BEST-ESTIMATE TRANSIENT ANALYSIS WITH SKETCH-INS/TRAC-BF1, ASSESSMENT AGAINST OECD/NEA BWR TURBINE TRIP BENCHMARK

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ABSTRACT

Incorporation of three-dimensional models of the reactor core into plant transient codes allows for a best-estimate analysis concerning the interactions between reactor core dynamics and plant systematic functions. SKETCH-INS/TRAC-BF1 is the coupled best-estimate BWR transient analysis code system, which utilizes a modern nodal method with introducing an assembly discontinuity factor for the neutron kinetics and a two-fluid six-equation model for the thermal hydraulics. Assessment on the code system for the capability to predict plant operational transients has been performed in this study against the OECD/NEA BWR turbine trip benchmark, where the following phenomena were realized with the simulation. (1) An acoustical pressure wave propagated in the main steam line, entering into the reactor pressure vessel, produced a hump in the reactor dome pressure rise. (2) The pressure wave propagated into the reactor core induced the fluctuation in core flow rates with out-of-phase, which caused void collapse in core region. (3) The reactivity insertion due to void collapse made the core power peak and the power generation shift upwards during the transient. Simulation on these phenomena demonstrated the capability of SKETCH-INS/TRAC-BF1 code system to predict the plant transients that include interactions between reactor core dynamics and plant systematic functions.

1. INTRODUCTION

Advancement in numerical methodology and computational technology has made feasible to develop code systems coupled reactor thermal hydraulics and core neutronics. Incorporation of three-dimensional models of the reactor core into plant transient codes allows for a best-estimate analysis concerning the interactions between reactor core dynamics and plant systematic functions. The Transient Reactor Analysis Code (TRAC) was developed as a best-estimate code with an aim to make the more realistic prediction excluding an excessive conservatism based on the most suitable field equations and constitutive equations in the state-of-the-art. TRAC-BF1[1] is the latest public domain boiling water reactor (BWR) version of TRAC. Although axial variation of neutron kinetics could be simulated in a one-dimensional core model available in TRAC-BF1, further improvement in core model was required to simulate reactor core dynamics coupled with neutronics and thermal hydraulics in time and space domain.

This paper presents the coupled best-estimate BWR transient analysis code system SKETCH-INS/TRAC-BF1, which utilizes a modern nodal method for the neutron kinetics and a two-fluid six-equation model for the thermal hydraulics. The code system has been originally developed in Japan Atomic Energy Research Institute (JAERI)[2] by a coupling of TRAC-BF1 with the three-dimensional neutron kinetics code SKETCH-N[3]. The coupled code system has been modified as SKETCH-INS/TRAC-BF1 in Nuclear Power Engineering Corporation (NUPEC)[4] to apply it to an audit analysis on the BWR stability[5] and other plant transients. Assessment on the code system for the capability to predict plant operational transients has been performed in this study against the OECD/NEA BWR turbine trip benchmark[6].

2. OUTLINE OF BWR TURBINE TRIP BENCHMARK

OECD/NEA has recently organized a BWR turbine trip benchmark project under the sponsorship of USNRC for the purpose of validating an advanced best-estimate plant transient analysis code providing a fully defined exercise with a comprehensive set of analytical specifications. Transient tests conducted in an actual plant are suitable to assess the code predictive capability of plant operational transients that include interactions between reactor core dynamics and plant systematic functions. In a reactor, fuel cooling and neutron moderation are governed by thermal hydraulics and core power generation by neutronics, where the core boundary conditions are affected by responses of plant systematic functions. The reference experimental data has been come from a test series conducted in the 1,100 MWe BWR Peach Bottom 2 nuclear power plant[7]. The plant specifications are summarized in Table 1.

Table 1. Peach Bottom 2 plant specifications

Item	Specification
Reactor type	1,100 MWe BWR
Rated core power	3,293 MWt
Rated core flow rate	12,915 kg/s
Rated dome pressure	7.03 MPa
Fuel assembles	764 assembles
Fuel type	7×7, 8×8 type
Control rods	185 rods

The turbine trip test[8] was conducted at the end of cycle 2 in April 1977 with the initial core power and flow rates being 61.6 and 80.9% of the rated, respectively. Table 2 shows the turbine trip test sequence. The test was done by manually tripping the turbine, which automatically initiated a rapid closure of the turbine stop valves (TSVs). The turbine bypass valves (TBVs) were opened to prevent the reactor from being overpressurized. A reactor scram initiation was intentionally delayed up to an averaged power range monitor (APRM) high flux scram set point of 95% of the rated in order to allow a considerable rise of the core fission power. The turbine trip in a BWR is a pressurization transient caused by a sudden steam flow blockage inducing the void collapse in core region in which the interactions between reactor core dynamics and plant systematic functions play an important role. The core dynamics during the transient are dominated by the void behavior, because it affects both the neutronics and the thermal hydraulics. The core power response is a result of neutronic and thermal-hydraulic interactions as affected by plant systematic functions. Code versus data comparisons were made for the responses of the plant integrated parameters during the transient.

Table 2. Turbine trip test sequence (time in sec)

TSVs	TBVs	Scram
0.0 : start to close 0.096: full closed	0.06: start to open 0.846: full open	0.63: initiated 0.75: start of motion 3.71: full insertion

3. SKETCH-INS/TRAC-BF1 CODE SYSTEM

The code system has been originally developed in JAERI[2] 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. Assessment on the coupled code system has been performed against the OECD/NEA BWR cold water injection benchmark[9]. The code system has been modified in detail as SKETCH-INS/TRAC-BF1 code system in NUPEC[4] to utilize it for the audit analysis on the BWR stability[5] and other plant transients. Table 3 summarizes the specifications of SKETCH-INS/TRAC-BF1 code system. The specifications are classified into four categories: neutron kinetics, thermal hydraulics, fuel heat transfer and plant system. As shown in the table, the specifications are in the state-of-the-art as a best-estimate analysis code coupled with neutronics and thermal hydraulics in time and space domain.

3.1 SKETCH-INS Code

The SKETCH-INS code is a modification of the SKETCH-N[3] code which was originally developed in JAERI. The SKETCH-INS code deals with neutron kinetics, which solves time-dependent diffusion equations in three-dimensional Cartesian coordinates. The code treats two neutron energy groups and six groups of delayed neutron precursors. In order to improve the spatial resolution accuracy, an assembly discontinuity factor (ADF) has been implemented in the code upon the original one. Reactivity feedback is taken into account with moderator density, fuel temperature, control rod motion and reactor scram. The ANS-1979 standard decay heat model has been implemented in the code. Direct gamma heating is taken into account for the in-channel active coolant flow. Numerical methods for the neutronic calculations are as follows. Polynomial and semi-analytical nodal method based on the nonlinear iteration procedure[10] 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[1] is the latest public domain BWR version of TRAC, which deals with thermal hydraulics, fuel heat transfer and plant system. Thermal hydraulics utilizes the two-fluid model that solves six balance equations of mass, momentum and energy for liquid and vapor phases. Two-phase flow in core region is treated as one-dimensional parallel vertical flows. Heat transfer model solves one-dimensional radial heat conduction equations. Heat transfer coefficients at the cladding surface are subdivided into each flow regime. Material properties are based on the code installed MATPRO correlations. TRAC-BF1 has major BWR component modules: Vessel, Channel, Separator-Dryer, Pump, Pipe, Valve and etc. The plant system can be composed of these component modules. The reactor pressure vessel is modeled with the Vessel component in three-dimensional cylindrical coordinates. Numerical methods for the thermal-hydraulic calculations are as follows. Standard finite differential method with staggered mesh is used for space integration of both fluid flow and heat conduction. Time integration of the fluid flow equations is performed by the semi-implicit scheme with the stability enhanced two-step (SETS) method.

3.3 Coupling Interface

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 reference[2].

Table 3. Specifications of SKETCH-INS/TRAC-BF1

Item		Specification
Neutron Kinetics	Dimension	3D XYZ（1 bundle/1 node）
	Equation	Time-dependent diffusion equation
	Prompt neutron	Two groups
	Delayed neutron	Six groups
	Spatial resolution	Assembly discontinuity factor considered
	Decay heat	ANS-1979 standard decay heat model
	Direct heating	Gamma heating in active coolant
	Numerical method	Spatial integration: modern nodal method Time integration: fully-implicit scheme
Thermal Hydraulics	Dimension	1D axial (multi-channel)
	Fluid model	Two-fluid model
	Equation	Six balance equations: mass, momentum and energy conservation for liquid and vapor phases
	Numerical method	Spatial integration: differential method Time integration: semi-implicit scheme
Fuel Heat Transfer	Dimension	1D radial (for each axial node)
	Equation	Heat conduction equation
	Heat transfer	Heat transfer coefficients in each flow regime
Plant System		Reactor vessel, Recirculation loop, Jet pump, Steam separator, Main steam line, etc.

4. ANALYSIS

4.1 Analytical Model

An extensive set of the data has been given in the benchmark specifications (Solis et al., 2001) with files on CD-ROM. An input data set includes general description of the reactor system, core neutronics data, thermal-hydraulic data and the plant systematic characteristics data during the transient. The Peach Bottom 2 plant nodalization model set up with the TRAC-BF1 code is shown in Figure. 1. The plant system was represented with the reactor connected with a recirculation loop, a main steam line and a feedwater line. The reactor was composed of the reactor pressure vessel, core and internal components. The pressure vessel was modeled with two radial rings and fifteen axial levels in cylindrical coordinates. The inner ring represented the core region, lower and upper plenum and the steam dome. One separator component was specified in the inner ring. The outer ring represented the reactor annular downcomer. One jet pump component was specified in the outer ring. The specification of each axial level was associated with the boundary of the reactor internal structures. The main steam lines were lumped into one line connected with TSVs, the steam bypass line and safety relief valves (SRVs) and ended by a turbine break. The steam bypass lines were lumped into one line connected with TBVs and ended by a condenser break. Homogeneous equilibrium mixture (HEM) critical flow was assumed for the steam bypass flow discharging through TBVs. The boundary conditions at a steady-state were given with a pressure boundary for the main steam line and a flow boundary for the feedwater line. The plant systematic functions such as the TSVs closure, TBVs opening, the reactor scram and feedwater flow transient were specified as the boundary conditions during the transient.

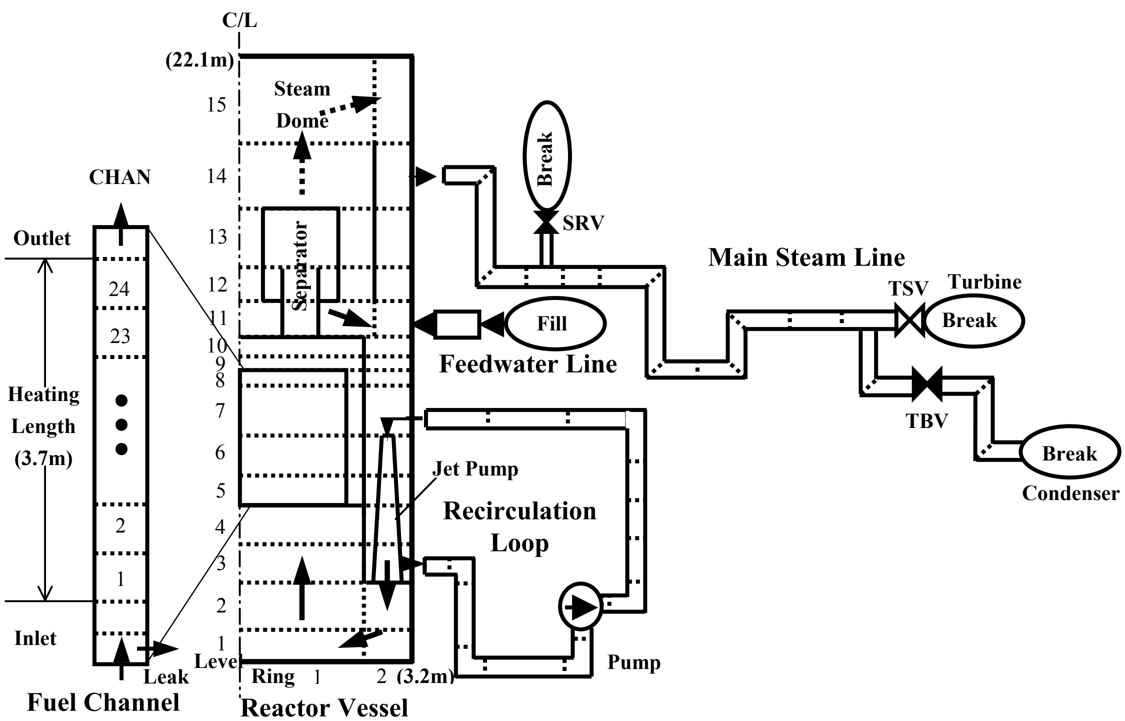
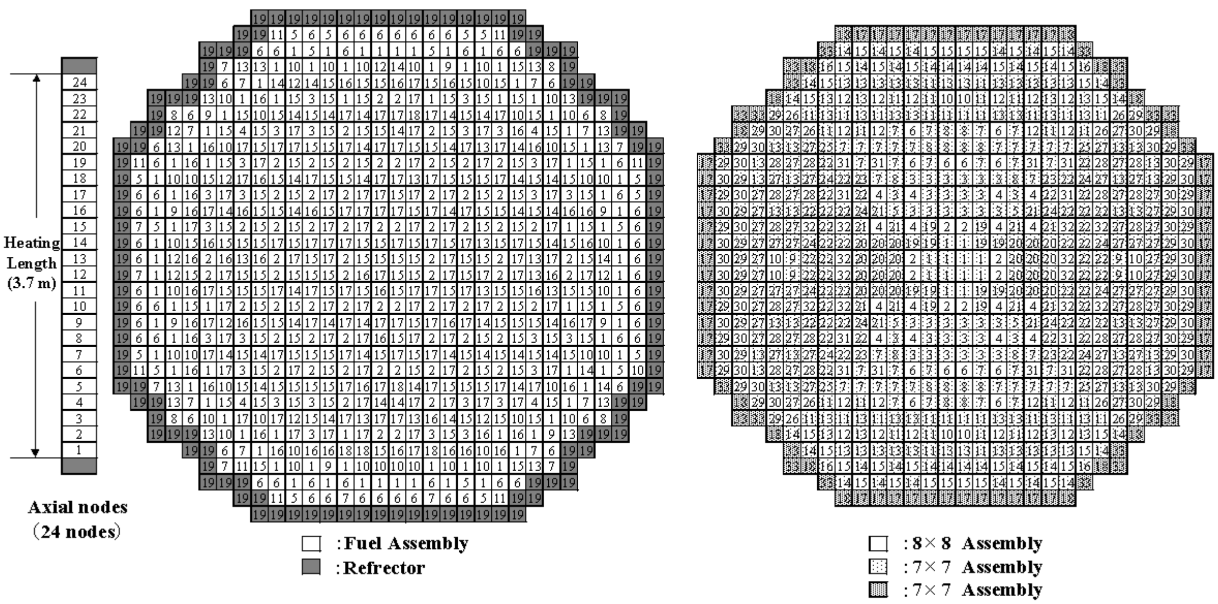


Figure 1. Peach Bottom 2 plant nodalization model

The Peach Bottom 2 core nodalization model is shown in Figure. 2. The neutronics nodalization model was set up with the SKETCH-INS code. There were 764 fuel assemblies loaded in the core region. The core region was nodalized into an individual fuel assembly divided by one radial node per assembly. Axially, the core region was divided into 24 nodes. In addition, the reflector layer enveloped the core region with each one node for the side, top and bottom. These assemblies were represented with 19 neutronic assembly types, with the distribution shown in Figure. 2(a). A set of diffusion coefficients, macroscopic cross-sections for scattering, absorption and fission, ADFs defined as a function of the moderator density and fuel temperature were provided from the benchmark cross-section library. Fuel assemblies in the core were represented with channel components in TRAC-BF1, as shown in Figure. 2(b), which were collapsed into 33 thermal-hydraulic channels. The heating length of the channel was divided into 24 nodes consistent with the neutronics nodalization. The core bypass flow was simulated with a leak path model from channel inlet to the core bypass region.



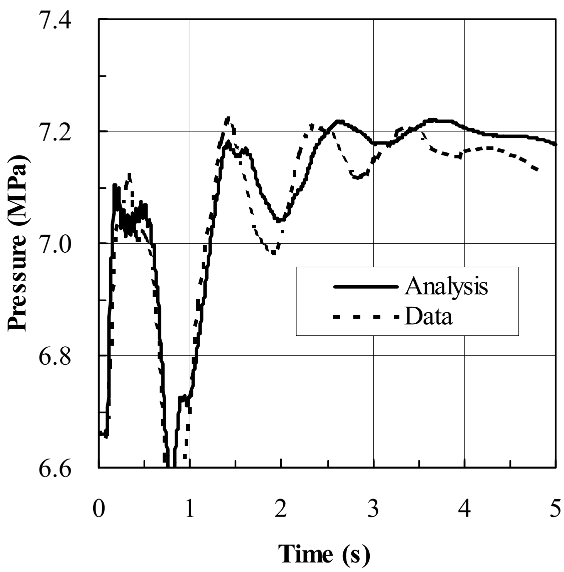
(a) Neutronic assembly types (19 types)

(b) Thermal-hydraulic channels (30 channels)

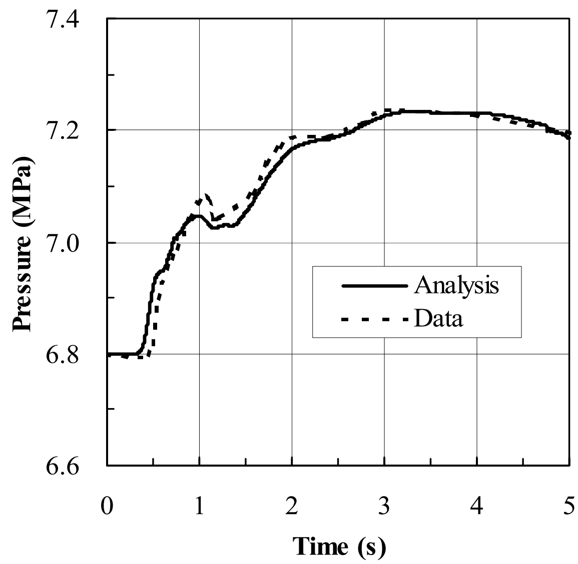
Figure 2. Peach Bottom 2 core nodalization model

4.2 Analytical Results

The analysis has simulated the BWR turbine trip transient progression. Code versus data comparisons were made for the responses of the plant integrated parameters during the transient. Time histories of turbine inlet pressure and reactor dome pressure are compared with the measured data in Figure.3. An acoustical pressure wave was clearly seen in the comparison of turbine inlet pressure. The oscillation period and the amplitude of pressure wave were simulated by the analysis. Once the TCVs were closed, the reactor dome pressure rose within a few seconds and the pressurization was moderated by the TBVs opening. An acoustical pressure wave propagated in the main steam line, entering into the reactor pressure vessel, produced a hump in the reactor dome pressure. The analysis simulated the behavior of reactor dome pressure that was dominated by the functions of TSVs and TBVs.



(a) Turbine inlet pressure



(b) Reactor dome pressure

Figure 3. Time histories of pressure

Time histories of reactor core flow rates are compared with the measured data in Figure.4. The analytical result of void fraction in average channel is also shown in th figure. The pressure wave propagated into the reactor core induced the fluctuation in core flow rates with out-of-phase as follows. The inlet core flow rate increased and outlet core flow rate decreased, which caused void collapse in the core region. When a pressure wave was away from the reactor vessel, the phases reversed resulting in void recovery. These void behaviors should induce the reactivity insertion into the core.

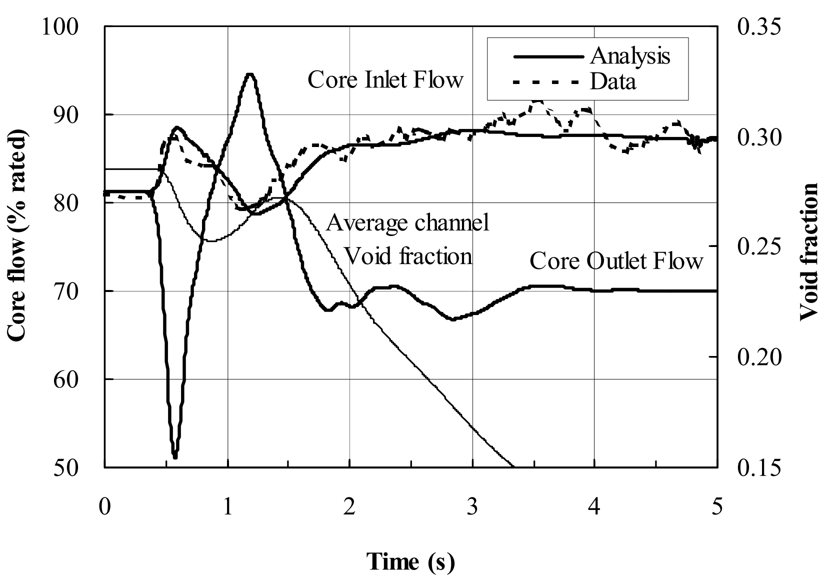


Figure 4. Time histories of core flow rates

Test data versus analytical results were compared in Figure. 5 for the reactor power. Time history of core fission power was compared with the APRM data. The core fission power once increased rapidly with the reactivity insertion and depressed due to the negative reactivity feedback followed by the reactor scram. Figure 6 shows snapshots of the core averaged axial power distribution during the transient. The core averaged axial power distribution was middle peaked at the initial steady-state, where the analyzed power shape was skewed slightly upwards than the P1 data. The core power generation shifted upwards during the transient with the axially varying reactivity insertion due to void collapse in the core region. The analysis simulated the axial transition of core fission power that was dominated by the space-dependent reactivity insertion due to void collapse.

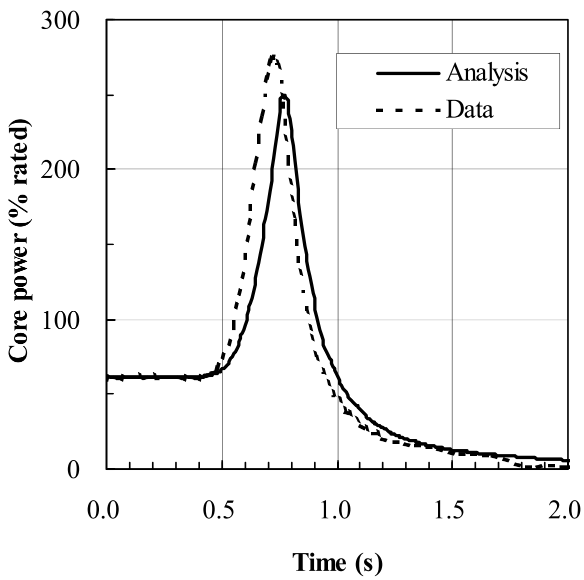


Figure 5. Time histories of core power

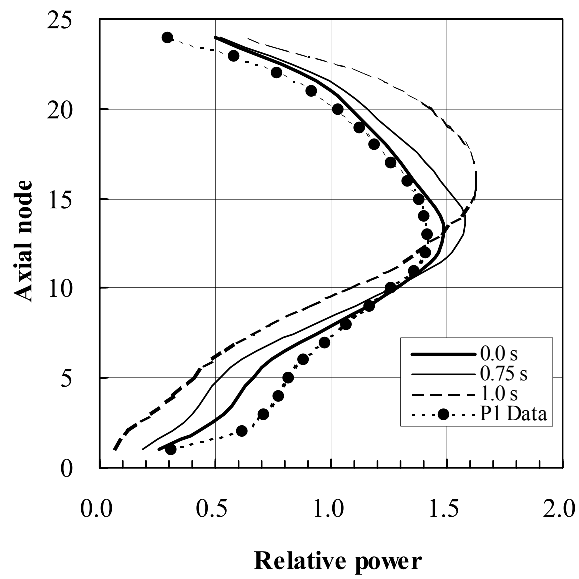


Figure 6. Transition of axial power distribution

Table 4 contains quantitative comparisons of the test data versus analytical results for peak responses of the reactor dome pressure and core fission power. The Initial peak of reactor dome pressure was underestimated by about 0.03 MPa and the maximum value was agreed well with the data. The peak core fission power was underestimated by relatively 10 % of the peak vale as compared with the data. It was recognized that the initial rate of pressure rise strongly affected the core fission power peak.

Table 4. Quantitative comparisons of peak responses

Parameter	Data	Analysis
Reactor dome pressure		
Time to initial peak (s)	1.09	1.00
Time to maximum (s)	3.02	3.28
Initial peak pressure rise (MPa)	0.28	0.25
Maximum pressure rise (MPa)	0.44	0.44
Reactor core power		
Time to peak (s)	0.73	0.77
	279	250

Peak core fission power (% rated)	

5. CONCLUSIONS

Assessment on SKETCH-INS/TRAC-BF1 code system for the capability to predict plant operational transients has been performed against the OECD/NEA BWR turbine trip benchmark, where the following phenomena were realized with the simulation. (1) An acoustical pressure wave propagated in the main steam line, entering into the reactor pressure vessel, produced a hump in the reactor dome pressure rise. (2) The pressure wave propagated into the reactor core induced the fluctuation in core flow rates with out-of-phase, which caused void collapse in core region. (3) The reactivity insertion due to void collapse made the core power peak and the power generation shift upwards during the transient. Simulation on these phenomena demonstrated the capability of the code system to predict the plant transients that include interactions between reactor core dynamics and plant systematic functions.

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