M&C 2001 Salt Lake City, Utah, USA, September 2001.

Time-Domain BWR Stability Analysis with SKETCH-INS/TRAC-BF1, Validation against OECD/NEA Ringhals 1 Stability Benchmark

Hideaki Utsuno, Hiroshi Yamada, Masahide Nishio, Katsuo Suzuki
Nuclear Power Engineering Corporation (NUPEC)
Fujita kanko toranomon bldg., 7F
17-1, 3-chome tranomon,
minato-ku, tokyo 105-0001, JAPAN
utsuno@nupec.or.jp, h-yamada@nupec.or.jp,
nishio@nupec.or.jp, kt-suzuki@nupec.or.jp

Keywords: BWR, Stability, Neutron Kinetics, Thermal Hydraulics

ABSTRACT

SKETCH-INS/TRAC-BF1 is a three-dimensional time-domain analysis code, which utilizes a nodal method with introducing an assembly discontinuity factor for the neutronics and a two-fluid six-equation model for the thermal hydraulics. Validation of SKETCH-INS/TRAC-BF1 has been performed against the data from OECD/NEA Ringhals 1 stability benchmark. Three-dimensional time-domain BWR stability analyses were performed on some test points for the core-wide oscillation mode and the regional oscillation mode. The analytical results of the stability decay ratio and the resonance frequency for both modes were compared with the test results. As a result, the standard deviation for the stability decay ratio between the analyses versus the test data was satisfactory as well as those obtained from the benchmark project. Fairly good agreement was obtained in the comparison of resonance frequencies, which means that the thermal hydraulics of two-phase flow in fuel channels were well simulated by the analysis. Regional oscillation, where one half of the core oscillates with out-of-phase against the other half, was well simulated by the analysis. These results show the validity of SKETCH-INS/TRAC-BF1 to predict the BWR stability.

1. INTRODUCTION

Core nuclear thermal-hydraulic stability in a Boiling Water Reactor (BWR) is one of the important issues to be considered in licensing evaluations. When such instability occurs, it may reduce the core thermal margin. The core may oscillate in two different modes. One is the fundamental mode, or core-wide mode, where neutron flux in the core oscillates together with in-phase. The other is the first harmonic mode, or regional mode, where neutron flux in one half of the core oscillates with out-of-phase against the other half. Both these modes of oscillations have occurred in actual operating BWRs.

TRAC-BF1/SKETCH-N (Asaka et al., 2000) code system has been recently developed in Japan Atomic Energy Research Institute (JAERI) by a coupling of the best estimate BWR transient analysis code TRAC-BF1 (Borkowski et al., 1992) with the three-dimensional neutron kinetics code SKETCH-N (Zimin and Ninokata, 1998). The code system has been modified as SKETCH-INS/TRAC-BF1 code system in NUPEC (2000a) to utilize it for the audit analysis on the BWR stability. SKETCH-INS/TRAC-BF1 deals with the BWR stability in three-dimensional time-domain, using a nodal method with an assembly discontinuity factor for the neutronics and a two-fluid six-equation model for the thermal hydraulics.

The purpose of this study is to show the validity of SKETCH-INS/TRAC-BF1 to predict BWR stability against the data from OECD/NEA Ringhals 1 stability benchmark, which is suitable for the code validation since it covers the various stability conditions. Three-dimensional time-domain BWR stability analyses were performed on some test points for the core-wide oscillation mode and the regional oscillation mode. The analytical results of the stability decay ratio and resonance frequency for both modes were compared with the test results. It was found that the agreement is quite sufficient.

2. Outline of OECD/NEA Stability Benchmark

OECD/NEA has recently organized BWR stability benchmark project providing a comprehensive and fully defined set of data, which can be used for the code validation. The benchmark data have been come from measurements performed during cycles 14, 15, 16 and 17 in the Swedish 800 MWe BWR Ringtails 1. The reactor system specifications are summarized in Table 1.

Table 1 Ringhals 1 reactor system specifications

Item	Specification
Reactor type	800 MWe BWR
Rated core power	2,270 MWt
Rated core flow	7,050 kg/s
Fuel bundles	648 bundles
Fuel type	8×8 type
Control rods	157 control rods
Recirculation loops	6 loops

An extensive set of the data has been given in the benchmark specifications (Levert, 1994) with files on CD-ROM. An input data set includes general description of the reactor system, core neutronic data, thermal-hydraulic data, recirculation loop data and computed distributions (power, burnup, void history, control rod history, etc.) at the measurement state points. Two group macro cross sections are given in the benchmark files for each fuel material composition. It also includes the input files for the CASMO, a fuel assembly calculation code based on the transport theory, thus providing an opportunity to generate macro cross section in the form suitable for user’s code.

Figure 1 shows state points where the stability measurements were taken in the power-flow map. The measurements comprise totally 41 state points among four cycles. The benchmark data contains stability parameters decay ratio and resonance frequency for both the fundamental mode (core-wide oscillation) and the first harmonic mode (regional oscillation), which were evaluated from the measured time-series data. Moreover, the spontaneous regional oscillation was observed in the case of cycle 14 point 9. Nine participants from eight countries participated in the benchmark with the both frequency-domain and time-domain stability analysis codes. Their results are summarized in the benchmark final report (Levert, 1996).

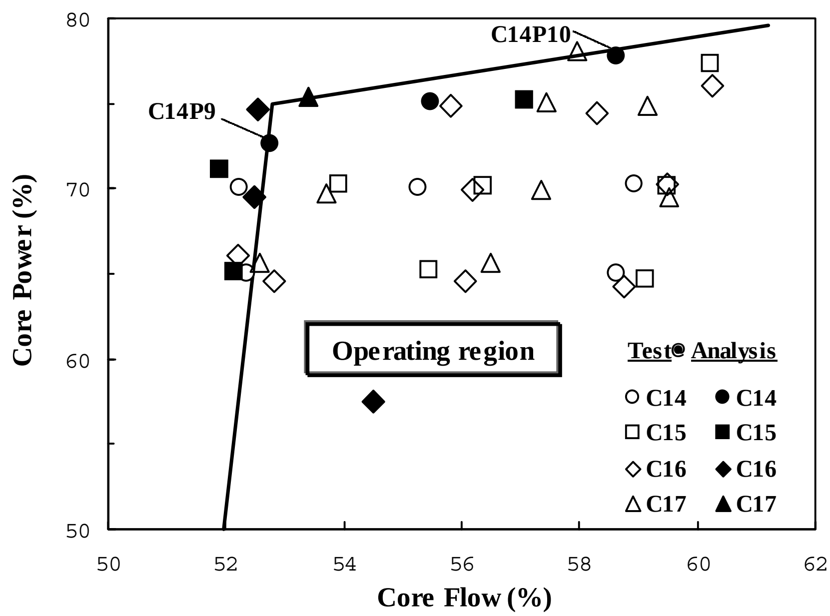


Fig. 1 Ringhals 1 stability test state points

In this study, analyses were performed on some test points (10 points), which were selected from the various test points as a boundary of higher core power or lower core flow sides in the wide operating region. Decay ratio and resonance frequency for both the core-wide oscillation and the regional oscillation were evaluated from the analyzed neutron flux response. In the following explanation, representing the analyzed cases, the analysis for the case of cycle 14 point 9 is mainly discussed since it was a characteristic case observed the spontaneous regional oscillation. Also, the analysis for the case of cycle 14 point 10 is shown representing a stable case.

3. SKETCH-INS/TRAC-BF1 Code System

TRAC-BF1/SKETCH-N (Asaka et al., 2000) code system has been recently developed in JAERI by a coupling of the best estimate BWR transient analysis code TRAC-BF1 with the three-dimensional neutron kinetics code SKETCH-N. The coupling between the codes is organized using an interface module based on the massage-passing library. Verification of the code system has been performed against the data from OECD/NEA BWR cold water injection benchmark (Zimin, 2000).

The code system has been modified in detail as SKETCH-INS/TRAC-BF1 code system in NUPEC (2000a) to utilize it for the audit analysis on the BWR stability. Table 2 summarizes the specifications of SKETCH-INS/TRAC-BF1 code system. The specifications are classified into four categories: neutronics, fuel heat transfer, thermal hydraulics and external core system. As shown in the table, the specifications are in the state of the art as for a three-dimensional time-domain analysis code coupled with neutronics and thermal hydraulics.

Table 2 Specifications of SKETCH-INS/TRAC-BF1

Item		Specification
Neutronics	Dimension	3D XYZ •1 bundle/1 mesh•
	Equation	Time-dependent diffusion equation
	Energy group	Two groups
	Delayed neutron	Six groups
	Spatial resolution	Discontinuity factor considered
	Numerical method	Spatial integration: nodal method Time integration: fully-implicit scheme
Fuel heat transfer	Dimension	1D radial (for each axial node)
	Equation	Heat conduction equation
	Heat transfer	Heat transfer coefficients: single phase: Dittus-Boelter nucleate boiling: Chen
Channel thermal hydraulics	Dimension	1D axial (multi-channel)
	Fluid model	Two-fluid model
	Equation	Six balance equations of mass, momentum and energy for liquid and vapor phases
	Numerical method	Spatial integration: differential method Time integration: semi-implicit scheme
External core system		Reactor vessel, Recirculation loop, Steam separator, etc.

3.1 SKETCH-INS Code

The SKETCH-INS (NUPEC, 2000a) code is a modification of the SKETCH-N code (Zimin and Ninokata, 1998) which was originally developed in JAERI. The SKETCH-INS code solves neutron diffusion equations in three-dimensional Cartesian coordinates for the steady-state and the kinetics. The code treats two neutron energy groups and six groups of delayed neutron precursors. In order to improve the spatial resolution accuracy, the assembly discontinuity factor is implemented in the code upon the original one. Numerical methods for the neutronics calculations are as follows. Polynomial and semi-analytical nodal method based on the nonlinear iteration procedure (Zimin, et al., 1998) is used for spatial integration of diffusion equations. Time integration of the neutron kinetics equations is performed by the fully-implicit scheme.

3.2 TRAC-BF1 Code

TRAC-BF1 (Borkowski, et al., 1992) is the latest public domain BWR version of the reactor transient analysis code TRAC, which utilizes a two-fluid six-equation model for the thermal hydraulics. TRAC-BF1 has modules of major BWR components: Vessel, Channel, Separator-Dryer, Pump, etc. The BWR system can be modeled with these component modules. The fuel channels in the core are modeled with the multiple Channel components, which solves six balance equations of mass, momentum and energy for liquid and vapor phases. Two-phase flow in the Channel component is treated as one-dimensional vertical flow. Fuel rod heat transfer module solves one-dimensional radial heat conduction equations. Heat transfer at the outer cladding surface is subdivided into single-phase liquid, nucleate boiling, transition boiling, film-boiling, single-phase vapor and condensation modes. The reactor pressure vessel is modeled with the Vessel component in three-dimensional cylindrical coordinates. Numerical methods for the thermal-hydraulic calculations are improved in the TRAC/BF1. Semi-implicit stability enhanced two-step (SETS) method is applied in time integration of the fluid flow equations. Standard finite differential method with staggered mesh is used for space integration of the both fluid flow and heat conduction.

3.3 Coupling Inteface Module

The coupling between the codes is organized using an interface module based on the massage-passing library called Parallel Virtual Machine (PVM). The codes are treated as separate processes. The interface module is responsible for the data exchange between the codes, data mapping between the spatial meshes of the codes and a synchronization of the time stepping. The interface module transfers power distribution from SKETCH-INS into TRAC-BF1, coolant densities and fuel temperatures are transferred in the opposite direction. The interface module also transfers the time step size. Detailed description of the interface module is given in (Asaka et al., 2000).

4. Analysis

4.1 Analytical Model

The neutronic calculation model has been developed for the SKETCH-INS code (NUPEC, 2000b) based on the benchmark specifications (Levert, 1994). The macro cross section set was generated by CASMO using the provided input files. The neutronic calculations on the core region were performed in Cartesian XYZ coordinates. There were 648 8×8 fuel bundles loaded in the core. The core region was modeled with one radial node per bundle of 15.3 cm width and 25 axial nodes of 14.7 cm height. In addition, the core region was enveloped in the reflector layer with each one node for the side, top and bottom of the core region.

The thermal-hydraulic nodalization model for the TRAC-BF1 code is shown in Fig. 2. The reactor was conposed of the pressure vessel, core, lower and upper plenum, separator and down comer, connected with recirculation loop, feedwater line and main steam line. The boundary conditions were given as the flow boundary for the feedwater line and the pressure boundary for the main steam line. The pressure vessel was modeled with 2 radial rings and 13 axial levels in cylindrical R•Z coordinates. The heating length of the fuel channel was divided into 25 nodes same as the neutronic model.

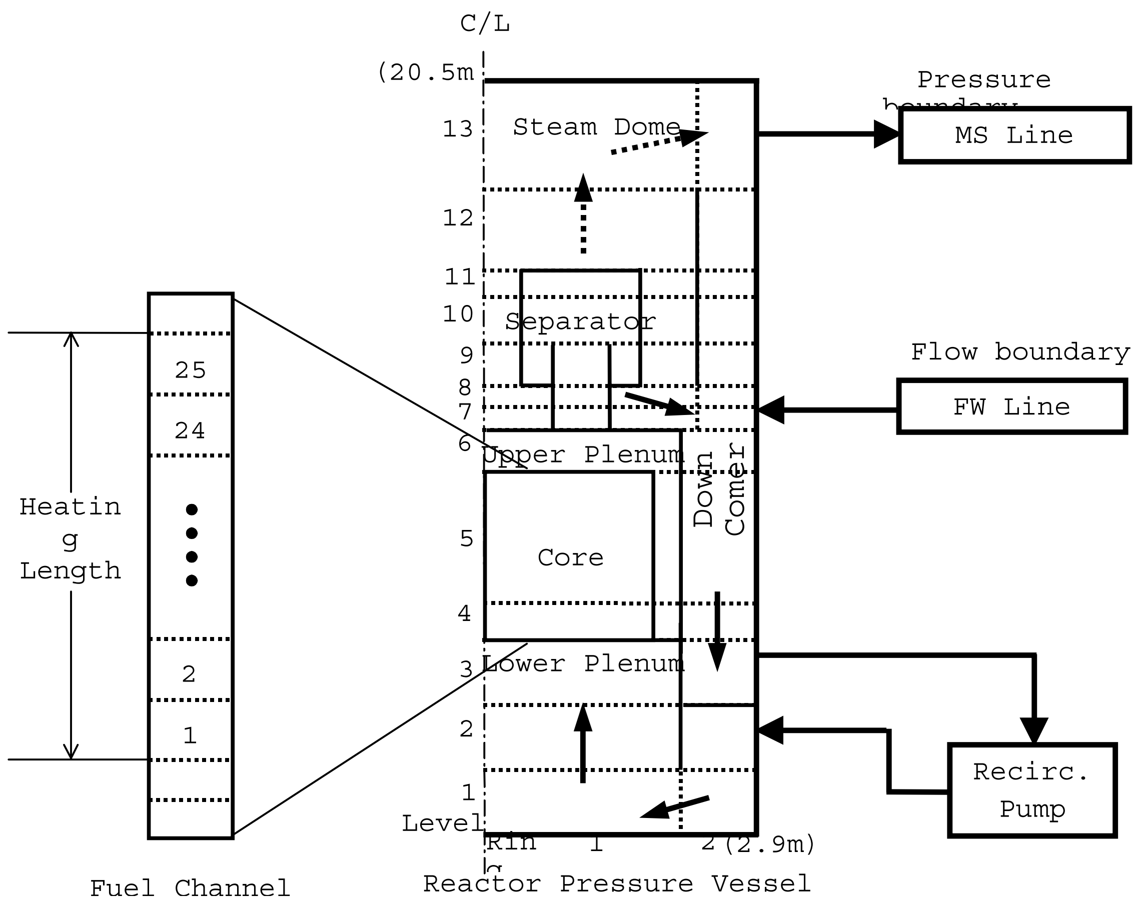
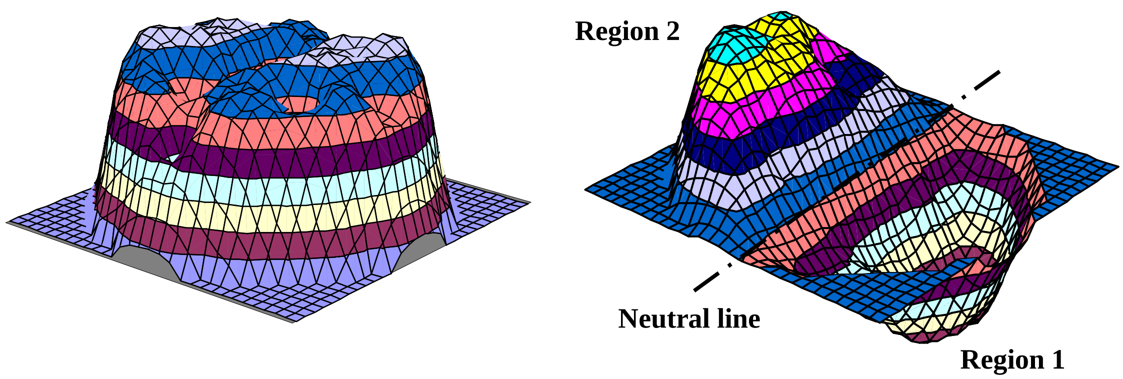


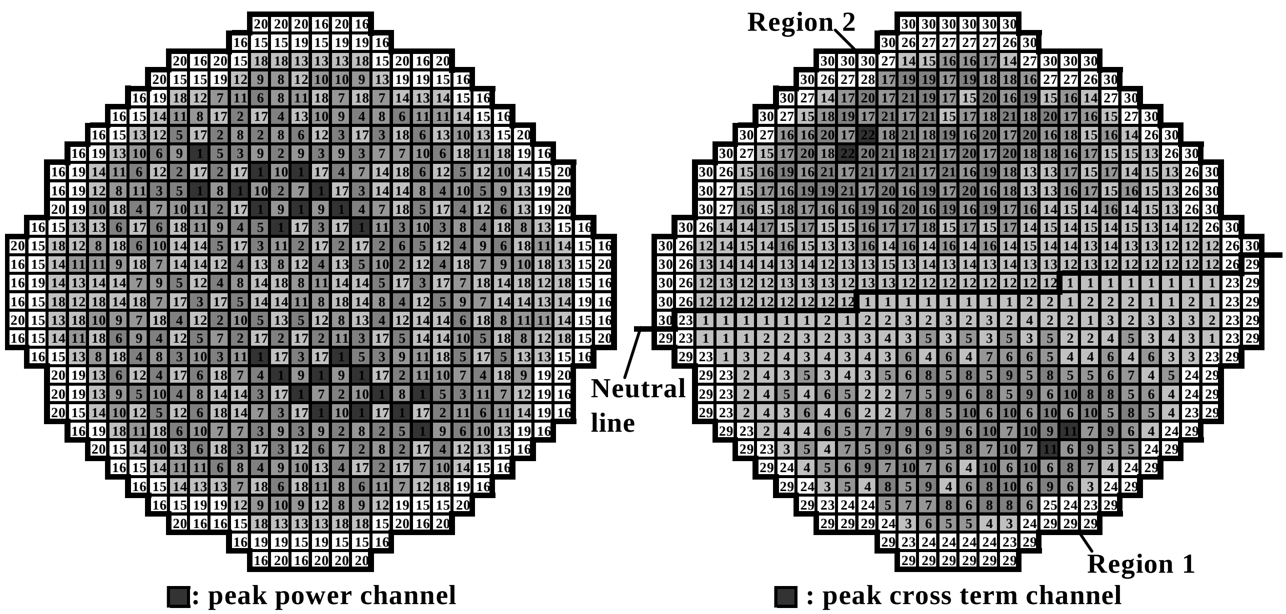
Fig. 2 Thermal-hydraulic nodalization model

The neutron flux harmonic modes of the fundamental mode and the first harmonic mode in the case of cycle 14 point 9 are shown in fig. 3. In the thermal-hydraulic calculations, fuel bundles in the core were collapsed into the multiple grouped thermal-hydraulic channels based on the harmonic modes. Figure 4 shows the thermal-hydraulic channel division. For the core-wide oscillation analysis, based on the relative power of the fundamental mode, the bundles were collapsed into 20 channels, where the integrated power in each channel division was made even. For the regional oscillation analysis, based on the cross term defined as a product of the relative power of the both modes, the bundles were collapsed into 30 channels symmetric against the neutral line of the first harmonic mode.



(a) Fundamental mode **(b)** First harmonic mode

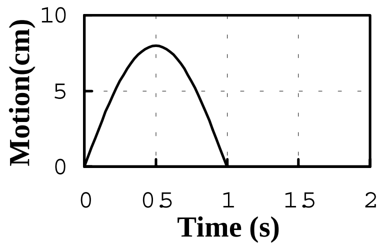
Fig. 3 Neutron flux harmonic modes



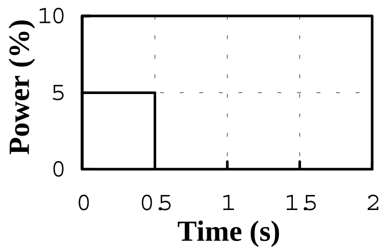
(a) Core-wide oscillation (20 channels) **(b)** Regional oscillation (30 channels)

Fig. 4 Thermal-hydraulic channel division

Neutron flux mode oscillations were excited by perturbations in the core at a steady-state. Two kinds of perturbation are shown in Fig. 5. One was a local disturbance by a sinusoidal movement for 1.0 s with the specified control rods. Figure 6 shows the control rod pattern in the case of cycle 14 point 9, where the number indicates an inserted rare of the control rod. Six partially inserted control rods were chosen for the perturbation located symmetric in the opposite sides. In order to induce the core-wide oscillation, each three control rods were moved together with in-phase resulting in a reactivity insertion of 8•, and the regional oscillation, they moved with out-of-phase against the each other. Another was a numerical disturbance by a 5 % step for 0.5 s in the core power. The power disturbances were added uniformly over the entire core for the core-wide oscillation, or with reversed phases against the opposite sides for the regional oscillation.



(a) Control rod motion



(b) Power disturbance

Fig. 5 Perturbations

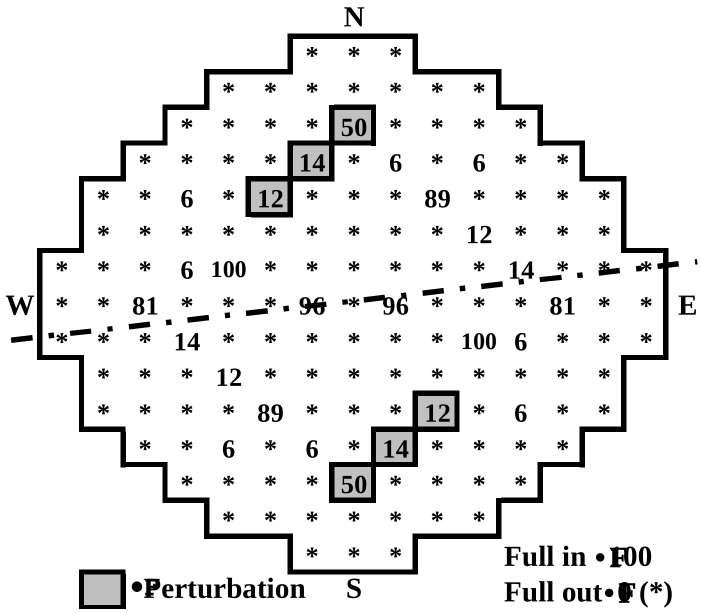
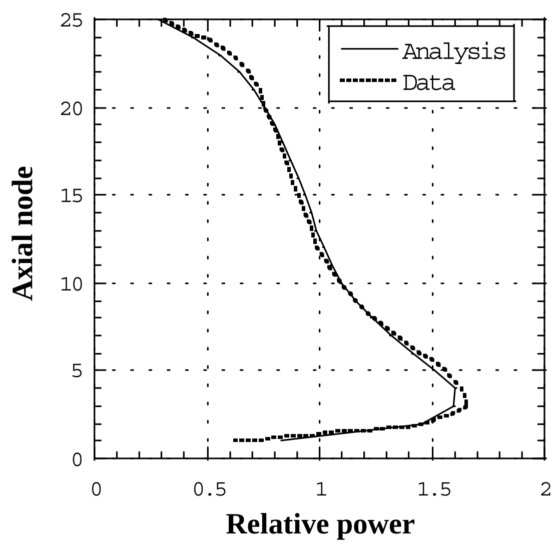


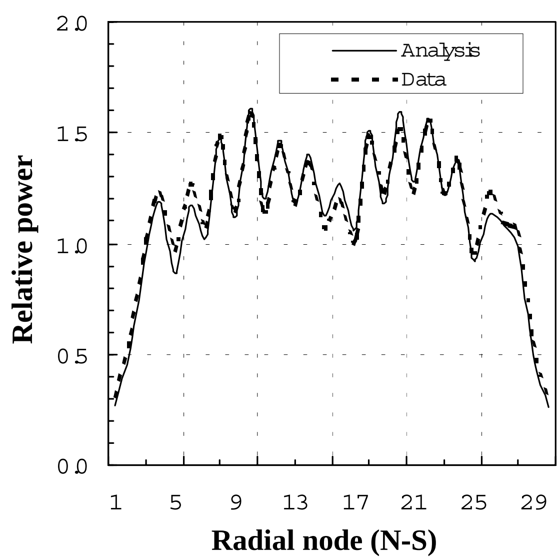
Fig. 6 Control rod pattern (cycle 14 point 9)

4.2 Steady-state Analysis

The core status at a steady-state of each test points were analyzed with SKETCH-INS/TRAC-BF1. Figure 7 shows the analyzed core power distributions in the case of cycle 14 point 9 in comparison with the benchmark reference value, which are regarded to be rather unstable conditions. Figure 7(a) shows the core averaged axial power



(a) Axial power distribution

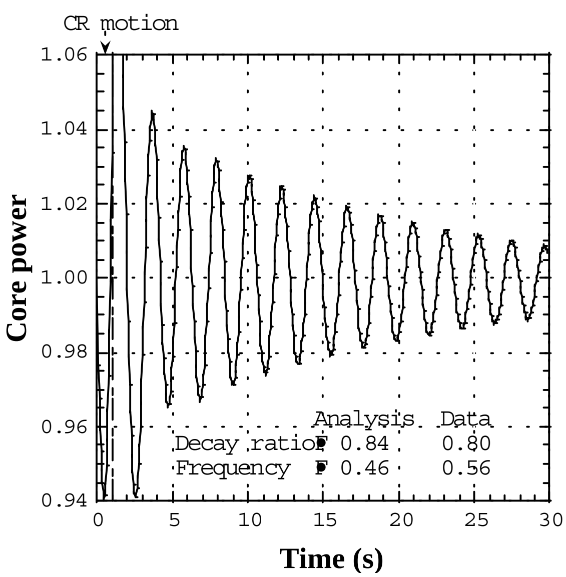


(b) Ragial power distribution

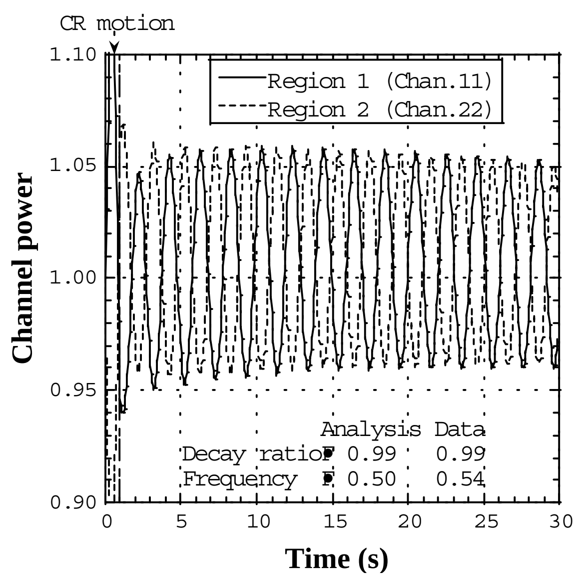
Fig. 7 Steady-state core power distributions (cycle 14 point 9) distribution. The power shape was bottom skewed, which made thermal-hydraulic characteristics unstable. Figure 7(b) shows the radial power distribution along with north to south axis. The radial power distribution was double peaked because the deeply inserted control rods were arranged from east to west orientation in the core. These power distributions were well simulated by the analysis.

4.3 Stability Analysis

The core responses to the perturbations were analyzed with SKETCH-INS/TRAC-BF1. Figure 8 shows the analyzed power responses to the control rods motion in the case of cycle 14 point 9. For the core-wide oscillation, the core power response is shown in Fig. 8(a). The dumping core-wide oscillation was simulated by the analysis with the analyzed decay ratio of 0.84. For the regional oscillation, the peak channel power responses in each region are shown in Fig. 8(b). The spontaneous regional oscillation, where the channel power oscillates with out-of-phase against the another channel with the decay ratio of 0.99, was well simulated by the analysis.



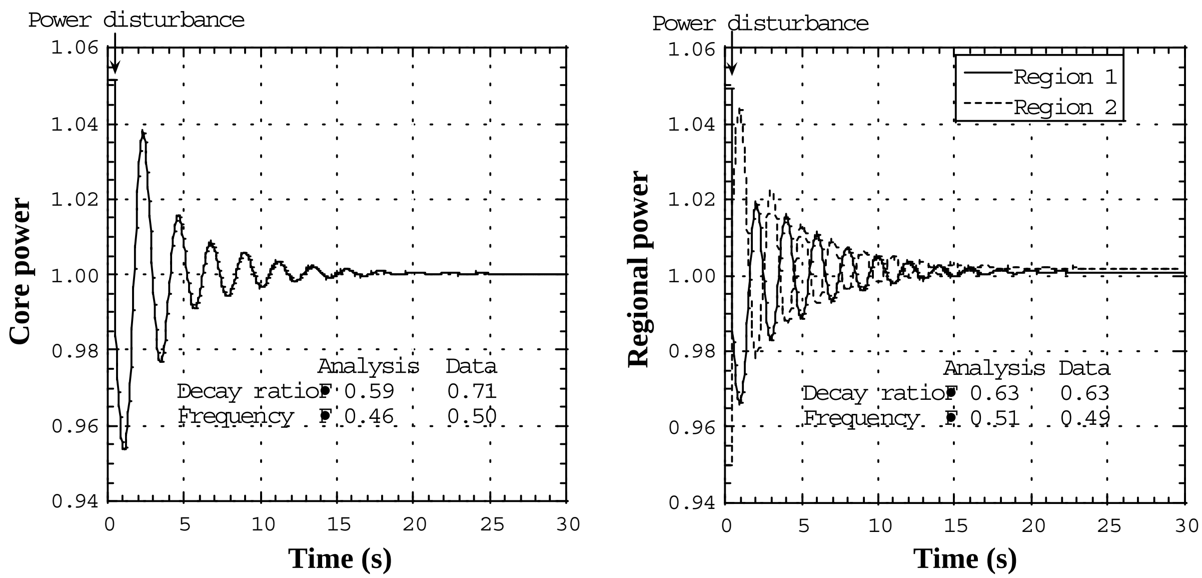
(a) Core power response



(b) Channel power response

Fig. 8 Power responses to the control rods motion (cycle 14 point 9)

Figure 9 shows the analyzed power responses to the power disturbances in the case of cycle 14 point 10. This case was a stable case. For the core-wide oscillation, the core power response is shown in Fig. 9(a). The dumping core-wide oscillation was simulated by the analysis with the analyzed decay ratio of 0.59. For the regional oscillation, the regional power responses in each region were shown in Fig. 9(b). The dumping regional oscillation with the decay ratio of 0.63 was well simulated by the analysis.



(a) Core power response

(b) Channel power response

Fig. 9 Power responses to the power disturbance (cycle 14 point 10)

The response of neutron flux to the power disturbance was examined in the case of cycle 14 point 9. The power disturbance was added with reversed phases against the opposite region. Figure 10 shows the spatial change of neutron flux distribution in time sequence as the deviation from a steady-state. The distribution at 0.5 s is one just after the power disturbance. The maximum amplitude of the oscillation appears at 2.0 s. After the half period, at 3.0 s, the phase of the distribution is reversed. After the period, at 4.0 s, the phase comes to the same repeatedly. It was recognized that, the regional oscillation is a phenomenon where the neutron flux oscillates with the first harmonic mode.

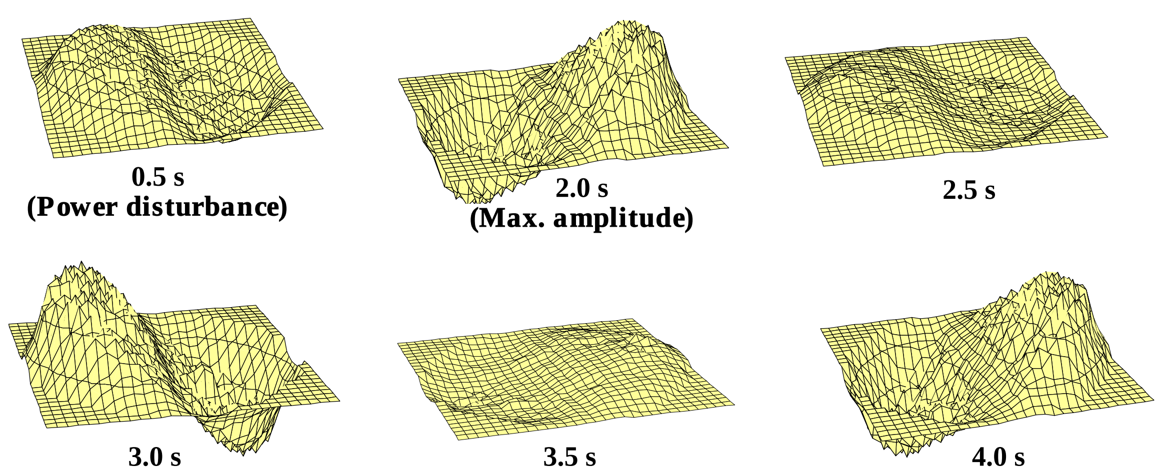
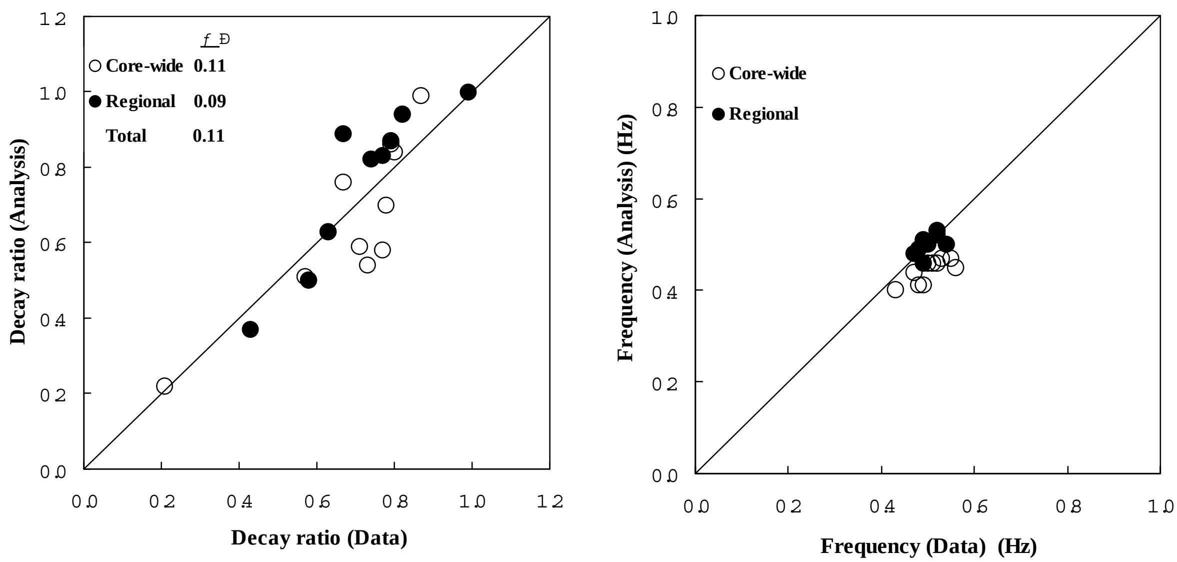


Fig. 10 Spatial change of neutron flux distribution (cycle 14 point 9)

Comparisons of decay ratio and resonance frequency between the analyses versus the test data are shown in Figs. 11(a) and (b). The core-wide oscillation is indicated by white plots and the regional oscillation black plots. Because the analysis was a best estimate, the test and the analysis should agree within the analytical accuracy. As shown in Fig. 11(a), the plotted data are along with the 45-degree line throughout the examined range of the decay ratio from 0.2 to 1.0. The decay ratio 1.0 is corresponding to occurrence of the spontaneous regional oscillation, which is well predicted by the analysis. The standard deviation between the analyses versus the test data was obtained as 0.11 for the core-wide oscillation, 0.09 for the regional oscillation and 0.11 for the total. As shown in Fig. 11(b), the resonance frequencies are concentrated on around 0.5 Hz for all cases and good agreements were obtained between the analyses and the test data, which means that the thermal hydraulics of two-phase flow in fuel channels were well simulated by the analysis.



(a) Comparisons of decay ratio **(b)** Comparisons of resonance frequency

Fig. 11 Comparisons between the analyses versus the test data

According to the benchmark final report (Levert, 1996), the total results from nine participants with five frequency-domain codes and four time-domain codes were as follows: the standard deviation of decay ratio is in the range of 0.06 to 0.10 for the core-wide oscillation, 0.08 to 0.14 for the regional oscillation. The standard deviation of the stability decay ratio obtained in this study is similar to those obtained from the benchmark project. The capability of SKETCH-INS/TRAC-BF1 to predict BWR stability was confirmed from this comparison.

5. Conclusions

Capability of SKETCH-INS/TRAC-BF1 has been studied against the data from OECD/NEA Ringhals 1 stability benchmark. Three-dimensional time-domain BWR stability analyses were performed on some test points, which were selected from the various test points as a boundary of higher core power or lower core flow sides in the wide operating region, for the core-wide oscillation mode and the regional oscillation mode. The analytical results of the stability decay ratio and the resonance frequency for both modes were compared with the test results. The following conclusions were obtained.

The standard deviation of the stability decay ratio between the analyses versus the test data was 0.11 for the core-wide oscillation and 0.09 for the regional oscillation in the wide range from the dumping oscillation with decay ratio 0.2 to the spontaneous oscillation. The standard deviation of decay ratio was satisfactory as well as those obtained from the benchmark project. Fairly good agreement was obtained in the comparison of resonance frequencies, which means that the thermal hydraulics of two-phase flow in fuel channels were well simulated by the analysis. The regional oscillation, where one half of the core oscillates with out-of-phase against the other half, was well simulated by the analysis on the case of cycle 14 point 9. These results show the validity of SKETCH-INS/TRAC-BF1 to predict the BWR stability.

ACKNOWLEDGMENTS

This study was performed under the sponsorship of the Ministry of Economy, Trade and Industry (METI).

REFERENCES

Asaka H., V.G. Zimin, Iguchi T. and Anoda Y., 2000. Coupling of the Thermal-Hydraulics TRAC Codes with 3D Neutron Kinetics Code SKETCH-N, Preliminary Proc. OECD/CSNI Workshop on Advanced Thermal-Hydraulic and Neutronics Codes: Current and Future Applications, Barcelona, Spain, 10-13 April 2000, vol. 2, pp. 1-15.

Borkowski, J. A., et al., TRAC-BF1/MOD1: An Advanced Best-Estimate Computer Program for BWR Accident Analysis, Model Description, NUREG/CR-4356, EGG-2626, Vol. 1, August 1992.

Lefvert, T., 1994. OECD/NEA Nuclear Science Committee BWR Stability Benchmark, Final Specifications, NEA/NSC/DOC(94)15.

Lefvert, T., 1996. OECD/NEA Nuclear Science Committee Ringhals 1 Stability Benchmark, Final Report, NEA/NSC/DOC(96)22.

NUPEC, 2000a. Modification of Three-Dimensional Plant Transient Analysis Code TRAC-BF1/SKETCH-N, INS/M99-12, March 2000 (in Japanese).

NUPEC, 2000b. Modification of Three-Dimensional Plant Transient Analysis Code TRAC-BF1/SKETCH-N (2) = Ringhals 1 Stability Analysis =, INS/M99-13, March 2000 (in Japanese).

Zimin, V.G. and Ninokata, H., 1998. Nodal Neutron Kinetics Model Based on Nonlinear Iteration Procedure for LWR Analysis. Ann. Nucl. Energy, **25**, 507-528.

Zimin, V.G., Ninokata, H. and Pogosbekyan, L., 1998. Polynomial and Semi-Analytic Nodal Methods for Nonlinear Iteration Procedure, Proc. ANS Int. Conf. On the Physics of Nuclear Science and Technology (PHYSOR), Long Island, New York, 5-8 October 1998, vol. 2, pp. 994-1002.

Zimin, V.G., Asaka, H. and Anoda, Y., 2000. Analysis of NEACRP 3D BWR Core Transient Benchmark, Proc. Int. Conf. on Supercomputing in Nuclear Applications (SNA 2000), Tokyo, Japan, 4-7 September 2000.