Analysis of Selected Rod Insertion Test in BWR Plant with Three Dimensional Transient Code TOSDYN-2

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Various SRI (Selected Rod Insertion) tests performed during the start-up of a 1,100 MW-class BWR/5 plant and many useful data were obtained. Those tests were simulated by the three-dimensional (3-D) transient code "TOSDYN-2" and the results were compared with the test data. The purpose of this study is to see the spatial effects which appeared during the transient phenomena with SRI, to confirm that SRI does not distort the power distribution locally and to verify the TOSDYN-2 code as the 3-D transient code.

Analytical results show good agreement, both qualitatively and quantitatively, with the 3-D effect test results, such as the transient response of spatial power distribution. From the test data and the analytical results, it is shown that the spatial effect by SRI is very smooth. Also, TOSDYN-2 is well qualified through this analysis as a 3-D BWR transient code.

KEYWORDS: selected rod insertion, BWR type reactors, TOSDYN-2, three dimensional kinetics, start-up test, LPRM signals, spatial dependence of power change

I. INTRODUCTION

Various Selected Rod Insertion (abbreviated as SRI in this paper) tests were performed during the start-up of a 1,100 MW-class BWR/5 plant. SRI has the effect of reducing reactor power without a scram by inserting the pre-selected control rods at a scram speed during a transient and avoids the low-flow/high-power regime, where stability is concerned. In these tests, 48 LPRM signals distributed throughout the entire core region were measured and recorded. These test data were expected to yield much important information regarding, for example, the propagation characteristics of power changes in the core and the detectability of local phenomena by the LPRM system. Especially for the core stability, the power distribution in the core is very important, so the above recorded test data are very useful.

The data have been analyzed using the 3-D kinetics code "TOSDYN-2". The TOSDYN-2 code was originally developed as a BWR core stability analysis code, but it has models of all the BWR components external to the core, including the control system. These are the same as those in the plant dynamics design code. Also the LPRM and APRM nuclear instrumentation models are modeled. So it is applicable to transient phenomena, such as above mentioned tests. There have already been several studies about the BWR transient phenomena with 3-D kinetics codes, but they were limited to the transient with scram, such as the failed scram(1), the turbine trip test(2)(3) and the load rejection test(4).

The purpose of this paper is to describe the 3-D effects seen in the SRI tests through analysis to confirm that the local power dis-
tortion would not appear, which has possibility to induce the neutron flux oscillation, and to show the validity of using the TOSDYN-2 code as a 3-D BWR transient code by a comparison of LPRM responses between the test results and the analytical results by TOSDYN-2.

II. TOSDYN-2 MODEL

The TOSDYN-2 code is a time domain model for an overall BWR plant system, as shown in Fig. 1(6).

Fig. 1 TOSDYN-2 model organization

It was developed mainly for BWR core stability analysis(6)~(8), but is applicable to other transient phenomena analysis(9). It includes a detailed core model made up of 3-D neutronics model(10), the axially 1-D and multi-channel thermal-hydraulics model, and the radially 1-D fuel heat transfer model. The models of the ex-core components, including the recirculation and control systems, are the same as those used in the BWR transient analysis design code(11). Data transfer among these models is shown in Fig. 2(6).

The neutron kinetics model assumes a modified one-group neutron diffusion theory with six delayed neutron groups. To reduce computation time in the neutronics calculation, four fuel bundles surrounding a control rod are averaged into one radial node and the prompt jump approximation is employed except in the case of super-critical phenomenon, such as a control rod drop accident. The response matrix method(12) is used in com-

Fig. 2 Data transfer among TOSDYN-2 each model
pensation for the above mentioned relatively coarse-mesh noding. The neutronics model restarts from the dump file of the detailed steady-state 3-D model (10), using the neutronics constants, the control rods insertion pattern, the fuel bundle types, distributions of in core parameters, such as the neutron flux distribution, the water density distribution and the fuel temperature distribution, and so on. The neutron flux distribution is obtained by solving the 3-D neutron diffusion equation and the delayed neutron equation with the water density distribution calculated by the thermal-hydraulic model, the fuel temperature distribution calculated by the fuel heat transfer model and the nuclear constants distribution due to the control rod movement at the transient state.

In the thermal-hydraulics and fuel heat transfer calculation, fuel bundles in the core region are grouped into several channel types. The bypass region can be treated as one channel type. These channel types are coupled to each other through flow redistribution so as to ensure the same pressure drop across each channel. The thermal-hydraulic model consists of five basic conservation equations based on the nonequilibrium separated flow model. Namely, continuity and energy equations for the mixture and vapor phases and a momentum equation for the mixture. The drift-flux relationship is used for the void-quality correlation. After continuity and energy equations are converged, the pressure drop across the channel is calculated by integrating the momentum equation.

The ex-core model includes all major BWR component models, as shown in Fig. 1, such as separator model, dryer model, pressure dome model, steam line model, recirculation model and four control systems, namely, recirculation flow, pressure regulation, feed water flow and water level controller. The following three ex-core parameters, core inlet flow, core inlet enthalpy and core pressure, are used for the thermal-hydraulic boundary conditions, as shown in Fig. 2.

III. PLANT SRI TESTS

The SRI tests were performed during the start-up of a 1,100 MW-class BWR/5 plant. The analysis was performed on the basis of the following three SRI tests:

1. TC#2: SRI test at 50% power
2. TC#3: Load rejection test with SRI at 75% power
3. TC#6: Load rejection test with SRI at 100% power.

In these tests, the fully-withdrawn control rods in the peripheral region of the core were fully inserted simultaneously. Twelve rods were inserted for TC#2 and TC#3, and 20 rods were inserted for TC#6. The position of the inserted rods in the core is shown in Fig. 3.

Fig. 3 Measured LPRMs and inserted rods location in the core

In TC#2 test, only SRI was performed and no special control system was in operation. However, in TC#3 and TC#6, the generator load was rejected. Then the steam control valve was closed rapidly and the bypass valve was opened to suppress the pressure rise, and,
further, the two recirculation pumps were tripped to suppress the power increase. Thus, TC#3 and TC#6 are compound transient phenomena.

In these tests, the 48 LPRM signals from the 12 LPRM strings shown in Fig. 3 were recorded along with various plant parameters. In one LPRM string, there are four axially-arranged monitors A, B, C and D, respectively, from the bottom.

IV. TEST ANALYSIS

The number of thermal-hydraulic channel types was determined by taking into account the radial power distribution and the orifice type. The core was divided into seven thermal-hydraulic groups, six heated channel groups and one bypass channel, as shown in Fig. 4.

Before the transient analysis, a steady state calculation was carried out on 3-D power distribution. Figure 5 shows a comparison between initial LPRM signals measured in the test and given by the analysis for TC#2. The results are in good agreement, within 5% evaluated by standard deviation of tests and analytical values, so it is clear that the initial conditions simulated in the test and the LPRM nuclear instrumentation model of the TOSDYN-2 code are adequate.

Figure 6 shows a comparison between analytical and test responses of one APRM and four LPRM signals in the same string (LPRM-(8) indicated in Fig. 3). In this plant, all LPRM signals are equally divided and averaged into six APRM signals. So, the APRM signal represents the core averaged neutron flux response. The TOSDYN-2 code simulates the transient response well, especially in the first few seconds during which the minimum value appears. The axially lower LPRM responds more quickly than the upper one’s in descending order (A->B->C->D). The step-shaped response, seen in the test data between about 3.0 and 7.0 s, does not have a clear correlation with other measured parameters, such as jet-pump flow, feed water flow, water level and so on. From the results of some parametric analyses, this kind of response does not represent the oscillatory phenomenon, but is due to the characteristic
change in the total reactivity, decided by the balance of in-channel void change, fuel temperature change and movement of the selected control rods in such a relatively slow power-increasing phenomenon which is difficult to simulate well.

In order to evaluate the 3-D effects in the test, the maximum change in the LPRMs was chosen as the parameter of relative change rate (RCR), defined by

$$\text{RCR}_j = \frac{(\text{LPRM}_{j,\text{max}} - \text{LPRM}_{j,\text{min}})}{\text{LPRM}_{j,\text{min}}} \times 100\%,$$

where LPRM$_{j,\text{max}}$ and LPRM$_{j,\text{min}}$ are the initial and minimum values of the $j$-th LPRM signal, respectively.

Figure 7 shows a comparison between test and analytical values of RCR. Good agreement, about 3% of standard deviation, is seen, which indicates that the 3-D power change can be well simulated throughout the core by the TOSDYN-2 code, because such agreement is sufficient for evaluating the 3-D power change during transition.

To clarify the spatial dependence of LPRM response, the RCRs at axial positions B and D are plotted in Fig. 8 vs. the reciprocal of average distance from the inserted control rods (RAD), as defined by

$$\text{RAD}_j = \left(\frac{1}{N} \sum_{i=1}^{N} \frac{1}{\sqrt{X_{ij}^2 + Y_{ij}^2}}\right),$$

where $X_{ij}$, $Y_{ij}$ are the node intervals (one node interval means one fuel bundle interval) between the $j$-th LPRM string and the $i$-th inserted control rod and $N$ is the number of inserted control rods. From the above definition, the smaller the RAD$_j$ value is, the further the $j$-th LPRM is apart from the inserted control rods in the peripheral region.
of the reactor core, in other words the closer it is to the central region of the core.

From Fig. 8, the following trends can be seen both in the test and analytical results:

(1) The RCR of LPRM response increases almost linearly as RAD increases, which indicates that the spatial power change is greater at the periphery of the core close to the inserted control rods, that it is smaller at the center of the core, and its profile is very smooth.

(2) The RCR of LPRM response is greater in the upper region (D) than in the lower region (B). This trend is more noticeable at the center of the core, far from the inserted control rods.

The LPRM response, therefore, depends on the distance from the inserted control rods and correlates well with the reciprocal of average distance (RAD), so the power distribution changes smoothly. In the case of such a SRI test, only the peripheral control rods are inserted, they directly affect only the peripheral fuel bundles, and the source of thermal neutron is kept almost constant in the center of the core. So the neutron flux distribution changes mostly due to the decrease in neutron flux at the peripheral core region, namely, the neutron current from the center region to the peripheral region, so the spatial power change rate is almost in inverse proportion to the distance from the inserted control rods.

Figure 9 shows the spatial dependence of the top (D) and bottom (A) LPRM responses at 9.5 s, when they reached almost the stationary state. In the central region of the core, the LPRM level has returned to almost the initial level, but in the peripheral region, it remains lower than previously and this trend is more noticeable in the axially lower region. So, after SRI, the axial power shapes in the central region are almost the same as those before SRI, and the axial power shapes in the peripheral region are more top-peaked than those before SRI. This fact implies that the core condition will not become less stable, from the viewpoint of axial power shape effect on stability.

Next, the compound transient tests in the rated condition were analyzed. Figure 10 shows a comparison between test and analytical results for APRM response and LPRM-(8) responses. In these figures, the transient response consists of 3 distinct sections: (1) between zero and about 1.2 s, the power decreases rapidly due to control rod insertion; (2) between 1.2 and 6.0 to 7.0 s, the power decreases slowly due to void generation by
the pump trip; and (3) after about 7.0 s, the power increases slowly due to void collapse.

By comparing the four LPRM responses, the time of the minimum value is found to be between 6.0 and 8.0 s, it is earlier in the upper region than in the lower position with the descending order (D→C→B→A).

This response is different from that of TC#2, as shown in Fig. 6. In TC#2, the minimum value was brought about only by control rod insertion, but in TC#6, it was determined by the void feedback effect, which is an effect of the competing power decrease and flow decrease due to the pump trip. Except that the APRM response is slightly higher in the analysis than in the test, there is good agreement between the transient responses both for the APRM and the LPRMs.

Figure 11 compares the LPRM-(21) responses in the tests and in the analysis displayed three-dimensionally. Values are normalized to the initial values to make the spatial behavior clearer. The power starts to decrease from the bottom of the core (A→B→C→D). At about 6.0 to 7.0 s, it reaches a minimum, then starts to increase slowly from the upper monitor (D→C→B→A). From this figure, it can be seen that the responses of axially located four LPRM monitors change at almost the same rates as the initial values, only monitor-(B), which is located at the nearest axial position to the boiling boundary, changes slightly larger than the other three monitors.

Figure 12 shows the spatial dependence of the relative change rate (RCR) of LPRM response (B and D monitors). The RCR of analysis agrees with that of test data, within 2% of standard deviation. From this figure, it can be seen that the spatial dependence of the power change of TC#6 is smaller than that of TC#2. Namely, the slope (proportional coefficient of the RCR with the reciprocal of average distance from inserted rods) of TC#2 is about three times larger than that of TC#6, and this trend is more noticeable at the upper monitor. That is, the change in radial power distribution at TC#6 is smaller than at TC#2 due to the flow reduction brought about by the recirculation pump trip. It is useful to plot the radial-node power distributions by analysis, since they are more detailed than LPRM signals, for seeing the change in radial power distribution during the transient, as shown in Fig. 13. The distributions are normalized to the initial values and the plots show that the radial power distribution changes very smoothly and that there is no power distortion in the core.

The spatial dependence of LPRM relative change rate is summarized in Fig. 14, with
In Fig. 14, the following four analytical results are shown:
(1)—TC#2 (50% power) without RPT
(2)—TC#6 (100% power) without RPT
(3)—TC#3 (75% power) with RPT
(4)—TC#6 (100% power) with RPT.

The effect of the flow decrease at RPT can be seen by comparing the results of (1), (2) and (3), (4) and the effect of power level can be seen by comparing the results of (1) and (2), or (3) and (4). From this figure, it can be seen that the flow decrease phenomenon mitigates the power distribution change in the core and the relative power change rate, so the spatial dependence is not affected by the power levels, but by the types of transient phenomena.
V. CONCLUSION

The SRI tests performed during start-up of a 1,100 MW-class BWR/5 plant were simulated by the 3-D kinetics code TOSDYN-2. The results were compared with the data obtained at the plant tests. The APRM response, that represents the core average neutron flux response and the LPRM responses, representing the spatial distributed neutron flux responses are in good agreement, both qualitatively and quantitatively with the test data.

Several 3-D effects, which were indicated by the test data were well simulated by the TOSDYN-2 code. These 3-D effects are as follow:

1. The relative power changes increase almost linearly as the reciprocal of average distance from inserted rods increases.
2. The relative power changes are greater at the axially upper region than at the axially lower region and this trend is more noticeable at the center of the core, far from the inserted rods.
3. The relative power changes are mitigated by the core global transient, such as the pump trip.
4. These spatial power responses are not affected by the power level but by the types of transient phenomena.

From this study, it can be concluded that in case the insertion rods are selected uniformly, the effect of SRI on the spatial power distribution is very smooth during the rod insertion and after SRI.

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