SIMULATION OF BOILING WATER REACTOR ONE-PUMP TRIP TRANSIENT BY SIMULATE-3K

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Abstract

To validate models relevant to the transitional in-core 3-D power behavior, the 3-D core kinetics code SIMULATE-3K was applied to an analysis of one-pump trip test performed in a BWR-5 plant which includes large distortion of the power distribution due to the local insertion of control rods. The core boundary conditions required by SIMULATE-3K were calculated by the plant system analysis code RETRAN-3D. It was found that the transitional APRM signal and LPRM signals calculated by SIMULATE-3K agreed well with the measured data. The results showed that the transitional in-core 3-D power behavior can be appropriately predicted by SIMULATE-3K.

1. Introduction

Recently in Japan, there has been a movement that the present conservative licensing analysis shall be replaced with the statistical safety analysis. Atomic Energy Society of Japan (AESJ) published “Standard Method for Safety Evaluation using Best Estimate Code Based on Uncertainty and Scaling Analyses with Statistical Approach” (hereafter named as AESJ-SSE (Statistical Safety Evaluation) method) as a standard for applying the best-estimate plus uncertainty (BEPU) method to the analyses of design basis transients and accidents [1][2].

Based on AESJ-SSE, validation of 3-D core kinetics analysis code SIMULATE-3K [3] is underway to confirm applicability to the best-estimate and statistic safety evaluation of design basis transients of boiling water reactors (BWRs). As a part of this work, the transitional in-core 3-D power behavior is one of important validation items. Prediction accuracy of the transitional in-core 3D power behavior, i.e. the temporal variation rate of local power, is closely related to prediction reliability of the boiling transition critical power ratio (CPR), which is a major safety parameter in the design basis transients.

In this validation, the experimental database was composed of a series of measured data obtained in the recirculation pump trip test (one pump trip test) performed in Kashiwazaki-Kariwa unit 3 nuclear power plant of Tokyo Electric Power Company. The calculated transitional in-core 3-D power distribution was converted into simulated LPRM (local power range monitor) signals and they were compared with the measured LPRM data. This test is suitable for validation of the transitional in-core 3-D power behavior, since localized flux shape distortion was observed due to eight selected rods insertion (SRI) at 10 seconds after an initiation of the event.

The following sections of this paper will describe an outline of the one pump trip test (section 2) and a validation method (section 3). After some validation results (section 4) are shown, the conclusions will be given (section 5).

2. Outline of test

The one pump trip test was performed as a part of the start-up test of the initial loaded core to demonstrate the transient response during large flow reduction. Table 1 shows test conditions. A priori selected eight control rods were inserted in the test, in order to protect the core against the neutronic-thermal-hydraulic instability and keep the reactor at a sufficiently stabilized condition. Figure 1(a) and (b) show the control rod patterns at the initial and the final state, respectively. After this selected rods insertion, the 3-D power distribution was locally distorted significantly during the test.

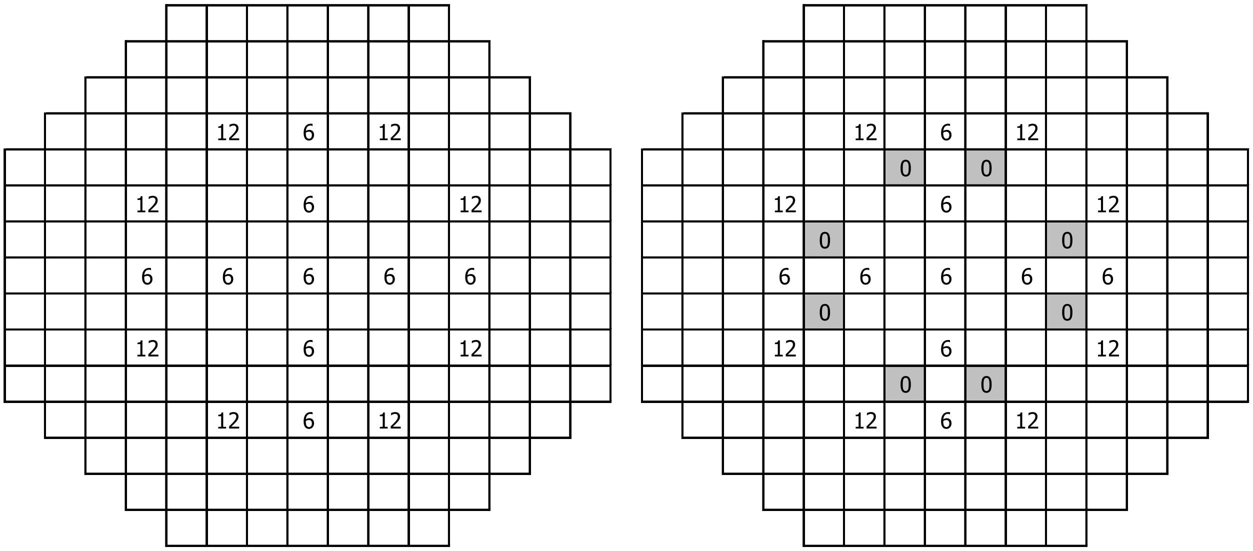
In this test, one recirculation pump trip triggers rapid decrease in the core flow rate, then the core power declines gradually due to delayed increase in the void fraction. Then the core power showed sudden decrease caused by the insertion of eight control rods at 10 seconds. Then the core power gradually recovers due to collapse of the void and settled at the steady state.

There are forty-three in-core detector strings in this plant, and each string has four LPRMs (Channel A-D) at fixed elevations as shown in Figure 2. The signals of major plant parameters including all LPRM responses, the jet pump flow, the dome pressure etc. were sampled by the data recorder, and stored as digital data. The measured LPRM signal data has sufficient temporal and spatial resolutions from the viewpoint of the code validation.

Figures 3(a) and (b) depict the planar axial averaged flux shapes at the initial and the final state. These shapes were smoothed from the point-wise LPRM responses. These figures indicate that the inserted eight control rods significantly distorted the local flux shape, which made this transient test a valuable benchmark for validating 3-D kinetic models.

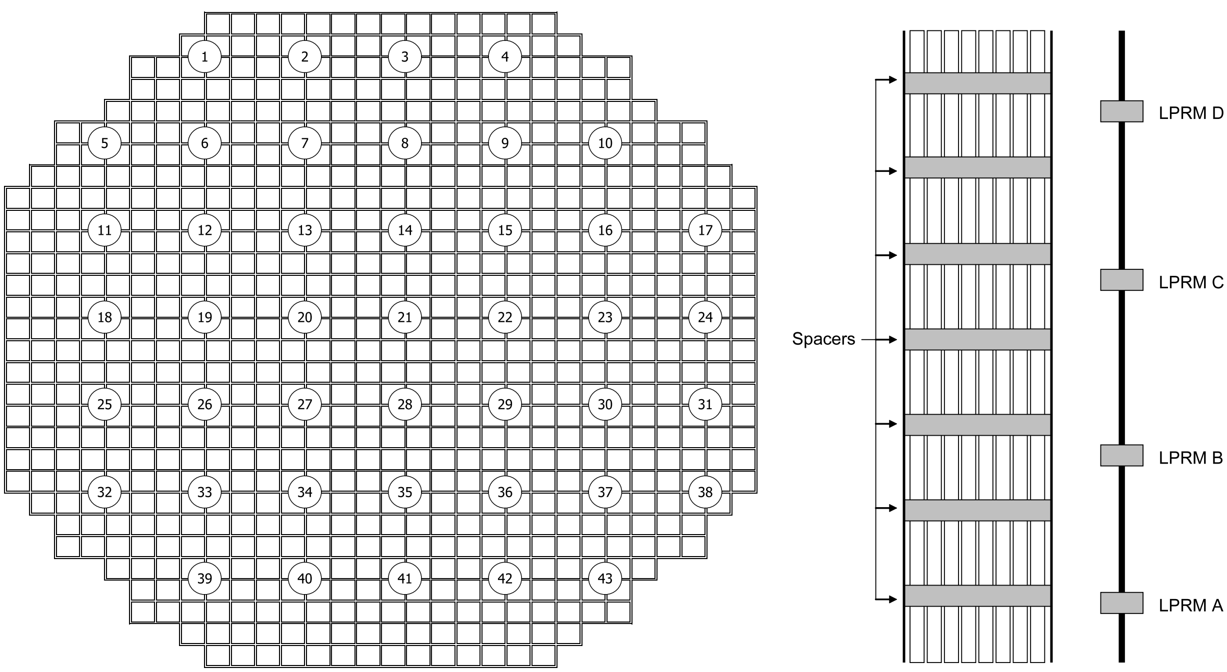
Table 1 Conditions of one pump trip test

Plant type	BWR-5
Rated power	3293 MW (thermal)
Rated flow	13417 kg/s
Number of fuel bundles	764
Type of fuel bundles	high burn-up 8x8 fuel (60 fuel rods and 1 water rod)
Initial condition	98% power, 86% flow
Tripped instrument	one of two recirculation pumps (loop A)
Suppression mechanism	eight selected rods insertion (SRI) at 10 seconds after the trip



(a) Initial state (b) Final state

Figure 1 Control rod pattern at initial and final states
*Integers denote position of control rods
(0:Fully insertion, 48:Fully withdrawn [shown as blank]).
Shaded area indicates locations of selected rods insertion (SRI)



(a) Planar LPRM locations (b) Axial LPRM locations

Figure 2 Planar and axial LPRM locations in BWR-5 plant

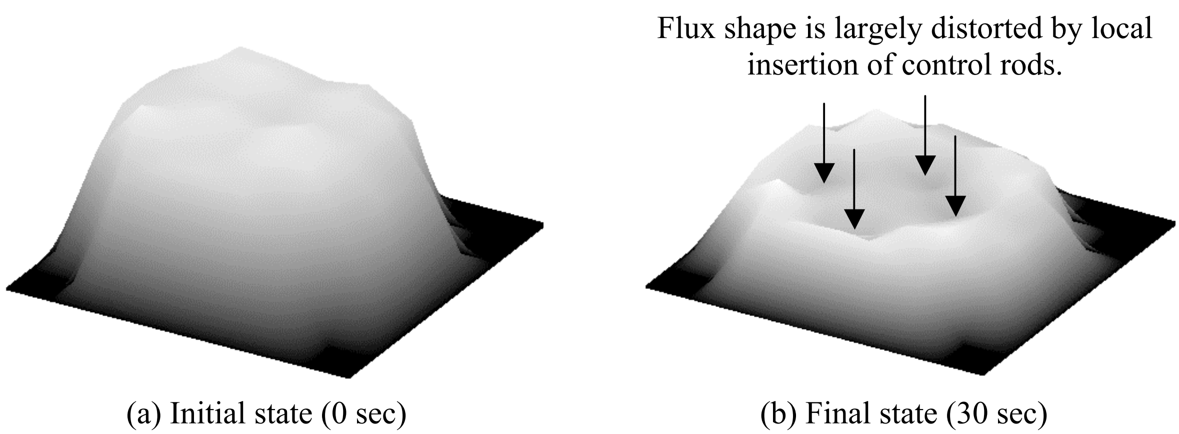


Figure 3 Flux distributions smoothed from average of LPRM responses at four axial locations

3. Validation method

3.1 Validation procedure

Figure 4 shows a validation procedure. Accuracy of the initial core power distribution and the temporal variation of core boundary condition during the test are paramount from the viewpoint of accurate prediction of the control rod worth, the void fraction and the fuel temperature. The initial core power distribution was calculated by the steady-state core analysis code SIMULATE-3 [4] based on the core tracking analysis with faithfully tracing the operating condition recorded by the on-line core simulator. Since core boundary conditions required by SIMULATE-3K were not directly measured, these parameters were calculated by the plant system analysis code RETRAN-3D [5]. In this calculation, the plant transient sequence of the one pump trip test from the initial condition was traced. The transitional LPRM responses calculated by SIMULATE-3K were compared with the corresponding measured data. Outlines of RETRAN-3D and SIMULATE-3K are described in section 3.2 and 3.3, respectively.

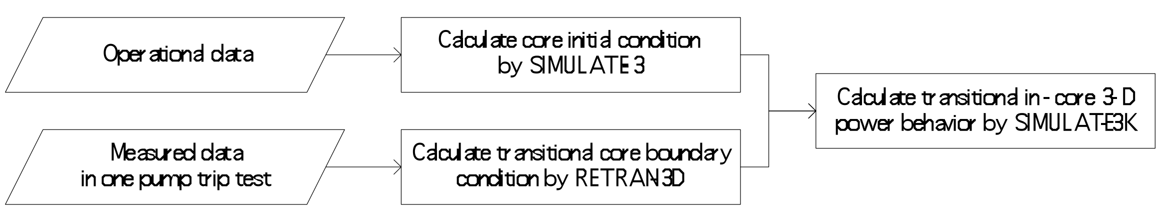


Figure 4 Validation procedure

3.2 RETRAN-3D plant model

RETRAN-3D simulates the plant system behavior. The code is capable of simulating a trip sequence from the steady-state condition and calculates the transient behavior of the entire plant system. The plant model includes the core, the upper plenum, the separator and dryer, the steam dome, main steam lines, the downcommer, the recirculation loop, and the lower plenum. The control system of the feedwater flow, the recirculation flow, and the dome pressure are also modeled to simulate the control mechanism during the transient.

As for the thermal-hydraulic model, 1-D model is applied. The governing equations are the mass, the momentum and the energy conservation equation. The void fraction is calculated by the drift-flux correlation. The pressure drop is calculated by correlations of the friction factor and the local losses.

Since the core power history can be estimated based on the averaged neutron flux data (i.e. averaged power range monitor: APRM signal) and the decay heat, the core power was not calculated by RETRAN-3D but given as the time-dependent boundary condition in this study.

3.3 SIMULATE-3K core model

SIMULATE-3K calculates the core kinetics. The code is implemented with the cross-section format conformable to the assembly-homogenized cross-sections calculated by the lattice code CASMO-4 [6], and the neutronic models consistent with those of SIMULATE-3, which are based on the steady-state two-group three-dimensional neutron diffusion equation discretized based on the advanced nodal method and the discontinuity factor. As for the kinetics model, the time-dependent neutron diffusion equation is solved by the frequency transform method. The reactivity feedback is calculated based on the variation of the cross-section mainly induced by changes of the control rod position, the void fraction and the fuel temperature.

As for the thermal-hydraulic model, the 1-D parallel channel model with 24 axial nodes is applied for all the individual fuel assemblies. The governing equations are the mass, the momentum and the energy conservation equation. The void fraction is calculated by the drift-flux correlation and the subcooled boiling model. The pressure drop is calculated by correlations of the friction factor and the local losses. The fuel heat transfer model is based on the 1-D radial heat conduction equation. Material properties are calculated as functions of core state variables. The heat transfer coefficient of the clad surface is calculated by correlations of the forced convection and the nucleate boiling heat transfer.

Core boundary conditions required by SIMULATE-3K are time tables of the inlet flow, the inlet temperature and the outlet pressure. In this study, these parameters were calculated by RETRAN-3D and given to SIMULATE-3K as the time-dependent boundary conditions.

4. Calculation results

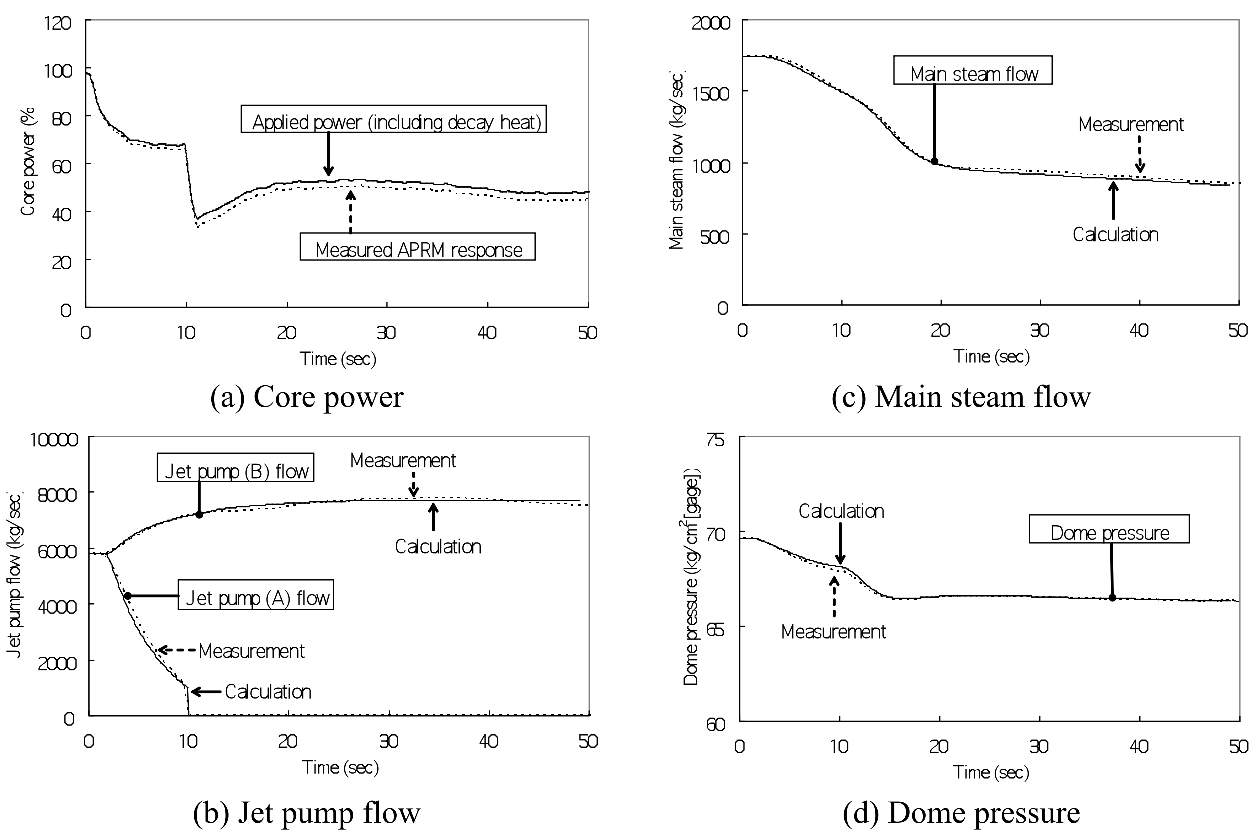
4.1 Core boundary condition

To ensure validity of the calculated core boundary conditions, trends of major plant parameters during this transient event were compared with the measured data as shown in Figure 5. Dashed lines are measured values and solid lines are calculated values. Note that the solid line of Figure 5 (a) is not calculated by RETRAN-3D, but separately evaluated as a summation of the

APRM data and the decay heat, and given as the boundary condition for the plant transient calculation.

The jet pump (A) flow decreases due to the recirculation pump trip, then the jet pump (B) flow gradually increases due to the reduction of the flow resistance. Considering that the APRM signal declines just after the trip, the observed delay of flow variation is attributed to an uncertainty in the time lag included in the flow measurement system. A notched trend observed in the jet pump (A) flow at about 10 seconds after the trip is due to the low cut filtering included in the measurement system. By taking into account the low cut filtering function and the delay time based on the measured jet pump flow trend, the calculated jet pump flow agreed well with the measured data. The main steam flow decreases with the core power, and it results in decline of the dome pressure. The calculated results agreed well with the measured data since the given core power was appropriate.

These results indicate that RETRAN-3D simulates the entire plant system behavior with sufficient accuracy. Core boundary conditions obtained in this calculation are imported to SIMULATE-3K as input data for the in-core neutron calculation in section 4.2.



(a) Core power (c) Main steam flow

(b) Jet pump flow (d) Dome pressure

Figure 5 Trends of major plant parameters

4.2 In-core power distribution

The in-core power distribution was calculated by SIMULATE-3K. Figure 6 shows the measured and calculated core averaged axial power profiles at the initial core condition. Prediction accuracy is good and the root mean square error (RMSE) is 5.2%. A closer observation indicates that the calculated power at the middle of the core is slightly overestimated and the power at the level of LPRM channel-A is underestimated about 10%.

The trend of APRM signal during the test is shown in figure 7. The measured APRM is a representative one among measured six APRM responses. It can be seen that the declining trend of the core power due to core flow reduction, the power decrease due to control rods insertion, and the power settling level are well predicted.

Figure 8 shows planar core maps of measured and calculated LPRM responses during initial 30 seconds after the trip. The calculated LPRM responses show fairly good agreement with the measured data including the large shift near the inserted control rods. Among these results, a relatively underestimated trend can be seen in LPRM channel-A responses.

Figure 9 shows correlations between measured and calculated LPRM responses at 0 seconds (initial state) and 30 seconds (final state). In both states, calculated responses show some underestimation in Channel-A while they show slight overestimation in other channels. This trend is consistent with errors observed in the initial core averaged axial power profile shown in figure 6, and is maintained through the transient event.

Therefore, the normalized LPRM variation rate (i.e. the LPRM signal value at the final state divided by the initial value) can be defined as an index of the local power variation. The normalized LPRM variation rate is closely related to the normalized variation rate of the CPR, which is one of major safety parameters in the design basis transients, along with the core flow or the local power peaking etc. Figure 10 shows the measured and calculated normalized LPRM variation rates. The calculated normalized LPRM variation rates agreed better with the measured data regardless of the LPRM location, and the RMSE was about 0.02. A slight underestimation at Channel-C should be considered as an uncertainty of the power variation rate in the case of the statistical safety analysis.

During the one pump trip test, the power distribution is globally changed by the decrease of the core flow and locally distorted by the local insertion of control rods. In other transient events, behaviors of the power distribution are much simpler and the initial power profile will be almost retained. As a consequence, it was validated that SIMULATE-3K is capable of predicting the normalized power variation rate with sufficient accuracy (as shown in figure 10), which is important in predicting the normalized CPR variation rate, for most design basis transients.

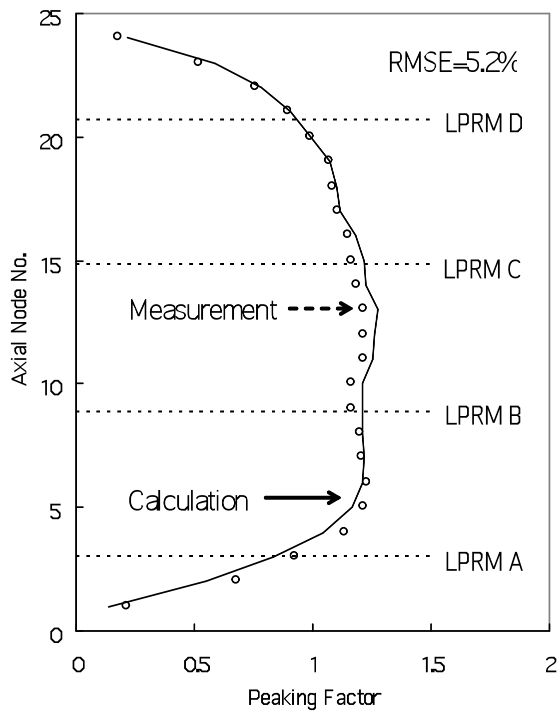


Figure 6 initial axial power profile

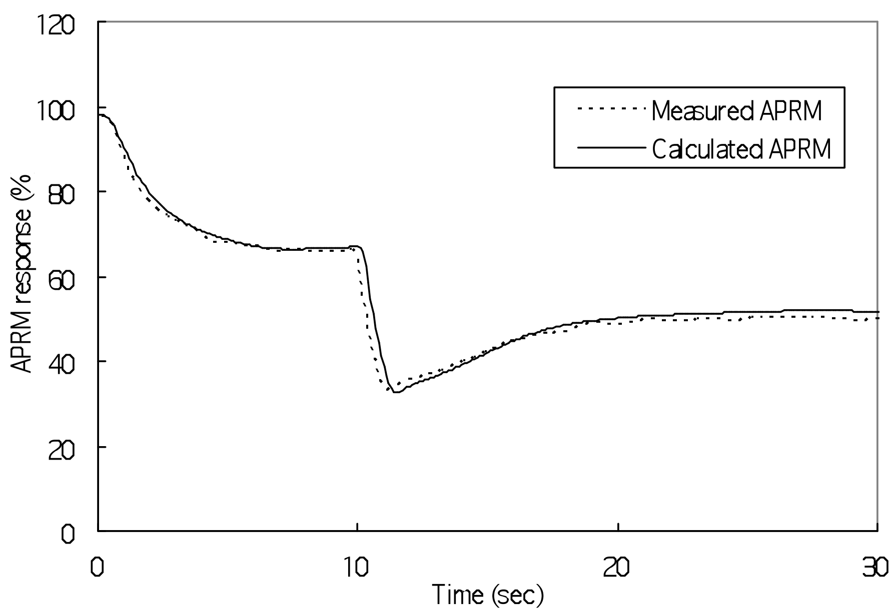
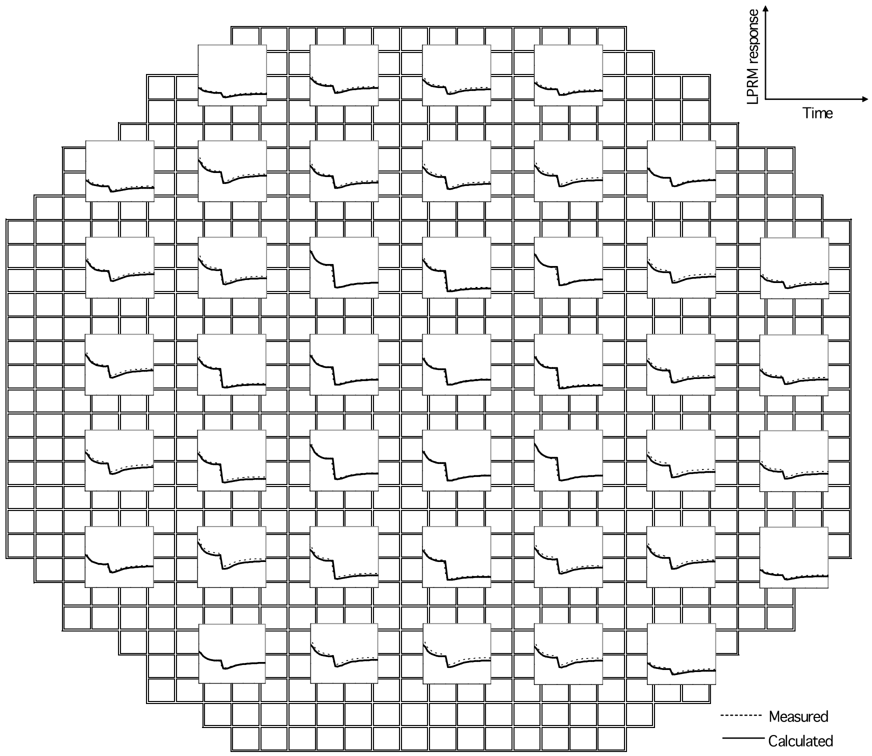
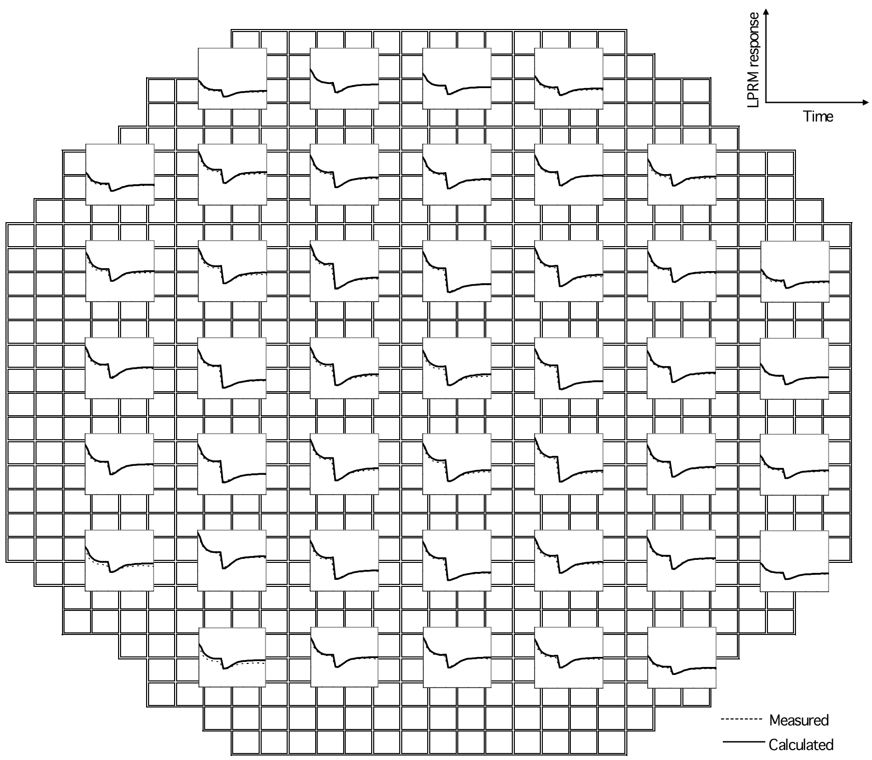


Figure 7 APRM response

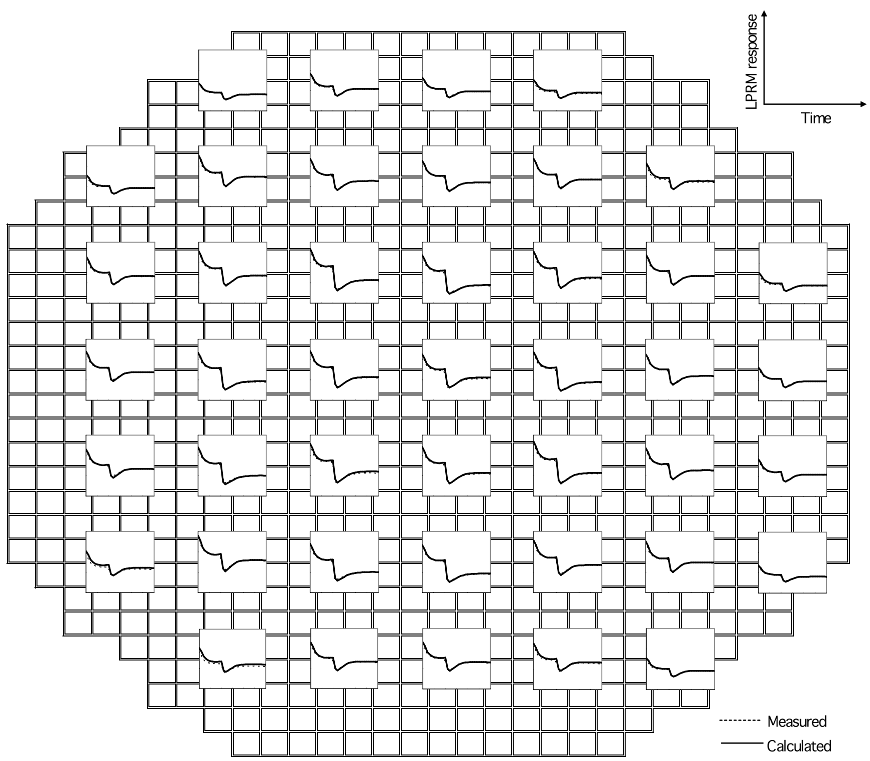


(a) LPRM responses (channel-A)

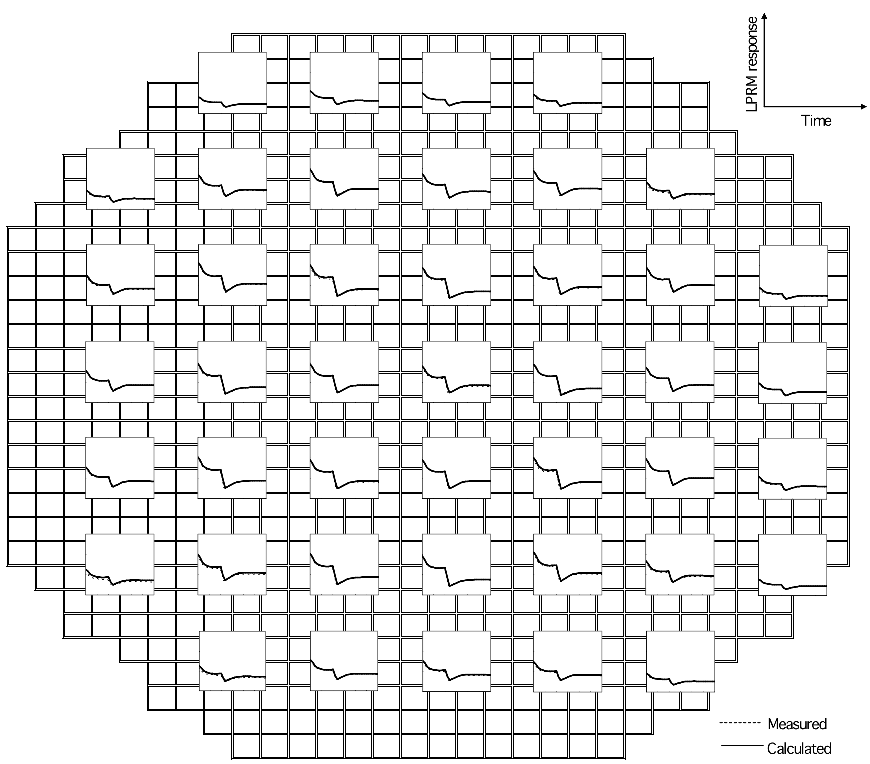


(b) LPRM responses (channel-B)

Figure 8 Planar maps of calculated and measured time series LPRM responses

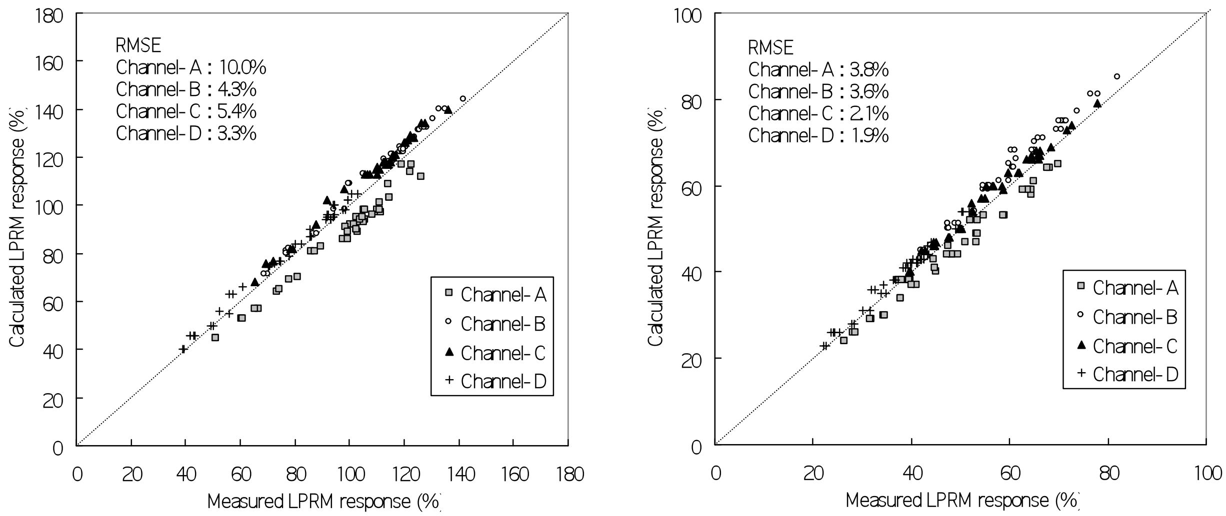


(c) LPRM responses (channel-C)



(d) LPRM responses (channel-D)

Figure 8 Planar maps of calculated and measured time series LPRM responses (cont.)



(a) Initial state (0sec) (b) Final state (30sec)

Figure 9 Correlation of measured and calculated LPRM responses

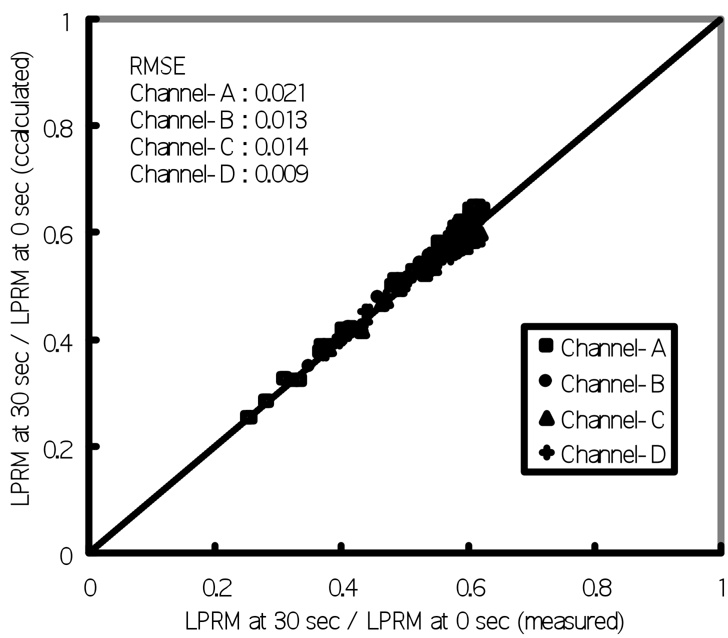


Figure 10 Correlation of measured and calculated normalized variation rate of LPRM responses

5. Conclusion

To validate models relevant to the transitional in-core 3-D power behavior, the 3-D core kinetics code SIMULATE-3K was applied to analyse the one-pump trip test performed in a BWR-5 plant which includes a large distortion of the power distribution due to the local insertion of control rods. The core boundary condition required by SIMULATE-3K was calculated by the plant transient analysis code RETRAN-3D. The APRM response and the LPRM responses calculated by SIMULATE-3K agreed well with the corresponding measured data. Furthermore, the normalized variation rates of the LPRM responses, which are important in predicting the normalized CPR variation rate, showed better agreement with the measured data.

These results indicate that SIMULATE-3K is capable of simulating the transitional in-core 3-D power behavior of the actual plant by giving appropriate core boundary conditions, even if the power distribution was largely distorted by the local insertion of control rods. Considering the power behaviors of other transient events are much simpler compared to the one pump trip test, it is expected that the normalized local power variation rate can be adequately predicted by SIMULATE-3K for most design basis transients.

6. References

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