Development of a CAD model for the European Sodium Fast Reactor in view of using the thermal hydraulics code TRACE

Final thesis

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Contents

Abstract

The current thesis was made to describe the CAD model and its use for the Horizon-2020 ESFR-SMART project (European Sodium Fast Reactor Safety Measures Assessment and Research Tools), a sodium cooled fast reactor development. There are two main purposes for the CAD model, such as to create a conceptual design based on previous experiences, so that it is easy to visualize it for all the project members. Moreover, to produce a model which can provide crucial input information for further research such as for the use of TRACE thermal hydraulic code. To create this 3D CAD model, the software used was Autodesk Inventor, which is a product development 3D CAD software. After the model was finished, volume and surface information was extracted manually from Inventor for the use of TRACE. During the thesis project, important achievements were made. First, a CAD model was created presenting all the primary systems of the reactor, the secondary cooling system and the decay heat removal system. Following this, information was extracted for further analysis and used in TRACE. Overall, the project proved the importance of such a model in helping for further concept development and providing input information for computational research tools.

1. Introduction

1.1. Generation 4 technology

The today`s nuclear reactors are mainly generation 2 and generation 3 reactors with a few newly deployed generation 3+. These reactors, although can differ greatly, use only a few percent of the uranium fuel to produce energy. This feature has two main drawbacks. First, relatively high amount of waste is produced. Secondly, the known uranium reserves, which can be extracted relatively cheaply with the current technology at the 2014 level of requirement, would last around 135 years only [1]. To tackle these problems, generation 4 reactor technology was introduced to produce less depleted uranium and to use up most of the fertile fuel in the reactor. Apart from this purpose, the new reactor designs were meant to be cheaper to build, safer to run and more efficient in energy production. In this way, Generation 4 technology is an improved way for utilizing nuclear fuel and to produce energy.

To develop and to choose the next generation nuclear reactor concepts, the Generation IV International Forum (GIF) was initiated in 2000, where 14 countries were represented as part of the forum [2]. This forum reduced the high number of new generation 4 concepts to only 6 possible nuclear reactor technology and $6 billion was allocated over the next 15 years for [2] the research and development of these concepts.

The 6 different types of generation 4 reactors that were chosen are [2]:

- Gas-cooled fast reactors
- Lead-cooled fast reactors
- Molten salt reactors
- Sodium-cooled fast reactors
- Supercritical water-cooled reactors
- Very high-temperature gas reactors

Four of these reactor technologies are using fast or epithermal (Molten salt reactor) neutron spectrum during operation and only two of them use slow neutron as the current conventional nuclear reactors.

1.2. Sodium cooled fast reactors

1.2.1. Reactor Coolant

Fast reactor technology has been researched since the early 1950´s and experimental reactors which used different coolant mediums were built. All of these reactors used some sort of liquid metal but, in most of the cases, specifically sodium. Sodium cooled fast reactor technology uses liquid sodium as the coolant in the reactor. This generation 4 reactor technology is the one that has been in use for the longest period of time. Sodium cooled fast reactor operation has an operating experience of around 300 years [3]. The fast reactors have a relatively high power density associated with their core which demands a coolant with very effective heat removal properties such as liquid metals. As liquid sodium is a relatively good heat transfer medium, in most of the ever created liquid metal cooled reactors, sodium is the coolant material used due to its high thermal conductivity (75 W/m·K), low melting point (98 Celsius) and boiling temperature (892 Celsius, which should be avoided).

1.2.2. History

The first Sodium Cooled fast nuclear reactor in the world that produced electricity was the EBR-I, which was first critical in 1951 in Idaho, USA. It had a 0.2 MWth nominal full power capacity and it was finally shut down in 1963. Following this reactor, the main objective was to increase the nominal power, which was achieved by EBR-II, Fermi and FFTF experimental fast reactors.

In other countries, there was also a growing interest towards sodium cooled fast reactors, as a result, some experimental reactors were built. For example, the UK made the DFR (1959) and PFR (1974). The USSR built the BR-10 (1958), BOR-60 (1968) and BN-600 (1980). To continue with, France made Rapsodie (1967), Phenix (1973) and Superphenix 1 (1985). Equally, Japan built the Joyo (1977) and MONJU (1994) [3]. Just to mention a few countries and their projects.

In the west, the research on this technology slightly slowed down in the past 15 years and many projects have been cancelled, particularly in the US, Italy, Germany and France. In contrast, there is a growing interest in the technology in the developing countries, with the strongest will from China and India to further proceed with the development. [4].

1.2.3. SFR technology

A characteristic feature of the Sodium Cooled Fast Reactor Technology is the coolant circuit structure of the reactor. In such structure, the heat from the primary sodium is conducted to a secondary sodium loop. Then, this secondary sodium loop is exchanging the heat with the water and makes the steam for the steam turbines. This extra sodium loop is crucial since the sodium reacts strongly with water, causing easily sodium fire. In case of a fire in the primary sodium, it would contain radioactive material, which would be much more dangerous than a secondary sodium fire, characterizing the incident as a nuclear accident and not as a chemical accident.

There are two different concepts for Sodium Fast Nuclear Reactors, namely the loop and the pool type concepts. The main difference between the two is that, in the loop type concept, the intermediate heat exchangers between the two sodium loops are outside of the reactor vessel, whilst, in the pool type, it is situated inside of the reactor vessel. These two designs have different kind of advantages which correspond to the main difference between them. On one hand, in the loop type concept, since the intermediate heat exchangers are outside of the primary reactor vessel, in the event of a leak in any of the heat exchangers, it is easier to fix the problem. Moreover, the size of the primary reactor vessel can be smaller as it does not have to include the intermediate heat exchangers. Furthermore, by having the intermediate heat exchangers outside of the vessel, they can be placed above the reactor vessel, which enhances the natural circulation in the system.

On the other hand, in the pool type concept, since the reactor vessel is bigger than in the loop design, as it contains the heat exchangers, there is more primary sodium available in the system. This means that, in the case of an accident, there is more coolant material available to dissipate the decay heat from the reactor core. Secondly, as the heat exchangers are in the main vessel, in case of a leak in any of the heat exchangers, the sodium will stay in the reactor vessel and would not contaminate the surroundings. Therefore, the risk of sodium leakage to the surroundings is greatly reduced.

Most of the first experimental SFR reactors were made following the loop type design. Nonetheless, the latest type designs and future SFR concepts, except in Japan, are based on the pool type concept.

In Figure 1.1, a schematic can be seen about a pool type SFR design concept. As the figure shows, there are two separated sodium pools in the primary reactor vessel that makes the primary coolant circuit. From the cold sodium pool, the primary pump sucks the sodium pushing it towards the reactor core to cool down the fuel assemblies in the core. Above the reactor core, there is another pool where the sodium that has cooled down the core accumulates. This pool is called the hot sodium pool. The sodium from this pool is cooled down as it travels through the heat exchanger (intermediate heat exchanger) back to the cold sodium pool. The two sodium pools are connected through this heat exchanger, creating the primary circuit. The heat from the primary hot sodium is exchanged with the secondary sodium which flows through the heat exchanger.

The secondary sodium loop consists of a steam generator, a pump and the previously mentioned heat exchanger. The sodium in the secondary loop does not consist of radioactive material since it flows through the steam generator to heat up the water and to create steam. From this point, the generated steam is used exactly in the same way as in any other power plant, namely that the steam runs a turbine, which is connected to a generator to produce electricity. After the steam was used for the turbine, the condenser turns the steam back to water and a pump pushes it back to the steam generator.

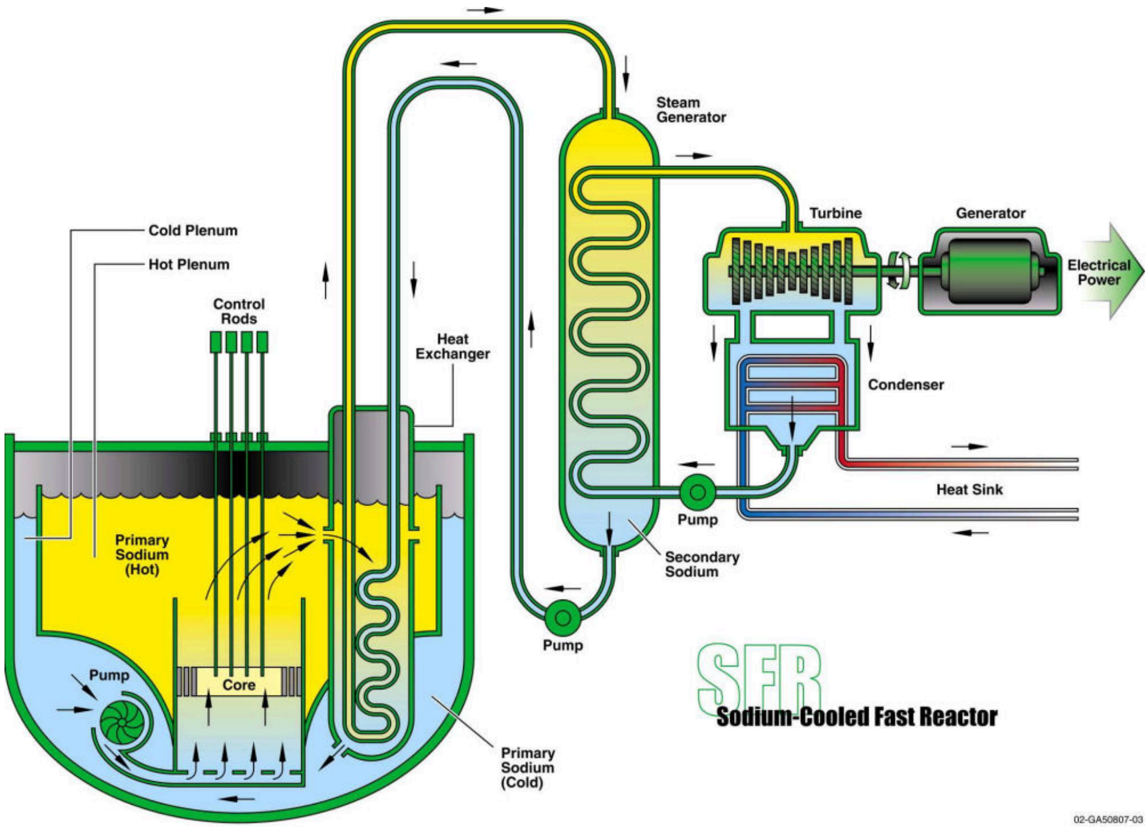

Figure 1.1 Sodium cooled fast reactor concept [5]

1.3. The project

This current thesis was based on the preparation for the Horizon-2020 ESFR-SMART (European Sodium Fast Reactor Safety Measures Assessment and Research Tools) project, which is related to the Sodium-cooled fast reactors in the sense of safety analysis. The concept of the ESFR is a 3600MWt or 1500MWe high power reactor, which is very favorable economically compared to the previous smaller sodium cooled reactors.

The aim of the ESFR-SMART project is to enhance the safety of generation 4, particularly the commercial-size ESFR, and to increase the public acceptance by proving that the safety level of the future commercial-size SFR reactors is significantly higher than the today's traditional reactors. [6]

The 5 specific aims of the ESFR-SMART project are [6]:

- To produce new experimental data for the use of validation and calibration of research tools

- To provide test and qualifications for the instrumentation used by the reactor protection system

- To provide data from the project for the validation and calibration of computational tools to support safety assessment for future SFR designs

-To choose, carry out and evaluate new mechanisms for the safety of commercial-size ESFR, using the GIF methodologies, the FP7 CP-ESFR project legacy and the codes that have been validated and calibrated and are in accordance with the update of the European and international safety frameworks taking into account the Fukushima accident;

-To support and create links between new networks, particularly, between the European sodium facilities and European students working on SFR technology.

In this sense, this thesis is aiming at the CAD model development for the ESFR-SMART project. Such CAD model was done by the use of Autodesk Inventor software. The main source of information for the design came from a previous project on ESFR [Gen09]. Furthermore, the EFR [Jul99] (European Fast Reactor) and Indian SFR studies [Raj15] served as the base to create a working concept.

The main use of the model was twofold. Firstly, it provided a starting point for the relevant decision makers for further discussion on the concept and to create a more detailed conceptual design in the future. Secondly, a very important role of the model was to research how feasible is to extract information out of the CAD files to create input files for different computational tools, more precisely, in this case, for TRACE thermal hydraulic code.

1.4. CAD system

For the model development, the so-called Autodesk Inventor software was used. Inventor is a computer aided design (CAD) software to create 3D concept models for various purposes such as prototypes, simulations, visualization and production. Regarding this thesis, the main use of the software was to create a 3D model for visualization of the concept which can be used as a base for further concept development. Furthermore, this 3D model was created to provide crucial input information (volume, size and geometry) to run different computer codes, in particular for this thesis, TRACE thermal hydraulic code.

The model was divided up into different elements to create it. Each of the elements was created individually, which was then assembled into a more complex system of elements. For example, all parts of the primary system in the reactor vessel were made piece by piece and then they were built together into one working unit. Similarly, the secondary circuits, the intermediate heat exchanger, the steam generator, the secondary pump, the sodium dump vessel and all the connecting tubes were made separately and assembled together into one piece. This approach is useful in the case of modifications on the model since one part of the system can be modified separately and then connected to the already established system at any time. Moreover, in an assembly, it is always easier to modify the position of the elements and just extend the tubing to match the new position of an element of the system.

Another useful feature of the model is that many of the systems are repeated, which brings the possibility to use different patterns or mirroring in the model. This means that after creating one element, it can be copied in a certain pattern or mirrored by a certain working plane in the model. For example, many of the systems such as the secondary circuits, the decay heat removals, the pumps, etc., were made with a circular pattern since they are repeating elements around the main axis of the model. This simplification allows to modify only one element, which was used for patterning, and, by modifying it, all the other elements are changed as well. These simplifications eased the process to change the conceptual design and to modify the model as it required.

Inventor is also capable of showing information about the model properties. The material can be set accurately, allowing the calculation of the weight of the model, and provide data for stress analyses and volume/surface area information to run different simulations. It is also possible to export the original Inventor CAD files into different file formats which can be used then directly by different codes as input files. As an example, after the original files are converted into STL files, which is a file format where only surface information is saved from the model, it can be used for OpenFOAM CFD analysis without further modifications.

For the current discussion, for TRACE code, the important information from the model was the volume and surface information since this is the input information which has to be provided for the code.

1.5. TRACE thermal hydraulic code

The thermal hydraulic code which is used at Paul Scherrer Institut (PSI) for the safety analysis is TRACE (TRACE/RELAP Advanced Computation Engine). TRACE was developed by the Nuclear Regulatory commission from the US, previously called TRACE-M. [7].

The original purpose of TRACE was to simulate accident scenarios for BWRs and PWRs, such as operational transients, loss-of-coolant accidents (LOCAs), etc. The code is also capable of modeling what is going to happen in experimental facilities made to simulate transients in reactor systems. The original code was modified at PSI to enable it to run two-phase flow simulations with sodium coolant [8]. The language which is used to program TRACE is the standard Fortran 90.

As an important part of the current work was to analyze the possibility of data extraction for TRACE input file, it is important to know how the reactor is modelled in TRACE and what the input parameters which must be provided from the CAD files are.

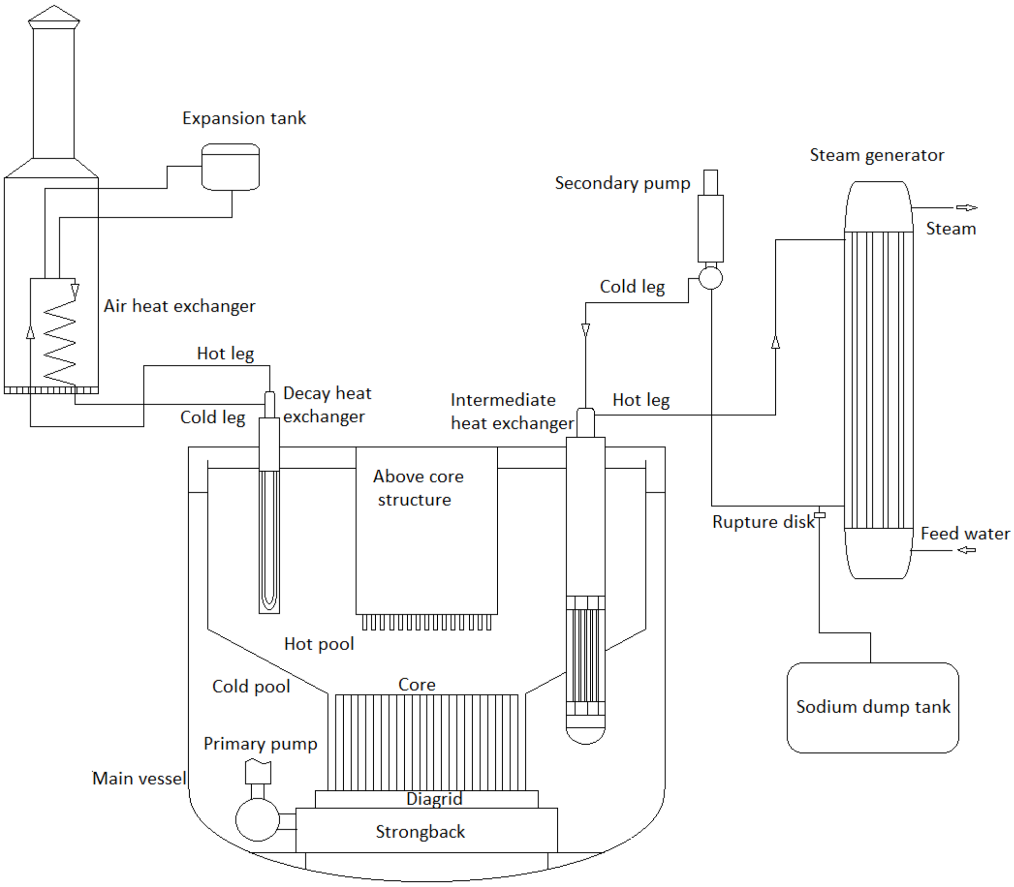

Fig. 1.2 Sodium Fast Reactor schematic drawing

The first step to create an accurate model is to create a schematic diagram of the system which should be modelled. In Figure 1.2, an example drawing can be seen for the ESFR reactor. On this drawing, all the main system components are present and the connections are visualized as well. The next step is to divide up the system into smaller units for more detailed analysis.

To model the reactor in TRACE, different simple components are available. Each element of the system can be recreated using the simple components found in TRACE. This means that the system must be divided up into such elements that it is possible to substitute the element with some components from TRACE. The most often used available hydraulic components for TRACE are the following: VESSEL, PUMP, PIPE, TEE, VALVE. All of these components are 1-dimensional ones except the VESSEL, which allows a 3-dimensional computation. Other important components are the HTSTR (heat structure) and the POWER component, which are usually used together to model the generated or deposited energy in the solid structure, e.g. Heat structure. The boundary conditions in the system are provided by the means of FILL and BREAK components. The FILL component provides the desired coolant flow, whereas the BREAK component allows the user to set a pressure boundary for the system [9].

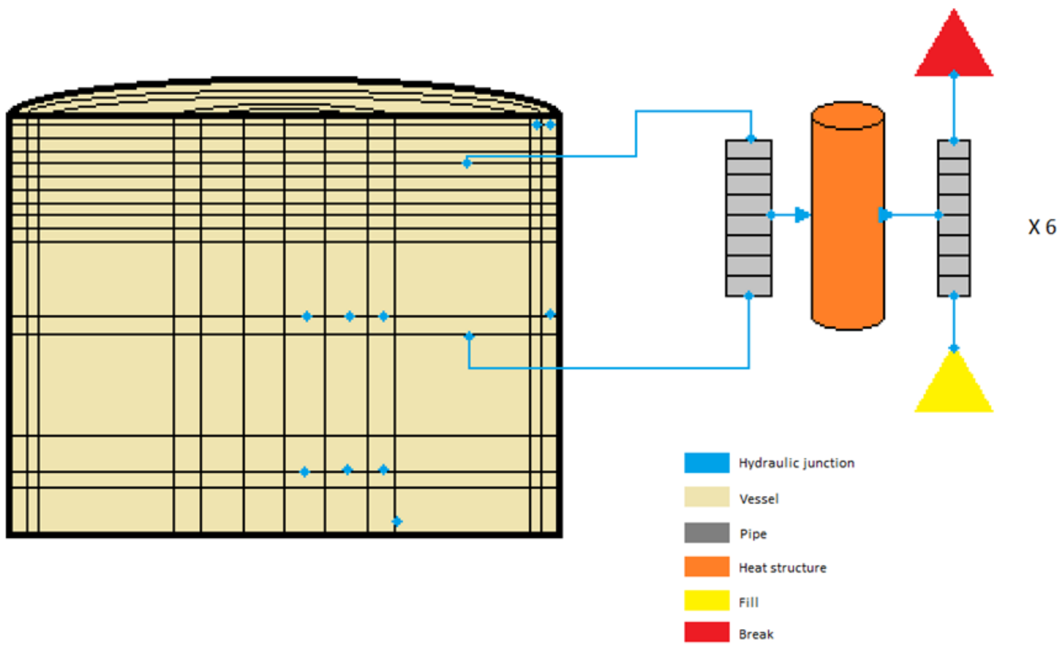

Figure 1.3 Noding diagram and hydraulic components for Intermediate heat exchanger

In Figure 1.3, at the right side of the picture, an example can be seen to show how a system element can be subdivided into basic hydraulic components. As the picture shows, an intermediate heat exchanger can be subdivided into a pipe, which symbolizes the sodium in the primary system, another pipe to allocate the sodium in the secondary system and a heat structure between the two pipes to provide the heat transfer between the two. The coolant flow inside the pipe for the secondary sodium is set by the FILL component. Besides, a BREAK component is set to the same pipe to control the pressure in the secondary part of the heat exchanger. The component set up can be nodalized into smaller physical volumes called cells. In Figure 3, this can be seen on the left side of the picture, where the vessel is divided up by horizontal and vertical lines into smaller cells. The size of the cell is dependent of its position. Usually, the size is advised to be between 0.1-3m, smaller where the spatial variation is anticipated to be smaller and higher where the spatial variation is expected to be lower [9]. Inside the cells the conduction, fluid and kinetics equations are averaged out. To run the simulation, data about the cells should be provided such as cell lengths, cell fluid volumes, fluid flow area, flow channel hydraulic diameter, etc.

2. CAD model

To acquire the required information about the input data for TRACE and to produce a conceptual design for further discussion and development, a 3D CAD model was created. On this model, all the main elements of the primary system, the secondary coolant loops and the decay heat removal system are shown.

2.1. Main system with Secondary circuits

In Figure 2.1, the main system is shown with the secondary sodium loops to produce steam; furthermore, the decay heat removal circuits are visible as well. As it is shown in the picture, there are 6 secondary sodium loops, 6 decay heat removal loops and 3 primary pumps connected to the vessel. On the picture, the blue tubes represent the cold legs, whereas the red tubes represent the hot legs.

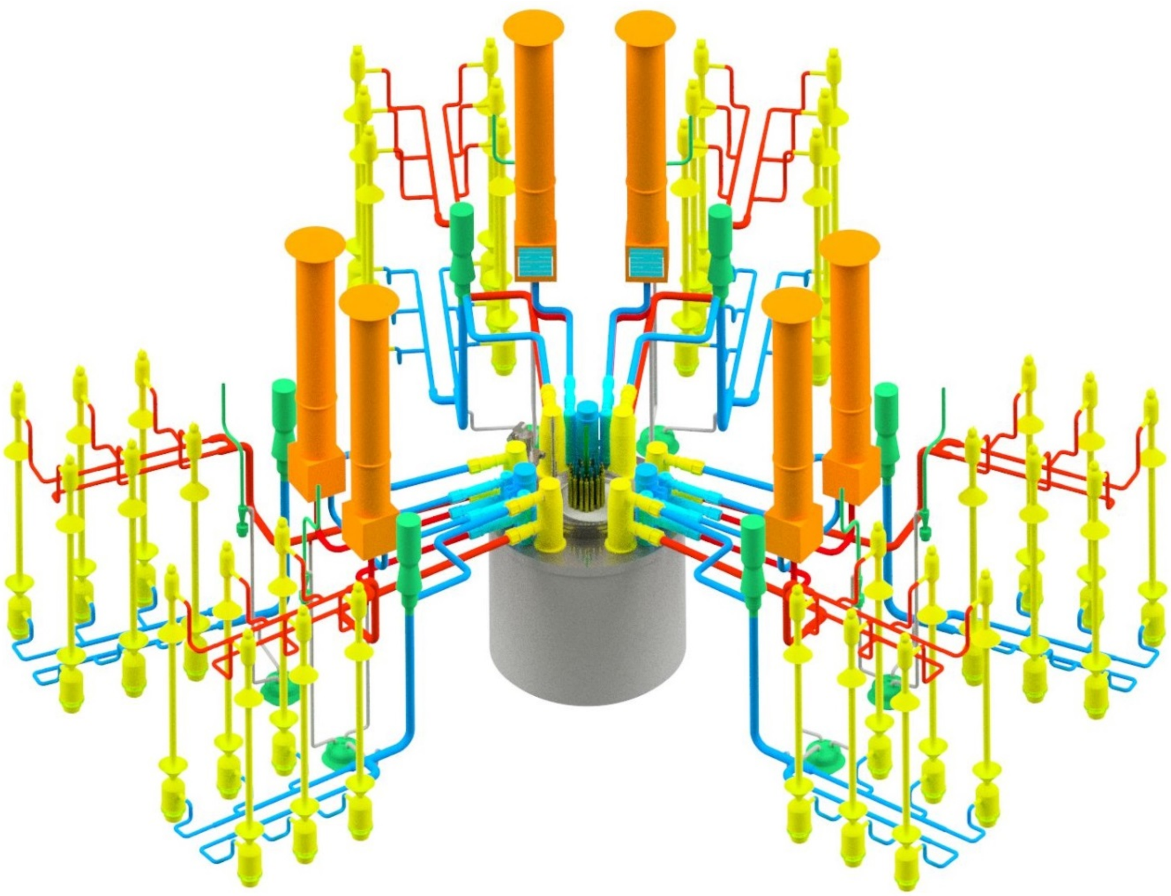

Figure 2.1 Primary and secondary systems

2.2. Primary system

In Figure 2.2, the primary system is shown with its components. At the middle, the Inner vessel divides up the sodium into the hot pool and cold pool. The sodium situated above the core makes up the hot pool, whereas the sodium below creates the cold pool. The primary pump hangs into the cold pool, sucking the cold sodium and pushing it into the strongback. From the strongback, the sodium flows into the diagrid through 18 flexible sleeves. From the diagrid, the sodium flows into the shroud tubes which hold the fuel assemblies. Following the shroud tubes, most of the sodium flows up into the fuel assemblies, whereas a minor amount flows up through the inter-assembly gaps between the fuel assemblies. This upward flow has the purpose of cooling down the core of the reactor. Moreover, a part of the sodium flows downwards to the core catcher and the vessel cooling pipes. Some cooling is provided to the upper part of the reactor vessel through these pipes.

After the sodium left the reactor core, it gets into the hot pool. From the hot pool, the sodium is sucked through the intermediate heat exchanger, which provides the connection between the hot and cold pool. As the sodium cools down and gets into the cold pool, the circulation follows the path which was described previously. Some of the sodium, as it leaves the reactor core, gets into the "Above core structure", in which the control rod drive lines are situated.

The decay heat exchanger hangs into the hot pool and provides a passive way of heat exchange from the hot sodium region. This decay heat removal system is based on natural circulation, so, even at normal conditions, a small amount of heat is taken away by the system from the primary sodium.

Regarding the size of the vessel, it is 17360 mm in diameter and its height, including the roof, is 18000 mm.

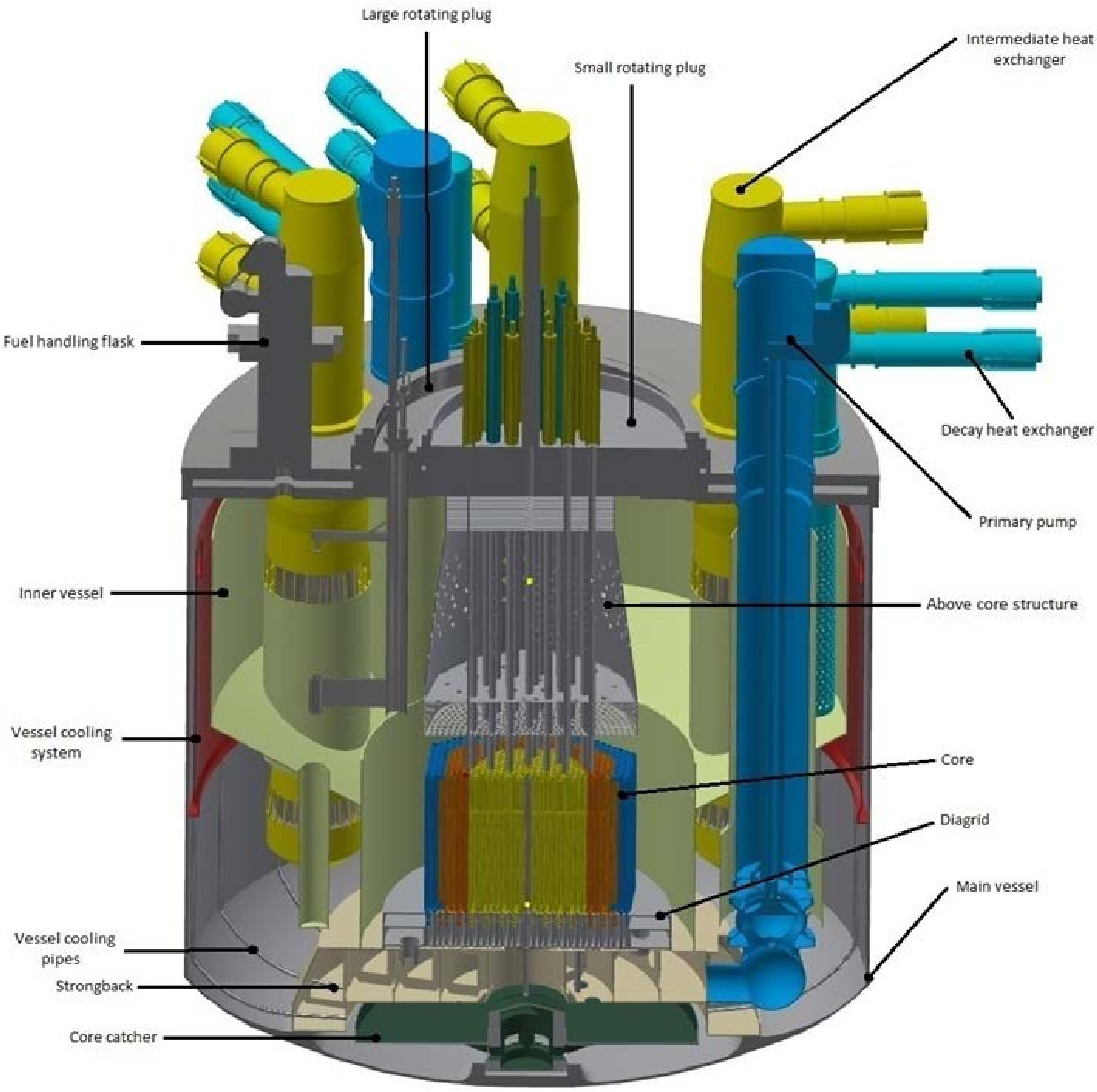

Figure 2.2 Primary system

2.3. Reactor core

The reactor core is presented in Figure 2.3. The core is divided up into 3 different regions. Firstly, the inner fuel assemblies, which consist of fertile material at the bottom and fissile on top of it, can be seen. Secondly, the outer core assemblies, which have its biggest difference compared to the inner core assemblies by having a larger amount of fissile material above the lower blanket, can be seen. Lastly, around the outer core, the shielding/reflector can be seen. Furthermore, in the core, there are 13 empty slots which are left empty to provide a free path for corium towards the core catcher in the case of a severe accident.

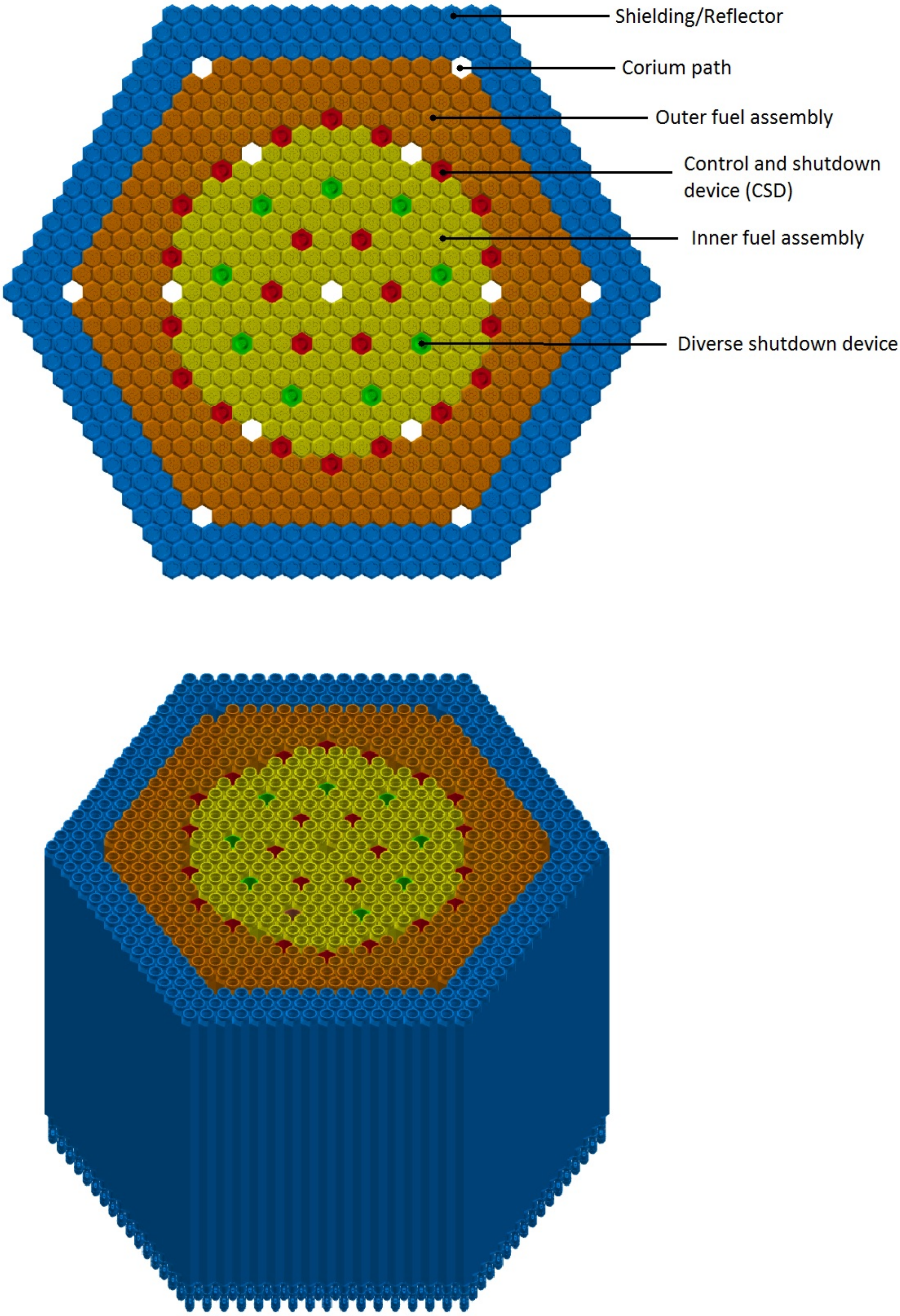

Figure 2.3 Core layout

In the core, the hexagonal wrapper tubes are also shown to accommodate the 2 different types of control rods. There are 24 hexagonal wrappers with a hexagonal inner side for the Control and Shutdown rods and there are 9 hexagonal wrappers with a cylindrical inner side for the

Diverse shutdown devices. The size of the reactor core with the shielding around is about 6000 mm in diameter and has the height of about 4500 mm.

In Figure 2.4, a modified core structure can be seen. This is a good example to show how easy it is to modify an already existing design to make a few changes on the concept and to present the results. A slight change is shown on the layout of the wrapper tubes of the two different control rod devices. First, the number of Diverse shutdown device (DSD) has decreased from 9 to 6. Secondly, the layout of Control and shutdown device (CSD) has changed as well to be completely symmetric. Thirdly, the empty spaces provided for the corium at the periphery between the inner and outer core have moved one slot closer to the center of the core. The inner and outer core layout otherwise stayed the same.

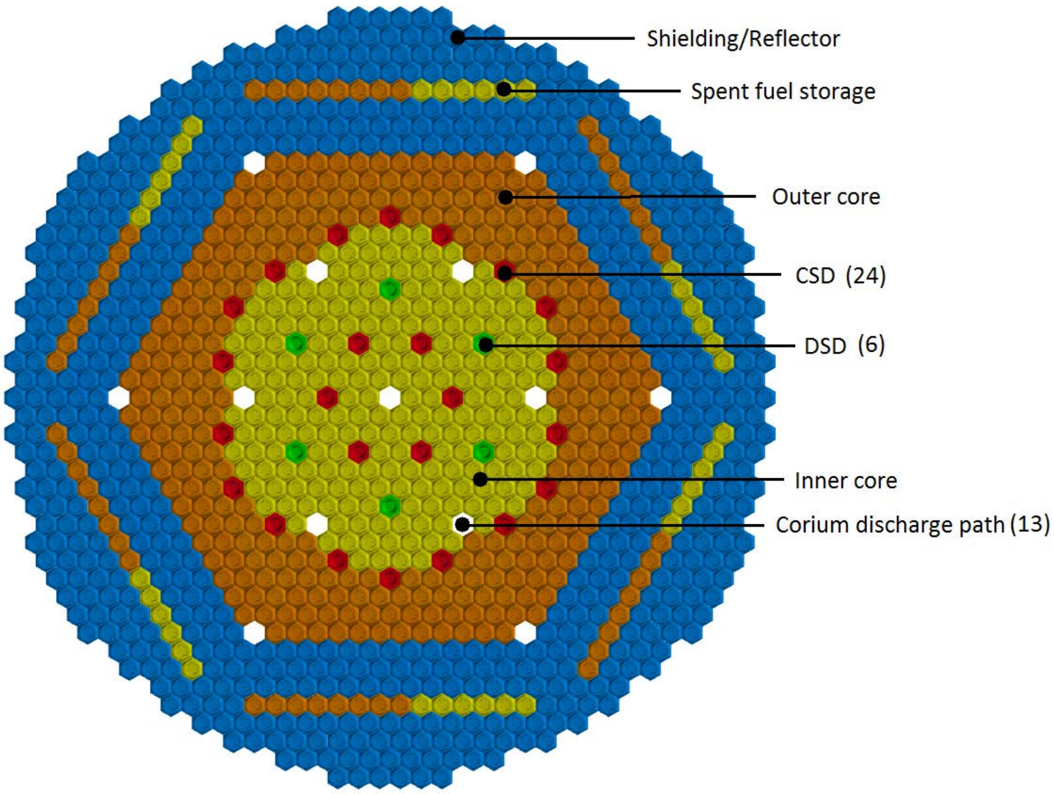

Figure 2.4 Modified reactor core

Perhaps the biggest change on the core is the allocated place for internal spent fuel storage. With this modification, it is possible to keep the spent fuel inside the primary sodium, where its heat production reduces with time. The symmetry of the core allows using a 6-batches scheme, in which three steps take place every year. Firstly, the spent fuel from the internal storage is discharged from the reactor and put into the external storage. Secondly, for each of the 60°-sectors, 6 fuel assemblies from the inner core zone and 8 from the outer core zone are moved to the internal storage. Thirdly, the fresh fuel assemblies are loaded into the core. The core was also extended with a few layers of extra shielding, which is very useful to protect the primary system components, such as intermediate heat exchangers and primary pumps, from extra radiation damage and extend their possible lifetime.

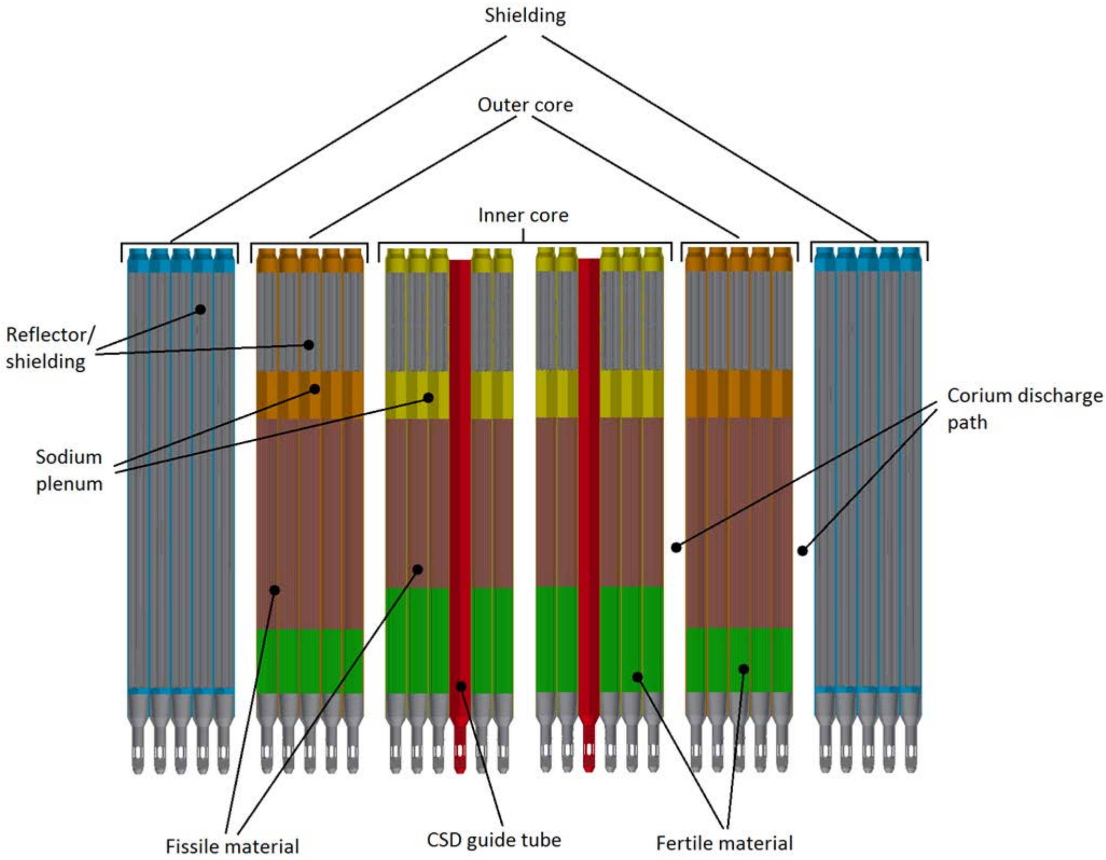

Figure 2.5 Modified reactor core regions

To show the structure and different regions of the fuel assembly, it was sliced with a plane at the middle. Figure 2.5 shows the plotted results. As the picture shows, at the sides, the shielding pins are running through the whole hexagonal wrappers. Besides, shielding is provided at the top of each Breeding and Fuel assemblies as well. In-between the different sections, the spaces show the empty slots for the corium in the case of a severe accident. Through those channels, there is a free path provided downwards, towards the core catcher. After the shielding, the next section is the outer core, where the green color shows the height of the fertile material, whilst the brown region shows the height of the fissile material. Moving inwards inside the core, the next section is the inner core. The main difference between the inner and outer core is the ratio of fissile and fertile material, the inner core having a higher ratio of fertile to fissile. This difference can also be seen clearly in the picture.

2.4. Fuel Assembly

Figure 2.6 represents an example of the fuel assembly in the core. As the sodium flows into the diagrid and to the shroud tubes, the resulting hydraulic pressure is such that it pressurizes the spike body of the assembly and, therefore, it is held in position.

The sodium coolant flows into the assembly through the coolant inlet ports to cool down the fuel pins. The amount of coolant flowing through the assembly can be adjusted by the gag assembly, depending on the position of the assembly inside the core. The gag assembly is made of cylindrical layers of honeycomb shape flow restrictors.

Between the fuel pins in the assembly, there are spacer wires to keep the position and the distance between the pins. Above the fuel, there is a sodium plenum, with a size of around 400 mm height, which is followed by the neutron shield pins. The size of the fuel pins is 8.2 mm in diameter and 2400 mm in length. The size of the spacer wires is 1.2 mm in diameter and, on a helical way, it wraps the fuel pin around along the whole length of it. Between the shielding pins, the same spacer wire technology is used as for the fuel pins. To continue with, at the top of the assembly, a lifting head is situated to ease the refueling/defueling processes.

In Figure 2.7, a magnified picture of the bottom part of the fuel assembly is shown. By having this picture, more details can be seen in the fuel assembly, such as that there are labyrinth seals above and under the coolant inlet port. These seals are different in size. The one at the top is bigger in diameter than the one at the bottom. By having this difference in size, the flow is regulated, letting only a smaller amount of coolant to flow upwards through the inter-assembly gap and forcing a higher amount downwards to be used for the vessel cooling system. The figure also shows the honeycomb shape flow restrictors in a more detailed manner. Such part is built up by having layers of flow restrictors welded together, which gives the possibility to change the orientation of the restrictor to adjust the pressure loss in the specific assembly.

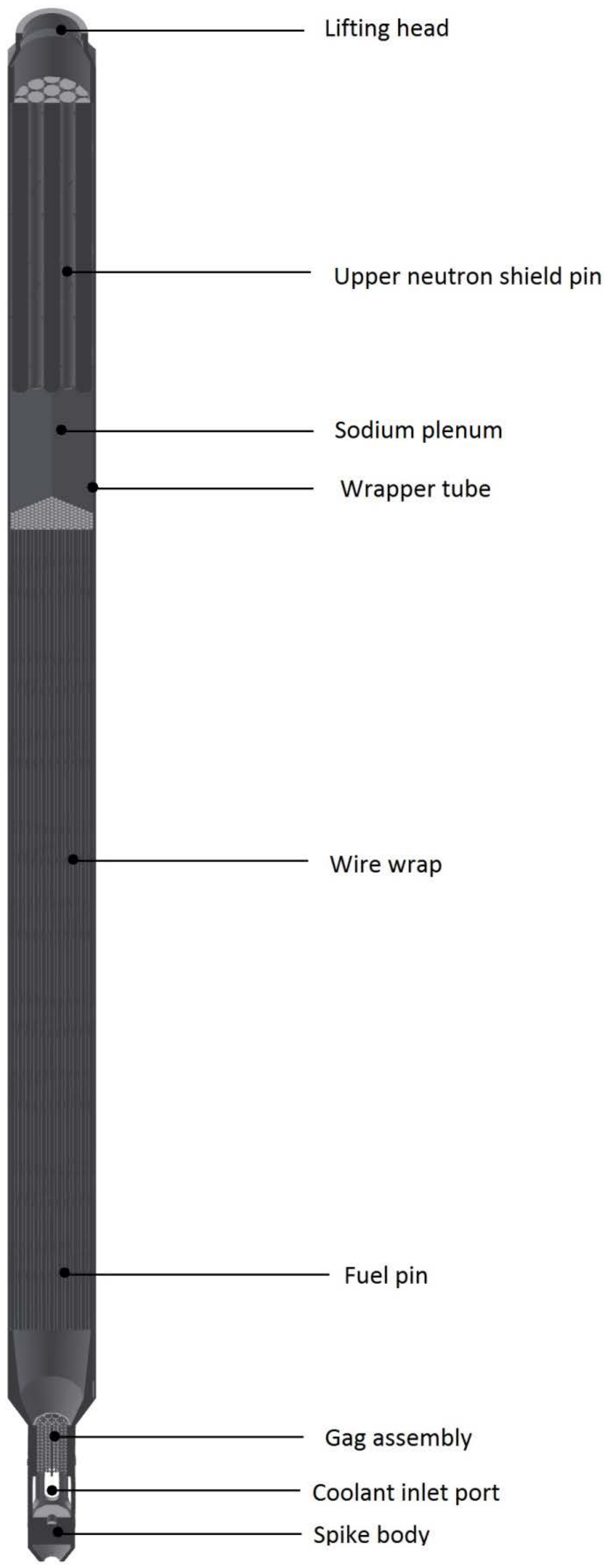

Figure 2.6 Fuel assembly

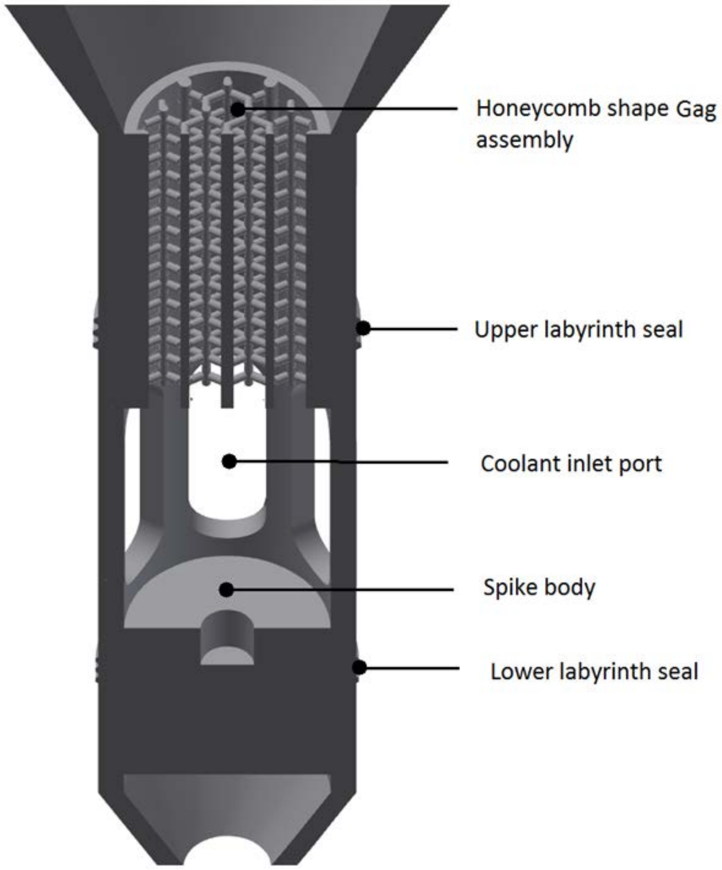

Figure 2.7 Fuel assembly bottom

2.5. Intermediate Heat Exchanger

In Figure 2.8, the intermediate heat exchanger is presented. The intermediate heat exchanger is the device which provides the connection between the primary sodium circuit and the secondary sodium loop. The secondary sodium, which comes from the steam generators, flows into the intermediate heat exchanger at the top, as it is shown by the blue arrow, then flows down inside the main tube at the middle.

From the bottom of the device, the sodium flows into the tube bundle. This tube bundle is surrounded by the primary sodium which exchanges its heat with the secondary sodium through the tubes. The hot primary sodium enters into the heat exchanger at around the middle of the device and, as it passes by the tube bundles, it gradually cools down and leaves the system at the bottom of the device. During this process, the secondary sodium heats up while it is flowing upwards in the tubes. After it comes out from the tube bundle, it goes through a mixing plate to average out the temperature of the coolant. In the end, the hot secondary sodium leaves the intermediate heat exchanger and goes towards the steam generators.

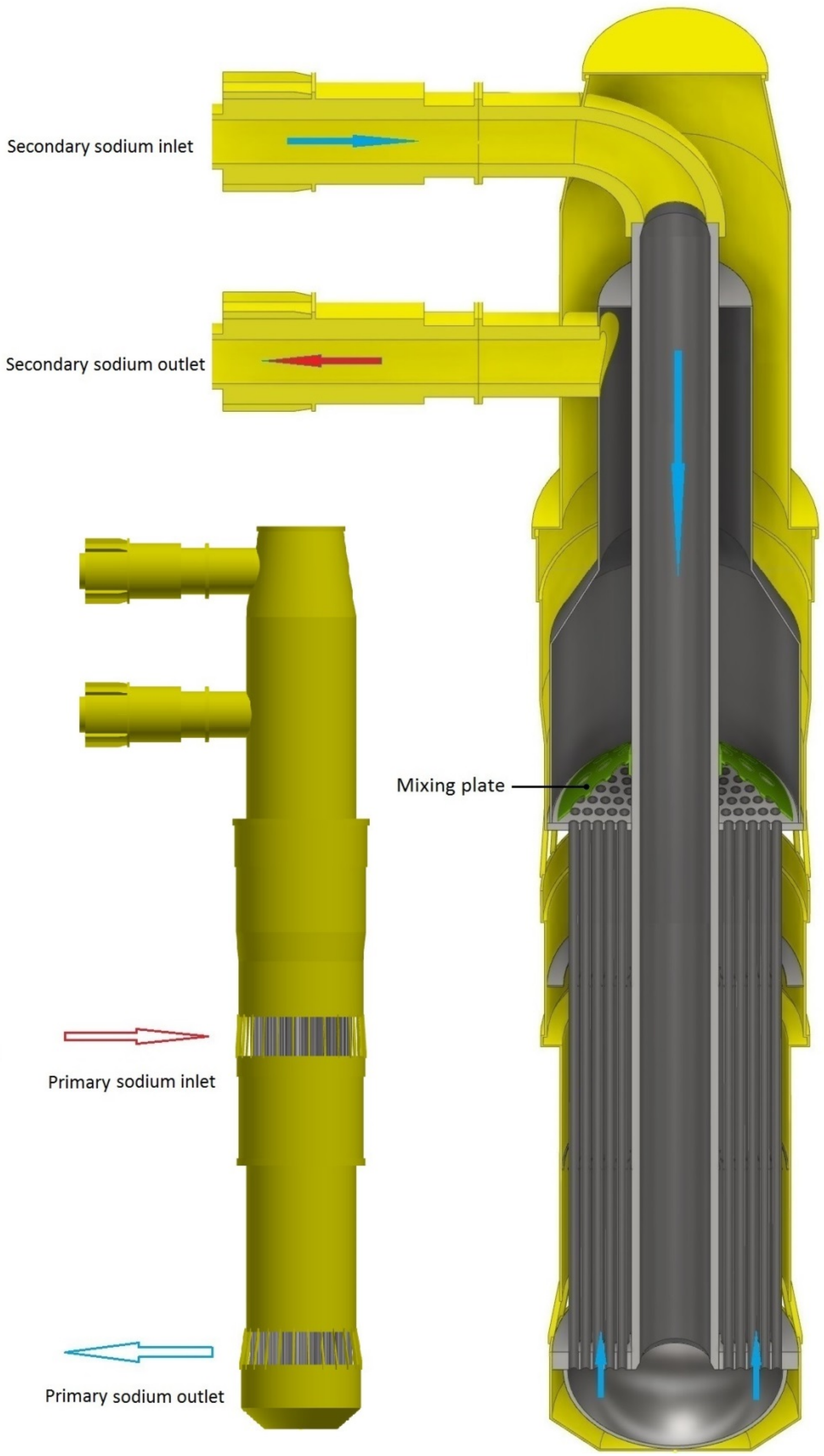

Figure 2.8 Intermediate heat exchanger

2.6. Strongback

The strongback of the reactor is shown in Figure 2.9. The main purpose of the strongback is to hold the weight of the diagrid, the reactor core and the inner vessel. With its bottom outer periphery, the strongback sits on the bottom of the reactor vessel. It is made of circular shape metal sheets connected with horizontal cylinders.

At the centre of the strongback, an opening is left so that, in the case of a severe accident, it can provide the main path for the melted core towards the core catcher. Extra holes are provided for sodium coolant flowing towards the vessel cooling system. Around these holes, another set of smaller diameter holes is present as well, providing a way for the corium discharge tubes, which are part of the diagrid. Moreover, a final set of 18 holes can be seen which are the biggest in diameter and closer to the periphery. The diagrid and the strongback are connected through those holes and flexible sleeves to provide the coolant for the core of the reactor.

At the side of the strongback, 3 openings can be seen. They are there to provide a space for the primary pumps to be connected to the strongback.

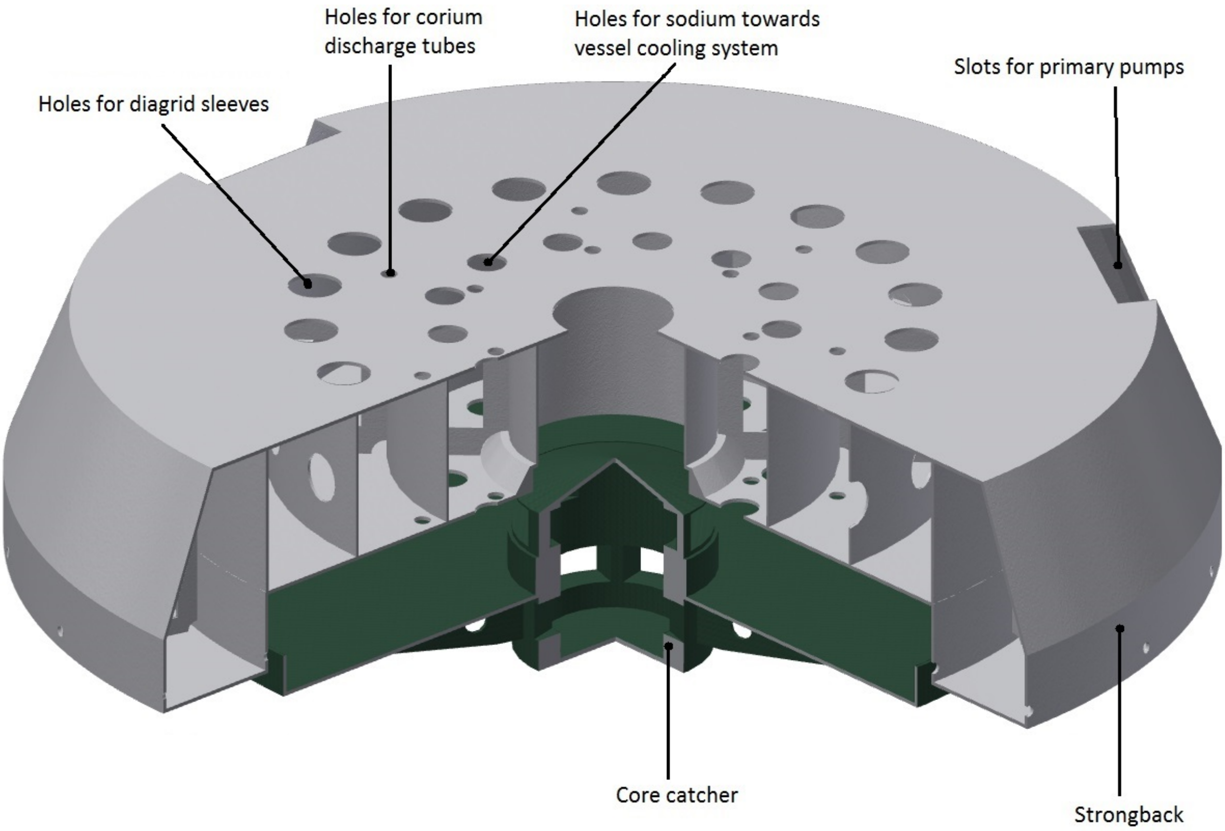

Figure 2.9 Strongback and core catcher

2.7. Diagrid

The following part is the diagrid which is shown in Figure 2.10. The diagrid sits on the top of the strongback. The connecting points between the two parts are the periphery of the diagrid and a disk shape flange from the corium guide tubes. The weight of the core is conveyed through these resting points toward the strongback.

There are 18 flexible sleeves connected to the diagrid. Through these sleeves, the sodium coolant is flowing from the strongback to the diagrid. As the coolant gets into the diagrid, it flows into the main area where the shroud tubes are situated. The shroud tubes provide a support for the fuel assemblies to keep them in position and to provide a path for the excess sodium to flow downwards, towards the reactor vessel coolant tubes.

Other parts of the diagrid are the corium discharge tubes. These discharge tubes are restricting the sodium to not to flow through them with the means of a thin metal disk at the bottom, but, in the case of a severe accident, these disks melt easily and let the corium to pass by.

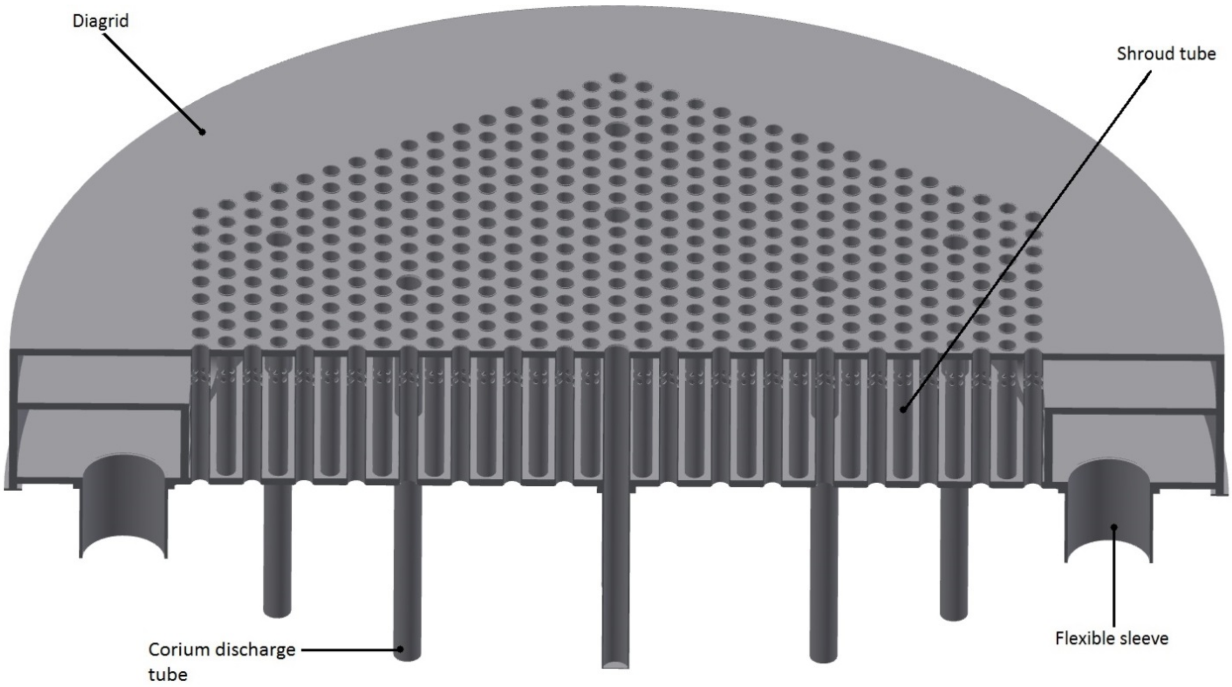

Figure 2.10 Diagrid

2.8. Core catcher

The core catcher, as it can be seen in Figure 2.11, is a device made for the most severe nuclear reactor accident, when the produced heat can no longer be extracted sufficiently from the reactor core and, as a result, core meltdown occurs.

The device is situated under the strongback at the bottom of the reactor since this is the last safety system to protect the main vessel from the melted corium. Its shape is designed to store the corium with the highest surface area possible so that it would not go critical and it would slowly cool down by the primary sodium.

A conical shape is located in the centre of the core catcher for a very crucial reason. Since the highest amount of corium would come from the middle of the reactor core, this conical shape is meant to catch the coming corium at the centre to distribute it as evenly as possible on its bottom circular surface. If such distribution would not occur, the corium would accumulate at the middle, meaning a risk of criticality. Moreover, the very core catcher could melt through.

In addition, at the middle section of the catcher, there are certain openings which facilitate the natural circulation, which cools down the present corium on the core catcher. Moreover, to help the natural circulation to establish, the outermost periphery of the device is not in contact with the bottom part of the strongback but there is rather a small gap.

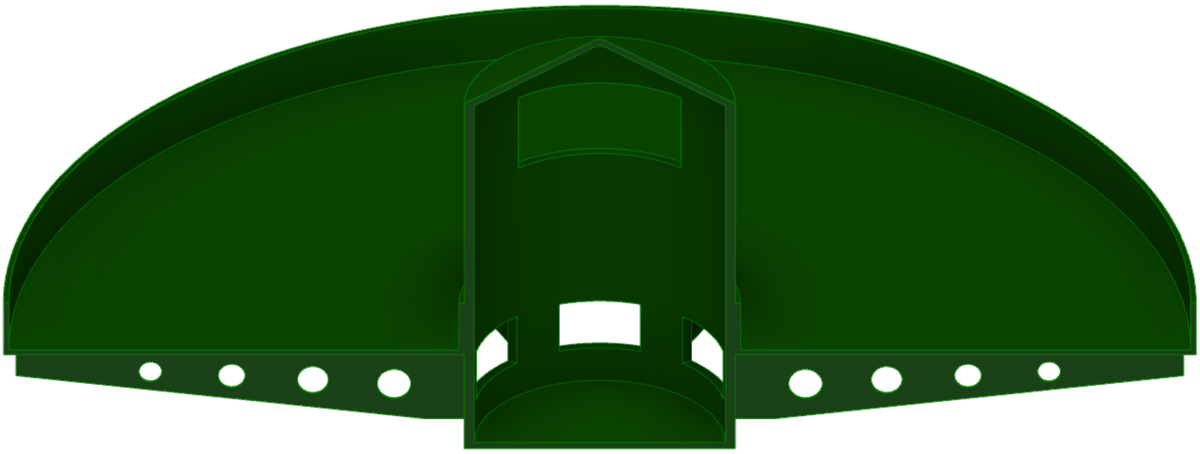

Figure 2.11 Core catcher

2.9. Primary pump

The following unit is the primary pump, shown in Figure 2.12. There are 3 primary pumps connected to the strongback, providing the coolant for the reactor core.

The pump hangs into the cold sodium pool from where it sucks the sodium out of the pool and pushes it into the strongback. For such suction, it starts with a driving motor at the top of the pump outside of the vessel. This driving motor is connected to the main shaft which runs through the pump body and ends in an impeller. The whole shaft is supported close to the bottom by a hydrostatic bearing to reduce the resonance created by the rotating motion of the shaft and impeller.

The sodium inlet on the pump is just above the impeller, where the primary cold sodium is sucked in from. The sodium from the bottom sphere of the pump is directed into the strongback through one single horizontal tube.

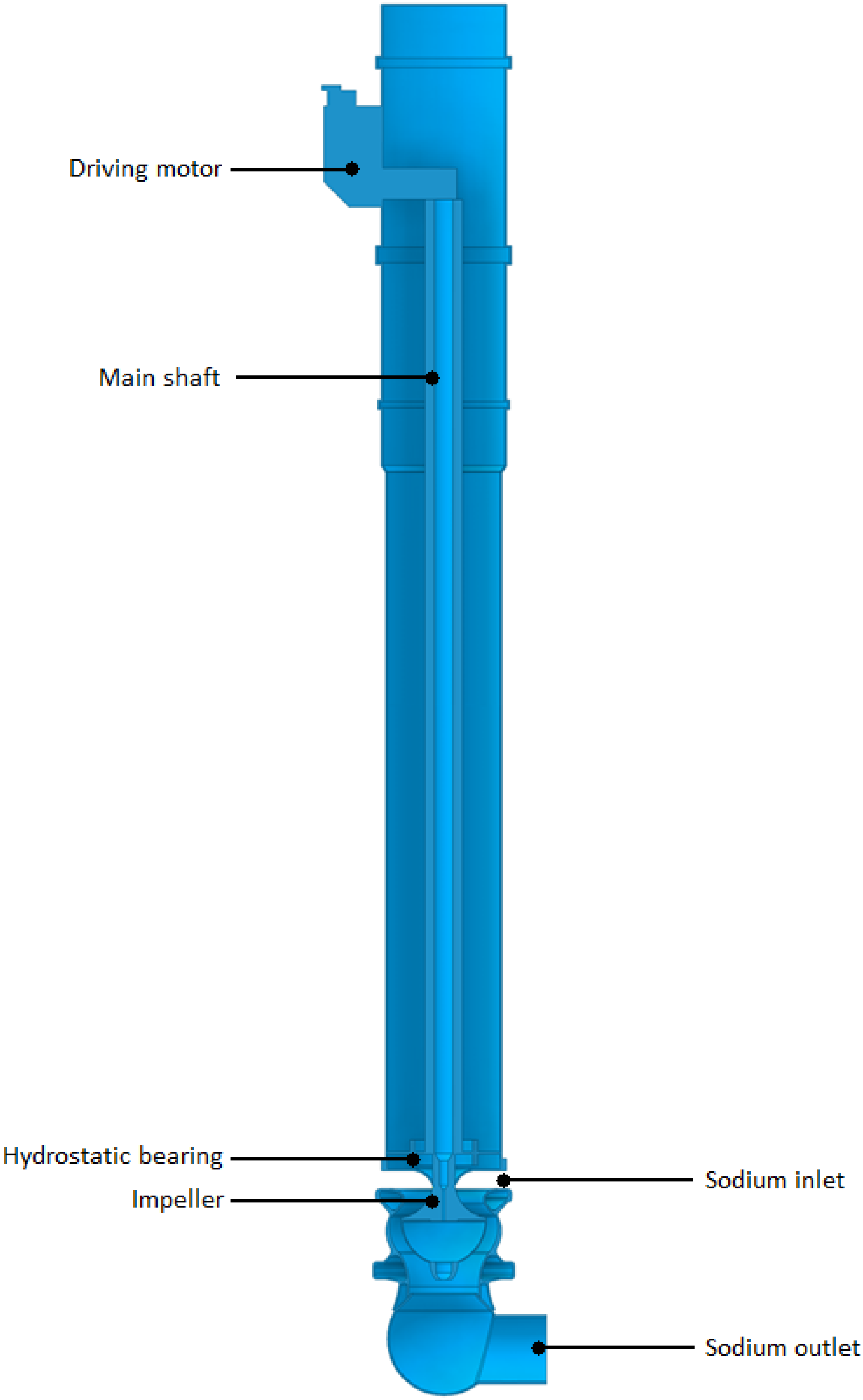

Figure 2.12 Primary pump

2.10. Flow path

To show the path of the sodium coolant from the primary pump through the strongback and the diagrid, Figure 2.13 is provided. As it is shown in the picture, the primary pump is connected to the second ring of the strongback. Between the second and third rings, holes are cut on the wall between the rings where the coolant finds its way through.

Then, from the strongback, the coolant flows upwards into the diagrid through the 18 flexible sleeves connecting the two devices together. As the flow gets into the diagrid, it flows into the shroud tubes. From the shroud tubes, the coolant is distributed into 3 different directions. Most of the coolant flows into the fuel assembly upwards towards the core and the hot pool. To continue with, a smaller portion of the sodium flows downwards between the shroud tube and the lower labyrinth seal on the fuel assembly bottom. To finish with, the smallest amount of sodium flows upwards again between the shroud tubes and the upper labyrinth on the fuel assembly bottom and, from there, it goes up through the inter-assembly gap between the hexagonal wrappers. The sodium that flows down is used for cooling as well. It flows through the vessel cooling tubes and cools down the top of the reactor vessel.

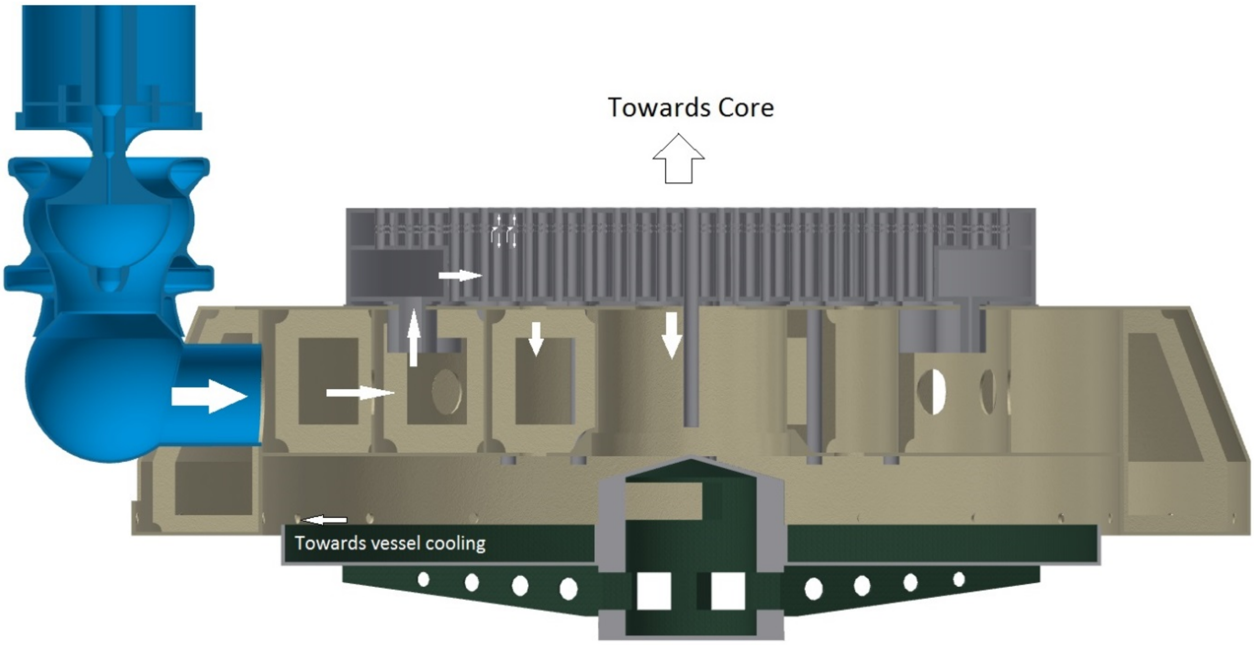

Figure 2.13 Sodium path from primary pump through Strongback and Diagrid

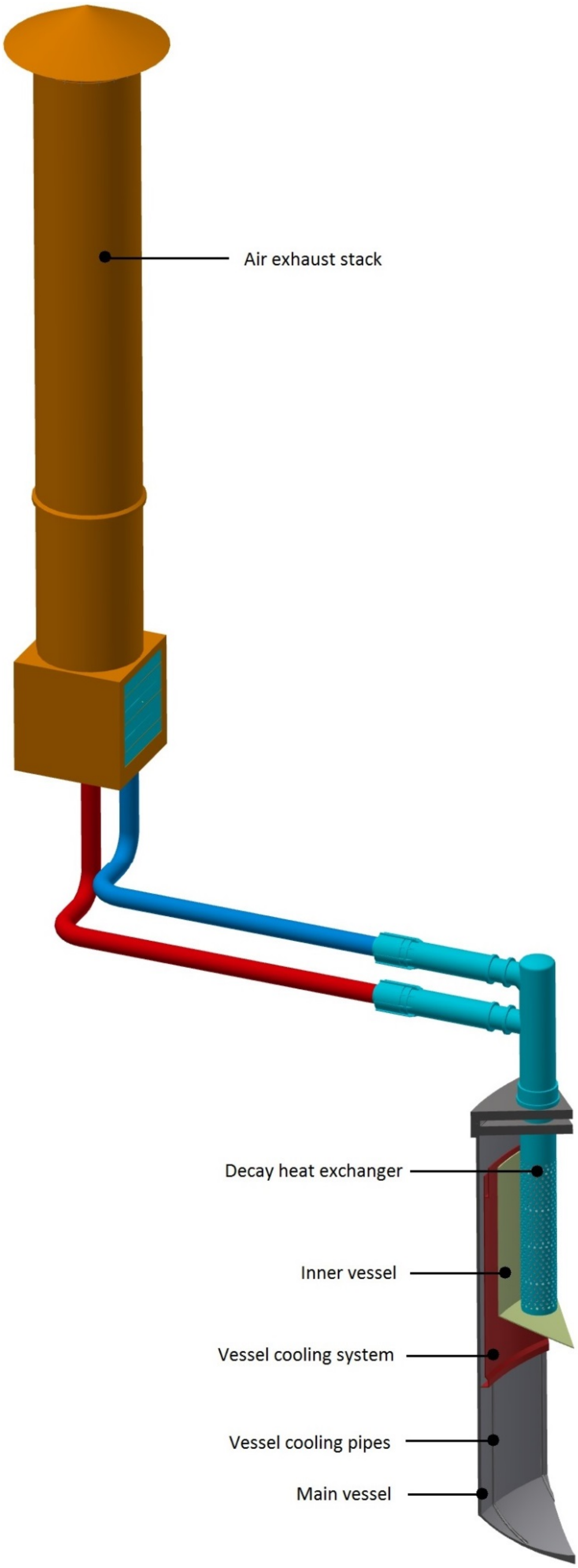

Figure 2.14 Decay heat removal system

2.11. Decay heat removal loop

In Figure 2.14, the decay heat removal system can be seen. The system consists of a decay heat exchanger, an air exhaust stack and the connecting tubes between the two.

The circulation is based on natural circulation for the loop so no pumps are used to circulate the flow. As the figure shows, the decay heat exchanger is immersed in the primary hot sodium pool, having sodium always in its heat exchanging region. The blue tube represents the path for the cold sodium, whereas the red represents the path for the hot sodium. The air exhaust stack is outside of the reactor building and high enough so that the higher velocity wind naturally sucks out the air from the stack.

As the air is continuously sucked out from the stack, there is an air damper close to the bottom of the stack which is controlled by electro-pneumatical means to regulate the cooling and the circulation.

2.12. Decay heat exchanger

The decay heat exchanger is presented in Figure 2.15. The primary hot sodium enters such device through perforations at its skirt. At this perforated region, the heat exchange inside the device happens. As the hot primary sodium passes by the tube bundles, which contain the secondary sodium that goes from the circuit towards the air exhaust stack, the primary sodium cools down and leaves the decay heat exchanger through the holes at its skirt.

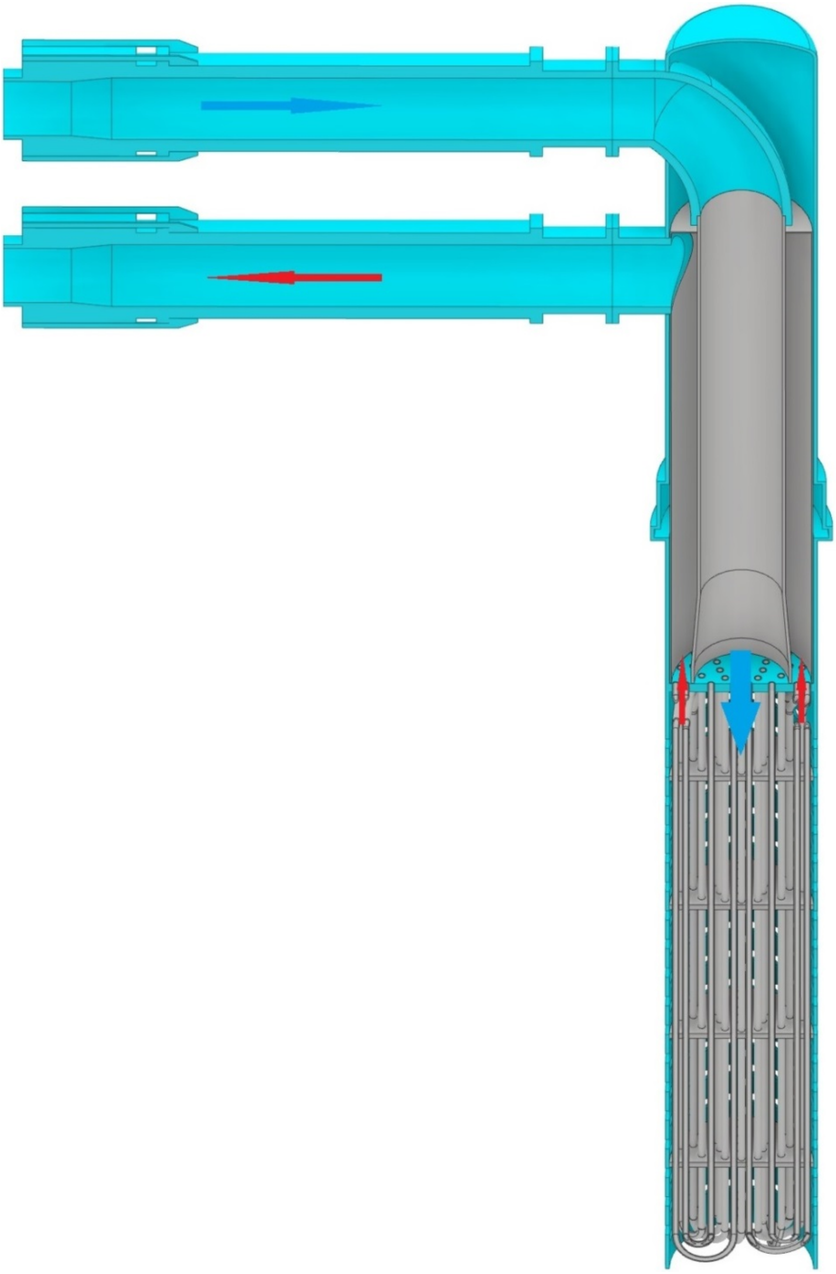

Figure 2.15 Decay heat exchanger

The secondary sodium enters the heat exchanger through the main middle tube (blue arrows). Following that, it enters the tube bundle through a divider plate. As the coolant flows downwards, its temperature gradually increases. After it has passed through the U shape bottom part of the tube, the increased temperature sodium flows upwards (red arrows) at the periphery of the heat exchanger. Before the heated up secondary sodium would come out from the tube bundles, there is a bend at the top of the tube to accommodate the thermal expansion in the system.

After the sodium leaves the bundle, it leaves the system towards the air exhaust stack to cool down the secondary sodium. This circulation is maintained by natural means.

2.13. Decay heat removal stack

In Figure 2.16, the air exhaust stack can be seen. From the decay heat exchanger, the hot sodium enters the sodium/air heat exchanger inside the stack. This sodium inlet is at the top of the heat exchanger and at the bottom of the expansion vessel. This helps the natural circulation since the cooled sodium, which is heavier, would naturally flow downward.

After the sodium enters the heat exchanger, it flows downwards inside the spiraled tubes which connect the bottom and the top of the heat exchanger. While the sodium travels in these tubes, the circulating air cools down the coolant medium inside the tubes. Following this, the sodium flows back from the bottom of the heat exchanger towards the decay heat exchanger, which is inside the reactor vessel.

To adjust the performance of the decay heat exchanger loop, an air damper is present to control the air flow. In normal operating conditions, the air damper is closed completely to minimize the circulation of air coolant in the system, reducing the cooling performance of the decay heat loop. When the decay heat removal system is needed, the air dampers are opened fully to allow the air to flow through. This mechanism is controlled by electro-pneumatical valves connected to the air dampers.

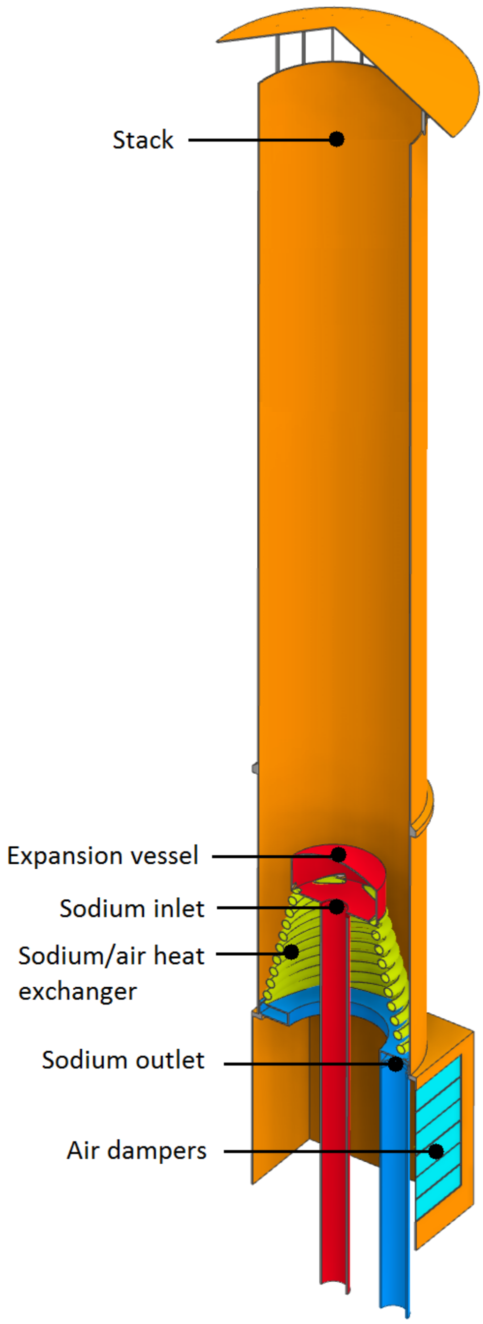

Figure 2.16 Decay heat removal stack

2.14. Secondary sodium loop

The secondary sodium loop is shown in Figure 2.17. It consists of the intermediate heat exchanger, the secondary pump, a dump vessel, a cyclone separator, 6 steam generators and the connecting tubes.

The main purpose of the secondary sodium loop is to extract the heat from the primary sodium circuit and convey it to the steam generators. As the sodium in this loop does not have a direct connection with the primary sodium, it does not contain any radioactive material so that any leak in the steam generator could not cause a fire with radioactive material, reducing greatly the probability of a nuclear accident.

The hot secondary sodium from the intermediate heat exchanger flows towards the top of the steam generators. After the sodium flowed through the steam generators and exchanged its heat with the water and steam, the cold sodium comes out at the bottom of the steam generators. The circulation is maintained by the secondary pump which is allocated at the cold leg of the system.

In addition, a dump vessel is present in the system with the capacity to hold a high amount of sodium. This capacity is crucial since, in the case of an accident and overpressure present in the system, the secondary sodium can flow into the dump vessel through a rapture disk. The dump vessel is connected to a cyclone separator which is used to purify the sodium.

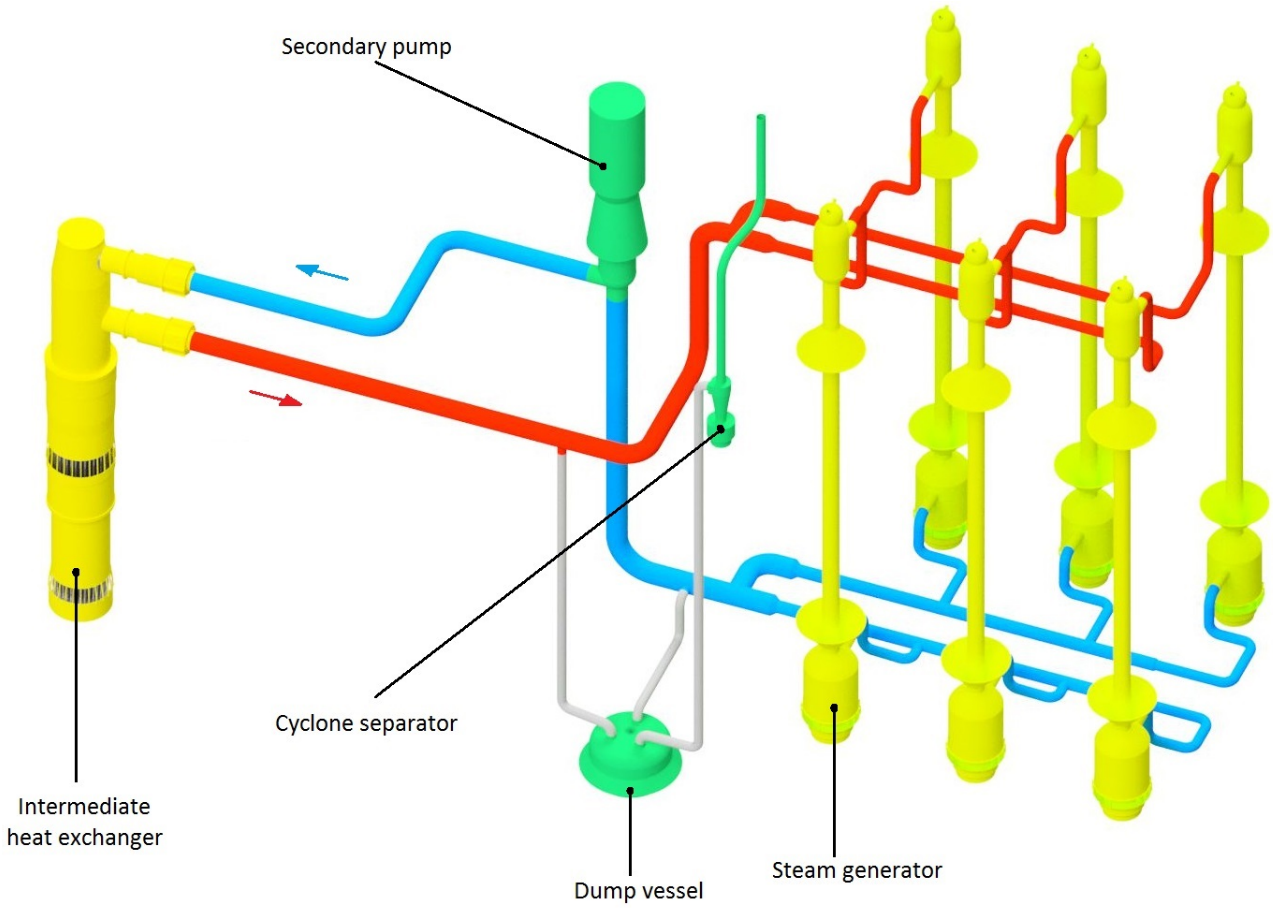

Figure 2.17 Secondary sodium loop

2.15. Steam generator

The steam generator is plotted in Figure 2.18. For each of the secondary sodium loops, 6 steam generators are corresponding to create steam to run the turbines.

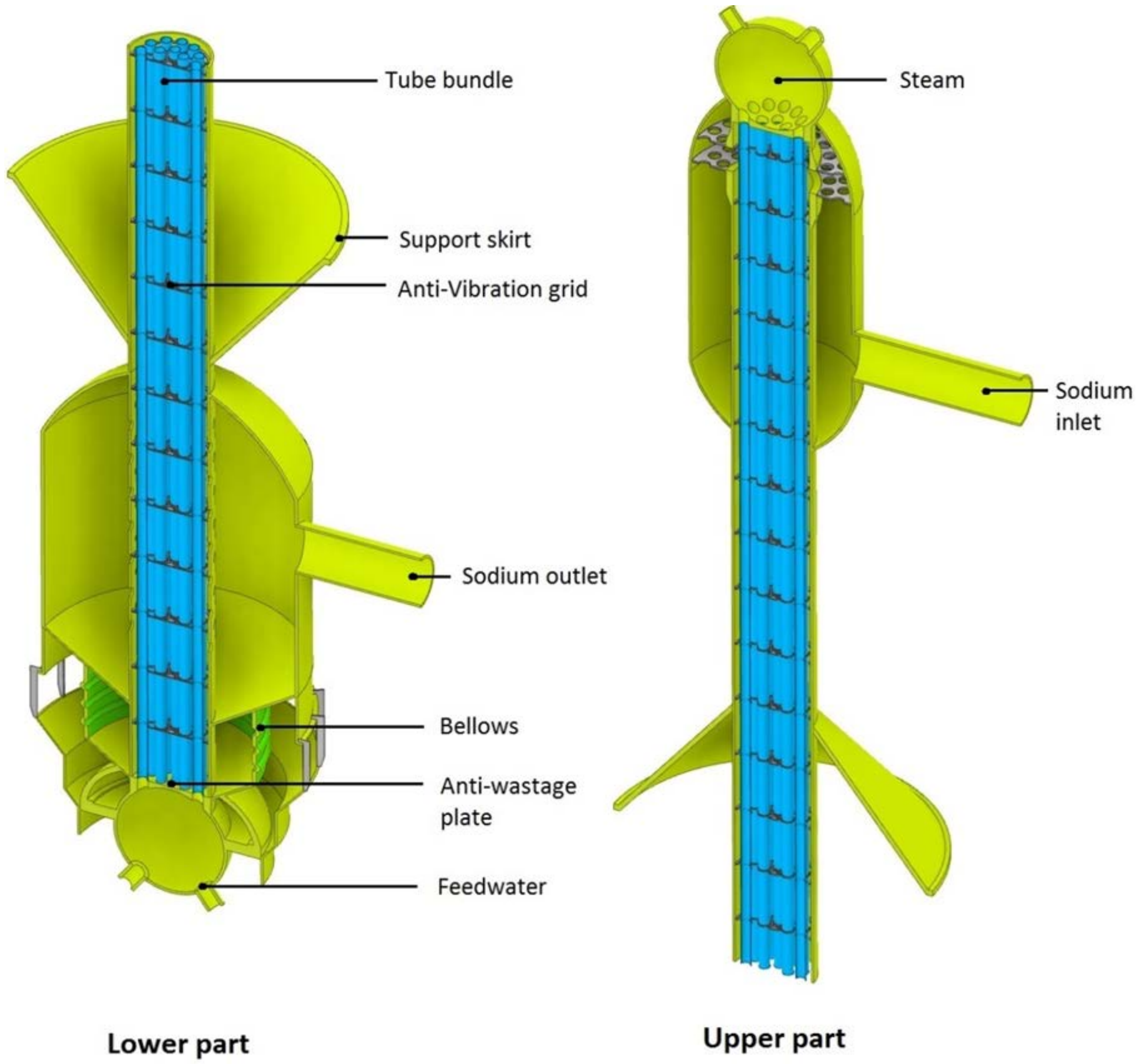

Figure 2.18 Steam generator

On one hand, the secondary hot sodium enters the device at the upper part and, through an opening on the main tube, it flows downwards, surrounding the tube bundle. On the other hand, water enters the device at the bottom of the steam generator and flows upwards inside the tube bundle. As the sodium flows downwards, it exchanges its heat with the water inside the tube bundle. During this heat exchange, the water turns to steam and leaves the steam generator at the top, cooling down the secondary sodium. On the lower part of the steam generator, the main tube is perforated so that the cold sodium can leave the device and flow back toward the intermediate heat exchanger.

At the bottom part of the steam generator, there is a so called “bellows” which allows the system to compensate for thermal expansion since the straight tubes would not accommodate this problem otherwise. The bellows allow some expansion and contraction by having its sides concertinaed.

2.16. Small rotating plug with “Above core structure”

The final piece presented here is the “Above core structure” with the small rotating plug shown in Figure 2.19. This part is responsible to accommodate the control rods, the control rod driving mechanisms and some instrumentation to monitor the reactor.

In the “Above core structure”, part of the sodium coolant arising from the core flows through the porous plate. Since there is a baffle plate situated after the porous plate restricting the flow towards the middle region of the structure, only a small amount of sodium continues its path upwards and the rest of the sodium comes out from the side of the “Above core structure” shell.

To continue with, some of the flow from the core goes up through the shroud tubes and surrounds the control rods inside the tubes. This shroud tubes provide a guide for the control rods as they move along to control the nuclear reactions. The sodium from the shroud tubes then leaves through the perforations on the side of the tubes and flows into the middle region of the “Above core structure”.

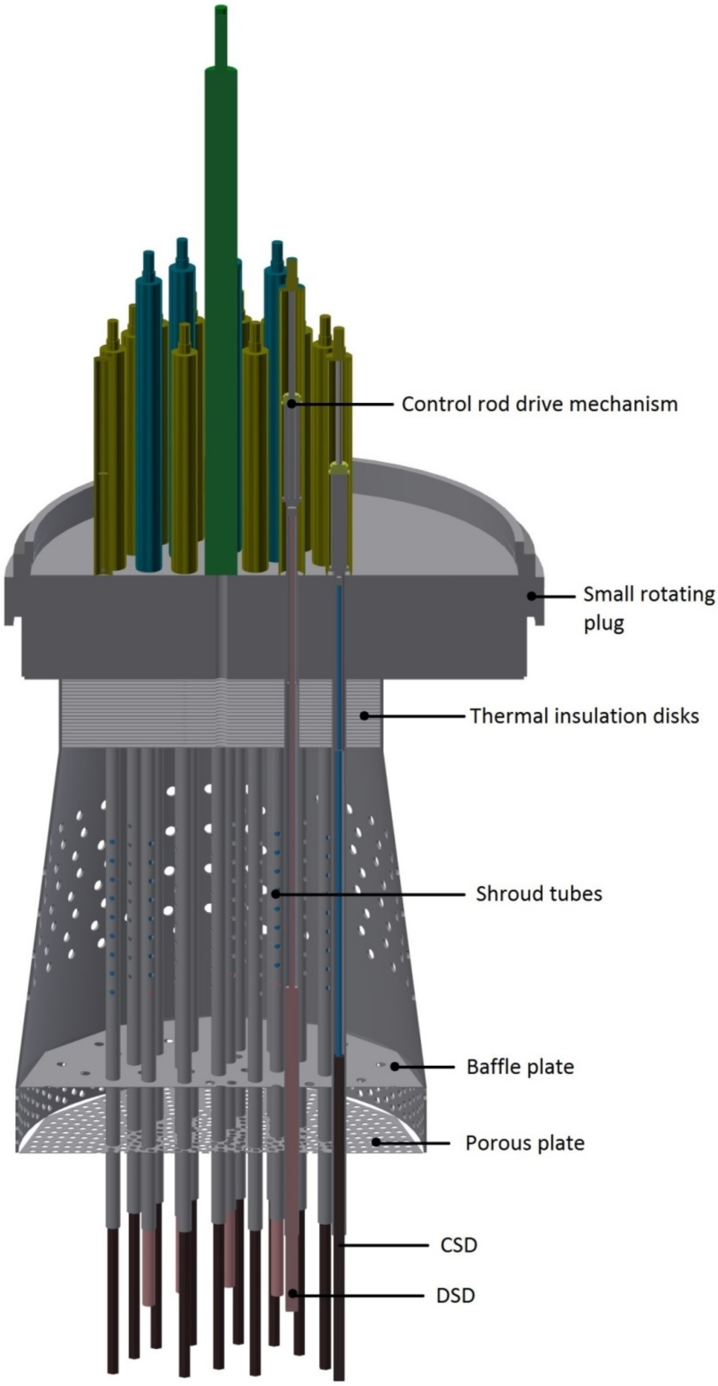

Figure 2.19 Above core structure and small rotating plug

3. TRACE model

In order to obtain the right input data from the CAD model, information about physical properties are required for each cell to obtain an accurate simulation. For this purpose, it is important to know how the TRACE model is divided into components and how the components are nodalized into smaller cells.

In Figure 3.1, the nodalized diagram of the reactor vessel concept can be seen. The nodalization was based on a previous work which was done on ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) [10]. Such work is built on a very similar fashion as the ESFR reactor concept, allowing to use a similar TRACE model for the current purposes.

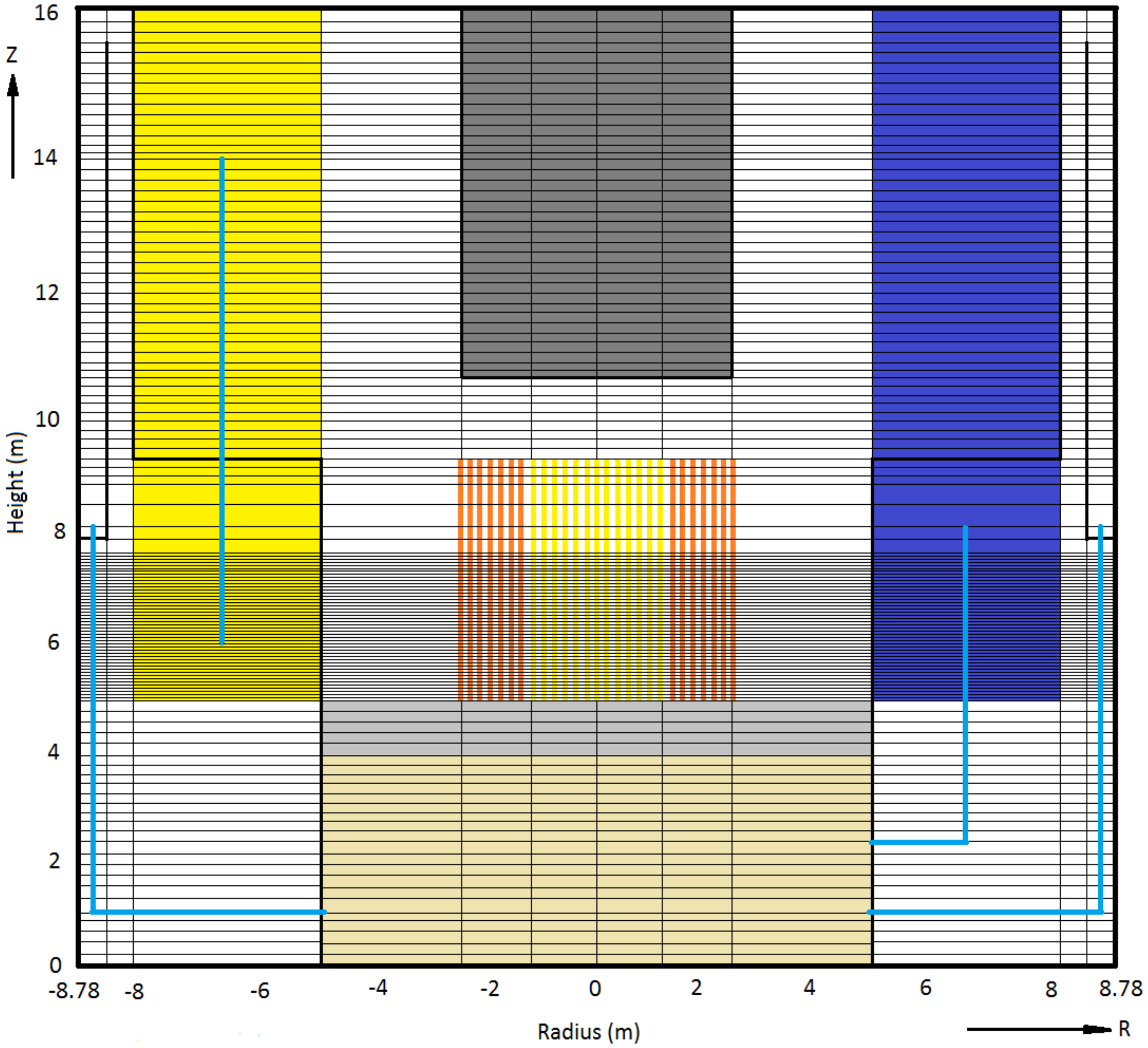

Figure 3.1 TRACE model cells, radial cross section

The figure shows the vessel component of the model which allows a 3D cylindrical geometry (radial, axial and azimuthal directions) to model the primary system pool of the reactor. As it can be seen, on a radial direction, the model is divided up into 6 different sections, whereas, on the axial direction, there are 120 divisions. In Table 3.1, a detailed description of the cells of the model is provided.

<table>
<tr><th colspan="5">Axial nodalization</th></tr>
<tr><th>Z</th><th>Lower surface [m]</th><th>Upper surface [m]</th><th>Cell height [m]</th><th>Cell outer surface specificities</th></tr>
<tr><td>1 to 25</td><td>0.0</td><td>5.0</td><td>0.2</td><td>Z=20 : Strongback top/Diagrid bottom
Z=25 : Diagrid top/Core inlet
Z=15 : Primary pump inlet
Core inner FA fuel zone bottom</td></tr>
<tr><td>26 to 77</td><td>5</td><td>7.6</td><td>0.05</td><td>Z=53 : Intermediate heat exchanger outlet/Core inner FA fuel zone top</td></tr>
<tr><td>78</td><td>7.6</td><td>8.0</td><td>0.4</td><td>Core inner FA fuel sodium plenum top</td></tr>
<tr><td>79 to 80</td><td>8.0</td><td>8.85</td><td>0.425</td><td>Z=79 :Vessel cooling pipes outlet/Core inner FA neutron shield top</td></tr>
<tr><td>81 to 82</td><td>8.85</td><td>9.4</td><td>0.275</td><td>Core inner FA head top</td></tr>
<tr><td>83</td><td>9.4</td><td>9.6</td><td>0.2</td><td>REDAN surface</td></tr>
<tr><td>83 to 120</td><td>9.6</td><td>16.0</td><td>0.2</td><td>Z=100: Intermediate heat exchanger inlet
Z=118 : Vessel cooling system opening
Z=120 : Pool top</td></tr>
<tr><th colspan="5">Radial nodalization</th></tr>
<tr><th>R</th><th>Inner surface [m]</th><th>Outer surface [m]</th><th colspan="2">Cell outer surface specificities</th></tr>
<tr><td>1</td><td>0</td><td>1.4</td><td colspan="2">Inner core outer surface/Outer core inner surface</td></tr>
<tr><td>2</td><td>1.4</td><td>2.5</td><td colspan="2">Outer core outer surface/reflector region inner surface</td></tr>
<tr><td>3</td><td>2.5</td><td>3.29</td><td colspan="2">Primary pump outlet/Vessel cooling pipes inlet</td></tr>
<tr><td>4</td><td>3.29</td><td>4.25</td><td colspan="2">Core outer surface/REDAN inner surface</td></tr>
<tr><td>5</td><td>4.25</td><td>8.11</td><td colspan="2">REDAN outer surface/PP inner surface/IHX and DHX inner surface</td></tr>
<tr><td>6</td><td>8.11</td><td>8.53</td><td colspan="2">PP outer surface/IHX and DHX outer surface/Vessel cooling system walls inner surface</td></tr>
<tr><td>7</td><td>8.53</td><td>8.78</td><td colspan="2">Vessel cooling system walls outer surface/Pool inner surface</td></tr>
<tr><th colspan="5">Azimuthal nodalization</th></tr>
<tr><th>Θ</th><th>Inner surface [rad]</th><th>Outer surface [rad]</th><th colspan="2">Cell specificities (the azimuthal cell was modeled for R=5 and Z>20)</th></tr>
<tr><td>1</td><td>0</td><td>$\pi/6$</td><td colspan="2">Free sodium</td></tr>
<tr><td>2</td><td>$\pi/6$</td><td>$\pi/3$</td><td colspan="2">Intermediate heat exchnager #1</td></tr>
<tr><td>3</td><td>$\pi/3$</td><td>$\pi/2$</td><td colspan="2">Primary pump #1 + Decay heat removals #1</td></tr>
<tr><td>4</td><td>$\pi/2$</td><td>$2\pi/3$</td><td colspan="2">Intermediate heat exchnager #2</td></tr>
<tr><td>5</td><td>$2\pi/3$</td><td>$5\pi/6$</td><td colspan="2">Free sodium</td></tr>
<tr><td>6</td><td>$\pi/6$</td><td>$\pi/6$</td><td colspan="2">Intermediate heat exchnager #3</td></tr>
<tr><td>7</td><td>$5\pi/6$</td><td>π</td><td colspan="2">Primary pump #2 + Decay heat removals #2</td></tr>
<tr><td>8</td><td>π</td><td>$7\pi/6$</td><td colspan="2">Intermediate heat exchnager #4</td></tr>
<tr><td>9</td><td>$7\pi/6$</td><td>$4\pi/3$</td><td colspan="2">Free sodium</td></tr>
<tr><td>10</td><td>$3\pi/2$</td><td>$5\pi/3$</td><td colspan="2">Intermediate heat exchnager #5</td></tr>
<tr><td>11</td><td>$5\pi/3$</td><td>$11\pi/6$</td><td colspan="2">Primary pump #3 + Decay heat removals #3</td></tr>
<tr><td>7</td><td>$5\pi/6$</td><td>π</td><td colspan="2">Primary pump #2 + Decay heat removals #2</td></tr>
<tr><td>8</td><td>π</td><td>$7\pi/6$</td><td colspan="2">Intermediate heat exchnager #4</td></tr>
<tr><td>9</td><td>$7\pi/6$</td><td>$4\pi/3$</td><td colspan="2">Free sodium</td></tr>
<tr><td>10</td><td>$3\pi/2$</td><td>$5\pi/3$</td><td colspan="2">Intermediate heat exchnager #5</td></tr>
<tr><td>11</td><td>$5\pi/3$</td><td>$11\pi/6$</td><td colspan="2">Primary pump #3 + Decay heat removals #3</td></tr>
<tr><td>12</td><td>$11\pi/6$</td><td>2π</td><td colspan="2">Intermediate heat exchnager #6</td></tr>
</table>

Table 3.1 VESSEL component nodalisation

In this primary system model, the system components and the sodium flow paths of the system are shown in Figure 3.1. The yellow color represents the intermediate heat exchanger, the dark blue shows the primary pump of the system, the dark gray symbolizes the “Above core structure”, the beige represents the strongback and the light gray shows the diagrid. The bold black lines represent the no-flow boundary conditions. The light blue lines show the different sodium paths simulated with 1D pipes. First, the light blue line coming from the bottom to the side of the vessel represents the vessel cooling tubes. Second, the vertical light blue line at the left of the figure represents the intermediate heat exchanger. Lastly, the line coming from the primary pump towards the strongback shows the sodium path through the primary pump. The inner core fuel assemblies are represented by yellow lines, whereas the outer core breeder assemblies are shown as orange lines.

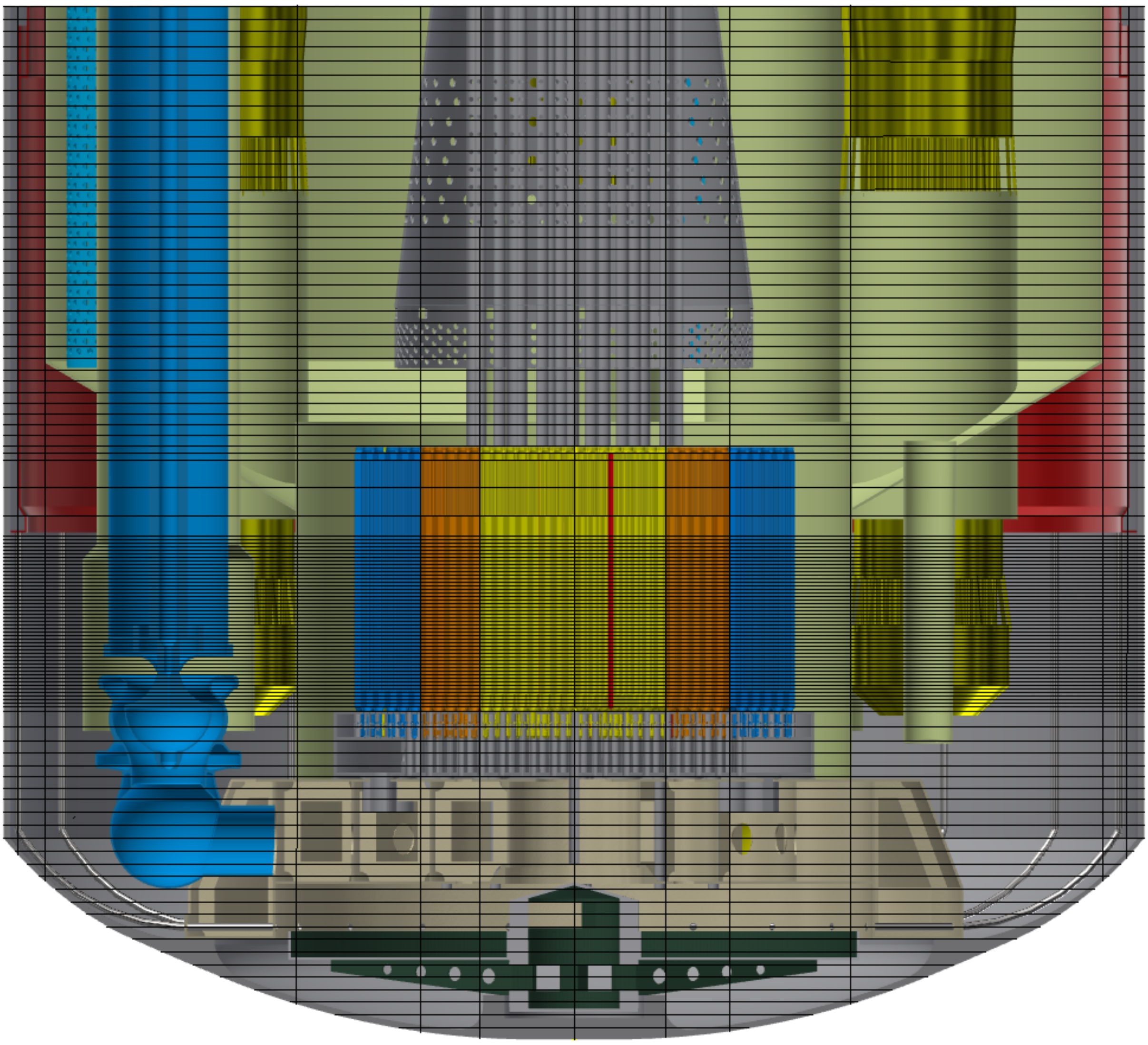

Figure 3.2 Inventor model, radial cross section with TRACE cell division

After having the TRACE model with the nodalization, the same nodalized picture was made from the CAD model as well. This is shown in Figure 3.2. As it is shown in the figure, all the cell boundaries were kept and presented here also. The axial distances of the cells depend on how much spatial variation is expected at the specific region. For example, at the bottom of the primary system around the strongback and diagrid, only smaller change is anticipated, so there are cells for every 0.2 m. In contrast, around the fuel zone where the fuel heats up the sodium, there is a much denser cell distribution, namely 0.05m. This is because it is expected that the spatial variation is higher and the coolant properties will change considerably at that region.

To obtain accurate information about properties of a specific cell such as free volume, sodium volume, porosity and heat transfer surface area, the CAD model can be divided and sliced into pieces described by the TRACE model. By doing so, a very detailed analysis can be done on the model depending on how dense the cell distribution is for TRACE.

In Figure 3.3, an example can be seen for an axial cell taken from CAD. The picture represents the top axial cell (Z=120) from the TRACE model sliced out from the whole model to get the input information based on this one cell.

At the top of the figure, the case is shown where the reactor is filled with sodium, whereas a sodium free slice is presented at the bottom. To obtain porosity information about the whole cell, the volume information from the sodium free cell must be obtained as a first step. This information is directly available from Inventor. The provided volume information only shows the volume of material of the actual cell. In the next step, the cell can be filled with sodium and the volume information can be extracted from this case as well. By having these two pieces of volume information, they can be subtracted from each other which would give the volume of the sodium present in the cell.

To calculate the porosity information, the sodium volume must be divided by the volume of the whole cell filled with sodium. This will give the ratio of the cell occupied by the coolant material.

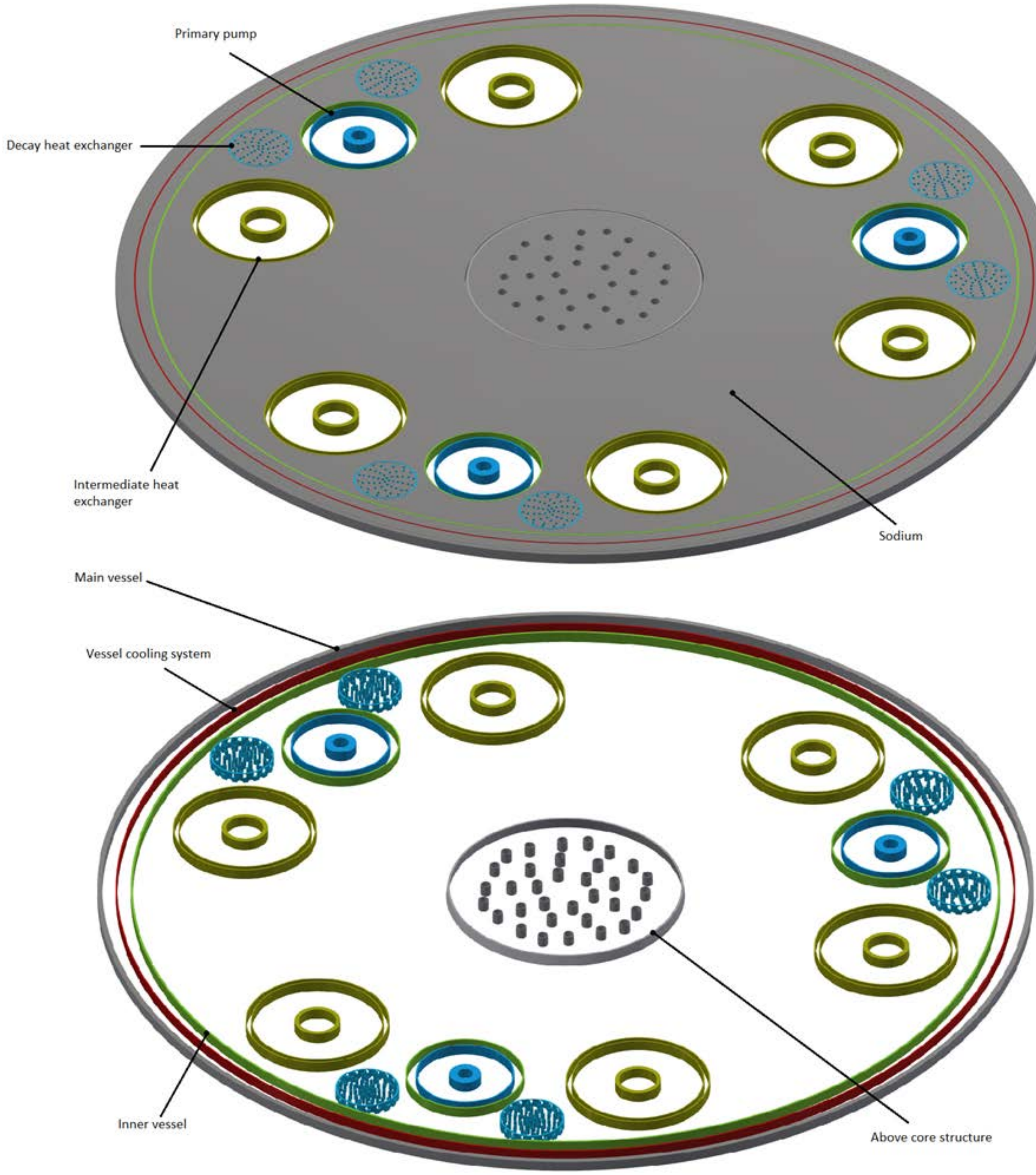

Figure 3.3 Top axial TRACE cell, obtained from CAD model

To further simplify and to obtain more detailed information about the VESSEL component, it was divided up into 12 sections in the azimuthal direction.

In Figure 3.4, these sectors are shown. The figure was taken from the axial cell Z=90. From the 12 sections, 3 sections correspond to the areas where there is only free sodium, 6 of the sections contain the intermediate heat exchangers (IHX) and 3 of the sections contain the primary pumps, which are coupled with 2 decay heat exchanger each. The description of the sections is provided in Table 3.1. The core is also divided into sectors in the same way as the rest of the vessel. The 2 outer bold black circles represent the inner vessel (inner) and the main vessel wall (outer). Between those 2 circles, there is another black circle with a thinner line simbolising the top of the reactor cooling system.

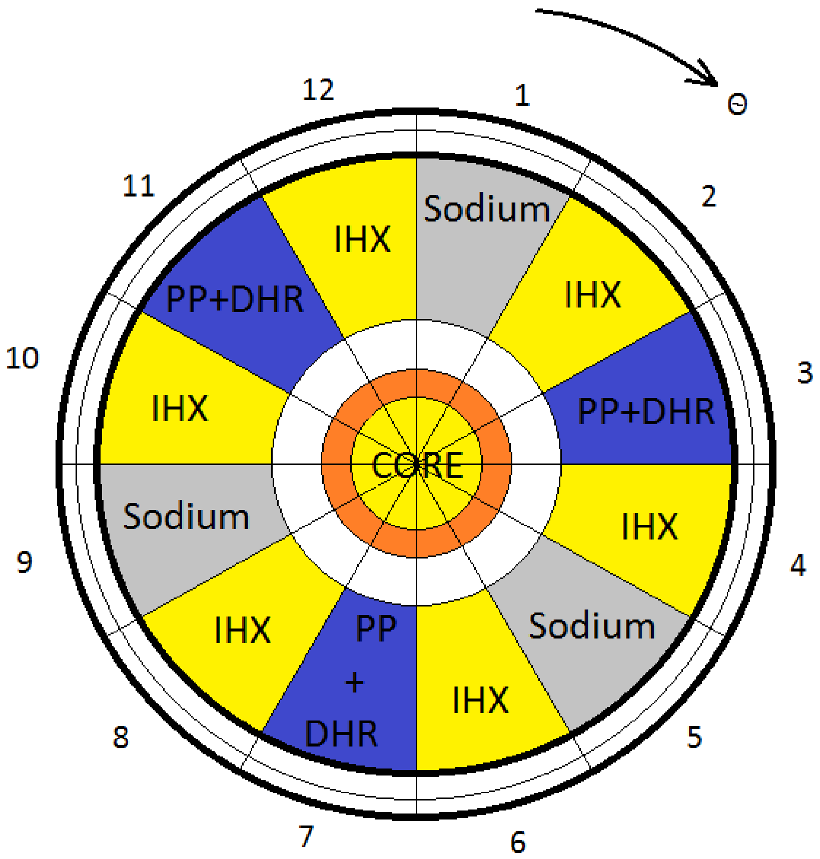

Figure 3.4 TRACE azimuthal VESSEL component division

By having divided the VESSEL into symmetrical sectors, the TRACE model is simplified. There are sectors which are repeating, therefore, it is enough to get the input information for one repeating sector cluster and copy the rest.

To show the same sector division on the CAD model, Figure 3.5 is presented. The setting of sectors in this figure is the same as in Figure 3.4 (TRACE azimuthal division). Nevertheless, there are some differences between the two figures. First, the size of the sectors on the CAD model are different from the ones on TRACE. Furthermore, the size of the sector on the CAD model differ from each other depending on which elements it contains. The widest sectors are the ones with sodium and the ones with the primary pump together with the decay heat exchangers. Second, on the CAD model, it is possible to have a more detailed visualization of the elements than on TRACE. For example, on the CAD model, it is possible to visualize a circle (In reactor fuel handling station) in one of the sodium sectors (sector 5), which differs from the other two sodium sectors, having to be built separately.

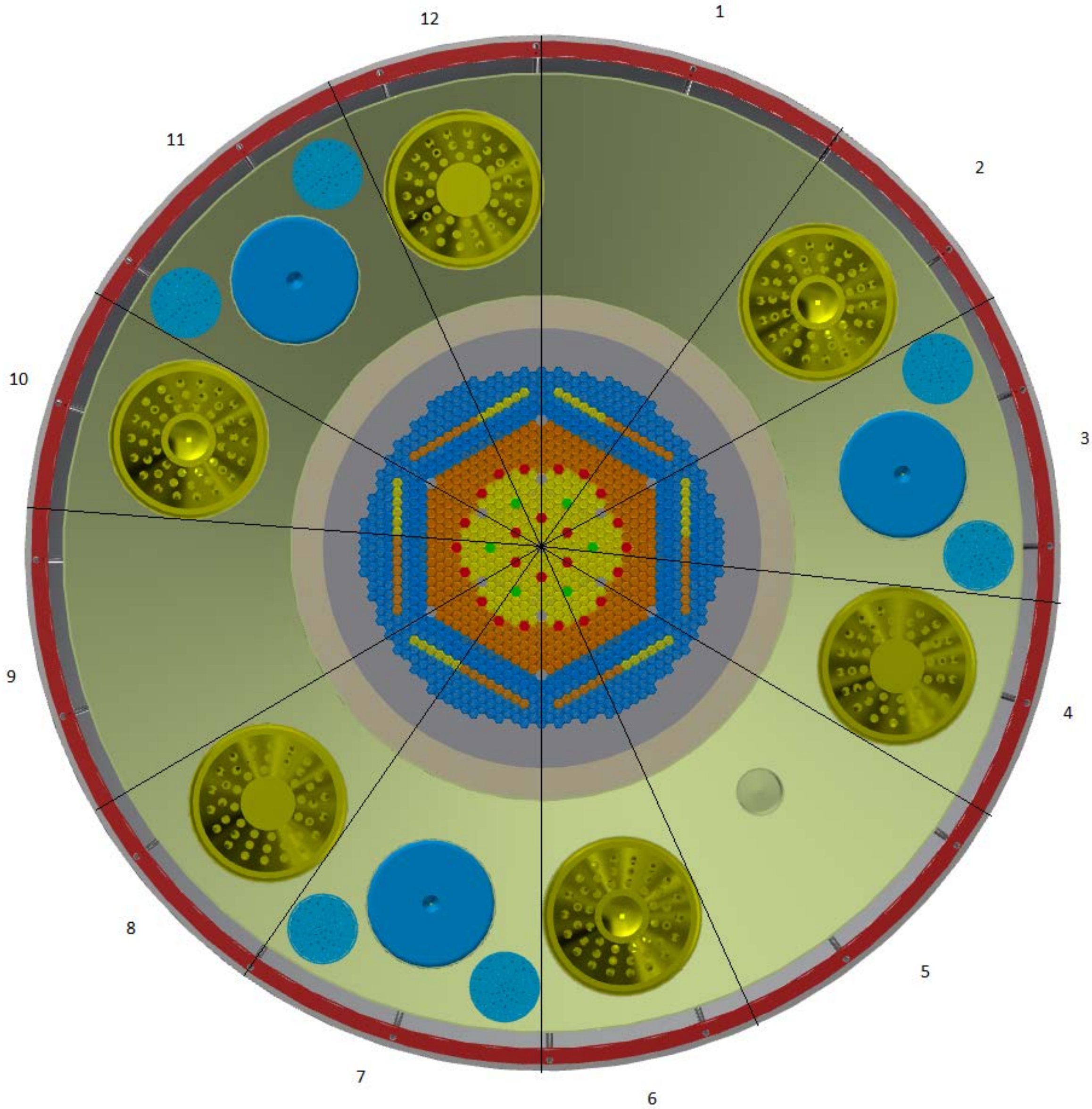

Figure 3.5 Azimuthal sectors on CAD model

By having a 3D CAD model of the reactor, it can be seen very prominently what kind of simplifications can be done on the model to speed up the input file creation process and shorten the time to run the model. On such a model, the visualization of smaller details made possible reveal some differences in the symmetry of the reactor.

In Figure 3.6, the same cell is shown with sodium (on the left) and without sodium (on the right) as it was shown in Figure 3.3. The difference between the two figures is that Figure 3.6 shows a particular sector out of the whole cell, allowing to obtain even more detailed input information for TRACE based on this small element. The method to calculate and obtain the required data is the same as it was described previously.

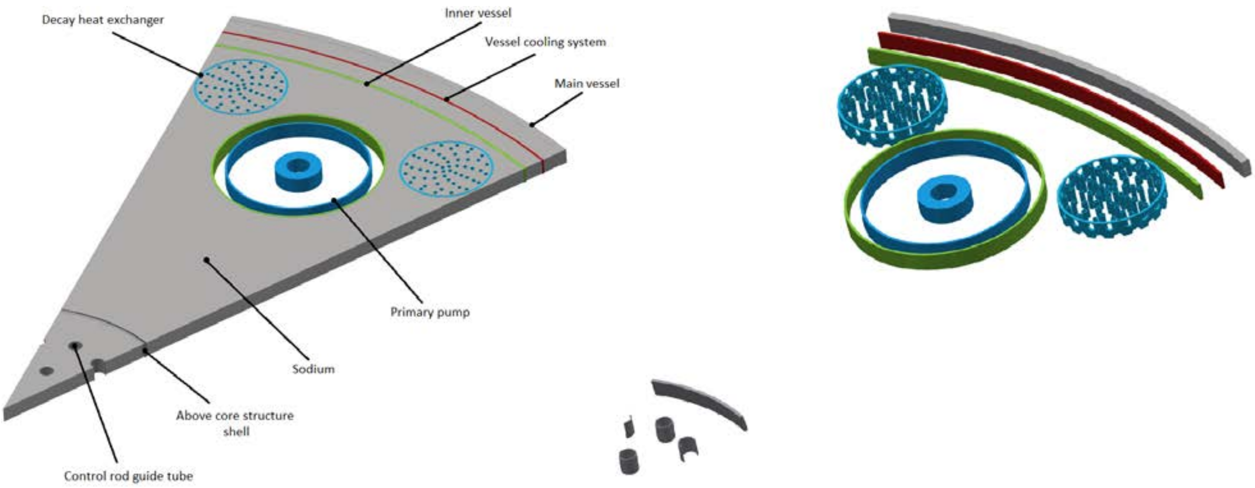

Figure 3.6 CAD model top axial cell (Z=120), sector 3, filled with sodium slice and free volume slice

Other pieces of information which must be deducted from the CAD file are the flow area and hydraulic diameter data, which are essential for a thermal hydraulic calculation. These properties are shown in Table 3.2 and Table 3.3. The flow area can be taken directly from the CAD file but Inventor cannot display the hydraulic diameter of an element, thus it must be calculated. The calculation was done by using the following equation:

$$D_h = \frac{4A}{P_w}$$

Where D_h is the hydraulic diameter, A is the flow area and P_w is the wetted perimeter. Both A and P_w can be obtained from the CAD file directly.

Pipe type	Number of pipes	Axial zone	Height [m]	Flow area [m^2]	Hydraulic diameter [m]	Cell height [m]
Inner fuel assmbly	216	Foot	0.6	0.003	0.009	0.2
		Fission gas plenum	1.2	0.009	0.004	0.05
		Fissile fuel zone	1	0.009	0.004	0.05
		Fertile fuel zone	0.4	0.009	0.004	0.05
		Sodium plenum	0.4	0.026	0.174	0.4
		Axial shielding + reflector	0.85	0.008	0.012	0.425
		Head	0.275	0.02	0.159	0.275
Outer fuel assembly	270	Foot	0.6	0.003	0.009	0.2
		Fission gas plenum	1.2	0.009	0.004	0.05
		Fertile fuel zone	1.4	0.009	0.004	0.05
		Sodium plenum	0.4	0.026	0.174	0.4
		Axial shielding + reflector	0.85	0.008	0.012	0.425
		Head	0.275	0.02	0.159	0.275

Table 3.2 Hydraulic parameters of Fuel assemblies

Pipe type	Θ sectors	Number of pipes	Length [m]	Flow area [m^2]	Hydraulic diameter [m]
Pump vertical part	3,7,11	3	2.67	1.14	0.90
Pump horizontal part	3,7,11	3	1.65	0.82	1.02
Primary IHX	2,4,6,8,10,12	6	7.5	4.01	0.11
Secondary IHX	2,4,6,8,10,12	6	7.5	0.7	0.08
Vessel cooling system pipe: horizontal part	All	20	4	0.01	0.1
Vessel cooling system pipe: vertical part	All	20	6.1	0.01	0.1

Table 3.3 Hydraulic parameters of system elements

In Table 3.4, the calculated porosities are presented for the main thermal hydraulic elements of the system. It is important to calculate this as the primary system in TRACE is represented by allocating a specific porosity for each cell. By using the CAD model, this porosity calculation can be done in a much more detailed manner. Therefore, it is not necessary to use average porosity values for the same element used in different cells. The actual porosity can be calculated for each cell individually instead.

This individual cell porosity calculation was not done as part of the current work because of the shortness of the time frame. Likewise, to show the porosity calculation, only an average porosity was determined for each of the primary system elements.

As it was described before, as a first step, the free volume of an element was determined, providing the volume of the material which builds up the element. Following this, the part was filled with sodium and the material content was determined again. From this point, the sodium content and the porosity could be calculated easily. In addition to the volume information, Inventor displays the error information on the calculated values, which help to calculate the error on the resulting porosities. The error propagation formula which was used for this purpose is the following:

$$\sigma_{result} = \sqrt{\sigma_{sodium} + \sigma_{element+sodium}}$$

Where σ_{result} is the final error on the result, σ_{sodium} is the error on the volume of the sodium and $\sigma_{element+sodium}$is the error on the volume of the system element together with the sodium.

Component	Porosity	Error on Porosity (%)
Strongback	0.92	0.07
Diagrid	0.79	0.76
Core	0.33	0.14
Pump	0.55	0.04
Decay heat exchanger	0.84	0.19
Intermediate heat exchanger	0.71	0.07
Above core structure	0.96	0.09

Table 3.4 Porosities of VESSEL elements

4. Conclusions

In conclusion, it can be pointed out that a CAD model for such a complicated system as pool type Sodium Fast Reactor is useful in many ways. More specifically, through the current work, it has been proven that the CAD model is useful in two particular ways. On one hand, the CAD model proved beneficial for the development of a reactor concept, which can be used for more detailed designs. On the other hand, the CAD model can provide essential input information to run different simulations in order to analyse the safety and feasibility of a specific design idea. Moreover, it is possible to make some quick modifications to the design and then test right away whether the new concept works any better than the previous one.

All of the uses of the CAD model mentioned above were proved and tested through this work. As the European Sodium Fast Reactor (ESFR) model was created, it was used to present a new idea for members who are going to work together on the ESFR-SMART project. To see how useful the model can be for simulation codes, input information was created for the TRACE system code. Moreover, it was proved that, by having a 3D model, more precise data can be obtained and, as such, more accurate simulations can be run.

Unfortunately, the time frame of the current work did not allow to obtain all the data for a whole TRACE primary system simulation. Moreover, as a future work, a script could be made to extract the required information from the CAD software since it can be a lengthy process by doing it manually.

During the preparation of the model, many different ideas for future use of the model were suggested. For instance, the model could be used for other system codes or for CFD (Computational Fluid Dynamic) simulations. As the model is already made and it is relatively easy to make, it would not be necessary to use another software to create the geometry for a fluid dynamic simulation. Instead, it could be taken directly from the CAD software.

Another useful application of the model is the possibility to use the model for 3D printing for further demonstration purposes. The usage of the model for 3D printing was also researched during this work and the actual printing is under process. The CAD software can easily scale the model for the required size, making it possible to export the file into the right format directly. After this, it can be sent right away to a 3D printer to create a real touchable object which is very convenient to familiarise the public with the reactor design. Nevertheless, although the project is at the beginning of its stage, it was already possible to realize that the current printing technology available does not allow to print the model with all of its details. This is because the model has to be scaled down from the 17.3 m main diameter of the vessel to only 0.3 m (chosen size for the mock up) in this specific case. This scaling down would result in that some smaller features of the reactor would not be possible to be printed because of printing resolution issues. At this point, an advantage of the CAD model is made evident since the model can be quickly modified to allow the 3D printing to happen.

Another further use of the model is to create introductory videos about the reactor. The CAD software has a built in video editing tool which would make it possible to further clarify any design concept and to demonstrate the reactor components to the public. This feature has not been assessed yet.

In the author`s opinion, it was demonstrated that a CAD software is very useful not only in the production stages but also in research facilities.

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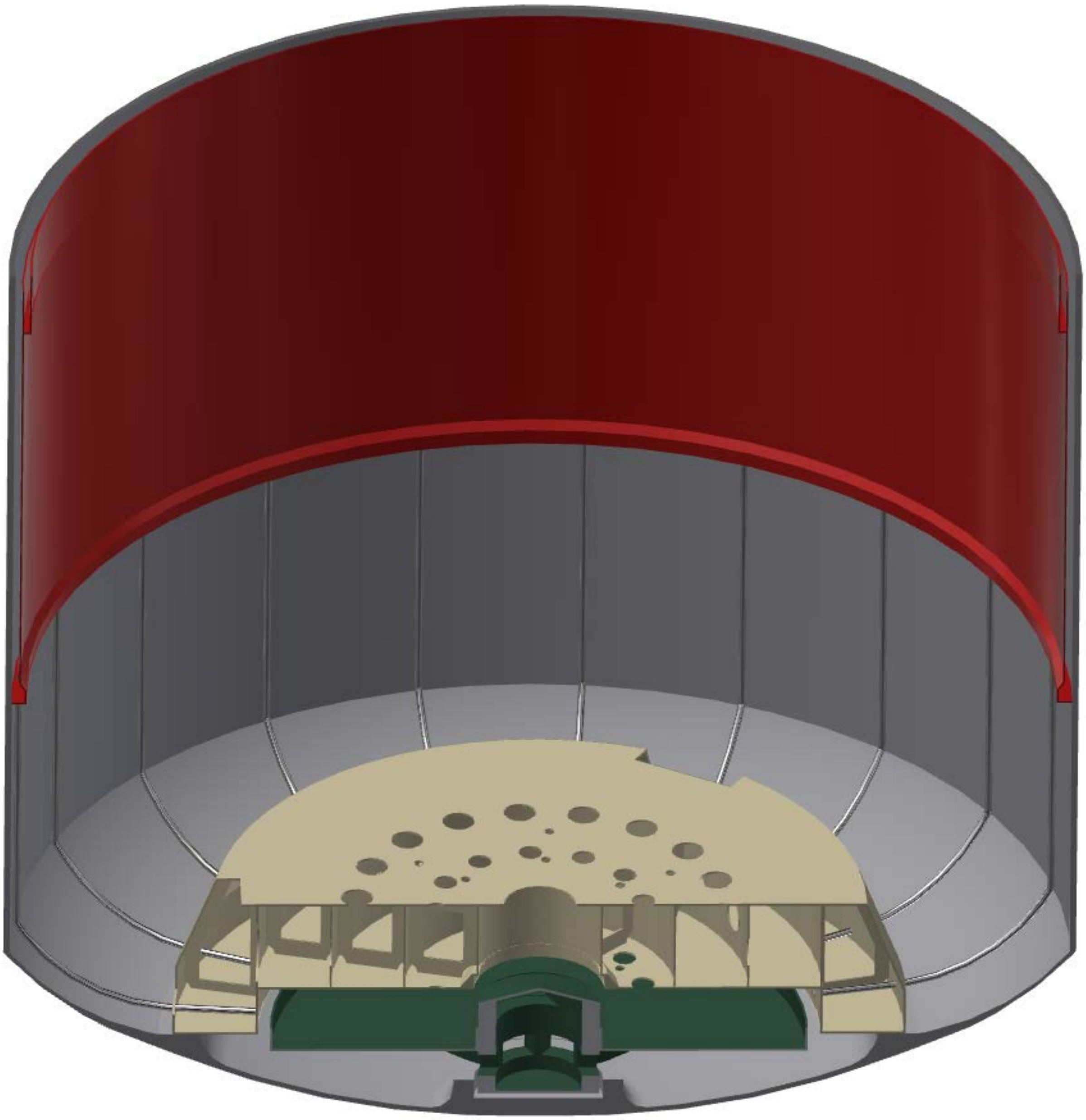
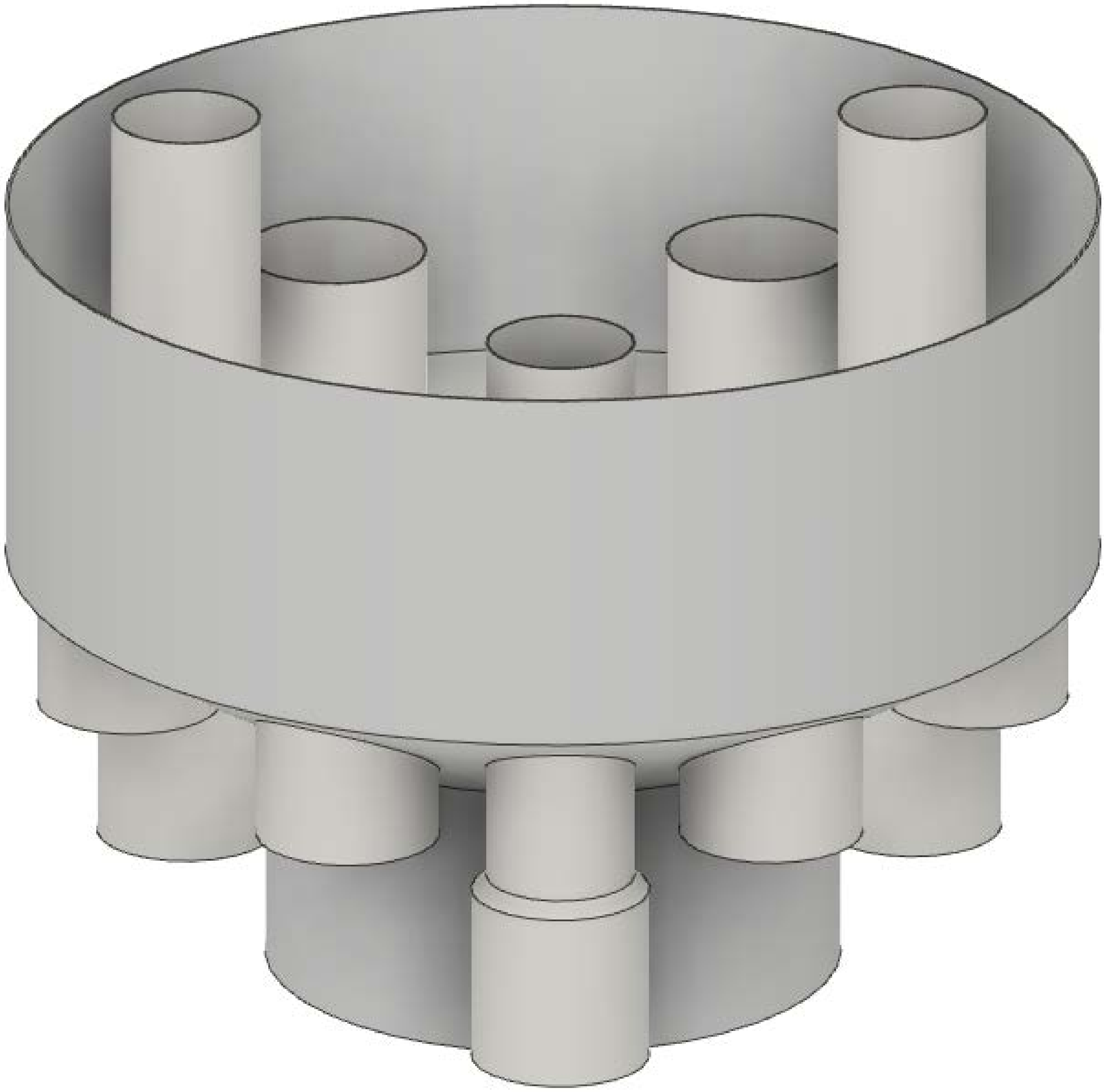
Appendices

A - Vessel Cooling system



B – Inner vessel



C – Large rotating plug

D – Porosity calculation table for main components

Main part	Sub part	Volume (mm2)	Error (%)	Porosity	Error on porosity (%)	
Intermediate heat exchanger	Sliced part	8693754000	0.06835	0.71	0.068516	
	Sliced part with sodium	29994415000	0.00477			
Decay heat exchanger	Sliced part	1049845000	0.15897	0.84	0.187658	
	Sliced part with sodium	6747945000	0.09972			
Primary pump	Sliced part	5733077000	0.03171	0.55	0.034773	
	Sliced part with sodium	12710802000	0.01427			
Diagrid	Part without soidum	8741230000	0.74812	0.79	0.759211	
	Part with soidum	50595098000	0.1293			
Strongback	Part without soidum	16390860000	0.07157	0.92	0.071785	
	Part with soidum	2.18412E+11	0.00555			
Above core structure	Sliced ACS	4542285600	0.08861	0.96	0.088695	
	Sliced ACS with primary sodium	1.05801E+11	0.00388			
Core	Fuel assembly	72565215.87	0.05122	0.25	0.053969	216
	Fuel assembly with sodium	168916924.4	0.017			
	Breeder assembly	77918630.87	0.0497	0.31	0.053646	280
	Breeder assembly with sodium	191318083.2	0.0202			
	Reflector	78662258.94	0.08323	0.31	0.089956	270
	Reflector with sodium	191877914.8	0.03412			
	Empty slots			1	0	13
	Contol rod guide tubes			0.88	0.077693	33
Average core porosity: 0.32682			Error on average core posority: 0.14113			