Simulation of Severe Nuclear Accidents

Submitted by

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In partial fulfilment of the

requirements for the Degree of

Bachelor of Science (Honors) in Physics

National University of Singapore

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AY 2015/16

Abstract

Severe nuclear accidents comprise of complicated physical phenomena that occur at distinct moments of the accident progression. Computer simulations are crucial tools to describe and explain how a nuclear reactor and its containment might behave when subjected to severe accident conditions. They are used to back design proposals or specific actions to reduce the consequences of severe accidents.

In this project, we used ASTEC (Accident Source Term Evaluation Code) and evaluated the behavior of a nuclear reactor pressure vessel subjected to different design modifications. We specifically looked into the late phase of core degradation where the time of rupture following a severe accident was an important consideration.

We improved the given code for the French 900 MW power plant by optimizing the mesh elements. We found that increasing the total thickness of the pressure vessel increases the rupture time as expected. We convinced ourselves that the values obtained were reasonable by performing a back of the envelop calculation. We demonstrated that strategic thickening of the pressure vessel gives comparable rupture times relative to increasing the total thickness. We showed that ex-vessel cooling can be used to increase the rupture time as expected. However, limitations of the code prevented us from displaying asymptotic rupture time. Further investigation revealed asymptotic rupture time when corium mass was reduced. Noisy rupture time results was also smoothed out by proper axial meshing during corium mass variation.

We conclude by discussing the shortcomings and limitations of ASTEC. They include the limited number of element meshes in the lower plenum code, the lack of flexibility in the heat removal code and lack of direct data export. Possible future work and adaptations such as the full PWR900 simulator and Molten Corium Concrete Interaction are also briefly examined.

Acknowledgements

I would like to express my gratitude to Associate Professor Chung Keng Yeow for this final year project. His calm and collected advice certainly smoothed some of my edges. I am thankful for his insights in matters even beyond this project.

I would like to acknowledge Professor Lim Hock for passing on encouragement and advice before the start of this final year project. A push in the right direction goes a long way.

I would like to thank Hoe Wei Qi for the various exchanges and conversations throughout the final year project. Moments of serendipity are most often chanced upon when talking with friends.

I apologize to my professors who taught my last semester modules because I frequently skipped your lectures. FYP consumed my life. I'm sorry.

I remember the builders who worked so hard at night (from 6pm-10pm) without fail to upgrade S14. Their constant hammering, drilling and other various loud ($100 \pm 10\,\mathrm{dB}$) noises served to remind me that I was not alone in this world as I stayed up late in my lab. But seriously, who tripped the fire alarm for over an hour?

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1 Introduction

There has been a growing resurgence of nuclear energy in the past decade due to several coinciding factors. The growing worldwide electricity needs combined with the pressure on reducing carbon emissions create a dilemma, particularly for rapidly developing countries. Because of this, some countries are looking and turning to nuclear energy for energy security. Developed countries, such as USA, UK and France, are also seeking to supplement their nuclear energy with late third-generation units. [1] However, the scale of these operations are dwarfed in comparison to projects in East Asia. In countries such as China, India and South Korea, more than a hundred further nuclear power plants have been proposed.

1.1 Regional Context

	Reactors planned	**Reactors proposed**	**Research reactors**
Indonesia	1	4	3
Malaysia	0	2	1
Philippines	0	1	1
Thailand	0	5	2
Vietnam	4	6	1

Table 1: Nuclear power in Southeast Asia [2]

While there are no nuclear power plants in Southeast Asia, some countries in the region have included nuclear power in their long-term plans. The table above summarizes the developing nuclear programmes in ASEAN.

Vietnam is the most active in this development. [3] It has signed two inter-governmental agreements to build nuclear power plants. The first with Russia to build its first nuclear power plant (Ninh Thuan 1). The second with Japan to build its second nuclear power plant. They were originally planned to come online in 2020 and 2025 respectively. However, due to further safety concerns, they have been delayed and pushed back to 2028.

Singapore will not be pursuing nuclear energy at this point of time as present technology has been deemed not suitable for the island. However, the country is strengthening its capabilities in order to become a regional player and to protect its people.

1.2 Motivation

During the operation of any nuclear power plant, safety is of the highest concern. [4] One of the most critical factors is the possibility of severe accidents. However, it is extremely difficult to conduct experiments to gather data in this regard. Therefore, the nuclear energy community developed severe accident analysis and codes which are then validated via international experiments. These codes are primarily used for training purposes, development and validation of accident management programmes, design and validation of severe accident mitigation systems and most recently, plant simulators.

One such tool is ASTEC (Accident Source Term Evaluation Code) which we will use in this project. This is a newly acquired tool in Singapore. Thus, there are three main objectives for this final year project.

1. To gain sufficient proficiency in ASTEC through understanding and optimizing the code.

2. To be able to reasonably explain the results of the simulation and relate to known physics.

3. To test simple improvement ideas to the design of a nuclear power plant.

Nuclear reactors have a diverse range of designs. Hence, there is a need to focus the study on a single type. The French 900 MWe Pressurized Water Reactor was chosen. This design is well established with around 30 plants in France and 10 plants exported out of France. During this period, the 900 MWe fleet in France are undergoing inspections to ascertain their operational status (from 2009 to 2020). So far, 5 plants have been approved for operating longer, some with minor upgrades to make them more resistant to core-melt. Thus, we will also be looking at issues of core-melt. [5]

2 Nuclear Energy

Nuclear fission reactions is the primary process to produce electric power in nuclear power plants. Therefore, studying these reactions would be a fitting starting point to understand nuclear energy. In order to realize the immense amounts of energy generated in relation to the mass of fuel used in nuclear power, we shall compare it with fossil fuels. Note the combustion of coal to produce electricity given by [6]

$$C + O_2 \ \rightarrow CO_2 + 4\ \text{eV}\,. \tag{1}$$

A nuclear fission reaction generates approximately 200 MeV per uranium atom. Thus, a nuclear fission reaction releases about 50 million times more energy.

When considering typical 1000 MW electrical plants, a coal plant will consume 10000 tons of fuel per day whereas a nuclear plant would consume 20 tons a year. Moreover, the supply chains of these two plants are different. A coal plant requires constant resupply whereas a nuclear plant is shut down for refueling once every 1-2 years.

Of course the radioactive waste from nuclear plants is much more harmful than by-products of operating coal plants. However, the quantity of waste is far lesser. Half of the nuclear fuel becomes deadly waste which must be carefully disposed of (10 tons per year). 5% of coal burned becomes ash dirt which must be stored in a landfill (500 tons per day). Perhaps the most alarming environmental impact is the thousands of tons of CO_2 released by a coal plant every day. We see now that nuclear power is actually orders of magnitude greener than fossil fuel power.

2.1 Nuclear Fission

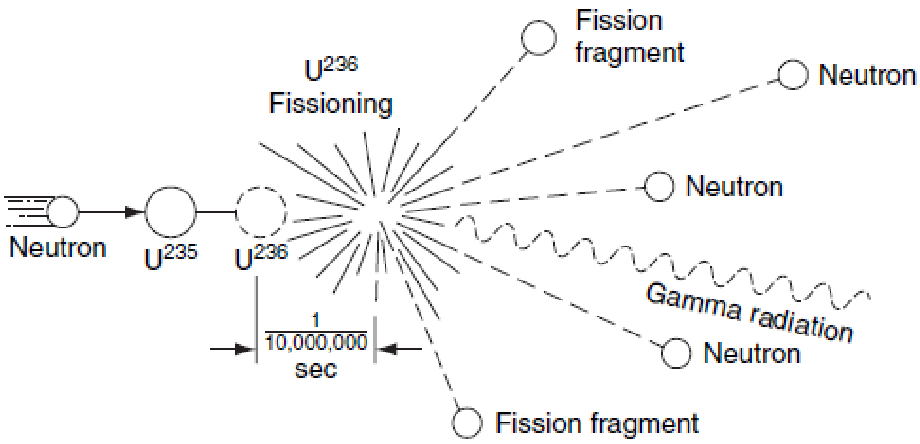


Figure 1: A fission reaction [6]

A typical nuclear fission reaction is shown in the above figure. A neutron reacts with a heavy nucleus. From the reaction come approximately 200 MeV, a few neutrons and fission fragments. These fission fragments undergo radioactive decay and produce further fission products. The few neutrons produced in a nuclear fission undergo more collision events. These neutrons may cause further nuclear fission reactions and so on. Hence, this neutron multiplication drives the chain reaction of nuclear fission.

Nuclear fission produces many diverse pairs of fission fragments. A typical nuclear fission is given by

$$\mathrm{n} + {}^{235}_{92}\mathrm{U} \;\rightarrow\; {}^{140}_{54}\mathrm{Xe} + {}^{94}_{38}\mathrm{Sr} + 2\mathrm{n} + 200\ \mathrm{MeV}\,. \tag{2}$$

These fragments are unstable and predominately decay through beta emission and gamma radiation. In order to reach stability, these fragments undergo a decay chain. A typical decay chain is given by

$${}^{140}_{54}\mathrm{Xe} \xrightarrow{\beta} {}^{140}_{55}\mathrm{Cs} \xrightarrow{\beta} {}^{140}_{56}\mathrm{Ba} \xrightarrow{\beta} {}^{140}_{57}\mathrm{La} \xrightarrow{\beta} {}^{140}_{58}\mathrm{Ce}\,. \tag{3}$$

All of the fission products have decay chains. Thus, even after nuclear fission has been stopped, radioactive decay will linger and generate significant amounts of decay heat. This decay heat can be approximated by the Wigner-Way formula [6] as

$$P_d(t) = 0.0622\ P_0 \left[t^{-0.2} - (t_0 + t)^{-0.2}\right]\,, \tag{4}$$

where $P_d(t)$ is the power generation due to the decay heat, P_0 is the power before shutdown, t_0 is the time of power operation before shutdown and t is the time since shutdown in seconds. This decay heat is roughly calculated using the empirical mean energies of the beta emissions, gamma radiation and their respective rates per nuclear fission.

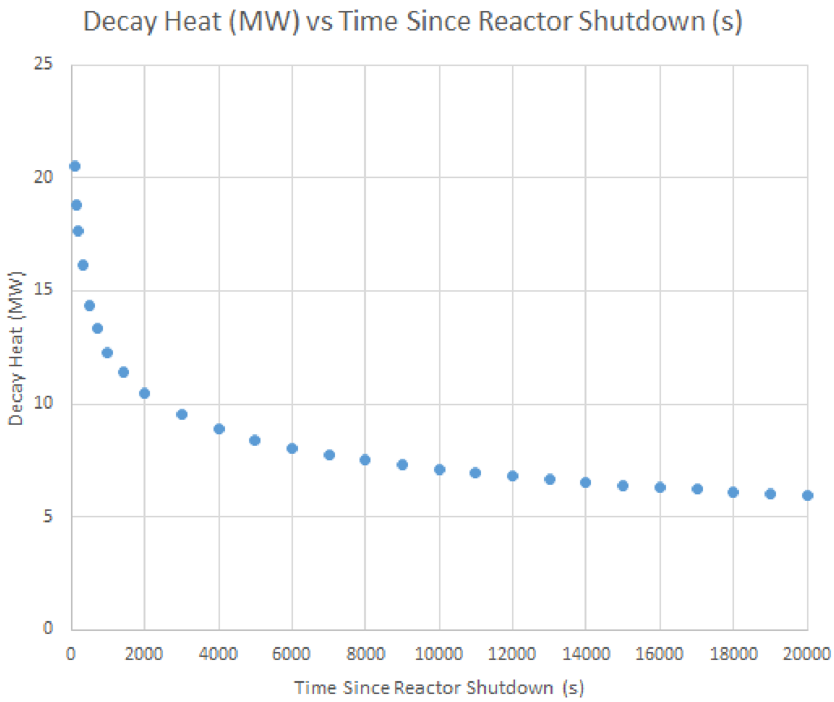


Figure 2: Wigner Way Formula plot for a 1 year old reactor from equation 4

From the above figure, we see that the decay heat is still in the MW range. Hence, cooling is still necessary to avoid accidents following stoppage of nuclear fission.

2.2 Nuclear Reactors

Nuclear reactors have been devised with a broad assortment of configurations. However, most nuclear reactors vessels are cylindrical in shape with coolant running through pipes down the axial length of the core. The core is enclosed by the vessel and is composed of the fuel assemblies. Actually, the full description of the core is far more complicated. Fortunately, the exact details are not important to us for this project.

Nuclear reactors are also divided into categories according to what they use as coolant and/or moderator. Moderators are used to slow down fast neutrons (>0.1 MeV) to thermal neutrons (<1.0 eV). Light water reactors use normal water as both coolant and moderator. Heavy water reactors, as the name implies, use D_2O instead. There are also graphite moderated reactors and newer generation nuclear reactors which use a gas coolant but their designs are quite distant from the water reactors.

Light water reactors are further divided into two categories: Pressurized water reactors (PWRs) and boiling water reactors (BWRs). More than half of the world's light water nuclear reactors are PWRs. This project focuses on a pressurized light water reactor. However, we shall discuss BWRs briefly at this stage to bring context to a subsection on Fukushima later.

2.2.1 Boiling Water Reactors

BWRs operate at lower pressures of around 700 bar. This allows boiling to occur in the coolant channels. As Figure 3 indicates, the water coolant is fed from the bottom up into the reactor vessel. It then generates steam which drives the turbine. The entire circuit consists of one primary loop. This has significant advantages and disadvantages when it comes to safety which we will elaborate on later.

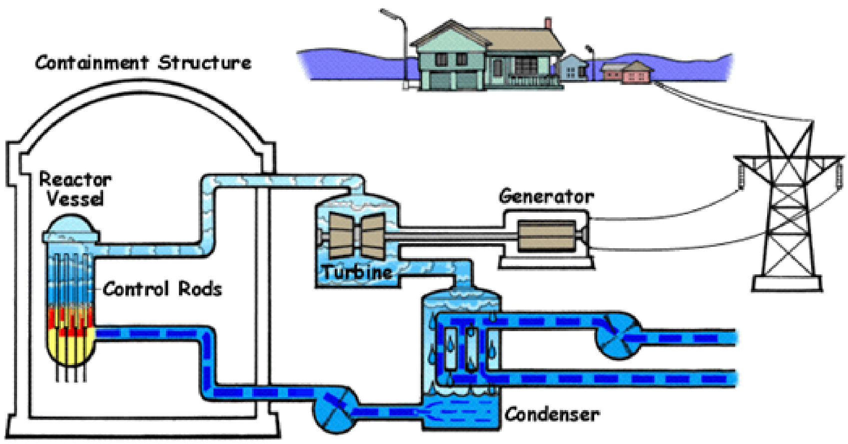


Figure 3: Boiling Water Reactor [6]

2.2.2 Pressurized Water Reactors

PWRs operate at higher pressures of around 1500 bar. As the operating temperature of the nuclear reactor is around 300 degrees Celsius, this allows the water to remain in a liquid state. As can be seen from Figure 3, there are two circuits. The primary coolant loop consists of the reactor vessel and the pressurizer. The secondary coolant loop consists of the steam generator, the turbine and the condenser. Hence, the water that mixes inside the nuclear reactor vessel is contained entirely within the primary loop. This is the stark contrast between the PWR and the BWR.

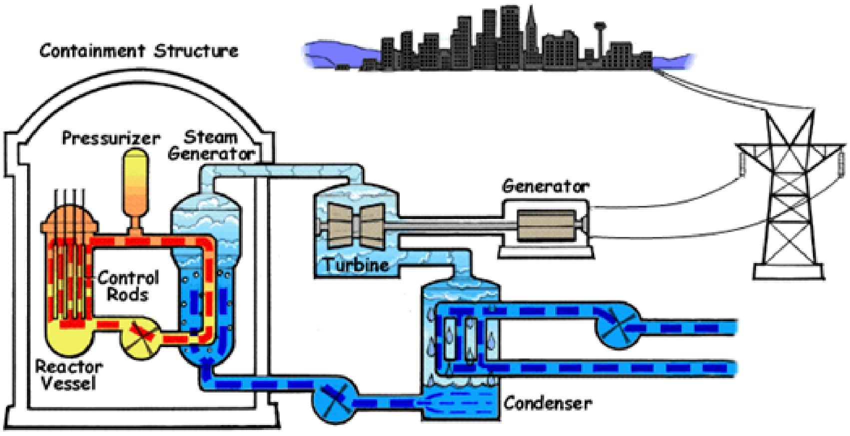


Figure 4: Pressurized Water Reactor [6]

2.3 Nuclear Accidents

The operation of nuclear power plants involve potential human and environmental risk due to harmful radioactive exposure. Hence, the nuclear industry adheres strictly to the safety of its facilities. They are always designed, constructed and used in such a way to prevent accidents. Furthermore, measures are taken to continuously improve the level of safety. This is done through periodic feedback and reassessments when advances in scientific knowledge and simulation tools are made. Unfortunately, the possibility of an accident occurring, over the relatively long term, cannot be excluded. At this junction, we introduce three infamous nuclear accidents in order to develop more relatable context for this project.

2.3.1 Chernobyl Accident

On 26th April 1986, the Chernobyl-4 reactor exploded. This was the worst accident ever in terms of human lives lost and cost in damages. The reactor was moderated by graphite. As mentioned earlier, the design is vastly different from PWRs or BWRs.

The accident was caused by a disregard of safety procedures and specific characteristics of the reactor design. [7] The explosion was caused by what is known as a criticality or reactivity accident. Uncontrolled chain nuclear fission reactions occurred and vaporized a large amount of water, causing a steam explosion. This released hazardous radioactive materials to the environment as there was no proper containment built around the reactor.

2.3.2 Fukushima Daiichi Accident

On 11th March 2011, the various units of the Fukushima nuclear power plant suffered from core reactor meltdowns and released radioactive material into the environment. This was primarily caused by the tsunami triggered by the Tohoku earthquake. All power sources were lost, hence, all cooling sources were lost as well, following the tsunami. The lack of decay heat removal eventually led to core degradation. The reactor vessel began to leak hydrogen which resulted in hydrogen explosions in the building of some units. However, there was no released of airborne radioactive materials as in the case of Chernobyl as Fukushima had proper containments.

In order to mitigate further pressure damages, they started to release coolant from the primary loop. However, due to the design of BWRs, this water was highly contaminated with radioactive materials and was the primary health concern.

2.3.3 Three Mile Island Accident (TMI)

On 28th March 1979, a core-melt accident occurred in the TMI-2 nuclear power plant. Although not as physically damaging as Chernobyl or Fukushima, historically, it marks an important change in the paradigm of nuclear safety. Previously, core degradation had not been seriously considered in depth at all.

TMI was a PWR. The accident was caused by a combination of human and instrumental error. [7] The necessary pressurizer valve was not closed and this led to the water in the primary loop to vaporize. This loss of coolant eventually led to core degradation. Re-flooding the reactor core was able to stop the vessel from rupturing. However, the re-flooded water was contaminated and some of it leaked out.

It must be emphasized that the vessel being full of water (due to re-flooding) could have led to a violent steam explosion. One explanation for why this did not occur could be that the slow flow (on the order of minutes) of melted materials into the water prevented any sudden behavior. Another factor could be that a crust layer formed between the molten corium (melted materials) and the vessel wall reduced heat exchange between them.

There are a few key learning points from the TMI accident which relates directly to our project.

1. Re-flooding may potentially stop vessel rupture but only if steam explosions can be avoided.
2. Past a certain molten corium mass, re-flooding will not stop vessel rupture.
3. The longer the molten corium mass stays in the vessel (prolonging rupture time), the re-flooding coolant has a greater chance of stopping vessel rupture entirely.

2.3.4 Loss Of Coolant Accidents (LOCAs)

As mentioned earlier, radioactive decay heat is not trivial even after nuclear fission has stopped. Therefore, nuclear accidents most often occur when there is a loss of coolant, typically known as LOCAs. From the examples we discussed above, we can also see that LOCAs can transit into more severe accidents. There are many definitions of a severe nuclear accident in the literature. However, we shall simply define a severe nuclear accident as nuclear reactor core degradation in this project. We do not concern ourselves with how the LOCA occurs in this project but rather how to deal with the severe accident that arises from it.

3 ASTEC

The ASTEC code (Accident Source Term Evaluation Code) has been jointly developed for several years by French Institut de Radioprotection et de Surete Nucleaire (IRSN) and the German Gesellschaft fur Anlagen und Reakotorsicherheit mbH (GRS). Its main applications are source term determination studies but it also provides accident management studies and physical analyses of experiments that improve the understanding of phenomenology. Since 2004, ASTEC is progressively becoming the reference for the European water-cooled reactors. It has numerous international validation experiments backing its physical models. Its software structure has high modularity which is particularly useful when a project wishes to focus on a single aspect.

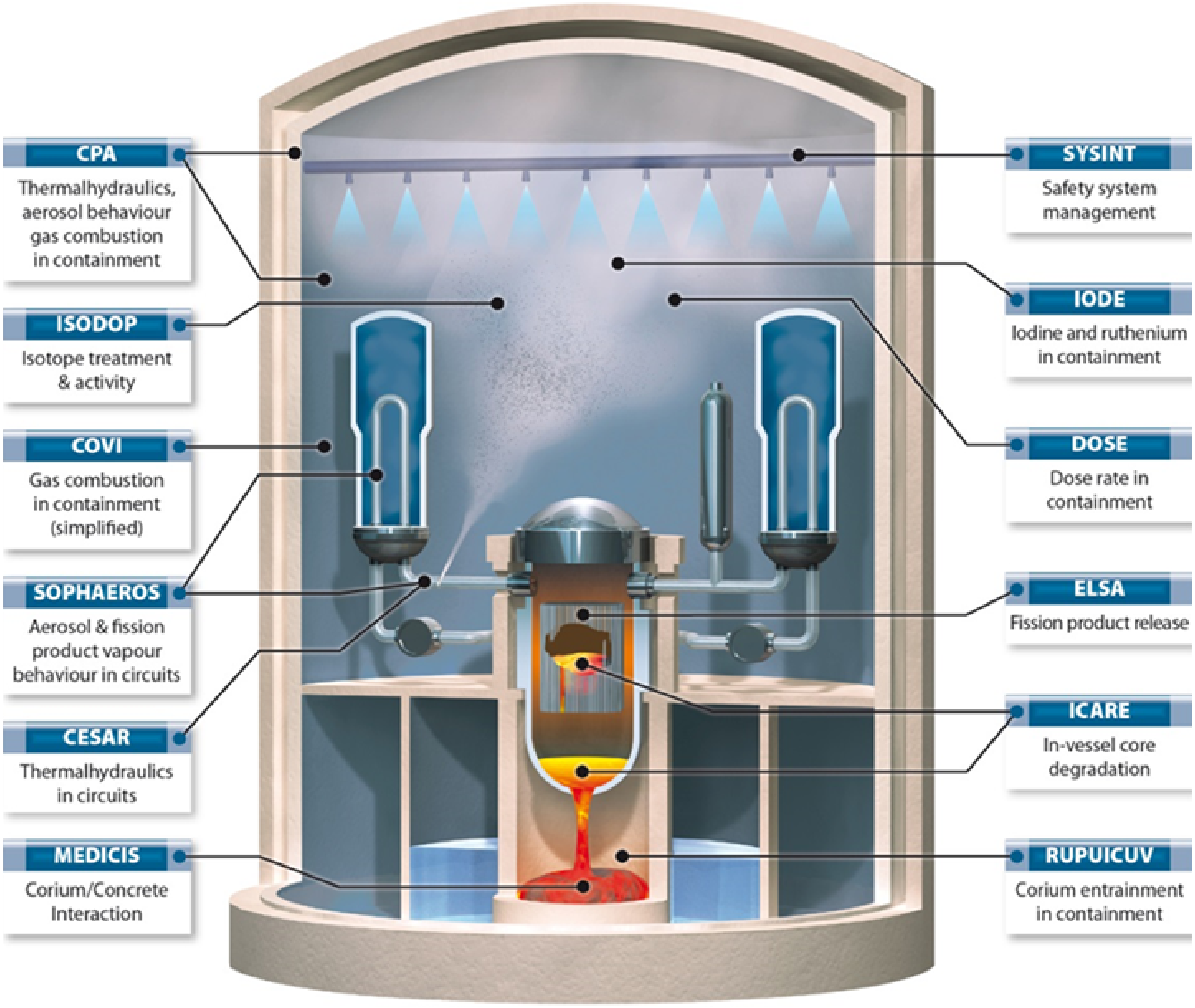


Figure 5: ASTEC Modular Structure [9]

The figure above shows the different modules of ASTEC and roughly indicates which part of the nuclear reactor it focuses on.

3.1 ICARE

When we discussed the TMI accident, we realized that prolonging the time before vessel rupture mitigates and may even halt vessel rupture entirely. Therefore, the module of choice is ICARE. ICARE contains the physical modeling of in-vessel core degradation. For the sake of brevity, we only discuss the general ideas of ICARE's computational models and leave the technical details out. [10]

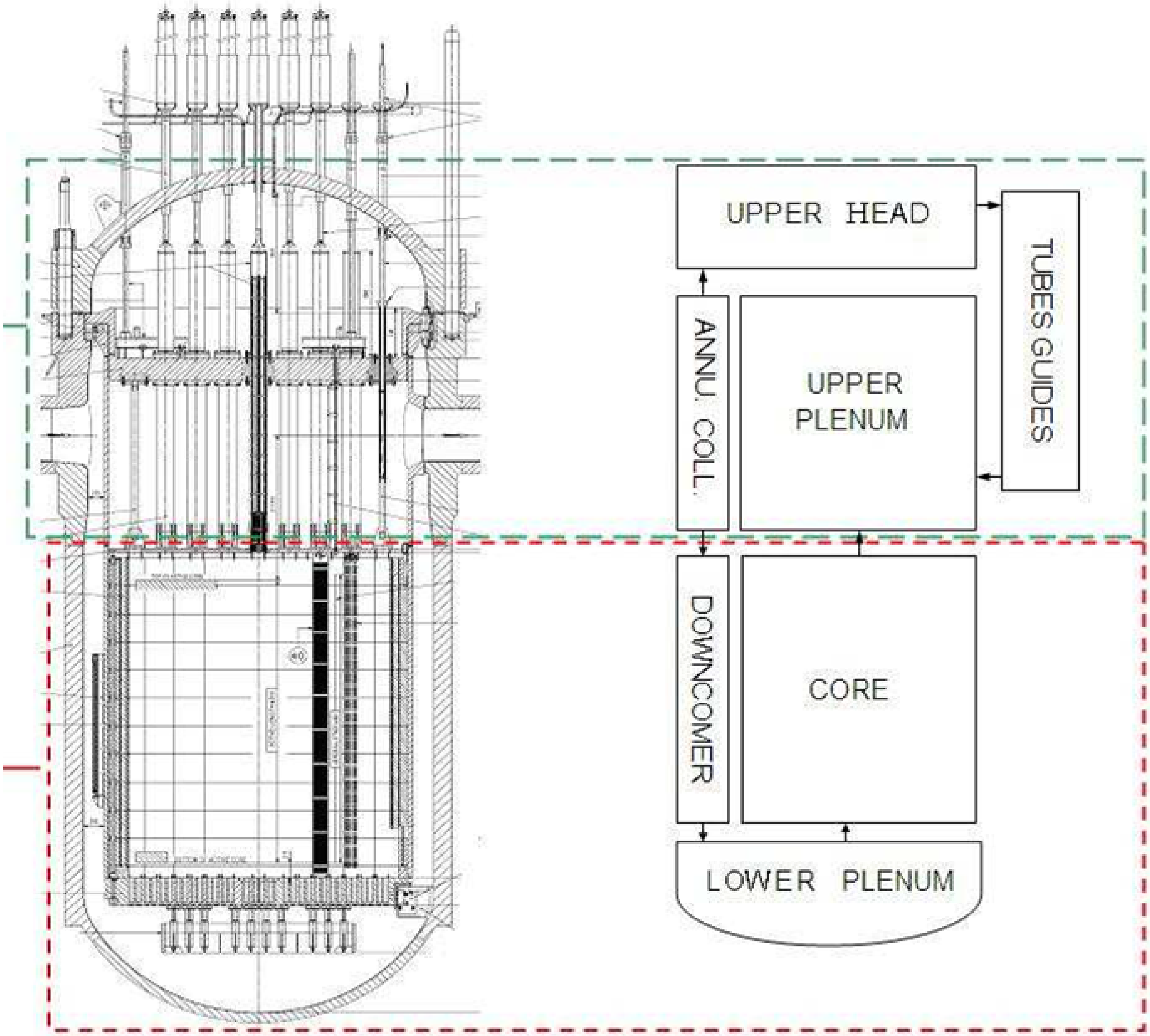


Figure 6: ICARE simulation represented by dotted red line. Dotted green line represents another module

Core degradation can be divided into two phases: the early phase and late phase. [10]

Early phase corresponds to the first degradation phase of the accident and covers the initial heat up phase before any material movement. This involves the initial stages where the fuel rod starts to heat up due to radioactive decay. The fuel rod starts to melt and eventually, dislocation occurs.

Late phase corresponds to an advanced degradation phase of the accident involving debris and molten pool formation. There is substantial melting, material relocation and eventually, rupture of the vessel. We narrow our focus once again to study only the simplified case of the lower plenum (or vessel lower head).

3.1.1 Vessel Lower Head

The vessel lower head can be meshed axially and radially. The meshing is made of truncated cones. Any shape of a reactor vessel lower head can be modeled. [10]

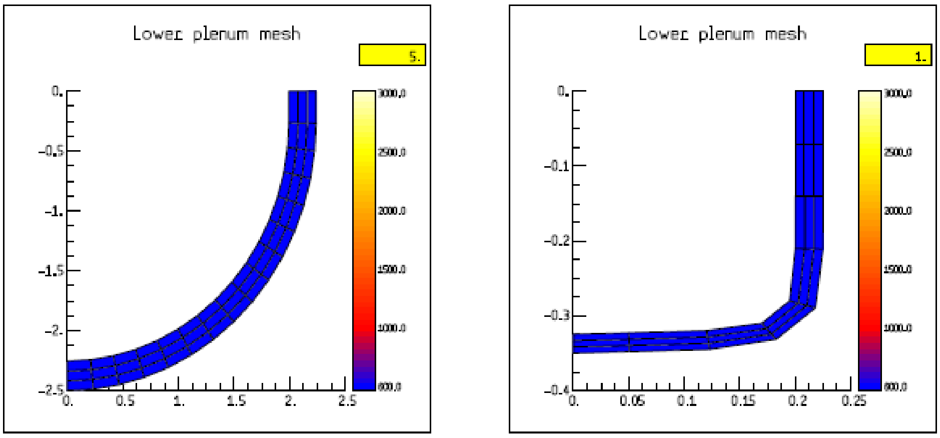


Figure 7: Meshing: hemispheric shape (left) vs ellipsoid shape (right) [10]

The internal structures in the lower head are modeled by a representative cylinder. They may represent the fuel rods and various others within the vessel lower head.

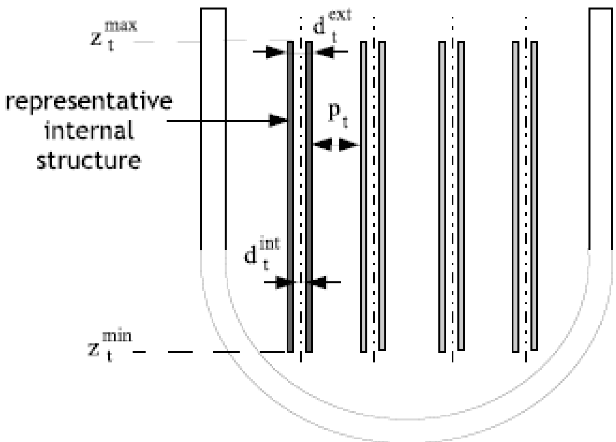


Figure 8: Representation of internal structures [10]

3.1.2 Corium Layers

When corium is in the lower plenum, the following physical phenomena are taken into account: quenching by residual water, molten pool formation, crust formation at the upper level and in contact with cold structures, radiation heat transfer, melting of structures and rupture of structures. All these phenomena lead to very complex corium configurations. To simplify this problem and according to the ASTEC code requirements, some assumptions are made.

A corium layer uniformly spreads in a horizontal way and forms a perfect horizontal layer. A corium layer is homogeneous in terms of composition and temperature. The heat transfers are made by direct contact with the vessel walls, the internal structures, and with the layers in contact. The vessel wall and the internal structures are colder than the corium: as a consequence, if the temperature is lower than the corium, a crust will form between the pool and the wall. The crust is not "physically" represented but acts as a thermal resistance for the thermal transfers between layers. We assume that the crust thickness is very small compared to the layer dimensions. [10]

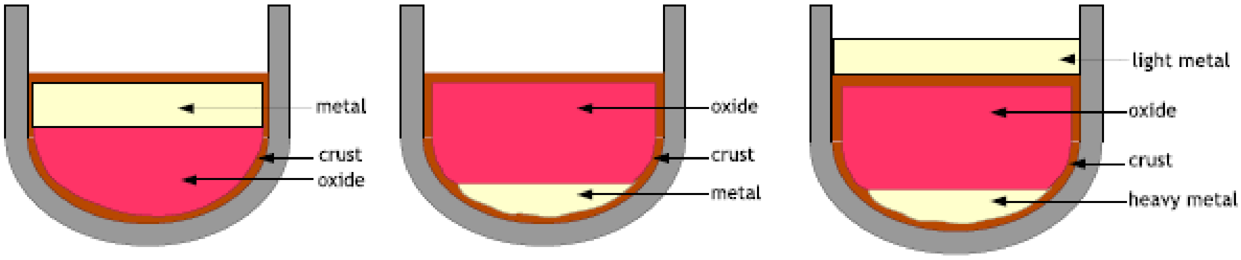


Figure 9: Corium layer configurations [10]

3.1.3 Heat Transfers

Thermal transfers for a dense corium layer depend on the characteristics of this layer, for example, its temperature and the power generated by fission products in this layer. However, more importantly, it also depends on the characteristics of the layers above and below it. There are four heat transfer formulations. [10]

1. Conductive heat transfers for solid layers.
2. Convective heat transfers generated by decay heat of fission products.
3. Convective heat transfers generated by temperature gradient between surrounding corium layers.
4. Axial conductive heat transfers and radial convective heat transfers for liquid layers.

3.1.4 Movement of Materials

The decanting model deals with an instantaneous movement of molten materials. They are often referred to as donors and receivers. In this case, the donors would be the internal structures and the vessel lower head which donates material when it melts. The receivers are then the dense corium layers. [10]

3.1.5 Lower Plenum Failure

In ICARE, the vessel rupture can be obtained using several criteria: temperature, molten fraction of vessel lower head and mechanical stress. [10] However, the main contribution to this failure is due to the high melting temperatures. Thus, it is our main focus when we discuss mitigation factors.

3.2 Test Cases

The task of writing a simulator code from scratch is a massive undertaking nor is generating a design of the vessel without realistic dimensions feasible. It is then fortunate that the specifications and designs of a French PWR900 is available in ASTEC. PWR900 refers to pressurized water reactor that produces 900 MW of electrical power.

We recall our motivation and elaborate further. These nuclear power plants were constructed around the early 1980s. Extending the life of these existing nuclear power plants are less costly than bringing in alternatives. There is considerable effort put into the specific study of these cases as they are important and necessary during inspections. The majority of effort and time was put into understanding, optimizing and subsequently modifying the design and code of this PWR900 simulation. From this point, when we refer to the simulation, we refer specifically to the PWR900 simulation. [11]

We will briefly elaborate the graphical output of ASTEC for the following figures. It is a 2D temperature field plot of a cross section of the reactor pressure vessel. The temperature scale is given by the color palette on the right hand side. The x and y coordinates gives the dimensions in metres (take note that the pressure vessel extends vertically beyond what is shown). The time, in seconds, is displayed on the top right hand side (often beginning from the time of the LOCA).

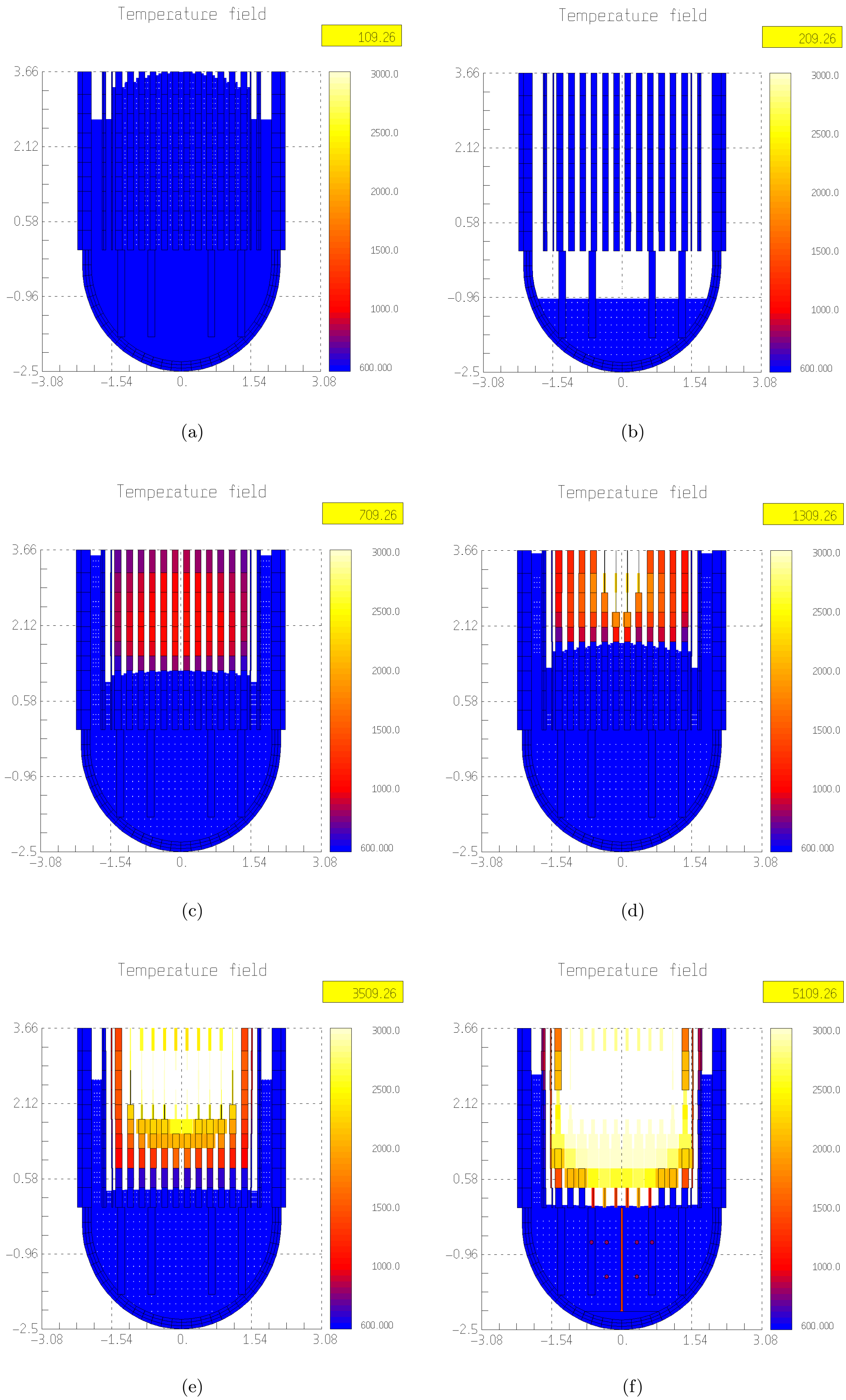
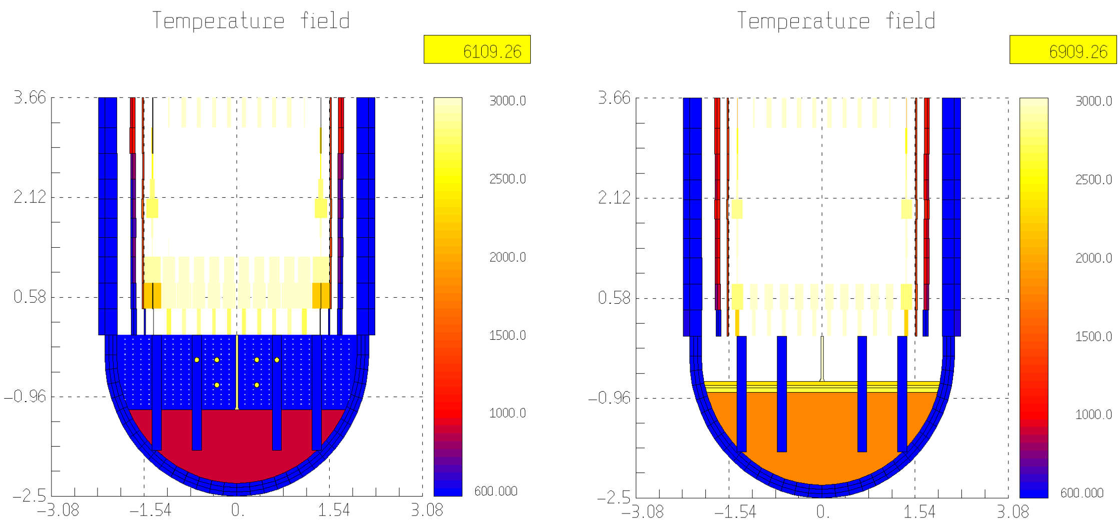


Figure 10: Temperature field of reactor core during different stages of core degradation



(a)

(b)

Figure 11: Temperature field of reactor core during corium build up in the lower plenum

The simulation begins with a LOCA with a leak in the pressurizer (Figure 10a). Hence, the water in the vessel is much more susceptible to vaporization (Figure 10b). There are some safeguard mechanisms which re-floods the pressure vessel (Figure 10c). However, the decay heat still runs away due to the lack of sufficient coolant (Figure 10d) and melting occurs (Figure 10e). The corium begins to flow into the lower plenum and begins to vaporize all remaining coolant (Figure 10f, 11a, 11b).

At this point, we are in the late phase of core degradation. In order to save time, we use a simplified simulation that begins at this point (Figure 13a). The lower plenum begins to heat up and melt (Figure 13b, 13c, 13d, 13e). Eventually, the pressure vessel ruptures (Figure 13f).

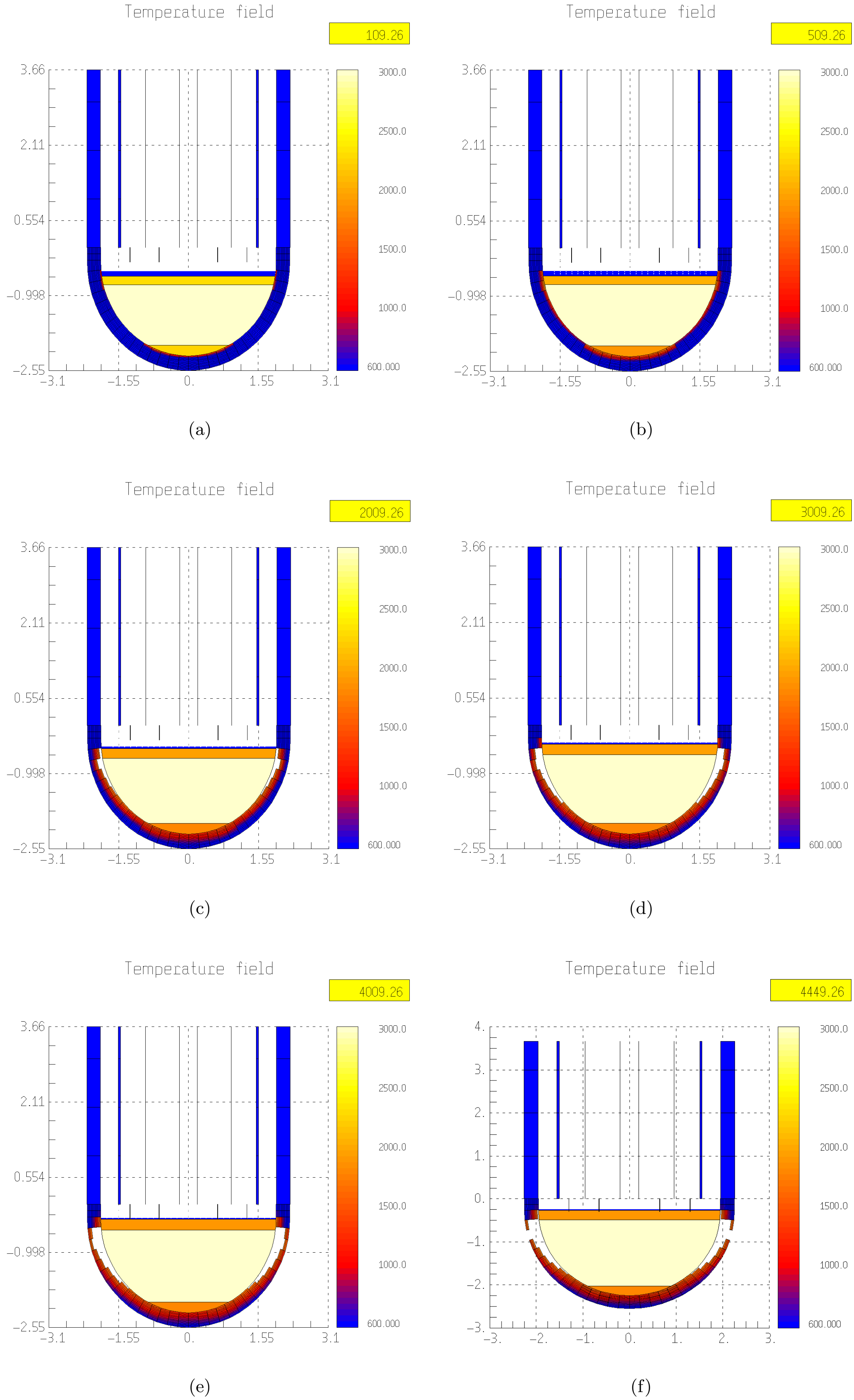


Figure 12: Temperature field of reactor core during very late phase

4 Simulation Work Done

We recall our motivation in order to divide the work done on the simulation into three broad categories: understanding, optimization and then application. It is also during this juncture that we point out that the majority of effort of this project was placed here due to the fact that there was no external guidance on the intricacies of ASTEC during the duration of this project.

Since the focus of this project is to deal with the late phase of core degradation, specifically, vessel rupture mitigation, we chose the simplified simulation that begins directly at the late phase. Overall, the lessons learnt from TMI-2 was brought here and the focus of study was to prolong rupture time.

4.1 Understanding

During this phase, we went through each line of the code to determine its function. There were a number of peculiar details of the simulation to note. The minor ones were quickly fixed to make more realistic sense. However, the major ones were the 'large' timestep and the number of meshes.

The timestep was set to 20 seconds. It is normally assumed in simulation that smaller timesteps gives more realistic results. Hence, we should reduce the timestep to a smaller time interval.

The number of meshes were under-utilized. The constraint of using ICARE module and ASTEC itself limits the number of meshes to a finite number. Hence, we should tune the number of meshes up to the highest limited value.

4.2 Optimization

The following question is then how small should the timestep be? How should the elements be meshed in order to maximize the result? It is also normally assumed in simulation that these proposed steps will increase computational cost. Therefore, we needed to balance all these outcomes and justify the actions to take.

4.2.1 Meshing

There are two main ways of meshing: axial and radial. The figures below clarify the stark contrast between having more axial meshes or more radial meshes.

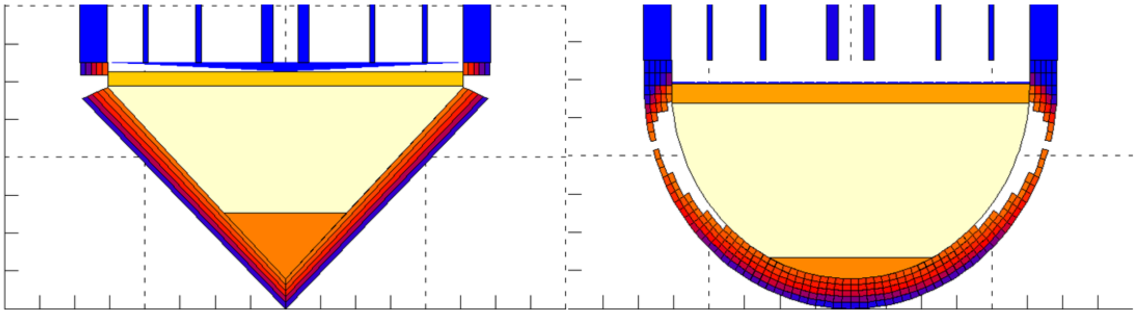


Figure 13: Axial meshing: 1 mesh vs 37 meshes

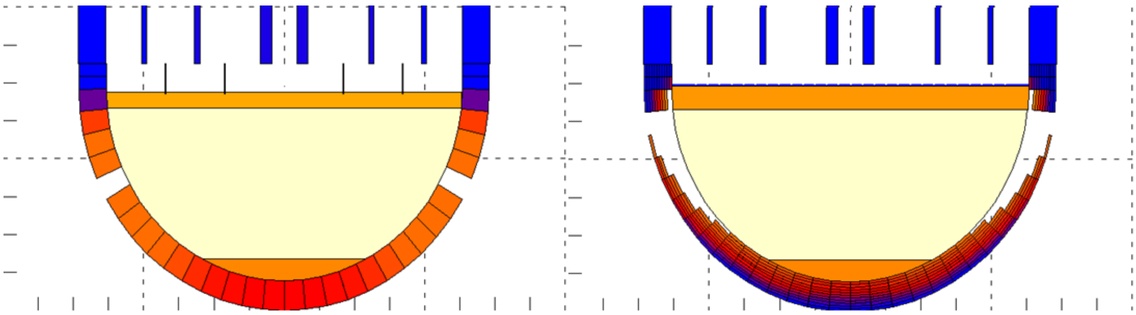


Figure 14: Radial meshing: 1 mesh vs 11 meshes

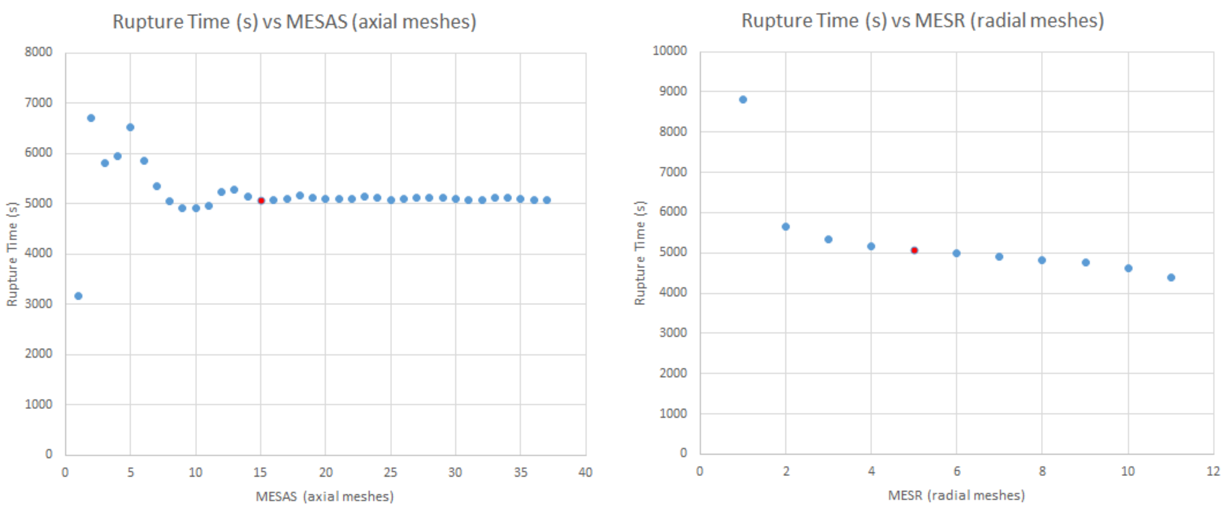


Figure 15: Default values marked with red point

We see from Figure 15 that varying the axial meshes gave us a converging trend and that varying the radial meshes gave us a decreasing trend. There was no significant computational cost when varying the two. Since there was no serious change when varying the axial meshes, we decided to optimize the simulation by maximizing the number of radial meshes while keeping the default number of axial meshes.

4.2.2 Timestep

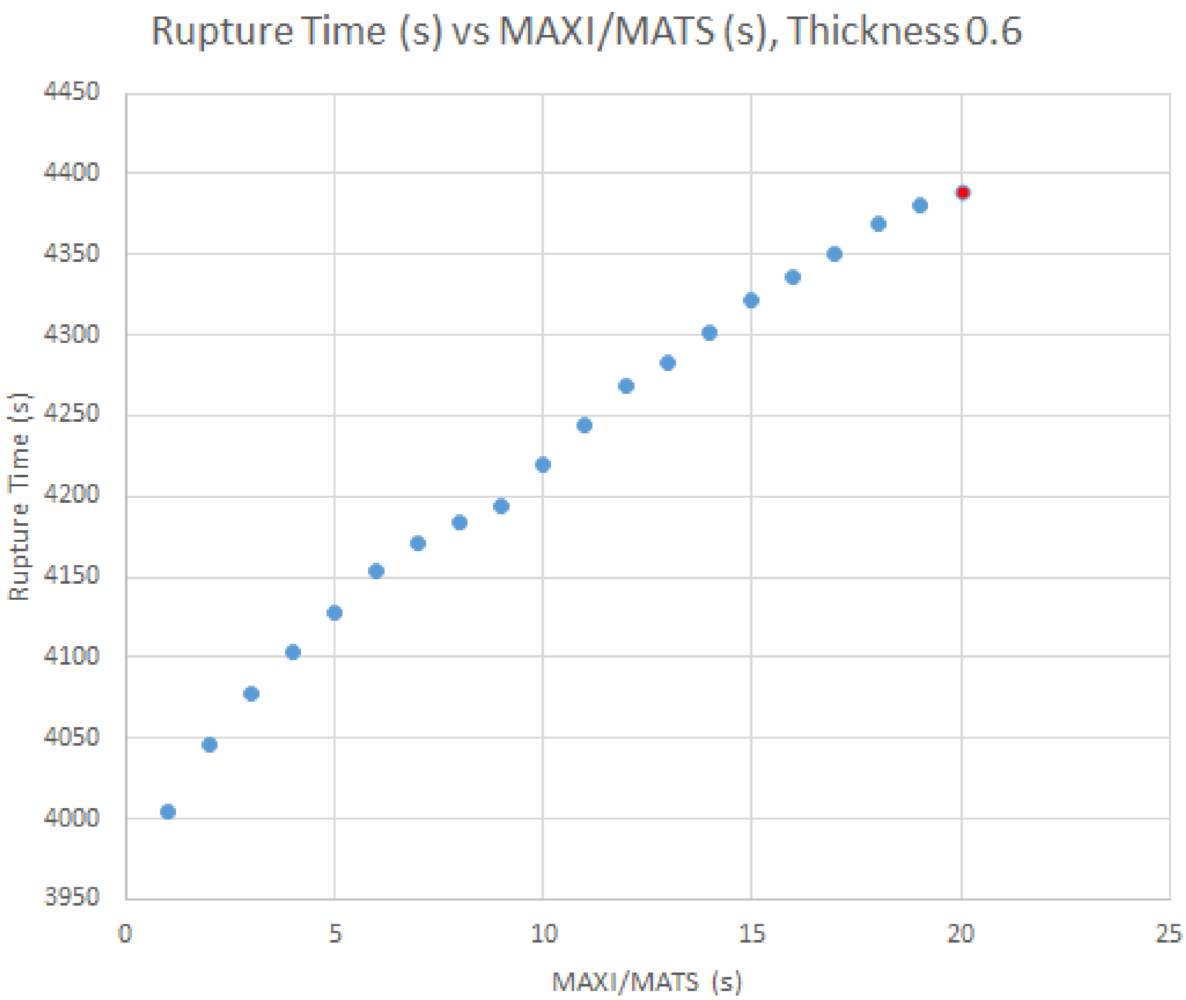


Figure 16: Default value marked by red point

We see from figure 16 that reducing the timestep reduced the rupture time. However, this came at a large computational cost. The simulation, for a timestep of 20 seconds, took a few seconds to run. The simulation, for a timestep of 1 second, took over three hours to run. Therefore, since the relative difference of the rupture times for the timesteps are small, roughly less than 10%, there was no variation made to the timestep in order to save computational cost.

4.3 Application: Total Thickening

After taking the necessary optimization steps mentioned above, we now felt confident to begin applying the simulation to tackle the problem of rupture time. The most straightforward answer would be to increase the thickness of the vessel walls. If there is more material to melt, it should lead to a longer rupture time.

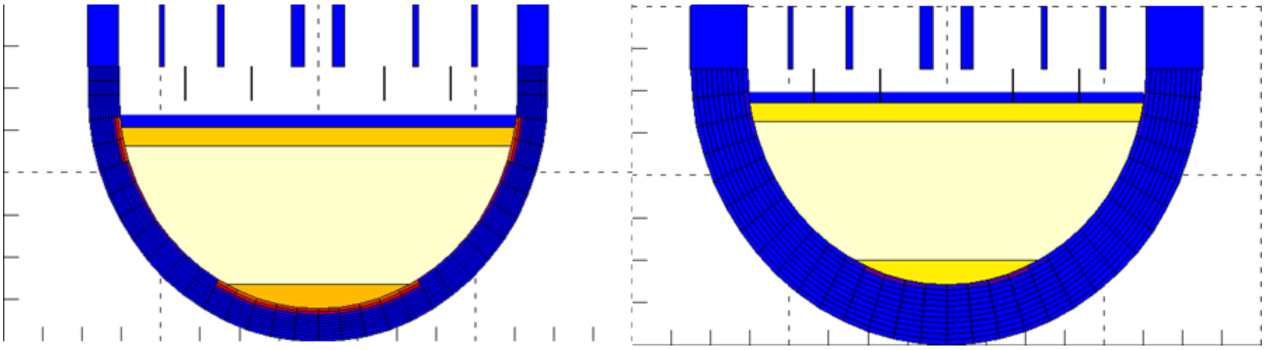


Figure 17: Vessel thickness: 0.3m on the left, 0.6m on the right

The original design of the 900PWR reactor had a 0.3m thickness. We varied this total thickness and recorded the resulting rupture times.

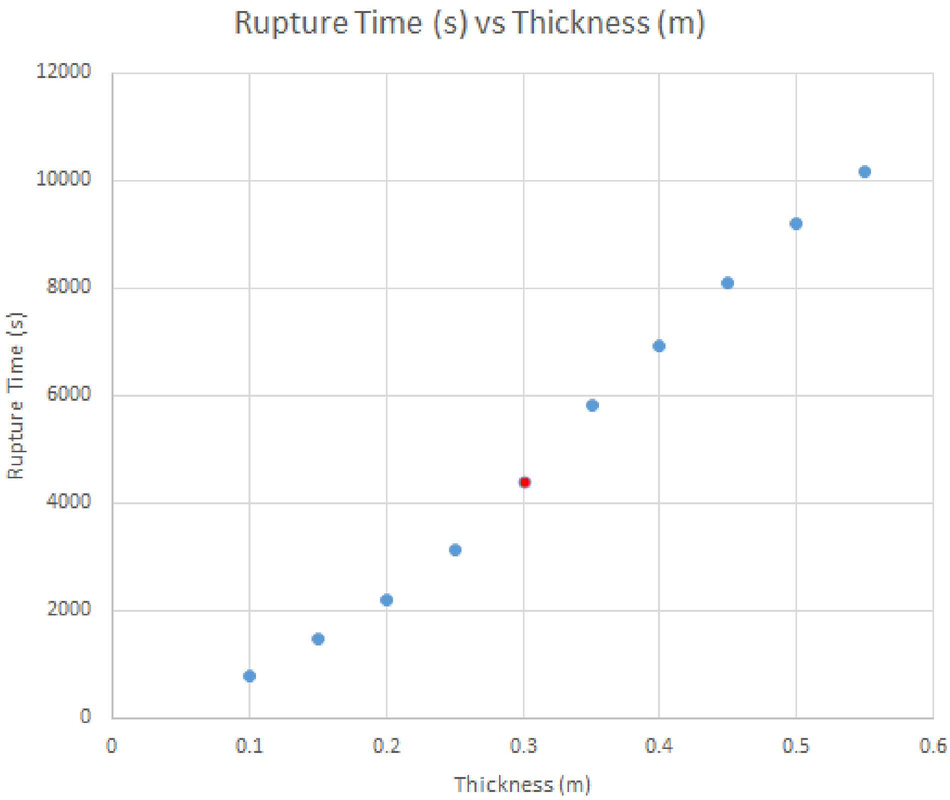


Figure 18: Total thickening plot

We see from figure 18 that there is an increasing trend which is expected. However, we now must ask ourselves whether these results are reasonable? We borrow our lessons from first year physics and recall the heat capacity equation

$$Q = mc\triangle T\,, \tag{5}$$

where Q is the total amount of heat required to change $\triangle T$ temperature of m amount of mass with c specific heat capacity. Since we know the decay heat of the corium inventory (in this simulation, it is set at a fixed value of 20MW), we can work out an approximate upper bound for the rupture time. It is given as

$$t_{\text{rupture}} = \frac{mc\triangle T}{P_{\text{decay}}}\,. \tag{6}$$

We can easily work out the mass of the lower plenum as it is a thick hemisphere. We also obtain the value of the specific heat capacity from the design datasheet of the PWR900 reactor (it is constructed out of a special grade of steel). We also know the initial temperature of the steel and the melting point.

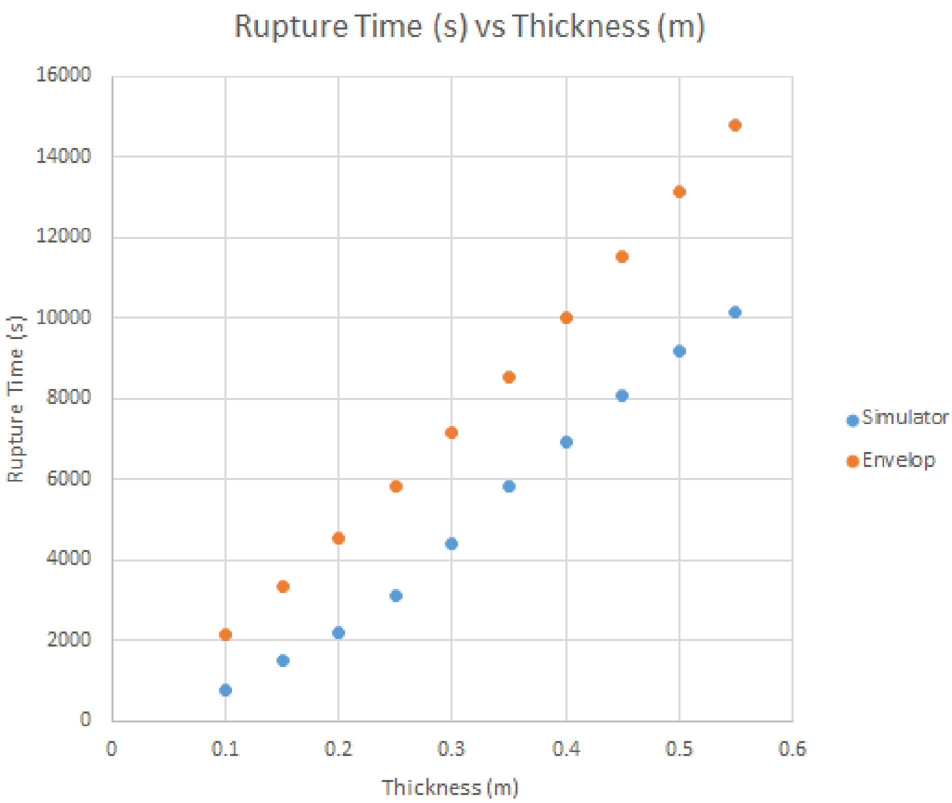


Figure 19: Simulator and back of envelope calculation

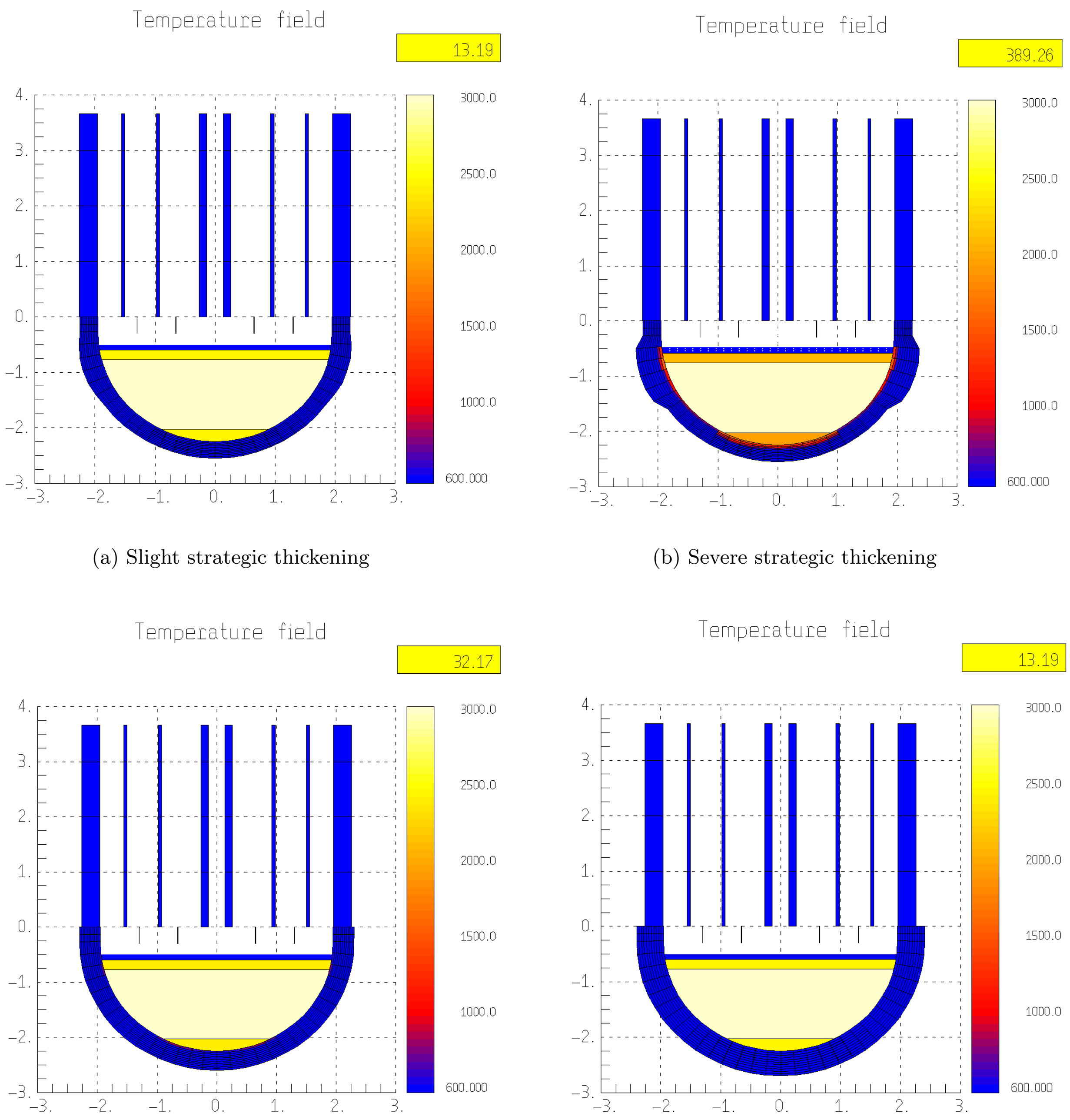
We see from Figure 19 the difference between our simulator results and from the back of the envelope calculation. There is a percentage difference of around 30%. It is roughly in the same order of magnitude although we did not take into account details such as convective or radiative heat transfers. We also assumed the total melting of the entire hemisphere where rupture actually only occurs at a segment. Thus, this calculation shows that the simulation is at least reasonable to predict the necessary thickness to required safety regulations.

However, there are other physical problems encountered when arbitrarily increasing the total thickness, especially during construction of such a thick vessel wall. It is not possible to cast such a thick wall due to the physical behavior of molten materials. [12] Hence, nuclear reactor vessels are welded together through a series of axial and radial welds, though they differ for the various companies. Regardless, thicker and larger vessel introduces more welding errors. Heat creep and deformation also occurs more often for thicker walls.

4.4 Application: Strategic Thickening

Arbitrarily increasing the thickness of the vessel walls is not the solution. The approach now is to thicken walls only at targeted locations where heat flux is the highest. It can be seen from the various simulations that rupture occurs near the top corium layer. This is because the less dense, hotter molten material floats to the top.

Figures 20a and 20b shows two progressing strategic thickening. At their thickest point, they correspond to the 0.35m and 0.45m thickness respectively. We found that strategic thickening rupture times were comparable to total thickening rupture times (Figure 21).



(a) Slight strategic thickening

(b) Severe strategic thickening

(c) Slight total Thickening

(d) Severe total thickening

Figure 20: Strategic thickening vs total thickening

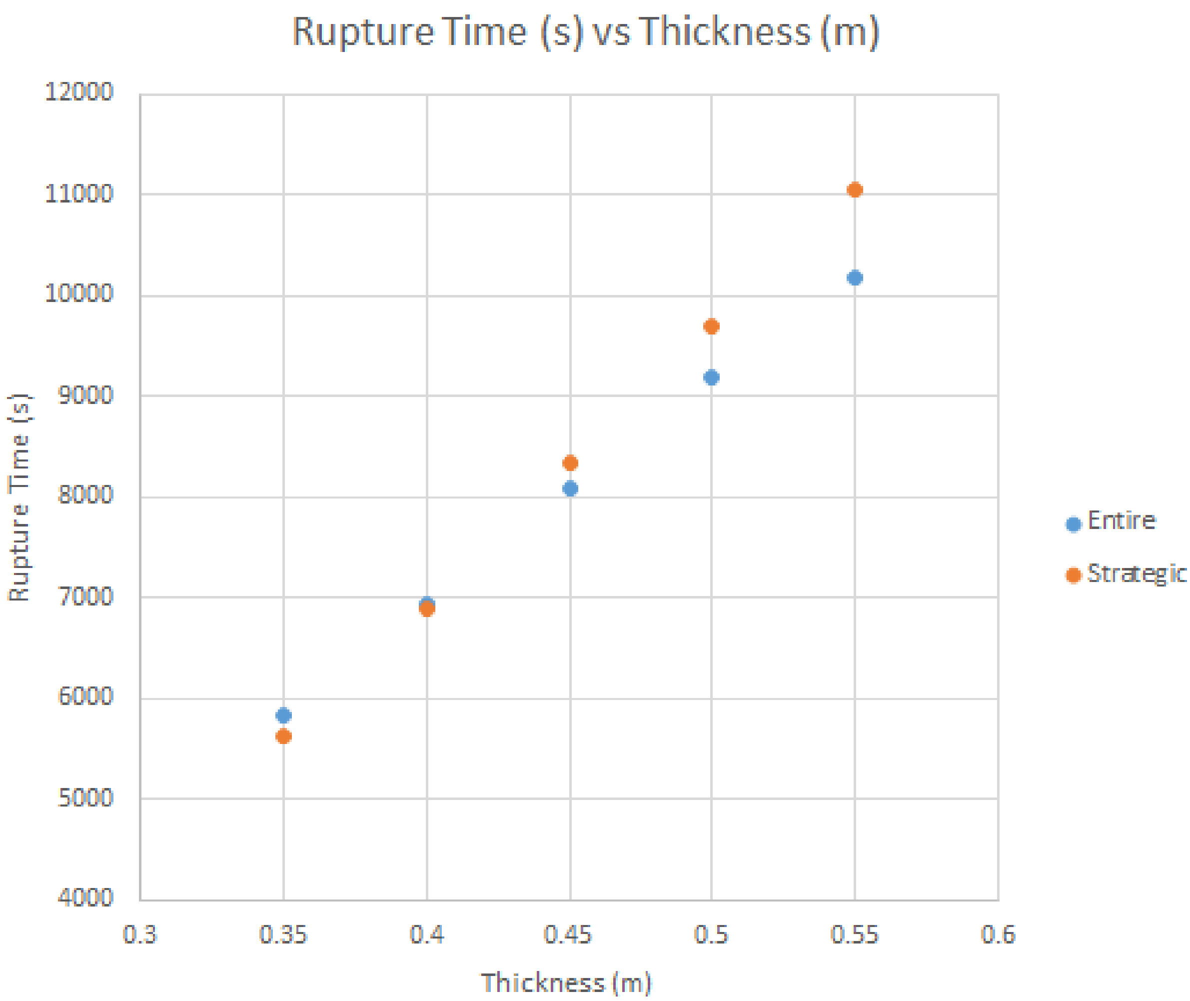


Figure 21: Rupture time vs thickness

This 'paste on' approach could be proposed as an additional safety measure when re-servicing and re-commissioning old reactors. The strategic thickening could be welded onto existing reactor vessels.

The interesting result to note is that strategic thickening seems to begin to perform better as thickness increases. A suggested explanation is that the strategic thickening increases the surface area of that region and radiates heat faster. The total thickening instead will have a smaller surface area to radiate heat but more surface area to conduct (via the top and bottom).

We attempt a ballpark calculation to convince ourselves. The surface area of the lump is given by

$$\pi r \times R \mathrm{d}\phi \rightarrow 2\pi^2 r R = A_{\mathrm{lump}}\,, \tag{7}$$

where we assume the line segment of the lump to be half a small circle of radius r. We then need to multiple by the radial displacement which is given by the small change in azimuthal angle and the radius of the lower plenum R. Without the lump, the line segment would have approximately been $2r$ and thus, the surface area would be

$$2r \times R\mathrm{d}\phi \to 4\pi r R = A_{\text{flat}} \,. \tag{8}$$

We approximate the extra surface area to conduct as half the surface area of two hollow discs with line segment as half thickness of the lower plenum, given by

$$2 \times \frac{\tau}{2} \times R\mathrm{d}\phi \to 2\pi\tau R = A_{\text{discs}} \,. \tag{9}$$

We combine these areas with the Stefan-Boltzmann law and the Fourier's Law and ask ourselves whether

$$A_{\text{lump}}\epsilon\sigma T^4 - A_{\text{flat}}\epsilon\sigma T^4 > kA_{\text{discs}}\frac{\triangle T}{\triangle x} \,? \tag{10}$$

We use $r = \frac{\tau}{2}$ for ease and use $\tau = 0.6$ and pluck the constants from the material database. R is roughly 2.25m. The thermal conductivity k is roughly 30. Assuming the temperature gradient ($\triangle T = 100\,\mathrm{K}$) across the axial elements ($\triangle x = 0.25\,\mathrm{m}$) and that the average temperature for the radiation is 1000 K, we find that the radiative heat flow is roughly 2.7 times the conductive heat flow at this thickness. Therefore, it seems likely that at thicker strategic thickening, radiative effects become more dominant.

4.5 Application: Ex-vessel Cooling

We can also use ex-vessel cooling since in-vessel cooling is more difficult due to instantaneous vaporization. Another disadvantage of in-vessel cooling is because steam vapor oxidizes Zirconium (found in the fuel assemblies). This generates significant heat and hydrogen.

The main idea of ex-vessel cooling is essentially spraying coolant on the vessel outside walls. The question we now ask ourselves is how much cooling do we actually need? Using the same values from the earlier back of the envelop calculation, we are able to approximate an upper bound for the cooling needed. We have the total decay heat and the mass of the lower plenum. If we make the same assumptions that all the decay heat transfers to the lower plenum uniformly, we find that the mass is being heated up at a rate of 300 W/kg.

$$\frac{P_{\text{decay}}}{m_{\text{lower plenum}}} \approx \frac{20 \text{ MW}}{66000 \text{ tons}} = 300 \text{ W/kg} \tag{11}$$

Therefore, if I plot rupture time against heat removal per unit mass, I expect to hit this asymptotic (because it will no longer rupture) rupture time as I approach this value.

The original optimized design of the simulation was used with no thickening at all (0.3 m thick). The only difference now is that the walls of the lower plenum are being cooled at a fixed, uniform rate.

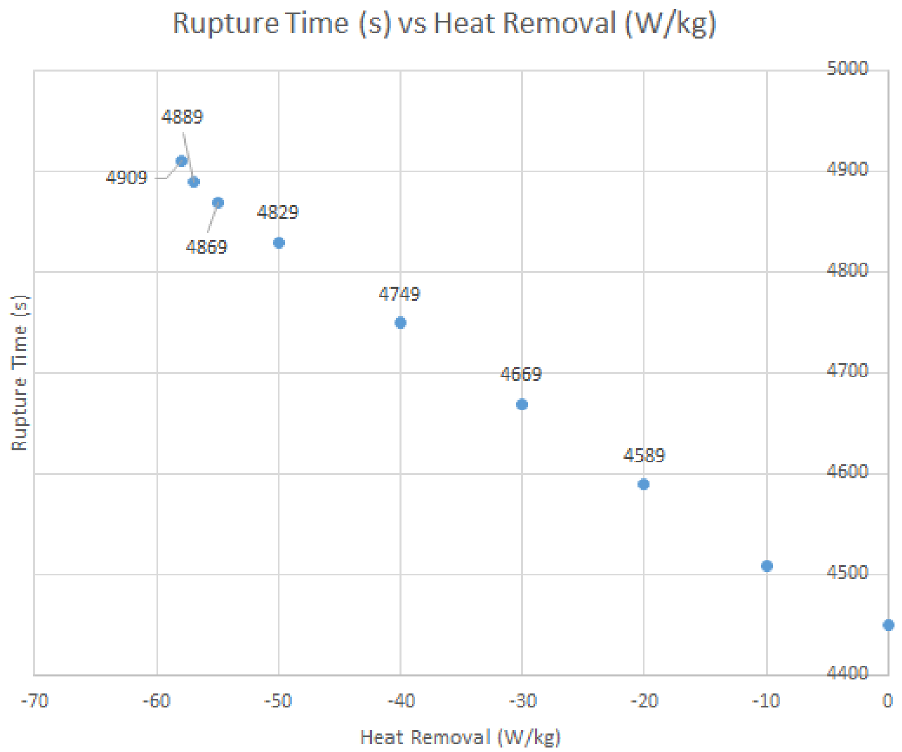


Figure 22: Lower plenum being cooled at a constant rate

We see that increasing the heat removal increases the rupture time (Figure 23). This is expected but not guaranteed as the cooling might have introduced mechanical stress. Unfortunately, due to the coding limitations of ASTEC, we were unable to use greater values of heat removal. We are unable to approach or observe asymptotic behavior since we are still quite far from our approximate upper bound for the cooling required to prevent rupture.

4.6 Application: Corium Levels

The alternative approach is then to vary the amount of corium mass in the lower plenum. Throughout the various simulations, the maximum (80 tons) corium mass was used. This is the maximum amount of corium due to the inventory of the nuclear reactor by design.

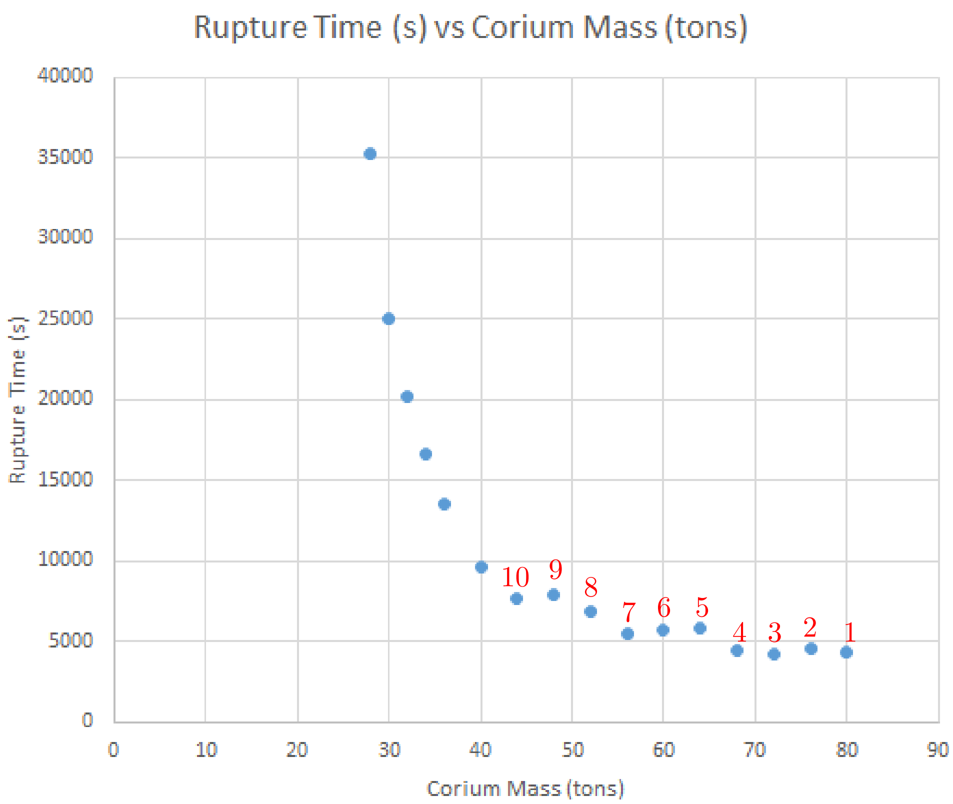
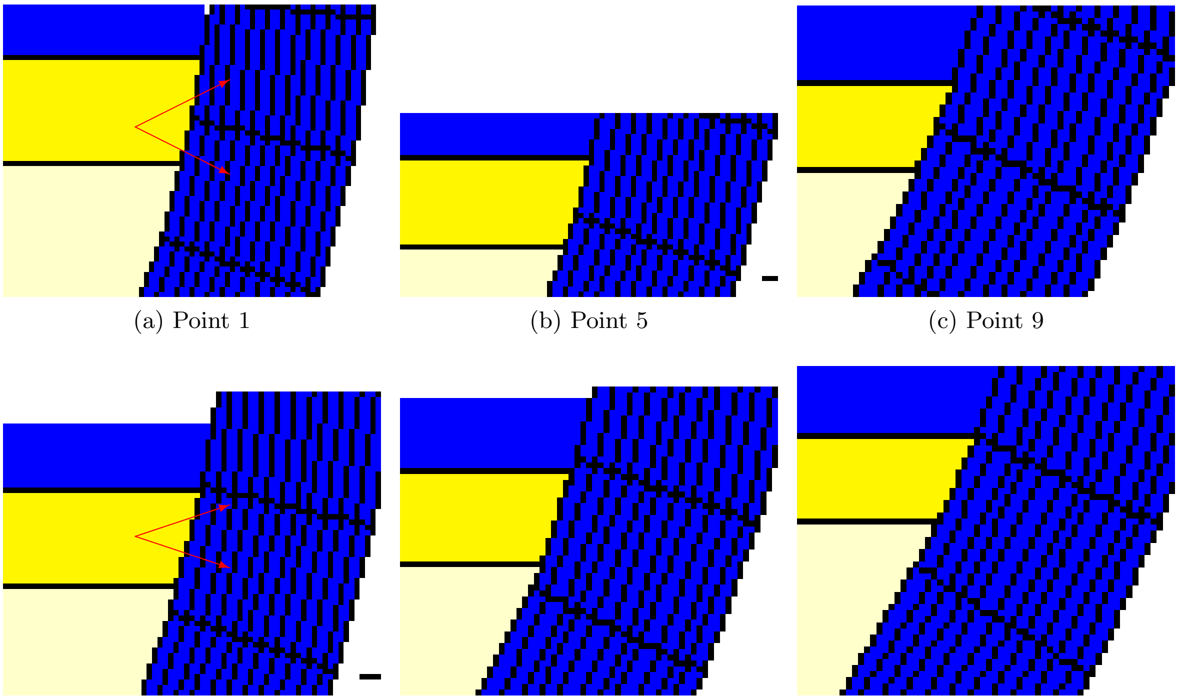


Figure 23: Points are numbered from right to left

We see from Figure 23 that reducing the corium mass increases the rupture time. It also displays asymptotic rupture time behavior as expected. Less corium mass equates to lesser decay heat, therefore, rupture time increases. However, we expected that the change in rupture time with respect to corium mass would always be a negative value. Instead, we find some 'noisy' points in the flatter region where it is sometimes positive.

This is most likely caused by the axial meshing of the simulation. The top most corium layer has the highest heat flux and is most responsible for melting the steel. If the axial meshes are larger than the corium level changes, rupture time is 'slowed down' because of the location of the top most layer. If the layer is located between two axial elements, it needs to transfer heat to both in order to melt them (Figures 24a, 24b, 24c). However, if the layer is located in one axial element, it only needs to transfer heat to that element and melting occurs faster (Figures 24d, 24e, 24f). For this scenario, we are dealing with small changes in the corium level. The top most layer is sometimes located between two axial elements and sometimes located in one axial element. In order to reduce this problem, we need to increase the number of axial meshes. For this scenario, axial meshing is more important than radial meshing.



(a) Point 1 (b) Point 5 (c) Point 9

(d) Point 3 (e) Point 7 (f) Point 10

Figure 24: (a),(b),(c) shows the high rupture time due to double contact. (e),(d),(f) shows the low rupture time due to single contact.

Therefore, we reduced the number of radial meshes and maximized the number of axial meshes (Figure 25). By making the axial meshes smaller than top most corium layer, we smoothen out the rupture times (Figure 26).

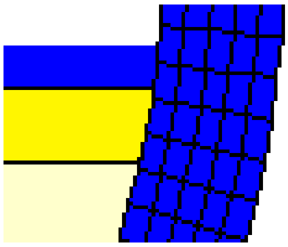


Figure 25: More axial meshes rather than radial meshes

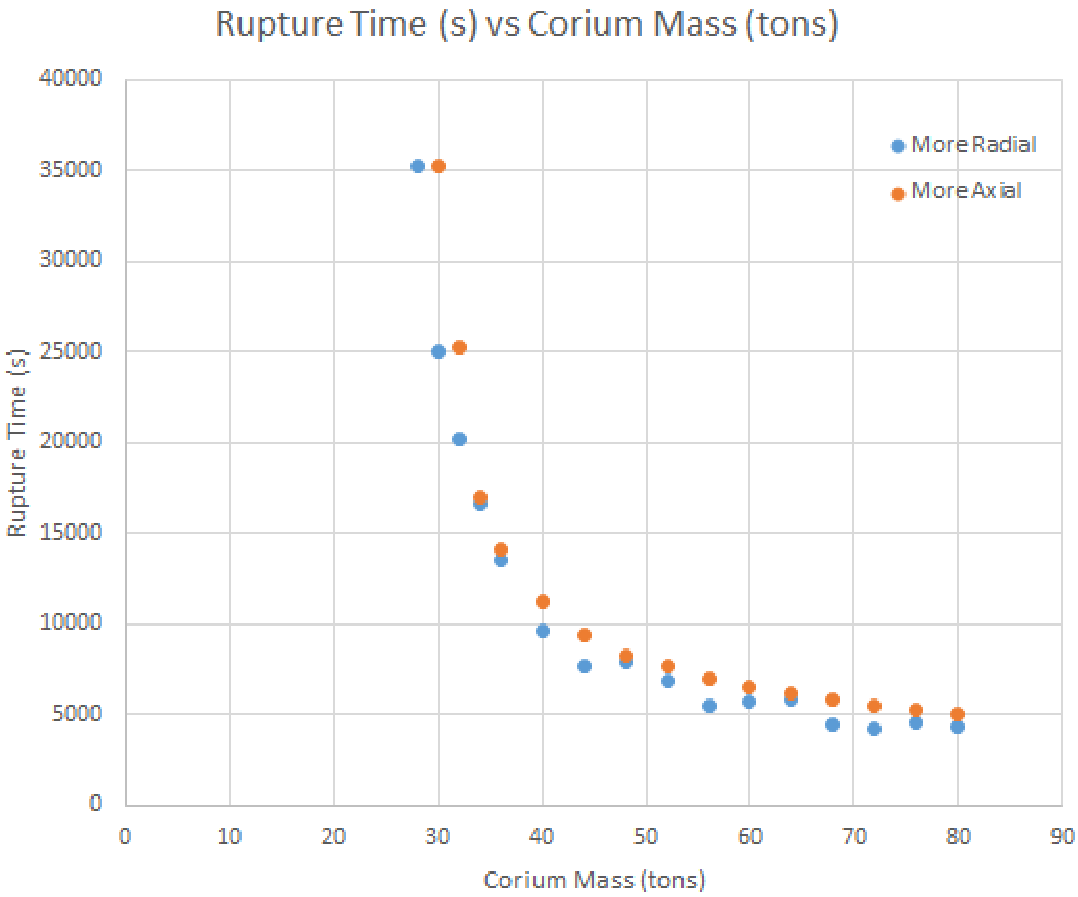


Figure 26: Rupture time vs corium mass, axial to radial mesh comparison

Even for simplified simulations, we realize that we have to be careful in choosing the meshing elements to suit the specific phenomena we want to study.

5 Concluding Remarks

We have achieved the objectives we set out at the beginning of this report. We used ASTEC to simulate a 900PWR reactor and showed that results were reasonable to known physics and eventually tested simple improvement ideas to prolong the rupture time.

Under the assumption that smaller timesteps and more element meshes gives a more realistic result, we found that the effective method to conduct the simulation was to fully utilize the radial meshes. Smaller timesteps did not invoke a significant difference in result even though it came at a huge computational cost.

We found that even though total thickening of the pressure vessel increases rupture time, it is inefficient when compared to strategic thickening as one uses significantly less material for a better result. Overall, the rupture times are reasonable as they are within the same order of magnitude as a result from a simple upper bound approximation.

We found that low levels of ex-vessel cooling gave the expected result and prolonged rupture time. Asymptotic rupture time was not found due to simulation limitations in the code.

We investigated that reducing the corium mass revealed asymptotic rupture time. The 'noisy' data was smoothened via alterations to the axial/radial meshing optimization. We appreciate that a one size fits all approach does not work for element meshing in specific phenomena.

5.1 Limitations

The progression of this project revealed several shortcomings and limitations of ICARE/ASTEC.

1. The maximum number of meshes allowed in the lower plenum is too small. Improving this aspect will make the simulation more robust and flexible.

2. The material properties of the lower plenum (for example the steel walls) are treated as one object in the code. This prevents the code from performing localized cooling and hence, made studying ex-vessel cooling difficult.

3. The export of data is extremely tedious and troublesome. The lack of direct data export of values to text files or graphs to portable graphics files made gathering data laborious and time-wasting.

5.2 Future Work

ASTEC V2.0 had been used through the course of this project. However, newer versions of ASTEC will be released in the months following the conclusion of this project. ASTEC V2.1 will have numerous corrections and upgrades to the pre-existing physical models. The improvements related to this project are:

1. Models to deal with upwards radiative exchanges in the lower part of the vessel after corium slumping into the lower head. Adding this model will most likely vary the results found in this project.

2. Generalisation of the material database for all ASTEC modules to use. This database allows the user to alter the materials called upon in the code. Currently, the user can only call from a small, fixed pool of materials. Therefore, one cannot test different types of steel.

3. Improvement of lower plenum modeling.

Possible continuation of the work completed in this project are:

1. Adapting the improved designs onto the full PWR900 simulator. The user would be able study the designs when more complicated phenomena such as re-flooding come into play.

2. If rupture cannot be avoided, studying the Molten Corium Concrete Interaction module (MCCI) is recommended as the next step for severe accident mitigation. Newer nuclear power plants rely on this as a defense in depth measure. For example, the European Pressurized Reactor's (EPR) core catcher system.

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A My Main Modified Simulator Code

For whomever it may help:

> “Code never lies, comments sometimes do”

Please read the documentation on the syntax if it confuses you. Most of the changes I have made should be straightforward to someone who understands the syntax. The comments have been made clearer by bolding the text. Parts of the code have also been ‘switched off’ by turning them into comments.

```
(namedata="tcshp1")
(compute=GETENV(computer))
(compile=GETENV(compiler))
!***********************************************************************
!                  ICARE STAND-ALONE CALCULATION
!                   --------------------------
!
!                     SIMPLIFIED LOWER PLENUM
!
!  Simulation du comportmeent du corium en fond de cuve
!  dans le but d’améliorer la modélisation actuelle
!
!  Scenario Haute Pression 1.1.1 de l’analyse EPS-2
!***********************************************************************

TITL "ASTEC V2 - TCSHP1"

!=======================================================================
! DATA BASE SAVING
!=======================================================================
!STRU SAVE FILE "pleasework.bin" FREQ 100. END
```

!This generates a bin file which can be used via postprocessing

```
!=======================================================================
! ON-LINE VISUALIZATION
!=======================================================================
```

```
!CALL "core_temperature_field.visu"
CALL "lower_head_corium_mass.visu"
CALL "menu.visu"

STRU VISU !This is used to construct a nicer axis
FILE  "core_temperature_field.vis"
CALL commun.com
BOUN 'YES'
TITL "Temperature field"
(palette=3)
CALL temperature.pal
XMIN -3.0 XMAX 3.0 XDEC 6
YMIN -3.0 YMAX 4.0  YDEC 7
END

!========================================================================!
! PLOT STRUCTURE FOR POST-PROCESSING
!========================================================================!
CALL "maint_icare.tool"
!This generates tcshp1˙icare˙windows.plot which gives some numbers such
! as the rupture time but not magma5 for some reason.

!=======================================================================
! GENERICS STRUCTURES ASTEC
!=======================================================================
(P0 = 150.D5)
(tgas = PROPERTY('STEAM','Tsat',582.78,1.5506D+07))
(tliq = 573.)

STRU SEQUENCE
TINI 0.
TIMA 300000. !To control the total time of simulation
TSTOP 'TRUP'
STRU MACR DTFI 1. MINI 1. MAXI 20. END !To control the timestep
END

STRU CALC_OPT
```

```
SC1 MODULIST 'ICARE' TERM
STRU ICARE MITS 1. MATS 20. END !To control the timestep
END

!=========================================================================
! BOUNDARY CONDITIONS
!=========================================================================

STRU CONNECTI
NAME 'CORE_OUT' TYPE 'BREAK'
FROM 'VESSEL' TO   'USER'
STAT 'ON'
MACR VAP1 MACR VAP2 MACR VAP3
MACR LIQ1 MACR LIQ2 MACR LIQ3
SR1 P 0. (P0) 3000. (P0) TERM
END
STRU CONNECTI
FROM 'VESSEL' TO 'USER'
TYPE 'BCTI'
STAT 'ON'
MACR 'VESSEL'
FACE 'EXTERNAL'
INST 0.
SR1 Z      0.   4. TERM
SR1 TIMP 350. 350. TERM
SR1 H    200. 200. TERM
END
STRU CONNECTI
FROM 'VESSEL' TO 'USER'
TYPE 'BCTI'
STAT 'ON'
MACR 'LOWER'
FACE 'EXTERNAL'
INST 0.
SR1 Z     -2.5  0. TERM
SR1 TIMP 350.  350. TERM
SR1 H    200.  200. TERM
```

```
END

!=========================================================================
! ICARE DATA BASE
!=========================================================================

STRU VESSEL

!--- options icare

STRU OPTI ARRA 'SQUARE' ENBA 'NO' UZRO 'MASSFRAC' PITC 1.26D-2 END

STRU NUME DELT 500. EPSI 1.D-3 END

!--- meshing

STRU DISC
SR1 AXIA 0. 1. 2. 3. 3.6576 TERM
SR1 RADI 0. 0.4 1.5 2.3 TERM
SC1 ROD  'FUEL1'  'CLAD1'  TERM
SC1 ROD  'FUEL2'  'CLAD2'  TERM
SC1 SHRO 'BAFFLE' TERM
SC1 SHRO 'VESSEL' TERM
STRU LOWE ZMIN -2.6 ZMAX 0. R 2.5 END
END

!--- physical objects definition

STRU MACR
NAME 'FUEL1' TYPG 'CYLINDER' !There are 4 sets of fuel plus cladding from center
ZMIN 0.      ZMAX 3.6576
DINT 0.      DEXT 8.D-3
WEIG 2376    R 0.2                 !WEIG 2376, there are 2376 of them
MATE 'UO2'   MASF 1.
PORO 0.      T 500.
END
```

```
STRU MACR NAME 'FUEL2' COPY 'FUEL1' WEIG 5280 R 0.8 END !WEIG 5280

STRU MACR
NAME 'CLAD1' TYPG 'CYLINDER'
ZMIN 0.      ZMAX 3.6576
DINT 9.D-3   DEXT 10.D-3
WEIG 2376    R 0.2                  !WEIG 2376
MATE 'ZRO2'  MASF 1.
PORO 0.      T 500.
END

STRU MACR NAME 'CLAD2' COPY 'CLAD1' WEIG 5280 R 0.8 END !WEIG 5280

STRU MACR
NAME 'BAFFLE' TYPG 'CYLINDER' !1st from the Outer Cylinder, graphically
ZMIN 0.       ZMAX 3.6576
DINT 3.0      DEXT 3.1
MATE 'STEEL'  MASF 1.
PORO 0.       T 500.
END

STRU MACR
NAME 'VESSEL' TYPG 'CYLINDER' !Outer Cylinder
ZMIN 0.       ZMAX 4.
DINT 3.9      DEXT 4.5
MATE 'STEEL'  MASF 1.
PORO 0.       T 500.
END

STRU MACR
NAME 'LOWER'
TYPG 'LOWERPLE' !Lower Plenum
TPLATE 500.
THERMEQ 'YES'
EQLAYER 2
!==============================================================
!This portion was used to manually build the lower plenum
```

```
!==============================================================
SR1 RINT 0. 0.206313 0.41031 0.609702 0.802249 0.98579 1.158265 1.317739 1.46242 1.590685 1.701093 1.7
SR1 ZINT -2.25 -2.23906 -2.20634 -2.15223 -2.07733 -1.98247 -1.86873 -1.73738 -1.58989 -1.42793 -1.253
SR1 THIC 0.3 0.3 0.3 0.3 0.3 0.3 0.3 0.3 0.3 0.3 0.3 0.3 0.3 0.3 0.3 0.3 0.3 0.3 TERM
MESR 11  !MESR = number of radial meshes of the lower head vessel
!==============================================================
!DINT 3.9      DEXT 4.5        !To control the thickness
!MESAS 15 !MESAC 2 !MESR 11 !MESAS = number of axial meshes
MATE 'STEEL' MASF 1.
T 500.
STRU LAYER !To control the corium levels, ensure mass ratio is constant 2:80:14
MASS 2.0D3 T 2500. !MAGMA1 Legend Item 2 on the legend of graph
MATE 'FE' MASF 1.
END
STRU LAYER
MASS 80.D3 T 3200. !MAGMA2 Legend Item 4
MATE 'UO2' MASF 1.
END
STRU LAYER
MASS 14.0D3 T 2500. !MAGMA3 Legend Item 5
MATE 'FE'  MASF 1.
END
STRU LSTRUCT !determines the penetration tubes in the lower plenum
ZMIN -0.3 ZMAX 0. !these things are the weird lines you see right above the water layer
NTUB 40   PITCH 0.001 !below and near the baffle and second set of fuel rods
DINT 0.   DEXT 0.005
MATE 'STEEL'  MASF 1.
T 573.
END
END

STRU MACR
NAME 'VAPO' TYPG 'FLUIDGAS' LOWE 'YES'
ZMIN -2.5 ZMAX 0.
GAS 'H2O' PP (P0) T (tgas) P (P0)
END
```

```
STRU MACR
NAME 'VAP1' TYPG 'FLUIDGAS'
R 0.2 ZMIN 0. ZMAX 3.6576
GAS 'H2O' PP (P0) T (tgas) P (P0)
END

STRU MACR NAME 'VAP2' COPY 'VAP1' R 0.95 END
STRU MACR NAME 'VAP3' COPY 'VAP1' R 1.75 END

STRU MACR
NAME 'LIQ0' TYPG 'FLUIDLIQ' LOWE 'YES' VELO 'LOOP'
ZMIN -2.5 ZMAX 0.
T (tliq) P (P0) VL 5.3 LEVEL -2.2
END

STRU MACR
NAME 'LIQ1' TYPG 'FLUIDLIQ'
R 0.2 ZMIN 0. ZMAX 3.6576
T (tliq) P (P0) VL 5.3 LEVEL 0.1
END

STRU MACR NAME 'LIQ2' COPY 'LIQ1' R 0.95 END
STRU MACR NAME 'LIQ3' COPY 'LIQ1' R 1.75 END

!--- residual power

STRU POWE
NAME 'CORI' TYPE 'MASS'
!  t1 t2 ....tn V1 V2 ...Vn
SR1 AMPL 0. 35000. 250. 250. TERM
SR1 PROF -2.55 0. 1. 1. TERM
MATE 'UO2'
MACR 'LOWER' FACT 1.
END

!   STRU POWE !TURN THIS ON TO HEAT UP PARTS OF THE BAFFLE
!      NAME 'COOLBAFF' TYPE 'TOTAL'
```

```
!  t1 t2 ....tn V1 V2 ...Vn
!      SR1 AMPL 0. 35000. 250.D5 250.D5 TERM
!      SR1 PROF 1. 3.0 1. 1. TERM
!      MATE 'STEEL'
!      MACR 'BAFFLE' FACT 1.
!   END !PROOF OF PRINCIPLE THAT LOCALISED HEATING/COOLING WORKS

STRU POWE !COOLING THE LOWER PLENUM
NAME 'COOL' TYPE 'MASS'
!  t1 t2 ....tn V1 V2 ...Vn
SR1 AMPL 0. 35000. -0 -0  TERM
SR1 PROF -2.0 -1.0 1 1 TERM !ATTEMPT AT LOCALISED COOLING. IT FAILED. IT COOLS ENTIRELY INSTEAD
MATE 'STEEL'
MACR 'LOWER' FACT 1.
END

!   STRU POWE !ATTEMPT AT USING A IF LOOP FOR COOLING TO AVOID THE SIMULATION ERROR T smaller than 0K
!      NAME 'COOLTEST' TYPE 'MASS'
!  t1 t2 ....tn V1 V2 ...Vn
!      SR1 AMPL 0. 35000.
!#begin INST IF (T>=600) -10 ELSE -0 #end
!#begin INST IF (T>=600) -10 ELSE (T<600) -0 #end TERM  !
!      SR1 PROF -0.5 3.0 1 1 TERM
!      MATE 'STEEL'
!      MACR 'LOWER' FACT 1.
!   END !THE ATTEMPT FAILED

STRU POWE
NAME 'RESI' TYPE 'TOTAL'
SR1 AMPL 0. 3000. 4.D5 4.D5 TERM
SR1 PROF 0. 3.6 1. 1. TERM
MATE 'UO2'
MACR 'FUEL1' FACT 1. MACR 'CLAD1'  FACT 1.
MACR 'FUEL2' FACT 1. MACR 'CLAD2'  FACT 1.
END
```

```
!--- thermal exchanges

STRU CONVLOWE FLUI VAP0 FLUI LIQ0 END

STRU CONV CONF BUNDLE
FLUIGEN GASPHASE   MACRGEN ROD    FACE EXTERNAL
ZMIN 0. ZMAX 3.6576
END
STRU CONV CONF BUNDLE
FLUIGEN LIQPHASE   MACRGEN ROD    FACE EXTERNAL
ZMIN 0. ZMAX 3.6576
END
STRU CONV CONF PIPE
FLUIGEN GASPHASE   MACRGEN SHROUD FACE INTERNAL
ZMIN 0. ZMAX 3.6576
END
STRU CONV CONF PIPE
FLUIGEN LIQPHASE   MACRGEN SHROUD FACE INTERNAL
ZMIN 0. ZMAX 3.6576
END
STRU CONV CONF PIPE
FLUIGEN GASPHASE   MACRGEN SHROUD FACE EXTERNAL
ZMIN 0. ZMAX 3.6576
END
STRU CONV CONF PIPE
FLUIGEN LIQPHASE   MACRGEN SHROUD FACE EXTERNAL
ZMIN 0. ZMAX 3.6576
END

STRU COND MACRGEN ROD     MACRGEN ROD
FACE    UP      FACE DOWN   END

STRU COND MACRGEN SHROUD  MACRGEN SHROUD
FACE    UP      FACE DOWN   END

STRU COND
```

```
MACR LOWER
MACR LOWER
END

STRU COND
MACR VESSEL FACE DOWN
MACR LOWER  FACE UP
END

STRU COND
MACR FUEL1
FACE EXTERNAL
MACR CLAD1
FACE INTERNAL
END

STRU COND
MACR FUEL2
FACE EXTERNAL
MACR CLAD2
FACE INTERNAL
END

STRU EXCHLOWE END

!--- degradation

STRU DECALOWE END

STRU DECA
SC1 FROM FUEL1 TERM
SC1 TOWA CLAD1 TERM
SC1 FACE EXTERNAL TERM
DIRE EXTERNAL
END

STRU DECA
```

```
SC1 FROM FUEL2 TERM
SC1 TOWA CLAD2 TERM
SC1 FACE EXTERNAL TERM
DIRE EXTERNAL
END

STRU CCYL MACRGEN 'ROD_FUEL' MACRGEN 'ROD_FUEL' END

STRU CAND
MACR CLAD2
FACE EXTERNAL
BLOC 1.
LIQF 0.
END

STRU CAND
MACR CLAD1
FACE EXTERNAL
BLOC 1.
LIQF 0.
END

STRU RUPTURE END

END

!=========================================================================
! THE END
!=========================================================================
```