Preliminary Proc. of the OECD/CSNI Workshop on Advanced Thermal-Hydraulic and Neutronics Codes: Current and Future Applications, Barcelona, Spain, 10-13 April, 2000, voil.2, pp. 1-15.

Coupling of the Thermal-Hydraulics TRAC Codes with 3D Neutron Kinetics Code SKETCH-N

Hideaki Asaka, Vyacheslav G. Zimin, Tadashi Iguchi and Yoshinari Anoda

Department of Reactor Safety Research
Japan Atomic Energy Research Institute
Tokai-mura, Ibaraki-ken, 319-1195, Japan,
Fax: +81-29-282-6728; Phone: +81-29-282-5357;
E-mail: asaka@lstf3.tokai.jaeri.go.jp

Abstract: Since 1997, the Japan Atomic Energy Research Institute (JAERI) has been developing the three-dimensional transient analysis codes: J-TRAC/SKETCH-N code system for pressurised water reactor (PWR), and the TRAC-BF1/ SKETCH-N for boiling water reactor (BWR). The J-TRAC code, developed by JAERI, has a framework based on the TRAC-PF1, which is best estimate system transient analysis code having the capability to simulate PWR system components and three-dimensional core in cylindrical R-θ-Z coordinates. The TRAC-BF1 is a BWR system analysis code. The neutronics code SKETCH-N can solve the steady-state and kinetics forms of neutron diffusion equations in X-Y-Z geometry. The efficient polynomial and semi-analytic nodal methods based on the nonlinear iteration procedure are used in the code. Coupling with the TRAC codes is performed using the interface module based on the message-passing library Parallel Virtual Machine (PVM). The interface module is responsible for data transfer between the codes, mapping of the data between different spatial meshes and synchronisation of the time stepping. Verification of the coupled TRAC/SKETCH code system has been performed against light water reactor (LWR) transient benchmark problems. J-TRAC/SKETCH has been verified by NEACRP PWR rod ejection benchmark and NEA/NSC PWR rod withdrawal benchmark. Verification of the TRAC-BF1/SKETCH-N has been performed using NEACRP BWR cold water injection benchmark. TRAC/SKETCH numerical results and a comparison with the other codes are given in the paper. The second part of the paper presents current status of the out-of-pile experiments at JAERI on BWR stability to assess and improve the TRAC-BF1 code

1. INTRODUCTION

Best-estimate analysis of many light water reactor (LWR) transients requires 3D neutron kinetics modelling of the reactor core (Diamond, 1996). To provide such capabilities the popular system transient analysis codes have been coupled with 3D neutron kinetics models: ATHLET code coupled with several neutron kinetics codes (Langenbuch et. al, 1996), RELAP5/NESTLE (Judd et al., 1994), RETRAN-3D/ARROTTA (Goose et al., 1998), TRAC-PF1/NEM (Bandini, 1998) etc. 3D LWR core transient benchmarks (Finnemann and Galati, 1992) were carried out by Nuclear Energy Agency through its Reactor Physics Committee (NEACRP) to give the first survey of the state of the art in this area. These benchmarks describe several control rod ejection transients in pressurised water reactor (PWR), and cold water injection and core pressurisation events in boiling water reactor (BWR). The benchmarks have been extensively used to verify coupled neutronics/thermal-hydraulics code systems; the first results of the benchmark participants are published in the benchmark report (Finneman et al., 1993). For PWR applications these benchmarks are followed by a PWR rod withdrawal benchmark (Fraikin and Finnemann, 1993) and a main steam line benchmark (K. Ivanov et. al., 1999), which is presently going on. BWR stability benchmark provides the experimental data of the Swedish reactor Ringhals 1 (Lefvert, 1994). This benchmark has been organised for validation of computer codes and models applied in the BWR stability analysis.

An interest to the best-estimate analysis of the reactivity-initiated accidents in PWR and out-of-phase instabilities in BWR results in the TRAC/SKETCH project in Japan Atomic Energy Research Institute (JAERI) has been started in 1997. 3D neutron kinetics code SKETCH-N (Zimin and Ninokata, 1998) has been coupled with the best estimate transient analysis codes: J-TRAC (Akimoto et. al., 1989) for PWR problems and TRAC-BF1 (Borkowski et. al., 1992) for BWR applications. The paper outlines the main features of the TRAC and SKETCH codes and describes the coupling interface module. Verification of the coupled J-TRAC/SKETCH code system has been performed against PWR rod ejection and rod withdrawal benchmarks. NEACRP BWR cold water injection benchmark has been computed by the TRAC-BF1/SKETCH code system. Numerical results of the TRAC/SKETCH codes are given and discussed in the paper.

The paper is organised as follows. Section 2 gives the main features of the both TRAC codes. Section 3 describes the nodal neutron kinetics code SKETCH-N. Section 4 outlines the main features of the interface module, which is responsible for the data transfer between the codes and synchronisation of the time stepping. Application of the J-TRAC/SKETCH-N code system for the NEA/NSC PWR rod withdrawal benchmark is described in the Section 5. Section 6 contains the description of the NEACRP BWR cold water injection benchmark and gives the numerical results computed by the TRAC-BF1/SKETCH-N code. Section 7 presents current status of out-of-pile BWR experiments performed in JAERI and their analyses by the TRAC-BF1 code. Section 8 draws the conclusions and outlines present activities.

2. J-TRAC AND TRAC-BF1 CODES

J-TRAC code (Akimoto et al., 1989) is the best-estimate two-fluid thermal-hydraulic code TRAC-PF1/Mod1 (Lilies et al., 1988) modified in JAERI to improve the simulation of reflooding phenomena during loss-of-coolant accidents. The verification of the reflooding model of the J-TRAC code is given in (Akimoto et. al., 1988), summary of the TRAC-PF1/MOD1 assessment results is presented in (Sahota and Addessio, 1985).

The J-TRAC code has models of major PWR components: Accumulator, Pressurizer, Pump, Steam Generator, Turbine, Vessel et al. Using these components and control system, a code user can simulate many PWR transients. The reactor core is modelled by the Vessel component, which solves six two-phase flow fluid dynamics equations in R-θ-Z geometry, heat conduction equations in the fuel rods and point kinetics equations to calculate the power distribution.

TRAC-BF1 is a version of the TRAC code developed for BWR analysis (Borkowski et al., 1992). The code additionally has specific BWR components: Channel, Jet Pump, Separator-Dryer, and Feedwater Heater. Numerical methods for fluid dynamics calculations are improved in the TRAC-BF1 version and 1D neutron kinetics model has been added as an option.

The neutron kinetics models significantly limit the best-estimate capabilities of the both TRAC codes. To remove this limitation a coupling with a 3D neutron kinetics code SKETCH-N has been performed.

3. SKETCH-N CODE

The SKETCH-N code (Zimin and Ninokata, 1998) solves neutron diffusion equations in X-Y-Z geometry for steady-state and neutron kinetics problems. The code can treat an arbitrary number of neutron energy groups and delayed neutron precursors. Polynomial, semi-analytic and analytic nodal methods based on the nonlinear iteration procedure can be used for spatial discretization of diffusion equations (Zimin et al., 1998). Time integration of the neutron kinetics problem is performed by the fully implicit scheme with an analytical treatment of the delayed neutron precursors. Steady-state eigenvalue problems are solved by inverse iterations with a Wielandt shift, the Chebyshev adaptive iterative procedure is used for the neutron kinetics problems. The block symmetric successive overrelaxation method (SSOR) is applied as a preconditioner in the both iterative procedures. Automatic time step control procedure based on the time step doubling technique is used in the code.

An extensive set of the steady-state and neutron kinetics Light Water Reactor (LWR) benchmarks has been calculated to verify the SKETCH-N code. The steady-state problems include: the classical 2D & 3D International Atomic Energy Agency PWR problems; 2D Biblis PWR checker-board-loaded core; 2D Zion-1 PWR problem with explicit modelling of the baffle; 2D 4-group Koeberg PWR checker-board-loaded core with realistic cross sections including up-scattering (Zimin et al., 1998). The neutron kinetics module has been verified against 3D Langenbuch-Maurer-Werner (LMW) operational transient in a small PWR model; and 2D & 3D super-prompt-critical rod drop accident in Boiling Water Reactor (BWR) (Zimin and Ninokata, 1998). The verification results show that the SKETCH-N code has acceptable accuracy and efficiency to be used in the LWR safety analysis and design.

4. COUPLING AND DATA TRANSFER BETWEEN THE CODES

A coupling of the SKETCH code with the both TRAC codes is performed using an interface module. The codes are treated as separate processes, and the interface module is responsible for the data exchange between the codes, data mapping between the spatial meshes of the codes and a synchronisation of the time stepping. The interface module is based on the message-passing library PVM (Geist et al., 1994). After performing the TRAC input, the TRAC enrols into PVM and spawns a child process SKETCH-N. When SKETCH-N is started, it gets the TRAC ID number under PVM and the codes can communicate to each other sending/receiving messages. At the beginner of time step, TRAC sends a message to SKETCH with the thermal-hydraulics core data and a proposal for the time step size. SKETCH receives the message, chooses a new time step size and performs the neutronics calculations. Then, SKETCH sends a message to TRAC with the power distribution and the used time step size. TRAC receives the message and performs the thermal-hydraulics calculation. The procedure is repeated till the end of the transient.

Actually time stepping procedure is slightly more complicated due to an automatic time step control based on the time step doubling technique as illustrated in Fig 1. SKETCH performs the calculations using two temporal meshes: fine temporal mesh, which consists of pairs of the time steps of equal length, and coarse temporal mesh with a double time step size. A comparison of the power distribution computed on these meshes gives an estimate of the SKETCH time step size. Finally, a time step size of the TRAC/SKETCH code system is selected as a minimum of the time step sizes proposed by the codes.

The TRAC and SKETCH-N codes usually perform calculations using different spatial meshes. Additionally J-TRAC code applies axially staggered meshes for the fluid dynamics and heat conduction models. The coupled TRAC/SKETCH-N calculations require a mapping of the data between the neutronics, heat conduction and fluid dynamics meshes. The mapping procedure based on the mapping matrix approach has been developed. For each pair of the spatial meshes a code user define two mapping matrices: a mapping matrix describing a correspondence of the 2D radial meshes and a mapping matrix for 1D axial meshes. Details of the mapping procedure is given in (Zimin et al., 1999), we only note that a choice of the spatial meshes of the codes can be arbitrary, the mapping procedure provides the consistent data exchange between the meshes.

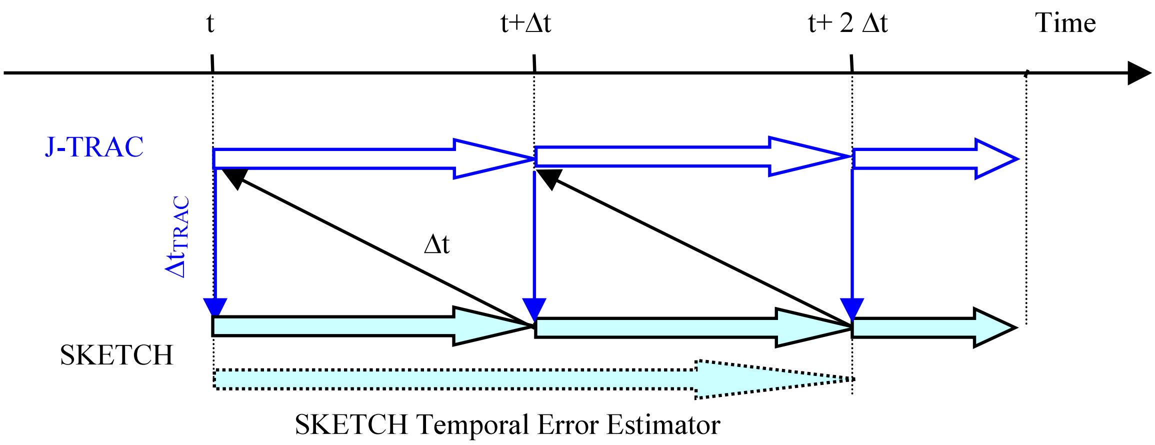


Figure 1. Time Stepping of the Coupled TRAC/SKETCH code system

The list of the variables, which transfer between the meshes, is given as follows

- from the neutronics mesh into the heat conduction mesh:

heat generation rate of the heat conduction nodes;

- from the neutronics mesh into the fluid dynamics mesh:
  heat generation rate of the fluid dynamics nodes due to direct coolant heating;
- from the heat conduction mesh into the neutronics mesh:
  average Doppler fuel temperature of the neutronics nodes;
- from the fluid dynamics mesh into the neutronics mesh:
  average coolant temperature and coolant density of the neutronics nodes.

Variables transferring from the TRAC code depend on the macro cross section model of the SKETCH and can be changed for a new reactor type or a new problem.

5. J-TRAC/SKETCH-N RESULTS OF PWR BENCHMARK ON UNCONTROLLED WITHDRAWAL OF CONTROL ROD

A coupling of the TRAC/SKETCH codes has been verified against international LWR benchmark problems. For verification of J-TRAC/SKETCH-N code we performed calculations of the NEACRP PWR rod ejection benchmark and NEA/NSC PWR rod withdrawal benchmark. The results of the rod ejection benchmark have been already published (Zimin et al., 1999) and the results of the rod withdrawal benchmark are given below.

PWR rod withdrawal benchmark (Fraikin and Finnemann, 1993) is an extension of the NEACRP PWR rod ejection benchmark (Finnemann and Galati, 1992). The reactor core model is the same; the only difference that a central control cluster added for symmetry in the rod ejection problem has been removed in the rod withdrawal benchmark. The reactor core configuration is illustrated in the Fig 2. The transient is initiated by a withdrawal of control rod banks from an initial critical core at the hot zero power (HZP). The scram signal is generated when the power level reaches 35 % of the nominal power. The control rods begin to fall after a time delay 0.6 seconds. All control rods participate to scram.

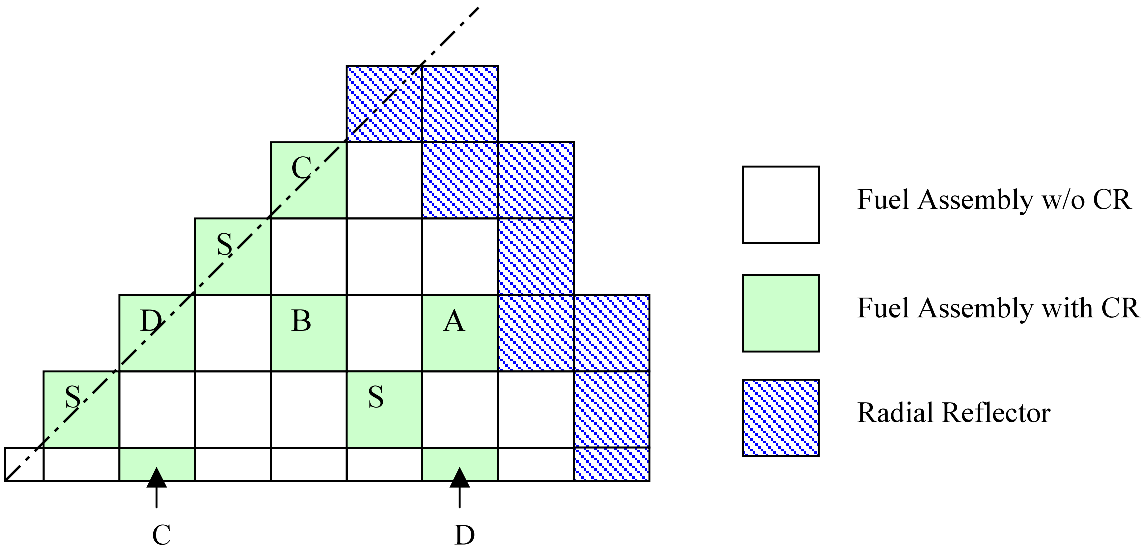


Figure 2. NEA/NSC PWR Rod Withdrawal Benchmark; Core Geometry and Arrangement of Control Rod Banks

Four cases are defined in the benchmark as follows:

Case A: single bank D is withdrawn; other banks (C, B, A, S) remain fully withdrawn till scram.

Case B: two banks (B and C) are withdrawn, banks A and D remain fully inserted, the bank (S) is fully withdrawn.

Case C is the same as the case B, except that the heat transfer coefficient between cladding and coolant is set to a constant.

Case D starts from the same initial conditions as the case B, but two peripheral banks A and B are withdrawn, banks C and D remains fully inserted and bank S are fully withdrawn till scram.

The cases B, C and D give similar results, only the cases A and B have been computed by the TRAC/SKETCH code system.

The benchmark report (Fraikin, 1997) provides the numerical results computed by ten participants from ten countries. The results include the reactor average data (reactor power, average Doppler fuel temperature etc.) as well as the most important local safety parameters: hot pellet enthalpy, maximum fuel temperature at the centerline of hot pellet, maximum hot pellet cladding outer surface temperature etc. The results are compared against the reference solution generated by the PANTHER code using fine temporal and spatial meshes. There is a good agreement between the codes on the reactor-averaged data, while there are some discrepancies in the hot pellet results.

5.1 SKETCH-N Model

The neutronics model has two neutron energy groups and six groups of delayed neutron precursors. Reactor core is specified at the beginner of cycle 1. The macro cross sections are given as polynomial functions of the boron concentration, effective fuel temperature, coolant density and coolant temperature. The SKETCH-N calculations were performed using a quarter-core representation. Several spatial meshes were used for neutronics calculations: 9x9x18 – 1 node per assembly (npa) and 18 axial layers specified in the benchmark; 9x9x30 – 1 npa and 30 axial layers; 17x17x30 – 4 npa and 30 axial layers. In the initial condition, a reactor is critical and a value of the critical boron concentration was computed. An automatic time step control procedure based on the time step doubling technique was used to determine the time step size. A temporal truncation error tolerance was set to 5×10^{-3}.

5.2 J-TRAC Model

The VESSEL component is used to simulate the reactor. Spatial mesh in R-θ plane contains 27 nodes (3 rings and 9 sectors), each of the 26 nodes simulates a single fuel assembly of the 1/8 core representation and 1 node models a radial reflector. Each of three small axial neutronics layers on the top and on the bottom of the reactor core are combined into the single fluid dynamics layer, the resulting axial spatial mesh has 14 layers. The radial heat conduction mesh has 9 zones in the fuel, 1 in the gap and 2 in the cladding. The reactor boundary conditions are given using the FILL components on the bottom, which define the mass flow rate and the BREAK components on the top, which specify the reactor pressure. Zero lateral cross flow is simulated setting to zero the cross flow areas in R and θ directions. The material properties defined in the benchmark are implemented as an option in the J-TRAC code. A direct coolant-heating model has been also developed for the VESSEL component of the J-TRAC code.

5.3 Numerical Results

Table 1 represents the steady-state results computed using different neutronics spatial meshes. Even with the coarsest spatial mesh 9x9x18, the J-TRAC/SKETCH-N results are in the excellent agreement with the PANTHER reference solution (Fraikin, 1997) confirming accuracy of the semi-analytic nodal method of the SKETCH-N code.

Before considering transient results one problem should be addressed. In the benchmark specification, a hot pellet is defined as the neutronics node with maximum power density at the *finest spatial mesh* available during the calculations. Following the definition a pin power reconstruction technique should be used to recover the power distribution inside fuel assembly and supplemental fuel rods should be used in the J-TRAC code to compute hot pellet fuel temperature and hot pellet fuel enthalpy. Pin power reconstruction is not presently available in the SKETCH code and hot pellet is defined as a node of the fuel assembly with maximum power density. The hot pellet thermal-hydraulics data are given respectively for this node.

Table 1. J-TRAC/SKETCH-N Steady-State Results of the NEA/NSC PWR Rod Withdrawal Benchmark.

Parameter	Case A			Case B			
	J-TRAC/ SKETCH-N		PANTHER	J-TRAC/SKETCH-N			PANTHER
	9x9x18	17x17x30	25x25x48	9x9x18	9x9x30	17x17x30	25x25x48
Critical boron concentration (ppm)	1268.5 *+5.8 ppm*	1262.8 *+0.1 ppm*	1262.7	795.6 *+ 2 ppm*	795.6 *+ 2 ppm*	793.7 *+0.1 ppm*	793.6
1D axial peak factor	1.512 *-0.07 %*	1.513 *0.0 %*	1.513	1.508 *0.06 %*	1.508 *0.06 %*	1.508 *0.06 %*	1.507
2D radial peak factor	1.235 *-0.6 %*	1.243 *+0.08 %*	1.242	1.902 *-0.5 %*	1.902 *-0.5 %*	1.911 *-0.06 %*	1.912
3D peak factor	1.871 *-0.5 %*	1.882 *+0.07 %*	1.880	2.872 *-0.5 %*	2.872 *-0.5 %*	2.885 *-0.03 %*	2.886

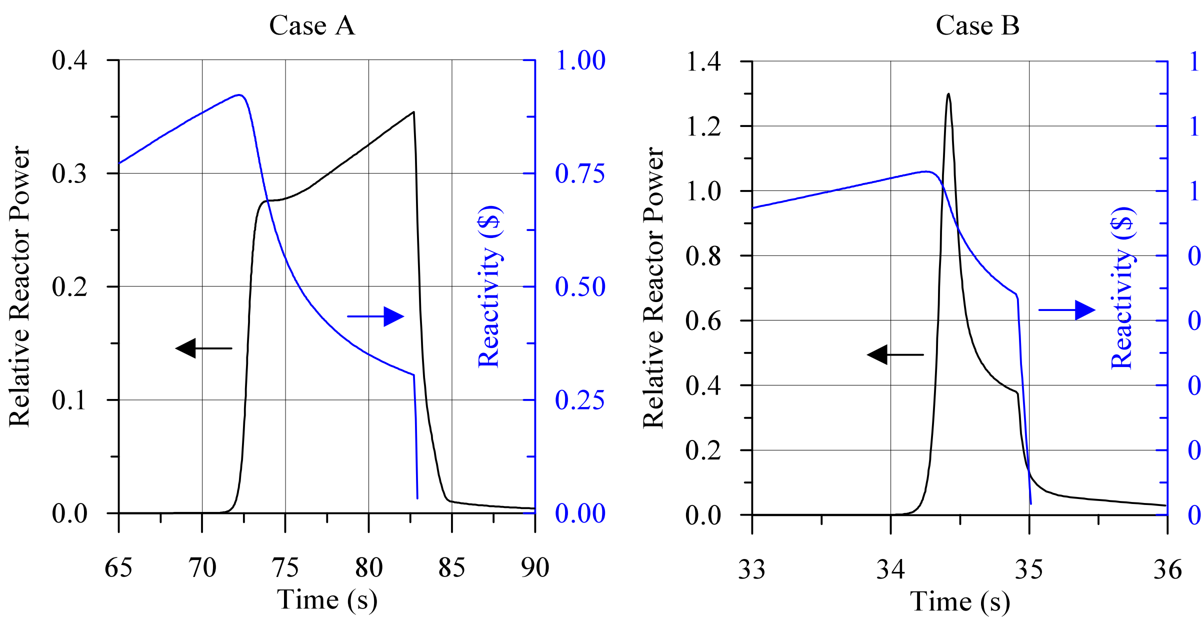


Figure 3. NEA/NSC PWR Rod Withdrawal Benchmark. Power and Reactivity versus Time

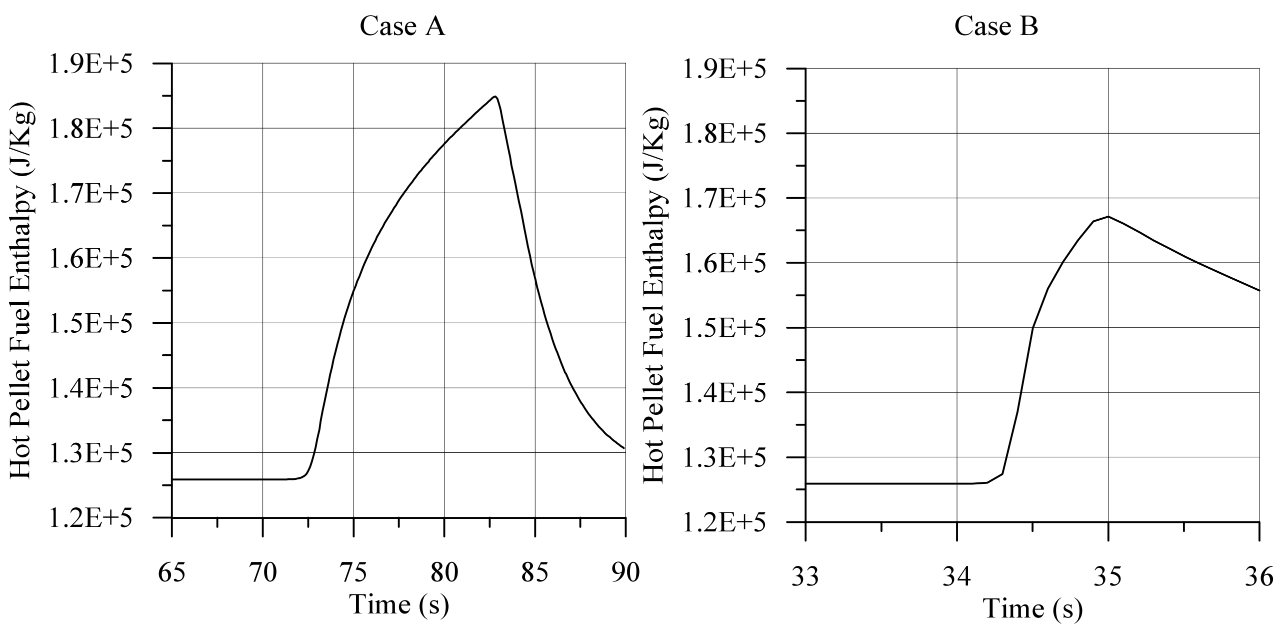


Figure 4. NEA/NSC PWR Rod Withdrawal Benchmark. Hot Pellet Fuel Enthalpy versus Time.

Fig. 3 shows the reactor power and reactivity versus time for the cases A and B. The transients are different in these two cases. The case B is a typical rod withdrawal transient usually considered for safety analysis. Two banks of the control rods are withdrawn. The power increase is very fast and exceeds significantly the trip level. In the case A, where only one rod bank is moving, the reactivity insertion rate is much lower, reactivity remains slightly below prompt-criticality. The power surge is terminated by the Doppler feedback when the power level slightly below 30 % of nominal, however the control rod withdrawal continues and power slowly increases till the scram level. Relatively high power level during a long period of time results that the energy deposited into the fuel, hot pellet fuel enthalpy and fuel temperatures are higher in the case A than in the case B. Fig. 4 illustrates this point showing the fuel enthalpy of the hot pellet for the both cases A and B.

Table 4 provides a comparison of the J-TRAC/SKETCH-N results with the reference PANTHER solution. The SKETCH calculations were performed using several spatial meshes. In the case A, there is an acceptable agreement with the PANTHER results even in the case of spatial mesh with 1 npa and 18 axial layers (mesh 9x9x18). In the hot pellet, the errors of maximum fission power, maximum fuel enthalpy, maximum centerline fuel temperature are -4 %, –6.7 % and –12 °C, respectively. An application of the finer spatial mesh (17x17x30) practically eliminates the error in the maximum fuel centerline temperature. The errors in power and fuel enthalpy are also decreased till –3 % and –4 %. The case B with larger reactivity insertion and faster power surge is more sensitive to a choice of the spatial mesh. The coarse spatial mesh (9x9x18) with 1 npa and 18 axial layer results in -16 % error in the reactor power peak, -28 % error in maximum fission power of the hot pellet and –14 % error in the maximum fuel enthalpy of the hot pellet. An application of the finer axial mesh (9x9x30) decreases the errors in the power peak till 2 % and the errors in the maximum power and fuel enthalpy of the hot pellet till -10 %. A refinement of the mesh in radial plane (4 npa, mesh 17x17x30) practically does not change these results. Case B illustrates that a rod cusping effect can be very significant in PWR transients, and a homogenization procedure for the partially rodded nodes used in the SKETCH-N code needs further improvement.

The differences in the hot pellet results between PANTHER and J-TRAC/SKETCH solutions are explained by the assembly-averaged definition of the hot pellet node in the J-TRAC/SKETCH code. Assembly-averaged data give the lower values of the hot pellet fission power and fuel enthalpy. An application of the pin power reconstruction procedure could significantly decrease these differences.

Table 2. J-TRAC/SKETCH-N Transient Results of the PWR Rod Withdrawal Benchmark.

Parameter	Case A			Case B			
	J-TRAC/SKETCH-N		PANTHER	J-TRAC/SKETCH-N			PANTHER
	9x9x18	17x17x30	25x25x48	9x9x18	9x9x30	17x17x30	25x25x48
Maximum fission power (%)	35.4 *-0.6 %*	35.4 *-0.6 %*	35.6	113.2 *-16 %*	132.2 *-1.9 %*	129.9 *-3.6 %*	134.8
Time of maximum fission power (s)	83.0 *+0.9 s*	82.7 *+0.6 s*	82.14	34.6 *+0.3 s*	34.34 *+ 0.04 s*	34.42 *+0.1 s*	34.30
Maximum fuel Doppler temperature (°C)	359.3 *+0.6 °C*	359.2 *+0.5 °C*	358.7	315.0 *-0.2 °C*	316.1 *+0.9 °C*	315.6 *+0.4 °C*	315.2
Max. fission power in hot pellet w/r to nominal power	0.84 *-4.2 %*	0.85 *-3.1 %*	0.8771	4.3 *-28 %*	5.4 *-10 %*	5.4 *-10 %*	6.0016
Maximum fuel enthalpy injected in hot pellet (J/Kg)	57502 *-6.7 %*	59005 *-4.3 %*	61662	39430 *-14 %*	41267 *-10 %*	41213 *-10 %*	45882
Time of maximum fuel enthalpy (s)	83.1 *+0.9 s*	82.8 *+0.6 s*	82.19	35.2 *+0.35 s*	34.9 *+0.05 s*	35.0 *+0.15 s*	34.85
Coolant temperature at outlet of hot channel (°C)	301.2 *-0.8 °C*	301.2 *-0.8 °C*	302.0	296 *-1.2 °C*	296 *-1.2 °C*	296 *-1.2 °C*	297.2
Maximum fuel temperature at centerline (°C)	620 *-12 °C*	629 *-3 °C*	631.8	474 *-4 °C*	482 *+ 4 °C*	482 *+ 4 °C*	478.2
Maximum cladding outer surface temperature (°C)	307 *-1.2 °C*	307 *-1.2 °C*	308.2	302 *-1.6 °C*	302 *-1.6 °C*	302 *-1.6 °C*	303.6

6. TRAC-BF1/SKETCH-N RESULTS OF NEACRP BWR COLD WATER INJECTION BENCHMARK

Verification of the TRAC-BF1/SKETCH code system is done using NEACRP BWR cold water injection benchmark. NEACRP BWR cold water injection (CWI) benchmark is proposed by H. Finnemann and A. Galati (1992) for verification of the coupled neutronics/thermal-hydraulics BWR codes. The reactor core consists of 185 fuel and 64 reflector macroelements, which represent four regular subassemblies homogenized with a control blade. Each macroelement is subdivided into 14 axial layers of 30.48 cm height. Cold water injection over the whole core is simulated by doubling the inlet water subcooling through an exponential increase with 2.6 s time constant. Final benchmark report (Finneman et al., 1993) provides the results of 8 industrial and national institutions from 5 countries participated in this benchmark. VTT Finland recently published some results by the TRAB-3D code (Kaloinen, Kyrki-Rajamaki, and Raty, 1999). In a comparison with the PWR benchmarks, the deviations of the results computed by different participants are large: about 600 pcm for the steady-state eigenvalue, and almost 200 % in the power peak. Oversimplified representation of the reactor core, large discrepancies in the results computed by the benchmark participants and an absence of the reference solution make this problem not very useful for the TRAC-BF1/SKETCH-N code verification. However, because there is no any other benchmarks for the coupled neutronics/thermalhydraulics BWR codes we decided to perform the TRAC-BF1/SKETCH-N calculation of this problem.

6.1 SKETCH-N Model

The neutronics model has two neutron energy groups and six groups of delayed neutron precursors. Reactor core is specified at the beginner of cycle 1. The macro cross sections are given as polynomial functions of water density and Doppler temperature. Criticality of the initial condition is set by dividing an average number of the neutrons per fission by the steady-state eigenvalue. The SKETCH-N calculations were performed using the spatial mesh 17x17x26 with 1 node per macroelement (npm) and 26 axial layers: top and bottom axial reflector layers are 30.48 cm height and 24 reactor core layers of 15.24 cm. An automatic time step control procedure has been used; a temporal truncation error tolerance was set to 5×10^{-3}.

6.2 TRAC-BF1 Model

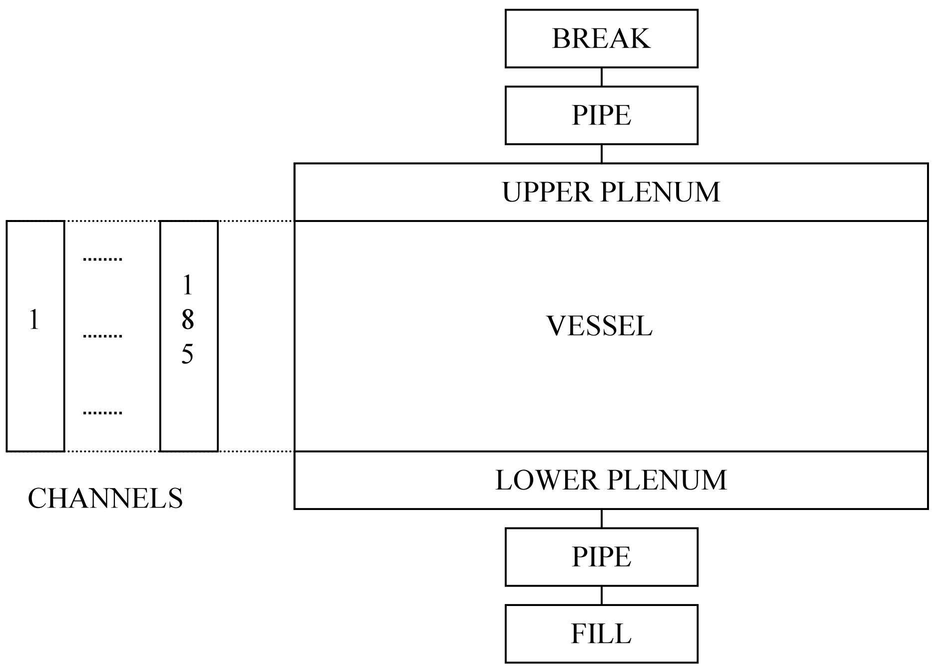


Figure 5. TRAC-BF1 Model for the BWR Cold Water Injection Benchmark.

The TRAC-BF1 reactor model is illustrated in Fig. 5. The VESSEL component with 185 CHANNELS is used to simulate the reactor core. Channel's axial mesh has 24 layers, which coincides with neutronics axial mesh in the reactor core. The radial heat conduction mesh for fuel rods has 6 zones in the fuel, 1 in the gap and 2 in the cladding. The reactor boundary conditions are given using the FILL component on the bottom, which defines the inlet coolant properties and the BREAK component on the top, which specify the reactor pressure of 6.7 MPa. The inlet water subcooling perturbation is simulated as the table of the FILL component by defining the inlet coolant temperature, velocity and pressure. The material properties defined in the benchmark are implemented as an option in the TRAC-BF1 code.

6.3 Numerical Results

Table 3 presents a summary of steady-state results computed by the TRAC-BF1/SKETCH-N code. The eigenvalue computed by TRAC-BF1/SKETCH is equal to 0.98439 and by 240 pcm below than the TRAB-3D value of 0.9868 (Kaloinen, Kyrki-Rajamaki, and Raty, 1999) and by 150 pcm below than the mean of 0.9859 of the benchmark participant results (Finneman et al., 1993).

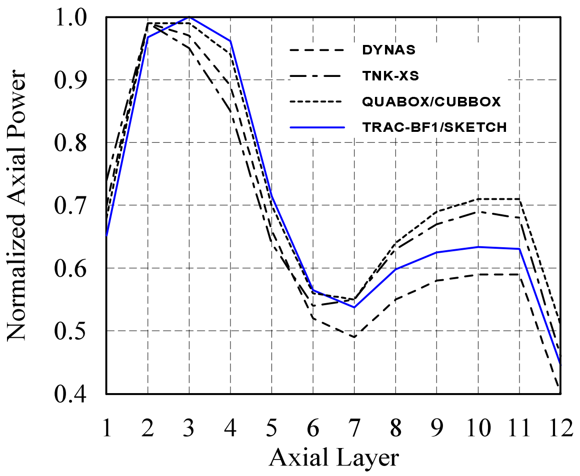
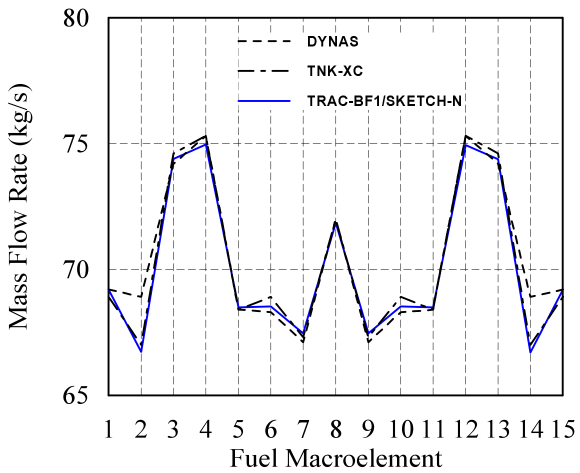


Figure 6. NEACRP BWR CWI Benchmark. Steady-State Normalized Axial Power Distribution.

Parameter	TRAC-BF1/ SKETCH
Eigenvalue	0.98439
1D power peaking factor	1.444
2D power peaking factor	1.955
3D power peaking factor	3.312
Averaged Doppler fuel temperature (°C)	499
Maximum Doppler fuel temperature (°C)	1105
Maximum centerline fuel temperature (°C)	2112
Average channel outlet coolant density (kg/m^3)	453
Average channel outlet coolant temperature (°C)	282.1

Table 3. TRAC-BF1/SKETCH-N Steady-State Results of the NEACRP BWR CWI Benchmark.



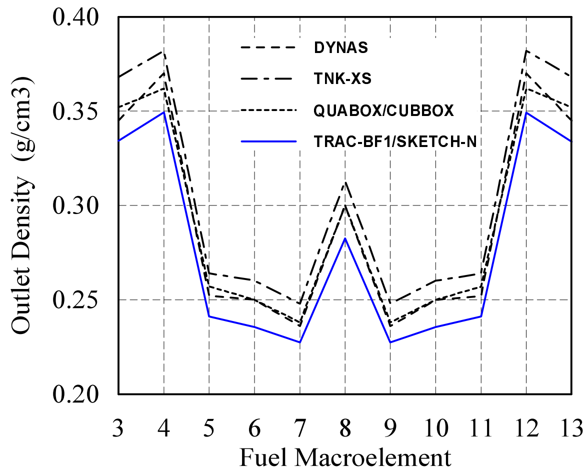


Figure 7. NEACRP BWR CWI Benchmark. Steady-State Mass Flow Rate and Outlet Density along the Vertical Traverse.

Fig. 6 shows the axial power distribution computed by the TRAC-BF1/SKETCH and several benchmark participant codes: DYNAS (NFI, Japan), TNK-XC (Siemens, USA) and QUABOX/CUBBOX (GRS, Germany). Fig. 7 compares the channel mass flow rate and outlet coolant density along the vertical traverse of the reactor core. TRAC-BF1/SKETCH gives lower values of the outlet density than the other codes.

Table 4 gives a summary of the TRAC-BF1/SKETCH transient results. Fig. 8 shows a comparison of the reactor power computed by SKETCH/TRAC-BF1 code system with the results of the TRAB-3D code (Kaloinen et al., 1998). A delay in the reactor power increase is attributed to the TRAC-BF1 reactor model. In the TRAC model, cold water is injected using the FILL component, as a result there is small time delay because the coolant has to pass through the inlet PIPE and LOWER PLENUM before reaching the CHANNEL inlet.

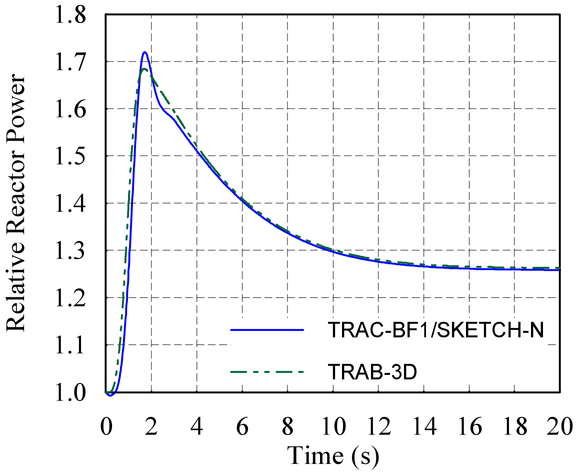


Figure 8. NEACRP BWR CWI Benchmark. Relative Reactor Power versus Time.

Parameter	TRAC-BF1/ SKETCH
Maximum fission power (%)	171.9
Time of maximum fission power (s)	1.69
3D power peaking factor at power peak	3.695
Reactor power at the time 20 s (%)	125.9
3D power peaking factor at the time 20 s	3.699
Difference in average Doppler fuel temp. at time 20 s vs. steady-state value (Δ°C)	64
Difference in maximum Doppler fuel temp. at time 20 s vs. steady-state value (Δ°C)	327
Difference in maximum centerline fuel temp. at time 20 s vs. steady-state (Δ°C)	793
Difference in outlet coolant temperature at time 20 s vs. steady-state value (Δ°C)	-2.3
Difference in outlet coolant density at time 20 s vs. steady-state value (Δkg/m^3)	20.7

Table 4. TRAC-BF1/SKETCH-N Transient Results of the BWR Cold Water Injection Benchmark

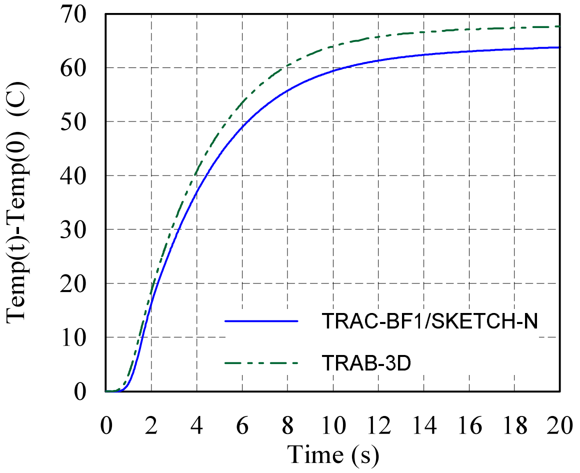


Figure 9. NEACRP BWR CWI Benchmark. Core-Averaged Doppler Fuel Temperature Difference with respect to the Steady-State Value.

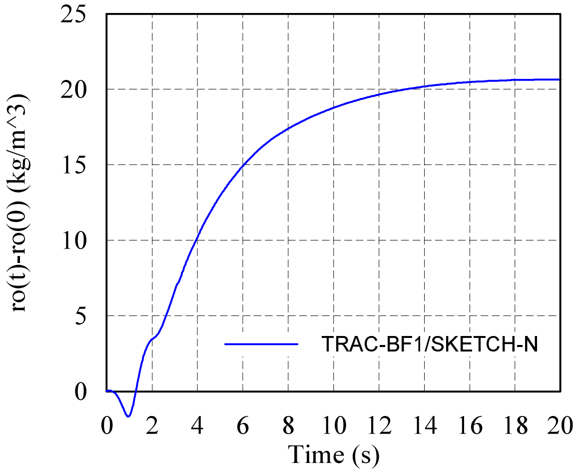


Figure 10. NEACRP BWR CWI Benchmark. Core-Averaged Channel Outlet Density Difference with respect to Steady-State Value.

Core-averaged Doppler fuel temperature computed by SKETCH/TRAC is compared with the TRAB-3D code in Fig. 9. The difference in average outlet channel density with respect to the steady-state value is given in Fig. 10. For compared global reactor parameters there is a close agreement of the TRAC/SKETCH results with the TRAB-3D results. The TRAC/SKETCH results are also close to the mean results of the BWR benchmark participants given in (Finneman et al., 1993). There are no local transient parameters available for a comparison for this benchmark. More detail analysis of this benchmark and a comparison with the TRAB-3D code will be given in the forthcoming paper (Zimin et. al., 2000).

7. EXPERIMENTAL ACTIVITIES

In order to assess and improve the TRAC-BF1 code, out-of-pile experiments on BWR stability are under way using two experimental facilities. A one of the two facilities, called Two Phase flow Test Facility (TPTF), has single channel core with a simulated natural circulation loop. The single-channel tests are focused on core-wide stability and density wave propagation phenomena. Another facility, called parallel-channel facility, has 3-pararel channels for out-of-phase stability tests with bypass channel. In the both the facilities, a core power control system, composed of a computer system furnished with a reactor kinetics model, was installed to provide the capability to simulate coupled neutronics thermal-hydraulics behaviour in a BWR core. This system controls the electric core power in real-time as a function of the measured core void fraction, fluid temperature and fuel rod temperature.

7.1 TPTF Experiment

The TPTF consists of a pressure vessel, core test section, power control system, inlet flow control system and natural circulation loop as shown in Fig. 11. The pressure vessel, 0.079m inner diameter, 6.2m long, includes the core test section with 24 electrically-heated rods and 8 non-heated rods, simulating a full-length (3.7m) PWR 17x17 type fuel bundle. The rods with an outer diameter of 9.5mm are arranged at 12.6mm pitch and supported by spacers at seven elevations. The test section is equipped with various two-phase flow instruments to measure temperatures, pressures and void fractions. Local differential pressures are measured at 9 locations in the core heated length of 3.7m. The core power supply system has a capability to provide forced oscillations of the total core power with radially and axially uniform power profile.

The density wave propagation test is being conducted by providing various core inlet flow rates under forced core power oscillation. The specific tests described herein were conducted at a core power oscillation (600KW±150kW) with a period of 4s, for core inlet mass fluxes ranging 120 to 450kg/m^{2}s under system pressure of 7MPa.

The experimental analysis was done with TRAC-BF1 alone using the measured thermal-hydraulic conditions as boundary conditions. The core test section of 4.1m between lower and upper plena was modelled with 41 volumes. The boundary conditions were imposed in terms of measured liquid velocity and temperature (subcooled) for the lower plenum, and measured pressure for the upper plenum. The natural circulation loop was excluded from the model. Most of the prediction results were reasonably well for the various core inlet mass fluxes. Figure 12 shows measured and predicted local core differential pressures at location near the core outlet (labelled DP8 in this figure) of the test section for inlet mass fluxes of 300 and 450□/m^{2}s, respectively. The differential pressures oscillated according to core power oscillations in both the experiment and prediction. The differential pressure first increased as core power started to increase from 600kW, and peaked when core power became maximum value of 750kW. In the most predictions, differential pressure responses were characterized by a continuous oscillation during the forced core power oscillation under the constant core inlet mass flux. However, it was found that the differential pressure behaviour was in discontinuity or irregularity at upper part of core only when the core inlet mass flux was nearly 300kg/m^{2}s, whereas it was not observed in the experiment as shown in Fig.12. The authors are now putting an effort to solve this problem.

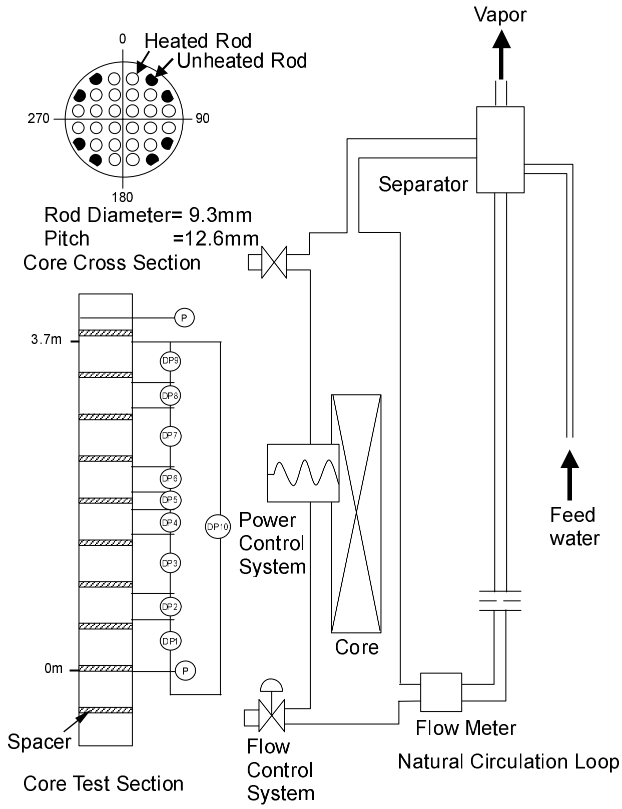


Figure 11. Two Phase Flow Test Facility.

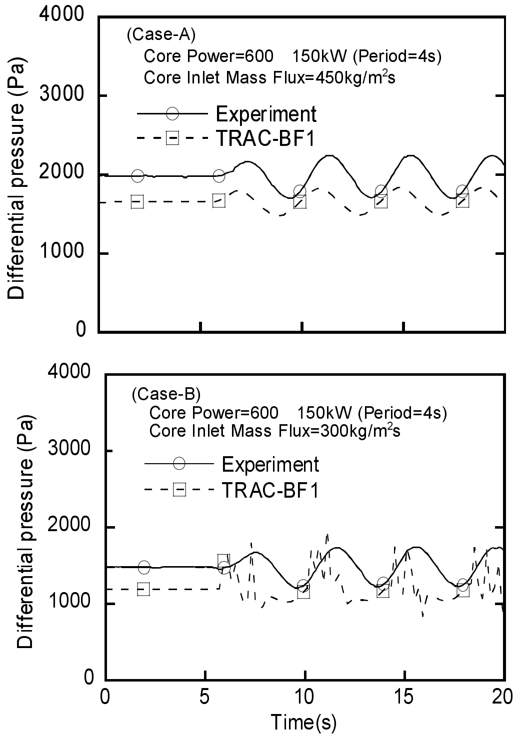


Figure 12. Measured and Predicted Local Differential Pressure (DP8).

7.2 Parallel-Channel Experiment

JAERI is performing experimental simulation of the nuclear and thermal-hydraulic coupled instabilities with a parallel channel test facility. The test facility is shown in Fig. 13. It consists of three channel test sections (2x2 bundle test sections) and one 4x4 bundle test section. Three electrically heated rods and one unheated rod are installed in each 2x2 bundle test section. The diameter and heated length of the heater rods are the same as those of nuclear fuels of conventional BWRs.

Instantaneous area-averaged void fraction in the test section is measured with impedance-type void meters. Electrical power of heater rods is calculated with point neutron kinetics by using measured void fraction. Thus, the nuclear and thermal-hydraulic coupling is simulated.

Experiment is undergoing. Following experiments were performed.

(1) Tests for channel stability
- Single channel, Two channels, or three channels
- Pressure=1-7MPa
- Mass flux=130-660kg/m^{2}s
- Kout/Kin=0.1-40

Kout/Kin is flow resistance ratio between outlet and inlet of test section, as shown in Fig.13. Figure 14 shows stability map of single-channel test under various Kout/Kin. Stability threshold increases with decrease of Kout/Kin. In the figure, boiling transition threshold is also shown. The boiling transition threshold also increases with decrease of Kout/Kin.

(2) Test for nuclear and thermal-hydraulic coupling
- Single channel
- Pressure=7MPa

- Mass flux=400 to 530kg/m^{2}s (step response)
- Kout/Kin=0.1-40

Figure 15 shows a test result. Initial mass flow rate of the test was 0.3kg/s. Then, mass flow rate was changed to 0.4kg/s and then to 0.3kg/s. Figure on the left side shows the result of a test without void feedback simulation. The increase of mass flow rate results in decrease of void fraction. The decrease of mass flow rate results in increase of void fraction. Both of mass flow rate and void fraction are stable. On the other hand, a figure on the right side shows the result of a test with void feedback simulation. The increase of mass flow rate results in decrease of void fraction, and then increase of power. Some oscillation of void fraction and power take place and continue. However, the oscillation decays in about 10 seconds in this test. Finally, void fraction is lower than initial value, and power is higher than initial value. Thus, increase of mass flow rate results in increase of power due to void feedback effect, and void fraction is higher than that without void feedback simulation.

According to other experiments, void fraction, mass flow rate, and power are more oscillatory under higher Kout/Kin. Thus, experimental result is much dependent on Kout/Kin.

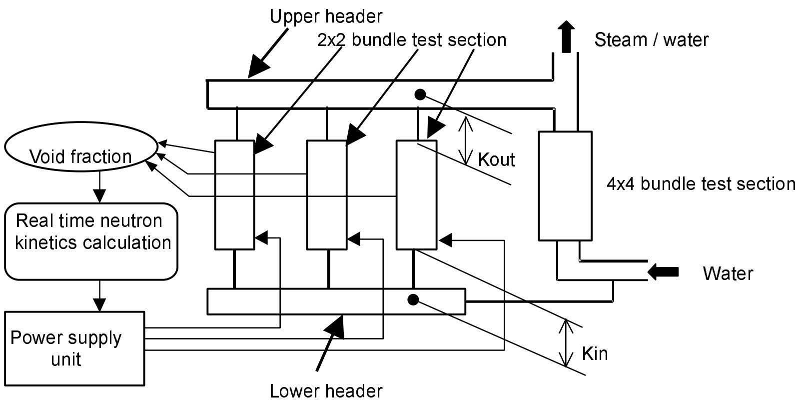


Figure 13. JAERI 3-channel test facility for nuclear and thermal-hydraulic coupling test.

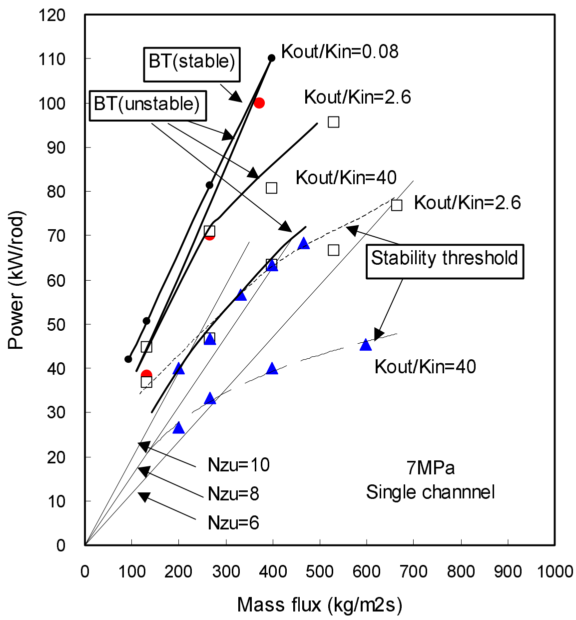


Figure 14. Stability map under single channel.

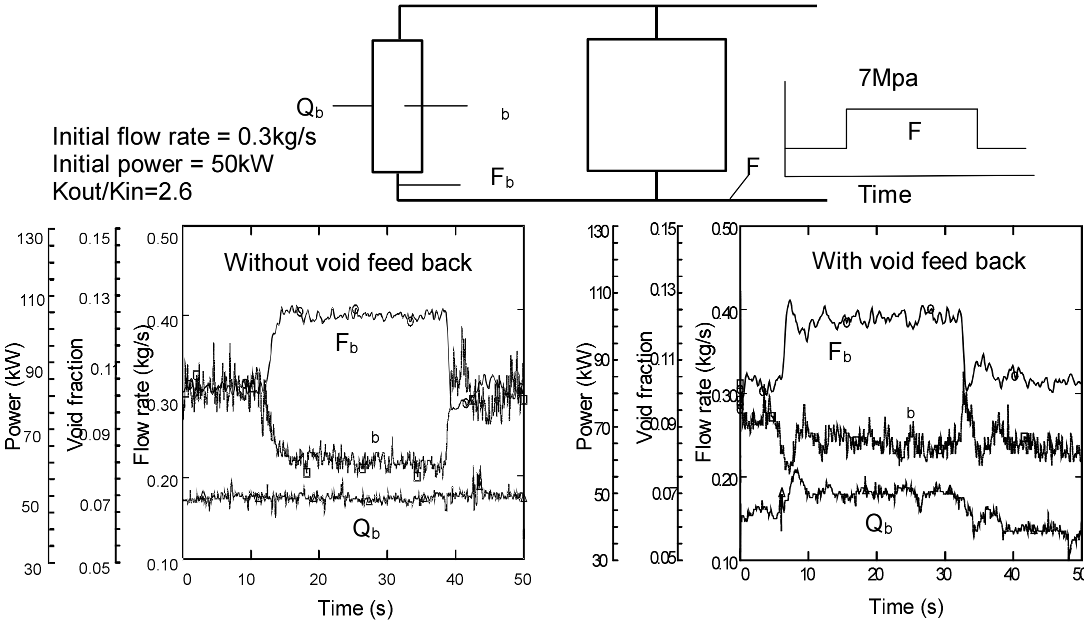


Figure 15. Nuclear and thermal-hydraulic test under step increase and decrease of flow rate.

8. CONCLUSIONS

3D neutron kinetics model SKETCH-N has been implemented into the transient analysis codes J-TRAC and TRAC-BF1. J-TRAC/SKETCH-N coupling has been verified using PWR benchmarks: NEACRP rod ejection benchmark and NEA/NSC uncontrolled rod withdrawal at zero power. A comparison with the PANTHER reference solution demonstrates acceptable accuracy of the J-TRAC/SKETCH-N code system for reactivity-initiated accidents. Verification of the TRAC-BF1/SKETCH-N coupling has been done by the BWR cold water injection benchmark. TRAC-BF1/SKETCH-N results are in a good agreement with the mean results of the benchmark participants and with the TRAB-3D results, which published recently. An absence of the reference solution for this problem and large discrepancies in the participant results prevent us from further conclusions. Presently Ringhals-1 stability benchmark is on the way for the TRAC-BF1/SKETCH code system assessment.

A progress has been made on the out-of-pile tests regarding BWR stability. Further TRAC-BF1 code verifications will be made through the single- and parallel-channel tests conducted at JAERI.

REFERENCES

Akimoto, H., Ohnuki, A., Kikuta, M., et al, 1988. Assessment of J-TRAC code with CCTF/SCTF test data. *Proc. of the Sixteen Water Reactor Safety Research Information Meeting*, October 24-27, 1988, NUREG/CP-0097, vol.4, pp.583-606.

Akimoto, H., et al. 18th report on the best-estimate code development. J-TRAC user manual (Version 62.1), JAERI-Memo-63-478, January 1989 *[In Japanese]*.

Bandini, B.R., Ivanov, K., Baratta, A., Steinke, R. G., 1998. A verification of a three-dimensional transient neutronics routine for the TRAC-PF1/MOD3 thermal-hydraulics system analysis code, *Nucl. Technol.,* **123**, 1-20.

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. 4, August 1992.

Diamond, D. Multi-dimensional reactor kinetics modelling, *Proc. of the OECD/CSNI Workshop on Transient Thermal-Hydraulics and Neutronics Codes Requirements, Annapolis, Maryland, USA, November 5-8, 1996,* NUREG/CP-0159, NEA/CSNI/R(97)4, pp. 495-505.

Finnemann, H., Galati, A. NEACRP 3-D LWR core transient benchmark. Final specifications, NEACRP-L-335 (Revision 1), January 1992.

Finnemann, H., Bauer, H., Galati, A. and Martinelli, R. Results of LWR core transient benchmarks, NEA/NSC/DOC(93)25, October 1993.
Fraikin, R., and H. Finnemann. NEA-NSC 3-D/1-D PWR core transient benchmark. Uncontrolled withdrawal of control rods a zero power. Final Specifications. NEA/NSC/ (93) 9, September 1993.
Fraikin, R. PWR benchmark on uncontrolled rods withdrawal at zero power. Final report. NEA/NSC/DOC(96)20, September 1997.
Geist, Al, et al. PVM 3 user's guide and reference manual, ORNL/TM-12187, September 1994.
Goose, G., Shatford, J.G., Agee, L.J., 1998. The RETRAN-3D code; pressurized water reactor multidimensional kinetics applications. *Nucl. Technol.,* **122**, 132-145.
Ivanov K. N., T. M. Beam, A. J. Barratta et. al. Pressurised water reactor main steam line break (MSLB) benchmark. Volume I: Final specifications. NEA/NSC/DOC(99)8, April 1999.
Kaloinen, E., R. Kyrki-Rajamaki, H. Raty "Calculations of BWR transients with 3D dynamics code TRAB-3D', *Proc. of the Int. Conf. on the Physics of Nuclear Science and Technology(PHYSOR),* October 5-8, 1998, Long Island, New York, American Nuclear Society, vol. 2, pp. 1444-1451.
Knight, M.P., Bryce, P., 1997. Derivation of a refined PANTHER solution to the NEACRP PWR rod-ejection transients. *Proc. of the Joint Int. Conf. on Mathematical Methods and Supercomputing for Nuclear Applications*, Saratoga Springs, N.Y., October 5-9, 1997, American Nuclear Society, vol. 1, pp. 302-313.
Judd, J.L., Weaver, W.L, Downar, T., Joo, H.G. A three-dimensional nodal neutron capabilities for RELAP5. *Proc. of the 1994 Topical Meeting on Advances in Reactor Physics*, April 11-15, 1994, Knoxville, Tennessee, American Nuclear Society, vol. II, pp. 269-280.
Langenbuch, S., Lizorkin, M., Rohde, M., and Velkov, K., 1996. 3D neutronics codes coupled with thermal-hydraulic system codes for PWR, BWR and VVER reactors, *Proc. of the OECD/SCNI Workshop on Transient Thermal Hydraulic and Neuronic Codes Requirements*, Annapolis, Maryland, November 5-8, 1996, US Nuclear Regulatory Commission, NUREG/CP-0159, NEA/CSNI/R(97)4, pp. 506-524.
Lefvert, T. BWR stability benchmark. Final specification. NEA/NSC/DOC(94)15, March 1994.
Liles, D.R., et al. TRAC-PF1/MOD1 correlations and models, NUREG/CR-5069, December 1988.
Sahota, M. S., Addessio, F. L. TRAC-PF1/MOD1 developmental assessment. Los Alamos National Laboratory; U.S. Nuclear Regulatory Commission, TI86-001251; LA-10445-MS ; NUREG/CR-4278, 1985.
Zimin, V.G., 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., Pogosbekyan L., 1998. Polynomial and semi-analytic nodal methods for nonlinear iteration procedure. *Proc. of the Int. Conf. on the Physics of Nuclear Science and Technology (PHYSOR)*, October 5-8, 1998, Long Island, New York, American Nuclear Society, vol. 2, pp. 994-1002.
Zimin, V. G., H. Asaka, Y. Anoda and M. Enomoto, 1999. Verification of the J-TRAC code with 3D Neutron Kinetics Model SKETCH-N for PWR Rod Ejection Analysis. *Proc. of the 9 International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH 9),* San Francisco, California, October 3-8, 1999.
Zimin, V. G., H. Asaka, Y. Anoda, E. Kaloinen, R. Kyrki-Rajamaki Analysis of NEACRP 3D BWR Core Transient Benchmark, *Proc.of the Intl. Conf. on Supercomputing in Nuclear Application, SNA 2000,* September 4-7, 2000, Tokyo, Japan [In preparation].