Concept for developing a research reactor analysis simulator based on a German reference reactor

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ABSTRACT

Deterministic safety analysis is a prevalent and important instrument to evaluate the safety of nuclear power plants and research reactors. This paper deals with the current state of science and technology in creating an input deck for deterministic safety analysis for research reactors using the ATHLET code. A modelling concept based on GRS experience in developing nuclear power plant simulators is outlined. The modelling concept is tested by modelling a German reference reactor. There are three research reactors in Germany that are suitable as a reference reactor: the TRIGA reactor FRMZ, the High-Flux reactor FRM-II and the MTR reactor BER-II. Each research reactor has a different design and hence, risk categorisation. The scope and level of detail of the safety analysis depends on the individual risk categorisation of the research reactor. Therefore, the reference system has a direct influence on the complexity of the simulator model and on the modelling concept. The rationale for the choice of the FRM-II as the most suitable reference system is given together with the current status of the analysis simulator.

1. Introduction

For more than 50 years, research reactors have been applied in nuclear science, technology and medicine in Germany. Currently, seven research reactors facilities are in operation – four smaller training reactors (one so-called homogenous thermal zero-power reactor and three Siemens training reactors) and three pool reactors (one MTR- (Material Test Reactor), one TRIGA-(Training, Research, Isotopes, General Atomic) and one high flux reactor. Like nuclear power plants, the national research reactors are subject to the German Atomic Energy Act, but they are not affected by the 13th law amendment setting fixed end dates by which nuclear power plants must be shut down. However, every ten years, research reactor operators are obligated to perform a periodic safety review and evaluation of their safety systems according to §19a of the Atomic Energy Act. As a part of this periodic safety review, a deterministic safety analysis has to be performed to evaluate the safety systems under accident conditions. The results have to be submitted to the regulatory authority. As a technical support organisation for regulatory bodies, GRS (Gesellschaft für Anlagen- und Reaktorsicherheit gGmbH) develops i.a. ATHLET (Analysis of thermal-hydraulics of leaks and transients) for transient and accident simulations and provides independent deterministic safety analyses for postulated events.
To perform a safety analysis, the code must be verified and validated. A wide range of system codes that are used for simulations of nuclear power plants are applicable to simulations of research reactors. In order to extend the application range of ATHLET to research reactor simulations, Hainoun developed a model to simulate steam formation in subcooled boiling regimes /HAI1994/. About the same time, GRS developed and implemented a model in ATHLET to simulate wall evaporation in subcooled and saturated nucleate boiling regimes. To activate the detailed simulation of subcooled nucleate boiling processes at low pressure, the user has to switch off the reduction of wall evaporation and condensation rate at heating and cooling surfaces for low pressures /ATH2016/, /GRS2009/.

Following these model options, research reactor simulations using ATHLET were successfully performed at national and international research institutes.
The modelling process and the complexity of the modelled systems differ for research reactors from those of power plants. There are also differences between the different types of research reactors. Therefore, it is important to ensure a qualified modelling concept specifically for research reactors. The modelling concept at GRS is presented in this paper.

2. Research reactor types in Germany

There are three research reactors in operation in Germany with a continuous thermal power of more than 50 kW. The reactor at Mainz University FRMZ is an open pool reactor of the TRIGA Mark II type. The BER-II at the Helmholtz-Centre in Berlin is an open pool reactor of the MTR type. The FRM-II, located at the Heinz Maier Leibnitz institute in Garching (near Munich), is a high flux reactor with a compact core. Table 2.1 shows the main core design characteristics of the three reactor types.

Table 2.1: Research reactor types in Germany

	BER-II /HZB2017/	**FRMZ** /GEP2016/,/JGU2017/		**FRM-II** /TUM2017/
		Continuous	*transient*	
Power [MW]	10	0.1	250 (0.03s)	20
Inventory [kg U]	9	2.7		8.1
Enrichment [% U235]	20	20		93
Max Flux, Thermal [n/(s·cm²)]	$2 * 10^{14}$	$4.2 * 10^{12}$	$1.0 * 10^{16}$	$5.0*10^{14}$
Max flux, Fast [n/(s·cm²)]	$1.4 * 10^{13}$	$4.8 * 10^{12}$	$1.2 * 10^{16}$	$8.0*10^{14}$
Fuel type	MTR	TRIGA		Involute
Fuel assemblies	30 (17/23 plates)	76		1 (113 plates)
Fuel material	U_3Si_2-Al dispersed	ZrH		U_3Si_2-Al dispersed
Cladding material	Al	Stainless steel		Al
Control rods	Hf 6 CR	Bc, B 3 CR		Hf 1 CR + 5 SHUT DOWN
Moderator	Light water	Light water, ZrH		Heavy water

As of today, there is no standardisation for the risk categorisation of research reactors. According to the IAEA, research reactor types may be categorised applying a graded approach based on factors like reactor power, reactivity control, amount and enrichment of fissile or fissionable material, inherent and additional safety features or radiological source term (potential for dose). Considering these criteria, the main characteristics of the three biggest German research reactors are analysed below, addressing why the FRM-II is chosen as model reference using a risk categorisation approach.

The TRIGA research reactor in Mainz features a prompt negative temperature coefficient provided by its fuel element design, which is based on a combination of low enrichment fuel (<20 % U-235) and zirconium hydride as moderator. Because of its inherently safe fuel design, the reactor can be pulsed to power levels of 250 MW returning within 30 ms to a safe low power level without any external aid, allowing natural convection cooling /JGU2017/. These design features provide the reactor with a high degree of safety that contributes to its lower risk categorisation.
The MTR research reactor in Berlin reduced its fuel enrichment level from HEU to LEU (<20 % U-235) in 2000. It currently contains 30 fuel elements each constructed of 23 (or 17 control assemblies) thin plates of a uranium silicate compound embedded in an aluminium mantle (U_3Si_2-Al) /HZB2017/. The fuel material is very similar to the one used at the FRM-II, but the differences in fuel geometry and enrichment level are significant.
The concept of the FRM-II reactor is based on the use of a compact core consisting of one fuel element with a high enrichment level (93% U-235), allowing a high neutron flux at a thermal power of 20 MW /TUM2017/. It has the highest power, the highest enrichment level and the highest neutron flux of the German reactors. Given these characteristics, the maximum heat flux on a hot point at the fuel plate surface can be up to 455 W/cm² at the beginning of the cycle /TUM1993/. To meet the safety criteria at these boundary conditions, the FRM-II has extensive active and passive safety systems. Its design is therefore more complex, which makes it more interesting and suitable as a reference system for a research reactor simulator model from a safety analysis point of view.

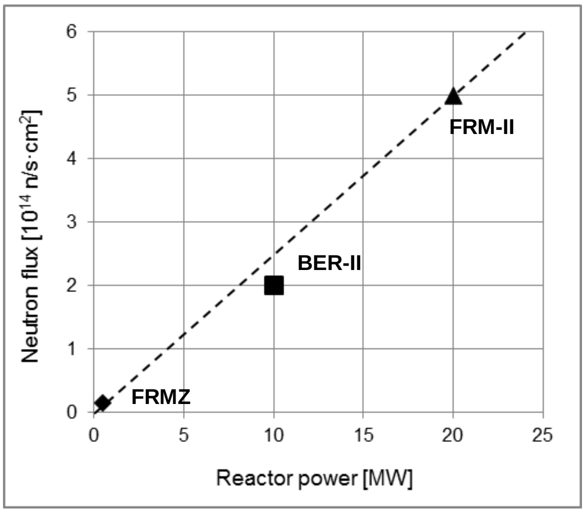


Figure 2-1: Relation between neutron flux density and reactor power for the three reactor types

For a classification based on the fuel type, the TRIGA reactor stands out with its inherently safe fuel design. In the reactors FRM-II and BER-II, dispersed U_3Si_2-Al fuel is used – LEU for the Berliner reactor and HEU for the Munich one. Both are reactors with a compact core structure and narrow plate fuel geometry with multi parallel channels. This type of research reactor is subject to the phenomenon of excursive flow instability. Thus, flow instability is considered as design limit for research reactors of plate type fuel. Given the high heat flux density of these reactors and the low system pressures, thermal hydraulic instabilities may occur during transients. In the narrow coolant channels void formation may arise, resulting into critical heating surface being exceeded locally and leading to damage of the fuel plates due to their low melting point of about 600°C. In both BER-II and FRM-II designs the fuel integrity is maintained and the safety margin against flow instability is much greater than the minimum required by the German Nuclear Technical Regulation (KTA) /HMI2001/, /TUM1993/. However, according to the IAEA transient comparison of HEU and LEU cores /IAE1992/, the minimum safety margin against flow instability is marginally larger for the LEU case. A classification based on the relation between core power and neutron flux is included in Figure 2-1. Using the IAEA factors for application of a graded approach, the main results of the categorisation are represented in the spider web diagram in Figure 2-2.

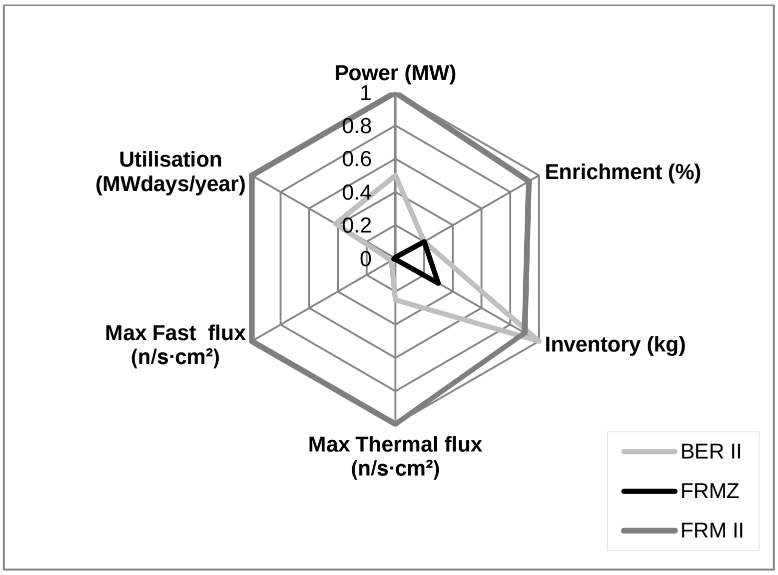


Figure 2-2 Spider web diagram based on factors considered for grading and categorisation according to IAEA

3. Generic modelling concept in ATHLET

After specification of the objectives and scope of the accident or transient analysis, the plant model has to be developed. Despite different research reactor designs, the main steps within the model developing process can be summarised in general terms. Therefore, a general process will be first described on a more abstract level, before details are highlighted in Chapter 4. In Figure 3-1, a simplified scheme of different steps within the plant modelling process is presented. Boxes with a grey background are further subcategories.

GRS is developing and operating a broad range of simulation programs for transient and accident analysis. As a first step, the user has to select an appropriate code or coupled code systems, which are capable to simulate the reactor facility behaviour. The presented flowchart focuses on the usage of ATHLET. The thermal-hydraulic code ATHLET uses the finite volume method and solves the partial differential equations matrix at discrete meshed volumes. The user has to build up the whole plant system to be simulated via this network of thermo-hydraulic volumes. To each control volume, heat structures can be added. Besides the thermal-hydraulic part, ATHLET provides a control and instrumentation module (GCSM). GCSM is mainly used for plant control simulation. Typical systems to be considered are the reactor protection system, limitation and control systems. Using analogue and logical signals, simplified modelling of plant components in GCSM is also possible. From available technical documentation, the user has on one side to define the thermal-hydraulic boundaries of the system to be modelled and on the other side, the possible representation of instrumentation and control systems.

The development of a detailed nodalisation scheme and transmission of plant control systems takes a large amount of human resources and requires experienced users. The nodalisation must meet different requirements that are e.g. sticking to the code guidelines and specific model options. Due to the complexity of the nodalisation development process, it is recommended to use a bottom-up approach, starting with the representation of a single component. Then, step by step, further elements and components have to be appended. Through this iterative process a higher input deck quality and less user errors are ensured. Based on the associated nodalisation scheme, corresponding plant data have to be transformed into the ATHLET format. This includes e.g. calculations of geometry data of flow paths as well as main thermal-hydraulic variables. After implementing a component or system, it is recommended to execute test calculations until the results are plausible. References of the plant data used should be documented within the input deck in order to make the input data comprehensible. With the implementation of further systems and components, the original input deck is gradually becoming more complex. This procedure has to be repeated for the implementation of control systems. When the input deck development is finished, the verification process has to be started.

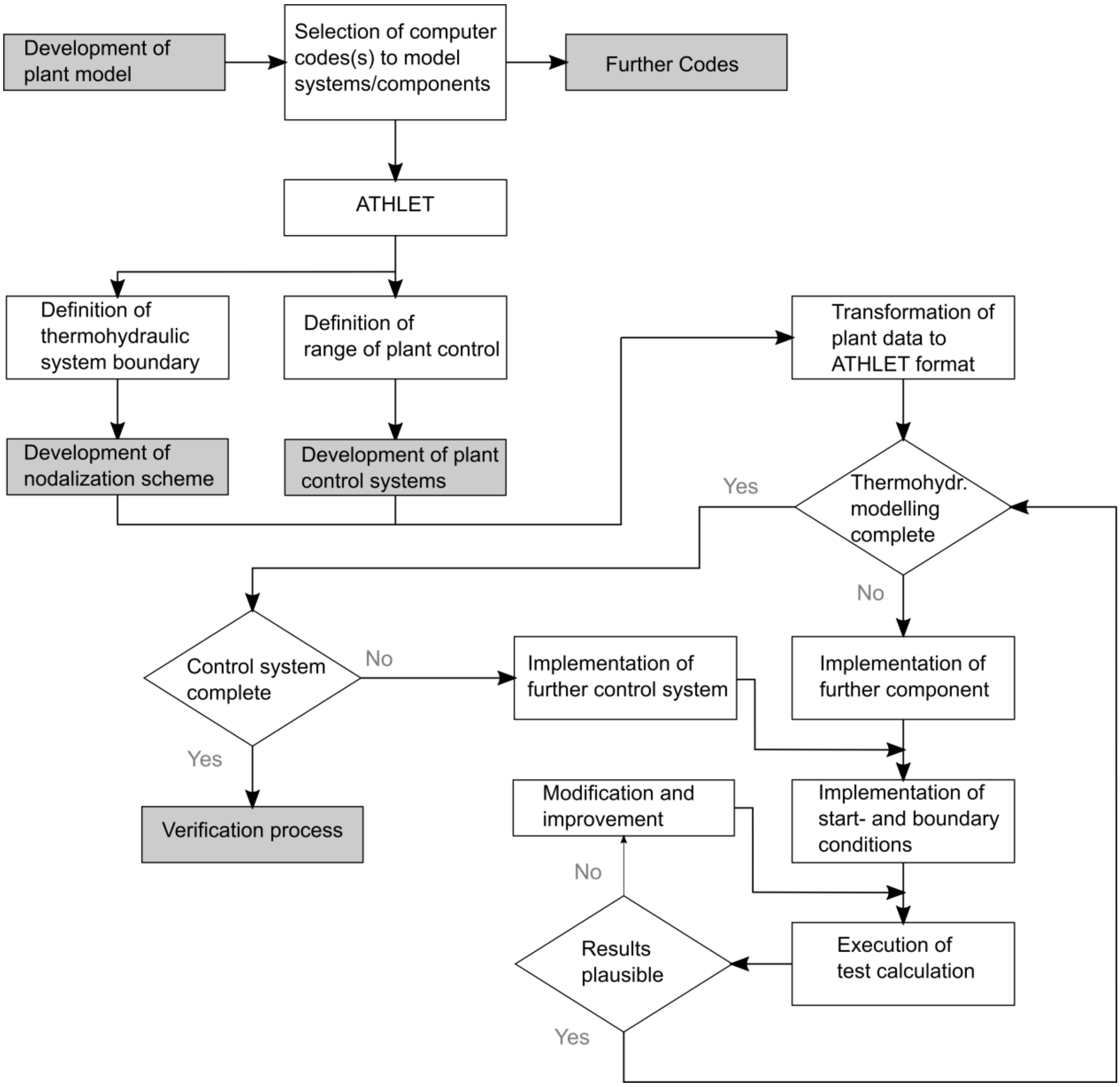
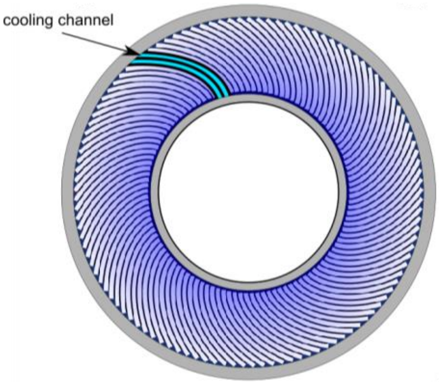


Figure 3-1: Generic modelling concept in ATHLET

4. Development of core nodalisation for the reference research reactor

Following the generic modelling concept in Chapter 3, the nodalisation development starts with the representation of a single plant component. Complex thermal-hydraulic conditions are present in the reactor core – the central safety element. Therefore the core is chosen as the start component. As outlined in Chapter 2, the FRM-II was chosen as reference research reactor for the generic research reactor input deck. The FRM-II has only one fuel element. Due to the involute geometry of the compact core, the cooling channels between the 113 fuel plates have a constant width of ca. 2.2 mm /BRE2012/. A schematic drawing of the compact core design is shown in Figure 4-1. Two cooling channels are emphasised. The active height of the cylindrical fuel assembly is 70 cm /BRE2012/. The height of the fuel element including cladding is 72 cm. To avoid power peaks, the fuel elements have two radial zones with different uranium densities, inner zone 3.0 g/cm², outer zone 1.5 g/cm². The different fuel zones are shown in Figure 4-2. The inner zone is marked in dark red and the outer zone is illustrated in bright red. Due to the density jump within the fuel elements, the core has a radial power profile.



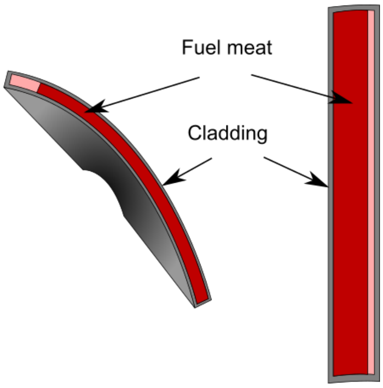


Figure 4-1: Scheme of reference compact core design; two cooling channels are emphasised in light blue

Figure 4-2: Scheme of top view (left) and front view (right) of reference fuel element; the density jump within the fuel meat is marked as light red

From a modelling point of view, a radial power profile within one fuel structure is difficult to simulate in AHTLET. The heat transfer package covers a wide range of single phase and two-phase flow conditions. Optionally, two-dimensional heat conduction can be simulated considering the axial direction of plates and cylinders. But there is no model implemented that simulates the heat flux radial along the fuel plate. Referring to available data of axial and radial power profile, a parallel channel approach with three thermo-fluid channels is used. The thermo-fluid channels are connected by cross connection objects. The radial power profile is described through separate heat structures. Due to the rotational symmetry of the reactor core, one core channel is representing 112 cooling channels during a multiplication factor. In addition, a penalised cooling channel is modelled to consider a hot channel peaking factor. Assuming a deviation in the manufacturing process of fuel plates, the penalised cooling channel has a reduced gap size. In Figure 4-3 on the left side, a scheme of one cooling channel is shown. The dashed lines indicate the subdivision of the cooling channel and the corresponding fuel plate into three segments. Following the presented flow chart in Chapter 3, the next step is the transformation of this nodalisation scheme into ATHLET format. Following the lumped parameter approach, the involute shape is transformed into three straight volumes containing the hydraulic diameter, area, volume as well as the length in z-direction as geometric data for each segment. The corresponding fuel plate segments are assigned to each volume. The fuel plate is divided in half at the centreline and coupled to the thermo-fluid channel with double length to consider the total heat surface in one cooling channel. At the centreline, the heat conduction object is adiabatic. In Figure 4-3 on the right side, the transformation of one cooling channel into the corresponding ATHLET model is pictured.

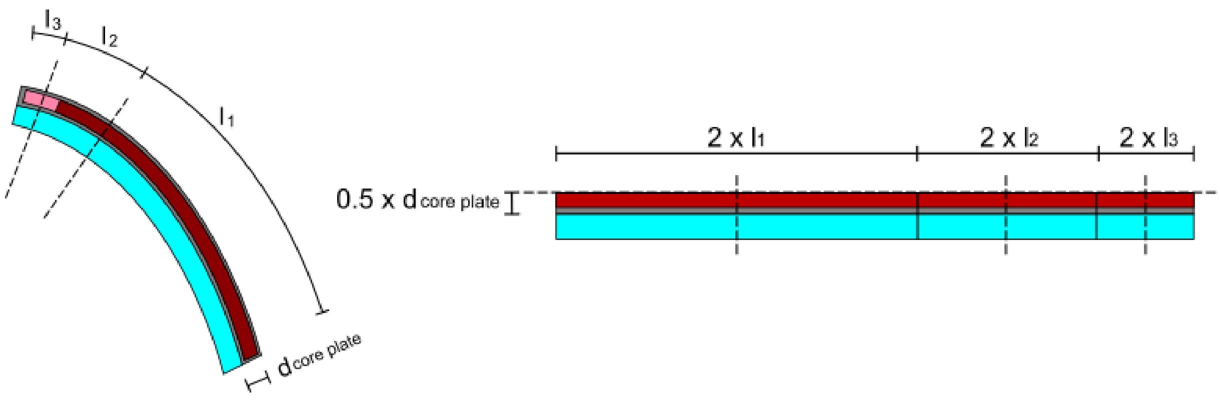


Figure 4-3: Scheme of one cooling channel subdivided into three segments (left) and model approach of corresponding heat structures in ATHLET (right)

5. First simulation results

According to the generic modelling concept described in Chapter 3, test calculations are to be executed to qualify the core model. Using the interactive analysis simulator ATLAS, first simulation runs were performed with the core model using stationary initial and boundary conditions. As mentioned before, two cooling channels are modelled represented by three thermo-fluid objects each and connected by cross connections. Next to the fuel channels, the bypass is modelled. Nominal conditions at BOL (begin of life) are assumed. The ATHLET-nodalisation is pictured in Figure 5-1.

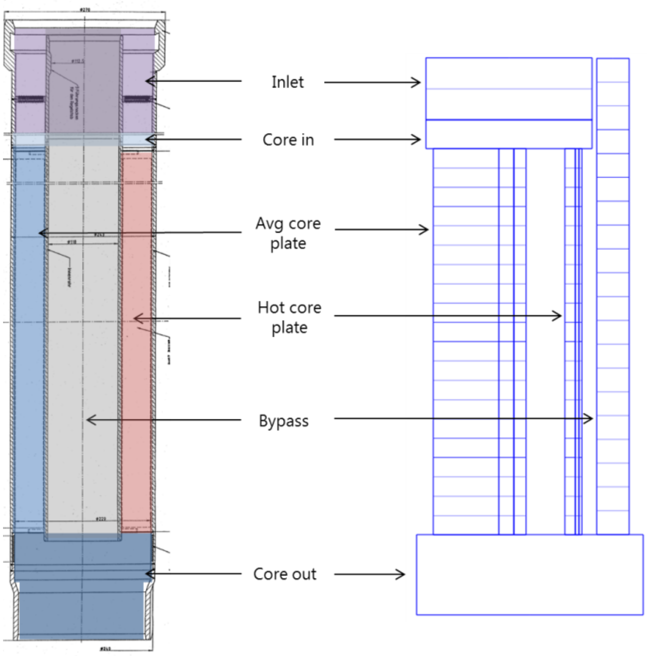


Figure 5-1: Assignment of FRM-II core geometry to ATHLET nodalisation

To validate the selected nodalisation, a comparison between calculated parameters and reference data is given in Table 5-1. There is agreement between the stationary data. Further, the pressure drop over the fuel elements is analysed. Calculations of /DÄU2012/ were used as comparative data. The pressure profile is presented in Figure 5-2. Agreement between GRS results and the reference data can be observed.

Table 5.1: Thermal-hydraulic data

	Mass flow core [kg/s]	Mass flow bypass [kg/s]	Fluid core inlet temperature [°C]	Fluid core outlet temperature [°C]	Core velocity [m/s]
Calculation	279.2	20.8	37.4	51.8	16.7
Reference /TUM1993/	~ 280	~ 20	~ 37	~ 53	~ 17

Further, the axial heat flux profile for each fuel plate section is averaged to compare the gradient over fuel plate length to the reference data published in /DÄU2012/. Although there are deviations in the amount of heat flux at the lower end of the fuel plate, the qualitative progress is comparable to the reference data. The difference might be a result of different nodalisations as well as the considered axial and radial power profiles.

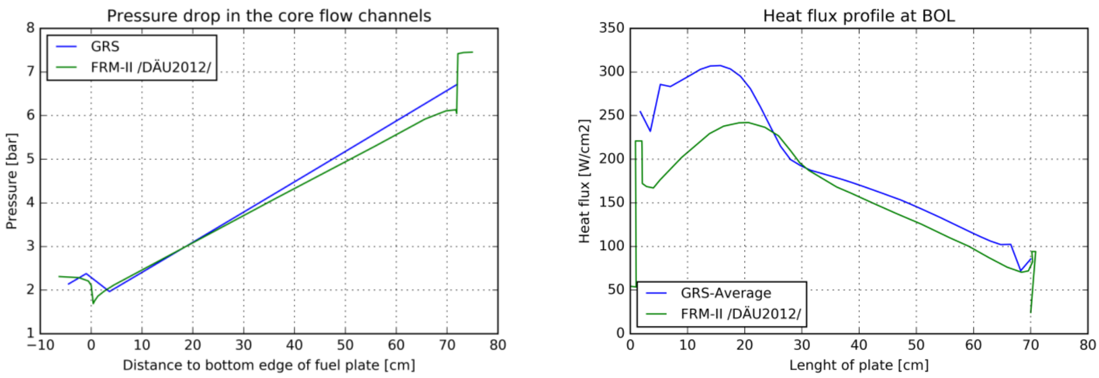


Figure 5-2: Calculated pressure drop (left) and heat flux profile (right) compared to reference data /DÄU2012/

6 Summary

Within this paper, a conceptual overview of how to model complex research reactor facilities using the thermal-hydraulic system code ATHLET is outlined. For this purpose, the current state of science and technology on creating an input deck for deterministic safety analysis is presented defining a general modelling concept in ATHLET. First steps of the presented modelling concept are executed modelling the reactor core of a reference research reactor. Therefore, possible reference facilities focusing on German research reactors are shortly described emphasising the core characteristics. The FRM-II was chosen as a reference reactor, due to its complex core geometry, high power and heat flux levels. At the current status, a nodalisation approach for the core channels and corresponding heat structures is presented.
Preliminary results of a steady state simulation are compared to reference data. The results show acceptable agreement. As next steps, further elements and components have to be appended to complete the plant nodalisation.

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