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Integrated Approach for Actual Safety Analysis

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1. Introduction

Actual trend in reactor safety deterministic analysis are evolving toward best estimate approach. Best estimate analyses imply use of best estimate codes and input data. The best estimate concept is not limited to thermal-hydraulics rather in general terms it covers many fields, likewise three dimensional neutron kinetics, structural analysis and containment performance evaluation.

The general frame is to put efforts in avoiding conservative assumptions performing analysis adopting the best tool available for each specific topic, all contributing to give an integrated evaluation of the plant response.

The needs to adopt an integrated approach in performing safety analysis come from the inherent complexity of a Nuclear Power Plant and from the tight interactions among the subsystems constituting the plant itself. These interactions directly involve the necessity to consider a broad spectrum of disciplines typically coming into play in different not interacting analyses.

An example of the integral approach is given in the present document. The integral approach has been pursued for the safety analyses of the 'post-Chernobyl modernized' Reactor Bolshoy Moshchnosty Kipyashiy (RBMK) specifically for Smolensk 3. These analyses were performed at the University of Pisa within the framework of a European Commission sponsored activity.

The mentioned analyses deal with events occurring in the primary circuit, as well as excluding those events originated from plant status different from the nominal operating conditions. Following the evaluation of the current state of the art in the safety analysis area, targets for the analysis were established together with suitable chains of computational tools. The availability of computational tools, including codes, nodalizations and boundary and initial conditions for the Smolensk 3 Nuclear Power Plant, brought to their application to the prediction of the selected transient evolutions that, however, are not classified as licensing studies.

The integrated approach for safety analysis yields to the evaluation of complex scenarios not predictable adopting just a single computational tool. Example is given considering the Multiple Pressure Tube Rupture (MPTR) event which constitute one of the main concern of this kind of plant.

The content of this document includes an introduction to the critical issues to be accounted for in the frame of an integral safety analysis approach; the selection of suitable computational tools to proper deal with the scenario subject of the investigation; an

approach on how to link (coupling issues) the selected tools; the use of intermediate code outcomes and interpretation of the global predicted plant behaviour. All the aspects presented in general terms are applied in the case study of a Multiple Pressure Tube Rupture having as reference plant the Smolensk 3 Nuclear Power Plant. The selected event may occur as a consequence of a fuel channel blockage which (if not detected) brought to the rupture of the affected pressure tube. The dynamic loads generated by its breach may lead to the rupture of the surrounding pressure tubes. Direct consequence of the pressure tube rupture is the pressurization of the reactor cavity which envelopes all the core. In the case of Multiple Pressure Tube Rupture event, involving a large number of pressure tubes, the lifting of the reactor cavity top may occur, putting in direct connection the core with the environment. The present example is a kind of analysis that cannot be performed if an integrated approach is not adopted.

2. Framework

The best estimate approach is the actual trend of the NPP deterministic analysis (International Atomic Energy Agency [IAEA], 2008). The concept of best estimate is generally applied to the software codes used in the analysis. However the best estimate approach concept has a broader meaning. It applies to the general framework of the analysis, and it involves not only the codes, but the kind of analyses to be performed, the approach to realize the models to be realized for the analyses, the input data including boundary and initial conditions also. The best estimate approach is not only connected with a calculation performed with a best estimate code. The result of the analysis is a best estimate evaluation, if all the aspects of the analysis (input data, systems models, results) are best estimate, in addition to the codes. As a consequence the use of a best estimate code, assuming not best estimate data or systems model cannot be considered a best estimate analysis.

A calculation of a complex system like a NPP, poses a lot of issues to perform a best estimate analysis. The main relevant aspect is constituted by the many areas involved in the analysis of a NPP. Knowledge in many technical areas are necessary. The solution can be obtained by “linking” in a single instrument of investigation the different tools developed for investigation in each of the different areas.

2.1 Complexity of the approach

The scope is the safety of nuclear power plants, is demonstration of the capability to keep the radiation exposure of personal and population within specified limits. It is ensured by maintaining the integrity of safety barriers, which are part of the plant defence in depth concept.

A series of barriers prevents the release of radioactive fission products from their source beyond the reactor containment and into the environment. In analyzing the NPP safety, it is essential to assess the integrity of these barriers and to decide to what degree the response of the whole NPP and its systems to a certain initiating event is acceptable from the viewpoint of the plant safety. The integrity of the safety barriers is related to certain threshold values, which are referred to as acceptance criteria. Design limits are adopted with a conservative margin so that the safety barrier integrity is guaranteed as long as the parameters do not exceed the relevant criteria. In the case of not efficacy of the barriers a radioactive release occurs and an evaluation of the dose to the workers and population is done (IAEA, 1996 and IAEA, 2000).

The complexity of the analysis is due to the involvement of a number of different technological areas requests a detailed identification of topics and targets together with a suitable connection with adopted codes and activities.

The nuclear technology sectors or computational areas relevant for NPP safety and design include the following areas: the system thermal-hydraulics, the computational fluid-dynamics, the structural mechanics, the neutron kinetics with the cross section generation and the fission product release and transport.

The interconnections among individual Technological Areas identify a chains of codes. Figure 1 gives an idea of the complexity of the activities and related technological areas necessary for a such analysis.

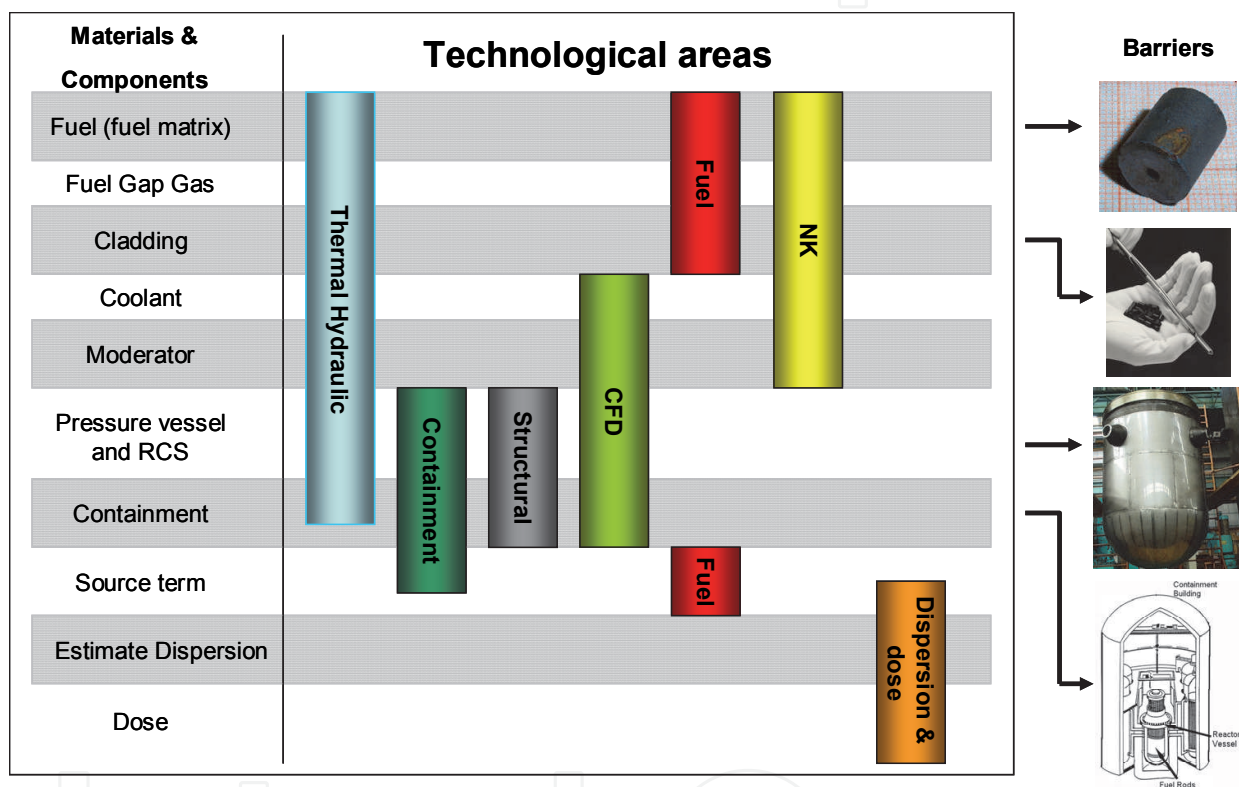


Fig. 1. Technological areas for the integrated an analysis

The effort to perform a such analysis is aimed to establish a connection with the regulatory or licensing environment. This connection must take into account the evolution of safety concepts following improvements of the technical knowledge, including the availability of powerful computational tools and of experimental evidences.

The framework constitutes by the development & qualification of computational tools is also related to relevant points like “physical phenomena understanding”, and “analysis of complex scenarios expected during accident conditions” considering the current licensing practices.

The strategic objective is the set-up of a suitable chain of codes to deal with accident scenarios. The motivation for the selection of individual accidents is given by expecting challenging phenomena for the concerned safety barrier. The concerned phenomena shall also be connected with the existing code typologies and capabilities. These codes are

supposed to be qualified for the prediction of individual accidents whose relevant and detailed boundary and initial conditions have been defined.

The list of phenomena, which are taking place during progression of an accident shall be analyzed, discussed and selected. Relevant information can be taken from international literature (e.g. IAEA, 2002) or from experimental tests. The operative objective is to demonstrate the capability of computational tools to reproduce relevant transient phenomena and to show that the same tools can be linked together.

Generally speaking, best estimate is associated to the TH SYS codes. About this kind of codes is clear the meaning of best estimate approach. Descriptions of this concept are largely diffused in international literature. The concept of best estimate is less clear about the codes related to the other technological areas. The general concept of best estimate approach is in avoiding any intentional conservatism. This concept is applied in all the aspects of the calculation: input data, conditions of the calculation, model of the systems and of course the code. From this point of view the aspects to be considered for each individual codes are:

- The physical modelling.
- The approximations that are made and their limitations.
- The used correlations.
- An assessment of uncertainties due to the physical models.
- The practice of application associated to these codes
- and their level of validation and/or certification.
- the associated impact on the drawing of safety analyses.

In such a complex analysis, requiring different codes, the data used as input for a code are derived from the result of another previous code calculation. So a relevant role is also played by the evaluation and selection (as input data in next calculations) of the results obtained by code application. The figure. 2 gives an idea of the links between the different technical areas.

Referring to the figure 2, some links are hereafter exemplified.

- Path a) the TH codes results are used to supply boundary data to the code for fuel evaluation.
- Path b) the TH codes supply the thermal hydraulic boundary conditions to the NK codes. The results of the NK codes are supplied to the TH code core component.
- Path c) the TH codes supply the thermal hydraulic boundary conditions to the CFD codes. The results of the CFD codes are supplied to the TH for evaluation of specific areas of the systems.
- Path d) the CFD codes supply the boundary conditions to the Structural code for evaluation of mechanical resistance of systems components.
- Path e) the results from the Containment code are supplied to the TH codes for calculation of the evolution of the accident in the reactor coolant system and containment.
- Path f) the results of the Structural code about the integrity of the systems (e.g. containment systems) are supplied to the containment codes.
- Path g) the results of the Containment code about possible failure and source terms are supplied to the codes for dispersion and dove evaluation.
- Path h) the results of the Fuel code about source terms are supplied to the codes for dispersion and dove evaluation.

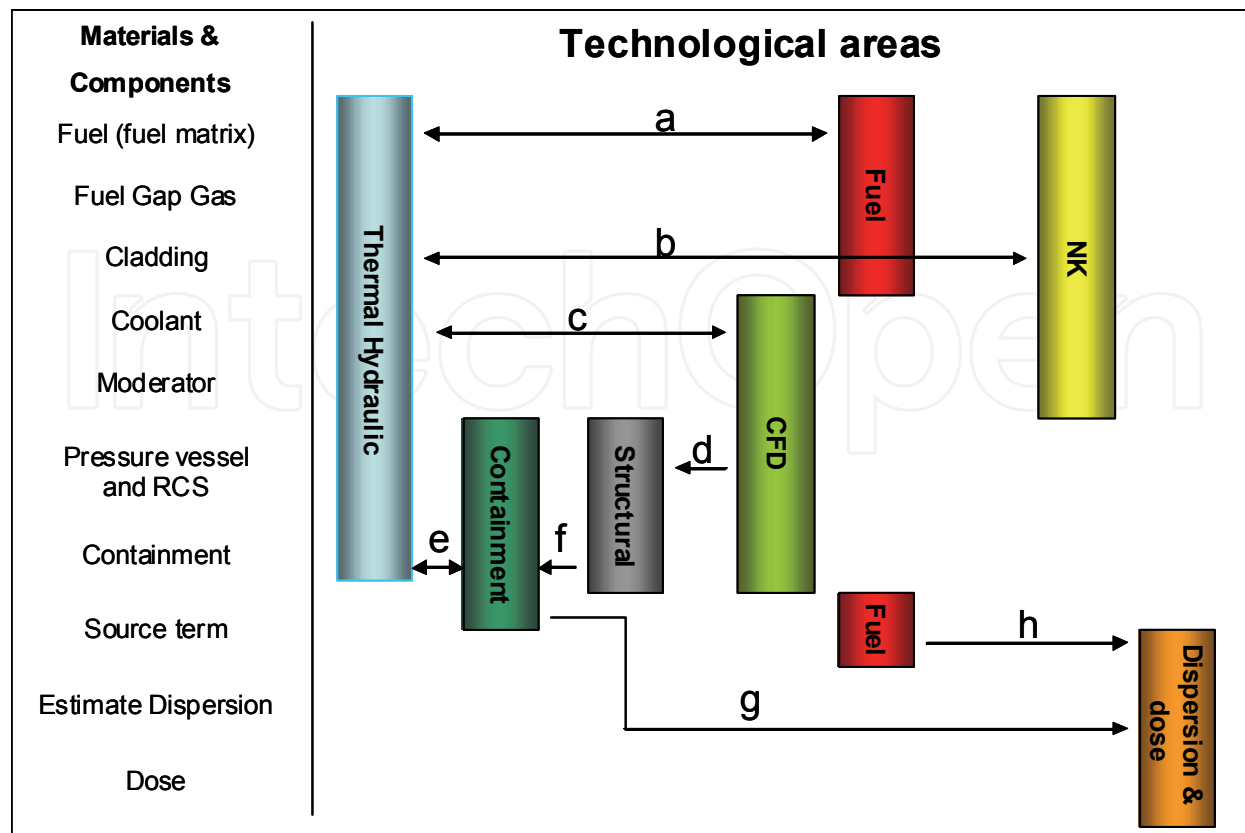


Fig. 2. Links between the different technical areas

2.2 Qualification and uncertainty

A relevant aspect in best estimate application is the qualification of the process of code application:

The following specific topics must be covered:

- Development process of generic codes and their capabilities;
- Developmental Assessment;
- Structure of specific codes
- Numerical methods;
- Description of input decks;
- Description of fundamental analytical problems;
- Analysis of fundamental problems;
- International Standard Problem Activity and benchmarks;
- Example of code results from applications to ITF;
- Plant accident and transient analyses application;
- Modalities for developing the nodalization;
- Description and use of nodalization qualification criteria;
- Qualitative and quantitative accuracy evaluation;
- Use of thresholds for the acceptability of results for the reference case;
- Description of the available uncertainty methodologies;
- Coupling methodologies.

A specific aspect of best estimate application is constitute by uncertainty evaluation (Wickett et al., 1998). For the TH codes specific methodologies were developed and applied.

International literature offers a spread documentation about this uncertainty methodologies for TH codes.

Concerning the codes not connected with the TH area the following items must be evaluated to derive the evaluation of the uncertainty.

- Description of the numerical methods. Generally the codes are validated versus some reference calculations and the related uncertainty is also given.
- International Standard Problem Activity and benchmarks. From the comparison with the result of other qualified codes can be estimated the uncertainty of the code.
- Code application to experimental tests.
- Code application to experimental tests in Plant accident and transient analyses.

Additional and relevant aspects to be also considered are:

- Procedure for developing the nodalization developed by the user or in the code manual.
- Description and use of nodalization qualification criteria.
- User experience

2.3 Computational tools needed in the analysis

The computational tools include:

- the best estimate computer codes;
- the nodalization including the procedures for the development and the qualification;
- the uncertainty methodology including the procedure for the qualification;
- the computational platforms for coupling and interfacing inputs and outputs from the concerned codes and nodalization.

An outline of the codes listed in the table below is provided in the table 1.

No	Field of application	Example of applications
1.	System Thermal-Hydraulics	All transients
2.	I&C Modelling	All transients (where I & C, i.e. control, limitation and protection systems, play a role).
3.	Computation Fluid Dynamics	Special detailed analyses of specific components and/or systems
4.	Structural Mechanics	PTS and structural mechanics integrity of the vessel wall.
5.	Fuel (mechanics)	All transients in relation to which the number of failed rods is calculated
6.	Neutron Physics (and supporting)	Transients analyzed by 3D coupled neutron kinetics - thermal-hydraulics: spatial or local neutron flux effects are relevant - transient conditions.
	Confinement	Severe accident
7.	Radiological Consequences (and supporting)	Environment diffusion and dose tot the population

Table 1. Outline of the codes needed in the analysis

All considered codes should be well established within the international community and some referenced document per each code should be provided that gives access to the peculiarities of the code.

Key issues for the application of the codes are represented by:

- a. the demonstration of the code qualification level;
- b. the demonstration of the current user capabilities in the use of the codes.

The quality demonstration of individual codes, item a), can be derived by several hundred worldwide available documents. In addition to such documents, per each code there are specific-additional qualification documents issued. The reference document provided per each code, gives one access to international qualification documents.

Connected with the above item a), the quality of the code application results is increased by a systematic and comprehensive application of independent codes for deriving the same result. All the codes should be applied by the users, item b), having experience (years) in the code application and results analysis. Code qualification cases shall be considered in order to prove the user capabilities in the application of the codes.

2.3.1 System Thermal-Hydraulics

The quantitative characterization of a system transient scenario constitutes the main role for the System Thermal-hydraulic (SYS-TH) code, consistently with the main objective for its development. The SYS-TH code gives the results connected with the thermal hydraulic parameters evolution of the NPP during a transient. The application of the SYS-TH code, because of the capability to represent all the systems in a quit compact and fast calculations, is typically also used to derive the initial conditions for the application of other more specific codes/tools.

These kinds of codes generally have embedded some additional capabilities:

- The multi-dimensional component in SYS-TH code developed to allow the user to more accurately model the multi-dimensional flow behaviour that can be exhibited in any component or region of a system.
- Neutron kinetic modules: the NK module can have from zero to three dimensions representation capabilities.
- Severe accident module: a limited capability can be included in simulating core damage occurrence and fission fragment distribution in the systems.

2.3.2 I&C modeling

The aim is to simulate the performance of the control, the limitation and the protection systems of the NPP. The simplified representation of the protection system only could be not sufficient for a detailed analysis. The Instrumentation and Control (I&C) can be modelled in the SYS-TH code. But the complexity of the control (also including limitation) systems request a more capable and flexible tool. Some applications have been done just realizing software (e.g. Fortran based software) coupled with the SYS-TH code.

In the I&C software the equations are solved to simulate the transient behaviour of the various transducers, actuators and logic of operation of each individual component that constitutes the control, the limitation and the protection systems of the NPP. The code receives the system information at each time step from the SYS-TH code related to any requested thermal-hydraulic variable (e.g. pressure, level, pressure drop, fluid temperature). The related information is processed, e.g. considering the inertia of the transducer or the

delay of the signal transmission, and commands for components (typically pumps, valves, control rods, heaters, etc.) modelled in SYS-TH are generated. With the new system configuration a new time step is calculated and the above process starts again.

2.3.3 Computational Fluid Dynamics

The main role of CFD is to support and validate the application of the SYS-TH in relation to the mixing phenomena and in calculating pressure drop coefficients at geometric discontinuities where information from experimental data is not adequate. The latter role is also relevant to the PTS study.

CFD features the following modelling capabilities:

- Steady-state and transient flows.
- Laminar and turbulent flows.
- Subsonic, transonic and supersonic flows.
- Heat transfer and thermal radiation.
- Buoyancy.
- Non-Newtonian flows.
- Transport of non-reacting scalar components.
- Multiphase flows.
- Combustion.
- Flows in multiple frames of reference.
- Particle tracking.

2.3.4 Structural mechanics

The structural mechanics code is used to calculate stress and strains in components other than the fuel rods. Two main uses are exemplified in order to summarize the role of the code:

- a. demonstration that dynamic loads, following transient scenarios, do not cause rupture/collapse of or the substantial deformation of the relevant component potentially affecting the coolability of the core;
 - b. calculation of stresses in the components relevant to prevent radioactive releases.
- Typical application is constituted by PTS analysis.

These tools are adopted to perform static and dynamic analyses of linear and non-linear problems (due to materials properties, geometry, contact between surface, etc.) in many fields of application (structural, thermal, electromagnetic, fluid-dynamic, etc.). It is possible to solve coupled problems as well as fluid-structure interaction, thermal-mechanical calculation. In addition several special purpose features are available, namely: fracture mechanics, composites, fatigue, beam analyses.

2.3.5 Fuel mechanics

The key goal for the use of the code is the evaluation of the integrity of the fuel claddings. The number of nuclear fuel rod claddings that are damaged following each transient constitutes the typical output from the code. The code is a computer program for the thermal and mechanical analysis of fuel rods in nuclear reactors. The code was specifically designed for the analysis of a whole rod. Code incorporates physical models of thermal and radiation densification of the fuel, models of fuel swelling, fuel cracking and relocation, a model of generation of fission gases, a model of redistribution of oxygen and plutonium,

and some other physical models. The code has the capabilities of analysis of all fuel rod types under normal, off-normal and accident conditions (deterministic and probabilistic).

2.3.6 Neutron physics

The transient (time dependent) three-dimensional calculation of the neutron flux following global or local perturbations constitutes the main goal from the use of the code. The neutron kinetics subroutines require as input the neutron cross-sections in the computational nodes of the kinetics mesh. A neutron cross-section model has been implemented that allows the neutron cross-sections to be parameterized as functions of SYS-TH code heat structure temperatures, fluid void fraction or fluid density, poison concentration, and fluid temperatures. Additional codes are necessary to (not exhaustive list):

- to derive macroscopic cross sections thus supporting the application of the Nestle code;
- to support and to validate calculation results (fluxes and several reaction rates in each point of the calculation domain and to perform criticality analyses);
- to calculate fuel cell calculation versus burn-up;
- to calculate the build up, decay, and processing of radioactive materials;
- to convert evaluated nuclear data file in continuous-energy or multi-group microscopic cross sections libraries.

2.3.7 Radiological consequences

The purpose is to simulate the impact of severe accidents at nuclear power plants on the surrounding environment. The principal phenomena considered are atmospheric transport, mitigation actions based on dose projections, dose accumulation by a number of pathways including food and water ingestion, early and latent health effects, and economic costs.

Several aspects must be considered:

- Calculation of the radioactivity inventory in the fuel elements.
- Tracking the transport of radioactivity products inside the primary system and the containment.
- Calculating the offsite radioactivity dispersion and the dose to the population.
- Calculating the onsite dispersion and the dose to the control room personnel

2.3.8 Nodalizations

The nodalizations are the result of a brainstorming process by the code-users, which connect each code with the physical system to be simulated. The process for developing a nodalization especially for a best estimate code does not necessarily require less effort than the process of development of the code itself. The same is true in relation to the qualification. Expert users develop the nodalization for an assigned purpose, provided that Best Practice Guidelines are followed whenever available. Sensitivity tests can be performed to demonstrate the nodalization quality and the achievement of mesh-independence of the results, which means that varying the node density (or the number of nodes) does not make the results change to a large extent. All nodalizations shall be developed according to suitable quality assurance procedures and criteria. The procedures are linked with the code characteristics and with the expertise of the users.

All nodalizations developed to apply the BE codes must be qualified according to current standards that are specific for each code. Plant nodalization should be developed according to predefined qualitative and quantitative acceptance criteria.

Three major steps in the process must be distinguished each one characterized by a number of sub-steps, by procedures and by acceptability thresholds:

1. Nodalization development: the nodalization must be characterized by 'geometric fidelity' with the modelled physical systems that are part of the NPP.
2. Acceptance of steady state.
3. The transient capability: the capability of the code-nodalization in simulating the phenomena of interest must be demonstrated

Qualitative and quantitative acceptability thresholds and criteria are adopted at step 1). Quantitative acceptability thresholds are adopted at step 2). Qualitative and quantitative accuracy evaluation is performed for step 3) with quantitative thresholds.

A simplified scheme of a procedure for the qualification of the nodalization is depicted in the figure 3. It is assumed that the code has fulfilled the validation and qualification process and a "frozen" version of the code has been made available to the final user. The steps of the diagram are described below.

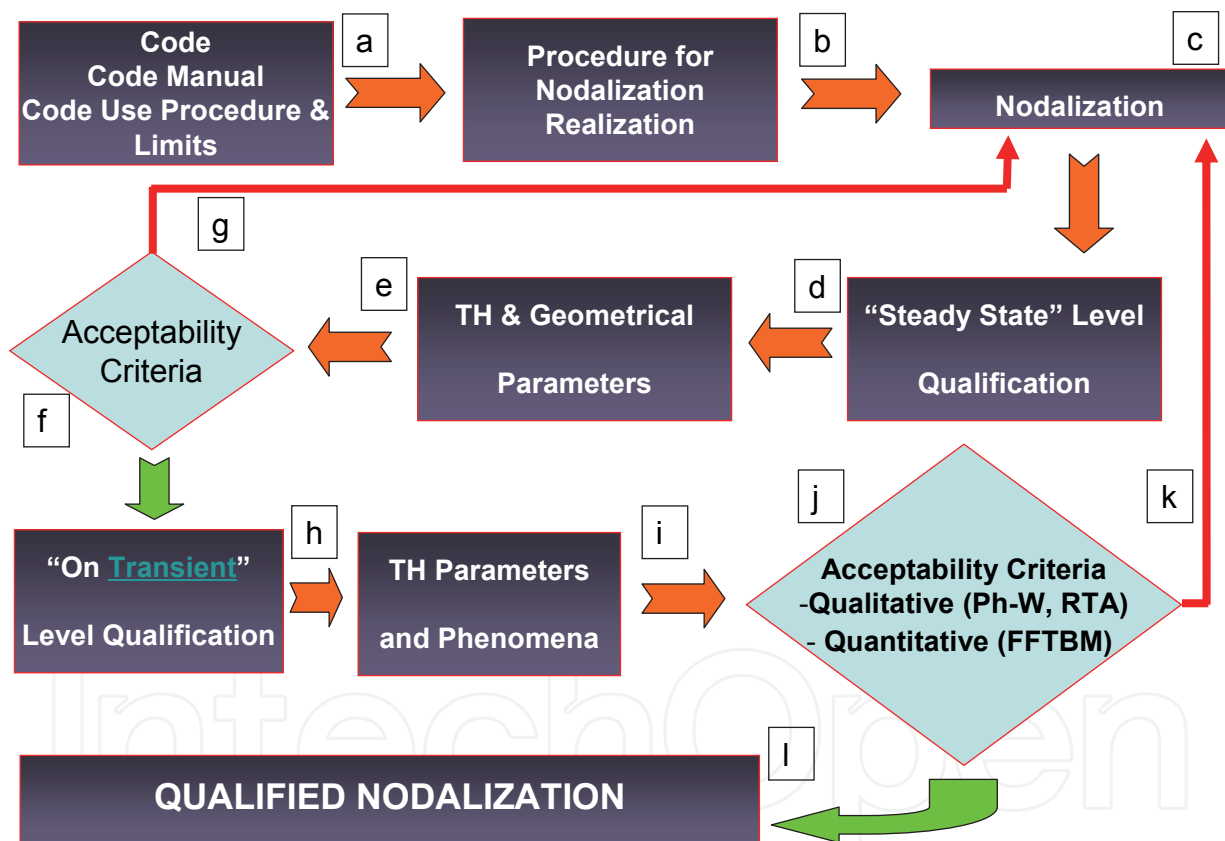


Fig. 3. Simplified scheme for nodalization qualification

Step "a": this step is related to the information available by the user manual and by the guidelines for the use of the code.

Step "b": user experience and developers recommendations are listed and considered.

Step "c": the nodalization must reproduce all the relevant parts of the reference plant; this includes geometrical and materials fidelity and consideration of components and logics.

Step "d": different checks are performed under this step mostly geometry related (does not require running the code-nodalization).

Step "e": different checks are performed under this step.

Step “f”: this is the step where the adopted acceptability criteria are applied to evaluate the comparison between hardware and implemented geometrical values in the nodalization and between the experimental and calculated steady-state parameters.

Step “g”: if one of the criteria in the step “f” are not fulfilled, a review of the nodalization (step “c”) must be performed. The path “g” must be repeated till all acceptability criteria are satisfied.

Step “h”: this step constitutes the “On Transient” level qualification and allows the verification of selected data that are relevant only during transient.

Step “i”: in this step the thermal-hydraulic parameters that are at the basis of the qualitative or quantitative accuracy evaluations are characterized.

Step “j”: checks are performed to evaluate the acceptability of the calculation, e.g. of the ‘Kv-scaled’ calculation both from qualitative and from quantitative points of view.

Step “k”: this path is actuated if any of the checks (qualitative and quantitative) is not fulfilled.

Step “l”: the obtained nodalization is used for the selected transient and the selected facility or plant. Any subsequent modification of the nodalization requires a new qualification process both at “steady state” and at “on transient” level.

3. Example of application: introduction to the analysis of the MPTR

The RBMK core is constituted by more than one-thousand pressurized channels housed into stacked graphite blocks and connected at the bottom and at the top by small diameter (D) and long length (L) pipes (less than 0.01 and more than 10 m, respectively) that end up into headers and drum separators. Control valves are installed in the bottom lines. Due to the large L/D value and to the presence of valves and other geometric discontinuities along the lines connecting with the pressure channels, the Fuel Channel Blockage (FCB) event is possible and already occurred in two documented NPP events. Previous investigations, have shown the relevance of these events for the safety technology, and the availability of proper computational technique for the analysis (NIKIET, 1983 and 1992).

The occurrence of the FCB event remains undetected for a few tens of seconds because of the lack of full monitoring for the individual channels. Therefore, fission power continues to be produced in the absence of cooling. This brings in subsequent times to fuel rod overheating, pressure tube failure, damage of the neighbouring graphite brick and ejection of damaged fuel. Following the pressure tube rupture, reactor cavity pressurization, radioactivity release into the same area and change of fluid properties occur that allow the detection of the event and cause the reactor scram at a time of a few tens of seconds depending upon the channel working conditions and the severity of the blockage.

Notwithstanding the scram and the full capability of the reactor designed safety features to keep cooled the core, the multiple pressure tube rupture (MPTR) issue is raised. The question to be answered is whether the ‘explosion’ of the blocked pressure tube damages not only the neighbour graphite bricks but propagates to other channels causing the potential for several channel failure.

In order to address the MPTR issue fuel channel thermal-hydraulics and three-dimensional (3D) neutron kinetics analyses have been performed, as well structural mechanics calculations for the graphite bricks and rings (graphite rings surround the pressure tube to accommodate for thermal and radiation induced expansions).

The bases for the analysis and the results of the study are presented. The conclusion, not reported within a licensing based format, is that the MPTR consequences are not expected to be relevant for the safety of the RBMK installations.

3.1 Execution of the analysis

The detailed knowledge of the RBMK system configuration was not spread in the Western world till the 1986 event. Afterwards, “information batches” of RBMK technology became available and were unavoidably evaluated in the light of the Chernobyl event. The results of recently completed project sponsored by European Commission (EC), with the participation of RBMK designers in Russia and the supervision of the national utility and the regulatory authority, allow to give an idea of RBMK current safety characteristics. The project has been made possible owing to the availability of sophisticate computational tools developed and qualified in the last decade. These include powerful computers, advanced numerical solution methods, techniques for developing input decks and for proving the qualification level. Following the identification and the characterization of bounding scenarios assuming to envelope all accident conditions relevant to RBMK safety technology, two main chains of codes have been set-up and utilized to perform safety analyses.

3.2 The computational tools

The computational tools include the numerical codes, the nodalizations and the relevant boundary and initial conditions related to the Smolensk 3 NPP in the present case. The application of computational tools requires systematic demonstration of quality and suitable documentation detail. However, within the scope of the performed activity, there is the ‘as-far-as-possible’ demonstration of quality for codes, the development of nodalizations, the implementation of boundary and initial conditions as available and the achievement of results from computer calculations. Furthermore, terms like ‘capable code’ and ‘suitable code’ have been introduced. A code is ‘capable’ when it is able to simulate the phenomena and the physical scenarios expected during the assigned NPP accident. A code is ‘suitable’ when a user can run the code addressing (or calculating) the expected phenomena within a reasonable time with reasonable resources. It should be noted that the term ‘capable’ is less binding for a code than the term ‘qualified’ and a quantification is provided for the items ‘reasonable resources’ and ‘reasonable time’.

3.3 The numerical codes

The numerical codes adopted are those listed in the third column of Table 2.

Identification		Codes adopted	Reasons for the selection
No.	ACRONYM explanation		
A1	LOCA-PH-FIGDH: LOCA in Pressure Header with failure to isolate GDH	Relap5	Largest primary system break with single failure. Challenging core cooling and the ECCS design
A2	LOCA-SL: LOCA originated by a break in Steam Line		Highest depressurization rate. Challenging core cooling and the ECCS design

Identification		Codes adopted	Reasons for the selection
A3	LOOP-ATWS: Loss of on Site Power with the ATWS condition		Challenging core cooling and the neutron kinetics model of the thermal-hydraulic system codes
A4	GDH-BLOCKAGE: Full blockage of the GDH		Check of the capability of the 'ECCS bypass' to cool the core
B1	GDH-BLOCKAGE-SA: Full blockage of the GDH with the 'Severe Accident' assumption of no bypass line available	Cocosys and Relap5	Challenging the venting capability of the reactor cavity (part of the confinement)
B2	LOCA-PH-FIGDH: See A1	Contain and Relap5	Challenging the ALS (part of the confinement) structural resistance (same as A1)
B3	LOCA-SL: See A2	Contain	Challenging the reactor building (part of the confinement) venting capability (same as A2)
C1	FC-BLOCKAGE: Full blockage of one fuel channel	Relap5-3D©/Nestle	Challenging the calculation of the local fission power generation (same as D1)
C2	GDH-BLOCKAGE: See A4		To assess and to understand the local core response (same as A4)
C3	CR-G-WITHDRAWAL: Continued withdrawal of a CR bank (or group)	Korsar-Bars	Challenging RIA (Reactivity Initiating Event)
C4	CPS-LOCA: Voiding (or LOCA) of the CPS	Relap5-3D©/Nestle	
D1	FC-BLOCKAGE: See C1	Relap5-Ansys Katran-U-Stack	Driving accident for the study. Challenging various areas and codes
D2	FC-LOCA: Rupture of one FC	Contain & Relap5 Fluent-Ansys Korsar-Rapta	To assess the ballooning model in the fuel pin mechanics area
E1	FC-BLOCKAGE: See C1	Cocosys Melcor	To assess the hydrogen and the fission products source term and transport (same as B1)
E2	GDH-BLOCKAGE-SA: See B1		To assess the hydrogen and the fission products source term and transport in one extreme conditions (same as B1)
F1	FC-BLOCKAGE: See C1	Relap5	To formulate the ICM proposal (same as D1)

Table 2. Adopted numerical codes

The area for the application of the codes can be deduced from the second column in the same table and from the diagrams in figure. 4 and figure 5 that are applicable for the Russian and the Western codes, respectively. Topological subjects relevant to the deterministic safety analysis of RBMK are identified in Figs. 4 and 5 and the correspondence with the range of application of numerical codes is established.

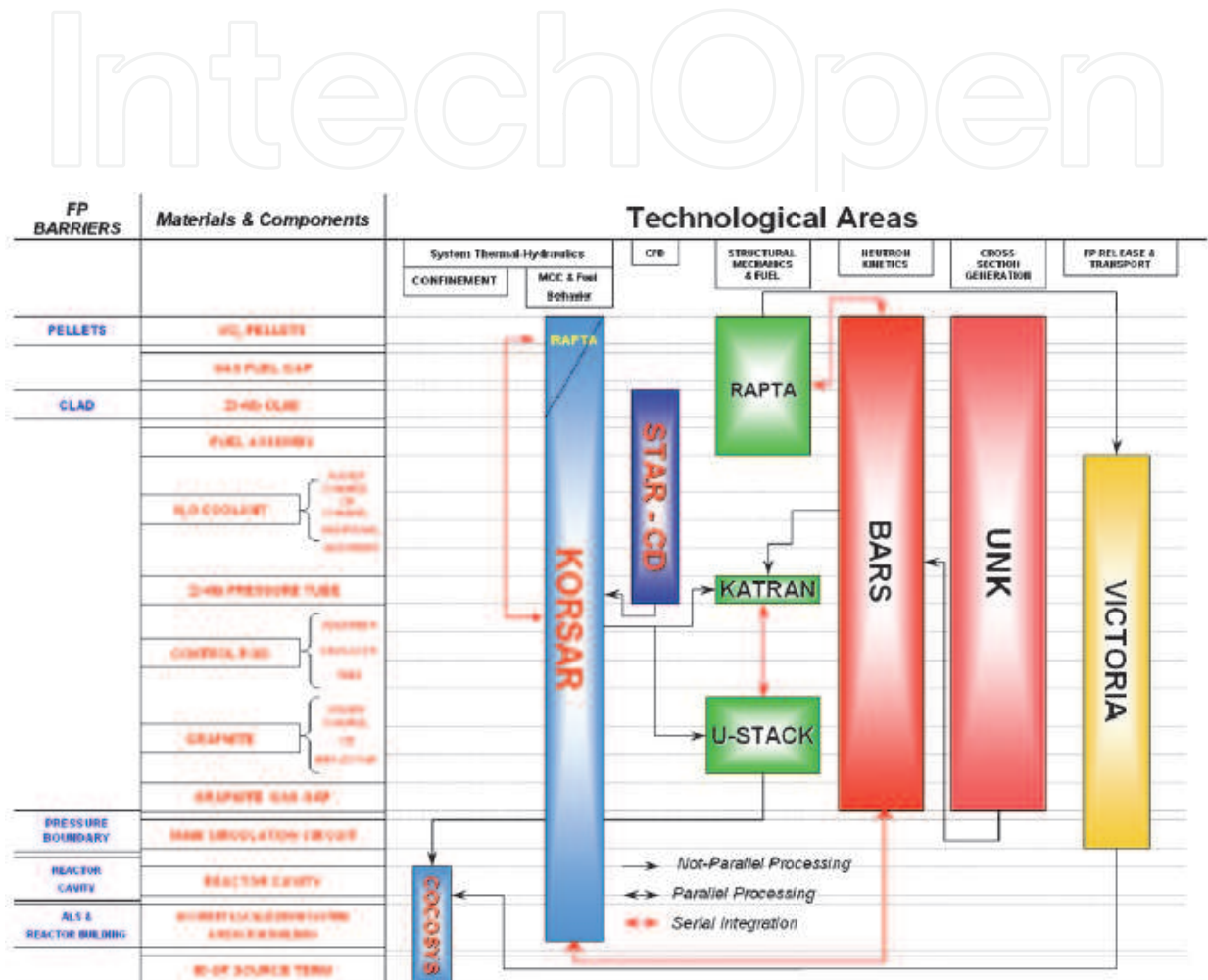


Fig. 4. Codes adopted by Russian group

The topological subjects include:

- Five fission product barriers: the fuel pellet, the clad, the pressure boundary of the primary cooling system and the confinement regions corresponding to the reactor cavity, the (ALS) and the reactor building.
- The materials and components constituting the NPP hardware: the coolant, the fuel and the moderator are examples of ‘materials’; the control rods, the pressure tube and the zones of the confinement are examples of ‘components’.

The technological areas (for deterministic safety analysis) include the system thermal-hydraulics, the computational fluid-dynamics, the structural mechanics, the neutron kinetics with the cross section generation and the fission product release and transport.

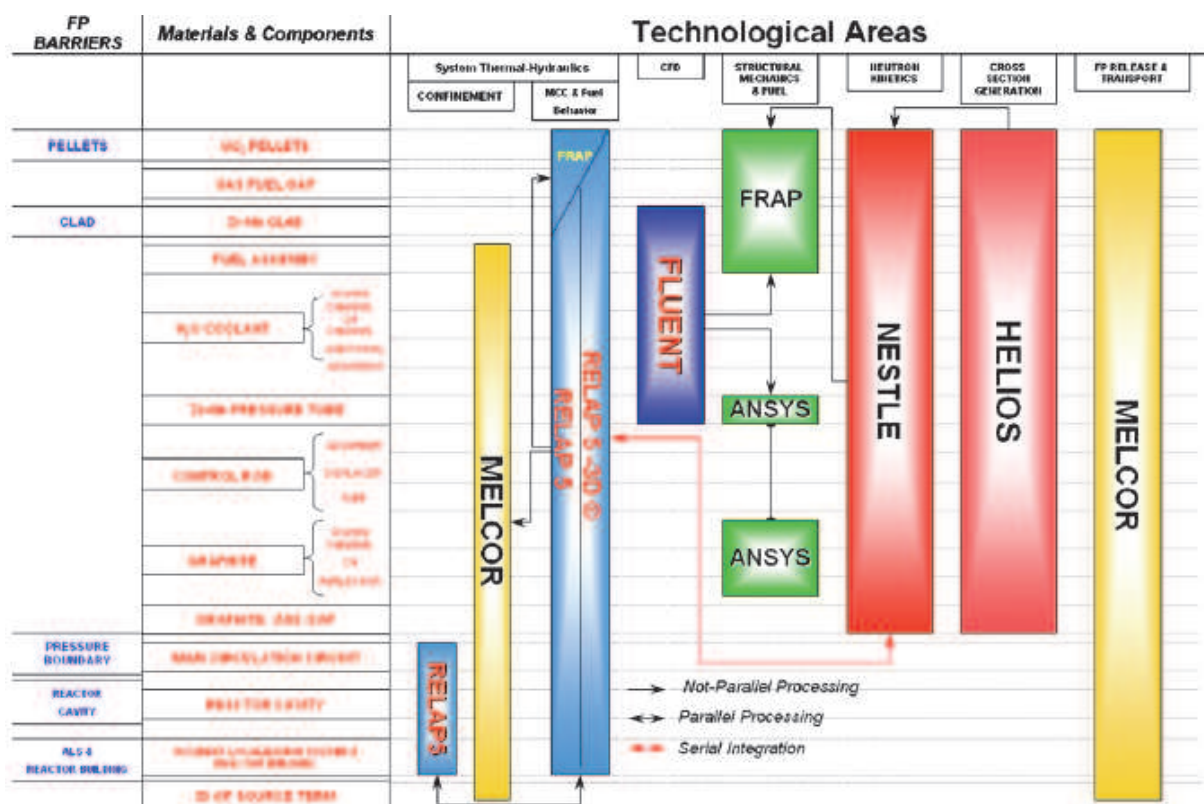


Fig. 5. Codes adopted by western group

3.4 The nodalizations

Nodalizations were developed for both Western and Russian codes by modelling the materials and components, by making reference to the technological areas and by considering the features of codes with the target of demonstrating codes capability and suitability, but also to assess the integrity of the fission product barriers. Nodalizations are typically the result of wide range brainstorming processes whose outcome depends upon the code features, the available computer power, the expertise of the user and the target for the analyses. An example of the realized nodalizations is reported in the table 3.

3.5 The boundary and the initial conditions

Boundary conditions for NPP accident analyses are constituted by huge amount of data ranging from in the present case the mass of water in the steam drum, to the individual fuel bundle burn-up, to the material properties of irradiated graphite, to the thickness and the Young module for the tank that encompasses the graphite stacks, to the free volume of the reactor cavity, to the net flow areas of the valves/openings connecting various zones of the confinement with the environment.

The boundary conditions for the MPTR issue is the accident scenario originated by the fuel channel blockage (FC-BLOCKAGE making reference to boundary conditions in the Smolensk-3 NPP unit.

3.6 The multidisciplinary problem associated with the FC-BLOCKAGE scenario

The background for addressing the multidisciplinary problem arising from the FC-BLOCKAGE and the MPTR include the presentation of following aspects:

No.	System or component	Code	Total number of nodes	Target for the nodalization
1	Main coolant circuit	Relap5	6491/52735*	Thermal-hydraulic performance of the RBMK primary loop in case of DBA and BDBA
2		Korsar	829/5256*	
3	Confinement system	Relap5	621/2625*	Thermal-hydraulic performance of the RBMK confinement in case of DBA and BDBA
4		Cocosys	37	
5	Reactor core	Nestle	29856	Three-dimensional calculation of fission power in the core
6		Bars	17416	
7	Part of a FC	Fluent	~10 ⁶	Calculation of fluid-dynamic forces acting upon fuel rods following FC break
8	FC Inlet Zone	Star-cd	3.7 10 ⁵	Calculation of pressure and velocity fields to assess the appearance of steam cavitation in control valve
9	Broken Single Channel and neighboring FC	Relap5	167/2810*	Calculation of break area and position and 'local' hydraulic parameters for primary system and confinement
10	Individual Pressure Tube and connected Graphite Stack	Ansys	4700	Stresses in pressure tube and in graphite stack consequent to the creep deformation of pressure tube
11	Fuel moderator and coolant	Helios	14 × 4 + 30 × 4 (zones)	To derive macroscopic cross section sets suitable for the use of three-dimensional neutron kinetics codes
12		MCNP	Not applicable	
13	Graphite stack as a whole	U_Stack	–	To address the MPTR issue by the deformation method

* Number of hydraulic nodes and number of mesh points for conduction heat transfer are distinguished.

Table 3. Realized nodalizations

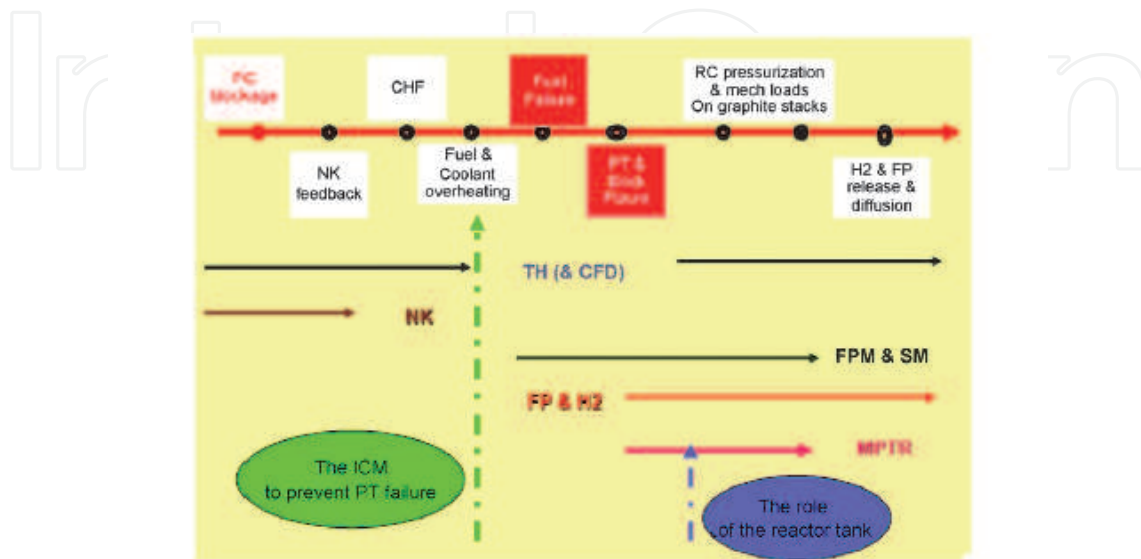
- study of the components and zones of the RBMK core region to make clear the concerned accident scenario,
- the characterization of the steady state operation of the reference RBMK boiling channel,
- the experience from the pressure tube (PT) rupture events in RBMK NPP,
- the phenomenological evolution of the transient.

An overview of the multidisciplinary problem associated with the blockage of one fuel channel scenario is given by the phenomenological aspects associated with the scenario originated by the blockage of one fuel channel in the RBMK NPP (i.e. FC-BLOCKAGE event). To this aim, phenomena are identified that characterize the progression of the event, the failure map for RBMK pressure tubes and the probable position for break elevation following FCBLOCKAGE.

The multidisciplinary nature and the demonstration of complexity for the concerned scenario is shortly highlighted in the figure 6 e table 4 and can be summarized in the following list:

- System Thermal-Hydraulics related to reactor coolant system
- Fuel pin mechanics to evaluate the fuel performance parameters including rod deformation following the FC-BLOCKAGE event in the RBMK fuel bundle.
- System Thermal-Hydraulics related to confinement
- Computational fluid dynamics to calculate the hydraulic loads acting upon the fuel rods following the rupture of the pressure tube occurring during the FC-BLOCKAGE event.
- Neutron kinetics for generation of average parameters for microscopic cross-sections as a functions of energy
- Neutron kinetics: 3D transient neutron flux to calculate the neutron kinetics parameters in the individual fuel channel and associated graphite stack following the FC-BLOCKAGE event.
- Structural mechanics to calculate stresses and strains in the pressure tube and in the graphite blocks following the rupture of the pressure tube occurring during the FC-BLOCKAGE event.

- Fission products generation to calculate the source term associated with the operation of a fuel channel of the RBMK, i.e. the amount of radioactivity that is released during the progression of the FC-BLOCKAGE event.
- Fission products transport to calculate the transport of the fission products generated as a consequence of the melting and the damage of a RBMK fuel bundle during the progression of the FC-BLOCKAGE event.



Nomenclature: CFD= Computational Fluid Dynamics, CHF = Critical Heat Flux, FC = Fuel Channel, FP = Fission Products, FPM = Fuel Pin Mechanics, ICM = Individual Channel Monitoring (see D'Auria et al. 2007e), MPTR = Multiple Pressure Tube Rupture, NK = Neutron Kinetics, PT = Pressure Tube, RC = Reactor Cavity, SM = Structural Mechanics, TH = Thermal-hydraulics.

Fig. 6. Scheme of the multidisciplinary approach

No.	Technological safety area		Codes	Notes
1	System Thermal-Hydraulics	Primary System	Relap5, Korsar	Also includes H2 generation and transport
2		Fuel	Frap, Rapta	
3		Confinement	Cocosys, Relap5	
4	Computational Fluid Dynamics		Fluent	The MPTR issue is addressed by the U_Stack code and by a procedure (given in chapter 6)
5		Structural Mechanics	Katran, Ansys (U_Stack) ^a	
6	Neutron Kinetics	Generation of average parameters (e.g., macroscopic cross-sections or 'λ-functions')	Njoy, Unk, Helios	
7		3D transient neutron flux	Bars, Nestle	
8	Fission Products	Generation	Refp, Melcor	
9		Transport	Cocosys, Melcor	

^a U_Stack code has capabilities to handle technological safety areas 1, 2, 3 and 5.

Table 4. Technical areas involved in the analysis

3.7 Results of the analysis

The scenario puts an enormous challenge to the codes: all key technological areas relevant to the deterministic reactor safety are involved. About 40 phenomena have been identified as characterizing the scenario and related computational tools have been evaluated.

However the possibility for the occurrence of the multiple pressure tube rupture (MPTR) was excluded.

4. Conclusions

The best estimate concept is defined as the efforts in avoiding conservative assumptions in performing analysis. It implies to adopt the best suitable tool available for each specific topic relevant for an analysis. In the case of an analysis related to a like NPP complex system it is necessary to “enlarge” the investigation in many technological areas. A direct consequence is constituted by the adoption of an integrated approach in performing safety analysis. A further relevant consequence is that the best estimate concept must be applied to a broad spectrum of disciplines.

This integrated best estimate approach for safety analysis means the availability of qualified tools and qualified users in many technical areas. The qualification has to be also applied to the coupling of the codes typically organized in a sort of “chain” including not only the code itself but also the input of the codes and the input and output data.

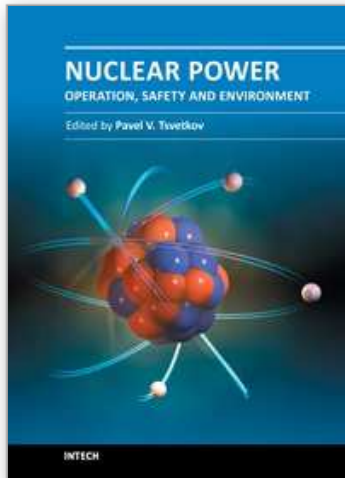
It is also relevant the availability of a suitable computational power necessary to perform the calculation with the different codes. The importance of this aspect is connected with the capability to include in the calculations all the details necessary to obtain a results to be considered as best estimate input for other linked codes.

The uncertainty is another relevant aspects. Best estimate application always requests uncertainty evaluation. The uncertainty evaluation is rather well developed for TH SYS code, but requires a focused and special effort in the case of all the other technical areas.

Summarizing, the best estimate concept applied in analysis for complex systems should be applied as an integrated approach in the meaning of application covering many technological areas and it requests a large effort in terms of technical competences, capability in qualified use of tools and user, and computational power.

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Today's nuclear reactors are safe and highly efficient energy systems that offer electricity and a multitude of co-generation energy products ranging from potable water to heat for industrial applications. At the same time, catastrophic earthquake and tsunami events in Japan resulted in the nuclear accident that forced us to rethink our approach to nuclear safety, design requirements and facilitated growing interests in advanced nuclear energy systems, next generation nuclear reactors, which are inherently capable to withstand natural disasters and avoid catastrophic consequences without any environmental impact. This book is one in a series of books on nuclear power published by InTech. Under the single-volume cover, we put together such topics as operation, safety, environment and radiation effects. The book is not offering a comprehensive coverage of the material in each area. Instead, selected themes are highlighted by authors of individual chapters representing contemporary interests worldwide. With all diversity of topics in 16 chapters, the integrated system analysis approach of nuclear power operation, safety and environment is the common thread. The goal of the book is to bring nuclear power to our readers as one of the promising energy sources that has a unique potential to meet energy demands with minimized environmental impact, near-zero carbon footprint, and competitive economics via robust potential applications. The book targets everyone as its potential readership groups - students, researchers and practitioners - who are interested to learn about nuclear power.

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