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THE DESIGN AND THERMAL-HYDRAULIC SIMULATION OF A NUWARD-LIKE SMR USING RELAP5

TESI DI LAUREA MAGISTRALE IN
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Abstract

This work delves deep into the design parameters of the E-SMR (small modular reactor design proposed by a consortium of companies within the ELSMOR project under the Euratom treaty of the European Union). A critical comparison of the parameters has been carried out with respect to the design values of the French Nuward SMR. Further on, verification of this data was done using the thermal-hydraulic code RELAP5, by running steady-state and transient simulations. Ultimately, the feasibility of implementing a passive safety system into the existing design of the SMR is studied.

Key-words: Nuward, SMR, RELAP5, Passive-safety

Abstract in italiano

Questo lavoro approfondisce i parametri di progettazione dell'E-SMR (progetto di un piccolo reattore modulare proposto da un consorzio di aziende nell'ambito del progetto ELSMOR ai sensi del trattato Euratom dell'Unione Europea). È stato effettuato un confronto critico dei parametri rispetto ai valori di progetto del Nuward SMR francese. Successivamente, la verifica di questi dati è stata effettuata utilizzando il codice termoidraulico RELAP5, eseguendo simulazioni di stato stazionario e transitorio. Infine, viene studiata la fattibilità dell'implementazione di un sistema di sicurezza passiva nella progettazione esistente dell'SMR.

Parole chiave: Nuward, SMR, RELAP5, Passive-safety

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1 Introduction

Small modular reactors (SMRs) have been one of the most prevalent topics circulating around the nuclear industry for almost a decade now. The question is really: why have these smaller versions of the conventional light water reactors captivated engineers, investors, and industrialists alike? One of the most obvious reasons is to combat climate change. However, there are many other solutions available to solve the same, such as renewables and electrically/battery run machines, but these reactors have many advantages over them and can also be implemented alongside them. Not only do they come with the high energy density of their predecessors but they are also easier to manufacture and assemble, making them quite an attractive option. Looking at the safety issue, it is currently being studied whether these miniature reactors can indeed be safer than the older PWRs and BWRs, the latter of which was under scrutiny after the accident at Fukushima. The ELSMOR project conducted by the European Union (EU) under the Euratom treaty aims to initiate research projects and study the underlying safety mechanisms that may be implemented in SMRs, all this with the help of a consortium of companies from all over the EU.

In the scope of this thesis work, the design parameters of a SMR design (European SMR or simply E-SMR) based on the French Nuward type will be discussed and supported with the help of steady-state and transient simulations using the thermal-hydraulic code RELAP5. Moreover, the possibility of implementing a passive safety system into the model will also be explored.

1.1 The ELSMOR Project

The development of Light Water Small Modular Reactors (LW-SMRs) is internationally at the point where various designs are being proposed to be built all around the world. European stakeholders, including the industry, regulators, support organizations and academic institutions are preparing themselves to respond to this change in the industry. The ELSMOR project aims to enhance the European capability to assess and develop the innovative SMR concepts and their novel safety features. It aims to investigate the safety of LW-SMRs and at the same time narrow down the necessary topics identified by the consortium to be the most important in ensuring the compliance of future SMRs to the safety objectives as established by the amended Directive 2009/71 Euratom. The ELSMOR project is built upon the expertise of the consortium that consists of technical support organizations, technical research centres, industrial partners and universities who have extensive experience with European nuclear safety analysis and the development and implementation of innovative nuclear technologies. The industrial partners include utilities, small medium enterprises (SMEs) as well as the consortium currently developing the French LW-SMR design. The experimental methods, results and data developed or obtained in the framework of the proposed project can be used as a basis for the study and pre-licensing analysis of any system for which similar T/H conditions occur. These outcomes should make the licensing process more fluid and comprehensive, and should also work from the regulator point of view [1].

The objectives of the ELSMOR project have been divided into 8 work packages (WPs). They have been listed as follows:

1. WP 1 focuses on the identification of advanced or innovative safety features of LW-SMRs that potentially pose challenges to establish safety demonstration approaches.

2. WP 2 has the objective of developing methodologies with qualitative and quantitative recommendations to support the safety demonstrations identified in WP 1.

3. WP 3 focuses on core cooling safety functions of integral LW-SMRs. The work is associated with safety analysis, development and assessment codes and models, and specifications of scaling or other requirements for tests and experiments to characterize the most promising passive systems.

4. WP 4 is the development, assessment and validation of analysis methods and tools for the safety demonstration of improved or innovative containment safety features.

5. WP 5 deals with the application of the methodologies and tools developed in WPs 2, 3 and 4. The implementation will be conducted on the safety features of the global design, but efforts will be put on the innovative safety systems that differ from the large PWR. The scope of this thesis work is under WP 5 and the implementation of the passive safety system is explored, in an effort to provide a benchmark for the E-SMR with the software RELAP5.

6. WP 6 focuses on stakeholder interaction and dissemination of results of the project to recommendations to stakeholders. Possibilities to integrate the IAEA's Institutional Strength-in-Depth (ISiD model) will be studied as well.

7. WP 7 targets the education and training of young students and researchers to get into R&D projects related to the concepts of the LW-SMR.

8. WP 8 concerns overall management and administration, coordination and execution of the ELSMOR project, consortium and project meetings.

2 SMR Design Descriptions

This chapter aims to give a detailed insight into why SMRs have been chosen as an alternative to conventional nuclear power plants and the basic design aspects of the same, along with their differences with respect to PWRs and BWRs, the Gen III LWR counterparts. The design parameters of IRIS and NuScale have been discussed in addition to some preliminary values of the Nuward SMR itself, which are developed further in the scope of this thesis.

2.1 Small Modular Reactors

2.1.1 Reasons for Adoption

Small modular reactors are advanced nuclear reactors manufactured on a smaller scale compared to conventional Pressurized Water Reactors or Boiling Water Reactors. Although, like their predecessors, they rely on the nuclear fission to generate heat and hence energy/electricity. They have a power capacity of approximately 300 MWe per built unit, which is around one-third the generating capacity of the PWRs and BWRs. The name suggests three things as shown: Small – physically only a fraction of the size of a conventional nuclear power plant. Modular – possible to assemble systems and components at factories and transport them as a unit to the specified location for installation. Reactors – as mentioned above, the ability of the system to harness the energy produced by nuclear fission [2].

The pressing issue at the moment is that the degree to which nuclear energy can contribute to the energy supply needs of society. This would depend on a couple of factors such as waste generated, safety, security, proliferation concerns and also capital cost of construction and initial investments. The SMRs address several of these issues owing to their smaller size. This in turn ensures shorter construction time and lesser initial capital investment. The components or in some cases the entire reactor (Micro Reactors) can be transported to isolated locations without advanced infrastructure and without a power grid. They can also be used with multiple modules to create a large capacity power plant [3].

According to [2], one of the major challenges today for accelerating energy access is infrastructure, especially the limited grid coverage in rural areas, and the costs of installing grid connections in remote locations. SMRs can be utilized in these scenarios wherein the areas lack sufficient lines of transmission and grid capacity. They can be installed into an existing grid or remotely-off grid, due to its smaller electrical generation capacity, providing low-carbon power for industries and the general population. Microreactors which are a miniature version of the SMRs themselves, are a more attractive option in this case. Their electrical output is on average 10 MWe per unit. Since they have a smaller footprint compared to SMRs, they can be placed in areas which do not have access to clean, reliable, and affordable energy sources. Moreover, they can be used as a backup power supply in case of emergencies or as a substitute to diesel powered generators in rural communities or remote businesses.

Figure 1: The UK SMR Power Plant Concept by Rolls Royce UK [4]

Although not commercialized yet, may SMRs have been developed by companies in different countries. Some of them are in the detailed design and development phase but a majority are still being conceptualized or at a basic design stage. A list of the current SMRs being developed are given by the following table (Table 1) recorded by the International Atomic Energy Agency (IAEA) as of 2020 [5].

There are six types of reactor designs that have been taken into consideration for further development and operation. In addition to the SMRs and Micro Modular Reactors (MMRs), there have been advances with innovation regarding High Temperature Gas Reactors (HTGRs), Fast Reactors (FRs) and Molten Salt Reactors (MSRs).

Table 1: List of current SMR designs [5]

Design	Output (Mwe)	Type	Designers	Country	Status
PART 1: WATER COOLED SMALL MODULAR REACTORS (LAND BASED)					
CAREM	30	PWR	CNEA	Argentina	Under construction
ACP100	100	PWR	CNNC	China	Detailed Design
CANDU SMR	300	PHWR	Candu Energy Inc (SNCLavalin Group)	Canada	Conceptual Design
CAP200	200	PWR	SNERDI/SPIC	China	Conceptual Design
DHR400	400 MW(t)	LWR (pool type)	CNNC	China	Basic Design
HAPPY200	200 MW(t)	PWR	SPIC	China	Detailed Design
TEPLATO RTM	50 MW(t)	HWR	UWB Pilsen & CIIRC CTU	Czech Republic	Conceptual Design
NUWARD	2 × 170	PWR	EDF, CEA, TA, Naval Group	France	Conceptual Design
IRIS	335	PWR	IRIS Consortium	Multiple	Basic Design
DMS	300	BWR	Hitachi-GE Nuclear Energy	Japan	Basic Design
IMR	350	PWR	MHI	Japan	Conceptual Design
SMART	107	PWR	KAERI and K.A.CARE	Korea and Saudi Arabia	Certified Design
RITM-200	2 x 53	PWR	JSC "Afrikantov OKBM"	Russia	Under Development
UNITHERM	6.6	PWR	NIKIET	Russia	Conceptual Design
VK-300	250	BWR	NIKIET	Russia	Detailed Design
KARAT-45	45 – 50	BWR	NIKIET	Russia	Conceptual Design
KARAT-100	100	BWR	NIKIET	Russia	Conceptual Design
RUTA-70	20 MW (t)	PWR	NIKIET	Russia	Conceptual Design

ELENA	68 kW (e)	PWR	National Research Centre "Kurchatov Institute"	Russia	Conceptual Design
UK SMR	443	PWR	Rolls-Royce and Partners	UK	Conceptual Design
NuScale	12 x 60	PWR	NuScale Power Inc.	USA	Under Regulatory Review
BWRX-300	270 -290	BWR	GE-Hitachi Nuclear Energy and Hitachi GE Nuclear Energy	USA, Japan	Pre-licensing
SMR-160	160	PWR	Holtec International	USA	Preliminary Design
W-SMR	225	PWR	Westinghouse Electric Company, LLC	USA	Conceptual Design
mPower	2 x 195	PWR	BWX Technologies, Inc	USA	Conceptual Design
PART 2: WATER COOLED SMALL MODULAR REACTORS (MARINE BASED)					
KLT-40S	2 x 35	PWR in Floating NPP	JSC Afrikantov OKBM	Russia	In Operation
RITM-200M	2 x 50	PWR in FNPP	JSC Afrikantov OKBM	Russia	Under Development
ACPR50S	50	PWR in FNPP	CGNPC	China	Conceptual Design
ABV-6E	6.9	PWR in FNPP	JSC Afrikantov OKBM	Russia	Final Design
VBER-300	325	PWR in FNPP	JSC Afrikantov OKBM	Russia	Licensing Stage
SHELF	6.6	PWR in Immersed NPP	NIKIET	Russia	Detailed Design
PART 3: HIGH TEMPERATURE GAS COOLED SMALL MODULAR REACTORS					
HTR-PM	210	HTGR	INET, Tsinghua University	China	Under Construction

StarCore	14/20/60	HTGR	StarCore Nuclear	Canada/UK /USA	Pre-Conceptual Design
GTHTR300	100 – 300	HTGR	JAEA	Japan	Pre-licensing
GT-MHR	288	HTGR	JSC Afrikantov OKBM	Russia	Preliminary Design
MHR-T	4 x 205.5	HTGR	JSC Afrikantov OKBM	Russia	Conceptual Design
MHR-100	25 – 87	HTGR	JSC Afrikantov OKBM	Russia	Conceptual Design
PBMR-400	165	HTGR	PBMR SOC Ltd	South Africa	Preliminary Design
A-HTR-100	50	HTGR	Eskom Holdings SOC Ltd.	South Africa	Conceptual Design
HTMR-100	35	HTGR	Steenkampskraal Thorium Limited	South Africa	Conceptual Design
Xe-100	82.5	HTGR	X-Energy LLC	USA	Basic Design
SC-HTGR	272	HTGR	Framatome, Inc.	USA	Conceptual Design
HTR-10	2.5	HTGR	INET, Tsinghua University	China	Operational
HTTR-30	30 (t)	HTGR	JAEA	Japan	Operational
RDE	3	HTGR	BATAN	Indonesia	Conceptual Design
PART 4: FAST NEUTRON SPECTRUM SMALL MODULAR REACTORS					
BREST-OD-300	300	LMFR	NIKIET	Russia	Detailed Design
ARC-100	100	Liquid Sodium	ARC Nuclear Canada, Inc.	Canada	Conceptual Design
4S	10	LMFR	Toshiba Corporation	Japan	Detailed Design
microURANUS	20	LBR	UNIST	Korea	Pre-Conceptual Design
LFR-AS-200	200	LMFR	Hydromine Nuclear Energy	Luxembourg	Preliminary Design
LFR-TL-X	5 – 20	LMFR	Hydromine Nuclear Energy	Luxembourg	Conceptual Design

SVBR	100	LMFR	JSC AKME Engineering	Russia	Detailed Design
SEALER	3	LMFR	LeadCold	Sweden	Conceptual Design
EM2	265	GMFR	General Atomics	USA	Conceptual Design
Westinghouse LFR	450	LMFR	Westinghouse Electric Company, LLC.	USA	Conceptual Design
SUPERSTAR	120	LMFR	Argonne National Laboratory	USA	Conceptual Design
PART 5: MOLTEN SALT SMALL MODULAR REACTORS					
Integral MSR	195	MSR	Terrestrial Energy Inc.	Canada	Conceptual Design
mTMSR-400	168	MSR	SINAP, CAS	China	Pre-Conceptual Design
CA Waste Burner 0.2.5	20 MW (t)	MSR	Copenhagen Atomics	Denmark	Conceptual Design
ThorCon	250	MSR	ThorCon International	International Consortium	Basic Design
FUJI	200	MSR	International Thorium Molten-Salt Forum: ITMSF	Japan	Experimental Phase
Stable Salt Reactor - Wasteburner	300	MSR	Moltex Energy	UK/Canada	Conceptual Design
LFTR	250	MSR	Flibe Energy, Inc.	USA	Conceptual Design
KP-FHR	140	Pebble-bed salt cooled reactor	KAIROS Power, LLC.	USA	Conceptual Design
Mk1 PB-FHR	100	FHR	University of California at Berkeley	USA	Pre-Conceptual Design
MCSFR	50 - 1200	MSR	Elysium Industries	USA/Canada	Conceptual Design
PART 6: MICRO MODULAR REACTORS					
Energy Well	8	FHTR	Centrum výzkumu Řež	Czech Republic	Pre-Conceptual Design

MoveluX	3 – 4	Heat Pipe	Toshiba Corporation	Japan	Conceptual Design
U-Battery	4	HTGR	Urenco	UK	Conceptual Design
Aurora	1.5	FR	OKLO, Inc.	USA	Conceptual Design
Westinghouse eVinci	2 – 3.5	Heat Pipe	Westinghouse Electric Company, LLC.	USA	Under Development
MMR	5 - 10	HTGR	Ultra-Safe Nuclear Corporation	USA	Preliminary Design

2.1.2 Differences in Design Aspects

In comparison to existing reactors, proposed SMR designs are generally simpler, and the safety concept for SMRs often relies more on passive systems and inherent safety characteristics of the reactor, such as low power and operating pressure. This means that in such cases no human intervention or external power or force is required to shut down systems, because passive systems rely on physical phenomena, such as natural circulation, convection, gravity, and self-pressurization. These increased safety margins, in some cases, eliminate or significantly lower the potential for unsafe releases of radioactivity to the environment and the public in case of an accident [2].

However, in the design of the IRIS and Nuward SMRs, pumps are still used as active systems, and the reasons for this will be discussed in the next subsections.

Some more major differences in terms of structural design are related to the arrangement of internal components. Due to its compact design, some of the components have been placed inside the main reactor pressure vessel (RPV). The RPV comprises the steam generator, pumps and pressurizer hence eliminating the need of a large primary circuit. Further explanations with regards to this will be given in Chapter 3, when the Relap models for each component are discussed.

2.1.3 Economics

One of the most important features that make SMRs an attractive investment option, is the economics itself. Not to go too much in detail, there are a few major aspects that will be covered in this subsection. The principal quantities that are of most utmost relevance according to [3] are the financial yield parameters such as net present value [$], internal rate of return (IRR [%]), LUEC (economic evaluation production cost) [$/MWh], and upfront investment [$]. Secondary values that are quite important as well are the risk associated with the entire investment and cash flow, capital cost [$/MWe (or) kWe] and construction time [years]. Due to the need for mass manufacturing of the SMRs, there are high fixed costs of capital and hence the economy of scale does not necessarily favour SMRs over large reactors. A study was conducted by [6] to solve this issue by considering other advantages of the SMR for example, the lower capital risk which brings it into competition. What truly makes it an investment option to consider is the smooth cash flow rate and the lower upfront investment. Typical values for capital costs were $5000/kWe for SMART (SMR design by the Korean Atomic Energy Research Institute (KAERI)) and $4000/kWe for NuScale, at the time. Looking at lead times and governmental support, the larger reactors have a longer lead time making the financing/maintenance costs more than 40% of capital costs. But this is not a problem if the government reimburses the amount of the utilities. However, SMRs have a much shorter lead time still making it the better investment option. For developed countries, this is a necessary step forward and in developing countries it is possibly the only option since grids cannot handle more than 1000 MWe single units (like the IRIS) [3].

As per OECD-NEA [7], the use of nuclear technologies is very competitive with other energy sources (coal-fired plants, gas-fired plants and renewables). Without the quantity of carbon tax [$/MWh] being considered SMRs don't seem to be the go-to option looking at investment costs and the net present value (NPV), Figure 2. Even if SMRs have to be implemented they can only be used as a complementary option to a combined cycle gas turbine power plant (CCGT), as CCGTs are more reliable and have lesser upfront costs, making them suitable for deployment for base load purposes [8].

As a comparison to large reactors, SMRs have a justifiable use in certain niche markets. For example, when the energy demand is around 1 – 3GWe. In this case, the "economy of multiples" would compensate for the economy of scale issue. For lesser power requirements in the range 300MWe – 1GWe, and for limited market space availability, SMRs are a good option and can even compete with coal plants and CCGTs [9]. Another scenario is when resources are limited in terms of water or if it is a disaster-prone area [10].

SMRs are a viable option for newcomers in the industry as well, since smaller reactors do not need a significant amount of prior knowledge and experience as compared to building a large reactor power plant [8].

Lastly SMRS could act as replacements for recently demolished/decommissioned fossil fuel power plants. A typical scenario suggested by [9], is the replacement of an old coal power plant due to high carbon tax towards the end of its lifetime and due to stricter environmental laws.

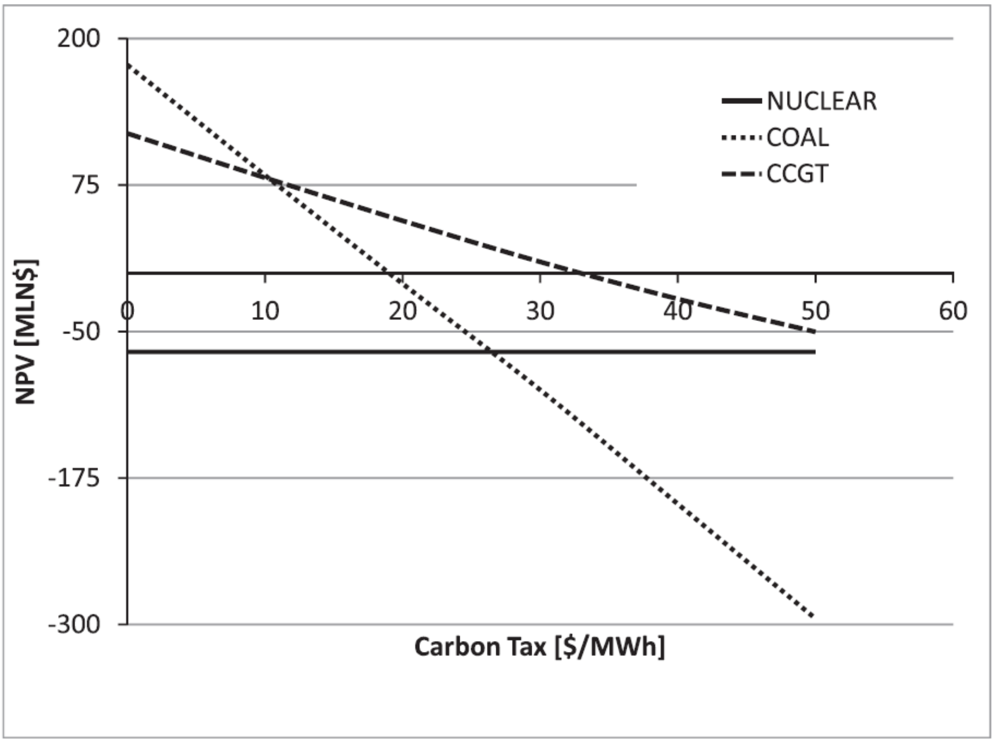

Figure 2: Impact of the carbon tax on the NPV, and comparison between different energy sources [8].

2.2 IRIS

The International Reactor Innovative and Secure (IRIS) is one of the SMRs mentioned in Table 1, that has been designed by an international consortium of companies, organizations, and institutions (totally 19 from 10 different countries). It is an advanced, integral, light-water cooled, pressurized reactor of medium generating capacity (1000 MWt or about 335 MWe). It has been under development ever since 1999, with Westinghouse Electric leading the project.

The IRIS has the ideal size for smaller energy grids as it allows introducing sequentially single modules in regions only requiring a few hundred MWs at a time. IRIS can also be deployed in multiple modules in areas requiring a larger amount of power increasing with time, thus fulfilling the needs of larger, developed countries as well. A top-down economic analysis which has been conducted for the reactor shows that the cost of generated electricity is competitive with other nuclear and non-nuclear power plants. With its features of moderate size and short construction time, IRIS is able to significantly reduce the financial burden and present a viable solution for the market with limited investment capital [11].

2.2.1 Design Specifications

The main design parameters of the IRIS have been listed in the following table considering that it has an integral primary circuit configuration.

Table 2: IRIS Design Parameters [11]

General Plant Data	
Core thermal power	1000 MWt
Power plant output, net	335 MWe
Nuclear Steam Supply System	
No. of coolant loops	Integral primary system
Steam temperature, pressure	317°C, 5.8 MPa
Feedwater temperature, pressure	224°C, 6.4 MPa
Reactor Coolant System	
Primary coolant flow rate	4700 kg/s
Reactor operating pressure	15.5 MPa
Core inlet/ outlet (riser) temperature	292°C / 330°C
Reactor Core	
Active core height	4.267 m
Fuel inventory	48.5 tU
Average linear heat rate	10.0 kW/m
Fuel material	Sintered UO2
No. of fuel assemblies	89
Rod array	Square, 17x17
No. of fuel rods/assembly	264
Outer diameter of fuel rods	9.5 mm
Enrichment	Up to 4.95 wt% U-235
Equilibrium cycle length	30-48 months
Average discharge burnup	Up to 60,000 MWd/tU
Reactor Pressure Vessel	
Cylindrical shell diameter	6.21 m
Wall thickness of shell	285 mm
Total height	21.3 m
Steam Generators	
Type	Helical coil tube bundle, once through, superheated
Number	8
Thermal capacity (each)	125 MWt

Reactor Coolant Pump	
Type	Spool type, fully immersed
Number	8
Pump head	19.8 m
Primary Containment	
Type	Pressure suppression, steel
Geometry	Spherical, 25 m diameter
Design pressure, temperature	1300 kPa, 200℃

The reactor pressure vessel (RPV) of the IRIS is an integral configuration which houses the fuel and control rods, but at the same time the components of the reactor coolant system (RCS). This includes: eight spool-type coolant pumps, eight modular helical coil steam generators, a steel reflector surrounding the core for improved neutron economy and internal shielding. Circumferential steel plates can be installed beyond the barrel in the downcomer region to enhance the shielding phenomenon. A pressurizer is located on top of the RPV head. This integral design eliminates the possibility of a loss of coolant accident as a design basis event [12].

The primary coolant flow path begins with the water moving upwards from the core and through the riser. The coolant is then sucked by the coolant pumps in the annular plenum. The flow is then directed towards the helical coil steam. The flow then continues through the annular downcomer and reaches the core inlet once again via a lower plenum completing the primary circuit.

The descriptions of the major components in the RPV are given below:

- **Pressurizer:** The pressurizer is integrated into the upper head of the RPV. It is defined by an insulated inverted top-hat structure and separates the circulating subcooled liquid from the saturated liquid in the pressurizer itself. Heater rods are present in the bottom portion of the inverted top-hat structure and are positioned outside the control rod drive mechanism (CRDM) drive line. The inverted top-hat contains holes which allows insurge and out surge of water to and from the pressurizer. Since the pressurizer is built into the RPV it contains more space than a conventional pressurizer used in a power plant. The water and steam volumes are respectively 71 m^3 and 49 m^3. The steam volume being almost 1.6 times bigger than the pressurizer steam space in the AP1000, although the core power is less than 1/3. This large ratio between the steam volume and the power is the main reason why there do not exist any sprayers to prevent pressurizer safety valves from opening up due to any design basis heating transient.

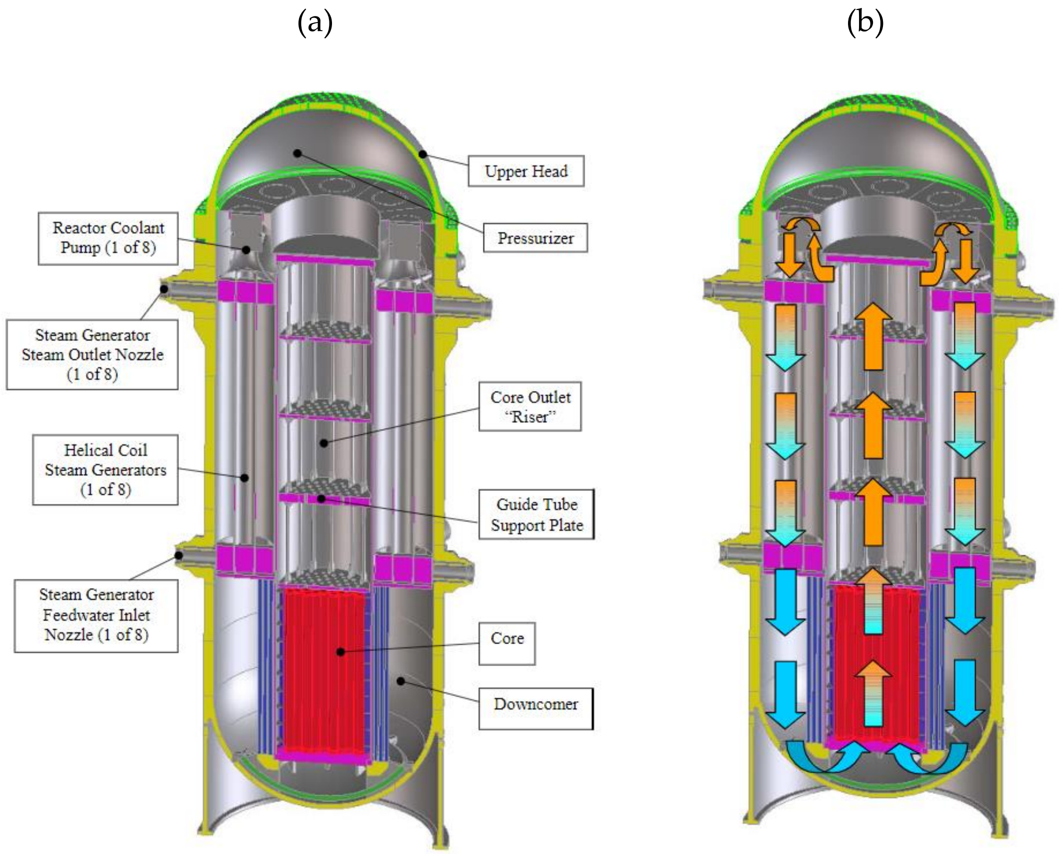

Figure 3: The IRIS RPV showing (a) the components and (b) the main coolant flow path.

- **Reactor Core:** The IRIS core and fuel characteristics are similar to those of a conventional Westinghouse PWR design. A single fuel assembly consists of 264 fuel rods. The central position is occupied by in-core instrumentation and 24 positions have guid thimbles for the control rods. The fuel assemblies themselves follow a 17x17 design. In total there 89 fuel assemblies are used with a fuel active length of 4.267 m for a low power density. The fuel used is UO_2, enriched to about 4.95% of U-235. Another important characteristic to be noted is that the fission gas plenum of the fuel rod is almost twice that of the PWR design hence eliminating concerns about internal overpressure.

- **Reactor Coolant Pumps (RCPs):** The pumps that are employed in IRIS are of an advanced spool-type. These are usually used in applications requiring high flow rates and a low developed head. The pump is located completely inside the RPV, apart from small penetrations for the necessary electrical cables and supply of cooling water for the pump components. The spool-type geometric configuration maximizes the rotating inertia and they also tend to have a high run-out flow capability. Both these features prevent the consequences of a Loss-

Of-Flow-Accident (LOFA). The IRIS integral design and low primary coolant pressure drops allow this pump type to be used even though it has a low developed head.

- **Steam Generators (SGs):** The SGs used in IRIS are the once-through helical coil tube bundle type with the primary fluid outside the tubes [13]. Eight SG modules are placed in the annular plenum region of the RPV, in between the riser and the RPV. Each of the SGs contain an inner column which supports the entire structure (tubes, lower feedwater header and upper steam header). Every module has 656 tubes designed specially to handle the RCS pressure. The performance characteristics (thermal, vibration, pressure losses) were investigated along with the determination of the operating characteristics domain for stable operation.

In Section 2.4, the influence of the design of IRIS on the E-SMR design will be discussed further, especially with respect to the pressurizer.

2.3 NuScale

The NuScale Nuclear Steam Supply System (NSSS) is a passive small modular pressurized water reactor designed by NuScale LLC. The design consists of an integral power module with a reactor core, two helical-coil steam generator tube bundles, and a pressurizer contained within a single reactor vessel much like the IRIS, along with a containment vessel immediately surrounding the reactor vessel. This design again eliminates the need for an external primary circuit. The NuScale compared to the IRIS depends on the natural circulation mechanism and hence does not require pumps.

A NuScale power plant generally consists between one to 12 Nuclear Power Modules (NPMs). Each NPM is rated at 160 MWt with an output of approximately 50 MWe (variable). The electrical output is usually dependent on environmental conditions and the type of purpose it serves [14].

2.3.1 Design Specifications

The following table, gives us some parameter data of an overall NuScale power plant having 12 NPMs.

Table 3: Characteristics of a 12-unit plant [15]

Overall Plant	
Net nominal power rating	540 MWe
Net station efficiency	> 28%
No. of power units	12
Nominal plant capacity factor	92% - 95%
Power Unit	
No. of reactors	One
Net nominal power rating	45 MWe
Steam generator number	Two, independent tube bundles
Steam generator type	Vertical helical tube
Steam cycle	Rankine
Turbine type	3600 rpm
Thermal power rating	160 MWt
Operating pressure	1850 psig
Reactor Core	
Fuel	UO2 (< 4.95% enrichment)
Refuelling intervals	24 months

The components present in he NuScale have been described briefly hence:

- **Reactor Pressure Vessel:** The RPV consists of a steel cylinder having a height of approximately 45 ft (≈ 13.716 m) and has been designed for an operating pressure of about 1850 psig (≈ 12.76 MPa). The upper and lower heads are elliptical similar to the IRIS, and the lower portion of the vessel has flanges to provide access for refuelling. The top of the upper head of the RPV provides support for the CRDMs. Nozzles on the upper head provide connections for relief valves, reactor vent valves and the pressurizer spray piping. A graphical view can be seen in Figure 4.

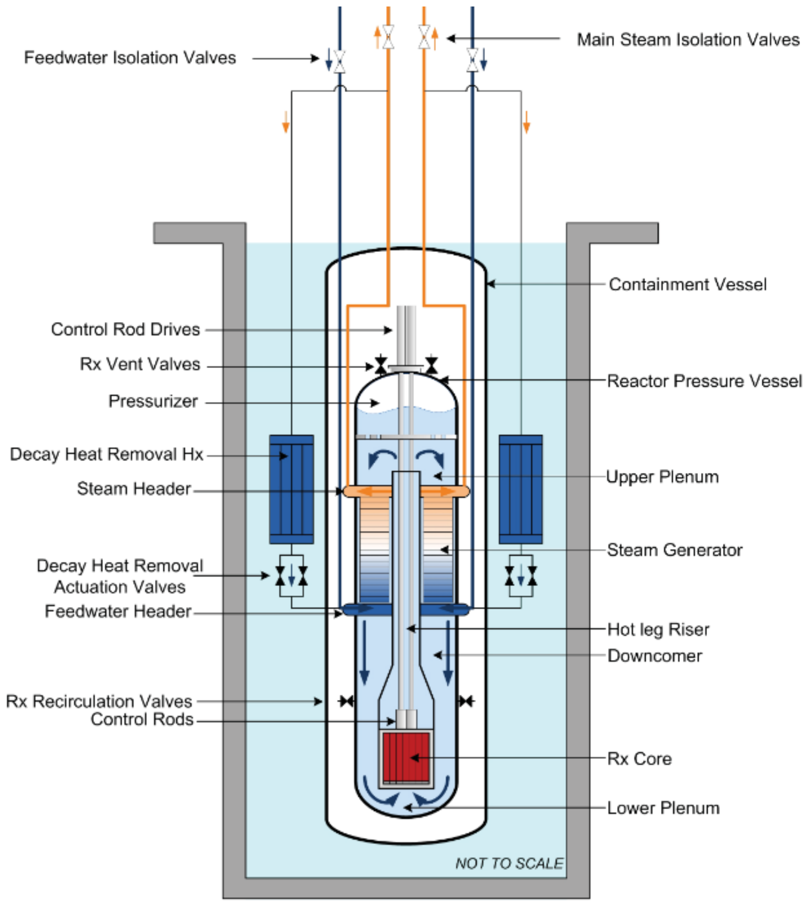

Figure 4: 2-D schematic of a single NuScale unit

- **Pressurizer:** This component as always provide the primary means for controlling the RCS pressure and needs to maintain a constant pressure during operation. A separator plate exists between the upper plenum and the pressurizer to prevent mixing between the saturated pressurizer water and the subcooled primary circuit water. The plate has orifices to regulate the in and out surge of water and acts as a thermal barrier.
  The reactor coolant pressure is increased by activating heaters which are installed above the separator plate. Pressure is decreased by employing the spray systems.

- **Reactor Core:** The core configuration of the NSSS has 37 fuel assemblies in total and 16 control rod assemblies. The fuel assembly design is modelled as the standard 17x17 PWR type with 24 guid thimble locations and space for a central instrumentation tube. The height of the assembly is normally half the height of the one in a standard PWR. The fuel is UO_2 with Gd_2O_3 as a burnable absorber homogeneously mixed within the fuel only for some selected rods. The enrichment is slightly below the U.S. manufacturing limit of 4.95%. More about the core is discussed in Chapter 3, wherein the E-SMR core design takes the NuScale and Areva core design as a base reference.

- **Steam Generators:** Like the IRIS each NSSS module has two once-through helical-coil SGs for steam production. The steam generators are placed in the annular plenum between the riser and the inner RPV. The SG consists of tubes connected to the upper and lower plenums with tube sheets. Preheated feedwater enters the lower SG plenum through nozzles on the RPV. As the feedwater rises through the interior there is heat transfer between the reactor coolant in the primary circuit which makes the feedwater change into a saturated state and eventually superheated steam. This steam then exits the SG and is directed toward a turbine.

Figure 5: 3-D rendered view of a NSSS module

2.4 Nuward

Nuward is a new (Gen III+) SMR design initiated by French consortium (EdF lead, with CEA, TechnicAtome and the Naval Group) which is in the early stages of conceptual design and development. It is projected to be deployed in the year of 2030. The reactor is another integral PWR type designed to be set-up with twin modules with each module generating 170 MWe (340 MWe in total). The entire NSSS is installed inside of a steel containment vessel [16], submerged in an underground water well allowing for enhanced in-factory manufacturing.

2.4.1 Design Specifications

The table below, displays some basic technical parameters about the Nuward SMR.

NOTE: These are initial parameters only, as of [5], as some of them may be changed later due to critical analyses over the course of this thesis work.

Table 4: Notable features of the Nuward SMR

Major Technical Data	
Coolant/moderator	Light water
Thermal/electrical capacity	2x540 MWt / 2x170 MWe
Operating pressure (primary/secondary)	15 MPa / 4.5 MPa
Core inlet/outlet coolant temperature	280°C / 307°C
Fuel type/ assembly array	UO2 / 17x17 square pitch arrangement
No. of fuel assemblies	76
Fuel enrichment	< 5%
Core discharge burnup (GWd/ton)	-
Refuelling cycle	24 months
Reactivity control mechanism	CRDMs, solid burnable poisons
Approach to safety systems	Passive
Design life	60 years
Plant footprint	3500, nuclear island including fuel storage pool
RPV height/diameter	13 m / 4 m

RPV weight	310 metric tonnes
Seismic design (SSE)	0.25 g
Distinguishing features	Highly compact NSSS and containment, boron-free design

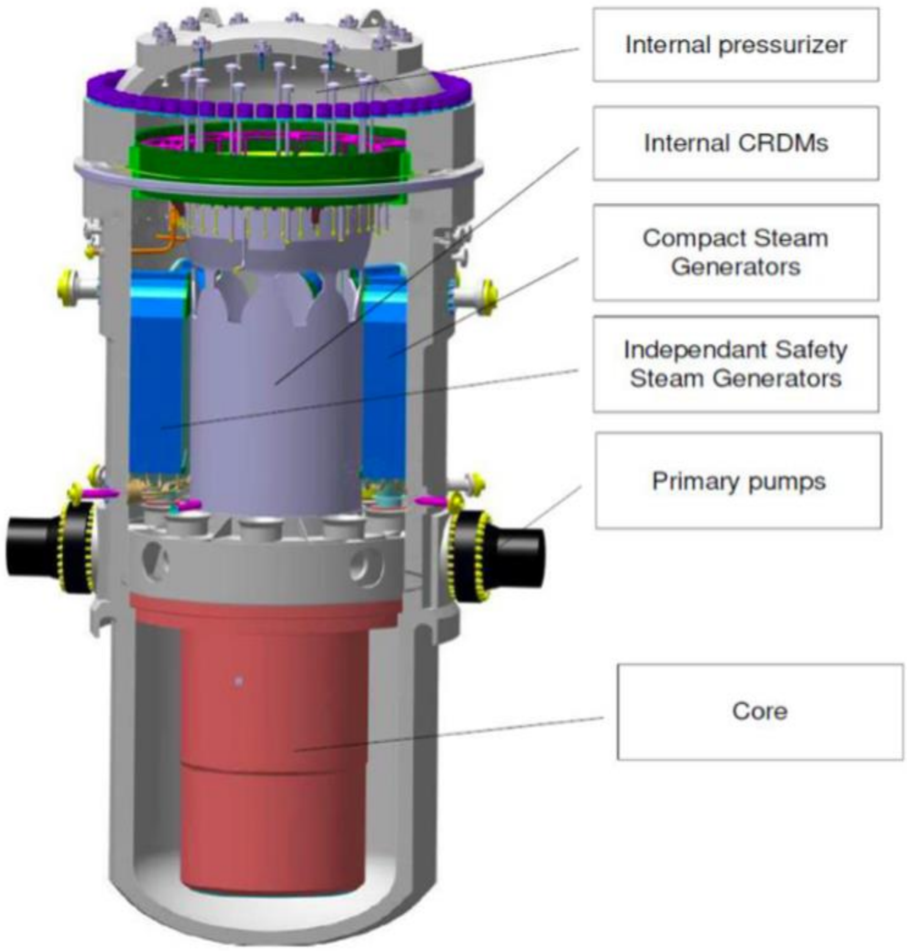

Figure 6: 3-D CAD model of the Nuward SMR RPV

The major components inside the RPV are the pressurizer above the upper plenum (derived from IRIS), the reactor core (derived from NuScale/Areva), compact plate steam generators (as opposed to the helical-coil type), and reactor coolant pumps for forced circulation.

- **Reactor Pressure Vessel and Internals:** The RPV of the Nuward is based off of an average 900 MWe PWR vessel for convenience of manufacturing and reduced procurement risk. The use of CRDMs and electrical penetrations for instrumentation and pressurizer heaters simplifies the vessel head manufacturing greatly. Another notable feature in Nuward during outages is that the components can easily be removed and replaced by refurbished/repaired ones. The removed components are later inspected and maintained outside of the outage critical path.
  The top head containing the pressurizer and the lower plenum of the RPV are of an elliptical shape similar to that of NuScale.

- **Reactor Coolant System and Pressurizer:** The primary coolant flow is internal and forced by 6 canned pumps which are mounted horizontally on the RPV outside the pump plenum (Figure 6). They are positioned in the cold-leg below the SGs for more efficient hydraulic conditions. The large reactor volume water and large volume of the pressurizer provide sufficient margins for operational transients and for normal day-to-day operation of the reactor.
  The pressurizer above the upper plenum has been separated with the help of a separator plate with holes, regulating in and out surges, and preventing the mixing of the saturated water with the subcooled primary coolant. Detailed design parameters will be discussed in Chapter 3 and how it has been partially derived from IRIS.
  By adopting the boron-free design, no primary water dilution is required during the whole production cycle.

- **Reactor Core and Fuel:** The reactor core is based on a proven PWR design having 17x17 square pitch fuel assemblies. UO_2 is the fuel and is enriched to slightly below 5% (U-235). All fuel assemblies are equipped with a Rod Cluster Control Assembly (RCCA) driven by an internal electrical CRDM. In Chapter 3 there will be a more detailed discussion about the core design and the RCCAs, and how they have been derived from NuScale and Westinghouse respectively.

- **Steam Generators:** The RCS of Nuward uses an innovative once-through compact plate-type heat exchanger for the steam generator. Specific features have been developed in the design and manufacturing process for them to be suitable for nuclear applications. The technology is of a Compact Steam Generator (CSG) which plays a key role in minimizing the overall size of the RPV. Six CSGs provide the turbine with the steam required during normal power operation and two Safety-CSGs (S-CSGs) ensure the safety cooling function (passive safety system) in case of accidents. They are part of an independent, closed and passive system that ensures the decay heat removal in all Design Basis Condition (DBC) accidents as well.

- **Boron-Free Control:** An exceptional feature of the Nuward SMR, is that the reactor does not use soluble boron for the reactivity control. This is feasible due to the small size of the core and the large coverage of the absorber (since the absorber is used to control almost every fuel rod). This in turn provides a large simplification of operation in normal (no boron related reactivity feedback) and accident situations (no dilution risk). This also decreases the amount of waste produced as there is no dilution during the fuel cycle.

2.5 RELAP5

The RELAP5 computer code is a light water reactor transient analysis code developed for the U.S. Nuclear Regulatory Commission (NRC) for use in rulemaking, licensing audit calculations, evaluation of operator guidelines, and as a basis for a nuclear plant analyser. This software has been used for various transient applications such as loss of coolant, transients without scram, loss of feedwater, loss of offsite power, station blackout, and turbine trip. RELAP5 is a highly generic code that, in addition to calculating the behaviour of a reactor coolant system during a transient, can be used to simulate different thermal-hydraulic transients in both nuclear and non-nuclear systems [17].

RELAP5/MOD3 code is the third major variant of the RELAP5 advanced thermal-hydraulic code that was released in 1979 [18]. The version used in the scope of this thesis is the RELAP5/MOD 3.2. It is used to simulate a steady-state model which will be used to analyse the behaviour of the system pressure and the inlet and outlet temperatures of the core, and a few other parameters. Further on, a transient run is also done varying the power and flow rates.

The RELAP5 code is coded in a modular fashion using a top-down structuring. The various models and procedures are isolated in separated subroutines. The top-level structure consists of input (INPUT), transient/steady-state (TRNCTL) and stripping (STRIP) blocks and can be seen in Figure 7. The input block processes the input file, check the input data, and prepares required data blocks for all program options. The transient/steady-state block handles transient and steady-state simulation options. The steady-state option is nothing but an accelerated transient until the time derivatives approach zero. It contains convergence testing algorithms to determine a satisfactory steady-state or divergence from the steady-state. If the transient technique is used alone, it would mean a steady-state from an initial condition would be identical to a plant transient from that initial condition. Finally, the strip block extracts simulation data from a restart plot file, and this data can in turn be post-processed and visualized with the help of other software [17].

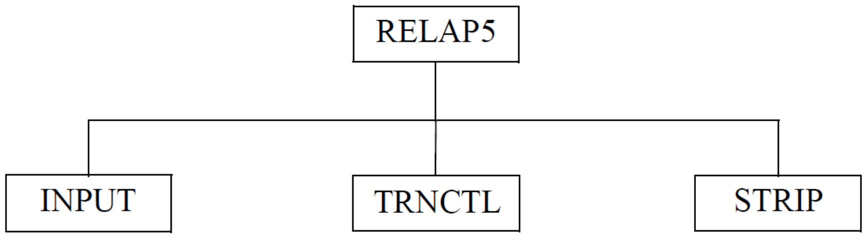

Figure 7: RELAP5 top-level structure

At first, to get the input file ready, values such as the flow area, volume, elevation, and hydraulic diameter need to be found out through initial intuition and given data. Further on, critical research and analysis can be done by going through relevant literature. Before nodalizing the problem, the components of the system must be identified (hydraulic components and heat structures accordingly).

The next step is to define and nodalize the problem. The code input nodalization should be defined so the most complete information set concerning the questions that motivated the study will be available. The model is then initialized, with all the initial and boundary conditions placed separately in another file, to secure the desired starting point for the problem investigation [18].

The process of nodalization is nothing but defining a boundary for the system that requires simulation. The boundary defines the extent of the model and the model is composed of volumes called “control volumes.” These are defined arbitrarily in a way that is most convenient for analysis of the problem. This process of defining the control volumes is known as “nodalization.” The complete nodalized model for the case explored in this thesis work is seen in Section 3.2.

Each hydraulic component is divided into discrete volumes, that are stream-tubes of equal size, which have inlet and outlet junctions, and junctions in between the respective volumes of the component as well. The junctions are used to connect the different volumes together along with different components (for components, can be single or time-dependent, more in Section 3.3). The code calculates the average fluid properties at the centre of the control volumes throughout the model and the fluid vector properties at the junctions.

3 Component Database and Model

This chapter aims to discuss the design parameters of the Nuward SMR in more detail. Firstly, a 2-D layout of the RPV is presented, along with a rough block scheme with dimensions which is then further developed by creating a nodalization diagram for the RELAP input. Most of the latter half of the chapter will be used to explain in detail each of the components, their respective RELAP models and justifications of the chosen parameters.

3.1 Nuward Layout

Using Figure 6 as reference, a discretization layout has been created for the Nuward SMR, displaying in a clearer manner the components and where exactly they are located in the RPV.

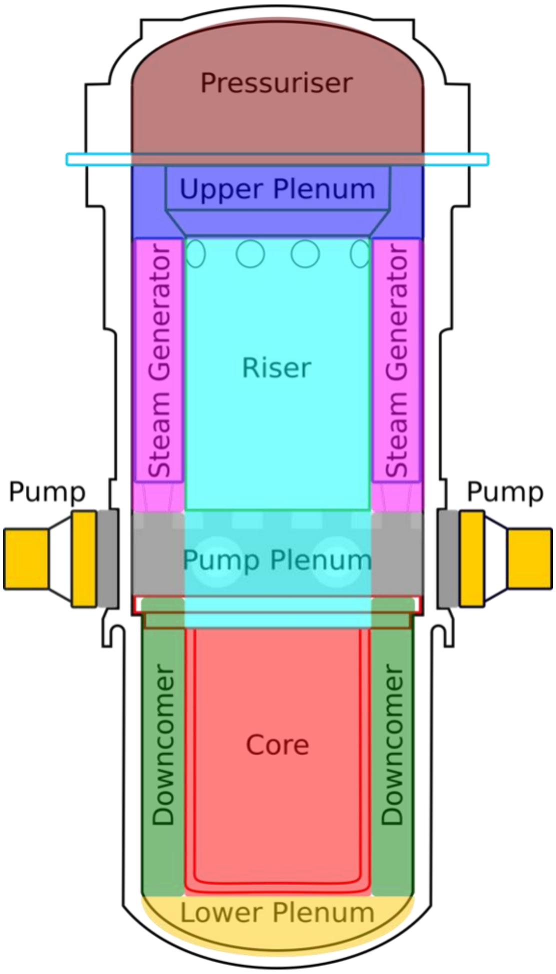

Figure 8: Nuward discretization

To obtain a well-defined RELAP model, the next step was to develop a rough dimensional scheme to show how much space each component takes inside of the RPV approximately. Judging from the CAD model (Figure 6) and the discretization scheme (Figure 8), approximated dimensions were decided on, as seen in Figure 9. The colour scheme can be deduced from Figure 8.

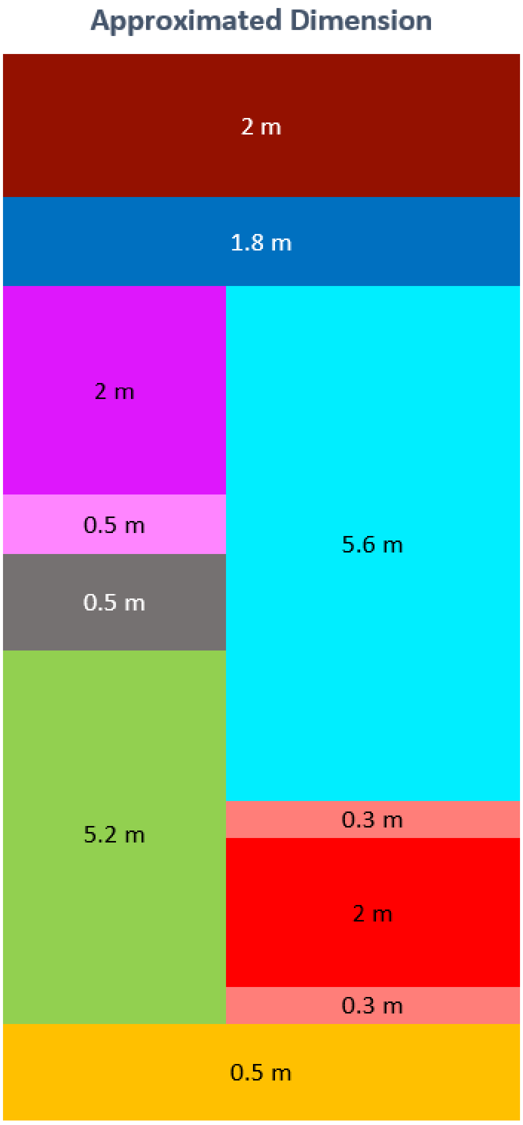

Figure 9: Approximated dimension scheme

3.2 RELAP5 Model

Using Figure 9, as a basis, a nodalization diagram was hence developed.

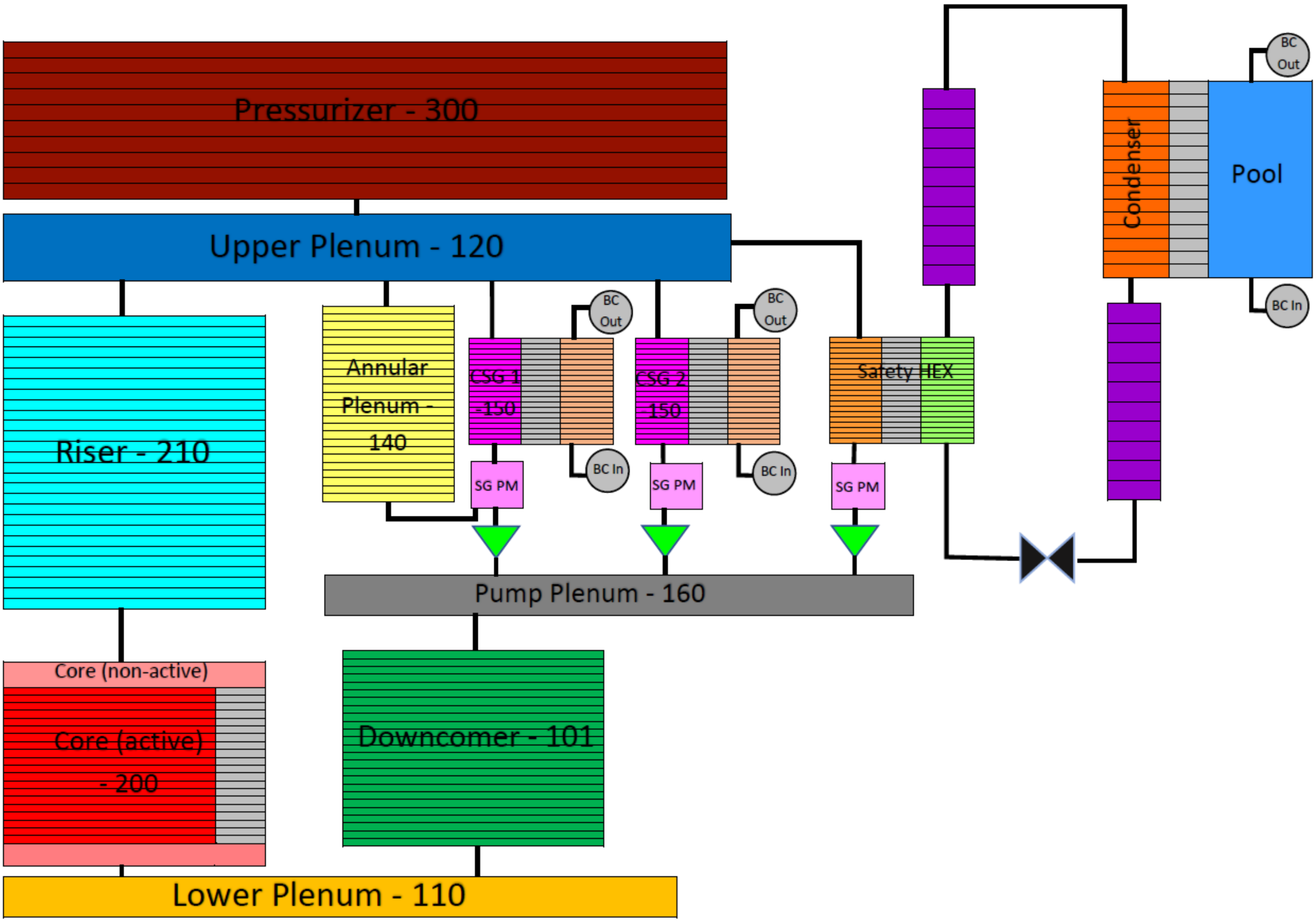

Figure 10: RELAP Nodalization diagram

In the above scheme the primary circuit is easily identifiable. The secondary circuit however has not been displayed explicitly. The boundary conditions in and out (BC in & BC out) signify the input and output conditions that are applied to the secondary part of the steam generator.

On the extreme right we can see the passive safety system which consists of the safety heat exchangers, a pool type condenser, and pipes for connections.

The following section describes each component of the primary and secondary circuits and their RELAP models in detail.

3.3 Design Parameters and Component-wise Model Setup

3.3.1 Input Parameters

Before getting into the component descriptions in detail, a list of input dimensions and parameter have been listed first. As per Table 5, input dimensions for the core, RPV and internals and containment (not relevant for the scope of this work) can be seen.

Table 5: Standard Input Dimensions

Dimension	**Internal Diameter [m]**	**External Diameter [m]**	**Height [m]**	**Number**
Fuel/Core				
No. of fuel rods per assembly	-	-	-	264
Fuel active height	-	-	2	-
RPV and internals				
Reactor vessel	3.65	3.95	12.5	-
Core barrel	2.35	2.45	5.2	-
CSGs	-	-	2	6
S-CSGs	-	-	2	2
Containment				
Containment	15	-	16	-

Below in Table 6, the input parameters for RELAP have been summarised. However, all the parameters except the power and the flow rate are checked again and verified in Chapter 4.

Table 6: General Physical Parameters

Parameter	Value	Unit
Core thermal power	540	MWt
Electrical output	170	MWe
Nominal coolant flow rate (primary)	3700	kg/s
Nominal coolant flow rate (secondary)	240	kg/s
Core inlet temperature	280/553.15	°C/K
Core outlet temperature	307/580.15	°C/K
Mean core temperature	293.5/566.65	°C/K
Primary pressure	15	MPa
Secondary pressure	4.5	MPa
Primary coolant inventory	65000 (approx.)	kg
Secondary coolant inventory (max)	2500 (approx.)	kg
Reactor internals mass	100000 (approx.)	kg

3.3.2 Downcomer

The downcomer is an annular component immediately after the pump plenum. The flow rate of the coolant is forced down by the pumps from the steam generator to the downcomer. The flow is then directed to the core via the lower plenum. The data for the downcomer is given below in Table 7.

The height has been computed visualizing Figure 6 and Figure 8 and taking a reasonable value hence. The annulus inner diameter corresponds to the core barrel outer diameter (calculation of which will be found in Section 3.3.4and the annulus outer diameter is nothing but the inner diameter of the RPV (Table 5).

Table 7: Downcomer data

Parameter	Value	Unit
Height (H_{dc})	5.2	m
Annulus inner diameter ($D_{in,ann}$)	2.45	m
Annulus outer diameter ($D_{out,ann}$)	3.65	m
Flow area (A_{dc})	5.7491	m^2
Volume (V_{dc})	29.8953	m^3
Hydraulic diameter ($D_{h,dc}$)	1.2	m
Roughness	1.0e-6	

The flow area through the downcomer is calculated as follows:

$$A_{dc} = \frac{\pi\left({D_{out,ann}}^2 - {D_{in,ann}}^2\right)}{4} \tag{3.1}$$

Hence, the volume is calculated by multiplying the flow area and the height of the downcomer:

$$V_{dc} = A_{dc} \cdot H_{dc} \tag{3.2}$$

The hydraulic diameter is given by the following formula:

$$D_{h,dc} = \frac{4 \cdot A_{dc}}{\pi(D_{in,ann} + D_{out,ann})} \tag{3.3}$$

The flow area and the hydraulic diameter are then input into the RELAP input file. The volume is usually always calculated by the software given certain conditions are met.

For the case of the downcomer, the component has been distinguished as an "annulus" type. It has been divided into 26 discrete volumes, each of them being 0.2 m in length, taking into consideration the total length of 5.2 m. The flow is later directed towards the lower plenum.

3.3.3 Lower Plenum

The lower plenum is an elliptical hydraulic component at the bottom most part of the RPV. Its purpose is to redirect the downward flow from the downcomer upward to the core. The component's physical data have been given below.

Table 8: Lower Plenum data

Parameter	Value	Unit
Height (H_{lp})	0.5	m
Ellipse major (and minor) axis (a_{lp})	3.65	m
Flow area (A_{lp})	5.7491	m^2
Volume (V_{lp})	3.488	m^3
Hydraulic diameter ($D_{h,lp}$)	1.2	m
Roughness	1.0e-6	

The height is computed similarly to that of the downcomer, judging a reasonable value from the CAD model (Figure 6). The major axis of the ellipsoid is nothing but the inner diameter of the RPV, and the minor axis has been assumed reasonably much like the height.

The flow area is given by:

$$A_{lp} = \pi\left({a_{lp}}^2\right) - \frac{\pi {D_{core,out}}^2}{4} \tag{3.4}$$

The volume is computed in a similar manner to that of the downcomer:

$$V_{lp} = A_{lp} \cdot H_{lp} \tag{3.5}$$

The hydraulic diameter is equal to the inner diameter of the RPV.

The lower plenum has been modelled on RELAP as a "branch" with two junctions, connecting the downcomer with the core inlet.

In the next section, the details of the core will be discussed.

3.3.4 Core

The core of the Nuward SMR is derived from a classical PWR. It is composed of assemblies of cylindrical fuel rods surrounded by the coolant that flows along the rod length. Two design features establish the principle thermal hydraulic characteristics of a reactor core, they are, the orientation and the degree of hydraulic isolation of an assembly from its neighbours [19]. The assemblies in the case of Nuward are vertical and are not hydraulically isolated in fuel channels (like in a BWR), for handling and structural purposes.

The principal characteristics of power reactor fuel bundles which make up majority of the core, are the array (layout and rod spacing) and the method of fuel pin separation and support along their span. Light water reactors (LWRs) in general have small fuel-to water volume ratios, since the coolant also serves as the moderator, and hence a large fuel rod pitch (distance between the centrelines of two adjacent fuel rods) [19]. The typical fuel assembly has spacer grids for precise arrangement of the fuel rods. The spring clips present in the grid contact and support the fuel rods. For Nuward, a square array is used for each assembly, 76 in total, with 264 fuel rods per assembly. The data facts for the core have been listed below.

Table 9: Core Parameters

Parameter	Value	Unit
Core Geometry		
Active fuel length (H_a)	78.74 / 2	inch / m
Active core diameter ($D_{in,core}$)	2.3556	m
No. of assemblies (N_A)	76	
Core volume (V_{core})	11.2771	m^3
Volumetric power density	47.8844	MW/m^3
Heat Flux		
Core average heat flux	451	kW/m^2
Hot assembly average heat flux	586	kW/m^2

Hot pin average heat flux	645	kW/m^2
Peak Factors		
Axial power peaking factor	1.45	
Hot assembly peaking factor	1.3	
Hot pin peaking factor	1.1	
Core peaking factor	2.0735	
Fuel Geometry		
Fuel rod length	85 / 2.159	inch / m
Fuel pellet outer diameter	0.3195 / 8.1153e-3	inch / m
Diametral gap	1.651e-4	m
Cladding inner diameter	0.326 / 8.2804e-3	inch / m
Cladding outer diameter	0.374 / 9.4996e-3	inch / m
Cladding thickness	6.096e-4	m
Guide tube outer diameter	0.482 / 1.2243e-2	inch / m
Instrument tube outer diameter	0482 / 1.2243e-2	inch / m
Fuel rod pitch	0.496 / 1.2598e-2	inch / m
Fuel assembly envelope side length	8.425 / 0.214	inch / m
Fuel assembly pitch (P_{fa})	8.466 / 0.215	inch / m
No. of fuel rods per assembly (N_{fr})	264	
No. of guide tubes per assembly (N_{gt})	24	
No. of instrument tubes per assembly (N_{ins})	1	
Spacer grids per assembly	5	
Overall fuel assembly length (H_{core})	2.6	m

Core Flow Area Quantities		
Assembly area (A_{ass})	4.624e-2	m^2
Fuel rod area per assembly (A_{fr})	1.8711e-2	m^2
Guide thimble area per assembly (A_{gt})	2.8253e-3	m^2
Instrumentation area per assembly (A_{ins})	1.1772e-4	m^2
Assembly flow area ($A_{ass,flow}$)	2.4586e-2	m^2
Core flow area (A_{core})	1.8686	m^2
Core Barrel Geometry		
Barrel inner diameter	2.3556	m
Barrel thickness (t_{barr})	0.05 (Figure 14)	m
Barrel outer diameter	2.4556	m
Baffle Assembly Sides	40 (Fuel assembly sides)	
Baffle Plate Thickness (t_{bp})	0.019 (Figure 14)	m
Hydraulic Diameter		
Core hydraulic diameter ($D_{h,core}$)	1.0984e-2	m
Assembly hydraulic diameter	1.1124e-2	m
Roughness	1.0e-6	

The fuel assembly used in the Nuward SMR has been based on the NuScale NuFuel-HTP2™ fuel assembly (Figure 11), a 17x17 PWR design which is in turn derived from the Areva assembly, and the height is approximately one-half of the typical PWR nuclear fuel. The assembly contains design features similar to those of proven HTP™ fuel designs. The fuel rod contains M5® alloy cladding with UO_2 pellets [20].

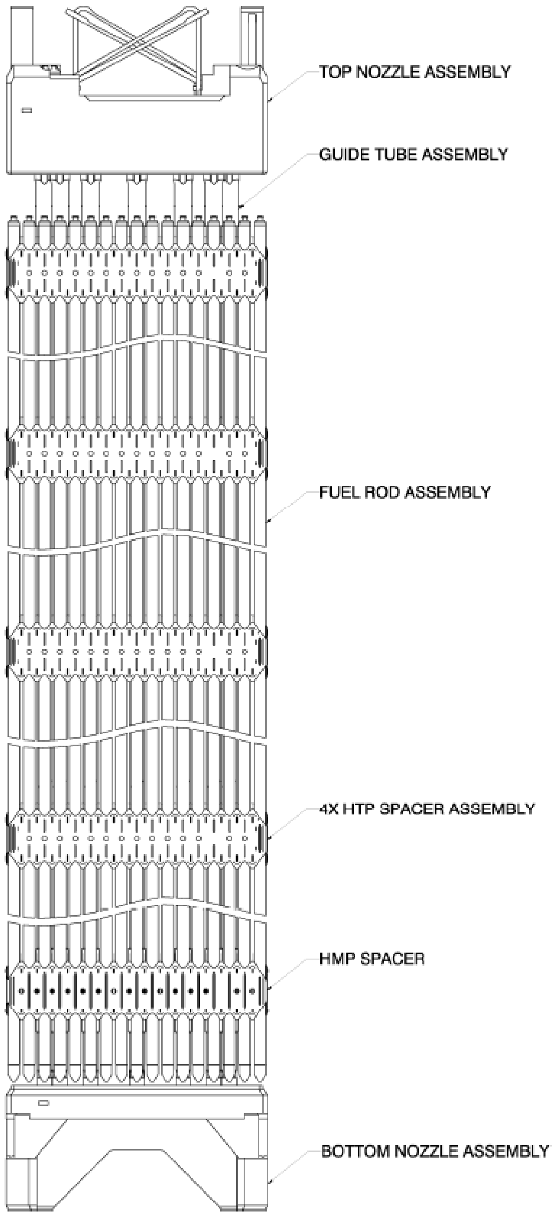

Figure 11: HTP2™ fuel assembly arrangement [20]

The assembly totally consists of five spacer grids, four of which are the HTP™ grid type and they have been welded to the guide tubes. The bottom most spacer grid is of the HMP™ type, and it is held with rings that have been welded to the guide tubes instead. The top four spacer grids are made of Zircaloy-4 strips and are welded with the guide tube assembly at the intersections. This is done to limit axial movement and align them with the adjacent fuel assemblies. These grids are identical to the Areva 17x17 PWR product. The HMP™ spacer grid on the other hand is made of a low cobalt, precipitation-hardened Alloy 718 strip material which provides enhanced strength and relaxation characteristics. These grids are held by Zircaloy-4 sleeves which are spot welded to the guide tube assembly. These spacer grids are also identical to the one used by Areva for the PWR [20].

The top nozzle consists of a 304L stainless steel frame that acts as an interface between the upper core internals and the core components providing a smooth transition for the coolant flow. The hole flow pattern in the nozzle ensures a low pressure drop along with satisfying strength requirements. It is attached to the fuel assembly with a quick disconnect feature which allows for easy detachment. Furthermore, the bottom nozzle also made up of 304L stainless steel consists of a cast frame of ribs. 24 holes are available for the connection of the guide tubes to the nozzle using cap screws and a centre hole is provided for the instrument tube [20].

The cross section of a single fuel assembly and of the complete set of fuel assemblies (76) are given below. The number of baffle assembly sides (40) are clearly visible in Figure 13.

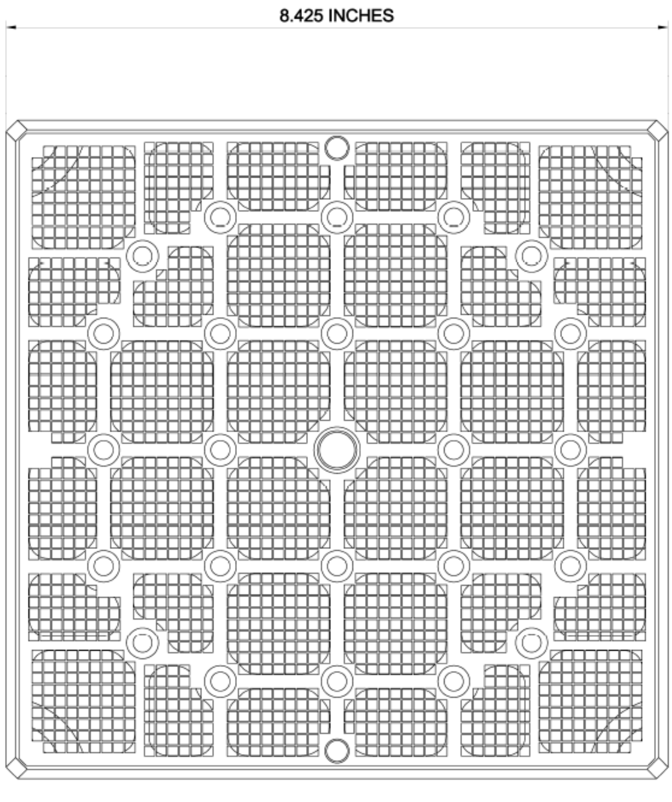

Figure 12: Cross-section of a single fuel-assembly with fuel rods, guide tubes, and instrumentation [20]

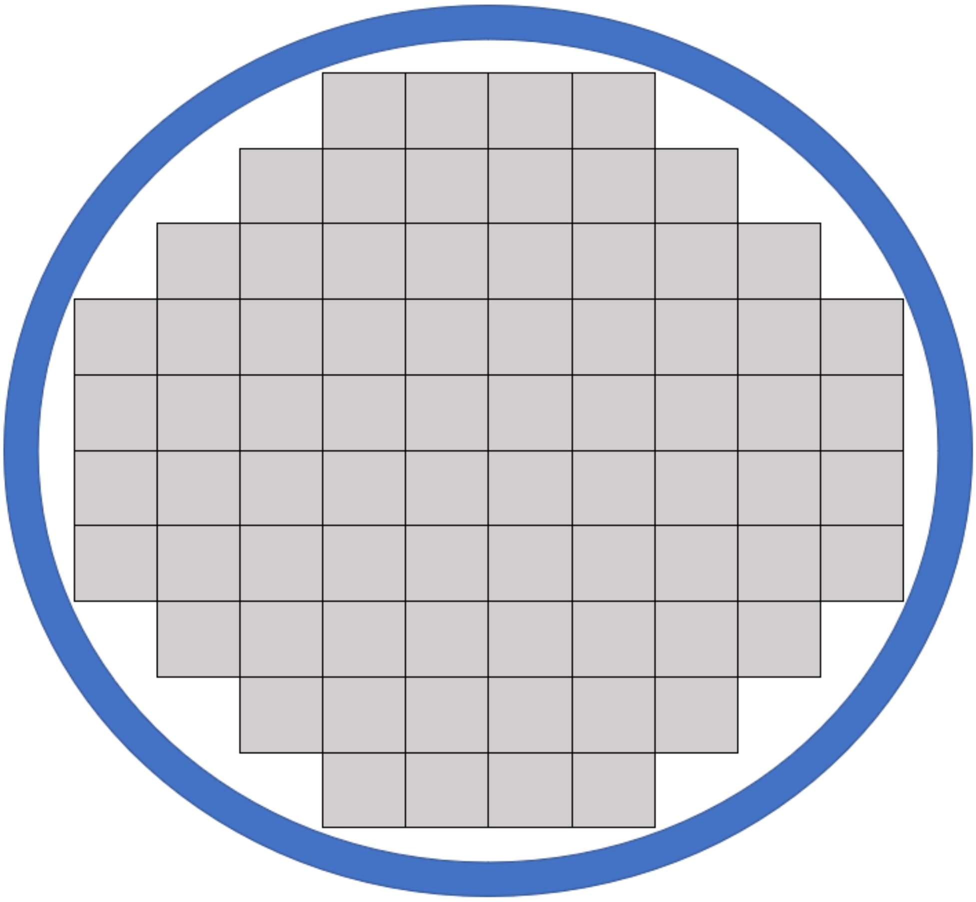

Figure 13: Core fuel assembly arrangement (76 assemblies) with outer barrel

The guide tubes are made of a Zircaloy-4 MONOBLOC™ material and have a constant outer diameter. The upper portion of the guide tube has a large internal diameter that allows for rapid insertion of the Rod Cluster-Control Assembly (RCCA) in case of a reactor trip. The lower portion has a smaller inner diameter to decelerate the RCCA to limit impact forces on the fuel assembly during the trip. The tube has four holes located at the bottom for a cooling flow for the inserted RCCAs. The connection of the guide tube with the bottom nozzle with the cap screw is derived from existing PWR designs. Finally, the single and central instrument tube is made of Zircaloy-4, and it has a constant inner and outer diameter. It provides guidance for the in-core instrumentation. The tube is not attached to top or bottom nozzles, so its axial position is determined by sleeves welded above and below the bottom HMP™ grid [20].

The calculations for the active core diameter and the process of arriving at an educated assumption for the barrel thickness have been described below with a few calculations.

Firstly, the core inner diameter ($D_{i,core}$) is calculated using the fuel assembly pitch length as shown in Figure 14.

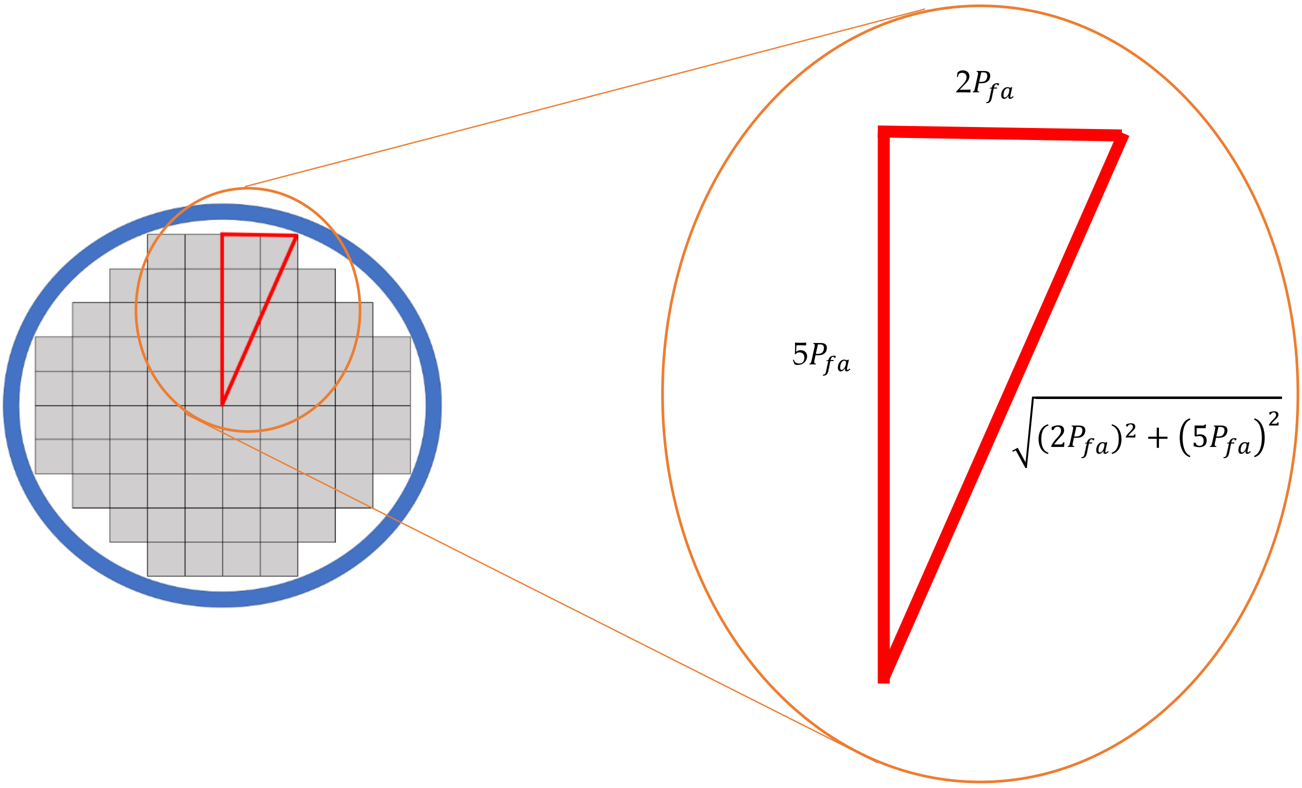

Figure 14: Schematic for the core inner diameter calculation

The active core diameter includes the length shown in Figure 14 and the baffle plate thickness shown in Figure 15.

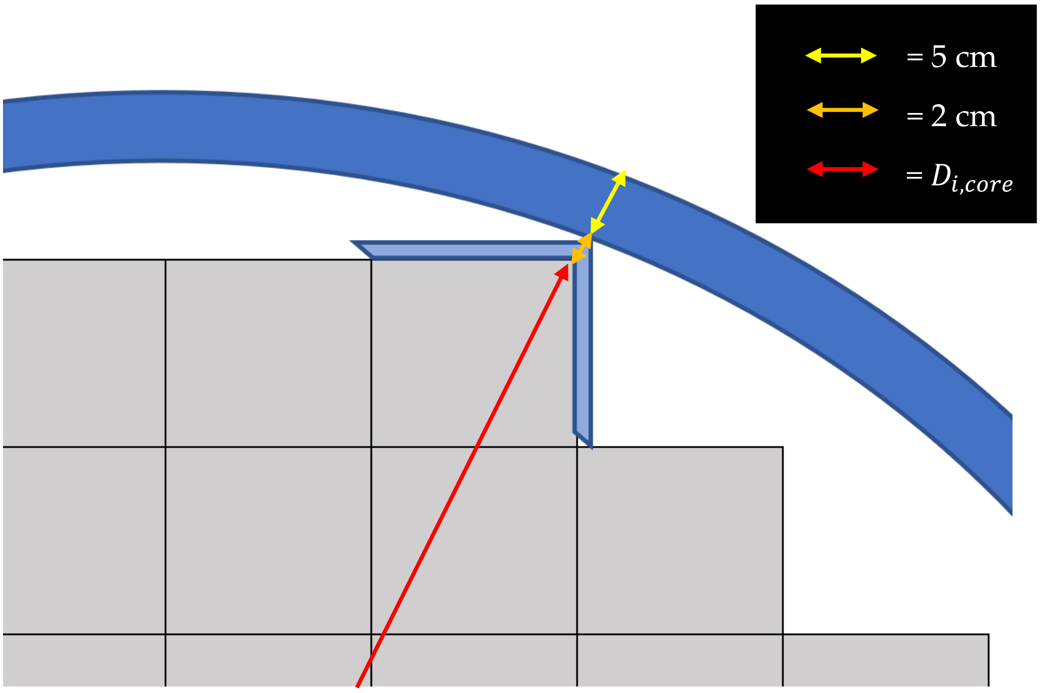

Figure 15: Thicknesses of baffle steel lining and core barrel

$$P_{fa} = 0.215\ m \tag{3.6}$$

$$D_{i,core} = 2 \cdot \sqrt{(2P_{fa})^2 + (5P_{fa})^2} = 2.3156\ m \tag{3.7}$$

The active core diameter is hence calculated by adding the inner core diameter of Equation (3.7) to the baffle plate thickness. The core outer diameter ($D_{out,core}$) is calculated by simply adding the barrel thickness to the active core diameter.

$$D_{in,core} = D_{i,core} + 2 \cdot t_{bp} = 2.3556\ m \tag{3.8}$$

$$D_{out,core} = D_{i,core} + 2 \cdot t_{bp} + 2 \cdot t_{barr} = 2.4556\ m \tag{3.9}$$

The heat flux and peak factor values have been listed for completeness but are not described in detail since they do not have relevance in the scope of this thesis.

The fuel pellet geometry specifications are visible in the following figure.

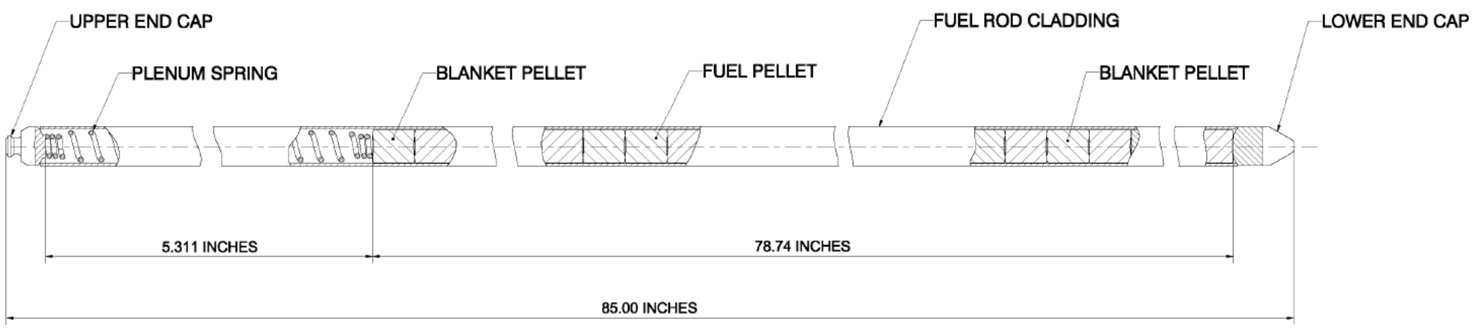

Figure 16: Typical fuel rod structure

In addition to the active fuel length of 2 m, lengths of 0.3 m have been taken at the top and bottom of the core for having a reasonable structural margin for other components. At the top, space has been considered for the gas plenum, top nozzle, and upper core plate, while at the bottom there are the bottom nozzle, lower core plate and support structures. Hence, the total length of the core including the active and non-active lengths is $\approx \mathbf{2.6\ m}$.

The core flow area has been calculated with a bottom-up approach. First, the areas of the individual fuel rods, guide tubes and instrumentation tube are calculated and subsequently their values for an entire assembly. Hence, the assembly flow area is obtained by subtracting the total areas of the fuel rods, guide thimbles and instrumentation. Referring to the notation in Table 9, we have the following equations.

$$A_{ass,flow} = A_{ass} - A_{fr} - A_{gt} - A_{ins} \qquad (3.10)$$

$$A_{core} = A_{ass,flow} \cdot N_A \qquad (3.11)$$

The volume is calculated in a similar manner to that of the downcomer by multiplying the core area using the active diameter with the overall assembly length.

The RELAP model of the core has been divided into three sections, and they are the core bottom head which is directly connected to the lower plenum, an intermediate core which is the active part, and a core top head which is connected to the riser. The bottom and top heads are modelled as the type "pipe" of 0.3 m each discretised into 3 volumes. The active part is also modelled as a pipe of 2 m consisting of 20 volumes for a more detailed analysis.

Each pipe volume in the core has also been given heat structure properties. The heat structure provides a means by which the heat transferred across a surface boundary can be calculated. In the case of the core, it would be the heat transferred due to the thermal power produced. There are 20 axial heat structures similar to the component itself and each axial heat structure is divided into 5 radial regions.

Next, the riser component will be discussed.

3.3.5 Riser

The riser is an intermediate component in the Nuward SMR acting as a channel for the flow between the core and the upper parts of the RPV (upper plenum and the steam generators). It also houses the major parts of the RCCAs and the instrumentation tube. Therefore, in this section the details of the RCCAs will be investigated in more detail.

The data for the riser and the RCCA is given below.

Table 10: Riser Parameters

Parameter	Value	Unit
Riser Geometry		
Riser height (H_{riser})	5.6	m
Internal diameter ($D_{in,core}$)	2.3556	m
Flow area (A_{riser})	0.79	m^2
Volume (V_{riser})	4.41	m^3
Hydraulic diameter ($D_{h,riser}$)	4.1961e-2	m
Roughness	1.0e-6	
RCCA Geometry		
No. of RCCA tubes (N_{RCCA})	76	
Outer diameter of drive-line ($D_{out,dl}$)	1.84 / 4.67e-2	inch / m
RCCA tubes side (s_{RCCA})	0.214	m
RCCA tube wall thickness (t_{RCCA})	0.006	m
No. of support columns (N_{SC})	8	
Diameter of a support column (D_{SC})	0.105	m
RCCA length (L_{RCCA})	161 / 4.1 (Figure 17)	inch / m

The RCCAs provide a rapid means for reactivity control during both normal and accident conditions. They are divided into two categories based on their operation i.e. control banks and shutdown banks. The control banks are inserted or withdrawn depending on the reactivity changes during operation of the reactor and can also be scrammed to provide shutdown capability. The shutdown banks are reserved solely for shutdown purposes and are always fully withdrawn from the core when the reactor is in the critical condition. They are inserted only if there is a reactor trip. The two important design criteria for the RCCAs are the reactivity worth that is required for the application, and the total power peaking factor. An RCCA always consists of a group of individual neutron absorber rods fastened at the top end to a common spider assembly. A cross-sectional view is shown in Figure 17. The absorber material used in the rods is Ag-In-Cd (Silver-Indium-Cadmium) and is in the ratio 80%-15%-5%. The spider assembly is in the form of a central hub with radial vanes containing cylindrical fingers from which the absorber rods are suspended. All components of the assembly are mostly made up of 304 and 308 stainless steels. To handle the RCCA, a long, hollow, and grooved drive shaft is attached to the spider hub of each of the assemblies by a split coupling. The drive shaft extends through the upper core support structure and through a reactor head penetration to the CRDM [21].

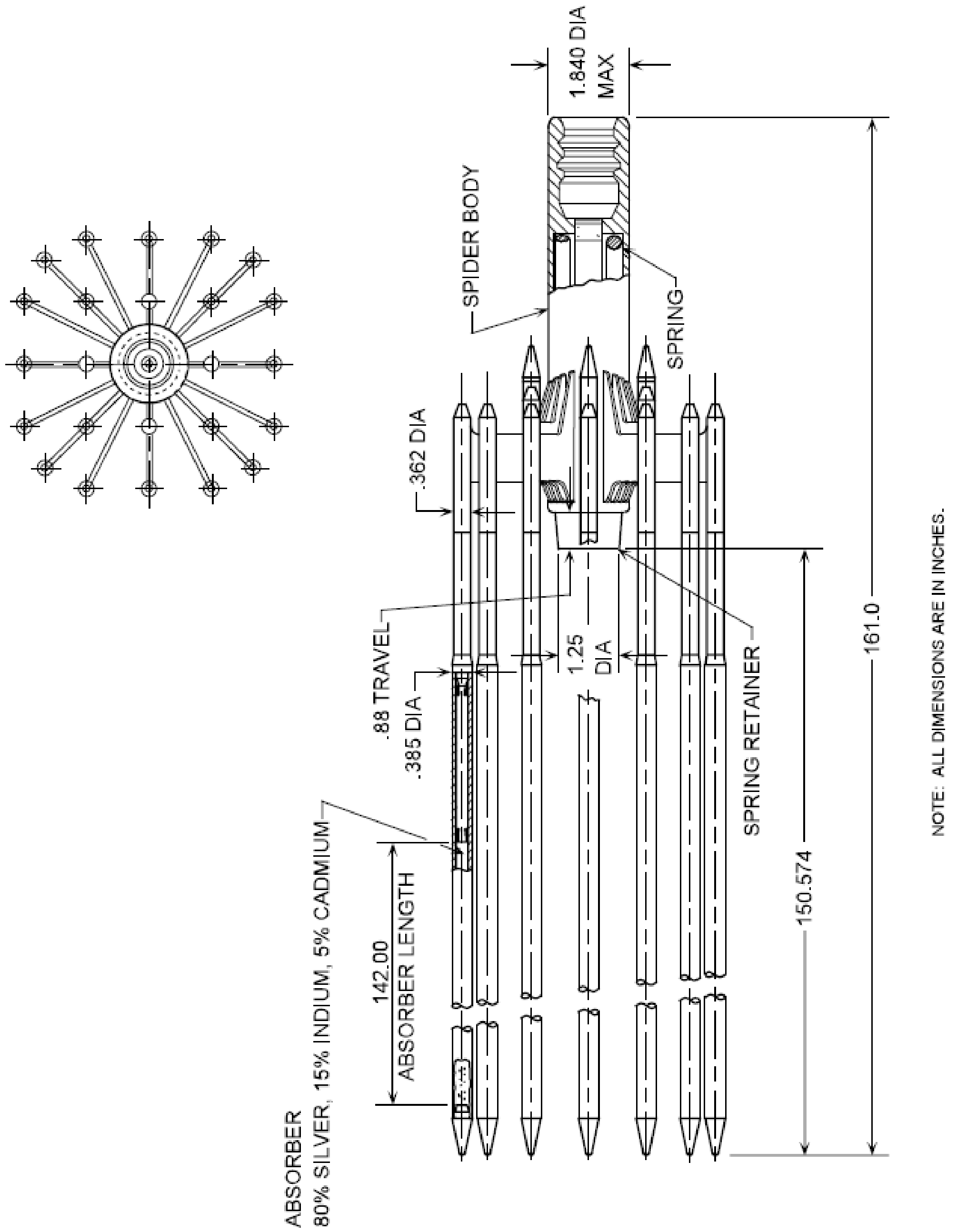

Figure 17: Sectional view of a typical Rod Cluster-Control Assembly (RCCA) [21]

The riser also contains support columns so in order to calculate the flow area, the area occupied by the support columns and the RCCAs are subtracted from the cross-sectional area of the riser as follows:

$$A_{riser} = \frac{\pi D_{in,core}^2}{4} - s_{RCCA} \cdot N_{RCCA} - \frac{\pi D_{SC}^2}{4} \cdot N_{SC} \quad (3.12)$$

The RELAP model of the riser is again of the type "pipe." Its length is 5.6 m and it has been divided into 28 discrete volumes. The component is connected to the upper plenum above it.

3.3.6 Upper Plenum

The upper plenum acts similarly to the lower plenum and is used to direct the flow towards the steam generators. Additionally, it also ensures separation between the subcooled coolant and the saturated coolant which is in the pressurizer. The component's data is the following table.

Table 11: Upper Plenum parameters

Parameter	Value	Unit
Height (H_{up})	1.8	m
Internal diameter ($D_{in,RPV}$)	3.65	m
Flow area (A_{up})	10.46	m^2
Volume (V_{up})	18.8342	m^3
Hydraulic diameter ($D_{h,up}$)	3.65	m
Roughness	1.0e-6	

The flow area is a simple calculation using the inner diameter of the RPV.

$$A_{up} = \frac{\pi D_{in,RPV}^2}{4} \tag{3.13}$$

The hydraulic diameter is again equal to the inner diameter of the RPV.

The upper plenum has been modelled as a "branch" element in RELAP. It acts as a connection between the riser and the pressurizer, annular plenum, and the steam generators.

3.3.7 Pressurizer

The pressurizer is the component at the top of the RPV head located right above the upper plenum. It is usually filled with saturated water for little-less than half of the total volume. The other part is occupied by steam. The main significance of the pressurizer is to regulate the pressure of the system during regular operation and during accident scenarios. In case the water level rises, increasing the pressure, the sprayers are employed to decrease the pressure by condensing steam. If the water level decreases and reduces pressure below a certain threshold, the heaters are employed present at the bottom walls of the pressurizer. This evaporates water to increase the pressure by producing more steam.

The design of the Nuward pressurizer is largely based on the IRIS pressurizer, with scaled down geometric and physical values. The only differences are that the Nuward pressurizer incorporates sprayers due to its smaller size, and makes use of a flat separator plate instead of an inverse top hat (Figure 18) since the cavities for the pumps are not required.

The separator plate not only regulates the surge flows, but also a) reduces thermal stresses in the head closure flange and its seals and maintains seal tightness, b) provides an effective thermal insulation to minimize heat transfer and maintain an adequate saturated water layer and c) provides structural support for the CRDM drive-lines and instrumentation tubes [22].

The table for the data of the pressurizer is given below.

Table 12: Pressurizer parameters

Parameter	Value	Units
Geometry		
Height (H_{pr})	2	m
Inner diameter (base) ($D_{in,pr}$)	3.65	m
Separator plate hole (surge orifice) diameter (D_{so})	0.05	m
No. of surge orifices (N_{so})	8	
Base Flow area (A_{pr})	0.0157	m^2
Volume (V_{pr})	14.4866	m^3
Hydraulic diameter ($D_{h,pr}$)	0.4	m
Roughness	1.0e-6	
Levels		
Initial water level	0.8	m
Initial water volume	10.4634	m^3
Initial steam height	1.2	m
Initial steam volume	4.0232	m^3
Physical parameters		
Pressure	15	MPa
Initial Temperature	553.15	K

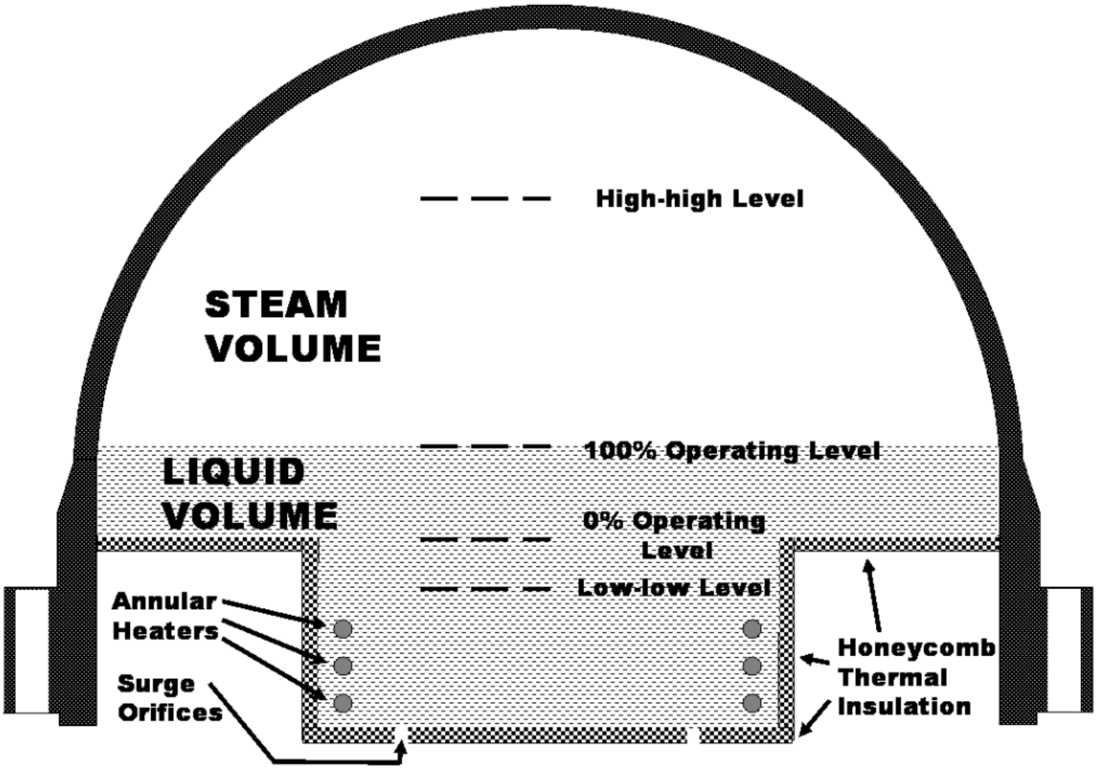

Figure 18: IRIS Pressurizer [26]

The RELAP model of the pressurizer is of the "pipe" type since a separate component has not been introduced yet in RELAP5/MOD 3.2. It is divided into 10 volumes considering its height of 2 m (each volume of 0.2 m). Initially half of the volumes are occupied by saturated water and the other half by saturated steam. For the steady-state analysis the system pressure is observed and it is seen whether it oscillates and then rests around the operating pressure.

To control the pressure, two trip valves (type "valve -> trpvlv") have been connected to the pressurizer volume and these valves further lead to time-dependent volumes ("tmdpvol") filled with water and steam respectively. If the pressure exceeds 15.2 MPa, the valve connected to the water volume opens supplying water to the pressurizer much like a sprayer function. If the pressure goes below 14.8 MPa then the valve connected to the steam volume opens up, injecting steam into the pressurizer identical to a heater.

The flow area of the pressurizer is calculated using the surge orifice diameter.

$$A_{pr} = N_{so} \cdot \frac{\pi D_{so}^2}{4} \tag{3.14}$$

The hydraulic diameter is calculated the conventional way as follows.

$$D_{h,pr} = \frac{4A_{pr}}{N_{so}\pi D_{so}} \tag{3.15}$$

3.3.8 Annular Plenum

The annular plenum is a dead space in between the steam generators. Water is accumulated here, but as a first guess, there is a partial flow rate from this component down to the pump plenum, through the steam generator lower head. The details of the plenum are given in the table below.

Table 13: Annular Plenum parameters

Parameter	Value	Unit
Height (H_{ap})	2.5	m
Inner diameter ($D_{in,ap}$)	2.4556	m
Outer diameter ($D_{out,ap}$)	3.65	m
Steam generator occupied area ($A_{occ,SG}$)	2.86	m^2
Flow area (A_{ap})	2.89	m^2
Volume (V_{ap})	7.2227	m^3
Hydraulic diameter ($D_{h,ap}$)	0.41	m
Roughness	1.0e-6	

The calculation of the flow area is quite straightforward. The occupied area of the steam generators is subtracted from the entire area of the annulus.

$$A_{ap} = \frac{\pi(D_{out,ap}^2 - D_{in,ap}^2)}{4} - A_{occ,SG} \quad (3.16)$$

The hydraulic diameter is calculated with the formula below:

$$D_{h,ap} = \frac{4A_{ap}}{\pi(D_{in,ap} + D_{out,ap})} \quad (3.17)$$

It is modelled in RELAP as a "pipe," and is consecutively connected to the steam generator lower head. Its height is taken as 2.5 m which is equal to that of the steam generator. It has been divided into 25 volumes.

3.3.9 Steam Generators

The steam generators used in the Nuward SMR are of the plate-compact heat exchanger type. Research has been conducted at the Massachusetts Institute of Technology (MIT) regarding the benefits of the CSGs for use in SMRs. The two designs that have been discussed in this section include the rectangular CSG design from Areva and the semi-circular design by Heatric™.

The CSGs have diffusion-bonded channels and they typically present hydraulic diameters of around 0.5 – 6 mm (referred to as mini-channels). These types of heat exchangers can withstand the pressure and temperature of a conventional steam generator. They are once-through steam generators wherein the water is completely evaporated and the steam is in a superheated state [23].

The CSG design has vertical and straight channels. The rectangular one has the dimensions of 4 mm x 2 mm and 1 mm metal thickness. The semi-circular one on the other hand has a 2 mm diameter and 2.65 mm pitch as shown in Figure 19 (Heatric single-phase HX design). As of now, the hot and cold channels have the same dimensions.

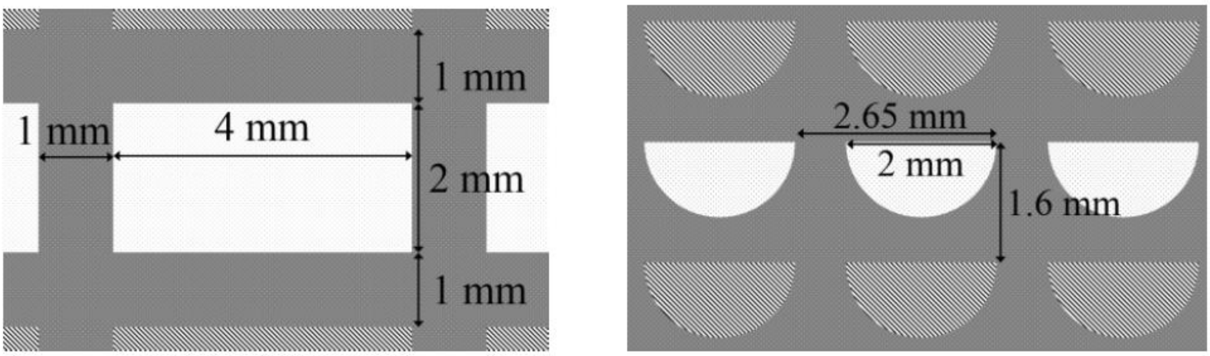

Figure 19: The two designs of the CSGs (rectangular and semi-circular channels) [23]

The data for the two steam generators have been given in the following table.

Table 14: Steam generator parameters

Parameter	**Value (Rectangular)**	**Value (Semi-circular)**	**Units**
Primary flow rate	3700		kg/s
Secondary flow rate	294.6		kg/s
Channel			
Channel length	4.0e-3	Diameter = 2.0e-3	m
Channel width	2.0e-3	-	m
Channel perimeter	1.2e-2	3.14e-3	m
Channel flow area	8.0e-6	1.57e-6	m^2
No. of channels	28818	147165	
Heat Exchanger			
Thermal power per HX	90.53	130.2	MW/m^3
Active height	2	2	m
Flow area	0.1152	2.26e-2	m^2
Heated area	345.816	756.6624	m^2
No. of heat exchangers	6	6	
Total power	540	NA	MW
Total HX flow area	0.6916	0.1356	m^2

Hydraulic Diameter			
Channel hydraulic diameter	2.6667e-3	2.0e-3	m
Channel heated diameter	2.6667e-3	2.0e-3	m
Roughness	1.0e-6		

The design used for the RELAP simulation is the one with the rectangular cross-section, as the power density is in line with the total thermal power of the Nuward SMR (540 MW). However, in further studies the semi-circular version may be used. Currently, a brazed-plate type heat exchanger is being investigated for use in the Nuward. For convenience, the six SGs have been modelled as one single SG.

The semi-circular design is a more compact design compared to the rectangular one, however, it has a higher thermal inertia making it more complicated to use. In comparison to the helical-coil steam generator of the IRIS, the CSG provides a power density almost 40 times higher [23].

The CSG in RELAP has been modelled as a “pipe,” and it has been divided into 20 volumes considering its height of 2 m. The lower head of the steam generator having a height of 0.5 m has been modelled as a branch, which acts as a connection between the annular plenum and steam generators with the pumps and pump plenum.

3.3.10 The Secondary Circuit

The secondary side of the steam generator has been modelled in RELAP as another “pipe” which is connected to the primary side by means of a heat structure for power transfer. It also consists of 20 discrete volumes as it has the same height of the primary side. The circuit consists of “time-dependent volumes (tmdpvol)” in place of the feedwater inventory and the turbine (for the steam outlet). The feedwater is forced to the steam generator with the help of a “time-dependent junction (tmdpjun)” by fixing the flow rate of 240 kg/s. The steam is then directed towards the second time-dependent volume through a “single-junction (sngljun).”

Since the secondary side design of Nuward is not available explicitly for the public, some reasonable assumptions have been made.

The data for the secondary circuit is given in the following table.

Table 15: Secondary-side parameters

Parameter	Value	Units
Pressure	4.5	MPa
Saturation temperature	257.5	°C
Feedwater inlet temperature (initial)	164 (93.5°C subcooled heating)	°C
Steam outlet temperature (initial)	301.5 (44°C superheating)	°C
Secondary flow rate	240	kg/s
Feedwater inventory	2500	kg

3.3.11 Passive-Safety System

The introduction of the passive-safety systems is a recent development in advanced reactors small modular reactors. These systems are mainly employed to remove the residual core decay heat in adverse conditions. It is observed that the induction of these safety systems in new power plants would greatly reduce complexity and capital cost. It enhances safety of the plant during worst-case accident scenarios and eliminates the use of active safety systems and the costs coupled with the installation of the same [24].

The Passive-Safety System of the E-SMR (also called the Decay Heat Removal System or DHRS) is connected to the primary circuit in a similar manner to that of the secondary side. Two safety steam generators exist in the annulus of the primary side. In the RELAP model, the passive-safety side of the steam generators are again connected by the means of heat structures to the primary side. It is initially taken to be filled half with saturated water and the other half with saturated steam. Likewise, the entire system is initially taken to be at saturation at 50°C. Further on, a pipe connects the S-CSG to a condenser, data of which is listed in the table below, which is in turn connected to a pool filled with water at standard temperature and pressure. This condenser is again connected by the means of a pipe to the inlet of the S-CSG.

Table 16: Passive-Safety System Parameters

Parameter	Value	Units
Temperature	50	℃
Saturation pressure	0.01235	MPa
Condenser height	2	m
No. of tubes	5	
Tube diameter	0.0508	m
Pool height	6	m
Pool flow area	0.833	m^2
Pool volume	5	m^3
Pool operating pressure	0.1	MPa

3.3.12 Containment

The Nuward Containment is made of a small steel containment submerged in a large water pool which provides lots of thermal inertia. In normal operating conditions, apart from the outage period, the containment is kept isolated without external venting (only with internal circulation). The containment pressure is kept at a lower value than the atmospheric pressure in a static way. Mechanical heat removal is not needed since the containment is passively cooled by submerging it in the external pool both in normal and accidental situations [16].

The containment has not been modelled on RELAP but has been summarised here however.

The design parameters of the containment are listed below:

Table 17: Containment parameters

Parameter	Value	Units
Internal diameter	15	m
Height	16	m
Operating pressure	(Slightly lower than) 1	MPa
Wall thickness	1.75 / 0.0445 [25]	inch / m

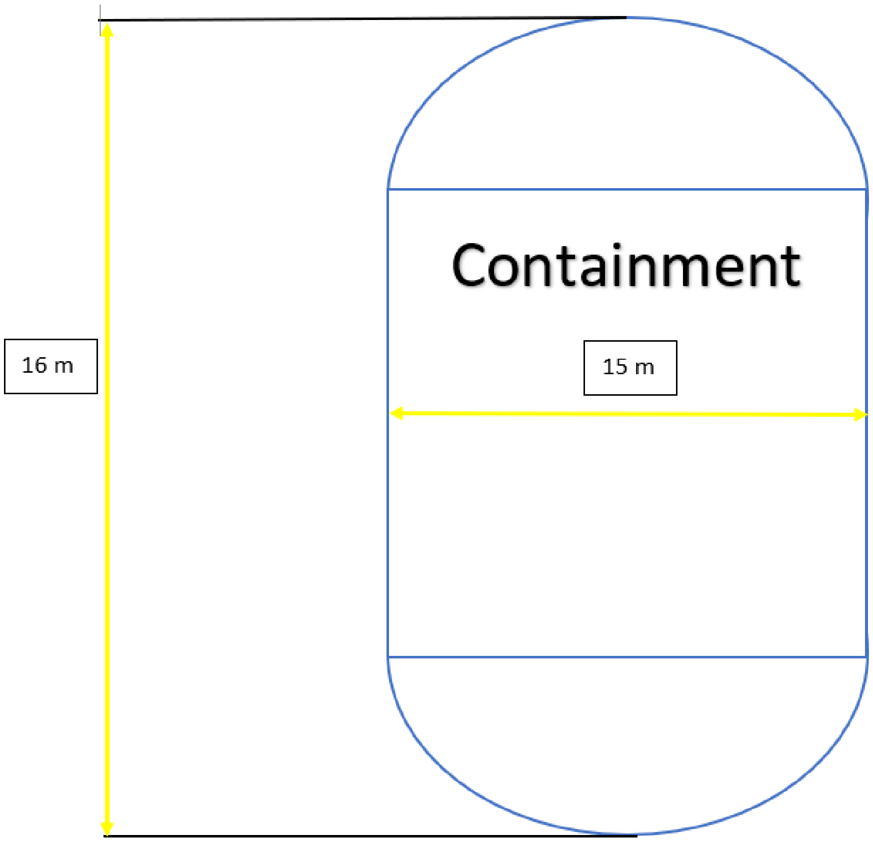

Figure 20: Containment dimensions

4 Simulation and Results Analysis

In this section, the results for the steady-state and transient simulations have been recorded and analysed.

4.1 Steady-state Analysis

The steady-state simulation for the Nuward SMR has been run considering nominal power conditions. The Relap model which has been discussed in the previous was run using the version RELAP5/MOD 3.2. The quantities that were observed were the primary system pressure (pressure in the pressurizer), secondary side pressure drop, pressure drops in each component in the primary system, core inlet and outlet temperatures, and the SG (primary and secondary) inlet and outlet temperatures.

The simulation was run for a total of 8000s, with time steps in between (50s, 100s, 200s, 400s, 600s, 1000s, 2000s and 4000s). The power of 540 MW is forced as an input for the core and it is initially a ramp dependent on time as shown below:

$$P(t) = \begin{cases} 0, & t = 0 \\ 540\ MW, & 50\ s \leq t \leq 8000\ s \end{cases} \tag{4.1}$$

This power is then almost perfectly transferred to the secondary side through the steam generator. The flow rate is input initially as a constant parameter throughout all the time steps starting from 0 s so it is independent of time. This is done using a time-dependent junction ("tmdpjun") to fix the parameter, before the pump plenum.

$$\dot{m} = 3700\ kg/s$$

On the secondary side a similar approach is considered for a flow rate of 240 kg/s.

The primary system pressure is calculated at the pressurizer component as this gives us the most accurate value without pressure drops coming into play. The plot against time is given below.

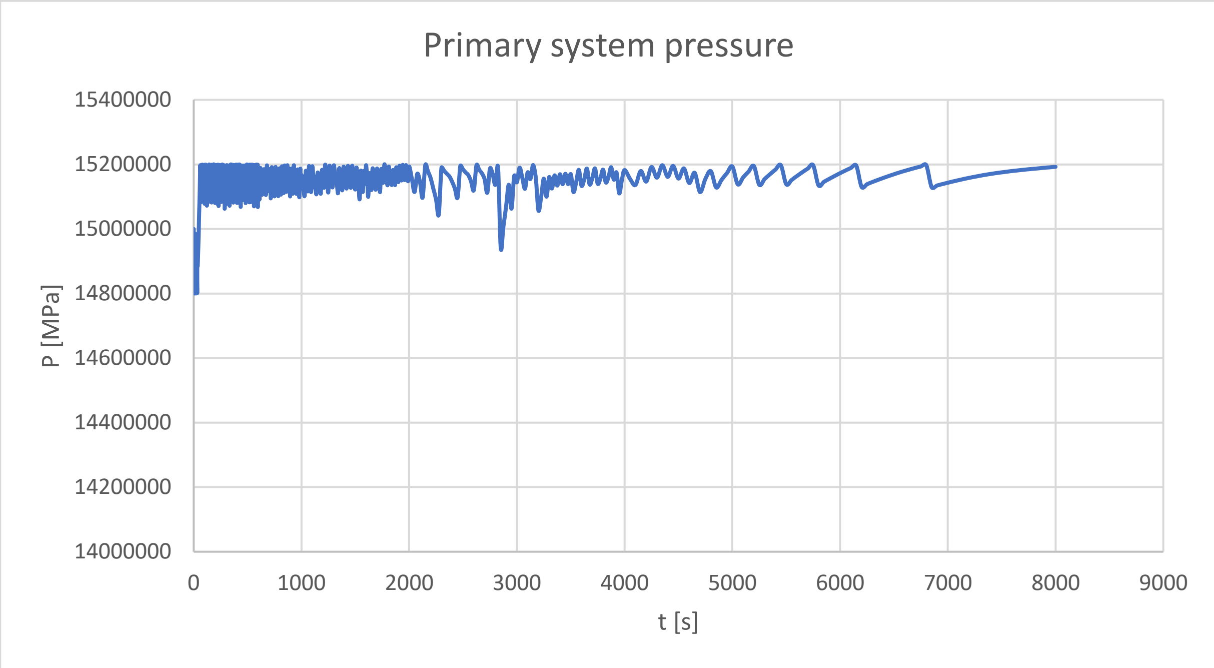

Figure 21: Primary system pressure (pressurizer)

Since there are trip control valves that have been connected to the pressurizer to control the pressure, it is noticed that it does not increase beyond 15.2 MPa and decrease beyond 14.8 MPa. Initially there is a sudden increase in the pressure possibly due to the increase in power until 50 s. The pressure fluctuates a lot around approximately 15.15 MPa for about 2000 s and then reaches a steady value of around 15.12 MPa.

The next plot deals with quite common parameters encountered in the nuclear sector mostly with LWRs. The core inlet and outlet temperatures have been compared with the SG inlet and outlet temperatures. The initial trend of a sudden decrease is common to all the temperatures. This could be due to the power increase and hence the effect of the temperature reactivity feedback. As the power becomes constant after 50 s, the temperature steadily increases and then reaches a steady-state value just around 5000 s. The unidentical nature between the core outlet and the SG inlet is due to the heat transfer surface area of the SG. This difference can be reduced by reducing the heat transfer area probably by opting for another CSG design. The exact numbers and comparisons with design values have been listed in Table 18.

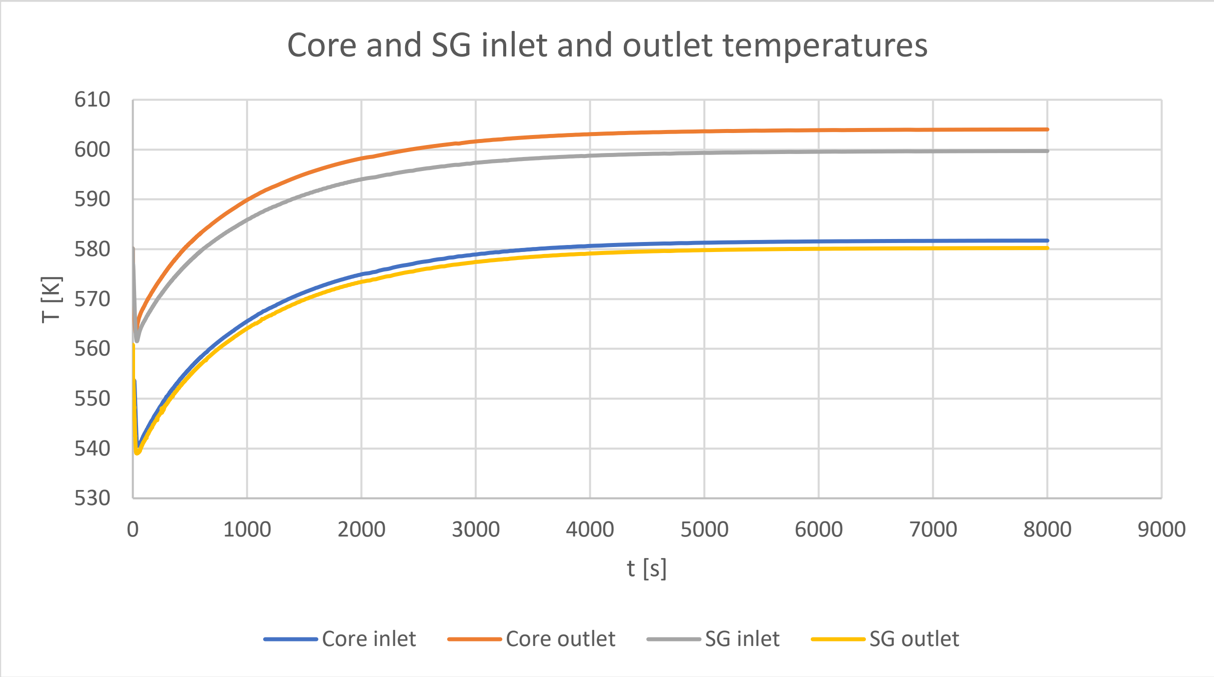

Figure 22: Core and SG temperatures

Furthermore, the pressure drops have been displayed for each component in Figure 23, according to the steady-state run. The values for the major components have been listed below:

$$\Delta P_{core} = -28\ kPa$$

$$\Delta P_{riser} = -37.9\ kPa$$

$$\Delta P_{SG} = -276\ kPa$$

$$\Delta P_{PP} = -3\ kPa$$

$$\Delta P_{dc} = 37.3\ kPa$$

The net pressure drop will not be equal to zero as this value will be used to decide an appropriate specification for the pump that is to be installed before the pump plenum.

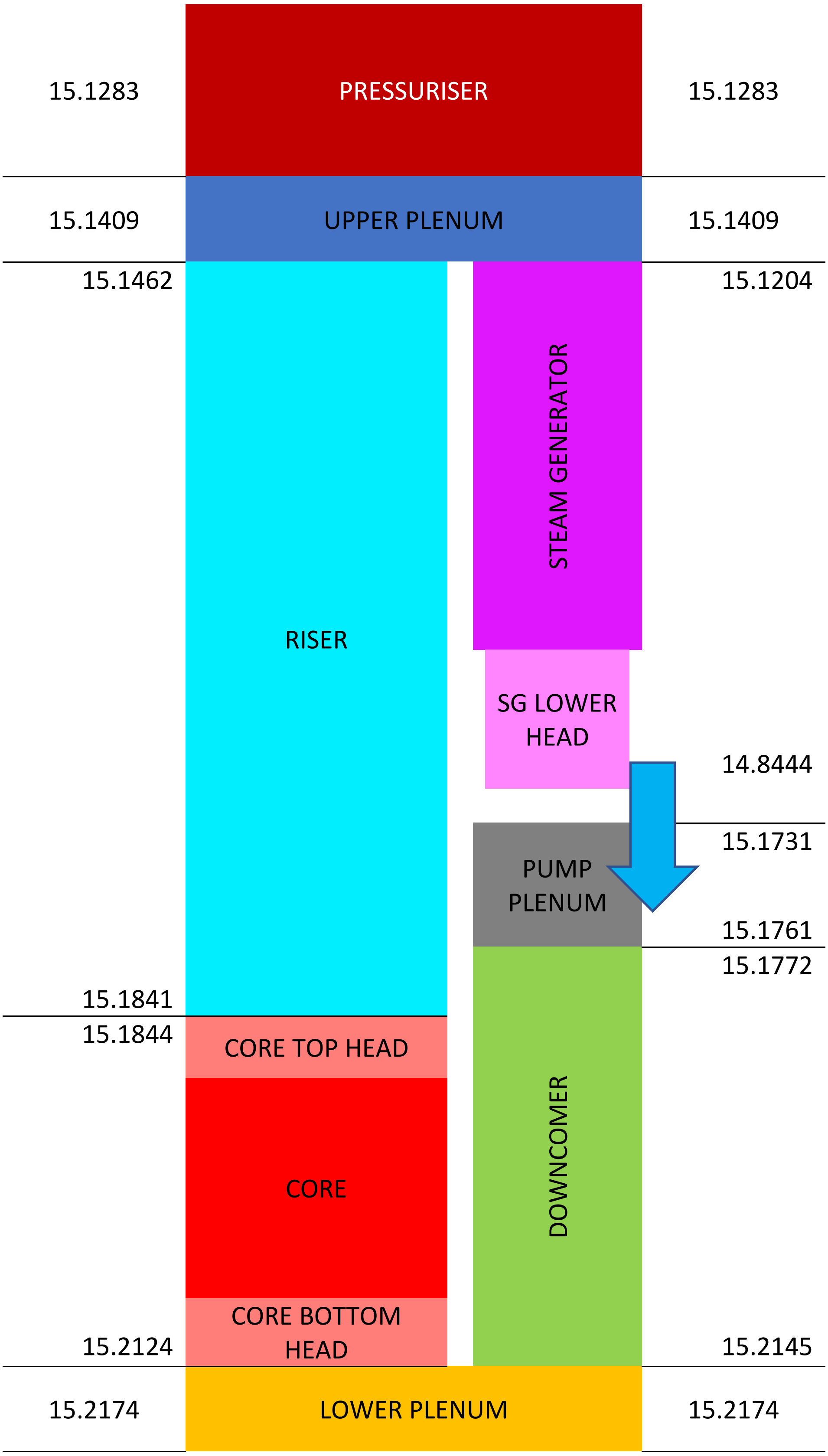

Figure 23: Pressure drops for each component of the E-SMR. All values in [MPa].

Next, the secondary side SG inlet and outlet temperatures have been plotted.

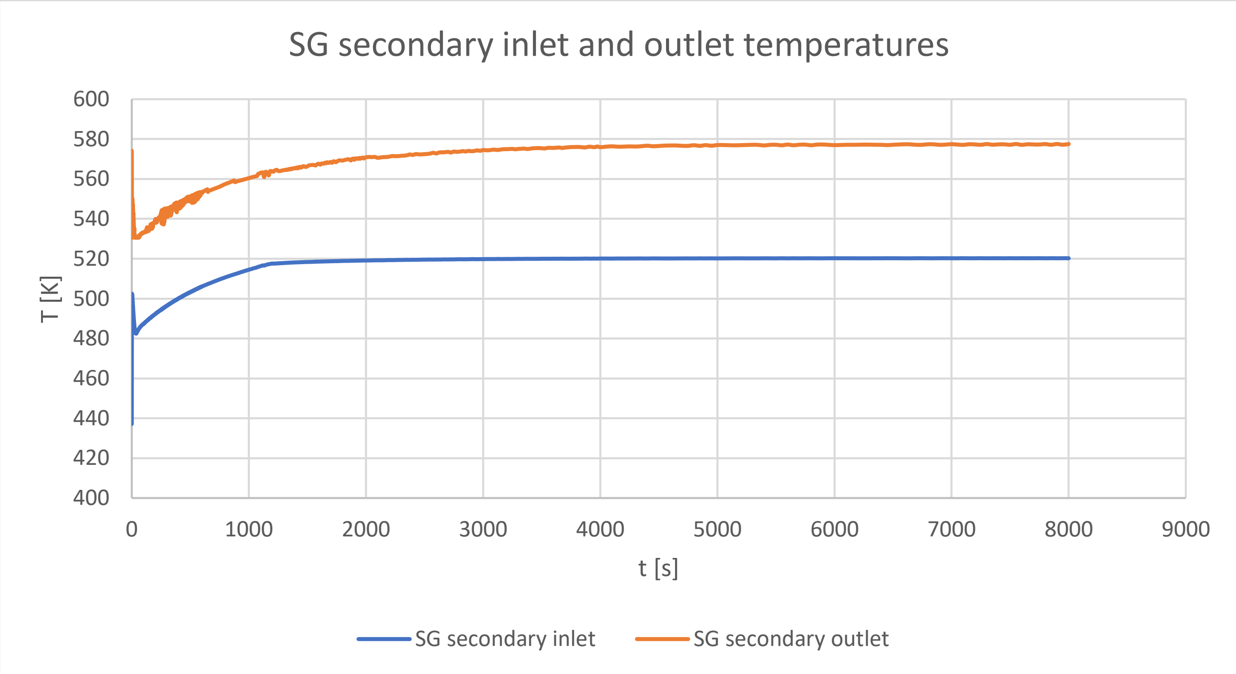

Figure 24: SG secondary temperatures

Compared to the initial temperature values that were set, the steady-state values show a different result. The inlet temperature is subcooled as expected and reaches a steady value of around 520 K (247.5°C) i.e., 10°C of subcooling and the outlet reaches a value of about 578 K (305.5°C) i.e., 48°C of superheating. This difference turns out to be quite reasonable although there needs to be a substantial amount of subcooling initially in order to maintain subcooled water in the first two volumes of the SG. The axial temperature distribution in the SG secondary is given below.

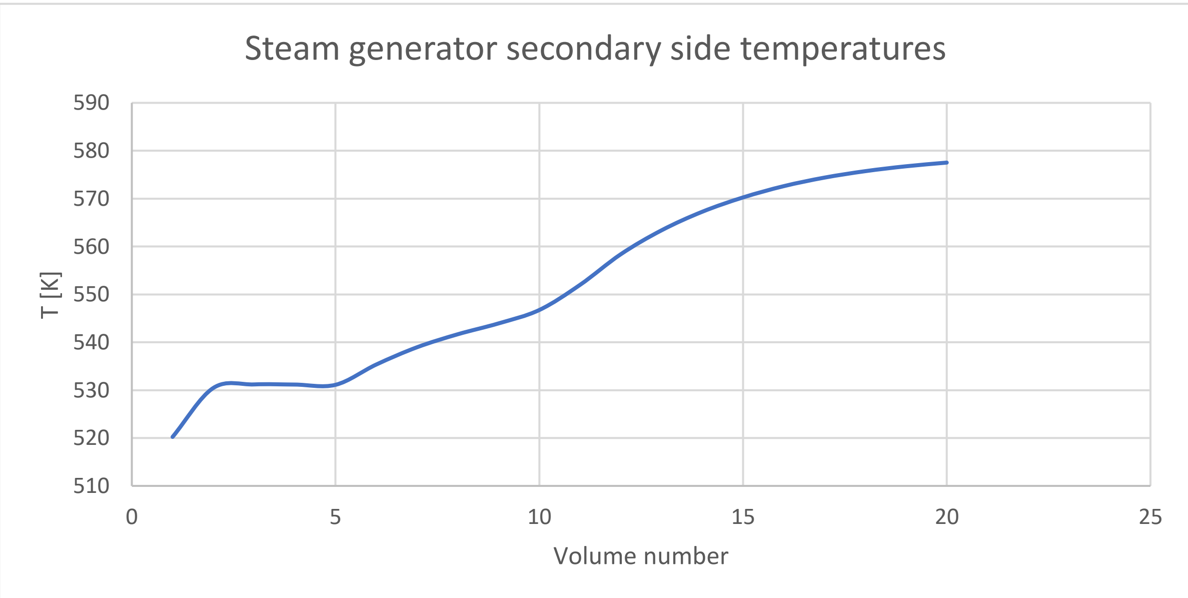

Figure 25: Secondary SG axial temperature distribution (20 volumes)

Table 18: Comparison between design values and RELAP simulated values

Parameter	**Design Value** [5]	**RELAP5**
Primary pressure	15 MPa	15.1 MPa
Core inlet temperature	280 °C	309 °C
Core outlet temperature	307 °C	331 °C
SG inlet temperature	307 °C	327 °C
SG outlet temperature	280 °C	307 °C
Core pressure drop	-	- 28 kPa
SG primary pressure drop	-	- 276 kPa
SG secondary pressure drop	-	- 46.7 kPa
Primary flow rate	3700 kg/s	3700 kg/s
Secondary flow rate	240 kg/s	240 kg/s

5 Conclusion and future developments

The design values of the Nuward SMR taken from [5], have been verified using the thermal-hydraulic code RELAP5. The steady-state values seem more or less matching although there are a few differences due to some design modifications, such as in the temperatures and flow rate of the secondary side of the SG and the addition of the annular plenum. The pressure drops of all the major components of the E-SMR have also been investigated in an effort to understand the ideal pump specifications that are required for this application.

Further developing this topic, transients related to accident scenarios could be studied for example the Loss of Flow Accident (LOFA), Station Black-Out (SBO) or the infamous Loss of Coolant Accident (LOCA). There could also be an additional scenario wherein the feedwater supply in the secondary side is cut off, and the transient effects can hence be observed on the primary side.

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List of symbols

Variable	Description	SI unit
H_{dc}	Downcomer height	m
$D_{in,ann}$	Annulus inner diameter	m
$D_{out,ann}$	Annulus outer diameter	m
A_{dc}	Downcomer flow area	m^2
V_{dc}	Downcomer volume	m^3
$D_{h,dc}$	Hydraulic diameter	m
H_{lp}	Lower plenum height	m
a_{lp}	Lower plenum major and minor axis	m
A_{lp}	Lower plenum flow area	m^2
V_{lp}	Lower plenum volume	m^3
$D_{h,lp}$	Lower plenum hydraulic diameter	m
H_a	Active fuel length	m
$D_{in,core}$	Active core diameter	m
N_A	No. of fuel assemblies	-
V_{core}	Core volume	m^3
P_{fa}	Fuel assembly pitch	m
N_{fr}	No. of fuel rods er assembly	-
N_{gt}	No. of guide tubes per assembly	-
N_{ins}	No. of instrument tubes per assembly	-
H_{core}	Overall fuel assembly length	m
A_{ass}	Assembly area	m^2
A_{fr}	Fuel rod area per assembly	m^2
A_{gt}	Guide thimble area per assembly	m^2
A_{ins}	Instrumentation area per assembly	m^2
$A_{ass,flow}$	Assembly flow area	m^2
A_{core}	Core flow area	m^2
t_{barr}	Core barrel thickness	m

t_{bp}	Baffle plate thickness	m
$D_{h,core}$	Core hydraulic diameter	m
H_{riser}	Riser height	m
A_{riser}	Riser flow area	m^2
V_{riser}	Riser volume	m^3
$D_{h,riser}$	Riser hydraulic diameter	m
N_{RCCA}	No. of RCCA tubes	-
s_{RCCA}	RCCA tube side length	m
N_{SC}	No. of support columns	-
D_{SC}	Support column diameter	m
H_{up}	Upper plenum height	m
$D_{in,RPV}$	RPV inner diameter	m
A_{up}	Upper plenum flow area	m^2
V_{up}	Upper plenum volume	m^3
$D_{h,up}$	Upper plenum hydraulic diameter	m
H_{pr}	Pressurizer height	m
$D_{in,pr}$	Pressurizer base inner diameter	m
D_{so}	Surge orifice diameter	
N_{so}	No. of surge orifices	-
A_{pr}	Pressurizer base flow area	m^2
V_{pr}	Pressurizer volume	m^3
$D_{h,pr}$	Pressurizer hydraulic diameter	m
H_{ap}	Annular plenum height	m
$A_{occ,SG}$	SG occupied area	m^2
A_{ap}	Annular plenum flow area	m^2
V_{ap}	Annular plenum volume	m^3
$D_{h,ap}$	Annular plenum hydraulic diameter	m
P	Power	MW
$\dot{m}$	Mass flow rate	kg/s
ΔP_{core}	Core pressure drop	kPa
ΔP_{riser}	Riser pressure drop	kPa
ΔP_{SG}	SG pressure drop	kPa
ΔP_{PP}	Pump plenum pressure drop	kPa
ΔP_{dc}	Downcomer pressure drop	kPa

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