VALIDATION OF THERMAL HYDRAULIC COMPUTER CODES FOR ADVANCED LIGHT WATER REACTOR

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Abstract

The Czech Republic operates 4 WWER-440 units, two WWER-1000 units are being finalised (one of them is undergoing commissioning). Thermal-hydraulics Department of the Nuclear Research Institute Řež performs accident analyses for these plants using a number of computer codes. To model the primary and secondary circuits behaviour the system codes ATHLET, CATHARE, RELAP, TRAC are applied. Containment and pressure-suppressure system are modelled with RALOC and MELCOR codes, the reactor power calculations (point and space-neutron kinetics) are made with DYN3D, NESTLE and CDF codes (FLUENT, TRIO) are used for some specific problems. An integral part of the current Czech project „New Energy Sources" is selection of a new nuclear source. Within this and the preceding projects financed by the Czech Ministry of Industry and Trade and the EU PHARE, the Department carries and has carried out the systematic validation of thermal-hydraulic and reactor physics computer codes applying data obtained on several experimental facilities as well as the real operational data. The paper provides a concise information on these activities of the NRI and its Thermal-hydraulics Department. A detailed example of the system code validation and the consequent utilisation of the results for a real NPP purposes is included.

1. INTRODUCTION

Until 1989, either home or modified Russian computer codes were used in the NRI for analysis of NPP safety,. In 1989, as first Western code, we obtained RELAP5 different Mods and in 1992, ATHLET and CATHARE, in 1995 — computer code TRAC.

Later on, containment codes MELCOR, DRASYS and RALOC, and advanced CFD computer codes (FLUENT, TRIO).

So, at the beginning, the fairly important part of our work consisted in learning how to use these codes and in modifying them for the hardware available at the NRI. The next stage consisted in the preparation of input data for the NPPs with WWER reactors and QA of input data decks.

Since all presented system codes have been developed and verified for western NPPs, their verification on Czech NPPs was an important stage. This verification was performed mainly on experimental facilities with parameters similar to those of WWER reactors, and on measurements performed during start-up of the NPPs. For this verification served, especially the ATHLET, CATHARE, RELAP, MELCOR and RALOC codes. We also participated in different computations of experiments organized under the auspices of OECD (ISP), EU (PHARE projects, 5th framework projects) and national project (sponsored Ministry of Trade and Industry).

Our activity in the field of Advanced Reactor started by participation in the IAEA project Thermohydraulic Relationships for Advanced Water-Cooled Reactors. This activity, which was focused to determination of CHF, followed former projects taking place in the Czech Republic, both experimental and methodological.

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TABLE I. LIST OF COMPUTER CODES USED FOR WWER 1000 IN NRI

	CODE title	**used by organisation country of origin developed for**	**type of code what is modelled**	**State of utilisation types of processes**	**state of code validation**
1	RELAP5	NRI,USA,PWR	system code primary and secondary circuits	LOCA, LOCA secondary. primary-to-secondary, transients, ATWS SGTR	experimental facility measurements on NPP
2	CATHARE	NRI,France, PWR	system code primary and secondary circuits	Loss of Flow	experimental facility measurements on NPP
3	ATHLET 1.2 A		system code primary and secondary circuits	LBLOCA,SBLOCA, ATWS	experimental facility measurements on NPP
	ATHLET CD	NRI,Germany, PWR, WWER	System code, include core degradation model	LBLOCA	
4	DYNAMIKA-ÚJV	NRI,Russian, modified in Czech Republic WWER	primary and secondary circuits transients	Transients of WWER 440/213	measurement on NPP loss of flow
5	MELCOR	NRI,USA,PWR	primary and secondary circuits,	DBA, BDBA Containment, bubble condenser	against experimental test facility
6	DYN3D	NRI, Germany WWER	reactor, 3D neutron kinetics, simple models of coolant mixing	RIA	against experimental test facility
7	DRASYS RALOC	NRI, Germany PWR, WWER	containment, hermetic boxes	Containment, bubble condenser	against experimental facility
	FLUENT	NRI, US,	general fluid dynamics	Selected local problems	against experimental facility
	TRIO	NRI, France	general fluid dynamics	Starting implementation	

TABLE II. STATE OF ASSESSMENT OF COMPUTER CODES

CODE title	state of code verification for WWER in Czech Republic	type of experimental facility	process
RELAP 5	performed continuously	PMK- Hungary PACTEL-Finland NPP-Dukovany RVS-Řež Russien-NPP ISB-2	Small LOCA primary-secondary natural circulation loss of flow
CATHARE	performed continuously	ISP-38 ISP-42PANDA	Shut-down state
ATHLET 1.2A	performed continuously	NPP PMK- Hungary ISB	Loss of flow SBLOCA
DYN3D	performed continuously	LR0-NRI Řež	RIA, Turbine trip
DRASYS	performed continuously	PHARE 2.13, Bubble condenser qualification	LBLOCA
RALOC	performed continuously	ISP-42 PHARE 2.13, Bubble condenser qualification	LBLOCA
MELCOR	performed continuously	PHARE 2.13, Bubble condenser qualification	LBLOCA
FLUENT	performed continuously	ISP-43	Boron dilution
TRIO			

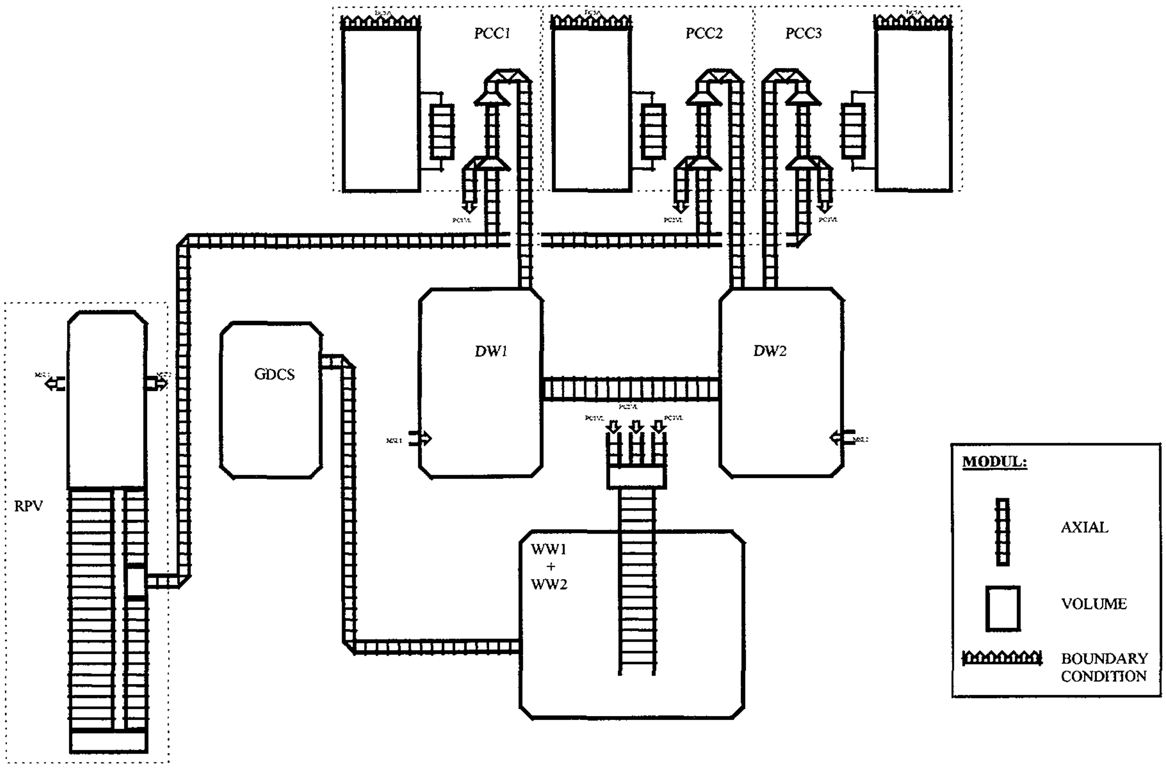

FIG.1. Nodalization of phase A (PANDA — CATHARE nodalization).

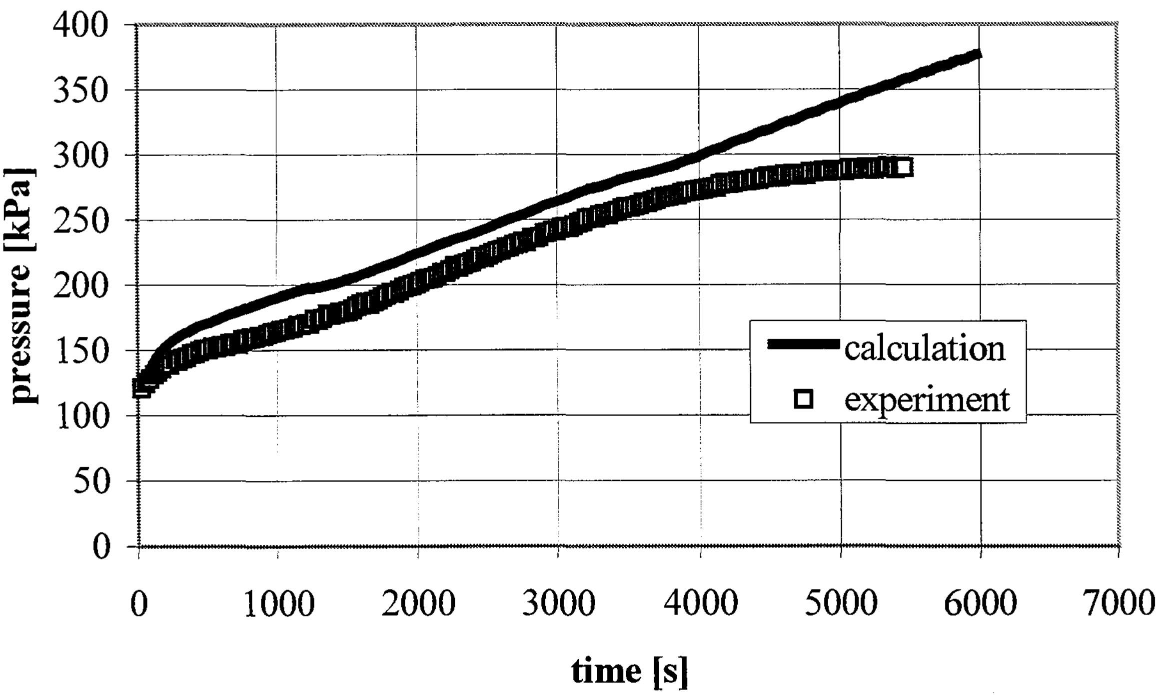

FIG. 2. PANDA — CATHARE reactor pressure, phase A.

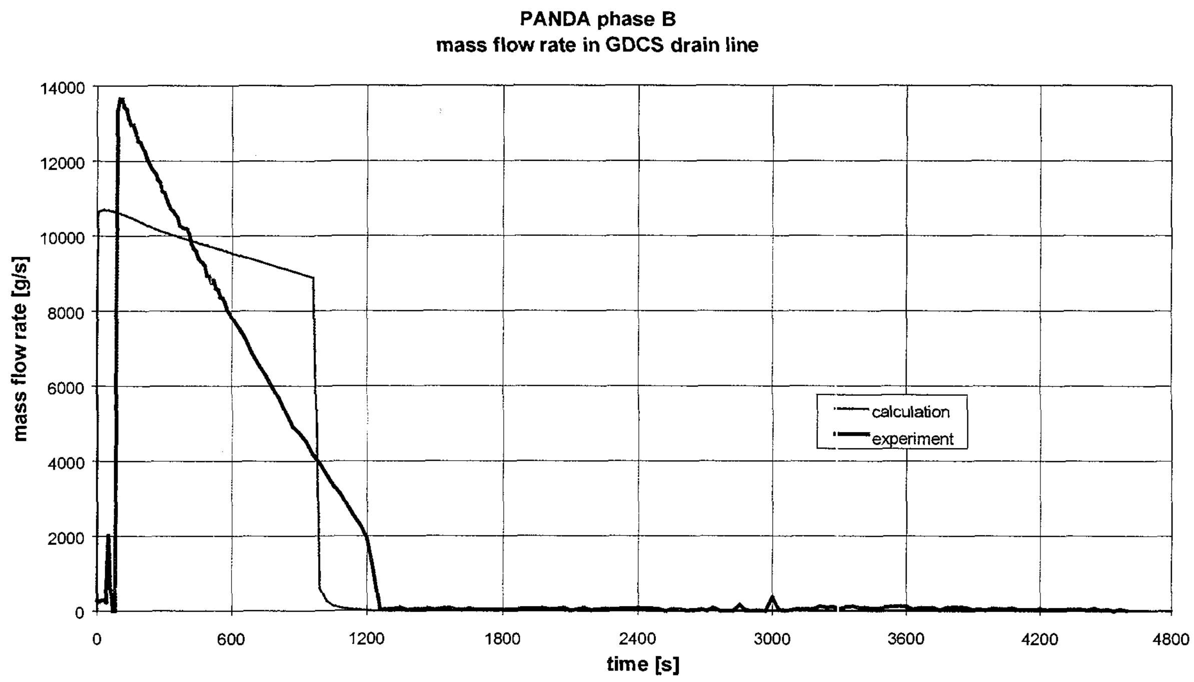

FIG. 3. PANDA-RALOC phase B, mass flow rate in GDCS drain line.

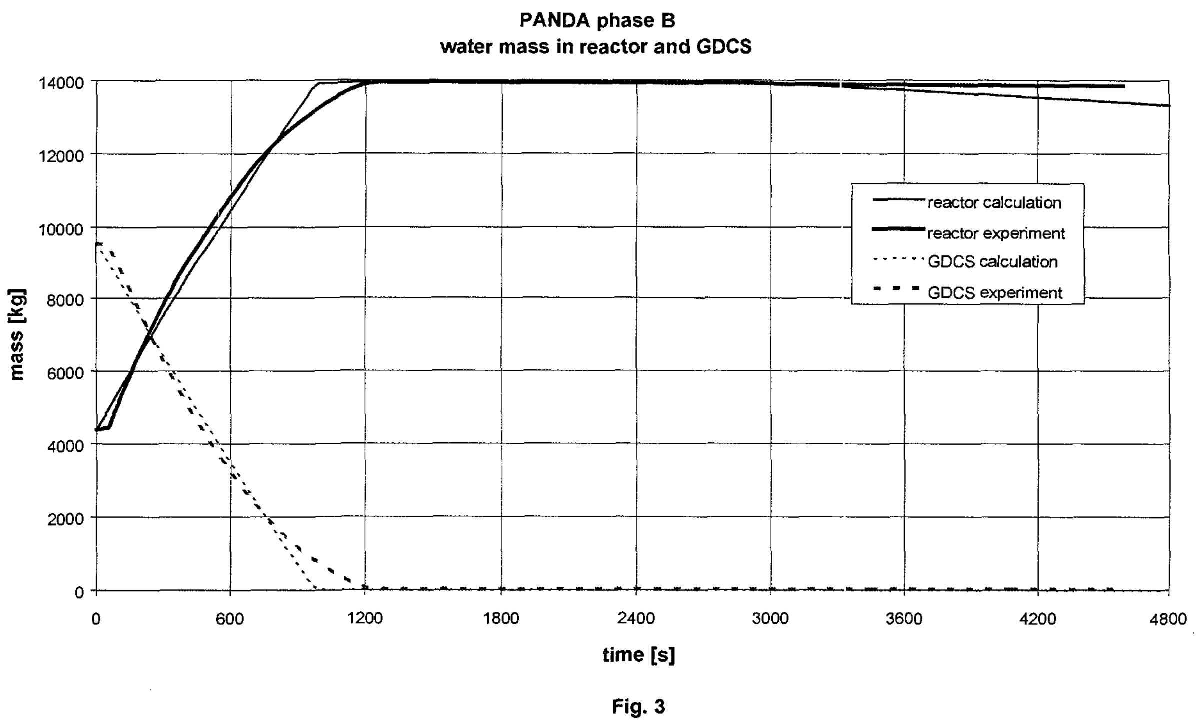

FIG. 4. PANDA-RALOC phase B, water mass in reactor and GDCS.

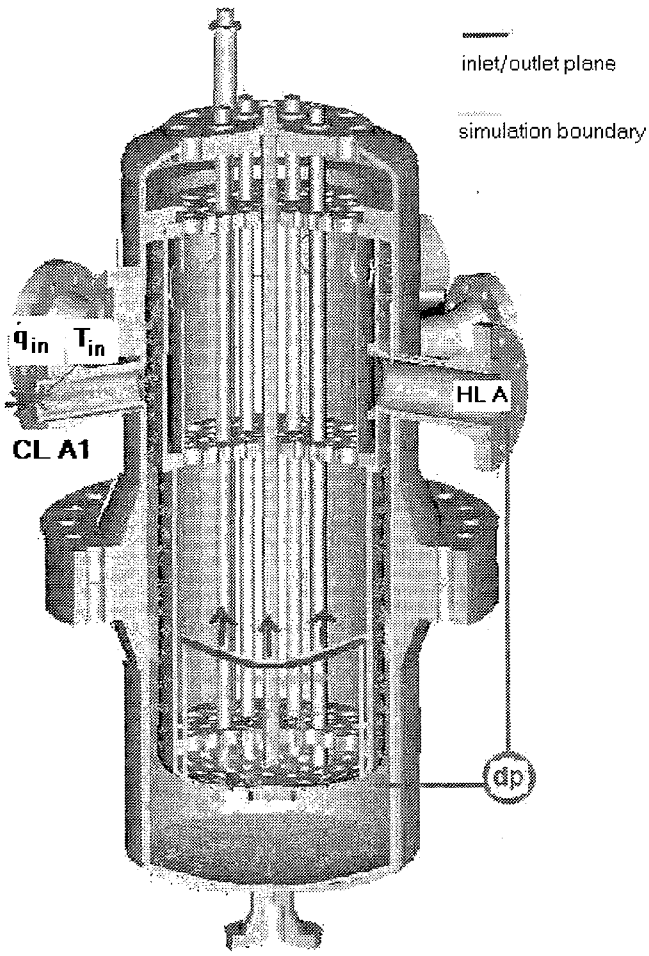

FIG. 5. ISP-43 solution region and boundary conditions.

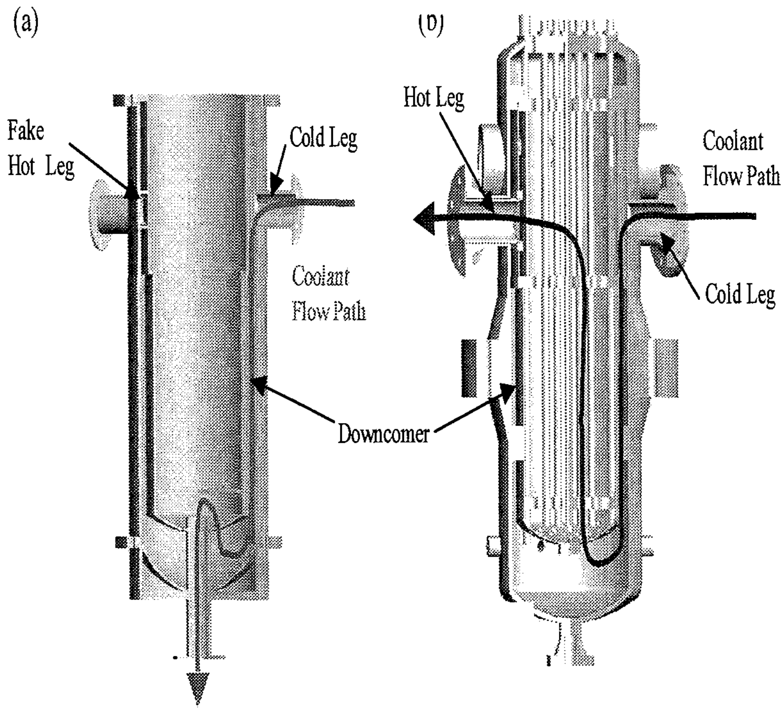

FIG. 6. ISP43- Cross-sectional view of the (a) visualization facility and (b) UM 2 × 4 loop facility vessels.

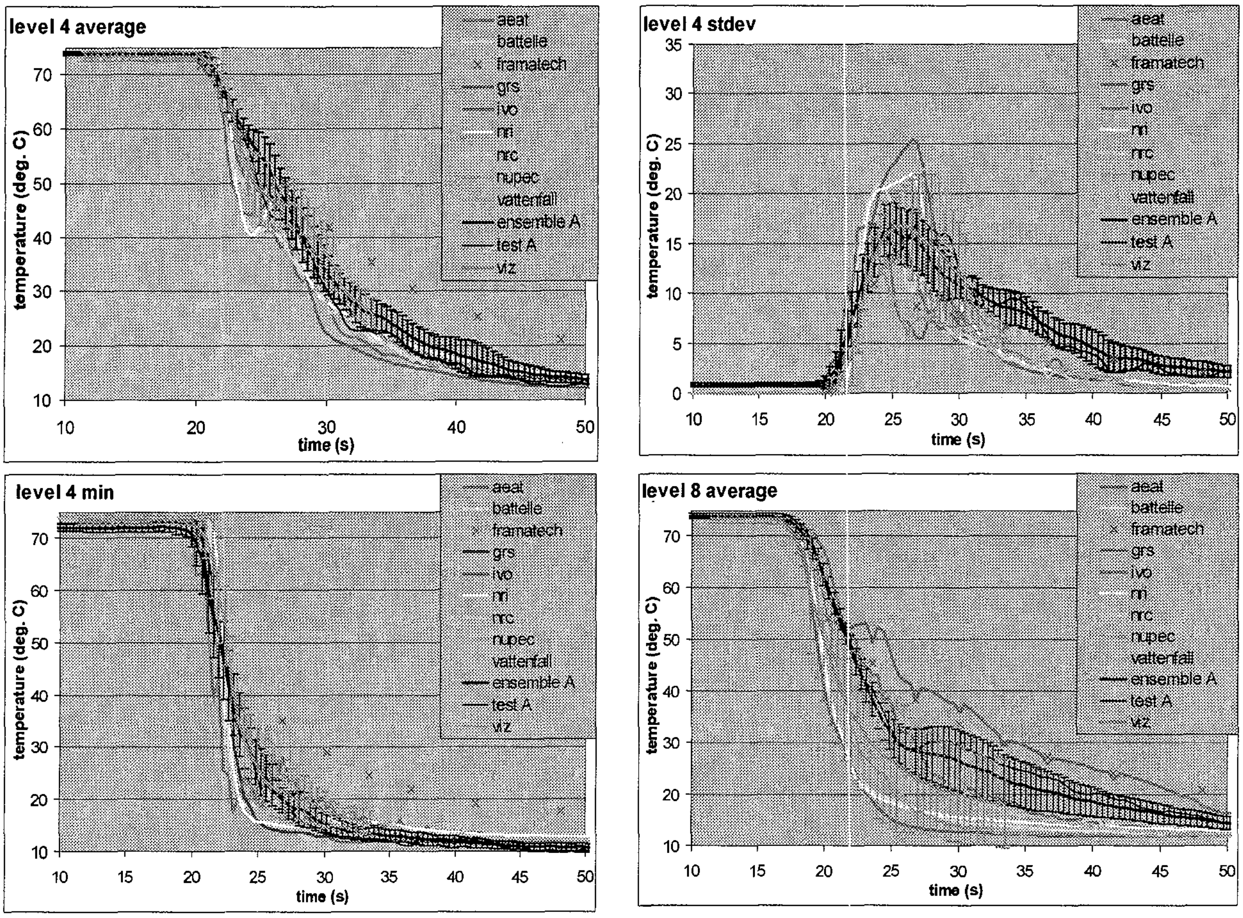

FIG. 7. ISP-43 Comparison of the results.

An experimental program was carried out in the Czech Republic, where Boiling crisis and Critical Heat Flux (CHF) were measured on the facilities that simulated the fuel assemblies of the former Soviet's Pressurised Water Reactors (PWR) WWER-440 and WWER-1000. The large part of experiments related to the CHF was performed at Skoda Plzeň Ltd, Nuclear Machinery Plant. The NRI started a complex of research activities in this field at the end of seventies.

The following topics have been developed systematically:

- Critical heat flux data bank for tubes, annuli and rod bundles
- Computational system based on the CHF data bank for testing critical heat flux correlations
- Subchannel analysis code for the calculation of coolant local thermo hydraulic conditions based on the rod bundles data obtained from the CHF data bank (subchannel CHF data bank)
- Computer code CALPER — a thermal hydraulic subchannel analysis code for the assessment of coolant local conditions in the fuel assemblies and in the core of PWR/WWER-type nuclear reactors
- System code calculations with CHF correlation for the safety analyses.

During the last years, a number of methodologies of improving NPP safety is under development within Czech Ministry of Industry and Trade, and computer codes are validated on experimental facilities. One of the important projects consists in searching new power sources and is also focused on the development of new methodologies including validation of computer codes.

2. BRIEF DESCRIPTION OF THE MAIN COMPUTER CODES

2.1. Computer code ATHLET

To perform analyses described above the advanced thermal hydraulic code ATHLET was applied, this code is used at the NRI since 1993. The code has been verified for the WWER accident analyses, in the first place by computations performed by the code-developer organization — GRS.

Significant advantage of the ATHLET code is the possibility to couple it with some other computer codes:

- 3D neutron kinetics codes (DYN 3D, BIPR 8)
- simple containment computer code CONDRU
- possibility of computer code for accurate calculation of reflood during LB LOCA

2.2. Computer code RELAP

This code was kindly transmitted to the Czech Republic by the US NRC. RELAP5 code is used for thermal hydraulic analyses of NPPs accidents, especially for loss of coolant accidents "small LOCA" and "medium LOCA", LB LOCA, primary-to-secondary leakages, and secondary side leakages. The RELAP5 (MOD2/RMA and later — MOD2.5 up to MOD3.2) is used at the NRI since 1989 for both analyses of NPP accidents and modeling of experimental facilities (pre and post-tests) which is very important for the code verification and generally for getting used to the code capability. Basic characteristics of the RELAP5 versions presented above are as follows:

- 1D model of two-phase flow including
- two mass conservation equations
- two energy conservation equations
- two momentum conservation equations
- equation for noncondensable gases
- boron transport
- point model of neutron kinetics
- hydrodynamics of the system, modeled using basic NAP components.

General features of the code and the way how the input data are introduced enables its application for various types of facilities.

In 1995 the model was modified to take into care of NPP Temelín analyses with new Westinghouse-supplied fuel, as well as for the verification of that of Zaporozhska NPP.

3. THE EXAMPLES OF COMPUTER CODES VALIDATION

3.1. The ISP-42 (PANDA)

PANDA is a large scale facility, which has been constructed at the Paul Scherrer Institute for the investigations of both overall dynamic response and the key phenomena of passive containment systems during long term heat removal phase for Advanced Light Water Reactors.

The whole ISP consists of six phases representing different scenarios:

Phase A:	Passive Containment Cooling System Start-Up
Phase B:	Gravity-driven Cooling System Discharge
Phase C:	Long-term Passive Decay Heat Removal
Phase D:	Overload at Pure-Steam Conditions
Phase E:	Release of Hidden Air
Phase F:	Release of Light Gas into the Reactor Pressure Vessel

3.1.1. PRE-TEST CALCULATIONS OF PHASES A,B,C,D WITH CATHARE CODE (Malacka, paper on PANDA Workshop OECD PSI JULY 2000)

Pre-test calculations of phases A,B,C,D were performed with code CATHARE 2 v1.4U_r1.51. Standard version of code was used without any modification of released version.

The nodalizations for phases B,C,D are similar. The standard modules of CATHARE (axial module, volume module, boundary condition module) were used. There are several common features in all nodalizations:

(1) The wet well volumes were joined into one volume.

(2) Instead of three PCC vent lines (or two in case of phase D) ends in the wet well the one common line is used ending in the wet well. For this, one auxiliary volume is used.

(3) In all axial and volume modules the walls are also modelled.

(4) Secondary sides of heat exchangers (PCC pools) are modelled as one volume module connected with one axial module.

(5) The standard CATHARE boundary condition BC5A module is used for open level at the PCC pools.

Problems, which were encoutered:

a) Initialisation

Problems with initialisation lead to a necessity to simplify input model. There were also problems to reach initial states of each transients. The auxiliary transients were used including sinks or injections of water, vapor and air. Those auxiliary transients could influence beginning of real calculated transients.

b) Heat exchange at passive containment cooler (PCC)

It was not possible to reach the right value of heat exchange at PCC. Mainly it is caused by used version of CATHARE code. The heat exchange in the calculations is very underestimated which influenced the courses of transients.

c) During calculations of phase C and D, when the water temperature at PCC pools reaches the saturated value, sudden unrealistic flow out of water from the pools follows. To prevent this event the pressure above the water level at pools was increased to the value of 130 kPa. The saturated temperature is then not reached, but there is also no evaporation of water from the pools which again influenced the results of pre-test calculations.

An example of calculation is on the figure, where there is a comparison of calculated and experimental reactor pressure during phase A. On this figure the problems, mentioned above, are illustrated. At the beginning of transient the calculated pressure is going above the experimental one due to auxiliary transient (see problem of initialisation). At the end of transient the experimental pressure decreases but calculated pressure goes up due to underestimation of heat exchange at PCC.

3.1.2. RALOC Model of ISP-42 in PANDA Facility (Simonkova, Paper on PANDA Workshop OECD July 2000 PSI)

The chosen phases A, B and F were calculated with code RALOC MOD 4.0 cycl AG. The basic objects simulating the individual parts of the PANDA facility represent **zones,** *juntions* and heat structures marked on diagram by different font style; structures are crosshatched. Each of the six vessels is simulated by one zone and the connection pipes between two drywells and two wetwells represents also one volume. Each of the three passive containment coolers is simulated by four volumes. The pipelines between vessels and coolers are modelled by various types of junctions. The action of valves is simulated by external control conditions. To all considered zones there are coupled heat structures representing the walls of vessels and modeling the heat transfer and conduction via walls. The heating power of reactor vessel and helium supply into reactor are modelled as a time dependent heat and He injection respectively.

During the pre-test calculations we met some problems which are listed as follows:

- The steam/air mixture from drywells entering in upper collector drum of PCC unit is not condensed in the tube bundle without definition of fluid parts with zero initial water mass in all PCC zones.
- Because it is not possible to connect the vent pipes to different zones (in case of wetwell 2) it was necessary to join together the units PCC2 and PCC3 into one unit PCC23.
- Some discrepancies were found in input values of cross section area of junctions without and with valves.

On the figures there is comparison of calculated and experimental results for phase B. There is compared the mass flow rate in GDCS drain line and its effect on the increase of the liquid level in reactor and the decrease of water mass in GDCS. In spite of different calculated and measured GDCS drain line mass flow the differencies between corresponding water masses in reactor and GDCS are very small.

3.2. Computer code FLUENT, ISP-43 (Muhlbauer: International Standard Problem ISP-43. Comparison of Pretest Calculations with Experimental Results. Report NRI 11 464, November 2000)

Increased power of present computers and progress in numerical methods and programming enables application of more sophisticated computer codes to some industrial problems. Before such aplication is made, the computer codes must be validated, especially when solving the problems of nuclear safety. Also the NRI therefore started validation and application of Computational Fluid Dynamics (CFD) codes to some selected problems encountered in NPP safety analyses. The commercial code FLUENT 5 was the first code undergoing such validation.

In the period of 1998-99, two sets of experiments focused on problems of rapid decrease of concentration of boric acid in reactor coolant at nuclear reactor core inlet were performed at the University of Maryland, US, under the auspices of OECD. The situation, when there is an inadvertent supply of boron-deficient water into the reactor vessel, could lead to a rapid (very probably local) increase of reactor core power in reactor, operated at nominal power, or to a start of fission reaction in shut-down reactor (secondary criticality). In the above mentioned experiments the transport of boron-deficient coolant through reactor downcomer and lower plenum was simulated by flow of cold water into a model of reactor vessel. These experiments were selected as the International Standard Problem ISP-43 and organisations, involved in thermal — hydraulic calculations of nuclear reactors, were invited to participate in their computer simulation. Altogether 10 groups took part in this problem with various CFD codes. The participants obtained only data on geometry of the experimental facility, and initial and boundary conditions.

A scheme of the experimental facilities is in Figs.. The University of Maryland (UM), College Park 2 × 4 Thermal-Hydraulic Loop and its plexiglass replica are scaled down models of the Three Mile Island Unit 2 Babcock&Wilcox pressurized water reactor. Cold water enters the vessel through the cold leg CL A1 and leaves it via hot leg HL A.. Two situations have been tested: front mixing (infinite volume of cold water), test A, and more realistic slug mixing (finite volume of cold water), test B. Average temperature at the downcomer outlet (24 thermocouples at the centre of the downcomer gap) was selected as the primary figure of merit for comparisons.

The NRI group selected very simple input model since the ISP-43 represented our first larger application of the FLUENT 5 code and mainly of the GAMBIT pre-processor. After several attempts we decided not to model the flaps in the lower part of the downcomer, the perforated bottom of the core barrel, lower support plate, and the heater rods, and spent the capacity of the computer on the rest of the domain. Also the outlet plane was situated at the position of the support plate, quite near the downcomer outlet. Hexahedral control volumes were used throughout the domain with the exception of the region of cold leg nozzles, where unstructured tetrahedral mesh was generated.

In Figures main results of calculations of all participants of the ISP-43 together with the experimental data are presented for the test A. Despite the very simplified input model, our results are well within the range of results of the other participants, and follow some significant features of the experiment. In the post-test phase, reasons of discrepancies should be identified, and corresponding lessons learned.

3.3. Conclusions

Experience accumulated in the process of the advanced codes application for various simulated accidents and transients allows us to believe that these codes can be used, with a great degree of confidence, not only for the Safety Reports purposes, but also to support simulators which will be in operation at both Czech power plants and also for new type of reactors .