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Master Thesis

Validation of nodalisation-approach against PSB test facility experiments for a SGTR scenario with RELAP5/SCDAPSIM 4.1 Thermal Hydraulic System Code

Submitted by

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Affidavit

I hereby declare that I have authored this master thesis independently, and that I have not used any assistance other than that which is permitted. The work contained herein is my own except where explicitly stated otherwise. All ideas taken in wording or in basic content from unpublished sources or from published literature are duly identified and cited, and the precise references included.

I further declare that this master thesis has not been submitted, in whole or in part, in the same or a similar form, to any other educational institution as part of the requirements for an academic degree.

I hereby confirm that I am familiar with the standards of Scientific Integrity and with the guidelines of Good Scientific Practice, and that this work fully complies with these standards and guidelines.

Vienna, 05.01.2022 Lukas ANZENGRUBER *(manu propria)*

This thesis is dedicated to my parents Sabine & Joachim,

my grandparents Maria & Franz, Inge & Gabriel

my girlfriend Kathrin

Thank you for everything you have done for me.

Preface

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Table of content

Abstract

The comparison of experimental results with the predictions of Thermal Hydraulic System (TH-SYS) codes is an essential part of the validation process of said codes. The purpose of this work is to validate a TH-SYS code with data from an experiment at the PSB-VVER Integral Test Facility, a 1:300 in volume and 1:1 in height scaled experimental facility for the VVER-1000. VVER-1000 is one of the most common reactor types in the world and manufactured by Gidropress in Russia. The chosen experiment simulates a steam generator hot header break with the assumption of a stuck open safety relief valve (BRU-A) at the initial opening. Such transients constitute a design extension condition A (DEC-A), characterised by multiple failures of safety systems but with the reactor core remaining intact. In addition, an accident management strategy was part of the experiment. After 1800s, cooldown of the primary side was simulated by depressurization of the intact steam generators at a rate of 60 K/h together with disconnecting the high pressure injection system (HPIS) and hydro accumulators (ACCs). RELAP5/SCDAPSIM 4.1 was used for the analysis. It is a best estimate thermal hydraulic system code with capabilities of modeling selected aspects of severe accidents. In the present work the capability of the code to predict the complex behavior of an integral test facility with accident management measures was evaluated. A qualitative comparison of experimental and calculated results, as well as the quantitative comparison using FFTBM was performed to show the quality of the calculation results. The version of RELAP/SCDAPSIM used was able to reproduce the transient satisfactorily. The accident management measures set by the simulated operator were able to prevent the core from drying out and causing serious damage to the plant.

Kurzfassung

Der Vergleich von Versuchsergebnissen mit den Vorhersagen von Thermohydrauliksystem-Codes (THSYS) ist ein wesentlicher Bestandteil des Validierungsprozesses dieser Codes. Ziel dieser Arbeit ist die Validierung eines THSYS-Codes mit Daten aus einem Experiment in der „PSB-VVER Integral Test Facility“, einer 1:300 im Volumen und 1:1 in der Höhe skalierten Versuchsanlage für den WWER-1000. Der WWER-1000 ist ein weit verbreiteter Reaktortyp und wird vom russischen Hersteller Gidropress fabriziert. Das gewählte Experiment simuliert einen Bruch im Hot Header heiße Seite unter der Annahme, dass das Sicherheitsventil (BRU-A) nach Öffnung nicht mehr schließt. Es handelt sich dabei um einen sogenannten DEC-A (design extension condition) Transienten. Solche Transienten sind durch mehrfache Ausfälle von Sicherheitssystemen gekennzeichnet, wobei der Reaktorkern jedoch intakt bleibt. Darüber hinaus war eine Störfallmanagementstrategie Teil des Experiments. Nach 1800s wurde die Abkühlung der Primärseite simuliert, indem die intakten Dampferzeuger mit einer Rate von 60 K/h heruntergekühlt wurden, sowie das Hochdrucknotkühlsystem (HPIS) und die Druckspeicher (ACCs) abgeschaltet wurden. Für die Analyse wurde RELAP5/SCDAP 4.1 verwendet. Dabei handelt es sich um einen „Best-Estimate-Code“ für thermohydraulische Systeme, mit dem sich auch ausgewählte Aspekte von Unfällen mit schweren Kernschäden modellieren lassen. In der vorliegenden Arbeit wurde die Fähigkeit des Codes zur Vorhersage des komplexen Verhaltens einer integralen Versuchsanlage mit Unfallschutzmaßnahmen bewertet. Ein qualitativer Vergleich von experimentellen und berechneten Ergebnissen sowie ein quantitativer Vergleich mit FFTBM wurden durchgeführt, um die Qualität der Berechnungsergebnisse zu zeigen. Die verwendete Version von RELAP/SCDAPSIM konnte den Transienten zufriedenstellend wiedergeben. Die vom simulierten Operateur gesetzten Unfallmanagementmaßnahmen konnten ein Austrocknen des Kerns und ernste Schäden an der Anlage verhindern.

1. Introduction

The nuclear reactors of the VVER type, which are operated worldwide, have some unique features. These include a rather large coolant inventory in the primary (PS) and secondary side (SS), horizontally arranged steam generators (SG), and the reactor design (Bucalossi et al., 2012). In order to be able to simulate as many transients as possible and thus examine and confirm the safety of the systems and processes, the Integral Test Facility PSB-VVER was built (NEA, 2001). The NEA and the Committee on the Safety of Nuclear Installations (CSNI) have been issuing guidelines on the validation of integral test facilities and the development of validation matrices since 1987 (NEA, 1987).

In 1987, Integral Test Facility CSNI Code Validation Matrix ITF-CCVM was published, which was republished in 1996 in a greatly expanded version (NEA, 1996). This publication was followed a few years later by Separate Effect Test Facility CSNI Code Validation Matrix SETF-CCVM in 1994 (NEA, 1994). An extension to the CSNI matrices was the Verification Matrix for Thermal Hydraulic System Codes Applied for VVER Analysis, published in 1995. This was done on the initiative of the Minister of Research and Technology of Germany in cooperation with experts from OECD, CEEC and NIS countries. The 2001 paper Validation Matrix for the assessment of thermal-hydraulic codes for VVER LOCA and Transients is based on all 3 of the above publications and extends the code validation methods described therein (NEA, 2001). Overall trends in the field of severe reactor accident research were summarised by the Severe Accident Research Network (SARNET) in 2017 (Van Dorsselaere et al., 2017).

This master thesis is based on the findings of the TACIS Report 2006 (D'Auria et al., 2006a and 2006b), which was produced in cooperation between experts from the University of Pisa, Electrogorsk Research and Development Center for Nuclear Power Plants Safety (EREC), the Kurchatov Institute and Gidropress, the developing firm behind the VVER reactor.

Over the course of this thesis, a hot header break is investigated at the PSB-VVER integral test facility with the goal to validate a nodalisation with RELAP5/SCDAPSIM Mod 4.1. Hot header breaks are a primary to secondary side leak accident unique to VVER reactors due to their horizontal steam generators. Previous probabilistic safety assessments (PSA) have concluded that steam generator tube ruptures are one of the main causes of core meltdowns in VVER reactors (NEA, 2001). To control a PRImary to SEcondary leak (PRISE), the first thing to do is the isolation of the affected steam generator from the rest of the system to prevent contaminated coolant from entering the secondary side. Since the success of the isolation cannot be guaranteed, the risk of a failed isolation must be taken into consideration. As in the transient worked on in this paper, there is an additional risk that the BRU-A safety valve used for pressure relief will get stuck, thus resulting in the release of contaminated steam into the environment during the transient. The goal of the operator in this case is to minimise the release of contaminated steam to the environment, to prevent a major loss of coolant resulting in damage to the core. In the case of this work, these are simulated operator actions for accident management. Exactly which actions taken by the operator will be dealt with later in this thesis. Compared to other transients, this is a slow transient. The final consequences of this transient therefore depend on the intervention of the operator, as well as his timing, i.e. at what time of the experiment the processes are intervened in from "outside" (NEA, 2001).

As mentioned before RELAP5/SCDAPSIM Mod 4.1 is used for this thesis. This programme is a widely used code in the field of power plant calculations and thermal hydraulics in general, which is why it has already been sufficiently validated. Results from this code have already been compared with real data on a large scale (Allison and Hohorst, 2010). However, PSB-VVER has not yet been included in this process, although this integral test facility is ideally suited for the validation of nuclear power plant calculations, as a large number of different transients can be artificially produced on this system (Bucalossi et al., 2012).

Generally, not many calculations have yet been carried out with SCDAP in VVER and its test facilities. Often, the results of the calculations were proprietary and thus not publicly accessible and usable. This is slowly changing. In 2015, a paper entitled “Application of RELAP/SCDAPSIM to the analysis of station blackout transient with LBLOCA for VVER-1000” was published in the Science and technology journal of the Bulgarian nuclear society – BgNS (Vryashkova et al., 2015). A publication from 2018, which deals with improvements in methods for in-vessel retention strategies for high-power reactors, also uses SCDAP for VVER reactors (Fichot et al., 2018). The two Iranian researchers Salehi and Jahanfarnia (2020) modeled a double-ended loss of coolant accident with station blackout condition for a VVER-1000/V446, which is the same reactor type as in the NPP Buschehr in Iran, with

RELAP5/SCDAPSIM mod3.4. Also from Iran is a publication by Gharari et al. (2021), which uses RELAP5/SCDAPSIM to investigate sensitivity analysis and uncertainty analysis in hydrogen production in a VVER-1000/V446.

Although modelling of transients in the VVER Ecosystem using RELAP5/SCDAPSIM is becoming more common, validations of this code with the PSB-VVER ITF have not yet been carried out.

In addition to this work, a paper from the University of Natural Resources and Applied Life Sciences Vienna's ISR institute will be presented at the NURETH conference in early 2022 by Raphael Zimmerl which also deals with this topic (Zimmerl et al., 2021).

1.1. Use of software in the licensing of nuclear power plants

In the history of the use of nuclear energy, the need for and the demands on ever-improving software have constantly increased in order to be able to see accident risks and possible problems with the plant in advance. Experimental data from a test facility can be compared with the results of a computer code which mirrors the same facility. If the results of both the experimental data and calculated results agree sufficiently, the nodalisation approach can also be applied to larger power plants. Such test facilities enable tests on the effectiveness of safety systems in the event of incidents and accidents and therefore play a vital role in nuclear safety (Heralecky, 2014).

Although PWR and VVER reactors are relatively similar, which means that experimental results from PWR plants are also valuable for VVER, VVER reactors have some special characteristics which lead to regulatory authorities usually requiring their own validations for VVER plants (Bucalossi et al., 2012). The test facility (PSB-VVER), which is the subject of this master’s thesis, refers to a VVER-1000/320 reactor. The reference plant is located in Balakovo, Russia.

A detailed description of PSB-VVER can be found in the corresponding chapter 2.3. Since some tests for malfunctions and accidents could lead to damage or even the destruction of the nuclear power plant, the safety of the plants is checked by relying on test plants and available experimental data in the sense of licensing in order to provide sufficient proof of safety. There is a wide range of programs that can be used for this purpose. A selection of the most commonly used programs include: RELAP5, CATHARE, ATHLET and MELCOR (Coscarelli et al., 2013).

1.1.1. The reliability of calculation results with FFTBM

The reliability of the calculation results can be checked quantitatively, for example, by using the Fast Fourier Transform Based Method (FFTBM). The Fast Fourier Transform Based Method (FFTBM) is one of the oldest and most widely used methods for quantitative analysis of experimental data. The method was developed at the University of Pisa and was implemented in the programming language FORTRAN (D'Auria et al., 1990). FFTBM uses two main characteristics for assessing accuracy of calculation results: AA, which stands for average amplitude and can be described as an accuracy coefficient (D'Auria et al., 2006a). The value assumed by AA describes the error in the calculation of the considered variable. A calculated value of 0 would mean that the calculation results and the experimental data are equal. The second important factor in a quantitative analysis with FFTBM is the weighted factor (WF) for each variable. The weighted factors are specified before the calculation for the individual aspects of the calculation. In this work, these values were taken from the TACIS Report (2006b). The weighted factors are assigned according to the importance of aspects in the transient (D'Auria et al., 2006b). Important safety systems are weighted more heavily in order to be immediately noticeable in the event of discrepancies in the calculation.

The way FFTBM works can be explained as follows. The frequencies are calculated by a fast fourier transformation and then the average of the frequencies is calculated. This average can then be compared with the average of the experimental data. The difference between the two values then gives the Average Amplitude. As described below, values close to 0 are excellent.

The weighted factor value finally forms the framework for the quantitative evaluation of the results. The importance for the integrity of the system determines the weighted factor value of the parameter. Using the weighted factor and average amplitude, an AA_{tot} value can be calculated for the calculation. This AA_{tot} value can then be used to read how well the calculated results match those from the test facility.

From the previous work of Prosek et. al (2002) and D'Auria et al. (1990) one can assume the following framework for the results of a quantitative analysis with FFTBM:

$AA_{tot} \leq 0.3$ very good predictions

AA_{tot} between 0.3 and 0.5 good predictions

AA_{tot} between 0.5 and 0.7 poor predictions

AA_{tot} over 0.7 very poor predictions

In chapter 2.42.4 further points regarding the reliability and significance of such results are adressed.

1.2. Research Questions

The research questions to be investigated in the course of this work are:

- Can a hot header PRISE leakage accident in the PSB-VVER integral test facility be satisfactorily reproduced with RELAP5/SCDAPSIM 4.1.0 BUILD 12/2018?
- Which accident management measures can be taken by the operator to bring the system back under control and minimise the damage caused?

1.3. Structure

First, the background of the work is explained and the project from which the research for this work arose is described. Then the experimental facility and its technical features are explained in detail.

An introduction to the codes and programmes used form the next chapter. This is followed by an explanation of the transient studied and is rounded off by an explanation of the research methodology, validation strategy and validation results.

Finally, the results are discussed, and potential future research is outlined.

2. Data, methods and plant description

To explain the background of this research, the following chapters describe the R2CA project, the codes and programmes used, the transient studied, and the nodalisation approach used.

2.1. The R2CA project

The research for this thesis was carried out as part of the R2CA project. The project was funded under H2020-Euratom-1.1, included 17 organisations and 4,2 € million in funding, three quarters of which was provided by the European Union. The participating organisations are shown in Figure 1. R2CA is an acronym for "Reduction of Radiological Consequences of design basis & design extension accidents". The main objective of the project is to reduce conservatism in safety assessments of selected design scenarios, calculations of releases taking into account the best estimate plus uncertainties (BEPU), as well as an assessment of more realistic safety margins (Girault, 2020).

This large project was divided into several smaller work packages. One of the first work packages of the project dealt with the experimental basis and the research method used in the project. (Hózer et al., 2020).

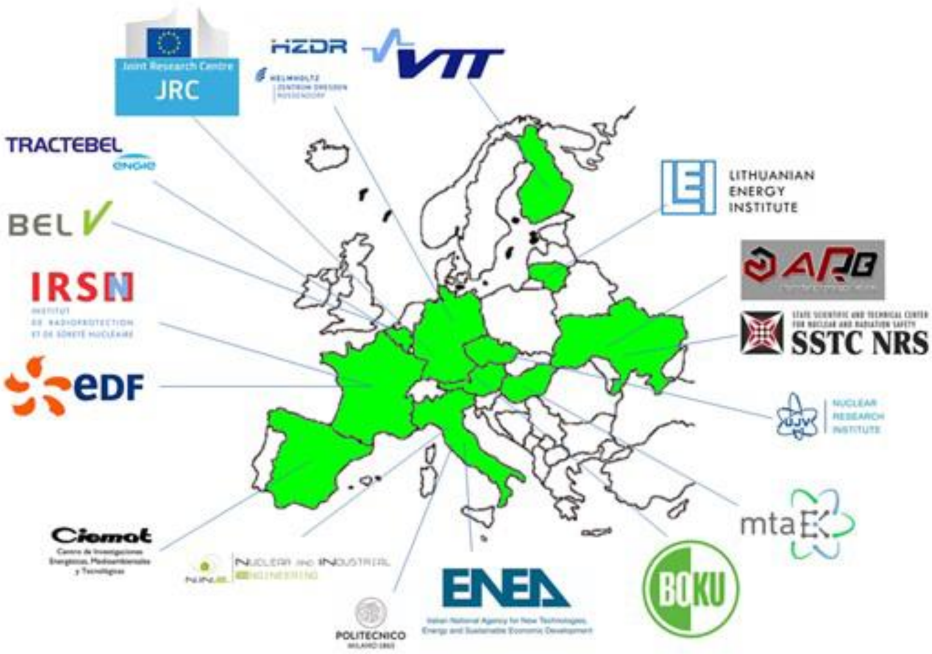


Figure 1 Participating organisations of the R2CA Project (IRSN, 2019)

2.2. Description of the VVER-1000

The VVER-1000/V-320 reactor in Unit 4 of the NPP Balakovo, Russia, forms the reference for the PSB test facility described in 2.3. It is a pressurised water reactor with a thermal output of around 3000 MW and an electrical output of 1000 MW. This reactor type consists of a primary circuit with 4 loops, each of which has a main circulation pump and a horizontal steam generator. The steam generators are fed by 3 separate feedwater systems, the main feedwater as standard, the auxiliary feedwater and the emergency feedwater system. All components of the primary circuit are located inside the containment building.

The primary circuit consists of the following elements(D'Auria et al., 2006a):

- First, the coolant comes from the steam generators via inlet nozzles into the reactor pressure vessel.
- Via the downcomer, which separates the outer area from the inner vessel, the coolant is transported into the lower plenum of the reactor pressure vessel.
- Via orifices in the inner vessel, the coolant reaches the core and then enters slots within the fuel support structure, which lead directly to the fuel assemblies.
- The coolant now flows directly through the open bundles in the core.
- After the coolant has passed through the reactor core, it enters the upper plenum. From there, the water enters the hot legs of the respective cooling loops.

The steam generators in VVER-1000 are horizontally aligned, u-tube and natural circulation type (see Figure 2,Figure 3 and Figure 4). The horizontal alignment of the VVER-1000 is an important difference to western reactor designs. A cross section of the steam generator can be seen in Figure 3. Especially with secondary side occurences, this large coolant inventory has an influence on the reaction in the transient state (D'Auria et al., 2006a).

VVER 1000 reactors are operated at 16 nuclear powerplants worldwide. The power plant in Zaporizhzhia, which was attacked by Russian forces on 03.03.2022, also operates 6 VVER 1000/320 reactors (World Nuclear News, 2022).

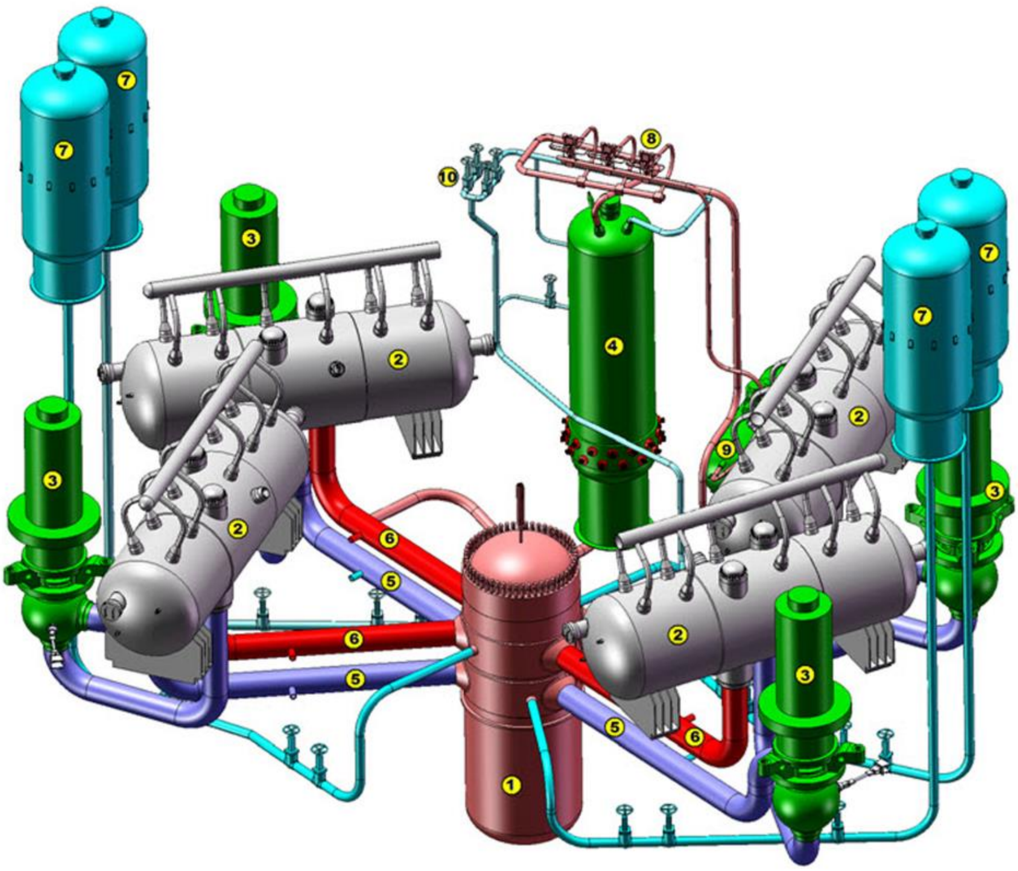
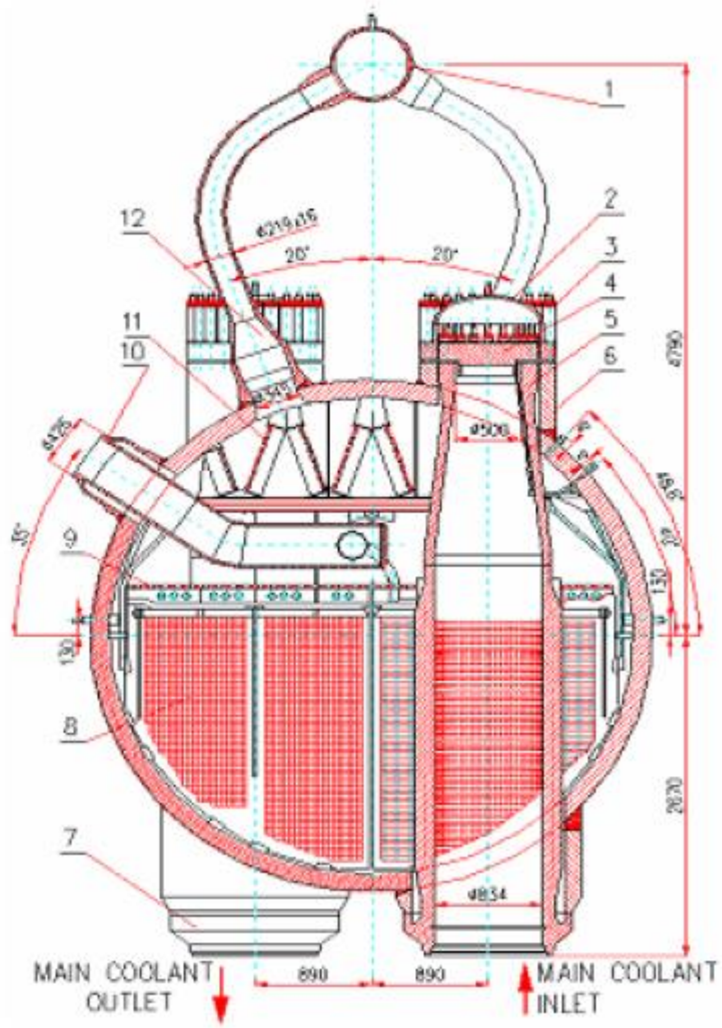


Figure 2 Primary Circuit of VVER 1000 reactor (Tabadar et al., 2016)

Shown in Figure 2 is the primary circuit of a VVER 1000 power plant: 1) reactor, 2) steam generators, 3) main coolant pump, 4) pressurizer, 5) cold leg, 6) hot leg, 7) accumulator, 8) pressurizer pulse safety device, 9) relief tank, 10) injection system. As can be seen in Figure 2, the steam generators of the power plant are laid out horizontally.



1. Steam header
2. Secondary Circuit (SC) cover
3. Nuts
4. Primary Circuit (PC) cover
5. SC header
6. PC header
7. Nozzles
8. Heat exchange tubes
9. Steam pressure equalization unit
10. Feedwater nozzle
11. Separation unit
12. Steam nozzle

Figure 3 Cross section VVER-1000 steam generator (D'Auria et al., 2006)

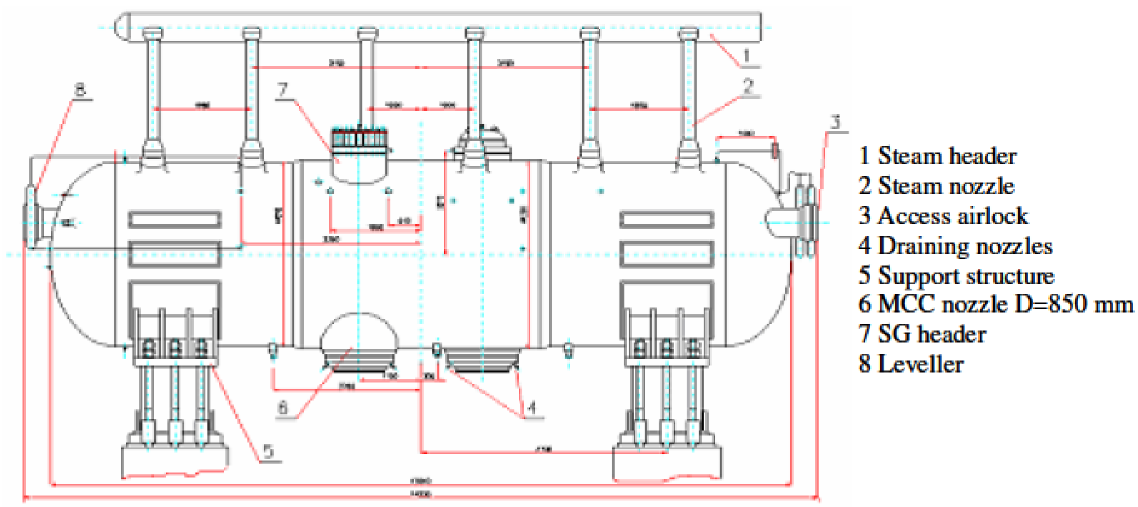


Figure 4 VVER -1000 steam generator - assembled view (D'Auria et al., 2006a)

2.3. The PSB-VVER experimental facility

PSB-VVER is an Integral Test Facility for the VVER reactor situated at the Electrogorsk Research Establishment near Moscow, Russia (Heralecky, 2014). The PSB-VVER Integral Test Facility is used for the experimental analysis of previously determined small and medium break LOCAs, as well as operational transients (Shahedi et al., 2010). The PSB-VVER simulates the VVER reactor with 4 loops with a volume scaling of 1:300 of the VVER-1000-320 at Balakovo power plant (Heralecky, 2014). It models the entire primary system and most of the secondary system, excluding turbine and condenser. The facility follows ”time preserving”, “full pressure”, “full height”, “full-linear-heat-generation-rate” and follows “power-to-volume” scaling laws (D’Auria et al., 2006a).

The plant was built for the following purposes (D’Auria et al., 2006a):

- Obtain experimental data for studying phenomena and processes specified in the verification matrices developed for VVER NPPs
- Assess the efficiency of the existing safety systems and verify engineering approaches proposed in new VVER NPP designs
- Check and evaluate the existing accident management recommendations and procedures
- Fill the experimental database used for thermal-hydraulic code validation

In this facility, the reactor pressure vessel is modelled through separate pipes. Core, DC, UP and core bypass are thus modelled by their own pipes.

In order to be able to simulate the steam generators in the PSB-VVER, they were built as so-called coil steam generators. The heat exchange in this system works in the same way as in the reference power plant, but was adapted to the scaling and space requirements of the PSB-VVER plant (Cherubini et al., 2004). The primary side of each steam generator consists of hot and cold collectors, and 34 steam generator tubes. Above the steam generators in the secondary side is a feed water tube. Main steam lines and the feed water system are also simulated in the secondary side. All steam generators come together to a single steam header (Shahedi et al., 2010).

The reactor core model is located in the upper plenum. The heated section of the reactor model (3.53m) is the same height as in the VVER reactor. There are 168 FRS (Fuel Rod Simulator Bundles) heated indirectly and a non-heated tube in the center grouped in a hexagonal shape (see Fig. 3). The simulated fuel rods are grouped into hexagonal bundles and these bundles are arranged in a triangular shape. The materials used for the FRS are stainless steel for the external cladding, nichrome for the heaters and Periscale as insulator. The initial core power is set at 1.5 MW (D'Auria et al., 2006b).

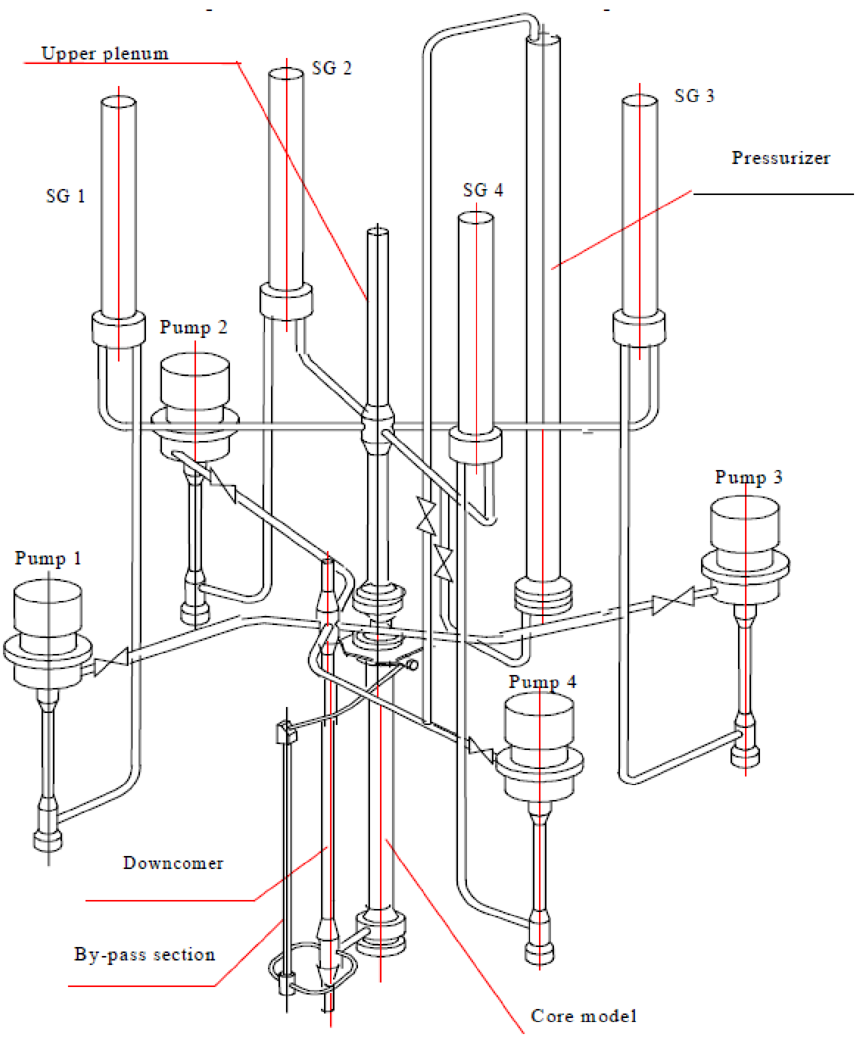
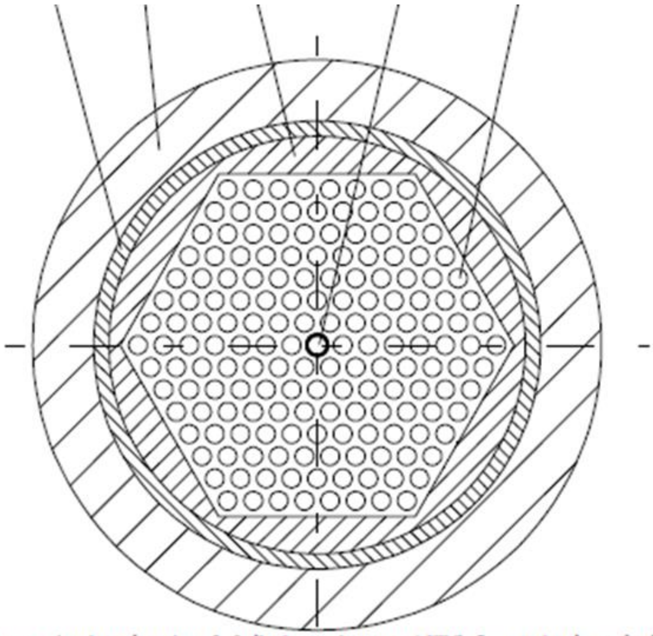


Figure 5 PSB-VVER: general overview of the facility (Cherubini et al., 2004)

Table 1 Main operational characteristics of PSB-VVER comparing to VVER-1000 (Heralecky and Blaha, 2011)

Parameter	Units	VVER-1000	PSB-VVER
Coolant	-	water	Water
Number of circ. Loops	-	4	4
	Primary circuit		
Pressure	MPa	15.7	15.7
Coolant temperature (CL/HL)	deg	290/320	290/320
Coolant flowrate	m^3/h	82485	<280
Core Power	MW	3000	10
	Secondary circuit		
Steam generator pressure	MPa	6.3	6.3
Feed water temperature	deg	220	<270
Thermal power of one SG	MW	750	2.5



1 core simulator housing; 2, 3 displacer elements;4 FRS; 5 central unheated tube

Figure 6 Arrangement of the Reactor Core Model in PSB-VVER / Thermocouples (Del Nevo et al., 2006a)

As shown in Table 1, the construction of the ITF focused on replicating the characteristics of the reference plant as closely and accurately as possible, while also saving space. The test plant has the same height as the reference plant. To make the scaling factors between VVER-

1000 and PSB-VVER easier to understand, the following table (Main design parameters of VVER-1000 and PSB-VVER) shows the Main Design Parameters of the two systems in comparison. The table shows how closely the ITF follows the specifications of the Reference Plant in order to be able to test processes and transients on a smaller scale to make statements about the behavior of the actual power plant.

Table 2 Main design parameters of VVER-1000 and PSB-VVER & Scaling Factor (Del Nevo et al., 2006)

Parameter	VVER-1000	PSB-VVER	Scaling Factor
Scale	1	1:300	
Heat losses, %	0.063	1.8	
Heat power, MW	3000	10	1:300
Primary circuit volume, m^3	370	1.23	1:300
Primary circuit pressure, MPa	15.7	15.7	1:1
Secondary circuit pressure, MPa	6.3	6.3	1:1
Coolant temperature, C°	290/320	290/320	1:1
Core length, m	3.53	3.53	1:1
Number of fuel rods	50856	169	1:300
Core Volume, m^3	14.8	$4.9*10^{-2}$	1:302
Upper plenum volume, m^3	61.2	$20.0*10^{-2}$	1:306
Down-comer volume, m^3	34.0	$11.0*10^{-2}$	1:309
Hot leg volume, m^3	22.8	$8*10^{-2}$	1:285
Cold leg volume, m^3	60	$24*10^{-2}$	1:250
Number of SGs	4	4	
Heat exchange surface between PS and SS, m^2	6115	18.2	1:336
Water volume in SG for primary circuit, m^3	21.0	$6.8*10^{-2}$	1:309
Primary side volume, m^3	20.6	0.0683	1:301
Secondary side volume, m^3	130.76	0.4799	1:272
PRZ Volume, m^3	79	$26.3*10^{-2}$	1:300
Number of hydroaccumulators	4	4	1:1
Number of pumps	4	4	1:1
Volume of hydroaccumulators, m^3	240	$80*10^{-2}$	1:300
Water volume in ACCU, m^3	200	$66.6*10^{-2}$	1:300

As shown in Table 2, the scaling factor of the ITF is in the range of 1:300. The aim of an ITF is always to demonstrate the capability of the codes used. If these capabilities can no longer be demonstrated, then a process of code improvement is triggered.

The following subsection provides an overview of the test facility's safety injection systems in PSB-VVER:

2.3.1. Safety injection systems overview

The safety system of the VVER 1000 consists of a 3 times 100% redundancy design with three nominally identical trains of equipment for each system. (D'Auria et al., 2006a). The high-pressure injection system (HPIS), the low pressure injection system (LPIS) and the containment spray system all draw water from a common containment sump below the plant (NRC, 2021). The PSB-VVER facility simulates the HPIS and LPIS, but not the containment spray system.

2.3.1.1. High pressure injection system (HPIS)

The maximum mass flow of the HPIS is 0.84 m^3/h. The actual mass flow for HPIS and LPIS depends on the pressure in the primary circuit. While the LPIS can feed a larger mass flow into the system, the strength of the HPIS is that it can feed at much higher pressure. Activation of the high-pressure safety system occurs when the difference between the saturation temperature and the temperature of the coolant is less than 10° K, or the pressure in the primary side is < 11 MPa.

A shutdown of the system occurs when one of the following points is reached:

- Coolant T_{sat} – Coolant T > 10 K or
- PS pressure < 1.765 MPa or
- Water level in corresponding tank < 0.25m

2.3.1.2. Low pressure injection system (LPIS)

The LPIS is used for coolant makeup in the event of a large pipe break and can be operated in the Residual Heat Removal (RHR) mode. LPIS in combination with RHR is also used to remove decay heat from the reactor coolant system after shutdown. The system can inject up to a pressure of 2.5 MPa with a max mass flow rate of 2.54 m^3/h. To initiate RHR mode of the LPIS there needs to be a series of operator actions taken from the main, or emergency control room (D'Auria et al., 2006a).

The system activates as soon as the pressure of the primary system drops below 2.43 MPa.

2.3.1.3. Hydro Accumulators (ACCs / HAs)

The plant's 4 hydro accumulators inject water into the Reactor Coolant System at a pressure of 5.9 MPa. 2 of the accumulators inject the coolant into the Upper Plenum and two HAs inject into the Downcomer of the reactor pressure vessel. Each of the accumulators has a capacity of 0.17 m^3. The pressure in the container is maintained by nitrogen gas. To prevent nitrogen gas from entering the primary circuit, fast-acting electric valves close the access to the accumulator after the coolant injection.

2.3.1.4. Make-Up System

The purpose of the makeup system is to provide chemical control and to stabilize the volume of the primary circuit and is essential for Normal Operation / Steady State.

Especially in case of Steam Generator Tube Rupture (SGTR) accidents the coolant pump plays a vital part in stabilizing the primary system and especially supporting the reactor coolant pump (RCP) to help maintaining the RCP seal integrity. The 4 trains of the make-up system are connected to the CL of each Loop and provide a max mass flow of 0.0672 m^3/h or 0.018kg/s (D'Auria et al., 2006a).

2.3.2. RELAP 5/SCDAPSIM

RELAP5 is a best estimate multi-dimensional, two-fluid, non-equilibrium system thermal hydraulics code with options for (International Atomic Energy Agency, 2019):

- Detailed fuel and severe accident behaviour models and correlation for LWR/PHWRs;
- Integrated uncertainty analysis;

- Coupled thermal hydraulics 3D reactor kinetics analysis;
- Alternative fluids and correlations for advanced fluid or reactor analysis.

SCDAP/RELAP5/MOD3.2 (Allison and Johnson, 1989) and RELAP/MOD3.3 (Fletcher and Schultz, 1992) form the basis for RELAP/SCDAPSIM. It is being developed within the framework of the SCDAP Development and Training Programme - SDTP for short (ISS, 2021). As of SCDAPSIM MOD 4.0, the programme has been completely rewritten to FORTRAN 90/95/2000 (Allison et al., 2008). Especially advanced analyses should benefit from this feature, as it makes it easier for users to couple with other analysis packages. All available RELAP/SCDAPSIM versions have been used to analyse manifold nuclear plant designs (Allison and Hohorst, 2010). Although calculations for VVER-1000 were carried out using RELAP/SCDAPSIM, the results of these calculations were proprietary and not publicly available for a long time. This has since changed as there have been publications modelling VVER type reactos in well-known scientific journals (Vryashkova et al., 2015).

SCDAPSIM uses the base code of RELAP and adds features which enhance usability and performance for its users. With each new version of the program, the developers add improvements, whether they numerical or help to improve the overall capability of the code. This has allowed SCDAPSIM to run more stable and faster than its relative RELAP5. Therefore SCDAPSIM can run on weaker equipment than was the case with RELAP 5 (Allison et al., 2008).

In this used RELAP5/SCDAPSIM 4.1.0 BUILD 12/2018 was used.

The interlocking of RELAP5 and SCDAP enables new possibilities in the calculation of transients. In the case of the two phenomena corium meltdown/relocation and core component balooning, there are changes in the hydrodynamic flow area used by RELAP 5. In the case of a narrowing of flow areas that make it difficult for the coolant to flow through the pipes, Darcy's Law for porous debris is applied. The size and porosity of the obstruction are calculated by SCDAP and conveyed to RELAP5 (Siefken et al., 2001). Another example of the interaction between SCDAP and RELAP5 is the release of non-condensable gases (hydrogen and fission products). SCDAP has the ability to release such non-condensable gases, while the RELAP5 part of the code calculates the movements within the pipelines of these gases (Siefken et al., 2001).

The two codes work hand in hand and use the capabilities of the other code to further improve the results. Other examples are (Siefken et al., 2001):

- RELAP5 calculates convective cooling and cooling conditions -> SCDAP calculates the heatup and damage of the reactor core and the lower head.
- The heat conduction model of SCDAP uses the convective heat transfer coefficients for the structures as boundary conditions, which were calculated by RELAP5.
- The SCDAP model for radiation heat transfer uses the RELAP5 calculations for coolant pressure.

The code itself is highly generic and allows for a wide variety of calculations for hydraulic and thermal transients in both nuclear and non-nuclear systems involving steam, water, non-condensable gases, and solid material. The basis is built on a one-dimensional, non-homogenous and non-equilibrium hydrodynamic model. RELAP5 uses six partial derivative balance equations. A non-condensable component in the steam phase and a non-volatile component (boron) in the liquid phase can be calculated. The thermal hydraulic model in RELAP5 solves eight field equations for eight primary dependent variables. These consist of pressure, phasic specific internal energies, vapor/gas volume fraction (void fraction), phasic velocities, noncondenseable quality and boron density (RELAP5 Development Team, 1999).

Certain special processes and phenomena, which would otherwise be difficult to calculate, are calculated in RELAP5 using quasi-steady models. Choking, stratification entrainment/pull-through, abrupt area change, user-specified form loss, cross flow junction, water packing mitigation scheme, counter-current flow limitation, mixture level tracking, thermal stratification, jet junction, magnetohydrodynamics, and variable volume are calculated in RELAP5 using custom models. The use of these models for complex phenomena results in large savings in calculation time (RELAP5 Development Team, 1999). A detailed description of the models can be found in the respective code manuals.

Equations in the code inside control volumes are solved by a partially implicit numeric scheme. Fluid scalar proprieties (pressure, density, void fraction, etc.) are viewed located in the center of the control volume. The fluid vector properties, for example velocities, are located at the junctions and are associated with mass and energy flows between control volumes (D’Auria et al., 2006a) using junctions to show model flow paths. The direction inside a control volume is positive from inlet to the outlet.

Heat flow is also modelled in a one-dimensional way. A staggered mesh is used to calculate temperatures and heat flux vectors. Heat structures and hydrodynamic control volumes are connected by heat flux and are calculated using a boiling heat transfer formulation. This is used for the simulation of pipe walls, heater elements, nuclear fuel pins and heat exchanger surfaces (D’Auria et al., 2006a). Only the RELAP part of the code was used in the course of this analysis.

2.4. Limitations of modelling and predictive capabilities

During the Fukushima nuclear disaster, it was not clear at times whether Tokyo would also have to be evacuated like other parts of the country. This would have resulted in 35 million people losing their homes. The consequences for the global economy would probably have been apocalyptic (Downer, 2015). The prime minister of Japan at the time, Kan Naoto, described the situation as "the existence of Japan as a sovereign state was at stake" (Sekiguchi, 2012).

From a policy perspective it is unthinkable that a state would rely on a technology that could endanger its capital in such a way. For this reason, certain requirements must be met (Downer, 2015):

- The probability of a nuclear disaster must be objectively calculatable
- This probability of an accident must be negligible, as in so low that there is no need for serious policy consideration

States thus base their decisions to build nuclear power plants on the assumption that such accidents can be “ruled out” mathematically with models. This approach is based in part on the 1975 Rasmussen Report (WASH-1400) (Nyman et al., 1996). What was remarkable about this report was that it excluded core meltdowns from the discussion about nuclear energy. Based on reliability calculations, it was shown that such accidents are too unlikely to require far reaching policy changes. Nuclear meltdowns were summarily declared from "credible" to "hypothetical". As soon as the study was published, it was criticized and eventually withdrawn by the NRC (Downer, 2015). The premise, however, remained. Probabilistic reliability calculations are a ubiquitous tool in licensing processes of nuclear power plants. Heads of states and decision-makers are often encouraged to accept these calculations unquestioningly and at face value. Whether certain nuclear power plants could have been built in the same locations, taking into account the risk of core meltdowns, is open to question (Downer, 2013). As M.V. Ramana (2011) points out, nuclear disasters occur much more frequently than calculations would suggest. A rough calculation, which includes previous major accidents and is shown in Downer (2015), comes to about one meltdown every 3,000 reactor years. On the other hand, Areva, a prominent French nuclear manufacturer, states that EPR reactors are expected to experience one core damage incident every 1.6 million years. Westinghouse assumes one core meltdown accident in 2 million years for its AP-1000 reactors (Ramana, 2011). Although reliability analyses and modeling are widely used in the licensing of nuclear power plants, the results of the calculations should not be accepted without question, as the predictive power of these models is limited to some extent. A study by the US NRC (2001)concluded that nuclear reliability assessments regularly underestimate the role of human actions.

Downer (2015) addresses 3 arguments in more detail regarding the limitations of these reliability calculations:

- Framing limitations: arising from the failure of these calculations to model all the variables that contribute to failures.
- Systemic limitations: arising from the failure of these calculations to recognize “failure” and “safety” as an emergent property of complex, tightly coupled technical systems.
- Epistemic limitations: arising from the fact that nuclear reliability calculations imply a level of certainty that is epistemologically implausible given the nature of the tests, theories, and models from which they are derived.

The arguments arising from the Framing Limitations coincide to a large extent with the concept of "Incompleteness Uncertainty", while Epistemic limitations arguments are roughly congruent with “Modeling Uncertainty” (International Atomic Energy Agency, 2014). Downer (2015) calls the issue this:

“They are premised on formal reliability calculations, and it does not take a degree in nuclear engineering to see that these calculations are necessarily “imperfect judgments” rather than “objective facts. The sociotechnical systems required to harness nuclear fission are complex and sophisticated. These systems must operate in an unbounded world that encompasses events far outside the traditional purview of […] Any full calculation of the likelihood of a nuclear accident would have to itemize and quantify all of these variables. And any calculation that sought to demonstrate that accidents are so unlikely as to be negligible (relative to the prospective hazards) would have to perform these tasks with an almost unimaginable level of precision.”

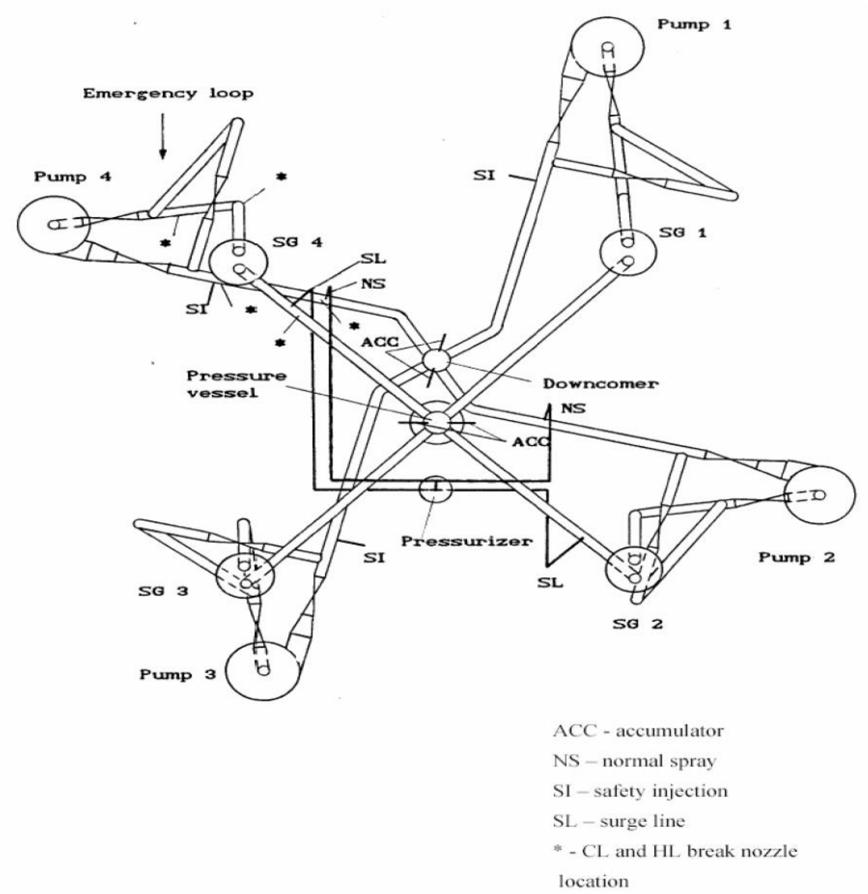


Figure 7 PSB-VVER view from top (D'Auria et al., 2006)

2.5. Simulated accident progression

To make the following events easier to comprehend, an overview of the plant from above can be gained from Figure 4. The figure shows the "reactor core" in the centre, as well as the entire primary system of the PSB, including the four Steam Generators. Instead of anticipated operational occurrences (AOOs), which are also categorised as Condition II and III events by the American Nuclear Society, the accident scenario dealt with here is a so-called postulated accident. This is an accident that is not expected to occur within the lifetime of a power plant, but must nevertheless be taken into account (U.S. NRC, 2007).

The transient, which is modelled in the course of this work, starts with a primary to secondary side leak in the SG of loop number 4. The diameter of the leak in the transient of the VVER 1000 power plant is 100 mm (D'Auria et al., 2006a). To be able to model this scenario in PSB, the diameter of the leak must be scaled to the PSB. In addition, it is assumed that the relief valves of the affected SG have a defect in the closing mechanism and therefore remain open. This scenario also simulates the operator actions aimed at avoiding a lack of coolant. The loss of water due to the break should be kept as low as possible to be able to cool the core for as long as possible. The operator therefore only runs a High Pressure Injection System (HPIS) (D'Auria et al., 2006a). Tables 3 and 4 show the boundary and initial conditions of the primary and secondary sides.

Table 3 Boundary and initial conditions of the transient - primary side (D'Auria et al., 2006a)

Parameter	Reference Value
Primary Circuit	
UP pressure, MPa	15.7
Coolant temperature	278/310[1]
Fuel rod assembly power, kW	1500
Bypass power, kW	15 [2]
Pressurizer coolant level, m	8.6

[1] The first value indicates the temperature at the downcomer entry. The second value refers to the measured temperature at the core exit.

[2] Suitabe to obtain coolant heatup on BP equal to 10 K

Table 4 Boundary and initial conditions - secondary side and hydroaccumulators (D'Auria et al., 2006a)

Secondary Circuit	
Pressure, MPa:	
SG-1	6.27
SG-2	6.27
SG-3	6.27
SG-4	6.27
Level, m:	
SG-1	2.25
SG-2	2.25
SG-3	2.25
SG-4	2.25
Hydroaccumulators	
Pressure, MPa:	
HA-1	5.9
HA-2	5.9
HA-3	5.9
HA-4	5.9
Level, m:	
HA-1	5.6
HA-2	5.6
HA-3	5.6
HA-4	5.6

2.5.1. Simulated operator action after 1800s

One of the challenges of the transient is to minimise the loss of coolant over the break. This is modelled in this scenario by shutting down two HPIS pumps. One HPIS pump continues to operate and feeds water into the CL of the intact Loop No 3. The LPIS pumps are not used. In addition, the HAs are isolated, the plant is cooled down by 60 K per hour using the intact BRU-

A relief valves, the makeup system is activated after 2700s, and the last HPIS pump is shut down by the simulated operator.

According to the Final Report Vol I. of the TACIS Project R2.03/97 (2006) the most challenging aspects for the code are:

- Heat transfer through SS, especially during filling and emptying of the SG
- Simulation of the mass flow rate through the affected BRU-A
- Critical break flow rate

Table 5 shows the most important operational devices and signals of the installation and the corresponding threshold points at which a certain action is to be performed by the code. More detailed information on the test procedure can be found later in the discussion and conclusion part of the master thesis.

Table 5 List of important operational devices and set points of this scenario

Event	Set point		
	Quantity	**Unit**	**Value**
Leak Opening	Time	s	0.0
SCRAM Signal	UP Pressure	MPa	13.7
PRZ Heater switched off	PRZ Level	m	<4.2
Closure of steam dump valve (BRU-A)	UP Pressure	MPa	13.7
Switch from FW to AUXFW	Time of turbine valve closure	s	+0
Auxiliary Feedwater deactivation	Level in SGs	m	<2.25
Auxiliary Feedwater deactivation	Level in SGs	m	<2.3
MCP coast down	Coolant saturation T – coolant T	°C	<10.0
HPIS actuation	Primary side pressure	MPa	<11.0
HPIS deactivation L1 and L4	Time	s	1800
HPIS deactivation L3	Time	s	2750
HA activation	UP Pressure	MPa	<6.0
HA deactivation	Time	s	1800
SSCS activation	Time	s	1800
Makeup system	Time	s	2700

2.6. Nodalisation approach

To be considered as a qualified nodalisation, there are certain requirements, that need to be fulfilled:

- The geometry of the experimental facility must correspond to the geometry of the reference plant
- It must be able to satisfactorily reproduce the steady state operation of the plant
- The system shows satisfying behaviour under time-dependent conditions

A standard procedure for creating qualified nodalisations can be found in Bayless et al. (Fichot et al., 2018). The process is based on two steps.

First, a qualification must be carried out in the Steady State. Then the qualification can be worked on in transient mode. The qualification in steady state is primarily about showing that the nodalisation can reflect the NPP in normal operation. The qualification of nodalisation in transient mode compares the time-dependent results of the calculation with the qualitative and quantitative experimental data from the test facility.

In this validation approach, RELAP5/SCDAPSIM 4.1.0 BUILD 12/2018 was used. The basic information which helped us to create our nodalization approach was taken from the TACIS reports (2006a and 2006b). This report also included an illustration of the nodalisation used for RELAP 5. Because of the similarities between RELAP5 and SCDAPSIM, we were able to create our input deck based on the nodalisation used in the TACIS report (D'Auria et al., 2006a)

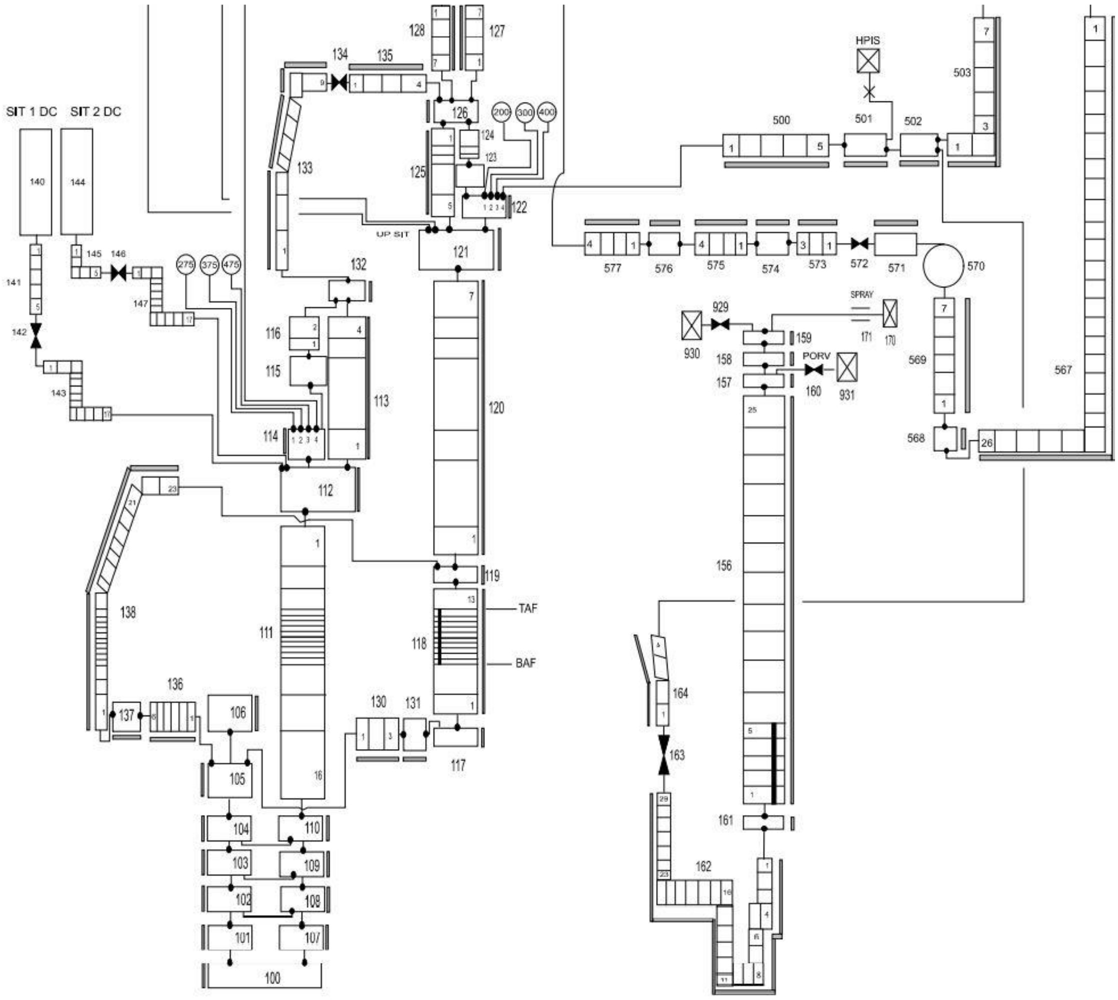


Figure 8 Nodalisation of the core and one loop (F. D'Auria et al., 2006a)

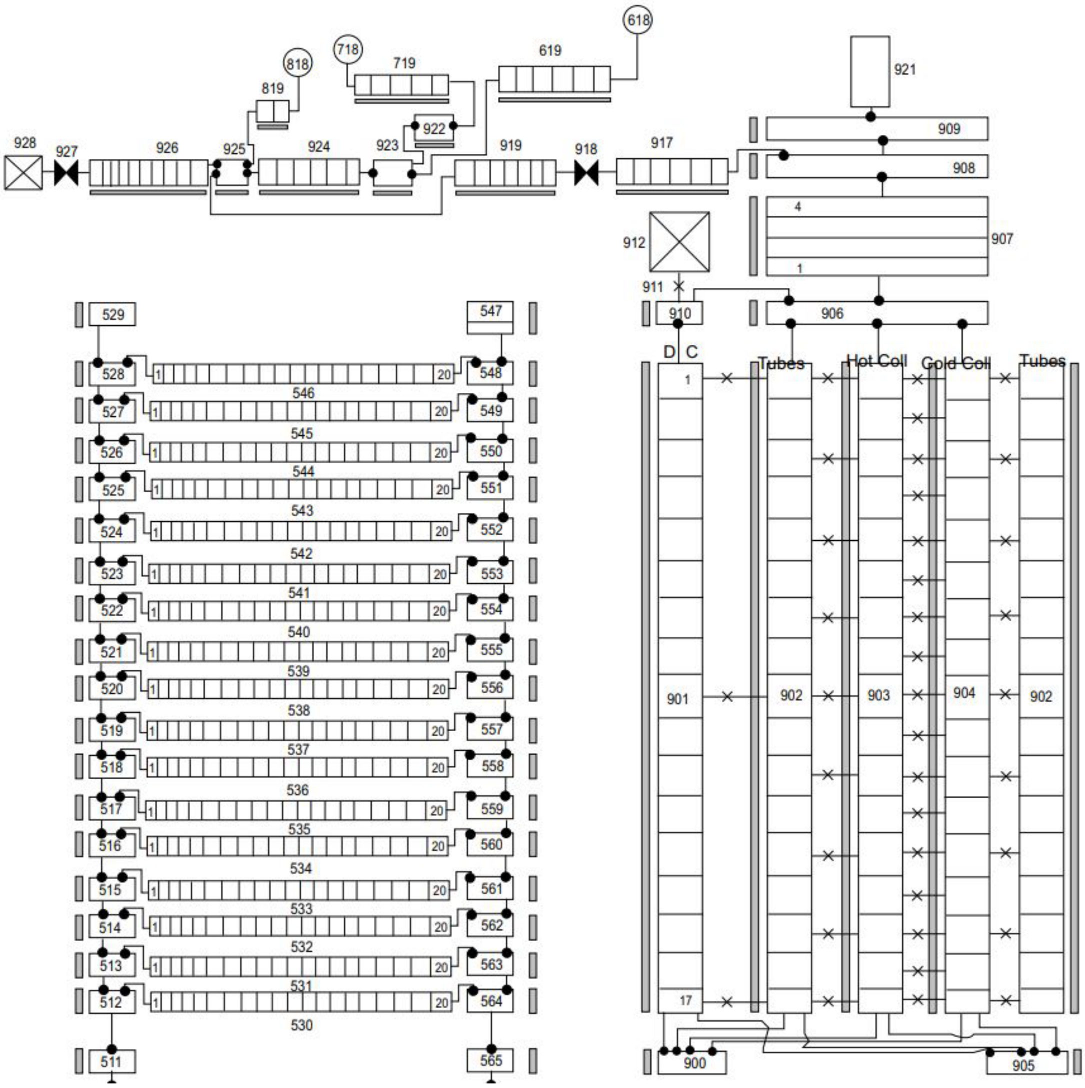


Figure 9 Nodalisation of SG 1 (F. D'Auria et al., 2006a)

Finally, the qualification of a nodalisation can be verified with a quantitative analysis. One of the oldest and most widely used methods is the analysis using FFTBM, which was mentioned earlier. The results of this quantitative analysis are shown later in this thesis.

3. Validation Results

Table 6 shows the main events of the transient. This table gives an impression of the comparison between the experimental data of the TACIS report and their respective counterpart in RELAP5-SCDAP.

Table 6 Main events of the transient (D'Auria et al., 2006a)

Event	Exp Time [s]	RELAP5-SCDAP Time [s]
Break Opening	0	0
Start of main coolant pump (MCP) coastdown in loop 4	13.3	11
Upper plenum pressure at 13.7 MPa (SCRAM Signal sent out)	30.3	22
Main steam isolation valve (MSIV) closure in loop 4 is initiated	35.3	34
Core / Core by-pass power reduction	35.9	20
Transition from feedwater to auxiliary feed water	38.3.	36
MSIV in loop4 completely closed	52	55
Start of MCP coastdown in loop 1 - 3	58.5	17
Start of high-pressure injection system (HPIS) trains in loop 1 + 3	69	73
Start of HPIS injection in loop 4	81	83
BRU-A opening in SG4	94	55
Primary side pressure at 5.9 MPa	662	498
Start of hydroaccumulator operation (primary side cooling and reactivity control)	658	500
Start of SSCS (controlled cooling of secondary side)	1803	1800
Isolation of the hydroaccummulators	1810	1800
Termination of HPIS in L1 and L4	1806	1800
Start of MAKE-UP system	2701	2700
Termination of HPIS in L3	2739	2730
Stop of experiment	4300	4300

The transient can be divided into the following 5 steps. The graphics shown here were also used in our paper for NURETH-19 (Zimmerl et al., 2021):

3.1. Phase 1

First, the simulation is kept in steady state for a prolonged period to show that this is simulated correctly, and that the simulation is in equilibrium without transients. This can be seen in the below Figure 10 before the 0 s mark.

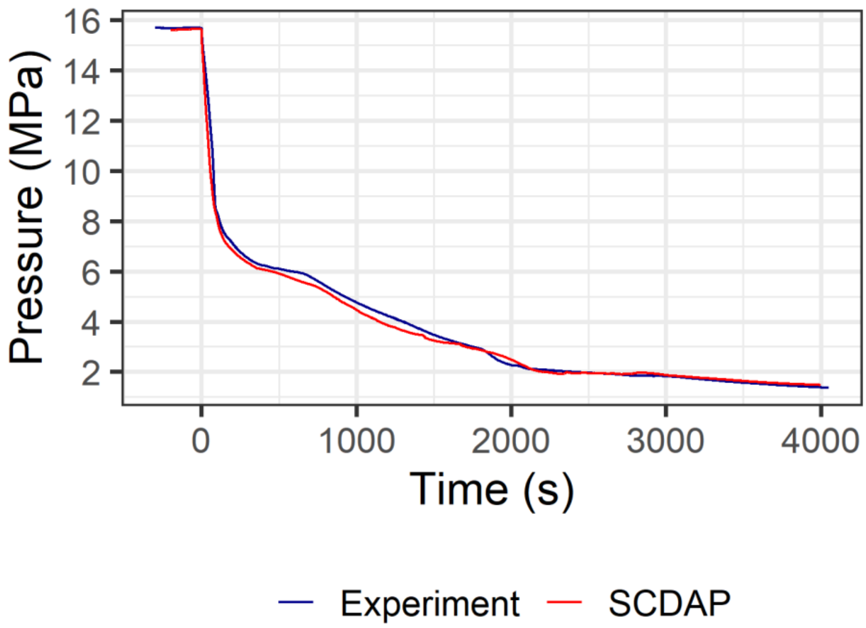


Figure 10 Primary side pressure

The actual transient begins by opening the breakvalve (Figure 12) at T=0 s. The complete opening of the breakvalve, which simulates the break in the SG, takes 6.2 s. The result of the rupture is the transfer of coolant from the primary side to the secondary side (PRImary to SEcondary: short PRISE), which results in a pressure reduction in the PS and an increase in the water level and pressure in the affected Steam Generator (Figure 11Figure 17).

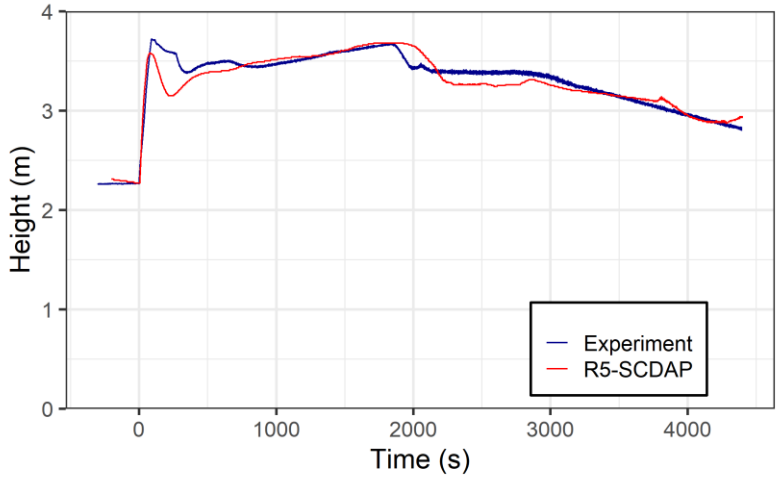


Figure 11 SG 4 Level

This can also be seen in Figure 10. Due to the drop in pressure in the primary side the PRZ heater's power is increased from 12 to 16.5 kW. As soon as the water level in the PRZ, which is measured from the bottom of the PRZ, reaches 4.20 m (see Figure 10), the PRZ heaters switch off. The reactor quick shutdown, SCRAM for short, is activated as soon as the pressure in the Upper Plenum falls below the reference value (= 13.7 MPa). The SCRAM causes the power reduction of the core simulator to start and the core bypass to be activated. Five seconds after the SCRAM, the signal to close the MSIV of the affected loop is sent out.

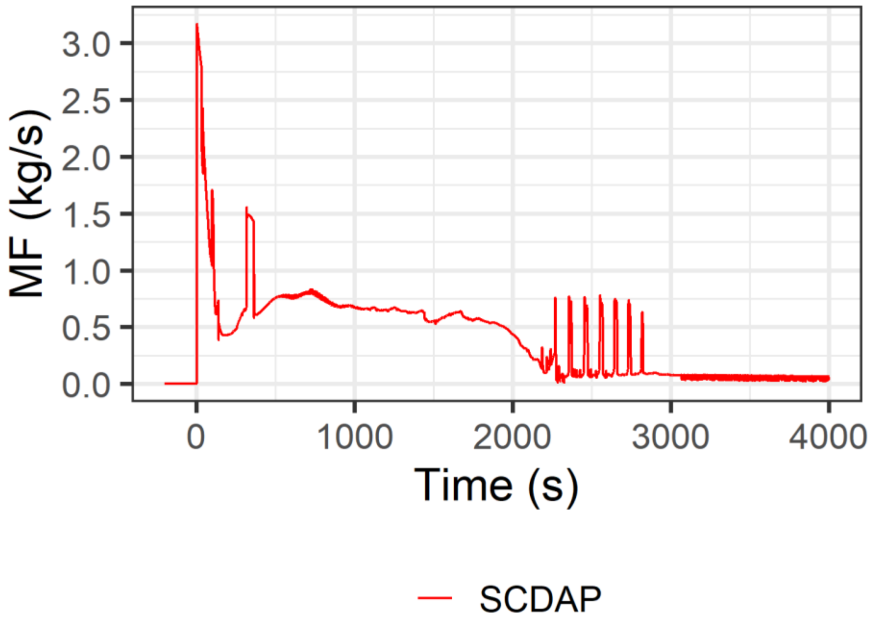


Figure 12 Breakflow from PS to SS

In addition, the change from feedwater to auxiliary feedwater begins, which should ensure that the water level in the intact steam generators remain constant.

When the limit of 2.8 m is reached in the affected steam generator, the main coolant pump of this loop is deactivated. All other main coolant pumps are deactivated when the low saturation margin signal is sent out (see Table 5). Next, the high pressure injection system is activated (see Figure 14 and Figure 16). In the reference NPP, the HPIS prevents core dry out and thus being damaged. This must also be absolutely prevented in normal nuclear power plants, which is why this is also very important in the experiment. The HPIS now injects coolant into loops 1, 2 and 3 after the pressure on the primary side at the core outlet drops below 11 MPa. (see Figure 10, Figure 18 and Figure 19). After 94 s in the experiment and 55 s in the calculation, the BRU-A valve opens in loop 4 (see Table 5). This is caused by reaching the limit of 7.16 MPa in the secondary side (see Figure 17). This in turn means that contaminated steam would be released into the environment at the real NPP (see Figure 15). Activation of HPIS in L4 occurs at t=81s in the experiment, and t=83s in RELAP5-SCDAP (see Tab 6). The difference between experiment and calculation seen in Figure 13 is due to the position of the measuring point in the PSB facility. Since this is not on the ground, we cannot tell at what point the pressurizer runs dry in the experiment.

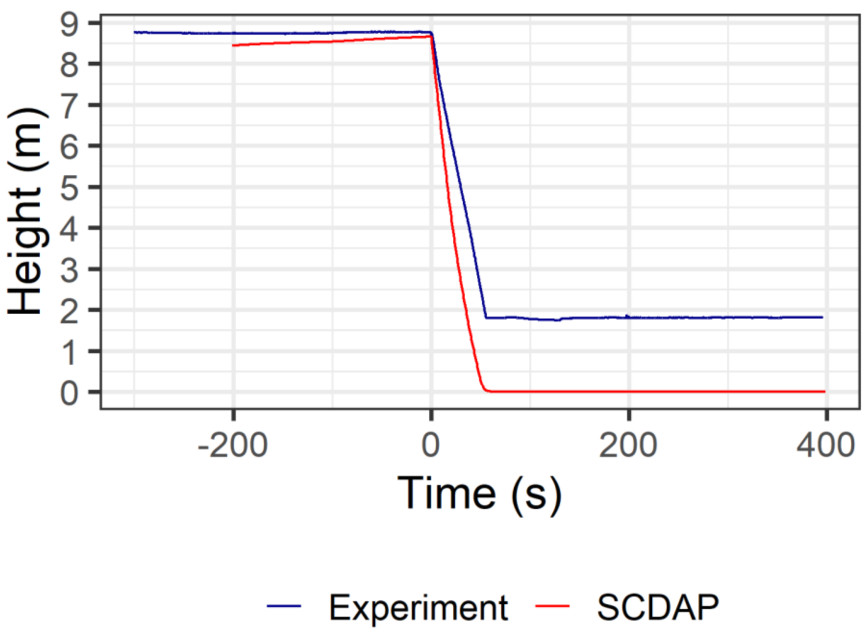


Figure 13 PRZ Level at start of the transient

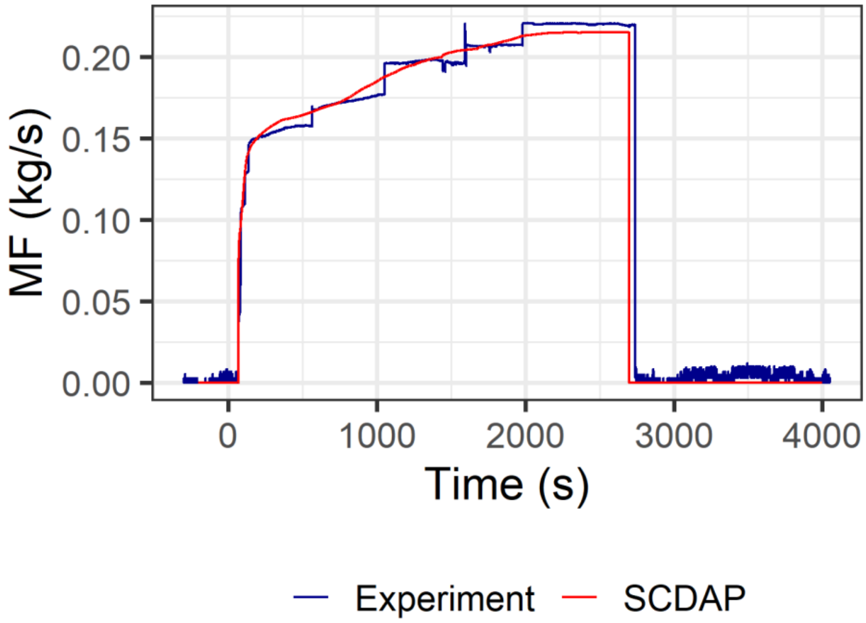


Figure 14 HPIS Loop 2 mass flow

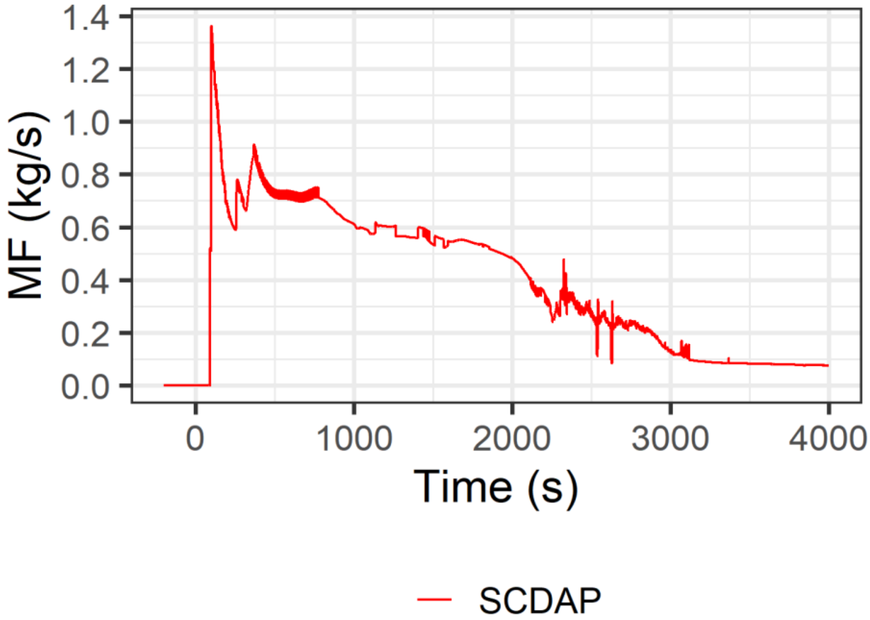


Figure 15 BRU-A SG4 mass flow to Environment

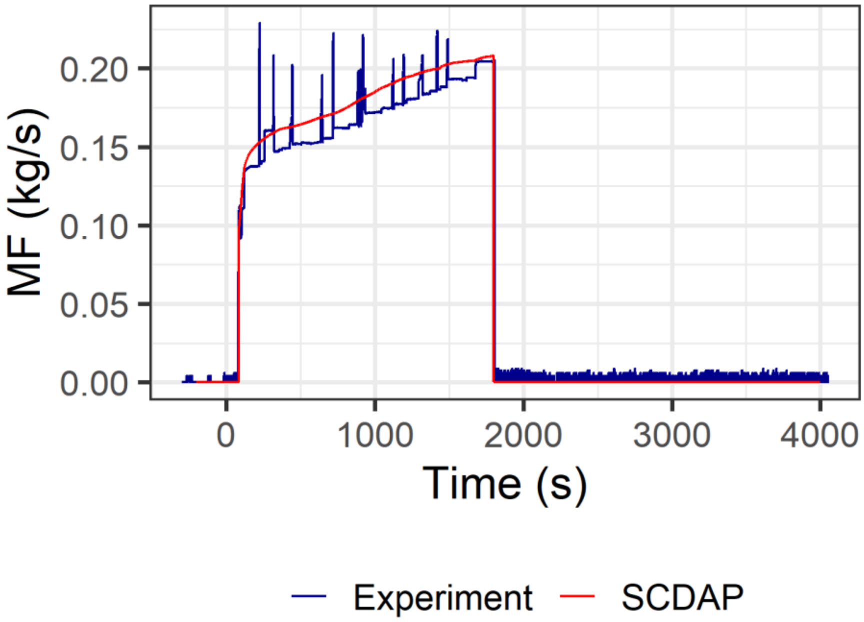


Figure 16 HPIS Loop 3 mass flow

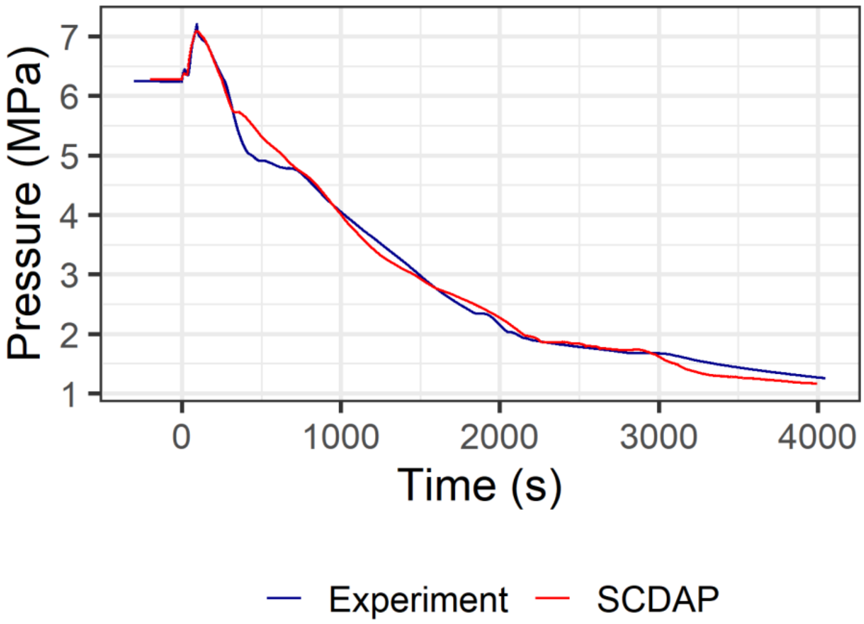


Figure 17 SG4 Pressure

3.2. Phase 2

Unfortunately, the BRU-A does not close as expected, due to a postulated single failure, when the pressure in the SG falls below 6.28 MPa. Thus, steam flows out of the BRU-A valve until the end of the transient as seen in Figure 15. In a real power plant, steam contaminated with fission products would escape from the secondary circuit into the environment via this valve, as the valve is outside the containment.

In the experiment, the main steam isolation valve of the affected loop is closed at 52s while it is closed at 55s in the calculation. The PRZ runs completely dry during this phase (Figure 13).

3.3. Phase 3

In phase 3, the hydroaccumulators (ACC) are activated at 658s in the experiment and at 500s in the calculation as soon as the PS pressure drops below 6 MPa. The earlier time of activation in the calculation is due to the faster pressure loss in the simulation (see Figure 10).

ACC-1 and ACC-3 introduce water in the upper plenum. ACC-2 and ACC-4 discharge their coolant into the DC (see Figure 5).

3.4. Phase 4

In addition to the automated actions of the integral test facility, operator actions are also carried out in this simulation. These accident management measures(AM) start at t=1800s. Exactly 30 minutes after opening of the breakvalve and the start of the transient. The SS cooling system simulates secondary side depressurisation, which is performed in the VVER-1000 using the BRU-A valves (D'Auria et al., 2006a).

Additionally, the HPIS trains in Loop 1 and 4 are disconnected, which means that they stop injecting coolant into the system. This is done to limit the loss of coolant.

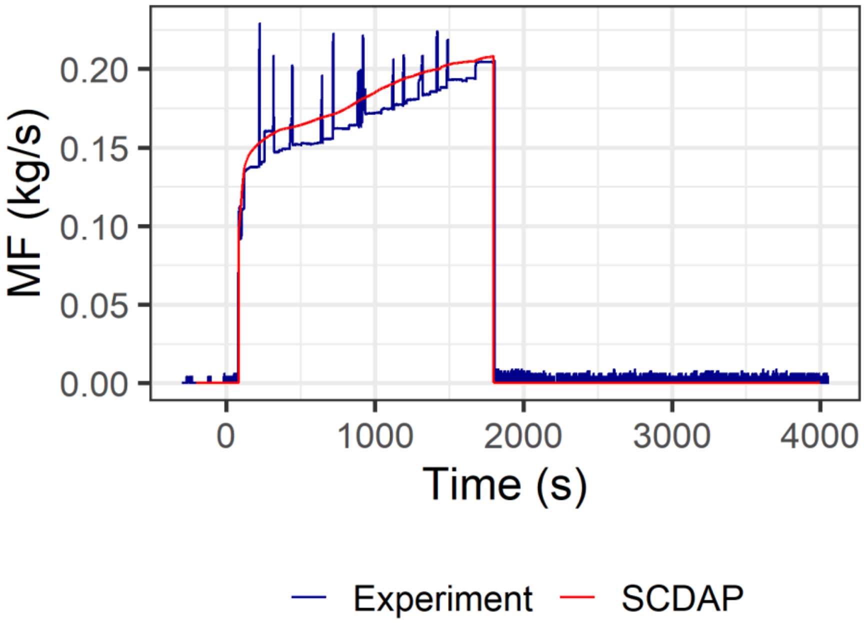


Figure 18 HPIS Loop 2 mass flow

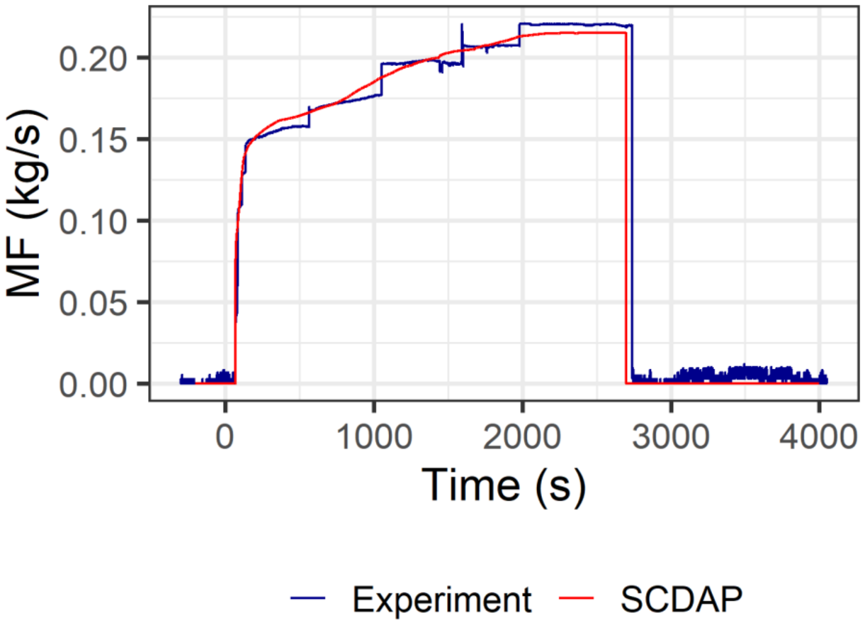


Figure 19 HPIS Loop 3 mass flow

3.5. Phase 5

900 seconds after the start of the AM measures, the procedure is changed. The operator activates the make-up system (see Table 6), which injects coolant into the cold leg of all 4 loops (t=2700s). 2730s after the start of the transient, the last running train of HPIS is disconnected from Loop 3 (see Figure 19).

The aim of the AM measures was to achieve steady cooling of the primary side. This could be achieved with the set AM measures. The transient was finally stopped at 4300s. Based on the simulation and the creation of a constant cooling for PS and SS, it can be concluded that the set AM measures were sufficient and effective. During the transient, 1742 kg of coolant passed through the breakvalve and 1742 kg of coolant was released into the environment through the BRU-A valve.

4. Discussion

In this chapter, the performance of the code is compared with the experimental data. With reference to some important parameters of the transient, it is determined whether the code can reproduce the results of the experiment well.

Absolute pressures, temperature of the coolant, water level within some components, pressure drops, mass flows, fuel rod temperatures are compared and then subjected to a quantitative analysis with FFTBM.

4.1. Coolant Temperatures

To analyse the fluid temperatures during the transient, the values in the lower plenum, the RPV outlet and the SG outlet were considered.

The values achieved in the calculation agree well with the experimental data of the test plant. Only between 500 s and 1800 s the values in the downcomer inlet (DC) were slightly underestimated in the calculation, which led to a small difference being notable there. This can be explained by the difference in distribution of the Emergency Core Cooling System (ECCS) water between experiment and calculation. An analysis with FFTBM also showed that the values for the core outlet (AA = 0.1) are more accurate than the values for the core inlet (AA = 0.17). However, both values are good from the point of view of quantitative analysis and below the threshold value of AA = 0.4. An explanation of FFTBM and the meaning of the values can be found in chapter 1.1.1 of this thesis.

4.2. Absolute Pressures

The absolute pressures of the simulation reflect the experimental data very well (see Figure 17Figure 10Figure 20 Figure 21). The absolute pressures and pressure curves could be reproduced very well by the code, especially in the primary side (see Figure 10). On the secondary side, minor deviations between experimental data and the calculation are evident. However, from an overall perspective, the code reproduces the pressures of the secondary side very well. In the affected loop 4, the pressure in the calculation drops slower (between 300 s and 750 s) than in the experiment (see Figure 17). In the unaffected circuit, the reference pressure from the experiment is not quite reached. (see Figure 20). The pressure also falls slightly faster in the calculation than in the experiment.

The secondary side cooling system, which is used to reduce the pressure after 1800 s on the secondary side, works well in this calculation (see Figure 20).

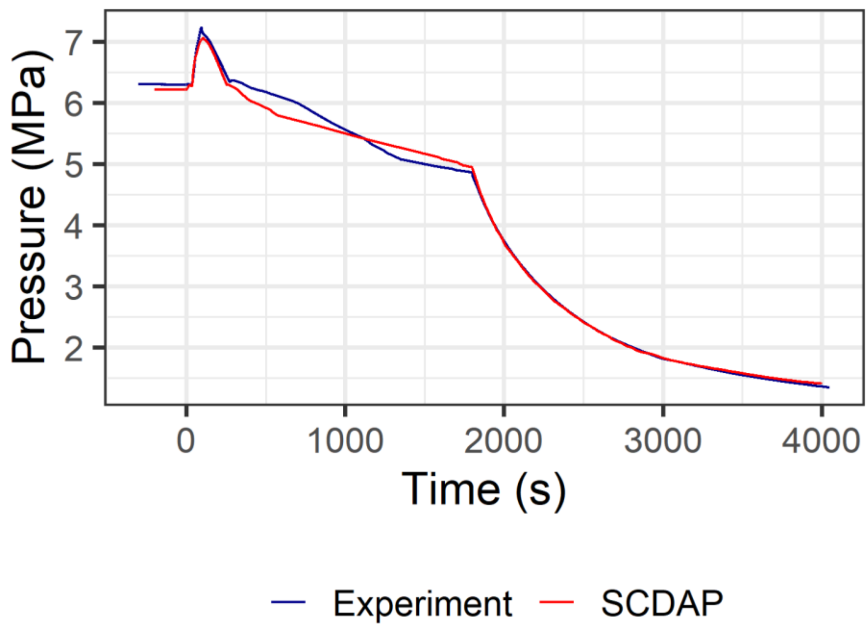


Figure 20 SG 3 pressure

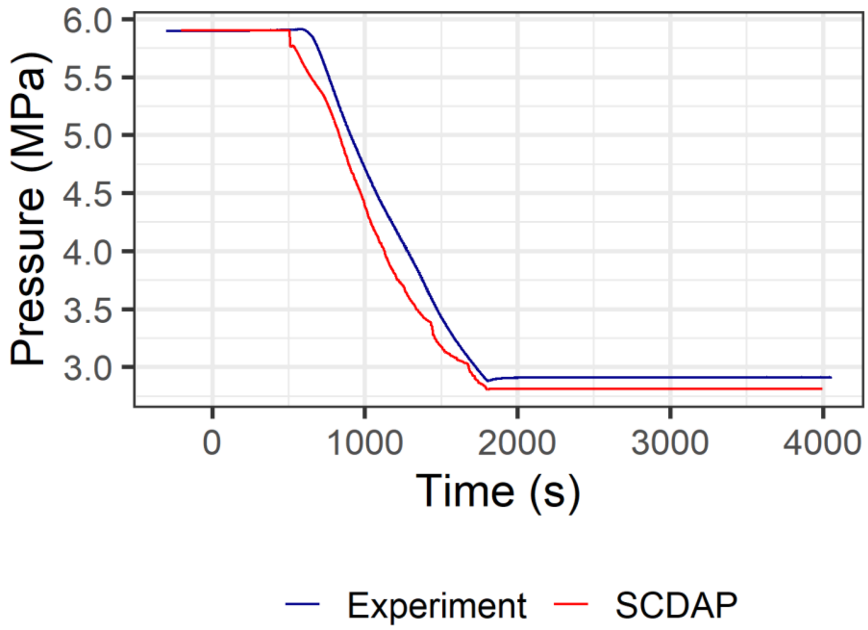


Figure 21 ACC 1 pressure

4.3. Coolant Levels

Since the opening of the ACCs depend on the PS pressure, the ACC empties into the PS a little earlier in the calculation. The pressure of 6 MPa (see Table 5), which marks the trigger point for the ACC, is reached about 150 s faster in the calculation than in the experiment. Since the simulated operator action, among other things, isolates the ACCs from the system, less water enters the system than in the experiment (see Figure 22).

The code was able to satisfactorily replicate the PRZ. Since the measuring point for the water level in the PSB test facility is not at the bottom of the PRZ, it is not possible to know the exact time when the PRZ runs dry in the experiment (see Figure 13). Overall, the curve of the water level within the PRZ during the experiment can be satisfactorily reproduced with the code.

The water level within the affected SG can be reproduced very well. Deviations at the beginning of the experiment and after the start of the operator actions, are visible (see Fig 16.).

As the secondary side cooling system constantly releases steam to the environment through the intact SGs, the water level of these SGs continuously decreases. This is mitigated by the auxiliary feed water (AFW). The simulated operator actions are able to control the level of the unaffected SGs with the help of the auxiliary feed water. There are fluctuations in the water level of the SGs, which then do not correspond exactly to the values of the experiment (see Figure 11, Figure 23 and**Fehler! Verweisquelle konnte nicht gefunden werden.**).

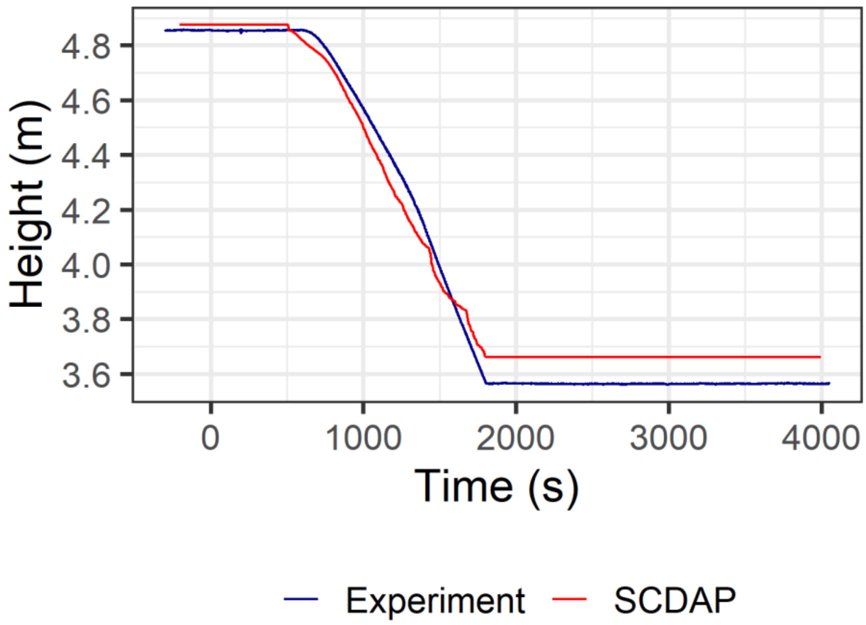


Figure 22 ACC 1 level

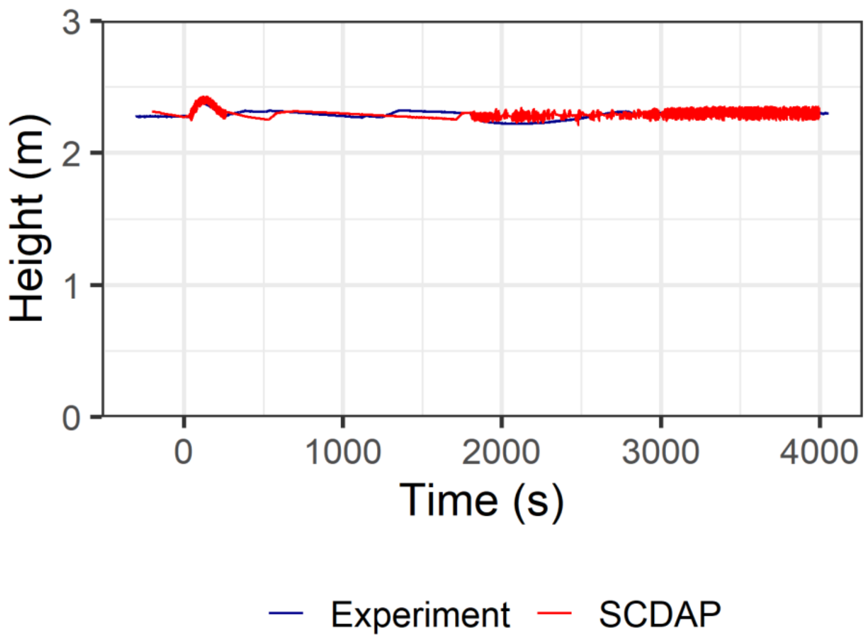


Figure 23 SG3 level

4.4. Pressure Drops

With regard to the pressure drops of the experiment, it can be said that the calculation predicts them well. However, when performing the quantitative analysis with FFTBM, it showed that there is a divergence between the experimental data and the calculation (see Table 7).

4.5. Quantitative analysis of the code with FFTBM

The quantitative analysis with FFTBM (described in chapter 1.1.1of this thesis) was carried out for 21 selected parameters. In addition to the individual parameters, an overall assessment of the results is carried out with the help of FFTBM.

The selection of the parameters and their weighting was taken from the TACIS report (D'Auria et al., 2006b). The respective threshold values for AA, which are essential for the assessment, are also illustrated in the aforementioned chapter of this thesis. The overall result of the entire calculation is satisfactory. The AA value for the entire calculation is well below the threshold for acceptable results. A result of AA = 0.265 is therefore good in this context. The specifications for the PS pressure are stricter. Here, an acceptability threshold of AA = 0.1 is assumed. The quantitative analysis of this parameter yields a result of AA = 0.087, which in turn means that the calculation also falls within the acceptability requirements here. It was even possible to achieve a value that is just below the values of the TACIS report. In the course of this report, however, RELAP5 was used for the same test scenario (D'Auria et al., 2006a). Whether the deviation is due to the usage different programmes or changes in the nodalisation was not investigated.

Table 7 Results of the quantitative analysis with FFTBM

Parameter	RELAP5-SCDAP		FFTBM weighting factor		
	AA	WF	W-exp	W-saf	W-norm
UP – Pressure	0.0870	0.024	1.0	1.0	1.0
SG 1 Pressure	0.0649	0.040	1.0	0.6	1.1
SG 4 Pressure	0.0809	0.030	1.0	0.6	1.1
ACC 1 Pressure	0.0704	0.034	1.0	1.0	1.1
ACC 1 Level	0.1258	0.039	0.8	0.9	0.6
SG 1 Level	0.2584	0.076	0.8	0.9	0.6
SG 4 Level	0.2268	0.038	0.8	0.9	0.6
PRZ Level	0.4215	0.049	0.8	0.9	0.6
DP UP	0.6360	0.052	0.7	0.7	0.5
DP Core	0.8650	0.049	0.7	0.7	0.5
DP loop seal 1	0.8613	0.049	0.7	0.7	0.5
DP loop seal 4	0.6768	0.076	0.7	0.7	0.5
DP SG 1 inlet and top	0.2216	0.014	0.7	0.7	0.5
Heater top level – T	0.5276	0.065	0.7	0.7	0.5
Cladding top level – T	0.4020	0.071	0.9	1.0	1.2
Cladding lower level – T	0.1777	0.039	0.9	1.0	1.2
Cladding middle level – T	0.1440	0.041	0.9	1.0	1.2
Core outlet fluid – T	0.1047	0.029	0.8	0.8	2.4
Core inlet fluid – T	0.1696	0.037	0.8	0.8	2.4
UH coolant – T	1.0360	0.018	0.8	0.8	2.4
ECCS L1 - MF	0.1540	0.324	0.8	0.9	0.9
Overall	0.2650	0.048	-	-	-

5. Conclusion

The validation of codes and the use of models in reactor safety analysis are essential. In order to have the possibility of validating such codes at all, test facilities are needed which either represent the entire power plant or relevant parts of it.

Accidents and incidents in reactors are not as rare as one might think at first glance. Of course, the huge accidents in Chernobyl and Fukushima are well known. In both cases, it was only after the accident that problems were identified that had not previously been taken into account in the safety assessment of reactors. In addition to these major accidents, however, there are many other incidents that lead to a reactor shutdown or worse. The consequences of such a shutdown always depend on the type and duration of this action. How the shutdown of a large power plant can affect the energy prices of individual countries or of an entire continent can currently be seen clearly in the case of the shutdown of the Civaux and Chooz power plants (EDF, 2021). This resulted in an energy loss of at least 1 TWh by the end of 2021. A huge increase in short term energy prices followed. The neighbouring countries were not spared from the resulting increase in energy prices. The timing of the shutdown was very unfavourable, as at that time electricity prices were already heading towards an all-time high and this only further fuelled the situation on the energy market.

As can be seen from this situation, shutting down the reactors for experiments is not an option. In this situation, the above-mentioned test facilities are needed to validate the codes there. Afterwards, the insights gained can be transferred to the reference plants. A nodalisation is then created for the reference plants and the accidents and incidents are modelled on a large scale.

However, as described in 2.3, the selected models have limitations that pose problems in the safety analysis. Therefore, the results of such analyses should not be blindly relied upon, as accidents in the past have repeatedly shown that the uncertainties in the safety analysis cannot simply be dismissed. A publication by the International Nuclear Risk Assessment Group (INRAG, 2021) sheds much more light on the risk we are facing in Europe than I could in the context of this thesis.

The aim of this work was to find out whether RELAP5/SCDAPSIM can reproduce this transient satisfactorily. Although SCDAPSIM has already been validated at many different plants and for a wide range of applications, no work has yet been done to validate the code at the PSB-VVER plant. In the course of the R2CA project, which was funded by European Comission, we were able to look closely at the PSB-VVER plant in Elektrogorsk (RU), which is based on a VVER-1000/320 reactor in Balakovo. The basis of this thesis was the TACIS report (D'Auria et al., 2006a, 2006b), which modelled the PSB-VVER system with RELAP 5, among others.

The aim of this thesis was to answer the following 2 research questions:

- Can a hot header PRISE leakage accident in the PSB-VVER system be satisfactorily reproduced with RELAP5/SCDAPSIM 4.1.0 BUILD 12/2018?

The results of the calculation, as well as their comparison with the experimental data, show that the code used has the ability to reproduce this type of accident. All important phenomena which occurred over the course of this transient during the experiments were also correctly predicted by the calculation.

In addition to the results of the calculation, as well as the subsequent comparison with experimental data, the quantitative analysis with the FFTBM (see 4.5) also shows that the calculations are within the threshold and that the results of the calculation can thus be considered satisfactory.

It can therefore be concluded that RELAP5/SCDAPSIM 4.1.0 BUILD 12/2018 can satisfactorily reproduce the postulated accident at hand.

TH-SYS codes are indispensable for the safety assessment of power plants. However, they are by no means a perfect tool to completely rule out accidents. The fact that accidents involving core meltdowns cannot be ruled out, as suggested in WASH-1400, is undisputed. However, recent nuclear accidents have also shown that nuclear power plants are not as safe as manufacturers such as Gidropress and Westinghouse like to make them out to be. The aforementioned INRAG study (2021) also shows that there are many other challenges and problems for nuclear power plants which are often not discussed by and with the general public. While it is already not unproblematic to draw conclusions from test facilities to normal power plants in the case of new plants, probabilistic and deterministic safety analysis only becomes more difficult and problematic in the cases of plants which have already aged.

- Which accident management measures can be taken by the operator to bring the system back under control and minimise the damage caused?

The accident management measures selected in the course of this transient, which started after 1800s, successfully brought the postulated accident under control and prevented the immediate danger of core damage.

Although steam was released into the environment, which would have caused contamination at the reference plant throughout the transient, it can be concluded that the selected measures were successful. Damage to the core could thus be prevented and over the course of the transient, the plant could be brought back to a stable state.

From the moment the accident management starts and the Emergency Core Cooling System is activated in the unaffected loops, a reduction of the pressure on the secondary side as well as the primary side is visible. At 4000s of the calculation, the pressure on the primary side drops below 1.5 MPa. The temperature at the core outlet is below 200 C at 4000s. The risk of the core running dry was averted throughout the entire transient. The temperature of the cladding at the core simulator never exceeded 322 °C, which means that damage can be ruled out.

All research questions could be answered satisfactorily in the course of the work.

However, when considering such results of calculations, it should never be forgotten that these are best estimate assumptions, which can by no means represent reality with certainty. Calculations are suitable for providing a framework for the safety analysis of nuclear power plants. Beyond that, however, they should not be seen as the absolute truth. As we have already seen with Chernobyl and Fukushima, severe nuclear accidents often result from previously unnoticed or unknown phenomena that appear inside a nuclear reactor. In addition, human error is constantly underestimated in the safety assessment of such plants (US NRC, 2001).

Since a major reduction of the nuclear fleet in Europe is unlikely at present, the only option is to continuously improve the test matrices for new test facilities. Especially the Generation IV reactors often mentioned in public will offer an exciting new field of research in this respect. The ongoing incorporation of new findings into the TH-SYS codes remains indispensable (Glaeser, 2008).

In addition to this thesis, as previously mentioned, a paper from our institute will be presented at NURETH 19 in March 2022 (Zimmerl et al., 2021). With the work at our institute and the first validation of PSB-VVER and RELAP5 / SCDAPSIM, we hope to be able to contribute a small part to improving the safety in these systems. The safety of the global reactor fleet can only be improved through constant validation and the incorporation of new findings and phenomena. Serious accidents at these reactors are unlikely but cannot be ruled out. The damage to the environment which could be caused by a major nuclear accident, especially in a highly populated area, is unimaginable.

I am currently continuing to work on the R2CA project. I work as part of the team at the University of Natural Resources and Applied Life Sciences on the implementation of innovative approaches and devices in the accident management of nuclear accidents.

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List of abbreviations

AA average amplitude

CL - Cold leg

DC downcomer

DP Pressure drop

ECCS emergency core cooling system

EREC Electrogorsk Research and Engineering Center

FFT Fast Fourier Transform

HA hydroaccumulator

HL - Hot leg

HPIS - High pressure coolant injection system

LOCA Loss of Coolant Accident

LPIS - Low pressure coolant injection system

NPP nuclear power plant

NO - Normal Operation

RHR residual hear removal

PRZ pressurizer

PS Primary Side

RPV reactor pressure vessel

SCRAMsafety control rod axe man (reactor shutdown)

SG Steam Generator

SS Secondary Side

SSCS Secondary Side Coolant System

T time

TH		thermal hydraulic
TV		turbine valve
UP	-	Upper plenum
VVER		water moderated water cooled reactor
WF		weighted frequency
ΔF		error function

List of figures / tables