Development of a software-based system for modelling research reactors using heuristics

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ABSTRACT

The present paper describes the development of a new method for fast modelling of selected research reactor types. Due to high diversity of research reactor designs, a rule-based software system is engineered to support the modelling for deterministic safety analysis utilising the system code ATHLET. The use of heuristics rules allows an adequate input deck generation despite limited data. This includes e.g. core layout and material data as well as neutron kinetic data for 1D representation of the core. Due to the knowledge-based implemented methods, the software is capable to generate the fundamental input deck automatically. The software is currently designed for selected research reactor types – namely MTR and TRIGA reactors – but possibilities for extensions are considered. Within this paper the new modelling strategy and its capabilities are highlighted. Therefore, examples of implemented methods using heuristics are described. Finally, first functionality is demonstrated by showing an exemplary research reactor generated by using the new modelling system. Preliminary simulation results of a loss of flow accidents are compared to experimental data.

1 Introduction

Evaluating safety of research reactors, deterministic safety analysis using thermal-hydraulic system codes are an important instrument to ensure safe utilisation during postulated initial events. The main group of safety analysis codes are based on the solution of partial differential equations – mass, momentum and energy balance. Using predominant one-dimensional approximations, the equation matrix is solved by finite element (e.g. RELAP) or finite volume approach (e.g. ATHLET). Therefore, the simulated system is discretised in space into a net of control volumes connected by junctions. This approach allows a wide range of code application due to free thermal-hydraulic nodalisation. Though, high responsible is transferred to the user presuming detailed code knowledge and complete plant descriptions to develop an adequate nodalisation scheme of the nuclear facility. Taking into account multidisciplinary phenomena, which can occur during accident conditions, the codes offer additional modules simulating i.a. heat transfer and neutron kinetics. In the system code ATHLET (Analysis of thermal-hydraulics of leaks and transients) – developed by GRS (Gesellschaft für Anlagen- und Reaktorsicherheit gGmbH) - the main modules are thermal fluid dynamics (TFD), heat transfer and heat conduction (HECU), neutron kinetics (NEUKIN) as well as plant control (GCSM). The user has to choose adequate input options out of a wide range of possibilities for each module. Consequently, modelling a whole plant system takes a large amount of reliable data and human resources.
Analysing foreign research reactors, technical support organisations and research institutes might be confronted with limited available information of plant data. In case of emerging safety related questions, the complex input data structure of safety analysis codes impede a fast response. To improve the nodalisation process and input data implementation, new

software is being developed in the present work. The fundamental elements of the input deck are generated automatically by few input data necessary. Hereby, the user is supported by making ad hoc decisions in the case of lack of appropriate data and time.

2 New Strategy of modelling research reactors

Heuristic methods can be used to achieve an appropriate modelling quality of research reactors despite incomplete data. To accomplish the goal, three main steps are identified covering the new strategy of modelling:

- Abstraction and modularisation of research reactor plant designs
- Concept of nodalisation
- Development of process for automation

The strategy for modelling research reactors is extensible to a wide range of safety analysis codes. For first application, the system code ATHLET was used. In this chapter, the sequential main steps of the modelling strategy are discussed. The main characteristics are highlighted concentrating on MTR design of research reactors.

2.1 Abstraction and modularisation of research reactor plant designs

Compared to nuclear power plants, research reactors have a wide range of designs and operation modes due to their different applications in the field of science, technology and medicine. Realising a heuristically process for research reactor modelling, a restriction in types covered have to be done. To date, 218 research reactors are operated around the world /RRDB2017/. The TRIGA (Training, Research, Isotopes, General Atomic) and MTR (Material Testing Reactors) reactors represent the most widely installed research reactor types. About 25 % of the research reactors are of MTR type and 21 % are of TRIGA design /RRDB2017/. Consequently, these types are selected as a model design basis. To reduce further design variety among MTR reactors, the main designs are abstracted to open core and tank-in pool reactors as pictured in Figure 2-1.

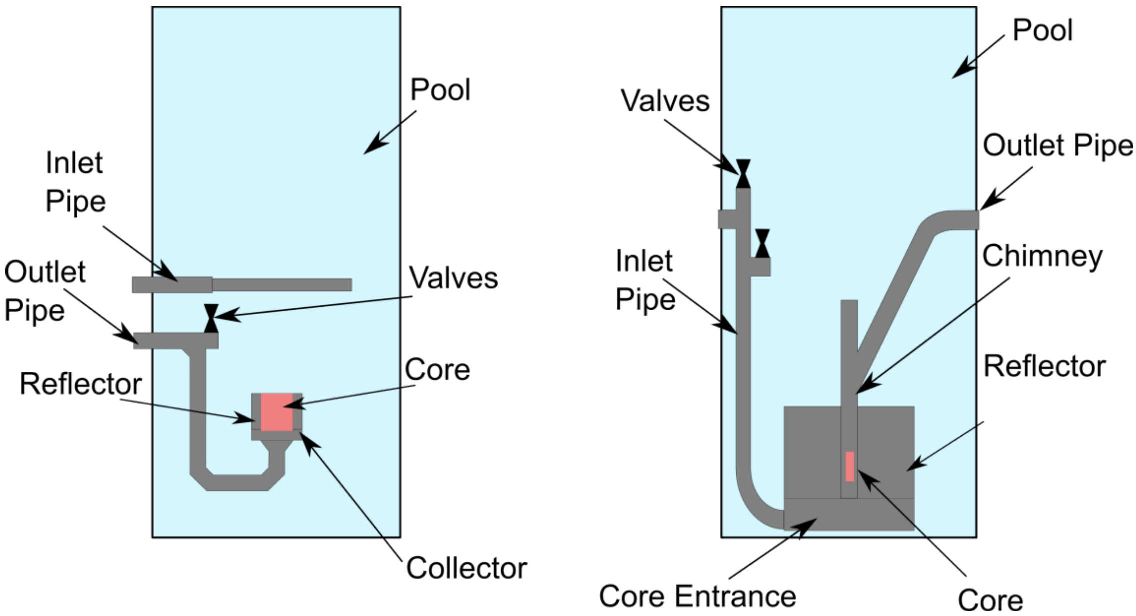

Figure 2-1: Generic design sense of MTR research reactors and modularisation of main components

Next to the abstraction technique, there are different advantages using a modularisation approach with respect to research reactor's main components:

- Higher flexibility in changing module characteristics without affecting other modules

- Support of user's mind set in object-oriented thinking
- Higher transferability between user and safety analysis code (basis for object-oriented software design)

The first level of modularisation is also pictured in Figure 2-1. On the second level, the reactor components are decomposed into their further elements. Focusing on the central component, the "reactor core", typical MTR research reactors have a cluster of multiple assemblies installed at the lower part of the reactor pool. In order to further reduce complexity, assemblies can be disassembled into several parallel arranged fuel plates and the assembly feet. The fuel plates are made of a fuel meat section containing the fissile material. The fuel meat is surrounded by cladding material. Within this work, the fuel meat and cladding material are the smallest units of which a fuel plate is made of. In order to enable the modelling process despite incomplete data, common characteristics of these core materials are collected. At the moment, low enriched uranium fuels (LEU < 20 wt% U235) are considered. Since the Reduced Enrichment for Research and Test Reactors (RERTR) started in 1978 in the USA, most state of the art reactors use LEU fuel /NAP2016/. In the year 2016, 74 operating research reactors using highly enriched uranium (HEU ≥ 20 wt%) were identified to be in scope of conversion programs /NAP2016/. Covering the main and qualified research reactor fuels with low enriched uranium (UAl_x-Al, U_3O_8-Al and U_3Si_2-Al) information about thermal properties were taken out of the IAEA research reactor core conversion guidebook /IAEA1992A/.
Next to thermal properties of fuel elements, neutron kinetic parameters have to be considered inducing the nuclear heat generation of nuclear reactor facilities. Assuming lack of available data, point kinetic approach is selected. Following data are deposited:

Table 2.1: Neutron point kinetics /IAEA1992B/

β_{eff}	L	β (1-6)		Λ (1-6)	
0.007275	43.74	(1) $2.7926*10^{-4}$	(4) $2.9627*10^{-3}$	(1) 0.0127	(4) 0.3121
		(2) $1.5178*10^{-3}$	(5) $9.4536*10^{-4}$	(2) 0.0317	(5) 1.3985
		(3) $1.3731*10^{-3}$	(6) $1.9716*10^{-4}$	(3) 0.1167	(6) 3.8521

Taking into account the feedback reactivity depending on fuel temperature, coolant temperature and coolant density, data referring to IAEA benchmark core specification is used:

Table 2.2: Reactivity feedback coefficients /IAEA1992B/

$\rho = f(T_{fluid})$		$\rho = f(\rho_{fluid})$		$\rho = f(T_{fuel})$	
T [°C]	Δρ*1000	ρ [kg/m³]	Δρ*1000	T [°C]	Δρ*1000
20	+1.478	1000	+2.011	20	+0.473
38	/	998	+1.500	38	/
50	-0.968	993	/	50	-0.309
75	-2.950	988	-1.475	75	-0.948
100	-4.881	975	-5.427	100	-1.567
		958	-10.76	200	-3.908
		900	-30.72		
		800	-72.65		

2.2 Concept of nodalisation

Within the safety analysis code ATHLET, the thermal-hydraulic nodalisation is represented by thermo-fluiddynamic objects (TFOs). There are different TFO types defined in ATHLET, classified into three basic categories:

- Pipe objects, simulating one-dimensional fluid flow

- Branch objects for the representation of major branching
- Special objects for simulation of components with special requirements, e.g. cross connections

Modelling a research reactor in ATHLET, the development of nodalisation is a compromise between required level of detail and acceptable computational effort considering the limitations of the code. Due to one-dimensional approximations, local phenomena are not simulated, but the integral characteristic of the reactor facility is represented. Consequently, a high level of detail in geometrical modelling of some components, e.g. reflector, is not mandatory for safety analysis with a one-dimensional code. Covering different simulations of initial events, e.g. blockage of one cooling channel in a fuel element, the reactor core should be considered in detail and simulated using individual thermal hydraulic channels. Consequently, a high resolution of a fuel assembly nodalisation is determined. Similar assemblies are grouped to reduce calculation time. In this manner, assemblies with various characteristics can be defined and modelled individually. This is necessary e.g. to take hot channel factors into account. In Figure 2-2, the applied nodalisation scheme for MTR fuel assemblies is presented. The nodalisation scheme is used for each individual assembly. Every fuel assembly is linked to a common branch before entering and leaving the reactor core. The fuel plates are modelled as Heat Conduction Objects (HCOs). Internal fuel plates are coupled on both sides to corresponding TFOs. External fuel plates are coupled one-sided to a TFO, representing a cooling channel. The other side is coupled to a single bypass channel next to the core channels. If not changed by the user, the core channels are axially divided by default into 20 nodes of equal length. The number of layers in the cladding material is defined as four on left and right side each and two layers are defined representing the fuel meat zone.

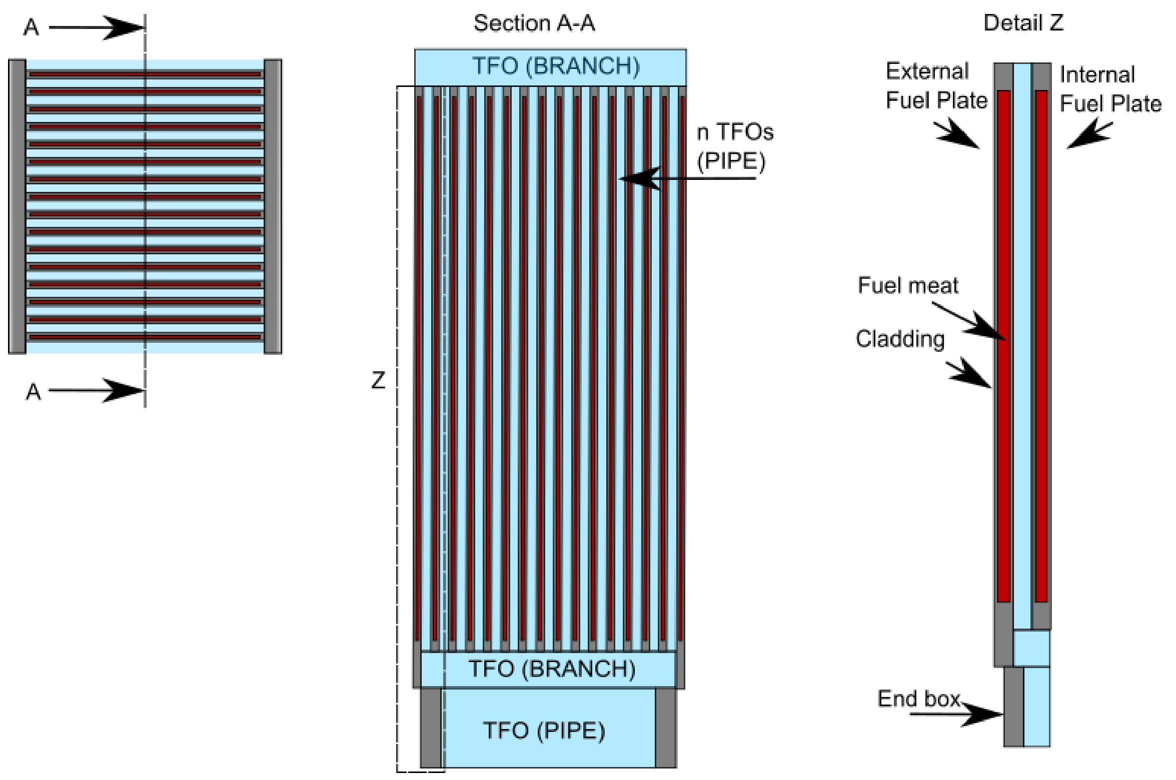

Figure 2-2: Nodalisation of MTR Fuel Assembly

As default setting, the axial power profile follows a sinus curve. The power profile in radial direction of the fuel plate is currently assumed to be homogeneous. While the geometry of

guide boxes and control plates are not considered in the generic model, the external reactivity is modelled by a signal in the general control simulation module of ATHLET.
The main system boundary of facility representation in the TFD part of the input deck is defined at the pool with inlet and outlet pipe. The model of pipe systems within the reactor pool depends on the corresponding reactor design (see Figure 2-1). The reactor core is built as the central element and further systems representing the coolant loop are set up accordingly. Regardless of the reactor design, the coolant system outside of the reactor pool is modelled by one pipe containing a coolant pump and a heat sink as heat exchangers. An example showing the total plant nodalisation is pictured in Chapter 3.

Developing the representation of the reactor components in ATHLET includes also the selection of adequate model options. In order to enable an automatically process of research reactor modelling, different ATHLET specific model options are collected and deposited. This includes general data, e.g. to identify different types of TFOs and specific model options, e.g. to simulate subcooled nucleate boiling processes at low pressure. Further detailed information about required ATHLET input data, are described in the ATHLET User's Manual /ATH2016/.

2.3 Development of process for automation

The last step of realising the new modelling strategy for research reactors is the development of software-based solution for automation. Therefore, the preliminary work described in the previous chapters is used. A brief overview of the software design and applied methods of programming is presented allowing a general understanding.

There are four main phases that are between user's input data and generated input deck executed by the software:

- process of user input
- build the research reactor model
- transform to ATHLET-format
- export as input deck

One of the main challenges is to define the key data, the user has to provide to run the software. An overview of required core and coolant system input data is given in Table 2-3.

Table 2.3: Overview of required core and coolant system input data

	Core Data	Coolant System Data
General	▪ Core type ▪ Core power ▪ Number of fuel assemblies ▪ Number of fuel plates	▪ Design type
Thermo-hydraulic	▪ Core mass flow ▪ Inlet temperature ▪ Outlet temperature ▪ Pressure loss	▪ total mass flow ▪ reference pressure ▪ effective pool water volume
Geometry	▪ Core lattice ▪ Assembly (x,y,z dims, z0) ▪ Element (x,y,z dims, z0) ▪ Fuel pitch	▪ Pipes (z0, zE, d) ▪ Positions of valves ▪ Pool dimensions (z0, zE, d)
Material	Identification of ▪ Fuel meat material ▪ Cladding material	

z0 = height of the component's bottom, zE = height of the component's top, d = diameter

Due to the fact, that it is impossible to foresee the user's amount of available plant data, publicly available data for different research reactors were initially used as a reference data base. Detailed technical documentations, such as safety analysis report, operating manual, system descriptions and schematics as well as technical drawings are assumed to be not accessible.

After processing user's input data, a data base of the modelled research reactor is created. In Figure 2-3 an entity relationship diagram is presented showing the logical structure of a created database of a research reactor. The research reactor is structured into two main parts: core and set of pipes. Following the previously described modularisation process in Chapter 2.1, the reactor core is composed of its different units. The other components within the research reactor (see Figure 2-1) are summarised under "pipework". The pipework is composed of different pipes, which are built up by pipe segments (horizontal, vertical, etc.). The pipe may also contain valves and pumps.

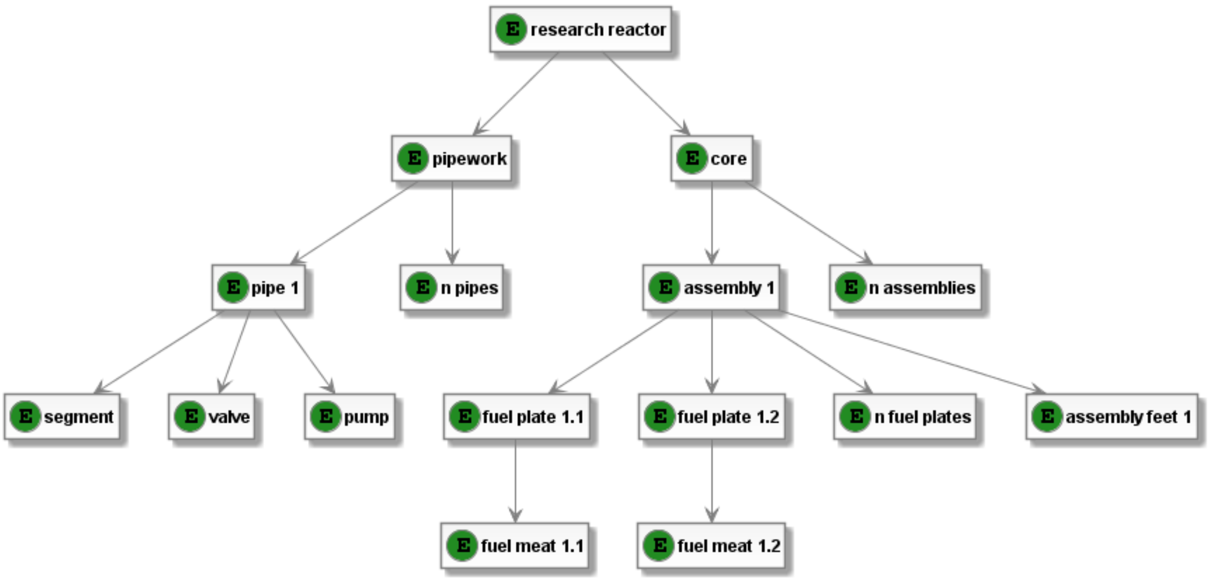

Figure 2-3: Entity relationship diagram: structure of created data base

Within the created data structure of research reactor, geometrical and object specific information are stored. But at the end of the second phase "build the research reactor model" there is no information about ATHLET specific requirements stored yet. Transforming and exporting the research reactor model into ATHLET format, a visitor design pattern is used. The visitor pattern decouples the research reactor data structure from extrinsic methods operating on the reactor objects. Different visit operations are applicable for different types. The visitors are running through the hierarchical research reactor data structure (see Figure 2-3) and if type matching is true, the visitors enable operations on the object without altering the classes. As a result, specific transformation requirements can be taken into account, e.g.: transforming MTR core channels into TFO data by calculating the free area between two adjacent fuel plates. Defining a separate object structure of visitors increases the software flexibility. If the software should be extended to a different safety analysis code, this is possible without changing the research reactor data structure. How the subsystem of visitor pattern collaborates with the reactor data structure is shown in Figure 2-4 using the example of core structure.

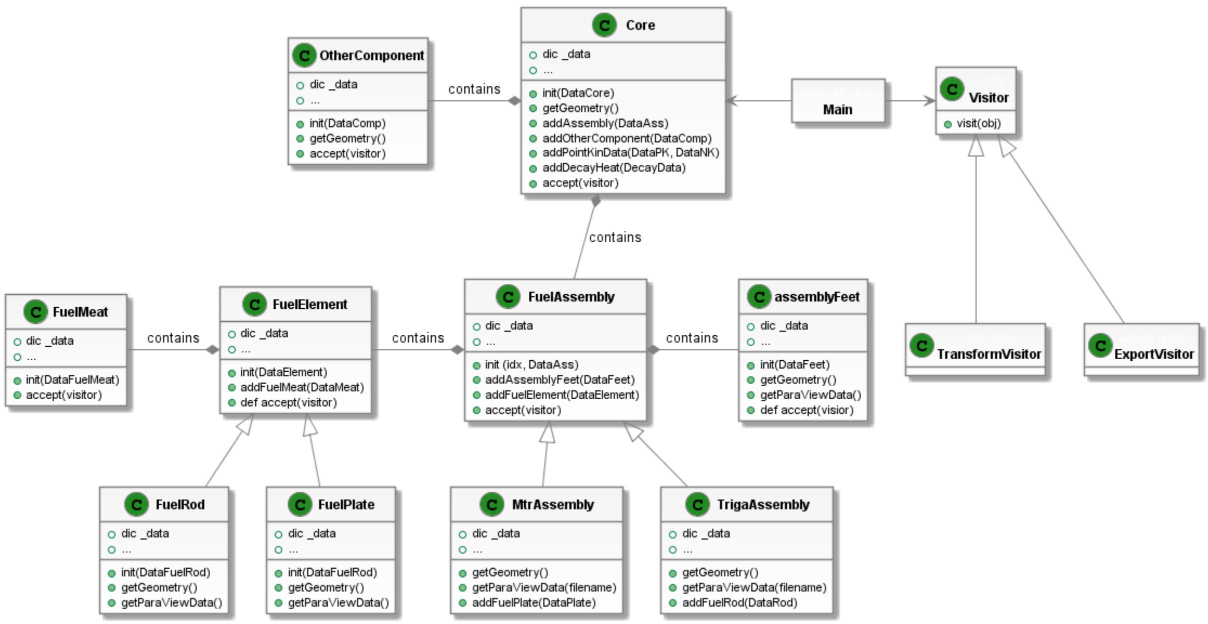

Figure 2-4: Example of collaboration between core components and visitors

3 Simulation of loss of flow transient with generated input deck

In this part, first functionality of the new modelling system is demonstrated by generating an exemplary MTR research reactor model. For this purpose, a reference research reactor was chosen. Accessible plant data and published experimental data to compare the simulation results were used as selection criteria. Providing technical details in /ABD2008A/ and comparative data in /ABD2008B/, the ETRR-2 was identified as a reference facility. Further, data are published in /ABD2015/ and /IAEA2005/. The ETRR-2 is a multipurpose research reactor located in Inshas, the Arab Republic of Egypt. It is used for radioisotope production and research activities as well as materials science. The most common core layout is pictured in Figure 3-1 on the left side. It consists of 29 fuel assemblies of MTR type with 19 fuel plates each. The pool reactor operates at 22 MW nominal power. Its design corresponds to the design shown in Figure 2-1 on the right side. Further description is presented in /ABD2008/.

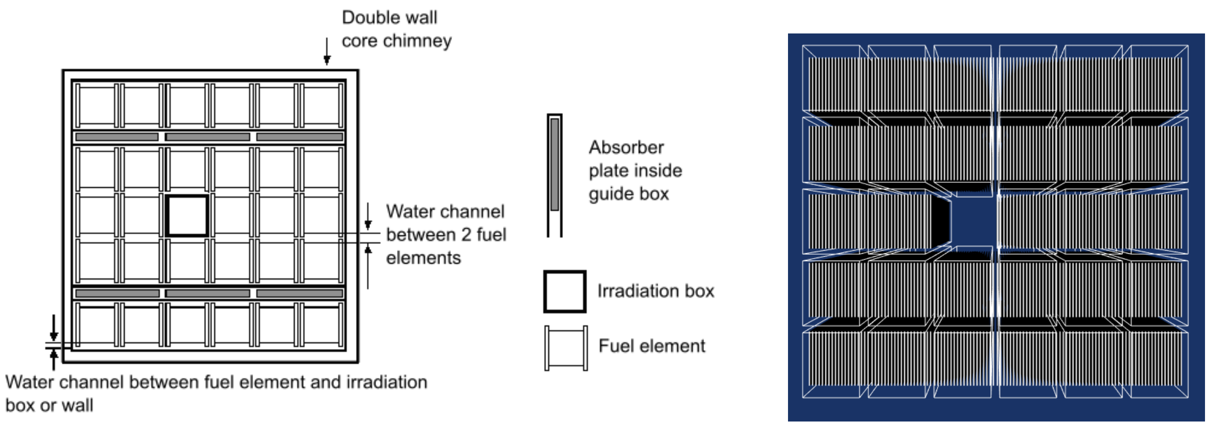

Figure 3-1: Core array ETRR-2/ABD2008/ (left) and core array generated by software for input deck generation (right)

In /ABD2008B/ experimental results of a loss of flow experiment are reported. The experiment was performed at a steady state total reactor power of 9.4 MW to measure the core inlet and outlet temperature profile. Scram was triggered manually and core cooling and secondary pump were tripped /ABD2008B/. The measured mass flows of the primary and secondary pumps coast down were provided as input data for thermal-hydraulic codes. While one flapper valve opened after 46 s, the other valve is defined as out of function and remained closed /ABD2008B/.

The main nodalisation of the generated ETRR-2 model in ATHLET is shown in Figure 3-2. On the left side, the coolant loop is presented in bright blue. The reactor pool is modelled with two pipes interconnected by cross-connections. The inner pool pipe is connected to the reactor chimney, which is marked in brown. The reactor core is modelled with two representative assemblies. Each is composed of 18 core cooling channels. One assembly represents 28 grouped average assemblies. The other assembly considers a hot channel factor on the 19 fuel plates plus one extra penalised fuel plate. The geometry of both assemblies is identically and the nodalisation is shown in Figure 3-2 on the right side.

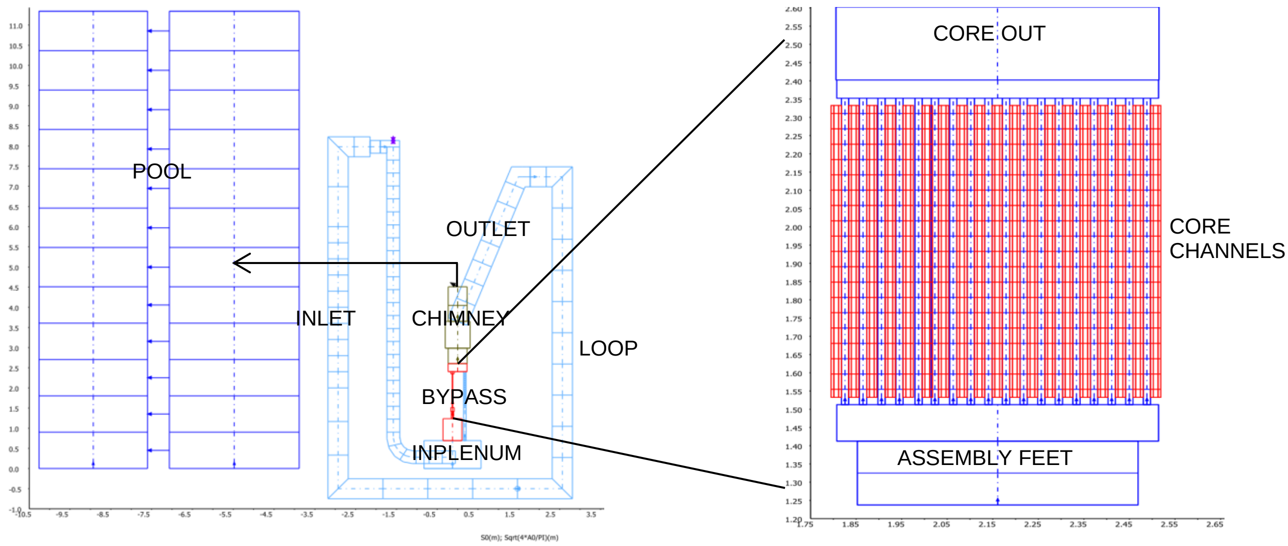

Figure 3-2: Overview of whole Nodalisation of the ETRR-2 (left) and one fuel assembly (right) with 18 core channels generated by the software for input deck generation

At the current state of development, some manually fine adjustments have to be done once the fundamental input deck is generated automatically. This includes e.g. the adaption of friction loss coefficients and addition of transient control signals. Before running the transient simulations, steady state calculations were performed. Thereby, the capability of the nodalisation to reproduce the thermal hydraulic plant conditions is checked. The initial conditions of the loss of flow experiment and the calculated parameters are compared in Table 3-1. There is agreement between the calculated and experimental stationary data.

Table 3.1: Thermal-hydraulic data

	Power [MW]	Loop mass flow [kg/s]	Core mass flow [kg/s]	Core outlet temperature [°C]	Core pressure drop [bar]	Reference pressure [bar]
Calculation	9.5	309.24	302.86	35.01	0.42	2.2
Reference /ABD2015/	9.4	309.24	302.87	34.9	0.31*	2.0

**in /ABD2015/ core pressure drop of 3.1 bar is mentioned, but in /IAEA2005/ 0.6 bar pressure drop at 100 % core power is referred*

Based on the steady state results, the loss of flow transient is simulated. The calculated results are compared to the experimental data. After scram activation, the primary and secondary pumps coasted down within 92 s (primary cooling system) and 11 s (secondary cooling system), following the given measured flow data /ABD2008B/. The point of time, where one flapper valve opened, is provided as well. In the ATHLET simulation, reactor scram was triggered at 1000 s simulation time. The coolant mass flow decreases until the flapper valve is opened at 1046 s. A small mass flow is established from the coolant loop to the pool through the valve. At about 1123 s, the mass flow reverses and establishes the natural circulation with approximately 5 kg/s. According to simulation results in /ABD2015/, the natural mass flow through the core reaches 50 kg/s (only time period up to 220 s after scram is plotted).
In Figure 3-4, the calculated core inlet and outlet temperature is compared to the experimental data. While the calculated core inlet temperature remain almost constant in the ATHLET simulation, the measured data increases until 31°C before the temperature decreases again. Focusing on the core outlet temperature, both temperatures decrease rapidly after scram activation. After that, the two temperature profiles start to rise again. In contrast to the experimental data, the ATHLET simulation remains at a higher temperature level at the end of the simulation than the core inlet temperature. As explained in /ABD2015/, local convection and thermal streaming and fluctuation became dominant at this point of time (about 1110 s) resulting in a temperature profile, that is not suitable for further comparison.
It may be concluded, that the generated ATHLET nodalisation is capable to simulate the loss of flow experiment during the first 100 s transient time. Further research is necessary to improve the simulated flap valve mass flow.

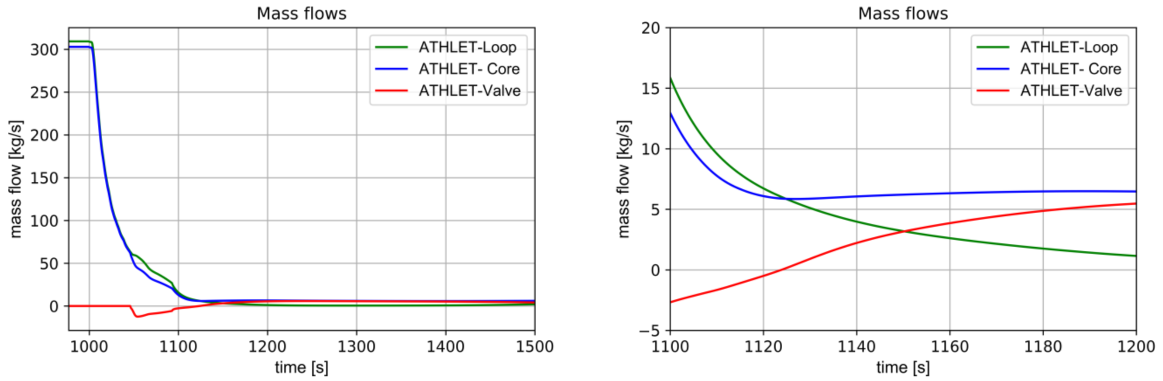

Figure 3-3: Coolant mass flows

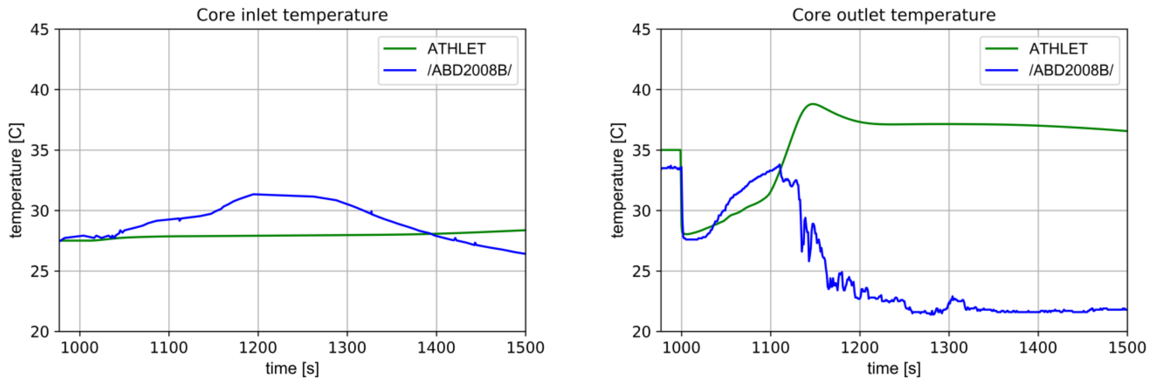

Figure 3-4: Core inlet (left) and core outlet (right) temperature

4 Summary

A new method based on a heuristic approach to model selected research reactor types in thermal hydraulic analysis codes is presented. This new approach allows a fast and reliable generation of the input deck's fundamental elements despite limited technical documentation. Focusing on MTR design, three main steps of developing process and the characteristics of the new method are highlighted. This includes the abstraction and modularisation of research reactor plant designs as well as the conception of a nodalisation. Finally, the development of the automation process is outlined. At the end of this paper, an exemplary MTR research reactor is presented, generated by the developed software-based system. Preliminary results of a loss of flow transient are compared to experimental data. Focusing on the stationary conditions, there is a good agreement between the calculated and experimental data. This underlines the basic functionality of the developed modelling system by generating a realistic plant model. Analysing the transient period, the measured temperature profiles and the calculated data show agreement in the short term. Further work is necessary in the way of modelling the natural circulation flap, which underestimates the established natural circulation mass flow. In future work, the nodalisation will be reviewed and tested against further safety transients and accidents.

5 References

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