



DIFFERENT CODES USED FOR REDUCING THERMAL HYDRAULICS PROBLEMS IN NUCLEAR REACTORS: A REVIEW

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ABSTRACT

The energy released in nuclear fission appears as kinetic energy of fission reaction products and finally as heat generated in the nuclear fuel elements. This heat must be removed from the fuel and reactor core and used to produce electrical power. The primary goals of thermal core design include achieving a high power density (to minimize core size), a high specific power (to minimize fuel inventory) and high coolant exit temperatures (to maximize thermodynamic efficiency). On the one hand, higher coolant flow rates will lead to better heat transfer coefficients and higher CHF limits. On the other hand, higher flow rates will also result in larger pressure drops across the core, hence larger required pumping powers and larger dynamic loads on the core components. Thus, the role of the hydrodynamic and thermal-hydraulic core analysis is to find proper working conditions that assure both safe and economical operation of the nuclear power plant. In the last four decades, large efforts have been undertaken to provide reliable thermal-hydraulic system codes for the analyses of transients and accidents in nuclear power plants. Whereas the first system codes, developed at the beginning of the 1970s, utilized the homogenous equilibrium model with three balance equations to describe the two-phase flow, nowadays the more advanced system codes are based on the so-called "two-fluid model" with separation of the water and vapor phases, resulting in systems with at least six balance equations. The wide experimental campaign, constituted by the integral and separate effect tests, conducted under the umbrella of the OECD/CSNI was at the basis of the development and validation of the thermal-hydraulic system codes by which they have reached the present high degree of maturity. However, notwithstanding the huge amounts of financial and human resources invested, the results predicted by the code are still affected by errors whose origins can be attributed to several reasons as model deficiencies, approximations in the numerical solution, nodalization effects, and imperfect knowledge of boundary and initial conditions.

Keywords: Thermal Hydraulics Codes, CFD, Coolant, Heat Transfer, Nuclear Reactor.

I. INTRODUCTION

In a power reactor, the energy produced in fission reaction manifests itself as heat to be removed by a coolant and utilized in a thermodynamic energy conversion cycle to produce electricity. Although this process is essentially the same as in any other steam plant configuration, the power density in a nuclear reactor core is typically four orders of magnitude higher than a fossil fueled plant and therefore it poses significant heat transfer challenges. Maximum power that can be obtained from a nuclear reactor is often limited by the characteristics of

heat transport system rather than nuclear considerations. Various factors that influence the power level include the coolant type, core configuration to maximize heat transfer surface area, coolant flow rate, thermo-physical properties of coolant and core materials, and consideration of material compatibilities. Thermal-hydraulics has played a vital role since the 1950s in the design, operation, performance and safety of nuclear power plants. As our knowledge in thermal-hydraulics progressed because of intense research and development (R&D) throughout the world, the design, operation and performance of nuclear power reactors have improved significantly. Authoritative reviews of role and challenges of thermal hydraulics for operating and advanced reactors have been conducted in the past. A review is also made of the computer codes developed for thermal-hydraulic analysis of nuclear reactors. The intention of this review is to compare these codes on the basis of their numerical method and physical models with particular attention to the two-phase flow and heat transfer characteristics. The thermal hydraulic analysis of nuclear reactors is largely performed by what are known as "System Codes". These codes predict the flows in the complex network of pipes, pumps and vessels that together form the thermal hydraulic systems of a nuclear reactor. Codes in this category include the US codes RELAP, TRAC and TRACE and the European codes CATHARE and ASTEC. Best-estimate thermal-hydraulic codes (e.g., RELAP, TRAC, CATHARE and ATHLET) are, in general, based on equations for two-phase flow which are typically resolved in Eulerian coordinates. Numerous computer codes have been written to calculate the thermal-hydraulic characteristics of the reactor core and the primary loop under steady-state and operational transient conditions as well as hypothetical accidents. New versions of some of these codes are still to come. The main purposes of the continuing effort in the development of such computer codes have been improved computational effectiveness and improved ability to predict the response of the core and the primary loop. Therefore, efforts have been continued to incorporate the recent models and methods of analysis in the areas of both hydrodynamics and heat transfer in two-phase flow to the extent that their prediction are reasonably reliable.

II. LIST OF REVIEWED THERMAL HYDRAULIC CODES

Name of Code	Reference Number	Name of Code	Reference Number
COBRA-I	1	RELAP5	12
COBRA- II	2	WOSUB	13
COBRA-III	3	RETRAN	14
COBRA-IIIC	4	TRAC	15
COBRA-IIIP	5	THERMIT	16
COBRA-IV-I	6	CATHARE	17
COBRA-DF	7	ATHLET	18
COBRA-TF	8	RELAP4-MOD6	19
RELAP2	9	RELAP4-MOD7	20
RELAP3	10	RELAP 4-EM	21
RELAP3B-MOD101	11	RELAP4-MOD5	22

However, such codes are fundamentally limited in that they are at heart only one-dimensional. If a part of the plant can reasonably modeled as one-dimensional flow in a pipe, these codes are excellent. However, there are plainly many important phenomena and locations where this one-dimensionality is not a good approximation. An obvious example of this might be flow within the bulky, 3-dimensional reactor vessel itself. There have been attempts to extend these System Codes to handle multi-dimensional effects. These have had some success, but there is naturally a trade-off between the fidelity of the representation and the computational complexity. It is over-simplified, but one might characterize a "3D System Code" as an array of one dimensional parallel pipes, allowed to interact 'sideways' with each other via some 'cross flow' coupling. The models so produced can be better than the original one-dimensional ones, but do not represent complex flows well. The most capable tool available to us for modeling these multi-dimensional effects is computational fluid dynamics (CFD). Modern CFD is able to produce high-quality predictions flows in complex geometries, but only with the use of large computing resources. It would be utterly impractical to build a CFD model of, for example, the entire primary circuit of a PWR. The most widely used of these codes, and the worldwide workhorse of nuclear reactor thermal analysis, is the RELAP suite, originating with the US NRC. Best-estimate system codes are currently used for the following:

- (i) safety analysis of accident scenarios;
- (ii) quantification of the conservative analyses margin;
- (iii) licensing purposes if the code is used together with a methodology to evaluate uncertainties; probabilistic safety analysis (PSA);
- (iv) development and verification of accident management procedures;
- (v) reactors design;
- (vi) analysis of operational events;
- (vii) core management investigation.

In the comparison that follows, both the advantages and drawbacks are noted in each code and ultimately it is attempted to assess the capability of each code for handling a specified case.

III. CLASSIFICATION OF NUCLEAR REACTOR THERMAL HYDRAULICS CODES

The existing thermal hydraulic codes may be classified under several categories as follows:

a) Capability of the system analysis

This contains two different classes of codes, namely, system component codes and loop codes. Basically, the hot channel or the fuel behavior codes are system component codes; however, some of these codes are extended to other situations far removed from subchannel (one channel) geometry. Integration of the down comer, jet pumps (in BWR's), bottom flooding, UHI and the like models into a component codes, makes it a vessel code. As distinct from the loop codes which are devised to analyze the whole primary side including reactor core and the secondary side, a variety of codes ranging from hot channel to vessel codes are called system component codes in this report.

b) Type of Two-Phase Flow Modeling

This part deals with the mathematical models used in thermal hydraulic codes to calculate the characteristics of the two-phase flow either in the reactor core or in the primary loop. The two pertinent methods in this respect, namely, the homogeneous equilibrium model and the two-fluid model fall in this category.

c) Range of Application

Since the capability of each code to handle flow and fuel rod calculations depends upon the mathematical models used to represent the physical situations as well as the numerical methods employed, codes can be classified in these respects into steady-state, transient and accident analysis (such as LOCA) codes. Naturally, the more demanding codes in this respect are ATWS and LOCA codes.

d) Type of Application

Codes may also be classified based upon their types, i.e., Best Estimate (BE) type and Evaluation Model (EM) type. The latter group are basically devised for the purpose of licensing. The type of nuclear reactor for which thermal hydraulic codes are devised (such as PWR, BWR and LMFBR) may be another category.

IV. CLASSIFICATIONS ACCORDING TO SYSTEM ANALYSIS CAPABILITY

4.1 Component Codes

A core thermal hydraulic assessment necessitates analysis of fluid passing axially along the parallel rod arrays. Such analysis is difficult to conduct due to the degree of freedom associated with parallel rod array and the two-phase flow and heat transfer involved in nuclear reactors. In addition, a radial and axial variation of the fuel rod power generation exacerbates this situation. Assumptions have been made to simplify the task of modeling the hydrodynamics and heat transfer characteristics of the rod arrays. Generally, there are three pertinent methods used in rod bundle thermal hydraulic analysis of the nuclear reactor core as well as heat exchangers, namely, (a)-subchannel analysis, (b)-porosity and distributed resistance approach and finally (c)-benchmark rod-bundle analysis which uses a boundary fitted coordinate system. The first approach is widely used in the subchannel codes such as COBRA, FLICA, HAMBO and THINC. Whereas the second approach is employed in THERMIT. The subchannel approach will be more elaborated upon here, while a discussion in detail of these three concepts is presented in Ref 21. In the subchannel approach, the rod array is considered to be subdivided into a number of parallel interacting flow sub channels between the rods. The fluid enthalpy and mass velocity is then found by solving the field or conservation equations for the control volume taken around the subchannel. Although a rod-centered system with subchannel boundaries defined by lines of "zero-shear stress" between rods (Fig. 1) seems to be well-defined control volumes, it has become customary to consider a coolant centered subchannel as a control volume.

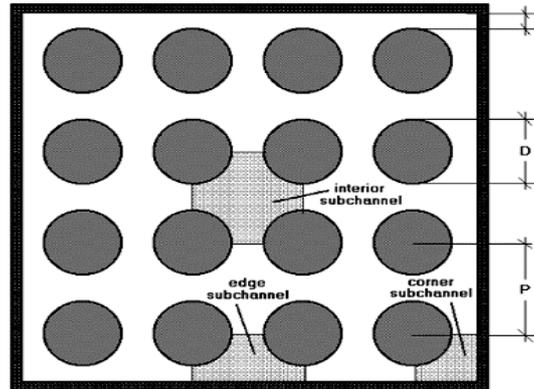


Fig 1 Coolant Centered Subchannel

4.2 Thermal Hydraulics Codes for Light Water Reactors

Around 350 Light Water Reactors (BWR and PWR) are operating throughout the world (NN, 2011). Apart from improvement in their capacity factors, many BWRs and PWRs have been uprated from their Original Licensed Thermal Power (OLTP). This has been achieved by combination of plant and fuel modifications, and by application of best-estimate (BE) Modeling and Simulation (M&S) capabilities that have enabled the recovery of conservatism in safety analysis. Reactor startup and shut down in the world as shown in Fig. 2.

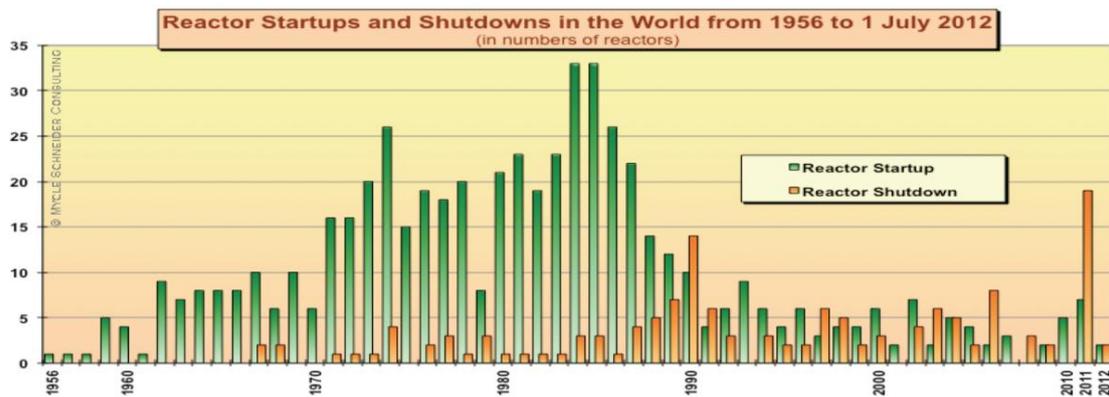


Fig 2 Reactor Startups and Shutdowns in the World

Other BWR and PWR fuel vendors have followed similar approaches with codes such as SRELAP and WCOBRA/TRAC (Leberig et al., 2009; Curca-Tivig, 2009; Yano et al., 2010; Westinghouse, 2007). The analytical methods used in the original design and licensing of the early BWRs were relatively simple; typically a four equation drift flux model was used for the thermal hydraulics, and point kinetics was used for transient nuclear kinetics. The calculations were done in sequential steps; one code would calculate the system response for the reactor primary system, one code would be used for the limiting fuel bundle to determine the thermal margins, and a separate code would be used for containment analysis if needed. Loss of Coolant Accident (LOCA) analysis is a typical example on such sequential analyses. Significant conservatisms were included in the analyses; conservatisms were typically included to bind the uncertainties in the analytical models and correlations. Most plants were therefore LOCA limited. At the same time the development of more advanced methods, as example based on the TRAC-BWR code (Borkowski et al., 1992) was initiated. These advanced models use two-fluid six-equation models for the thermal-hydraulics coupled with Best Estimate models for

interfacial and wall shear and heat transfer. The two-fluid model includes both a 1D and a multi-dimensional formulation. In addition models for non-condensable gasses and dissolved boron in the liquid phase were included. A 3D nuclear kinetics model consistent with GE's 3D core simulator PANAC and an improved fuel thermal-mechanical model were included. The applications abandoned the use of conservative models and switched to the best-estimate models coupled with a statistical quantification of total uncertainties from models, initial conditions and plant parameter uncertainties. The application process followed the Code Scaling Applicability and Uncertainty (CSAU) methodology (USNRC, 1989a) and USNRC RG 1.157 (USNRC, 1989b). TRACG, the GE proprietary version of TRAC-BWR code, was approved by USNRC for the reactor stability 'Detect and Suppress' solutions in 1996 (BWROG, 1996) and for operating transients in 2001 (Andersen et al., 2006b). The stability Detect and Suppress Solution – Confirmation Density (DSS-CD) for Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operation was approved in 2007 (GEH, 2007). In addition, TRACG has been used and approved by USNRC for transient, LOCA, stability and Anticipated Transients without Scram (ATWS) applications for the ESBWR (GEH, 2010a,b,c,d). The introduction of better models combined with the more rigorous statistical method used to quantify the uncertainty generated even further margin improvement for the BWR.

4.3 Thermal Hydraulic Codes for AHWR

The Indian advanced heavy water reactor (AHWR) is a 920 MW vertical pressure tube type boiling light water cooled and heavy water moderated system (Sinha and Kakodkar, 2006). Analysis has been carried out for simulating loss-of-coolant accident (LOCA) at inlet header in a natural circulation type reactor developed as the advanced heavy water reactor (AHWR). In this research it covers a case of LOCA due to 200% break at inlet header which is double ended rupture. The maximum clad surface temperature has been predicted in different cases by using the thermal hydraulic safety code RELAP5/Mod4.0. This reactor is equipped with emergency core cooling system (ECCS) and isolation condensers (ICs) to remove decay heat during LOCA. This ECCS provides cooling to fuel in passive mode during first fifteen minutes of LOCA and it is achieved by high pressure injection from advanced accumulator. Thermal hydraulics parameter analyze by using the advance RELAP5/MOD 4 simulation code which incorporated these advance models mentioned in below. The effect of these models are-

- ❖ Exact simulation of water road.
- ❖ Conditions in ECCS header.
- ❖ CCFL (Countercurrent Flow Limitations) model in the reactor core.
- ❖ Momentum mixing & heat structure model in ECCS header.
- ❖ Reactor kinetics (neutronics) modeling.

Postulated scenarios during Station Blackout are analyzed and mention in Tyagi et al. (2010b). This Study shows that the max clad temperature in hot pin and hot channel is within acceptable limit as per ECCS acceptance criteria during 200% large break LOCA. Results obtained by thermal hydraulic analysis of proposed natural circulation reactor are found to be within acceptable limits as mentioned above. Conclusion is based on deterministic assessment. Number of scenarios in different categories having been analyzed in design basis and beyond design basis categories. In the category of “decrease in coolant inventory” spectrum of breaks have been

analyzed at different location. 200% inlet header break is found to be enveloping initiating event in design basis category considering events in other categories also. Detailed ECCS acceptance criteria are shown in Fig 3. Maximum fuel rod clad temperature shall not exceed 1477 K.

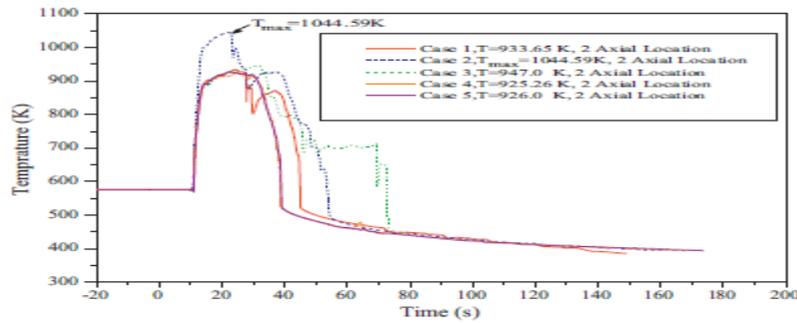


Fig 3 Hot Pin Clad Surface Temperature Transient at Maximum Axial Location for Hot Channel 05 Volume

4.4 Thermal-Hydraulic Code for Lead-Cooled Nuclear Fast Reactors

A new multi-physics simulation tool FRENETIC (Fast Reactor Neutronics/Thermal-hydraulics) is presented for the quasi-3D analysis of a lead-cooled fast reactor core with the hexagonal fuel element configuration, as currently proposed within the framework of the European project LEADER. The tool implements coupled neutronics (NE) and thermal-hydraulic (TH) models. In the NE module, a 2D + 1D full-core multi-group diffusion solver has been developed based on a coarse mesh nodal scheme and adapted to cope with the hexagonal geometry. In Fig. 4 we also distinguish between codes (as FRENETIC) applicable to the full core, or to the whole reactor (i.e., including also the external circuit(s) feeding the core

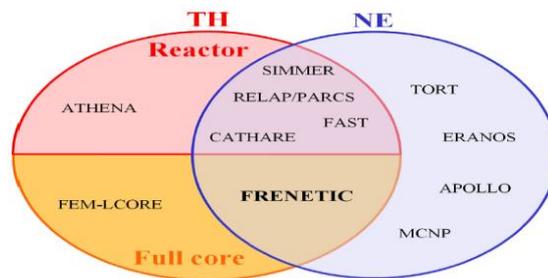


Fig 4 Representation of Available Tools for Thermal-Hydraulic Modeling

Table 1 Main Features of the FRENETIC Code Compared to the Already Available Tools for Coupled Neutronics/Thermal-Hydraulic Modeling of Nuclear Reactors

		SIMMER IV	RELAP5+PARCS	RELAP5-3D (+NESTLE)	FAST (TRACS/AAA+PARCS)	CATHARE	FRENETIC
Neutronic analysis	Transport theory	x				x	
	Diffusion approximation		x	x	x		x
	Dimension	3D	3D	1D-2D-3D	3D	3D	2D→3D
NE geometry	Cartesian/cylindrical	x	x	x	x	x	
	Hexagonal		x	x	x		x
Thermal hydraulic analysis	Fuel assembly	x					x
	Core	x	x	x	x	x	x
	System	x	x	x	x	x	
	Original working fluid	Na	Water	Water	He	Water	Pb & LBE
TH geometry	Dimension	3D	2D	3D	3D	1D-3D	1D+2D
	Cartesian/cylindrical	x	x	x	x	x	
Mechanical analysis	Hexagonal						x
		x			x	x	

The performance of FRENETIC has been tested in a few selected test cases reported here. The successful validation of the TH model in the case of a single hexagon, against experimental data from the ICE test section at the ENEA Brasimone research center, has already been presented elsewhere (Zanino et al., 2012; Bonifetto et al., 2012b). The fuel is assumed to be UO₂ (Popov et al., 2000), while the wrapper material is 18/12 stainless steel (similar to AISI 316). The number of hexagons (fuel elements) in the fuel region of the reference configuration is 451, unless otherwise noted, the reactor thermal power is 1500 MW and the axial power form factor $\phi = 1.3$ (Sobolev et al., 2007). The other main geometrical parameters, taken from (Sobolev et al., 2007; Mansani, 2012). The new code FRENETIC has been presented for the coupled neutronics/thermal-hydraulic analysis of lead-cooled fast nuclear reactors, as currently foreseen in the European program within the Generation IV framework. Both the neutronics and thermal-hydraulic modules of FRENETIC have been benchmarked against analytical solutions. The extension of the neutronics model to include 3D and transient capabilities is under way. In a separate, parallel effort, the multi-hexagon thermal-hydraulic model will be validated against experimental data. Eventually, the coupled neutronics/thermal-hydraulic model should also be validated against experimental data.

4.5 Thermal-Hydraulics Code for High Temperature Gas Cooled Reactor

DYN3D is a nodal diffusion code for 3D steady-state and transient analysis of Light Water Reactor (LWR) cores with hexagonal or square fuel element geometry. In addition to the neutron kinetics, it comprises of a thermal-hydraulics model for flow in parallel coolant channels. The possibilities of new computer generations allow a much more detailed analysis of complex physical and chemical interrelations which play a role in the determination of the plant behavior in normal operation and under accident conditions. Therefore, by using existing calculation methods for High Temperature Gas Cooled Reactors (HTGRs) and Light Water Reactors (LWRs), a new generation of simulation tools can be developed for HTGRs with pebble-shaped and prismatic fuel assemblies and for related systems for process heat and hydrogen generation. The RPT methodology allows combining the high spatial resolution of MC codes with the superior computational speed of deterministic lattice codes. Computational efficiency is essential, since for few-group cross section sets generation about hundreds to thousands branch-off calculations are required. At the first stage, a MC code is used to simulate an HTGR fuel element with explicitly described TRISO particles at the beginning of life (BOL) and at the nominal operational conditions.

A simplified version of the prismatic HTGR core developed by Idaho National Laboratory (INEEL) in the frame of Next Generation Nuclear Plant (NGNP) (MacDonald et al., 2003) project was considered as a reference core in this comparison. The NGNP core includes control rods channels asymmetrically distributed in the fuel blocks and the outer reflector. However, these control rods channels were not modeled in this study. The modified HTGR core layout is shown in Fig. 5. Materials specifications, TRISO particle layers dimensions, and packing fraction of TRISO particles were adopted from (DeHart and Ulses, 2009). The major core parameters are summarized in Table 2.

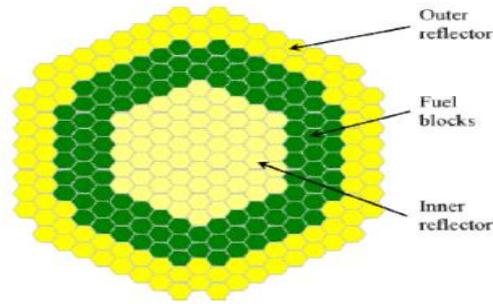


Fig. 5 Simplified HTGR Core Model

Table 2: The Simplified HTGR Core Data

Power, MW	600
Active core height, m	7.9
Number of fuel columns	102
Fuel assembly pitch, cm	36
Pin pitch, cm	1.88
Fuel channel diameter, cm	1.27
Coolant channel diameter, cm	1.588
Fuel compact diameter, cm	1.245
Compact height, cm	4.93
Number of TRISO particles per fuel compact	3000
TRISO packing fraction, %	19.72
Graphite matrix density, g/cm ³	1.75
TRISO particle data UO ₂ fuel density, g/cm ³	10.4
Enrichment, %	8.2
Outer TRISO radius, mm	0.455
Fuel kernel radius, mm	0.250
Coating materials C/C/SiC/C Coating thickness, mm	0.09/0.04/0.035/0.04
Coating densities, g/cm ³	1.05/1.90/3.18/1.9

A preliminary procedure for analysis of HTGR core using HELIOS and DYN3D codes was established and tested. Good agreement in keff and radial power distribution between DYN3D and reference MCNP results was observed. Nevertheless, several further steps should be taken in order to improve the accuracy of calculations:

- ❖ The neutron migration length in graphite moderated cores is significantly higher than that of LWR cores (~26 cm vs. ~7 cm), which results in a stronger spatial coupling between core regions. Thus, the neutron spectrum in fuel blocks is more affected by presence of graphite reflectors. Therefore it would be worthwhile to generate the fuel block cross sections by taking into account surrounding environment.
- ❖ In order to better account for up scattering events in graphite the number of thermal energy groups has to be increased and the coarse mesh group structure has to be optimized.

- ❖ For more realistic core configurations with asymmetrically allocated control rod channels it would be necessary to perform 3D full core calculations with refined triangular mesh.
- ❖ In order to better account for the neutron flux anisotropy in the core, the application of transport approximations higher than diffusion theory should be considered.

V. MAIN FEATURES AND LIMITATIONS OF THERMAL-HYDRAULIC SYSTEM CODES

The system thermal-hydraulic codes are based upon the solution of six balance equations for liquid and steam that are supplemented by a suitable set of constitutive equations. The balance equations are coupled with conduction heat transfer equations and with neutron kinetics equations (typically point kinetics). In the following sections, main problematic aspects, from the point of view of the user, of a thermal-hydraulic system code are highlighted.

5.1 System Nodalization

All major existing light water reactor (LWR) safety thermal hydraulics system codes follow the concept of a “free nodalization,” that is, the code user has to build up a detailed noding diagram which maps the whole system to be calculated into the frame of a one-dimensional thermal-hydraulic network. Due to the existing code limitations and to economic constraints, the development of such a nodalization represents always a compromise between the desired degree of resolution and an acceptable computational effort.

5.2 Code Options: Physical Model Parameters

Even though the number of user options has been largely reduced in the advanced codes, various possibilities exist about how the code can physically model specific phenomena. Some examples are as follows.

- (1) Choice between engineering type models for choking or use of code implicit calculation of critical two-phase flow conditions.
- (2) Two-phase flow characteristics of the main coolant pumps.
- (3) Pressure loss coefficient for pipes, pipe connections, valves, branches, and so forth.

5.3 Input Parameter Related to Specific System Characteristics

The assessment of LWR safety codes is mainly performed on the basis of experimental data coming from scaled integral or separate effect test facilities. Typically in these scaled-down facilities, specific effects, which might be small or even negligible for the full-size reactor case, can become as important as the major phenomena to be investigated. Examples are the release of the heat from the structures to the coolant, heat losses to the environment, or small bypass flows. Often, the quality of the prediction depends largely on the correct description of those effects which needs a very detailed representation of the structural materials and a good approximation of the local distribution of the heat losses. However, many times the importance of those effects is largely underestimated, and consequently, wrong conclusions are drawn from results based on incomplete representation of a small scale test facility.

5.4 Specification of Initial and Boundary Conditions

Most of the existing codes do not provide a steady-state option. In these cases pseudo-steady-state runs have to be performed using more or less artificial control systems in order to drive the code towards the specified initial conditions. The specification of stable initial and boundary conditions and the setting of related controllers require great care and detailed checking. If this is not done correctly, a large risk, that even small imbalances in the initial data will overwrite the following transient, exists especially for slow transients and small break LOCA calculations.

5.5 Specification of State and Transport Property Data

The calculation of state and transport properties is usually done implicitly by the code. However, in some cases, for example, in RELAP5, the code user can define the range of reference points for property tables, and therefore, can influence the accuracy of the prediction. This might be of importance especially in more difficult regions.

5.6 Selection of Parameters Determining Time Step Sizes

All the existing codes are using automatic procedures for the selection of time step sizes in order to provide convergence and accuracy of the prediction. Experience shows, however, that these procedures do not always guarantee stable numerical results, and therefore, the user might often force the code to take very small time steps in order to pass through trouble spots. In some cases, if this action is not taken, very large numerical errors can be introduced in the evolution of any transient scenario and are not always checked by the code user.

VI. CONCLUSION

In conclusion, the present status, of system codes development, assessment, and related uncertainty evaluation, is inadequate as far as the largest majority of design and safety problems of current water-cooled reactors are concerned. Anyway, new scientific goals must be achieved. To this aim, projects and programmes based on the development of system codes with multidimensional and multifluid capability and with "open" interfaces for an easy coupling with other codes in areas like neutronics (for implementing presently available 3D codes), CFD, structural mechanics (e.g., for pressurized thermal-shock studies), and containment constitute the new frontier of the scientific and engineering community in this field. However, taking into account that the development of such codes with measurable increased improvements in their capabilities may need several decades, it is an evident consequence that the existing system thermalhydraulic codes are going to be used for one or two decades in their present configuration. It is evident that all the progress has been made in the recent past is a consequence of experimental researches. After 30 years of validation through basic, separate and integral effect tests facilities and after code improvements, system codes are able to predict main phenomena of PWR & BWR transients with reasonable accuracy. Nowadays the attention should be focused more on developing procedures for a consistent application of a thermal hydraulic system code. This need has been highlighted in the paper and implies the drawing up of specific criteria through which the code-user, the nodalization and finally the calculated transient results can be qualified.

VII. FUTURE DIRECTIONS FOR THERMAL-HYDRAULIC R&D

Modern fuel design depends on a very accurate characterization of pressure drop, flow distribution and heat transfer characteristics. Computational Fluid Dynamics (CFD) while not suited for system calculations is very well suited for component performance and design optimization. Current CFD codes do not have adequate two-phase flow capabilities and Direct Numerical Simulation (DNS) is not a practical option at this time (Bestion, 2011). However, progress is being made and continued R&D is highly recommended. The formulation of a transport equation for the interfacial area (Ishii, 2011) is an example of promising thermal-hydraulic R&D effort since about 15 years and that should be further explored. Improved two-phase flow models and closure relations will allow detailed analysis and optimization of two-phase flow in BWR components, such as fuel bundles, upper plenum and chimney (for ESBWR), steam separators and steam dryers. Further development and application of subchannel methods, which are less sophisticated than CFD but more practical because of the reduced computer requirements, should also be continued for detailed thermal-hydraulic analysis in fuel assemblies. Thermal hydraulic R&D to improve the understanding and characterization of fluid structure interactions is needed. The increased power density coupled with the pressure drop characteristics and reduced thermal time constants of modern 10×10 fuels has reduced the margin to thermal-hydraulic instability. These are the key points for reducing thermal hydraulics problems in nuclear reactors:

- ❖ Advanced use of system scale codes and couplings.
- ❖ Optimization of flow.
- ❖ Design and Integration
- ❖ Optimization regards to thermal fatigue.
- ❖ Multi-physics 3×3 pin model
- ❖ Scaling from 3×3 pin modeling to larger geometries

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