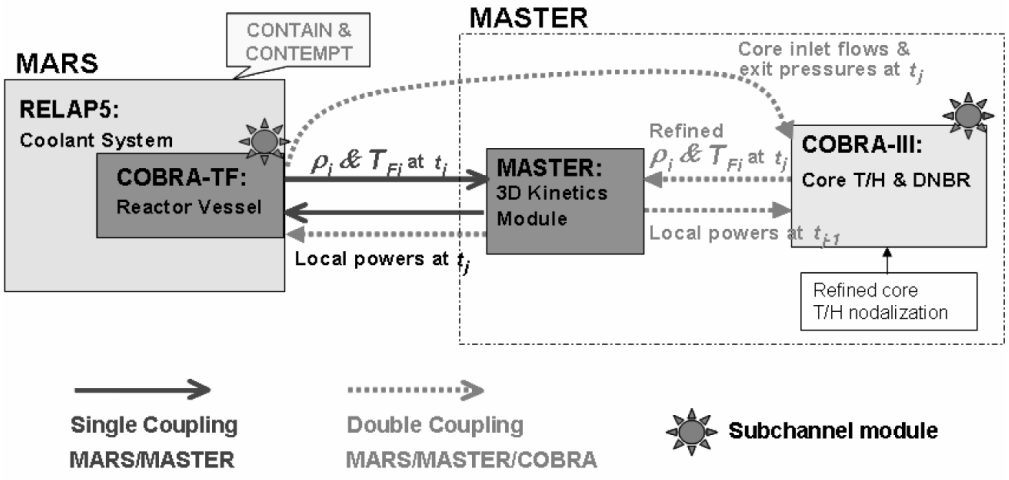


Fig. 1. The MARS Code Structure and Its Coupling Schemes

of the global core power behavior. However, the accuracy of the local power prediction is greatly enhanced in the double-coupling scheme. In the case of the OECD MSLB benchmark problem, a detailed thermal feedback using the double-coupling scheme reduces the peak local power by up to about 25 %, in comparison with that of the single -coupling scheme [7].

3. THE COUPLED HOT CHANNEL ANALYSIS CAPABILITY

There are two methods in MARS that can be used for a hot channel analysis: the simplified DNBR calculation model in MASTER and the MARS 3D module.

3.1 The Simplified DNBR Calculation Model in MASTER

This DNBR model can be used only when the double -coupling scheme between MARS and MASTER is activated. In this case, the COBRA-III/CP module simulates the core T/H conditions with a refined T/H mesh. However, since the COBRA-III/CP module is not a stand-alone code, but is adapted as the internal T/H solver of MASTER, the radial T/H mesh is limited to fuel assembly-sized nodes or a quadrant of an assembly and, thus, a detailed hot channel analysis using the subchannel meshes is not possible in its present form. Instead, MASTER employs a simplified DNBR calculation model, in which the CHF of the hot channel in each assembly is estimated from the coarse-channel T/H calculation results.

To conservatively estimate the hot channel CHF from the assembly average T/H calculation results, a special hot channel model is used that employs the pin-to-box factor and the flow penalty factor [7]. Specifically, the local enthalpy is estimated by multiplying the pin-to-box factor with the channel average enthalpy rise. The pin-to-box factor is defined as the ratio of the hottest pin power to the average pin power in the fuel assembly, and it is determined for each assembly from the pin power reconstruction calculation at each time step. The mass flux is reduced by a certain fraction for conservatism. The local enthalpy and the mass flux are then used to determine the CHF. The actual hot channel heat flux is obtained by multiplying the pin-to-box factor with the channel average heat flux. It is augmented by an uncertainty factor specified in the input. Then, the hot channel DNBR is determined.

3.2 The Hot Channel Analysis Using the MARS 3D Module

The MARS 3D module can be used for a hot channel analysis. It adopts a two-fluid, three-field model for two-phase flows on rectangular Cartesian or "subchannel" coordinates. Thus, the use of subchannel meshes is possible.

Generally, the fluid flow in the subchannels is an axially dominant one-dimensional flow. However, there is flow mixing between the adjacent channels and, in the case of a two-phase flow, the flow-mixing rate significantly increases. Therefore, the accuracy of the subchannel module is strongly dependent on the modeling of inter-channel exchanges. These exchange phenomena are generally delineated into three components [9]: diversion cross flow, turbulent mixing, and void drift. In the MARS 3D module, the diversion cross flow is modeled by solving the transverse momentum equations. For turbulent mixing and void drift between the adjacent subchannels, Lahey's model was employed and has been modified [10], based on the works of Kelly [11] and Hwang *et al.* [12]. In the modified model, the net mass flux of a gas phase from subchannel *i* to subchannel *j* caused by turbulent mixing and void drift is

$$w''_{g,i-j} = \left(\frac{\varepsilon}{l}\right)_{1\phi} \theta \left\{ (\alpha\rho)_{g,i} - (\alpha\rho)_{g,j} - K_{VD} \frac{G_i - G_j}{\overline{G}_{i,j}} \rho_{g,i-j} \right\}, \quad (1)$$

where ε is the eddy diffusivity and l is the subchannel mixing length. The term $(\varepsilon/l)_{1\phi}$ has a unit of velocity and is sometimes called single-phase "turbulent velocity." The term θ is a two-phase multiplier for the turbulent velocity. The terms α and ρ are the void fraction and density, respectively. The term G_i is the total mass flux at channel *i*. The term K_{VD} is the void drift coefficient. Similarly, the net mass flux of the liquid phase from subchannel *i* to subchannel *j* caused by turbulent mixing and void drift is

$$w''_{l,i-j} = \left(\frac{\varepsilon}{l}\right)_{1\phi} \theta \left\{ (\alpha\rho)_{l,i} - (\alpha\rho)_{l,j} + K_{VD} \frac{G_i - G_j}{\overline{G}_{i,j}} \rho_{l,i-j} \right\}. \quad (2)$$

For the entrained-liquid phase in the MARS 3D module, the mixing model is not applied. Equations (1) and (2) are added to the right-hand sides of the continuity equations for the vapor phase and the continuous liquid phase, respectively. In addition, energy and momentum exchange terms due to turbulent mixing and void drift are also taken into account in the governing equations.

Recently, the modified Lahey model in the MARS 3D module was assessed using the ISPRA 16-rod bundle test [13] and the GE 9-rod bundle test [14] data. These tests represent typical pressurized water reactor (PWR) and boiling water reactor (BWR) core T/H conditions, which were conducted at the pressures of 16.0 MPa and 6.9 MPa, respectively. From the results of the assessments [4, 10], it was found that the optimum void drift coefficient depends on the system pressure. To confirm the effect of the pressure on the void drift phenomena, subchannel-mixing tests that were performed under atmospheric pressure conditions [15] were also simulated using MARS. Finally, the void drift coefficient was represented as a function of the pressure:

$$K_{VD} = 0.112 + 16.4e^{-0.329P} \quad (3)$$

where P is pressure in MPa. This coefficient was chosen to minimize the root-mean-square error in the predictions of subchannel qualities or void fractions of the three tests.

In addition to the flow mixing model, the CHF correlation of the MARS 3D module was replaced with the AECL lookup table [16]. As a result, the MARS 3D module can be used for a hot channel analysis.

4. THE COUPLED CALCULATIONS OF AN MSLB ACCIDENT

Using the coupled feature of MARS, a hypothetical MSLB accident at YGN-3 was analyzed for demonstration purposes. The primary objective of the coupled calculation is to obtain realistic simulation results for an MSLB accident.

The major safety concerns of an MSLB accident are: (i) to minimize the possibility of a post-trip return-to-power and (ii) to minimize the off-site dose at the site exclusion area boundary. The off-site dose during an MSLB strongly depends on the fuel failure due to DNB before the reactor trip. So, an analysis focused on the DNB behavior before the reactor trip is called a "pre-trip DNB analysis." In this work, an MSLB was simulated in terms of a pre-trip DNB analysis.

Two calculations were done in this analysis. First, in *Step 1*, a coupled calculation of system T/H and 3D reactor kinetics with the refined core T/H nodalization feature was performed to obtain both the local DNBR behavior and the global system behavior. For *Step 1*, the following input data was prepared:

- The MARS input for the reactor coolant system (RCS), including the MARS 3D module input for the reactor vessel T/H.
- The MASTER input for the reactor kinetics, including the COBRA-III/CP input for the refined core T/H calculation.
- The mapping input for the neutronic and T/H meshes.

Transient DNBR behaviors are obtained from this calculation, and then the location of the minimum DNBR (MDNBR) is identified. Next, in *Step 2*, when a more accurate DNBR behavior is needed, another coupled calculation, with subchannel analysis capability, is carried out. For *Step 2*, the results of *Step 1* are used as boundary conditions.

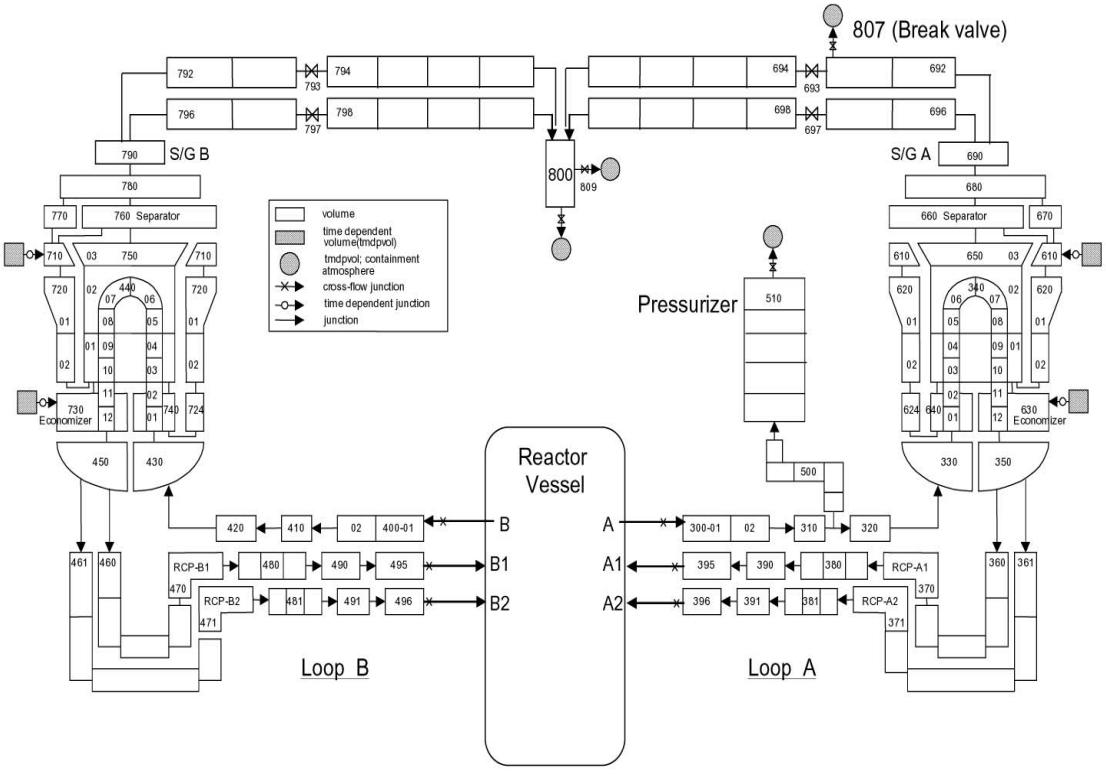


Fig. 2. MARS Nodalization for the YGN-3 Reactor Coolant System

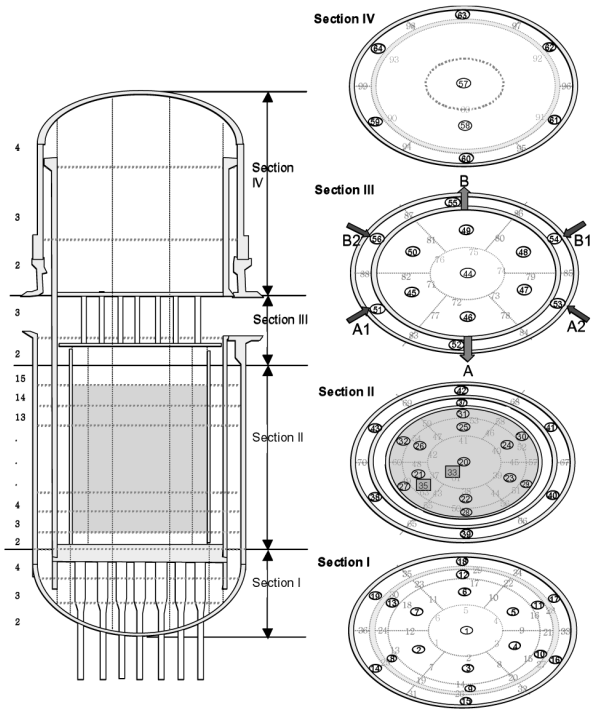


Fig. 3. MARS Nodalization for the Reactor Vessel

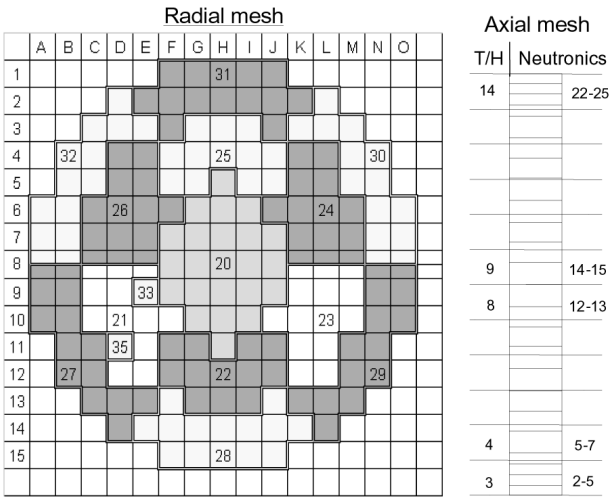


Fig. 4. Neutronic and T/H Meshes for the Reactor Core: A Square, Representing the Fuel Assembly, is Divided Into 4 Neutronic Meshes. Channels 20 Through 32, 33, and 35 are T/H Meshes

4.1 Input Models for the Coupled Calculation

Yonggwang Nuclear Unit 3 (YGN-3) is a two-loop, 2815 MWt pressurized water reactor plant [17]. Figure 2 shows the RCS nodalization, which consists of 161 volumes and 167 junctions. The break is assumed to occur at the main steam line of steam generator A. The break is located at the upstream of the main steam line isolation valve (MSIV). The break size is set to the flow area of the main steam line, but the break flow is actually limited by the flow restrictors that are installed at the exit of the steam generators.

The reactor vessel was modeled using the MARS 3D module. In the 3D module, single mesh "cells" are stacked upon one another to form a "channel." Channels can be grouped side-by-side to form a horizontal section of the reactor. Where a lateral flow exists, "gaps" are used to model the flow path between the adjacent channels. These groups of channels, referred to as a "section," are stacked upon one another to form the complete vessel mesh. Figure 3 shows the reactor vessel nodalization, which consists of 4 sections, 62 channels, and 99 gaps [18]. To model the asymmetric flow behaviors during the transient, the downcomer was divided into six circumferential channels. The active core region was divided into 13 channels; each channel has a fuel rod model, which represents the average fuel rod in the channel. Two hot fuel assemblies (channels 33 and 35) are also modeled to allow a closer look at the local phenomena. These are chosen from the core design data.

Figure 4 shows both the neutronic and T/H meshes for the reactor core. Each of the 177 fuel assemblies was individually modeled as four radial meshes; in addition, 264 radial reflector meshes were modeled. The fuel assemblies and the reflectors were divided into 26 axial neutronic meshes. The 15 radial meshes shown in Fig. 4 (i.e., channels 20 through 32, 33, and 35) are the T/H meshes for the core. Each of the channels 20 through 32 contains more than 10 fuel assemblies. Since the mesh structures for the neutronic and T/H calculations are different, a mapping input is required to define the correspondence between them. A stuck rod was assumed in the broken side to model the local power peaking. The results of the MSLB analysis strongly depend on the core design data. In this work, the postulated core design data of an 18 -month cycle was used [19].

The initial and boundary conditions are summarized in Table 1. The gap conductance was set to the maximum, and the reactor trip was assumed to be delayed for as long as possible by increasing the trip setpoint for the core protection calculator variable over power to 121 % of the rated power.

4.2 The Results of the Coupled 'System T/H – 3D Reactor Kinetics' Calculation

The MSLB accident begins by the instantaneous opening of break valve 807, shown in Fig. 2. As soon as the break occurs, the steam flows at the main steam lines increase rapidly, as shown in Fig. 5. The main steam line isolation signal occurs at 21.3 seconds when the steam

generator pressure decreases below 5.44 MPa. Because of the valve stroke time, the MSIVs are completely closed at 26.3 seconds. However, the steam in main steam line A1 continues to blow down, because the break is not isolated by the MSIVs. The overcooling caused by the MSLB in turn causes the primary-side coolant temperature to decrease, as shown in Fig. 6. This results in the depressurization of the primary side (See Fig. 7). Before the coolant in the steam generator U-tubes reaches the reactor core, coolant expansion inside the core resulting from the depressurization causes the total core power to decrease for a short period. However, as the coolant reaches the reactor core, the core power gradually increases, due to the positive reactivity insertion effect. Figure 8 shows the total core power behavior. When the core power reaches 3406 MW (121 % of the rated power) at 14.7 seconds after the break, the reactor trip signal occurs.

Figure 9 shows the radial power distribution at 0.0 s, 14.6 s, and 30 s. Before the reactor trip, the radial power distribution is almost symmetric. However, after the MSIV closure, an asymmetric core cooling occurs, resulting in an asymmetric radial power distribution, as shown in Fig. 9(c). The extent of the asymmetry increases until the broken-side steam generator empties.

Figure 10 shows the MDNBR behaviors in the hot fuel assembly and the hot channel calculated by MASTER. The pin-to-box factor and the flow penalty factor were 1.15 and 0.9, respectively. The initial MDNBR of the hot pin is 2.96 and the lowest value during the transient is 2.54, at around 14.6 seconds. The MDNBR rapidly increases after the reactor trip. Thus, the calculation was terminated at 30 seconds.

Table 1. Initial and Boundary Conditions for the YGN-3 MSLB Accident Analysis

Parameter	MARS	Comments
Core power, MWt	2871	The rated power: 2815 MWt
Core inlet coolant temp., K	571.7	
Core mass flow rate, kg/s	14193.4	
Pressurizer pressure, MPa	160.3	
Pressurizer water volume, m^3	30.99	
Gap HTC ($W/m^2.K$)	37851.7	For MDNBR calculation.
Steam generator inventory, kg	87015	
Core burn-up	EOC	
Auxiliary feed water	-	Supplied to the broken SG for 700 seconds after the MSLB occurs.

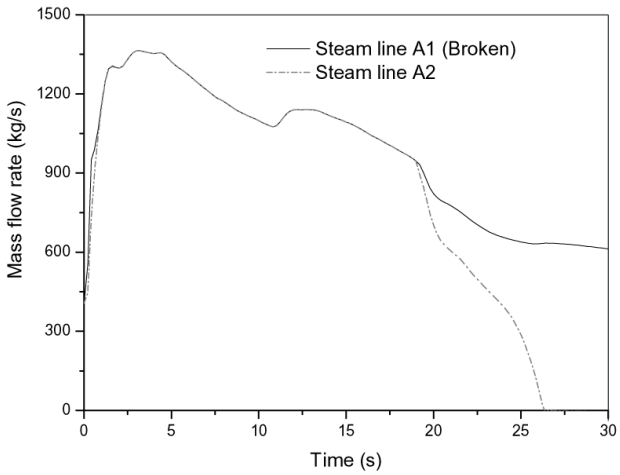


Fig. 5. Steam Flow at the Main Steam Lines

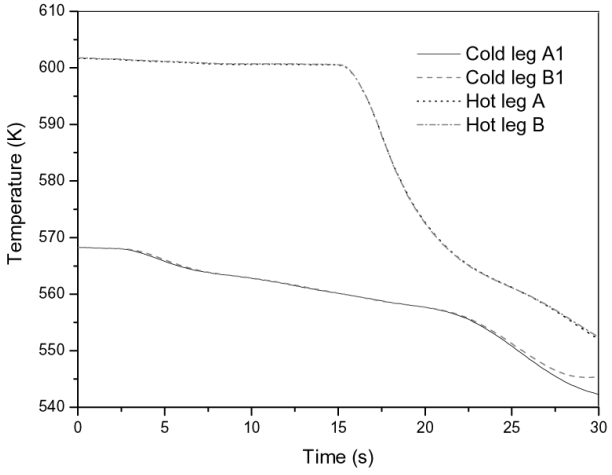


Fig. 6. RCS Coolant Temperatures

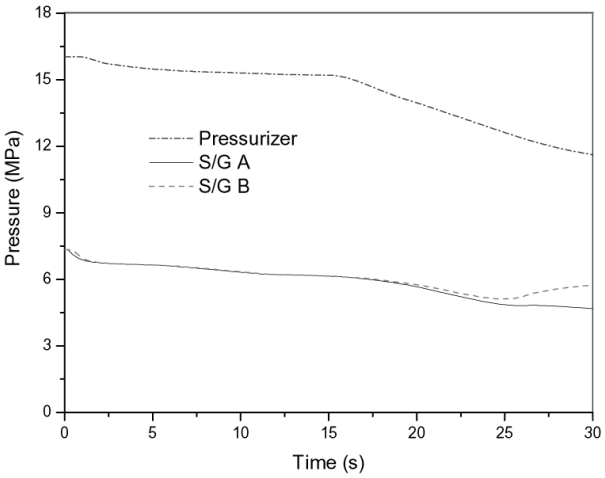


Fig. 7. Pressurizer and Steam Generator Pressures

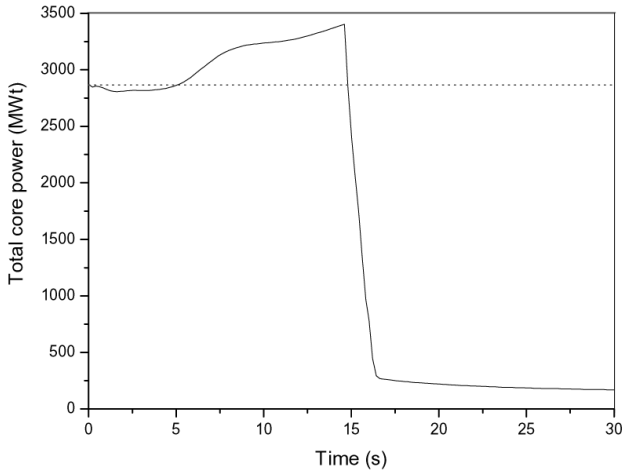
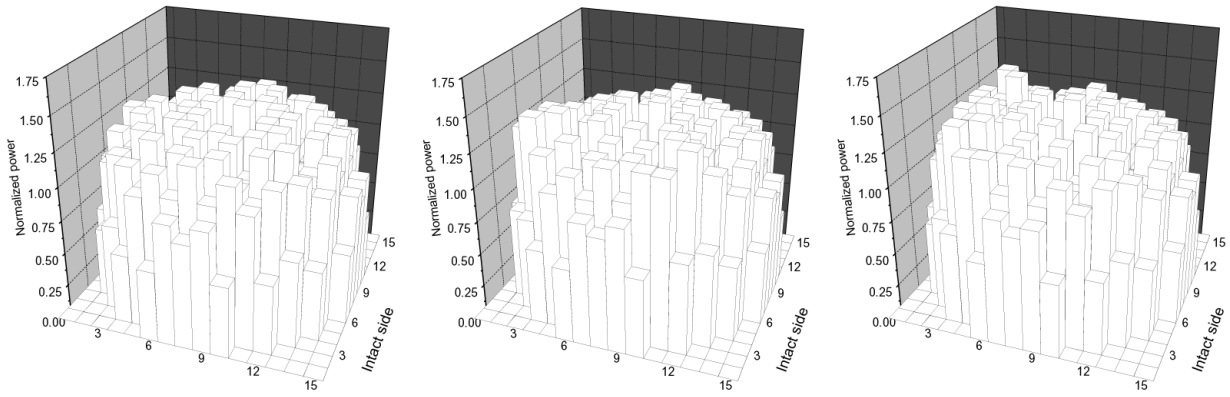


Fig. 8. Total Core Power Behavior



(a) 2871 MW at 0 s. (b) 3392 MW at 14.6 s. (c) 168.1 MW at 30 s.

Fig. 9. Radial Power Distribution at 0, 14.6 and 30 s

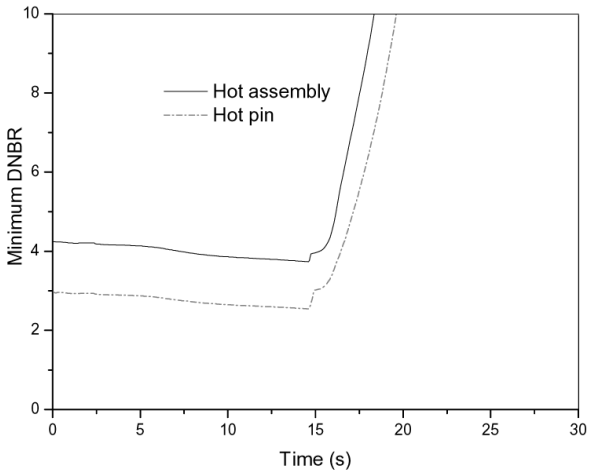


Fig. 10. MDNBR Behaviors

4.3 The Results of Detailed DNBR Calculation, Step 2

To obtain a more accurate DNBR prediction, an additional coupled calculation, Step 2, with subchannel analysis capability, is carried out. In Step 2, the T/H simulation is limited to the reactor core, including the lower and upper plenums, and more detailed T/H meshes are used for the reactor core.

From the results of Step 1, the MDBNR was found to occur at the fuel assembly D11 (channel 35 in Fig. 4). Therefore, in the Step 2 analysis, the quadrant of the hot assembly, D11, is divided into 18 subchannels and 4 lumped subchannels, as shown in Fig. 11. Therefore, the entire active core is modeled by 35 flow channels. In the subchannel meshes, 25 fuel rods are separately modeled. The MDNBR is expected to occur at one of these 25 rods. In each of the 4 lumped subchannels, an average rod model

is included. The optimum configuration of the subchannel mesh might be different according to the transients and pin power distribution, and this configuration warrants further discussion. The pin powers of the 25 rods in the subchannel meshes are provided by the MASTER code. To overcome the uncertainties in the pin power reconstruction, a user-specified input parameter is multiplied by the pin powers.

To impose the inlet and exit flow conditions of the reactor core, the lower and upper plenums are included in the T/H input model. The results of Step 1 are used as boundary conditions, including the channel-wise inlet flow rate and temperature (or enthalpy), and the exit pressure. The boundary conditions are specified using time-dependent volumes and junctions. The reactor trip time is also taken from the results of Step 1.

For the MASTER code, the same input data as Step 1 is used in the Step 2 calculation. However, the mapping input for the neutronic and T/H meshes is changed because of the changes in the core T/H input.

Using these input data, an additional coupled calculation was performed. Figure 12 compares the total core power behaviors of Step 1 and Step 2. Theoretically, the two results should be nearly the same because of the different core T/H models. However, the difference shown in Fig. 12 is greater than expected. This is could be because of errors incurred during the transfer of the boundary conditions from the Step 1 to Step 2 calculations. However, the deviations in the core powers are practically negligible.

In Figure 13, the MDNBR behaviors of Step 1 and Step 2 are compared. The CHF was calculated using the AECL lookup table in the two calculations. In the results of Step 2, the initial MDNBR is 3.29, and the lowest value is 2.91 at ~14.6 seconds, which occurs at rod "e3" of the hot subassembly shown in Fig. 11. The trend of the MDNBR behavior of Step 2 is nearly identical to that of Step 1, but it is always greater by a difference of ~0.33. This difference originates from the way the hot channel flow conditions are obtained.

As discussed in Section 3.1, the hot channel CHF in Step 1 is estimated from the results of the fuel assembly-sized channel T/H calculation. That is, a special hot channel model is needed, one which employs the pin-to-box factor and the flow penalty factor to conservatively estimate the hot channel CHF by increasing the local enthalpy rise along the hot channel and decreasing the mass flux. However, in the Step 2 calculation, the local flow conditions are realistically predicted by the MARS 3D module, which removes the conservatism introduced in Step 1.

5. CONCLUSIONS

The performance of the MARS code has been enhanced. In particular, the turbulent mixing and void drift models in the MARS 3D module, which are used for the flow-mixing phenomena in rod bundles, were improved. Then, the subchannel analysis feature was combined with the existing coupled calculation capability of MARS. As a result, MARS has a coupled calculation feature for the system T/H, hot channel analysis, 3D reactor kinetics, and

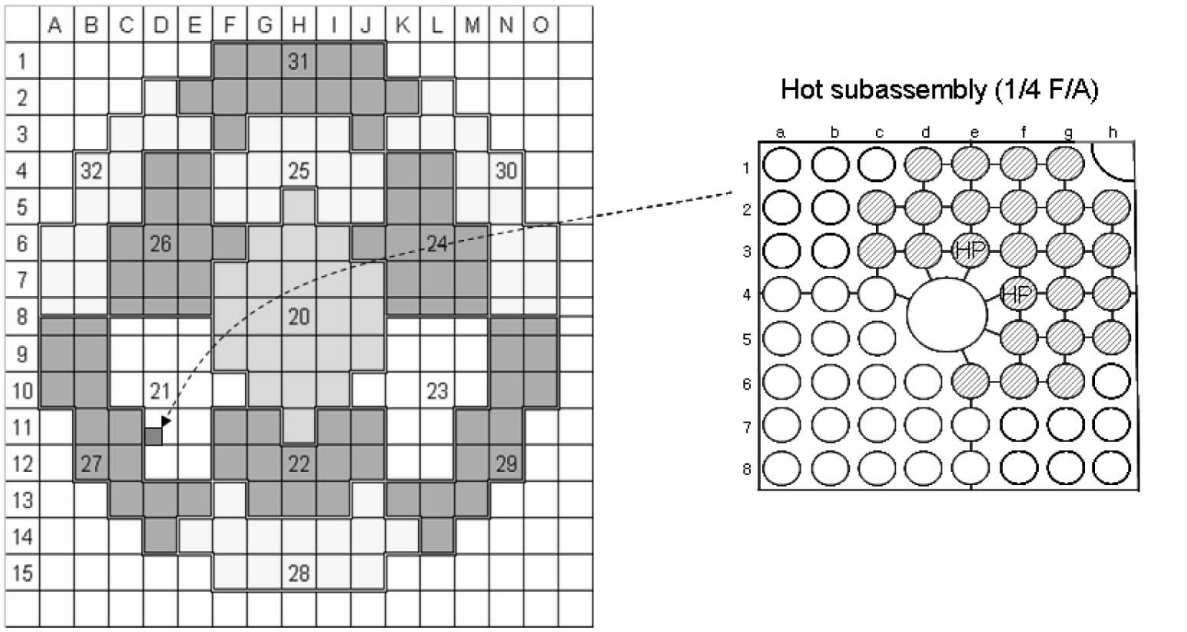


Fig. 11. Neutronic and T/H Meshes for the Reactor Core: The Subchannel Meshes for the Quadrant of the Hot Assembly are Added

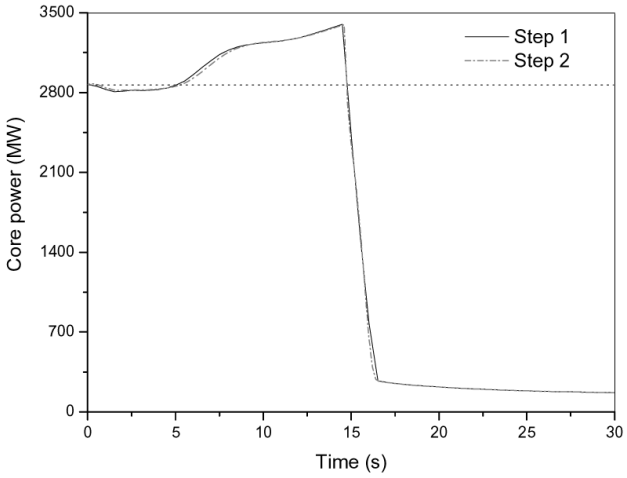


Fig. 12. Comparison of the Total Core Powers

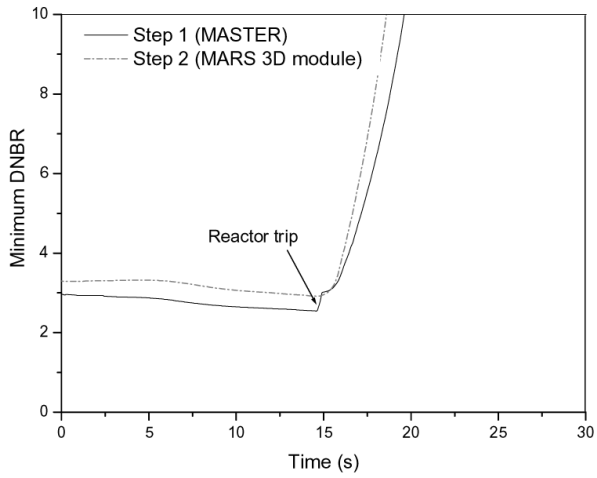


Fig. 13. Comparison of the MDNBR Behaviors

containment T/H. This feature allows for more realistic simulations of nuclear system transients.

In this paper, a demonstrative, coupled analysis for the MSLB accident was presented. The calculations were performed in two steps. First, a coupled calculation of the system T/H and 3D reactor kinetics, with the refined core T/H nodalization feature, was conducted to obtain the global system T/H and local DNBR behaviors. Next, an additional coupled calculation, with subchannel meshes, was carried out to obtain a more accurate DNBR prediction. In this second step, the T/H simulation scope was limited to the reactor core, and the results of the previous step were used as the boundary conditions for the reactor core T/H calculation.

The results of the coupled calculations using the MARS code seem to be very reasonable and promising. However, for practical applications, further development and validation of the application methods are needed.

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