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TRANSIENT AND FUEL PERFORMANCE ANALYSIS WITH VTT'S COUPLED CODE SYSTEM

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ABSTRACT

VTT maintains and further developes a comprehensive safety analysis code system ranging from the basic neutronic libraries to 3D transient analysis and fuel behaviour analysis codes. Some of the computer codes have been developed at VTT, others have been acquired from internationally recognized sources. The code system is based on various types of couplings between the relevant physival phenomena.

The main tools for analyses of reactor transients are presently the 3D reactor dynamics code HEXTRAN for cores with a hexagonal fuel assembly geometry and TRAB-3D for cores with a quadratic fuel assembly geometry. HEXTRAN has been applied to safety analyses of VVER-type reactors since early 1990's. TRAB-3D is the latest addition to the code system, and has been applied to BWR and PWR analyses in recent years.

The results from the 3D analysis can be used as boundary conditions for more detailed fuel rod analysis. For this purpose a general flow model GENFLO, developed at VTT, has been coupled with USNRC's FRAPTRAN fuel accident behaviour model. FRAPTRAN/GENFLO is quite a promising tool that can handle flow situations that are far too complicated for the models that are usually seen with fuel codes.

KEYWORDS: LWR, transients, coupling

1. INTRODUCTION

VTT started developing coupled codes as early as 1980 with the 1D coupled neutronics/thermal hydraulics BWR code TRAB [1], which was later further coupled to the system code SMABRE [2]. The 3D HEXTRAN [3] core dynamics code was coupled to SMABRE in 1991 and has been extensively used since. Finally, the square lattice stand-alone BWR dynamics code TRAB-3D [4] was coupled to SMABRE in 1997. The most recent coupled code system is FRAPTRAN-GENFLO [5], in which NRC's FRAPTRAN [6] code and VTT's general hydraulics model GENFLO [7] were coupled to a powerful tool for analyses of fuel behaviour during reactor transients.

Presently, VTT maintains and further developes a comprehensive safety analysis code system ranging from the basic neutronic libraries to 3D transient analysis and fuel behaviour analysis codes, as shown in Fig. 1. Some of the computer codes have been developed at VTT, others have been acquired from internationally recognized sources. The code system is based on various types of couplings between the relevant physical phenomena.

This paper describes firstly the coupled 3D neutronics/thermal hydraulics codes TRAB-3D and HEXTRAN, followed by an example load rejection test case, where a real BWR plant data was used for the code validation. In the second part of this paper, the fuel analysis code FRAPTRAN coupling with a thermal hydraulics model GENFLO is discussed together with an example case.

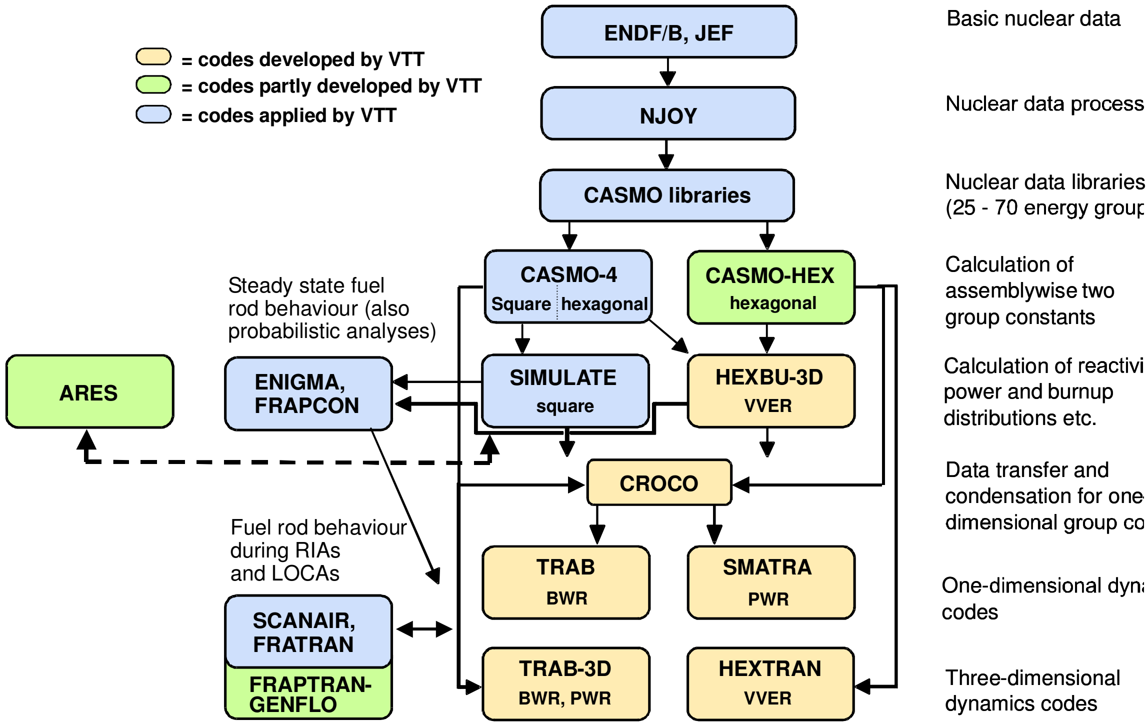

Figure 1. VTT code system for reactor analysis.

2. TRAB-3D AND HEXTRAN CODES

The main tools for analyses of reactor transients at VTT are presently the 3D reactor dynamics code HEXTRAN for cores with a hexagonal fuel assembly geometry and TRAB-3D for cores with a quadratic fuel assembly geometry. HEXTRAN has been applied to safety analyses of VVER-type reactors since early 1990's. TRAB-3D is the latest addition to the code system, and has been applied to BWR and PWR analyses in recent years. The nodal model of both HEXTRAN and TRAB-3D is based on the solution method of the steady-state hexagonal core simulator HEXBU-3D [8] with the solution method extended to quadratic geometry in TRAB-3D.

HEXTRAN and TRAB-3D are dynamics codes with 3D neutronics coupled to parallel 1D channel hydraulics. The two-group neutron diffusion equations are solved by a nodal expansion method in x-y-z geometry within the reactor core. A basic feature of the method is decoupling of the two-group equations into separate equations for two spatial modes and construction of group fluxes from characteristic solutions to these equations. The two solutions are called the fundamental or asymptotic mode, which has a fairly smooth behavior within a homogenized node, and the transient mode, which has a large and negative buckling in LWR reactors. The transient mode deviates significantly from zero only near material discontinuities.

The nodal equations are solved with a two-level iteration scheme where only one unknown per node, the average of fundamental mode, is determined in inner iterations. The nodal flux shapes are improved in outer iterations by recalculation of the coupling coefficients. Cross sections are computed from polynomial fittings to fuel and coolant temperature, coolant density and soluble boron density.

The thermal-hydraulic calculation of the reactor core is performed in parallel one-dimensional hydraulic channels that are usually coupled with one fuel assembly each. Channel hydraulics is based on conservation equations for steam and water mass, total enthalpy and total momentum, and on a selection of optional correlations describing e.g. non-equilibrium evaporation and condensation, slip, and one and two-phase friction. The phase velocities are related by an algebraic slip ratio or by the drift flux formalism. The thermal hydraulic solution methods are the same as in the one-dimensional code TRAB. During the hydraulics iterations a one-dimensional heat transfer calculation is made for an average fuel rod of each assembly. Fuel and cladding are discretised with several radial mesh points in the calculations, which are repeated at different axial elevations. The heat conduction equation is solved according to Fourier's law. Thermal properties of fuel pellet, gas gap and fuel cladding are functions of local temperature and burnup, and the heat transfer coefficient from cladding to coolant depends on the hydraulic regime. The fission power is divided into prompt and decay power parts, and part of the power can be dissipated into heat directly in the coolant.

The HEXTRAN/TRAB-3D core model can be used separately for VVER, PWR or BWR calculations. This feature enables the coupling of the codes with the external thermal hydraulic model SMABRE. SMABRE is a fast running thermal hydraulic system code with point kinetics. It was originally designed for analyses of small breaks in pressurised water reactors, but has had versatile applications in e.g. training simulators or coupled with one- and three-dimensional reactor dynamics core models. The code includes basic models for simulation of thermal hydraulics of light water reactor components: pipes, pumps, valves, ECC accumulators, water separators and steam dryers.

The traditional coupling concept at VTT is called the parallel coupling, the principles of which are illustrated in Fig. 2. The two coupled codes are running independently with a minimum amount of data transfer between the modules. The thermal hydraulics of the core is calculated with both codes in parallel, but the rest of the circulation system is solved only with SMABRE. The connection is carried out by data exchange once in a time-step for core inlet flow and outlet pressure into the direction of the 3D core, and core power distribution into the SMABRE direction. The first application realised with this principle was for the 1D neutronics in 1988 and for the 3D neutronics in 1991 – 1992.

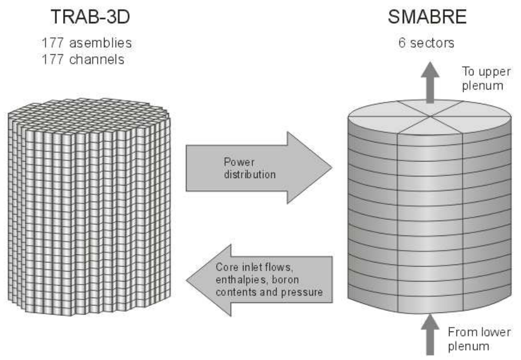

Figure 2. The parallel core coupling principle

TRAB-3D includes also models for the rest of the BWR pressure vessel thermal hydraulics using 1D channels, as well as models for steam lines, pumps and control systems. Thus, it can be used as a stand-alone dynamics code for BWR analysis. A schematic picture of the circuit components can be seen in Fig. 3.

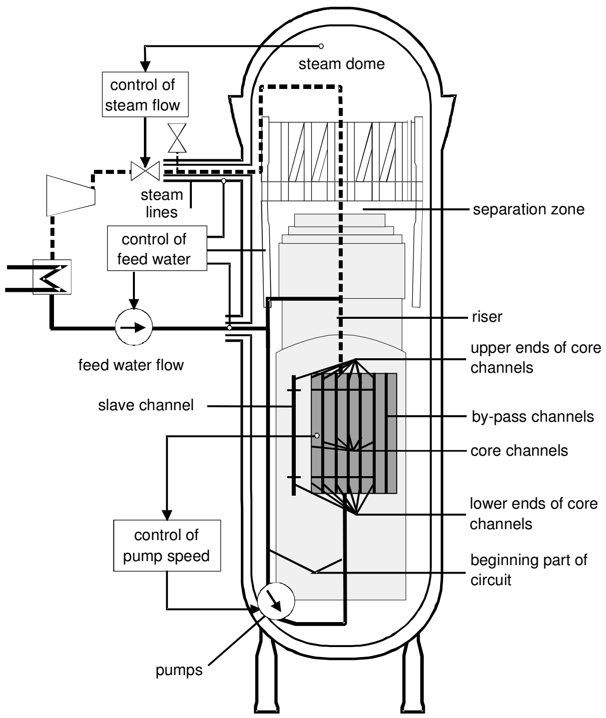

Figure 3. TRAB-3D BWR pressure vessel model

Recently a second method of TRAB-3D/SMABRE coupling has been developed. The intention has been to remove some limitations of the thermal hydraulics model, mainly to allow reversed flow in the core channels, and to be able to include transverse phenomena in the hydraulics. In this internal coupling mode SMABRE calculates also the detailed thermal hydraulics of the core, and provides the feedback parameters for neutronics. The interface between modules is realized inside the neutronic solution itself, and the SMABRE core hydraulics solution is performed inside the TRAB-3D iteration procedure. The coupling is rather complex, and required a lot of modifications into the internal solution procedures of both codes.

Both HEXTRAN and TRAB-3D are based on already validated and much used models. The validation of HEXTRAN consists of dynamics benchmark calculations against other codes, as well as comparisons with plant measurements from Loviisa VVER-440, and other VVER-440 and VVER-1000 plants [3,9,10].

The validation of TRAB-3D was started with the calculation of the OECD/NEACRP PWR and BWR core benchmarks [11]. The international benchmark activity and code-to-code comparisons were continued with the calculation of the OECD PWR main steam line break benchmark, in which TRAB-3D was for the first time coupled to the PWR circuit models of the SMABRE code [12]. VTT participated also in the calculation of the separate core exercise of the OECD BWR turbine trip benchmark.

The other line of the validation work has been the comparison of calculated results with plant measurement data. Three such cases, all for the Olkiluoto 1 plant, have been calculated. The first two of them were the pressurization transient in 1985 and the oscillation incident in 1987 [13], the third and most recent being the calculation of the load rejection test at Olkiluoto 1 in 1998.

Both TRAB-3D and HEXTRAN are in active use for safety analyses at VTT.

3. THE OLKILUOTO 1 LOAD REJECTION TEST

The load rejection test of June 16th 1998 was selected from the tests made in connection with the Olkiluoto plant power uprating, because data on local power measurements was available for this test, making the test a good case for validating 3D core models. The purpose of the test was to ensure that the plant is able to shift from full 2500 MW power operation to a 30% power level in a case of external load rejection. The plant should be able to feed the in-house load and dump the excess steam to the condensers. This is managed through a partial scram, where one scram rod group is inserted hydraulically in a few seconds and another one slowly electrically, the latter taking 260 seconds. Simultaneously the main circulation pump speed is decreased to the minimum level. In the test the plant functioned as planned.

The test has been calculated with TRAB-3D to 400 s from the beginning, which makes it, by far, the longest BWR transient calculation with VTT's dynamics codes until now. In the calculation the main interest is in the 3D effects in the core. The coolant circuit model was adopted from earlier Olkiluoto calculations as such. Measured values for the feedwater flow and temperature were used as transient boundary conditions, because during the test several manual operator actions were performed concerning feedwater flow.

The steady-state comparisons between the TRAB-3D power distribution and the one reconstructed from measurements show a maximum deviation of 4% for some individual fuel bundles, while for most bundles the deviation is clearly less, as can be seen in Fig. 4. Thus the calculated steady state is close to the real initial state of the reactor.

Figure 5 shows the calculated local neutron flux behaviour against measured data from local power range monitors (LPRM) for one radial location. The calculated value in the figures is an average of the thermal neutron flux of the four bundles surrounding the detector, there are no models for the actual detector response. The data is normalized under the assumption that the initial steady state values are correct. This assumption can be justified with the steady state power distribution comaprison of Fig. 5.

Average neutron flux behaviour against the data from one of the average power range monitor (APRM) measurement systems is shown in Fig. 6. Both the APRM signal and the calculated value are an average of 28 local detector signals at different radial and axial positions in the core. The comparison of measured and calculated dynamic behaviour of local power show that TRAB-3D calculates accurately the power distribution during the load rejection transient.

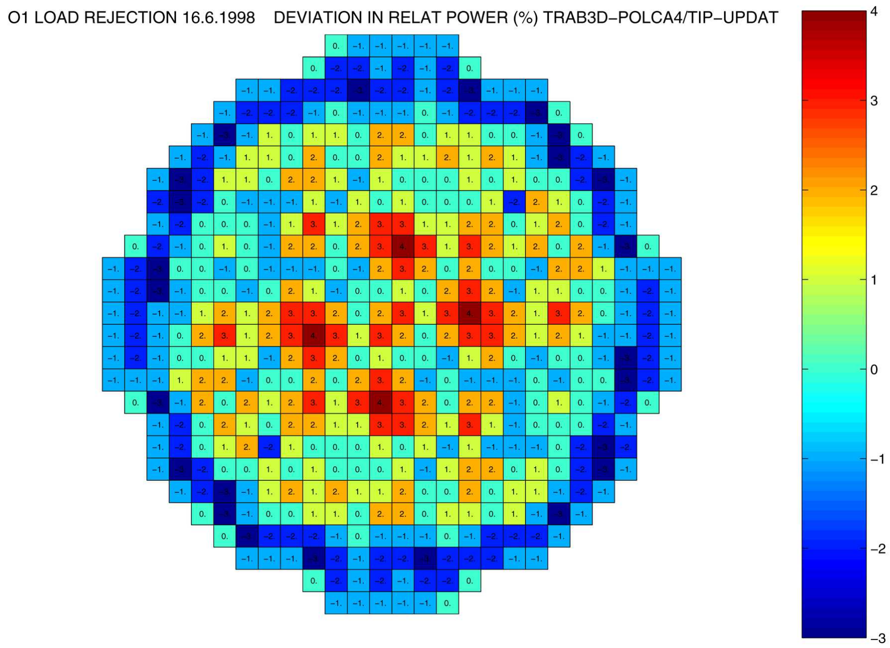

Figure 4. TRAB-3D power distribution deviation (%) from the power distribution reconstructed from measurements

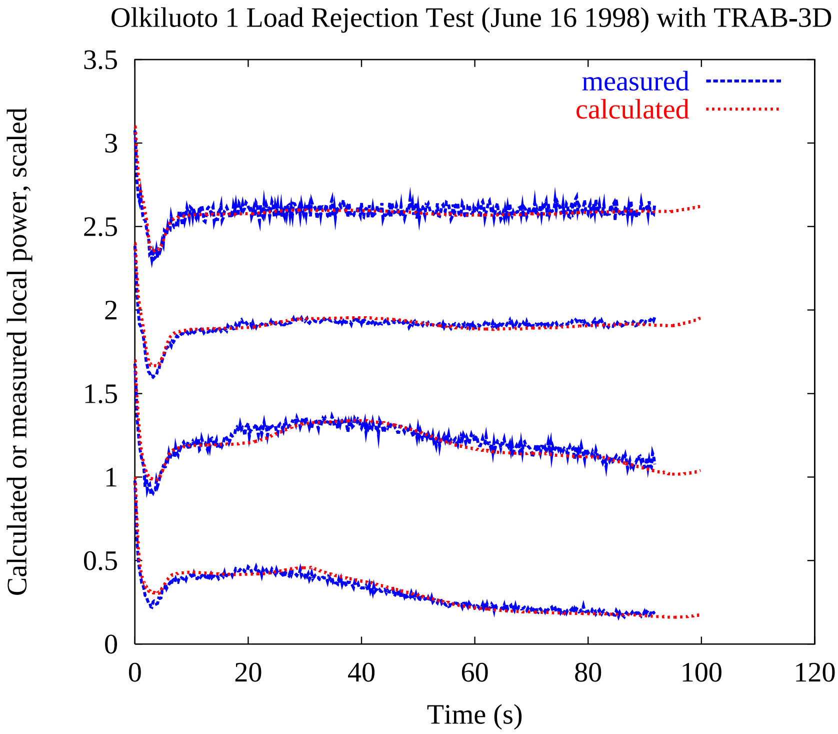

Figure 5. Calculated and measured local power at four axial locations

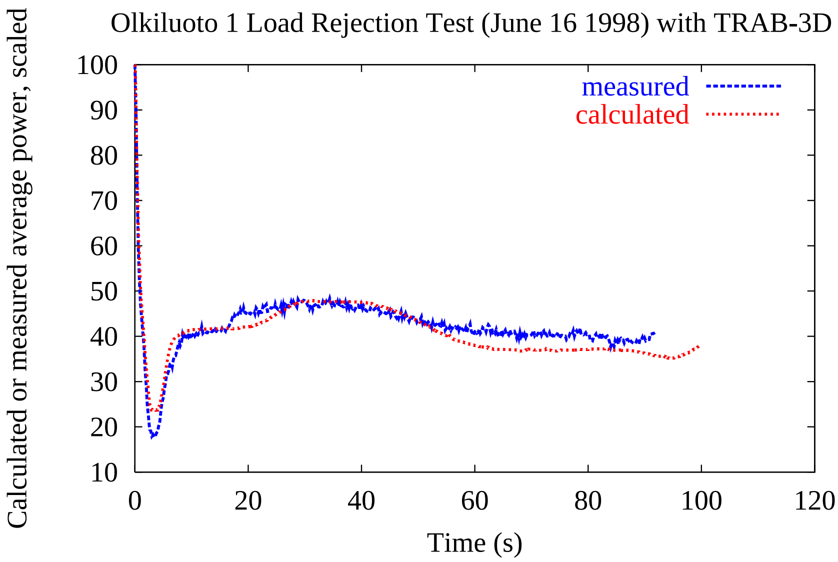

Figure 6. Calculated and measured average power, based on 28 local power measurements at 7 radial locations, each with 4 axial locations

4. THE FRAPTRAN-GENFLO CODE

The results from the 3D analysis can be used as boundary conditions for more detailed fuel rod analysis. For this purpose a general flow model GENFLO, developed at VTT, has been coupled with USNRC's FRAPTRAN fuel accident behaviour model [5].

The thermal hydraulic solution of GENFLO is based on the numerical solution developed for the SMABRE [2] code. In SMABRE the main interest was a reasonable accuracy of the physical models in small break LOCA and large break LOCA blowdown conditions, but for example the core heat transfer characteristics in reflooding conditions, and in high temperature conditions during severe accidents were not emphasized. GENFLO is a fast running five-equation thermohydraulic model, where the wetted wall heat transfer, dryout and post-dryout heat transfer, and quenching models are included.

GENFLO solves the coolant mass, momentum and energy conservation equations, including the calculation of the axial distributions of the fluid temperature and the void fraction. As a result, the fluid temperatures and heat transfer coefficients for each axial level at each time step are supplied for FRAPTRAN, which calculates temperatures and deformation of the fuel pellets and cladding, including possible ballooning. The fuel specific calculations are made for both codes by FRAPTRAN and the coolant specific calculations for both codes by GENFLO.

In the coupled code, FRAPTRAN is the master code calling GENFLO, which provides the thermal hydraulic conditions for the whole channel. This calculation is performed only once for each time step, even if a number of iterations is done in FRAPTRAN during the time step. In the beginning, GENFLO is required to make a steady state calculation before any coupled code calculation. In the coupled code calculation, FRAPTRAN dictates the time step length, typically 0.01 – 0.05 s, but the calculation is fast because GENFLO is non-iterative and effective numerical methods are applied.

The system behaviour and boundary conditions needed for a detailed core simulation may be calculated with various system codes such as RELAP5 or others. At VTT, also the three-dimensional BWR or PWR reactor dynamics codes TRAB-3D and the simulator APROS have been used. The data exchange between a system code, GENFLO, and FRAPTRAN is illustrated in Figure 7. At present, the boundary conditions provided for GENFLO from the system code are the mass flow and the enthalpy at the channel inlet, the pressure at the top of the channel, as well as the total power and the power profile of the fuel rod.

The new combination has been tested against VVER LOCA and BWR power oscillation cases and the difference in results may be occasionally dramatic. One of the future development targets can be modelling the effect of highly deformed (ballooned) fuel in local channel thermal hydraulics. Presently the FRAPTRAN-GENFLO code is also used for the pre and post test analyses of LOCA experiments performed at the OECD Halden reactor in Norway.

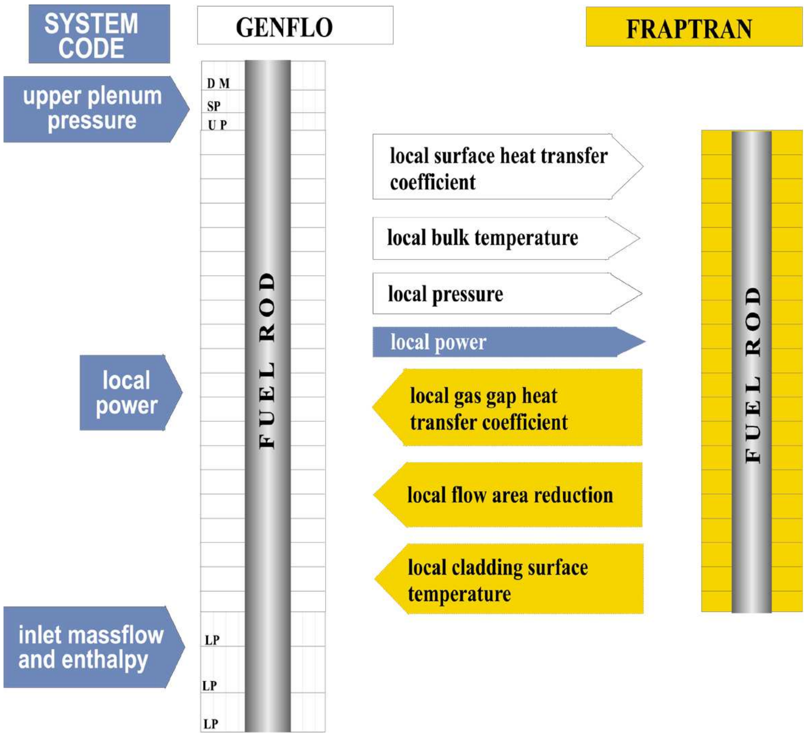

Figure 7. Coupling advanced thermal hydraulics model with a fuel accident performance code

5. BWR OSCILLATION CASE WITH FRAPTRAN-GENFLO

The example case for FRAPTRAN-GENFLO is for an ATWS at a BWR plant. The basis for the analysis is an oscillation incident in the Finnish Olkiluoto 1 BWR during reactor startup on February 22, 1987. The incident was safely terminated by normal operation of the reactor safety systems.

The incident has been simulated with the Finnish TRAB-3D code [13]. The results of the TRAB-3D calculation agree with measurements and earlier analyses. The oscillation frequency and the phase shift between the inlet and outlet flows in a channel of high relative power show good consistency. So do the out-of-phase oscillation of mass flows between high power channels and the core by-pass channel.

To test the performance of the new model combination, the case was also hypothetically extended assuming no actions of the safety system. The transient was recalculated with TRAB-3D as an ATWS case. The escalating oscillation phase of this calculation was chosen a subject of further studies in this FRAPTRAN-GENFLO analysis. The oscillations of boundary conditions were artificially continued in time and amplified, because the ATWS calculation with TRAB-3D showed a limit-cycle behavior where the oscillation amplitude did not grow beyond a certain limit. The total power in the hot assembly calculations with TRAB was multiplied by a factor of 1.3 for the hot rod analysis with FRAPTRAN-GENFLO. Also, the oscillating power was given a minimum value. Shown in Fig. 8 is the boundary condition of the channel inlet mass flow. The flow rate oscillation finally leads to temporary flow reversals (negative mass flow).

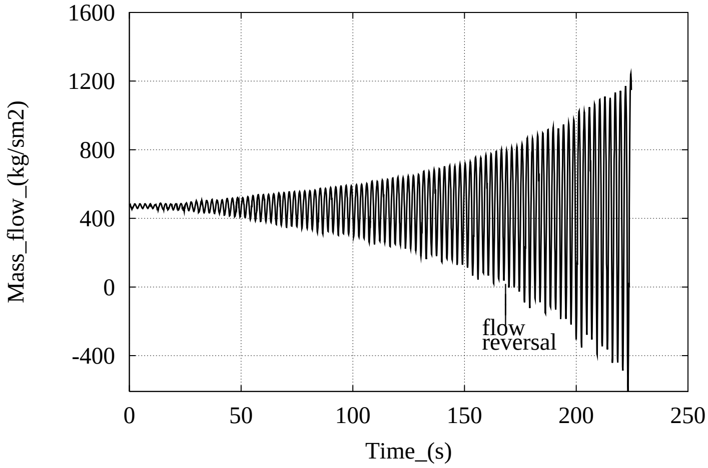

Figure 8. The mass flow rate boundary condition at channel inlet in hypothetical BWR instability case based on the TRAB-3D calculation of a real incident

The results of the FRAPTRAN-GENFLO calculations for this BWR instability case show that with a power cycle period of about 2s the fuel rod remains covered with water until the time of flow reversal. Then the quench front starts dropping in the channel. Before the flow reversal, only local or temporary dryout or DNB conditions may be achieved. The flow reversal soon leads to high cladding temperatures at the upper part of the fuel rod (1682K). The cladding temperature profiles are shown in Fig. 9. At the end of the calculation the cladding is quite soft and in contact with the fuel pellet (PCMI, pellet-to-cladding interaction). The plenum gas pressure remains below the fluid channel pressure during the whole transient, and no rod failure is predicted in spite of the fast deformation of the fuel rod (Figure 10).

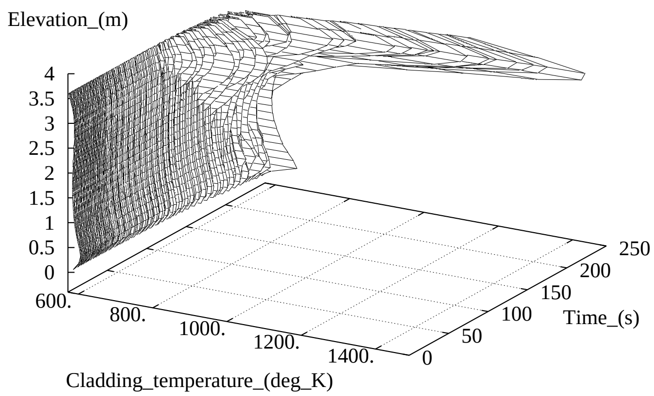

Figure 9. Cladding surface temperature profile in hypothetical BWR instability case

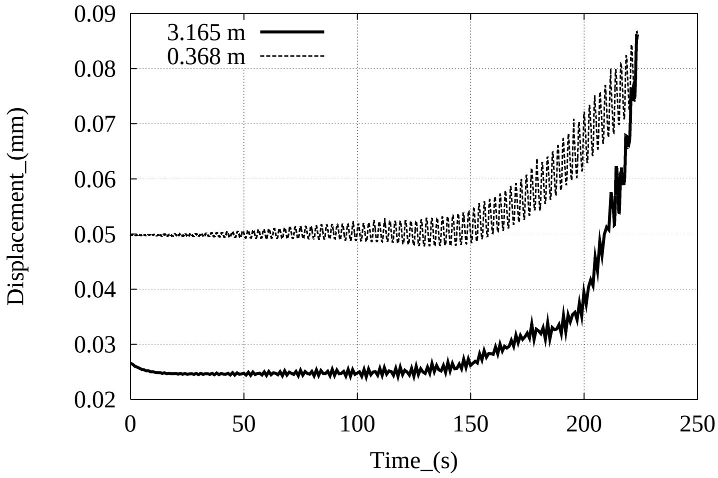

Figure 10. Displacement of cladding outer surface at two axial level in BWR instability case

6. CONCLUSIONS

In the core of a nuclear reactor, there are numerous different physical phenomena continuously interacting with each other. The neutronics, thermal hydraulics and fuel models must all be coupled together in order to analyse nuclear reactor transients. The development in computing capacity during recent years has made the couplings between more and more complicated physical models feasible.

At VTT, there is a long experience of developing and using coupled dynamics codes for scientific exercises, as well as safety analyses of real plants. The validation history of HEXTRAN and TRAB-3D, with and without the coupling to SMABRE, show that the models of these codes can calculate various types of nuclear reactors in different conditions with good accuracy. One validation case, the load rejection test at Olkiluoto 1 plant, is shortly described in this paper as an example. The results of the calculation of the load rejection test indicate the TRAB-3D is capable of calculating correctly the dynamic 3D power distribution. In earlier cases, only steady state power distributions have been compared to measurements.

Recently the development of the nuclear fuel models has been of great interest at VTT and elsewhere, mainly because of the increasing pressure to increase fuel burnup. The fuel models inside TRAB-3D and HEXTRAN are presently under development. Another direction in this field is the coupling of FRAPTRAN and GENFLO in order to improve the thermal hydraulics modeling inside the fuel behavior codes. This development also brings the fuel analysis codes more directly connected to the reactor physics and transient calculation system. The new coupled code is proving to be a proper tool for a detailed analysis of reactor transients, as can be seen with the oscillation example.

ACKNOWLEDGMENTS

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