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Computer codes in the safety analysis for nuclear power plants. Computational capabilities of thermal-hydraulic tools, using the example of the RELAP5 code

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Abstract

Safety is a paramount concern of the Nuclear Power Program in Poland. To this end there is a need to investigate the design of the proposed reactor and its operation principles and perform multiple analyses both before the reactor start-up (The Pre-Construction Safety Report (PCSR) and during its operational life. In the worldwide nuclear community hundreds of people are involved in this complicated and complex process. Due to the sophistication of the phenomena occurring during operation and accidents, the number of analyses is increasing rapidly. Currently, much interest in this field is focused on the use of computer codes and high computational power.

Keywords: Safety analysis, Neutronic analysis, Thermal-hydraulic analysis, Severe accident, CFD, RELAP5, MELCOR

1. Engineering computer codes for nuclear power applications

In the nuclear power industry the key issue is to assure the safety of the nuclear power plant, with simultaneous consideration given to the economic aspect of its operation. The aim in this context is the safety of the population at large, and electricity production becomes of secondary importance [1]. High standards and negligible radiation doses are demanded for the sake of both the plant personnel and the local community. The safety requirements and radiation limits are becoming more restrictive. Worth noting is the fact that the average dose received by a person due to natural background radiation is about 2.4 mSv/y and the radiation from a nuclear power plant is 0.001 mSv/y [2]. This situation, for the nuclear power industry, results in high demand for specialist analyses that are able to predict with high accuracy the radiation the environment receives from the reactor building.

Safety analysis plays an important role in evaluation of the radiation doses for the particular plant design and its location. The analyses are done over an extended period of time for defined object licensing, pre-construction, operation and decommissioning stages. Power plant safety is described in a quali-

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tative and quantitative manner for distinct issues, for example the result can be defined as the amount of specific isotopes in the core during the period of fuel burnup or the probability of radioactive releases to the environment due to accident sequences. To successfully develop the analysis for a nuclear power plant, the use of computer codes becomes a necessity. Since most of these analyses can be performed in parallel, the codes require high computational power asthis greatly increases the speed and efficiency of the calculations. To perform the calculations for the selected reactor design, there is a need to perform a few hundred simulations to prove that it fulfills requirements. The safety requirements are defined through: fundamentals, general requirements and specific safety requirements, which were formulated by the International Atomic Energy Agency (IAEA) and are considered worldwide safety standards.

2. Computational tools

There are a few significant groups of computational tools used for the purposes of safety analysis and they can be divided into: neutronic, structural, thermal-hydraulic and severe accident types. The safety analyses are conducted by three independent institutions: nuclear power plant utility, national regulatory body and the Technical Support Organisation—TSO. There are a large number of commercially available codes dedicated to safety analyses and it is a common practice to compare them, thereby leading to a fuller understanding of the phenomena present during the operation of a nuclear power plant. The comparative process is called benchmarking, and forms part of their validation and verification process. In addition, the validation and verification process involve experimental tests modeling followed by a comparison of the results with the real experimental data. On the basis of this kind of actions, the party responsible for developing the code is able to decide that the code is ready for commercial distribution. Traditionally, use of a commercial code application is connected with the necessity of purchasing a license which strictly defines the possible scope of applications and restricts the user – developer relationship. There is also a second type of license - a type of certificate - which specifies that the particular code is approved by the national regulatory body for safety analysis purposes.

2.1. Neutronic codes

Neutronic codes focus their applicability on phenomena that are present in the reactor core. They are capable of predicting reactions between the atoms of materials in the reactor pressure vessel and neutrons or radiation particles, which are the constituents of the fission process (α,β,γ). Througn those codes, users are able to evaluate the most important parameters, from the reactor physics point of view, such as the multiplication factor, neutron flux, isotopic changes in the core fuel or the fuel burnup. Due to the complexity of the reactor core design (3D), complexity of the calculation process, and the strictly related lengthy calculation time, supercomputers with high computational power play an important role.

Neutronic codes can be divided into a number of types, depending on the problem solving method: tdeterministic codes or the Monte Carlo method—based on the probability density functions. Neutronic codes are dedicated to different types of applications, and examples of use are:

1. creation of homogenous multigroup constants for deterministic reactor power simulation,
2. investigation of the fuel cycle, consisting of continuous analysis on the level of the fuel assemblies,
3. codes validation, which solve the neutron transport equation for the fuel assemblies,
4. neutronic calculations and fuel burnup for the research reactors on the whole core level,
5. educational and demonstration purposes, showing the physical phenomena present in the reactor core.

Neutronic codes, in various methods, are created to solve the neutron transport equation. This equation is a balance equation for the production and destruction of neutrons, through the absorption of neutrons in materials or escape of neutrons from the analyzed domain. Examples of the available codes for commercial use, which solve the transport equation through Monte Carlo method, are: Serpent code developed by VTT Technical Research

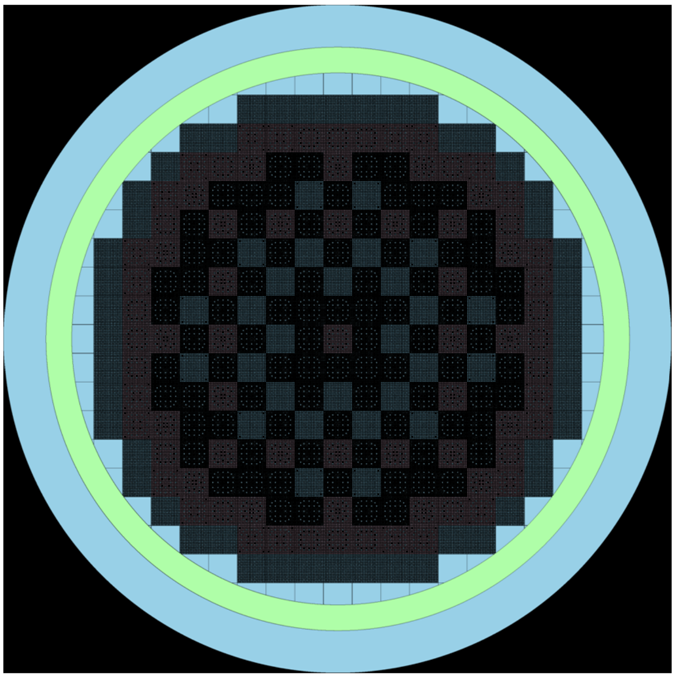

Figure 1: Core cross section analyzed using the Serpent code

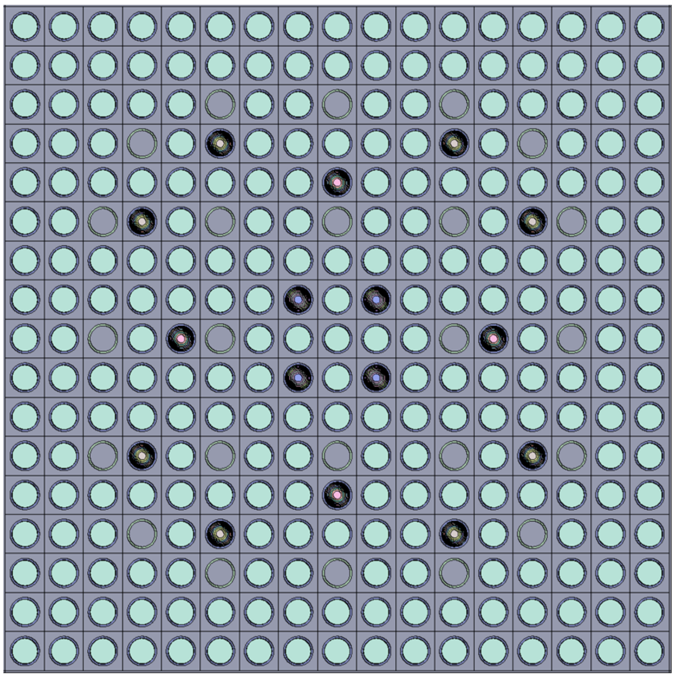

Figure 2: Fuel assembly cross section analyzed in the Serpent code

Centre in Finland, Monte Carlo N-Particle transport Code—MNCP developed by LANL (with extended version MNCPX) and KENO developed by ORNL code for criticality calculations. The deterministic codes are DRAGON 4 created at the Ecole Polytechnique de Montreal, WIMS developed by UKAEA and APOLLO-2 created jointly by the CEA, Farmatone and EdF.

2.2. *Structural codes*

Structural codes are responsible for predicting fuel behavior during reactor operation in various scenarios. Particular applications of the codes are:

1. calculations of the fuel rods behavior during irradiation
    (a) during constant irradiation,
    (b) in transient states,
    (c) evaluation of the source terms for accident analyses,
2. R&D applications,
3. fuel rod design,
4. design of new products and fuel cycles,
5. supporting fuel loading into the reactor core.

Most important for the structural code is its role in predicting events and phenomena. The phenomena which occur during normal and accidental fuel irradiation are: formation of oxides, variation of the fuel pellet temperature distribution, heat accommodation, cracks, porosity and fuel grain distribution. Additionally, during reactor operation, fission products are created and they constitute major heat sources in accident conditions. This provides the rationale for evaluating exact isotope concentrations in fuel, fuel pressure and possibilities of fuel structure damage followed by cladding failure.

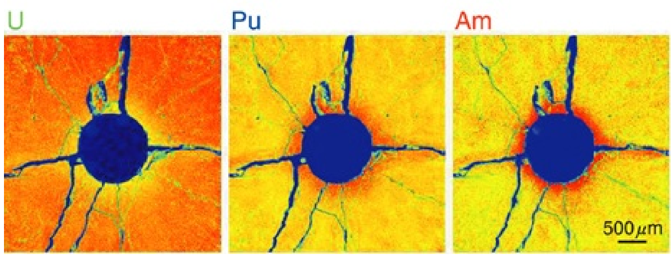

Figure 3: Elements distribution in the 24 hours irradiated %5Am-MOX fuel [3]

Structural codes used for fuel behavior prediction, are very detailed, and they are continuously increasing in number. Almost of the institutions focused on nuclear technology are developing their own structural code, for internal use. Those codes are usually designed for specific reactor types, distinguished by their core configuration or fuel rods design. Some examples of the codes used in research institutions and for licensing purposes are: ENIGMA—UK for reactors like PWR, FRAPCON, FRAPTAN—developed by US NRC for BWR and PWR for licensing and benchmarking, TRANSURANUS—ITU Germany used for fuel R&D and COSMOS—KAERI South Korea for fuel performance calculations.

2.3. Thermal-hydraulic codes

For the purposes of the safety analysis for nuclear power plants those tools are used as crucially-important codes. Analyses done on the fundamentals of the computer simulation results are analyses of the steady and transient operational states. The aim of those analyses is to indicate if the analyzed nuclear object with the available safety systems is able to withstand an accident sequence and what the potential consequences of an accident are, along with a related timescale. The simulation results play a key role in designing, licensing and operating the nuclear power plant. The codes are required because nuclear power plant systems work at a highly-sophisticated level that surpasses the capabilities of the human mind and simple, basic theoretical models. Safety analysis relies on conservative principles and requirements for system design and operation. Meeting those requirements ensures a high reliability level, stating that the risk associated with plant operation for workers and society is reasonably low [4]. With the increasing quality of data and the models implemented, it is possible to create a more realistic thermal-hydraulic analysis, which uses data from probabilistic codes to choose the most probable accident scenario.

The thermal-hydraulic codes used for safety analysis need to be adequately verified and validated. Verification describes the accuracy of the translation of physical equations to the computer code language. Validation determines the correctness of the mathematical models, which have to be a realistic representation of the system. Validation is usually performed by comparing the results obtained from the model and experiments. The validation process shows uncertainties and inaccuracies in models, which need to be taken into consideration later in the safety analysis process [4].

In thermal-hydraulic codes the aim is to determine modeled system parameters, in terms of fluid mechanics and heat transfer of the materials in the analyzed domain. Due to the precision of the calculations, there are four basic analysis scales: system, components, CFD and micro scale, which differ by the manner of division of the domain into calculation cells.

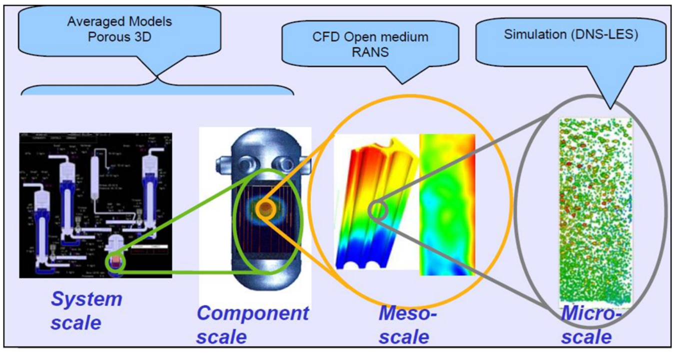

Figure 4: thermal-hydraulic codes for various analysis scale

System scale analysis focuses on the entire nuclear power plant design and usually the model represents key components of the reactor configurations like: steam generators (PWR), pressurizer (PWR), reactor pressure vessel, pumps and safety systems connected to the core cooling circuit. System scale codes are able to predict the overall response of operational events. The results of those simulations are the values of relevant system parameters such as: pressures, coolant, control volumes and temperatures of materials in the modeled structures as a function of time. The codes enable one to simulate the operation and behavior of the reactor, in particular the accident sequences, in order to evaluate the safety level of the nuclear power plant.

Simulations, performed using CFD codes, are midscale codes which are focused on finer discretization of the analyzed domain. As the division is larger and reaches millions of computation cells, the possible domain to analyze is decreased. The meshing of the simulated region is one of the more important aspects of the CFD analysis, because it greatly influences the results. What is more, in CFD analysis the emphasis is put on the turbulence presence during the fluid flow. Codes of this scale implement various turbulence models like Reynolds Averaged Navier Stokes or Large Eddy Simulation. They successfully allow one to calculate fluids mixing in the downcomer channel and the range of turbulence flows.

Examples of thermal-hydraulic codes in system scale are:

1. Relap5—The Reactor Excursion and Leak Analysis Program, which is a tool for the Loss of Coolant Analysis purposes and other transient sequences for PWR and BWR reactors. The code's applicability and capabilities focus on thermal-hydraulic phenomena in 1-D control volumes. The code was created and developed by the US NRC before the initiation of work on the TRACE [5] code. The code does not give multicore calculation possibilities.
2. CATHARE—The Code for Analysis of Thermalhydraulics during an Accident of Reactor and Safety Evaluation is a code for safety analysis purposes, accident management, operational procedures definition and reactor technology development. The code is also applicable to the determination of conservative margins in safety analyses and for the licensing of nuclear reactors. The code is able to run calculations on multiple computer nodes. CATHARE is the result of the joint work of AREVA_NP, CEA—French Energy Commission, EdF—nuclear power plants utility and IRSN—Nuclear Safety Institute [6].

CFD computer codes are represented by: NEPTUNE CFD—A New Software Platform for Advanced Nuclear Thermal Hydraulics—EDF, CEA, IRSN and AREVA, TRIO_U—CEA, ANSYS Fluent, OPENFOAM—developed by OpenCFD Ltd—open source code and TransaT—Transport phenomena Analysis Tool created by ASCOMP GmbH.

2.4. Severe accident codes

Severe accident codes are used to determine the behavior of the nuclear power plant during beyond design basis accidents, involving significant core degradation. A severe accident is considered as: accident conditions with more serious results than a design basis accident, leading to the significant core degradation and possible release of the radiation to the environment at levels above authorized limits.

This definition takes into consideration the most significant elements of a severe accident, which is the core material melting and threat of damage to physical barriers . According to the International Atomic Energy Agency severe accident conditions were moved to design basis conditions, which implies the necessity to take the core melting event and associated consequences into consideration for newly-built nuclear plants. The codes that are at present used for severe accident simulations reproduce phenomena of:

1. thermal-hydraulic behavior of the fluid and aerosols in the containment volume,
2. fission product behavior in the cooling circuit,
3. core melt progression,
4. ejection of the molten corium outside of the reactor pressure vessel,
5. interactions between corium material and concrete in the containment (mcci),
6. release of radiation to the environment.

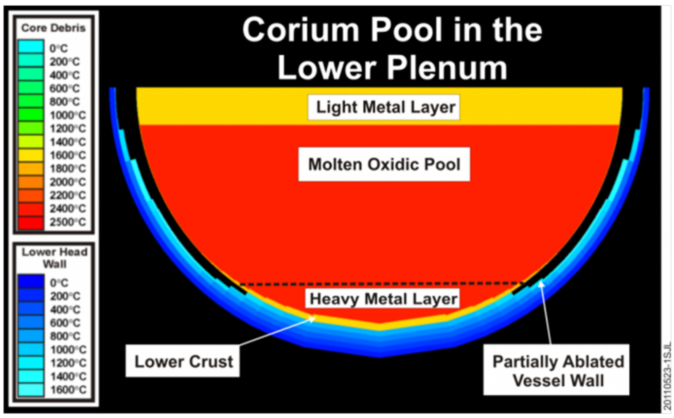

Figure 5: Model of the corium molten pool formation in the lower head of the RPV [7]

Severe accident analyses can be used for various purposes, in some cases they form part of the risk analysis for the nuclear power plant or are a required part of the safety analysis report under regulations of the national nuclear regulatory body. Phenomena present in severe accident conditions are very complex and their mechanisms and physics have still not been fully investigated. Accordingly, there is a degree of uncertainty associated with each of the severe accident analyses, which later has to be addressed and evaluated for each model created. Codes used for severe accident simulations have built-in modules specifically dedicated for sensitivity and uncertainty models.

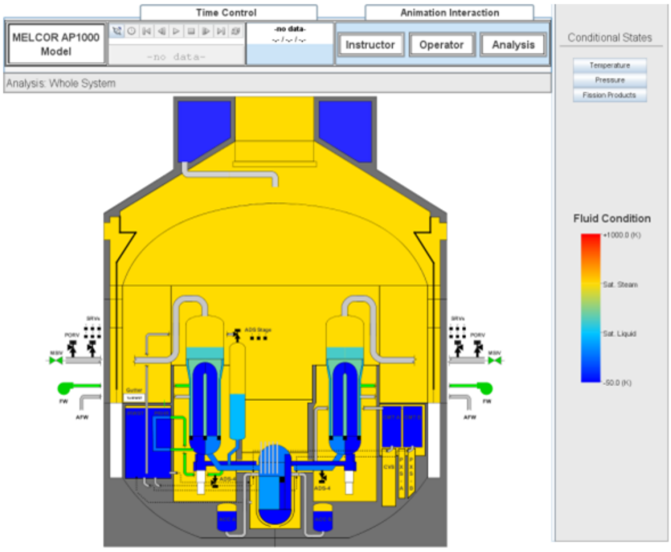

Figure 6: Simulator in the graphic interface connected to the MELCOR code [8]

Types of severe accident tools:

1. Integrated fast running codes:
   (a) MELCOR—developed by SNL, MAAP Electric Power Research Institute, ASTEC—created by IRSN and GRS.
2. Specific phenomena codes, primary coolant circuit analysis: ICARE/CATHARE, ATHLET_CD. RELAP/SCDAP:
   (a) Containment codes: CONTAIN, COCOSYS.
3. Specific integral code sets: SAMPSON, RELAP/SCDAP—CONTAIN—VICTORIA, ATHLET-CD—COCOSYS.

3. RELAP5 code

RELAP5 is a system code used for thermal-hydraulic calculation in Light Water Reactors (LWR). The code allows one to simulate steady states and transients, i.e., Loss of Coolant Accident (LOCA), Loss of Off-site Power (LOOP), turbine trip and Loss of Flow Accident (LOFA). It was developed in the United States by Idaho National Laboratories for the American Nuclear Regulatory Commission (NRC) for licensing issues in nuclear plants and related facilities. The code simulates almost all design basis accidents, which do not exceed the core melting event.

Mass and energy flow is calculated by one-dimension mass flow flux in pipe and via a conductivity model. The code contains many specific component models used in nuclear modeling, like point kinetics, electric heaters, jet pumps, turbines, separators, accumulators and control system logic. Basic components that the nodalization model can consist of are listed below:

1. Time Dependent Volume/Junction,
2. Single Volume/Junction,
3. Pipe,
4. Branch.

3.1. Nodalization

The model in the RELAP5 code is developed by defining connected elements by junctions, which are later called the nodalization. It corresponds with liquid flow paths, which we visualize by code elements. In each node the parameters for a particular phase of flow are calculated separately, solving equations for momentum, mass and energy conservation. The results obtained from one node (downstream) are used as boundary conditions for next node (upstream).

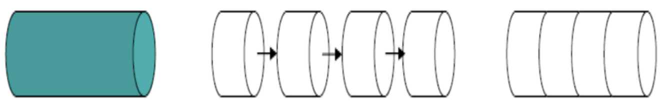

Figure 7: Example of RELAP5 nodalization technique

Appropriately completed nodalization of the model allows one to obtain adequate results, therefore it is important to devote much more time to preparing the model and to use guide requirements and experience gained during complex thermal-hydraulic systems modeling. The potential complexity of the created system should be taken into account. In RELAP5 one of the limitations is that the number of elements cannot exceed 999. Volumes should be fitted, in that way, to reflect the sizes of

the elements (flow area, elevation). Consideration should also be given to potential phenomena that will take place during the transient simulations.
Smaller volumes make the calculation time longer, but it allows one to describe the modeled system better. The length of neighboring volumes should not be twice longer as too large a volume can be overly simplified and introduce high uncertainties. In long parallel elements (such as the as riser and downcomer) in the reactor pressure vessel, steam generator or heat exchangers, the development of nodalization should use the slices method as a basis. This means that parallel volumes are divided on the same elevations and have the same length, as is shown on Fig 7. Development of the heat structures (passive and active) in the model should be implemented in accordance with this method—one control volume corresponds to one heat structure. In the case of complex system modeling, it is recommended to divide the model into separate components - a reactor pressure vessel, steam generator, pressurizer etc. RELAP5 beginners should learn alongside more experienced users who have used the NRC code for many years. RELAP5 is a deterministic code and it does not assess the results obtained—it is the user that is responsible for evaluating the results.

3.2. Heat structures

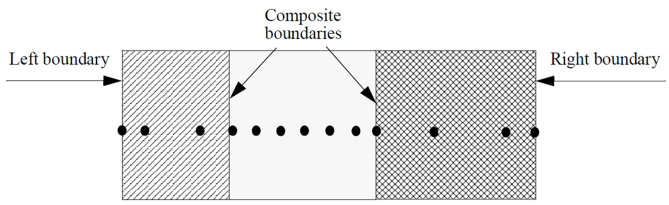

Figure 8: Mesh point layout in RELAP5 material model

Heat structures are applied in the model, always when heat transfer will be simulated in the analyzed domain. Through the use of heat structures many elements can be simulated, i.e., a channel's wall, electrical heater, nuclear fuel rod or heat transfer surface in a heat exchanger. Heat structures represent selected, solid portions of the thermal-hydrodynamic system. In solids, where there is no flow of mass and total system response is evaluated by heat transferred between the structures and the fluid, the temperature distributions in the structures are often an important indicator of the accident simulation. The conductor is connected to a particular control volume, as appropriate to the left and right boundary of the volume, giving direction to the heat flow in the material. The code calculates heat flux using suitable correlations according to conductivity law. The power of the electrical heaters or fuel rods can be modeled by surface heat flux or by volumetric power sources. On Fig. 8 it is shown that a wall can consist of more than one material, located between lthe eft and right heat structure boundary.

Table 1: Heat transfer modes

Mode	Heat transfer correlations
0	Convection to noncondensable-water mixture
1	Single-phase liquid convection at supercritical pressure
2	Single-phase liquid convection, subcooled wall, low void fractions
3	Subcooled nucleate boiling
4	Saturated nucleate boiling
5	Subcooled transition boiling.
6	Saturated transition boiling
7	Subcooled film boiling
8	Saturated film boiling
9	Single-phase vapor convection or supercritical pressure with the void fraction > zero
10	Condensation when the void is less than one
11	Condensation when the void equals one

The density of the mesh points have to be declared depending on the points of interest for the user, where results like: temperature, heat flux, heat transfer coefficient etc., can be verified. On the heat structure boundaries (depending on the previous declaration) it is possible to read heat flux, heat transfer coefficient, average temperature in the control volume or correlation which is used for heat transfer calculations. The correlations cover the various modes of heat transfer from a surface to a fluid, and the reverse heat transfer from fluid to surface. Since the correlations have specific names (Table 1), each of them corresponds to a number, which is easily written in the output file.

For some purposes a symmetry or insulated conditions boundary can be specified. At this surface there is no heat transfer and the temperature gradient is zero. This boundary condition is often used in cylindrical domains (fuel rod model), when the left-mesh model point is zero. Such application reduces calculation time by simplifying the model.

3.3. Work with the code

When starting to use the RELAP5 code iyou must familiarize yourself with the architecture, which is described in the Input Manual. The main structure consist of cards and each of them comprises words separated by a space. We can distinguish the words written in the form of a real number, integer or alphanumeric type.

```
1850000  inlet  tmdpjun
1850101  180010000  200000000  0.00781
1850200  1
1850201  0.    100.0   0.0   0.
1850202  500.  200.0   0.0   0.
```

Fragments of the input file are shown above. The first three characters of the zero word in each card is a component number (in this case the component 185) and the last four describe the type of card. The first and second word correspond to the name of the component and component type respectively. In the case of the junction (tmdpjun), in subsequent cards the connected components numbers are given with the flow area of the junction in square meters. Card 1850200 is responsible for the values entered in 185020N, defining whether the entered values will be implemented in the mass flow [kg/s] or in flow velocity [m/s] units. Selected value 1 means the mass flow rate, which will be changed over time from 100 kg/s to 200 kg/s for 500 seconds.

The next characteristic component is the reservoir of the emergency core cooling—accumulator (accum). Seemingly, the component appears to be a single volume control, but the code allows for more accurate description of the system. In reality it is quite a complex tank, where the water is pressured by a nitrogen pillow of pressure about 4 MPa. The tank

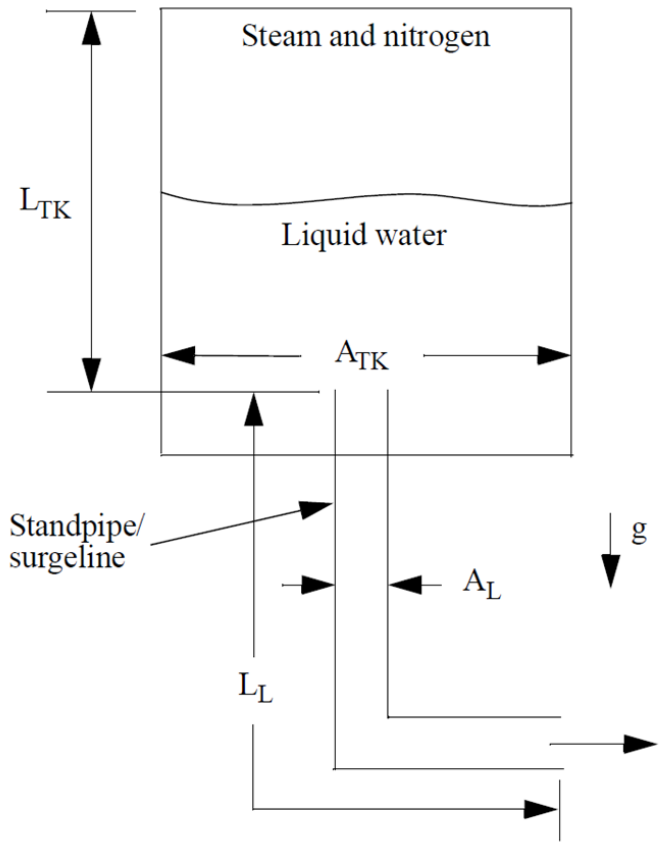

Figure 9: Schematic of a cylindrical accumulator with main components indicated

is connected to the primary side by a standpipe and surgeline—as is shown on Fig. 10. The accumulator is cut off by a check valve, so water does not flow from the primary side to the tank. The accumulator starts to work passively only when the pressure in the primary circuit drops below the pressure inside the accumulator. It does not require electricity and water is automatically delivered to the reactor core for cooling. As long as the tank and surge line are not empty, the accumulator is described as a lumped parameter. When the accumulator has no more water, the code automatically changes the properties of the volume from accum to a single volume snglvol.

The location of characteristic dimensions such flow area and length of volumes are shown on Fig. 9. Descriptions of the tank, standpipe and surgeline have to be properly implemented to avoid error in quantity of available water. Also the junction connecting the accumulator to the primary side has to be described correctly and can be connected only with one of the control volumes. In the next step the shape of

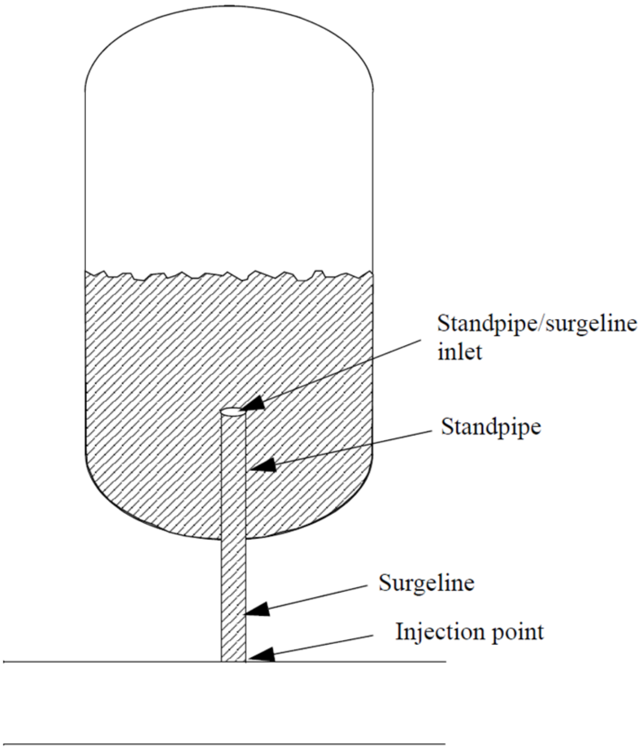

Figure 10: Schematic of an accumulator showing standpipe/-surgeline inlet

the accumulator is described (cylindrical or spherical) and the amount of the whole volume reserved for gas is stated. Each component of a defined volume has to contain information about the orientation. In this case, the user can change the vertical orientation by connecting the standpipe from the bottom or from the top of the accumulator. If the geometry is described, the initial conditions within the tank have to be specified, i.e., temperature, pressure and properties of the wall tank (thickness density, heat capacity).

The finished input file can be launched through the command line and the calculation will start if the input has no errors. After it starts the user gets an output file and restart file. The first one is recorded in text form and contains information about the processed input file and parameters calculated for particular components. If an error occurs during input processing, an appropriate phrase must be found in the output, and the nature of this error must be understood and improvements made. There is also a possibility for people who do not like to use text form files. A model can be prepared using the professional tool SNAP (Symbolic Nuclear Analysis Package), which can be helpful during modeling because it contains a graphical user interface (GUI). This tool reduces the probability of syntax errors or incorrect connections of the control volumes.

4. Summary

Calculation codes play a very important role in the nuclear industry. The growing safety requirements for nuclear reactors and the need for more exhaustive research into the phenomena occurring in them are leading to the use of computational tools that are able to solve many parallel equations. For that purpose high performance computers are being built, so that the calculating time can be greatly decreased.

In the safety analysis of nuclear power plants, many areas should be considered for the purposes of making a comprehensive assessment. The knowledge needed to fully understand phenomena is very wide and covers such issues as: physics of neutrons, materials, solid mechanics, fluid mechanics and heat transfer. For that reason computational codes are dedicated to specific areas and phenomena such as thermal-hydraulics, severe accidents or neutronics as well as results from another analysis can constitute input data for other calculations.

Thermal-hydraulic codes define the behavior of coolant in a nuclear reactor during emergency scenarios and normal operation. Many calculation tools can be distinguished depending on various scales, ranging from the micro scale, illustrating the turbulences at the micrometers up to the system codes, which evaluate the whole power plant with safety systems. Thermal-hydraulic codes are used to prepare a deterministic safety analysis, which is part of the safety assessment process of a nuclear plant.

One tool dealing with thermal-hydraulic calculations is the RELAP5 code, which is widely used in research and development for reactor technologies. It is used for boiling and pressurized water reactors, it can simulate loss of coolant scenarios or offside power up to the overheating and melting of the core. Using the code requires long term training, and the most demanding element of use is the ability to assess the correctness of the results. Currently, RELAP5 is no longer being developed, but in October 2011 the INL took the decision to start a new project, RELAP-7, which will replace the existing version and will become the main tool for safety simulations of reactor systems.

Acknowledgments

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