Investigating Hydrogen Production and Vessel Rupture Time When Reflooding Degraded Nuclear Reactor Cores

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Abstract

As the world turns increasingly towards nuclear power, nuclear safety research becomes incredibly pertinent. One of the accident scenarios studied is the loss-of-coolant accident (LOCA). In mitigating the consequences of LOCA, reflooding the core is commonly employed. Despite being an effective measure at delaying the rupture time of the vessel, hydrogen is also produced in the process. This presents a possible hydrogen combustion risk when the hydrogen escapes into the containment. Therefore, it is noteworthy to investigate if varying the reflood parameters (reflood time, duration and rate) will affect the amount of hydrogen produced, and if there are a set of possible values by which hydrogen produced is minimised and vessel rupture time is maximised.

This project uses Accident Source Term Evaluation Code (ASTEC) to run a testcase of the 900 MWe vessel. By varying the reflood time, duration and rates, results on the total hydrogen produced and the time of vessel rupture were collected. Results seem to show that there is an optimal time to reflood, but they are inconclusive with regards to reflood rate and duration. However, as ASTEC still needs work on reflooding a degraded core and there are limited experiments to compare the simulation results with, the results should be treated with care in regards to how well they actually represent real world phenomena.
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It’s not where you start; it’s where you finish. – Seesaw (1973)
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1 Introduction

As of March 2016, there are 442 nuclear reactors generating electricity in 30 countries and 66 new nuclear plants being constructed in 15 countries [1]. As nuclear reactors continue to rise in popularity, nuclear safety research becomes exceedingly important since severe accidents involving nuclear reactors may lead to devastating environmental consequences.

There are many aspects to nuclear safety research and one of them involves studying accident scenarios. One of these accident scenarios is known as LOCA. In the absence of a coolant, the core would begin to heat up and eventually degrade. With enough time, the vessel may eventually rupture. Therefore, there are several safety systems built into the nuclear power plant to help cope with LOCA, and injecting the coolant back into the core is a strategy that is commonly used. This is known as reflood.

While reflood does help in delaying the vessel rupture time and sometimes even prevent the vessel from rupturing, the steam produced from reflooding would react with the hot metals in the core to produce heat and hydrogen. The heat produced would exacerbate the core heat up and further degrade the core, while the hydrogen presents a combustion risk if it leaks into the containment which houses the vessel.

As a means to reduce hydrogen risk and delay vessel rupture, it is therefore worth taking a look at if varying the reflood parameters (reflood time, duration and rate) would have a significant impact on the amount of hydrogen produced and the vessel rupture time. In the course of the investigation, it is also noteworthy to investigate possible values in which reflood is optimised, thereby leading to the minimum amount of hydrogen produced and the maximum vessel rupture time.

Therefore, the objectives of this project are:

- To understand how ASTEC operates
- To be able to explain or propose possible justifications for the simulation results
- To use ASTEC to determine how changing the reflood parameters will affect hydrogen production and vessel rupture time
2 The Nuclear Power Plant

2.1 Background

Since the inception of the first commercial nuclear power plant in the 1950s, nuclear reactor technology has been constantly evolving [2]. Figure 2.1 gives an overview of how the nuclear reactors have developed over the years. The first generation of reactors, so called the generation I nuclear systems, included various designs that incorporated gas cooling with graphite moderation or prototype water cooling and moderation. Most of the current designs used today belong to Generation II. It includes the light water pressurized water reactors (PWRs) and the light water boiling water reactors (BWRs). Generation III reactors, of which some of them are currently being built and some are already operational, improve on designs from generation II, featuring improved performance, extended design lifetimes and are better adapted to cope with severe accidents. Generation IV reactors employ different designs and are currently under development by the Generation IV International Forum (GIF) [3]. Their designs include thermal and fast neutron spectra cores, closed and open fuel cycles. The first generation IV systems are expected to be deployed between 2030 and 2040.

For this project, light-water PWRs were studied.
2.2 Pressurised Water Reactor (PWR)

![Diagram of PWR](image)

*Figure 2.2: Layout of the PWR [4]*

Figure 2.2 shows a schematic layout of the PWR. Sustained nuclear fission reaction occurs in the core (1) and this produces heat. The reactor core is located within the reactor vessel (3), which is part of the primary loop (in red). The water in the vessel gets heated up by the core and is driven by the reactor coolant pumps (6) to the steam generators (5), transferring the heat from the primary loop to the steam generators. The water from the primary loop will not mix with the water at the steam generator. After which, the water is pumped back into the vessel and the process repeats. As the water is at a high temperature (about 300°C), high pressure (about 155 bar) is necessary to keep the water in liquid state [5]. This is done by a pressurizer (4). The steam generator is part of the secondary loop (in green and blue) and steam is created when it absorbs heat from the primary loop. The steam is transferred out of the containment (7) to the turbine (8), which transforms the steam’s heat energy into mechanical energy. The turbine drives the generator (9) and produces electricity. After the turbine, the steam is converted back into liquid form by the condenser (10) and carried back to the steam generators by the condensate extraction pumps (11). The process then repeats for the secondary loop.
In this example, water is the coolant being described. However, there are other possible substances that can act as coolant and these include molten metals and gases. Water can also act as a moderator for the core. Moderators slow down neutrons to enable sustainable nuclear fission reactions. Other types of moderators include graphite and heavy water. For this project, the reactors studied use water as both coolant and moderator.

2.3 Severe Accidents

A severe accident or core melt accident at a PWR is an accident in which extensive melting of the reactor core occurs, severely damaging the fuel rods in the process. This melting is due to the significant increase in temperature of the fuel rods when there is a loss of core cooling. In 1979, due to a series of failures, the second reactor at the Three Mile Island (TMI-2) nuclear power plant experienced a core melt accident. Fortunately, as both the vessel and the containment in TMI-2 were not breached, there were no significant environmental consequences. In general, if the core degradation within the reactor vessel cannot be stopped, a core melt accident may eventually lead to the loss of containment integrity, which will result in large releases of radioactivity into the environment. Due to the severity of radioactive releases, much effort is made to study severe accidents and to mitigate their consequences.

The various accident scenarios which may result in core melt include [5]:

- Loss-of-coolant accidents (LOCA)
- Loss-of-coolant accidents occurring with containment bypass (V-LOCA)
- Steam line break accidents (FWLB, SLB)
- Steam generator tube rupture accidents (SGTR)
- Total loss of heat sink or associated systems (H1)
- Station blackout (H3)
- Loss of onsite power (PDS)
- Transients involving automatic shutdown failure (ATWS)
- Reactor-coolant system (RCS) transients

For this project, only LOCA was studied.
3 In-Vessel Core Degradation

3.1 Description of Loss-of-Coolant Accident (LOCA)

LOCA are initiated by breaks in the reactor-coolant system (RCS) or any of the connecting circuits [5]. Vessel failure or failure of one or more steam generators is not included. These breaks cause the reactor coolant to leak out, which depressurizes the RCS, leading to an automatic reactor trip and activation of the safety injection system (SIS) to inject coolant back into the system. For large breaks, the rapid increase in pressure inside the containment automatically starts up the containment spray system (CSS) that sprays coolant into the containment to help mitigate the increase in pressure.

When LOCA occurs, the reactor immediately shuts down (called scram) by dropping control rods into the core. The control rods absorb neutrons, removing them from the fission process, thereby stopping the fission reactions from continuing. Even though the fission process is stopped, due to radioactive fission products present, the core will continue to heat up when these fission products decay. This heat produced is called the decay heat. For a 900 MWe vessel, the decay heat produced at the time of shutdown is about 5% of the total reactor power and it decreases with time. After an hour, it drops to 1% and after a month, it drops to 0.1% [6]. Assuming that one or more engineered safety systems have failed, the absence of water to cool down the core will eventually lead to the core degrading from the high temperature, in spite of the fall in decay heat over time.
3.2 Description of Reactor Core and Vessel

The reactor core consists of fuel assemblies, with the majority of each assembly containing fuel rods (Figure 3.1). Each assembly also includes tubes to contain the rods of a control rod cluster and a guide tube. The cladding is made up of zirconium alloy (zircaloy) as zirconium has low neutron-absorbing properties and good corrosion resistance. Within the rods, fuel pellets made of uranium dioxide (UO₂) or a mixture of uranium and plutonium oxides (MOX fuel) are stacked. Since naturally occurring uranium does not contain enough $^{235}\text{U}$ for sustainable nuclear reactions, enrichment is necessary to increase the amount of $^{235}\text{U}$ present in the fuel. The enrichment level of $^{235}\text{U}$ used in the PWRs varies between 3% and 4.5%.

Figure 3.1: Diagram of a fuel rod (left) and a fuel assembly (right) [5]
The core is located within a vessel (Figure 3.2) made of stainless steel, with an upper head that can be removed for refuelling. Inside the vessel, there are lower structures supporting the core, side structures (core barrel) which separate the hot water exiting the core from the cold water entering the vessel and upper structures that consist of the control rod guide tubes. They are collectively known as internals [5].

### 3.3 Chemical Processes

#### 3.3.1 Zircaloy Oxidation

As the core heats up, the zirconium in zircaloy will react with steam to produce an oxide, hydrogen gas and heat [7]:

\[
Zr + 2 \text{H}_2\text{O} \rightarrow \text{ZrO}_2 + 2 \text{H}_2
\]  

(3.1)
1 kg of Zr produces about 44 g of hydrogen gas. At temperatures above 1500 K, there is an exponential increase in the zircaloy oxidation rate as the heat produced by the oxidation accelerates the oxidation process which in turn produces more heat, creating a positive feedback loop [8]. This is a characteristic feature of severe accident experiments and plant simulations. The heat produced also results in a sharp increase in core temperatures. The total amount of energy generated by all oxidation is limited by the maximum rate of zircaloy oxidation as the total heat generated from zircaloy oxidation is sufficient in driving the peak core temperature beyond 3000K or the melting point of the fuel rod.

The oxidation rate of zircaloy is determined by the availability of steam in the core and the diffusion of oxygen into the zircaloy. Generally, the diffusion coefficient for zircaloy is characterized by an exponential function of temperature. For typical accident scenarios, the diffusion of oxygen into the zircaloy limits the oxidation at lower temperatures. When the peak temperature exceeds 1500 K, the positive feedback between oxidation rate and core temperature leads to little limitation on the oxidation rate and heating up. This is due to the fact that the diffusion rate increases with temperature and overwhelms the decrease in oxidation rate from the growing oxide thickness. As the core temperature rises further, the oxidation rate is then limited by the availability of steam and its diffusion to the surface of the zircaloy. In this scenario, the in-vessel thermohydraulic conditions become more important than the temperature dependence of the oxidation process. Specifically, the increase in hydrogen concentration in the upper core area and the decrease of steam formation (due to the decrease in amount of water in the core) are more effective in limiting the maximum oxidation rates, especially in the upper core area.

The total amount of oxidation possible at a given location is limited by the amount of zircaloy present and two other factors. Firstly, for initial heating rates above 0.3 – 0.5K/s, the formation of the protective oxide layer on the outer surface of the cladding would be limited. This permits the zircaloy to melt and relocate to the lower core area immediately after the melting point of zircaloy is reached. In this scenario, the oxidation at the original location of the zircaloy stops, since the zircaloy is completely removed. Although the relocating zircaloy can continue to oxidize, it cools down as it moves towards cooler regions of the core, which slows down the oxidation reaction. Secondly, for initial heating rates below 0.3 – 0.5K/s, the formation of the oxide layer prevents molten zircaloy from relocating, hence oxidizing the zircaloy in place. For intermediate initial heating rates, a combination of both processes can be observed.
3.3.2 Stainless Steel Oxidation

Stainless steel reacts with steam to produce oxides, hydrogen gas and heat. It may contribute about 10 – 15% of the total in-vessel hydrogen production [15]. The oxide layers formed are complex due to the alloying elements (Cr, Ni) and auxiliary elements (Mn, Mo, Ti, Si, V, Nb, etc.) playing a big part in influencing the growth and microstructure of the oxide. As a result, a unique schematic description of the reaction layer system does not exist.

Depending on the conditions, the main reactions that can occur are [7]:

\[
\begin{align*}
\text{Fe} + \text{H}_2\text{O} & \rightarrow \text{FeO} + \text{H}_2 \quad (3.2) \\
2 \text{Fe} + 3 \text{H}_2\text{O} & \rightarrow \text{Fe}_2\text{O}_3 + 3 \text{H}_2 \quad (3.3) \\
2 \text{Fe}_3\text{O}_4 + \text{H}_2\text{O} & \rightarrow 3 \text{Fe}_2\text{O}_3 + \text{H}_2 \quad (3.4) \\
\text{Fe} + 4 \text{H}_2\text{O} + 2 \text{Cr} & \rightarrow \text{FeCr}_2\text{O}_4 + 4 \text{H}_2 \quad (3.5) \\
\text{Fe}_3\text{O}_4 + 5 \text{H}_2\text{O} + 4 \text{Cr} & \rightarrow 2 \text{FeCr}_2\text{O}_4 + 5 \text{H}_2 \quad (3.6) \\
3 \text{H}_2\text{O} + 2 \text{Cr} & \rightarrow \text{Cr}_2\text{O}_3 + 3 \text{H}_2 \quad (3.7) \\
\text{H}_2\text{O} + \text{Ni} & \rightarrow \text{NiO} + \text{H}_2 \quad (3.8)
\end{align*}
\]

3.4 Mechanical Processes

Core degradation occurs gradually over a period of minutes to hours, over a range of temperatures from 1000 to 3000 K [8]. The exact timing and temperature depend heavily on the type of core materials, the initial uncover y and heating rates of the core, system pressure, and overall thermohydraulic response of the plant. Hence, the ranges quoted are only strictly applicable to western LWR designs with UO₂ zircaloy fuel rods, which are the designs that this project is studying. Many of the following changes described can occur concurrently in various regions of the core, since the core geometry varies primarily with the local core temperature and different parts of the core may be at different temperatures. Only the phenomena relevant to this project will be discussed in this section.

3.4.1 Melting and Relocation of Zircaloy Cladding

Above 2000 K, the cladding can melt and possibly drain into the lower core regions or lower plenum. The drainage of the molten cladding depends heavily on the early temperature history, as the formation of protective oxides on the outer surface of the cladding can delay or prevent relocation. For fast transients with heating rates above 0.3 – 0.5 K/s, or transients with low core water level, the
cladding will melt and drain into the lower core regions. Depending on the alloy and oxidation content of the material, the melting point of zircaloy typically lies between 2000 and 2200 K. For slower transients with heating rates below 0.3 – 0.5 K/s, the relocation of molten zircaloy is prevented by the formation of a protective oxide layer on the cladding’s outer surface.

The most significant consequence of this phenomenon is the reduction in hydrogen produced and heat generated due to the fall in zircaloy’s oxidation rate. The oxidation rate is reduced because the zircaloy moves into cooler lower regions of the core and as the molten zircaloy freezes, the blockages of metallic zircaloy have significantly less surface areas in contact with steam.

### 3.4.2 Melting and Slumping of the Fuel

When the temperatures of the fuel or oxidized cladding material reach their melting points, they will begin to slump lower in the core. Depending on the temperature gradients in the core and the location of the slumping material, the fuel and the oxidized cladding material will shift to cooler regions of the core until they freeze, leading to the formation of large blockages. These blockages can trap molten materials that are subsequently formed higher in the core or upper plenum. The exact temperature range in which these processes occur are dependant on the composition of the fuel and the oxidized cladding material. However, it is known that the formation of ceramic melts will occur below the melting point of the fuel, going as low as 2870 K for ceramic (U, Zr, O).

### 3.4.3 Relocation of Molten Pool Materials into the Lower Plenum

For the TMI-2 accident, the melt progressed through core bypass into the lower plenum. Analysis of the accident progression suggests that the melt relocation is caused by the penetration of the molten pool to the outer and lower periphery of the core and the inability of the frozen crust to enclose the melt. This is accompanied by the partial collapse of loose debris and fuel rod fragments that were supported by the upper crust. If the core is completely covered with water, melt relocation can still happen, in spite of the slowing or even prevention of the melt’s further movement due to the additional cooling.
3.4.4 Fragmentation of Embrittled Core Materials

Below 1500 K, experiments under design basis accident conditions have shown that the cladding will fail and the fuel pellets may fragment. Above 1500 K, the core geometry at the instant of reflood will have a strong impact on the subsequent change in geometry. For sections where molten metallic or ceramic melts have refrozen, the overall geometry of the material does not change much, even though there may be some cracking in the refrozen material. For sections where fuel rods are relatively intact with their peak temperatures below the melting point of the zircaloy, the cladding and fuel may fragment and partially collapse. For sections where the relatively intact fuel rods have temperatures above the melting point of zircaloy, the cladding will melt away but the fuel pellets will remain relatively unaffected.

The fragmentation of embrittled core materials leads to the failure of the protective oxide layer on the zircaloy cladding. This failure can lead to a sharp increase in the oxidation rate of the zircaloy under some conditions.

3.4.5 Heating and Failure of the Lower Head

Once a portion of the molten core materials or debris has relocated to the lower plenum, what happens next depends on several factors. If water is added to cool the vessel walls or debris, then what happens next depends on if there is water present in the lower plenum, the type of structures present and the location and method of water addition. If no water is added and the lower plenum is dry, the vessel may eventually fail as the debris or melt will continue to heat the lower head. The presence of water in the lower plenum complicates the situation. Ongoing research seeks to prove the hypothesis that water in the vessel is able to penetrate gaps between the debris and vessel walls, which prevents a vessel failure.

3.5 Reflooding

Reflooding, or quenching, is an important procedure in managing LOCA [9]. It involves injecting water back into the vessel to cool the degraded core. The reflooding of high temperature (above 1500K) and relatively intact fuel rods may lead to sharp increases in the temperatures of the fuel rods and surrounding core regions, in addition to an increase in hydrogen production, fission product release and melting. A wide range of reflooding experimental data and data from the TMI-2 accident have demonstrated these characteristic trends, even though this is still an active area of
research. It is shown that this behaviour is a result of the accelerated oxidation of the zircaloy structures, due to the cracking of the protective oxide layers, and the oxidation of newly exposed zircaloy layers and any molten zircaloy that may be present in the coolant channels. As a consequence, local hydrogen production rates may increase by an order of magnitude relative to the rates during the initial heating and melting of the core. Other processes in the reactor system can also be affected. The pressure of the system may rise due to the additional steam and hydrogen produced during reflooding. This was observed in the TMI-2 accident.

The effect of reflooding after large blockages of molten fuel are formed is not well understood quantitatively. However, analysis of the TMI-2 accident along with other supporting calculations have demonstrated that after peak core temperatures go beyond 2800 K with large melts of UO₂ and ZrO₂ already formed, reflooding is ineffective in addressing sustained core heating and the growth of molten pools. The primary reasons being a reduction in the heat transfer area and that the ceramic crust around the melt has low thermal conductivity. For the TMI-2 accident, the molten pool in the upper core area continued to grow, which eventually resulted in a portion of the melt relocating to the lower plenum, despite the core being completely covered with water.

### 3.6 Hydrogen Risk

When hydrogen, air and steam are present in the containment, hydrogen combustion may occur under certain conditions [5]. This gives rise to the possibility of compromising the integrity of the containment and its safety systems. Hydrogen is primarily generated by the oxidation of the metals within the reactor core and if the reactor vessel is breached, this also includes the oxidation of the metals in the corium pool or basement (the foundation which the containment is on) during the molten corium-concrete interaction phase. The distribution of hydrogen in the containment depends on the extent by which the containment atmosphere is mixed.

When the distribution is heterogeneous, there may be areas within the containment that have hydrogen concentrations above the flammability limit. The flammability limit can be determined from a Shapiro diagram similar to the one seen in Figure 3.3. Therefore, this presents an opportunity for ignition, upon which, may give rise to pressure loads that wreck the containment or its safety systems.

To reduce the probability of a hydrogen explosion, safety systems are built within the containment to decrease hydrogen concentration and to mix the hydrogen with the containment atmosphere.
Examples of these systems include the passive autocatalytic recombiners (PARs) which help to reduce the amount of hydrogen present, and the containment spray system (CSS) which homogenises the containment atmosphere.

As reflooding the core brings about a large amount of hydrogen from oxidation, it is noteworthy to investigate if varying the reflood parameters (timing, duration, rates) will lead to different amounts of hydrogen being produced and if there are certain “optimized” values that will minimize the amount of hydrogen, thereby reducing the likelihood of a hydrogen explosion.
4 Accident Source Term Evaluation Code (ASTEC)

4.1 Simulations in Nuclear Safety Research

In the 1970s, the United States started working on computer codes for numerical simulation of core melt accidents to support probabilistic studies. In the late 1980s, the Europeans and Japanese started to develop their own codes and probabilistic safety assessments. These codes are eventually used to explain and assess methods for preventing core melt accidents or to mitigate their impact, as well as to train reactor operators [5].

There are two types of codes:

- Integral codes (or software systems), which are designed to simulate the entire core melt accident, from the starting event to the end where fission product might be released out of the containment;

- Detailed or mechanistic codes, which are used to simulate the phenomena that arise in a specific accident phase such as fission product release, core damage or hydrogen combustion within the containment.

4.2 ASTEC Overview

The Accident Source Term Evaluation Code (ASTEC) is an integral code developed jointly by L'Institut de Radioprotection et de Sûreté Nucléaire (IRSN) in France and Gesellschaft für Anlagen und Reaktorsicherheit mbH (GRS) in Germany. It is designed to simulate the phenomena that happen in a water-cooled reactor during a severe accident. Its main applications are source term evaluations, nuclear reactors safety analysis, and severe accident management guidelines development. As such, it is used extensively in IRSN level 2 probabilistic safety assessments (PSA2) for its present fleet of 900 and 1300 MWe PWRs, and the European Pressurized Reactors (EPRs) that are currently under construction in a number of countries. It is also used for preparing and interpreting experimental programmes, particularly the Phébus FP integral test programme and the tests carried out as part of the International Source Term Programme (ISTP). ASTEC is the reference software within the European Severe Accident Research Network of Excellence (SARNET) and is also used by organisations in Canada, Russia, India, Korea and China.
ASTEC is modular in nature, with each of the modules simulating a set of physical phenomena or a reactor zone. The various modules are listed in Figure 4.1 above. Hence, there are two different running modes in ASTEC V2.0 [10]:

- Stand-alone mode, where each ASTEC module is ran independently, which is useful for module validation and calculation of separate effect tests;

- Coupled mode, where every (or some) ASTEC module is ran successively in a macro-time step, allowing explicit feedback between modules.

For this project, the stand-alone mode is used and the ICARE module is chosen.

Figure 4.1: The modules in ASTEC [10]
4.3 In-Vessel Core Degradation Module (ICARE)

4.3.1 Overview
The In-Vessel Core Degradation module (ICARE) describes both early and late phase in-vessel core degradation phenomena [7]. The early phase occurs near the beginning of the degradation process where the geometry of the core remains largely intact. The late phase refers to the stage where substantial melting and material relocation occur, resulting in the loss of geometry for a large part of the core.

The core degradation process is highly complex due to the phenomena being considered and the need to accurately represent the geometry at each time step. There are constant changes in different components due to processes such as chemical reactions, melting, failure, and relocation. Therefore, the components require a dynamic management. Figure 4.2 shows a diagram of the ICARE algorithm.

In the next few subsections, the relevant models used by ICARE are discussed.
Figure 4.2: ICARE algorithm [7]
4.3.2 Vessel Modelling

In ICARE, the following components are modelled (Figure 4.3): the main core, the upper plate of the core, the lower part of the core support (the lower plate, the support plate, the lower parts of the barrel), the lower plenum, the shell of the core and the annular downcomer [11].

![Figure 4.3: Longitudinal cross section of the vessel modelled by ICARE [11]](image)

The vessel is defined by geometrical objects called macro-components [7]. There are several types of macro-components, each corresponding to a well defined part of the reactor. The geometry of the macro-components is first defined in ICARE. Next, a 2D meshing is applied to these geometrical objects. Finally, the boundary conditions and all physical phenomena happening on or between these macro-components are described.
ICARE employs axi-symmetrical meshing for the cylindrical section of the vessel, which includes only axial and radial segmentations. Azimuthal discretisation is not considered. For the lower head, polar meshing is used and only one mesh is used for the fluids. An example of the meshing done can be seen in Figure 4.4 above.

4.3.3 Core Modelling

Radially from the center (Figure 4.5), the ICARE model includes the fuel assemblies, the baffle, the barrel, the heat shield, the vessel and the insulation. Axially from the top (Figure 4.3), it includes the upper plate, fuel assemblies, the lower plate, support columns and support plate.
The fuel and control rods are represented by vertical cylinders. The initial geometry is defined by users through specifying the axial rod extension and both the internal and external diameters. By default, these values are assumed to be stated for room temperature. This implies that the user input is updated by ICARE automatically at time zero to account for the initial thermal expansion according to the initial axial temperature profile that is supplied by the user.

In the core, fuel rods are grouped together to form different representative rods. The thermal, mechanical and chemical behaviour of the representative rods are assumed to correspond to the evolution of the several other rods they represent.

In Figure 4.6 above, rod 1 represents four fuel rods, rod 2 represents eight fuel rods, control rod represents four control rods, rod 4 represents another eight fuel rods and rod 5 represents the last eight fuel rods.
For reactor calculations, the core may be represented with five or six radial rings as seen in Figure 4.7 [7]. The bypass zone can be represented as the sixth or seventh ring. The downcomer portion is usually represented as the seventh or eighth ring.

### 4.3.4 Zircaloy Oxidation Model

The formation of brittle phases, specifically ZrO$_2$ and a high oxygen-stabilised α-phase (Figure 4.8), denoted by α-Zr(O), reduce the amount of oxygen diffusing into the underlying β-phase and they also weaken the zircaloy cladding to thermal shock and impact forces [7].
The growth of $\alpha$-Zr(O) and ZrO$_2$ layers is determined using experimental observations. Under unlimited steam conditions, numerous experiments showed that the oxidation process is controlled by the oxygen diffusion rate. Therefore, the chemical reaction kinetics (total oxygen mass gain and layers growth) can be modelled using parabolic laws, primarily above 1300 K and for sufficiently thick walls (before the $\beta$-phase becomes saturated with oxygen).

The parabolic law can be written as:

$$\frac{dX^2}{dt} = K_2$$

(4.1)

where $X$ is the total oxygen mass gain in both ZrO$_2$ and $\alpha$-Zr(O) layers or the ZrO$_2$ layer thickness.

$K_2$ is the kinetics constant and it verifies a classical Arrhenius formulation:

$$K_2 = Ae^{-\frac{B}{RT}}$$

(4.2)

where $T$ is the wall temperature, $A$ is the constant of the Arrhenius formulation, $B$ is the activation energy and $R$ is the perfect gas constant. The constants used are based on experimental observations and are specified in ICARE.
In this model, the oxygen concentration profiles in the three layers are not calculated. Therefore, additional assumptions are necessary to balance the oxygen mass in the cladding. These assumptions are:

- the oxide scale consists of stoichiometric ZrO$_2$ (no oxygen profile)
- the β-Zr phase is assumed to be oxygen free and will remain so
- $\alpha$-Zr(O) phase consists of $y$ mol % of oxygen and $(1 - y)$ mol % of Zr, with $y = 20$ by default

When the three layers are present, an additional assumption is required to manage the numerical problem under limited steam supply. Under steam starvation, the model assumes that the oxygen available will preferentially assist $\alpha$-Zr(O) to grow, implying that the growth of $\alpha$-Zr(O) is favoured at the expense of the growth of ZrO$_2$.

More details about the oxidation process can be found in Appendix A.

### 4.3.5. Stainless Steel Oxidation Model

Under normal operating conditions, a protective layer primarily composed of chromic oxide is formed on the surface of stainless steel, preventing oxygen diffusion through the metal and also the diffusion of the iron ions towards the surface [7]. However, during severe accidents, the protective ability decreases as the temperature rises. Hence, the oxidation model is only activated above approximately 850 K.

In ICARE, the mass fractions of Fe, Cr and Ni are fixed throughout the accident in a given run, implying that there is no Cr and Ni diffusing towards the surface of the metal. The values of these mass fractions can be changed at the start of the run, depending on the type of steel to be represented, but the initial distribution will remain constant through the oxidation process.

Similar to the zircaloy oxidation process, the stainless steel oxidation process also follows the parabolic law. For ICARE, the oxide layer is taken to be a perfect mixture of some of the following oxides: FeO, Fe$_3$O$_4$, Fe$_2$O$_3$, Cr$_2$O$_3$, NiO and FeCr$_2$O$_4$. 
5 Results

5.1 Description of Testcase

For this project, the ICARE testcase provided by ASTEC was used. This code is based on the French’s 900 MWe vessels. For a typical 900 MWe vessel, it is 12.3 m tall and 4 m wide internally. It contains 157 fuel assemblies and each assembly rod contains fuel pellets that have a total height of 3.66 m. Under normal operational conditions, the absolute reactor coolant pressure is 155 bar, the coolant flow rate is 5900 kg/s, the total amount of coolant (including those in the pressurizer) is 271 m³, the temperature of the coolant when entering the vessel is 559 K and the temperature when it exits is 596 K [5].

This testcase describes a LOCA at time zero caused by a break, which results in a drop in pressure in the vessel over time. The reason for the break is not specified, but it is not pertinent to the discussion. At time zero, scram has occurred, implying that the fuel rods are no longer undergoing fission and their only contribution to heating up the core is their decay heat. The decay heat is assumed to be constant. Reflood begins automatically at a specified time and mass flow rate. It lasts for a specified amount of time.

For this project, results of the hydrogen produced and the rupture time of the vessel are collected for reflooding under different timings, durations and mass flow rates.

The complete code can be found in Appendix D.

5.2 Analysis of One Simulation

In this section, a simulation of the default settings is analysed. Here, reflood began at 300 s with mass flow rate at 100 kg/s. It slowed down to 20 kg/s at 1000 s and stopped completely at 2000 s. The total amount of hydrogen produced was about 193 kg, with the zircaloy oxidation contributing about 180 kg and steel oxidation contributing about 13 kg. The vessel ruptured at about 9061 s. 75% of the total zirconium present and 13% of the total steel present were oxidised. The final state of the core can be seen in Figure 5.1(a) below. As can be seen, reflooding did not manage to preserve the core or prevent vessel rupture.
(a) Graph of temperature in each area of the core. Temperature is in kelvins.

(b) Graph of power produced by different reactions over time
(c) Graph of total hydrogen produced over time

(d) Graph of instantaneous hydrogen production over time

Figure 5.1: Graphs from the default testcase at time of rupture
Taking a look at the time evolution of hydrogen’s instantaneous production in Figure 5.1(d), it can be seen that there are two distinct sections. The first section (Section A) is a result of oxidation processes within the cylindrical section of the vessel and the second section (Section B) is a result of oxidation processes in the lower plenum (hemi-spherical bottom of the vessel).

Section A is characterised by somewhat continuous hydrogen production with occasional peaks. The hydrogen was primarily produced from the zircaloy cladding of the fuel rods oxidising, which can be seen in Figure 5.1(b). Hydrogen production rate increased with time as the heat produced from the oxidation reaction accelerated the oxidation process, which in turn created more heat and further accelerated the process. However, as the core continued to heat up, the fuel rods began to melt once the core temperature exceeded the melting points of the relevant materials. When the zircaloy claddings melted and relocated to cooler areas of the core, the molten zircaloy cooled down and froze, creating blockages that have significantly less surface area in contact with steam. This causes the rate of hydrogen produced to drop considerably. Therefore, the rate of hydrogen produced generally increased as time passed, but would exhibit occasional dips when the fuel rods melted and relocated elsewhere. Other possible causes of the peaks include the ballooning and rupture of the cladding which would have allowed the oxidation of the cladding’s inner surface, and the fragmentation of embrittled core materials upon reflood which might have led to the failure of the protective oxide layer.

As the molten zircaloy and the melted fuel relocated to the lower sections of the core, they froze and blocked the rest of the fuel rods below it from accessing the supply of steam. This can be seen in Figure 5.2 below, where the black meshes blocked the meshes below them. Therefore, this brought Section A to an end and created an area between Section A and B, where the hydrogen production rate was relatively negligible. It should be noted that hydrogen was still being produced, just at a much lower rate. As the fuel rods continued to melt and grow the molten pool, it eventually reached a point where it penetrated the lower plate and the magma began to relocate to the lower plenum. This restarted the hydrogen production as new areas were exposed to steam and this led into Section B.
Section B is characterised by discontinuous spikes. The hydrogen was mainly produced by the oxidation of the U-O-Zr magma and the oxidation of the debris, which can be seen in Figure 5.1(b). The spikes observed in the instantaneous hydrogen production rate were a combination of the spikes in oxidation of the U-O-Zr magma and the debris. Since the debris contained zircaloy and were modelled as particles, the time at which each individual particle began oxidising depended on whether the conditions were met for each individual particle. Hence, this created the spikes for debris oxidation. For the magma spikes, the movement of the relocating magma and the accompanying loose debris might have broken the crust around the magma, which created new areas for the oxidation process to kick in, hence creating the spikes.

When the relocation of the core materials completed, the magma in the lower plenum would form a protective crust as it oxidised, which effectively cut off the steam supply and the oxidation process became almost negligible. Therefore, the amount of hydrogen produced plateaued off and the heat generated by the oxidation processes became negligible relative to the decay heat.

Taking a look at the total hydrogen produced in Figure 5.1(c), it is apparent that the bulk of the hydrogen produced was due to zircaloy oxidation (about 93%). It can also be seen that Section A produced more hydrogen than Section B due to favourable initial conditions.
5.3 Effects of Reflood

The following results were obtained at the time of rupture when there was no reflood:

(a) Graph of temperature field

(b) Graph of power produced by different reactions over time
Figure 5.3: Graphs of no reflood at time of rupture

(c) Graph of total hydrogen produced over time

(d) Graph of instantaneous hydrogen production over time
The total amount of hydrogen produced was about 3 kg and the vessel rupture time was at 6506 s. 2% of the total zirconium present and 8% of the total steel present were oxidised. In comparison with the default reflood scenario discussed in the previous section, reflood introduced about 64 times more hydrogen and delayed the vessel rupture time by 2555 s. Therefore, it is clear that reflood is useful in cooling down the core, but it comes with hydrogen that needs to be dealt with.

5.4 Analysis of Trends from Varying Reflood Time

5.4.1 Overview

In this project, three reflood parameters were considered: reflood time, reflood duration and reflood rate. Due to insufficient data, the trends observed from varying reflood duration and rate will not be discussed in the main section of the report. The results and discussions for those two parameters can be found in Appendix B and C respectively.

Each simulation took a minimum of 20 mins to run and on average could take up to an hour. The longest simulation took approximately 12 hours as ASTEC took much smaller times steps to ensure that the results converge. The simulations stopped once the vessel ruptured.

As ASTEC relies on numerical models, there were some simulations where the results did not converge. ASTEC would continually reduce the time step in an attempt to solve the convergence issue, but only up to a minimum time step. After which, the programme would terminate and no results would be obtained. To cope with such a scenario, another simulation was run at 50 s later, meaning that if the results for reflood at 500 s did not converge, a simulation of reflood at 550 s would be executed.

For varying reflood time, the time step between each data point was chosen to be 500 s. The reflood rates were kept in their default values, meaning that reflood started at 100 kg/s, slowed down to 20 kg/s 700 s later and stopped completely after another 1000 s. All data points have simulations that converged, with the exception of simulations for default rates at 0 s and 2000 s. Therefore, simulations at 50 s and 2050 s were executed in place.

Since it is possible to consider the no reflood scenario as a scenario whereby reflood occurred after the vessel ruptured, the trends should eventually tend towards the results with no reflood.
5.4.2 Total Hydrogen Produced

Graph 5.1: Graph of total hydrogen produced against reflood time. \( m = 1 \) is the default reflood rates, while \( m = 2 \) is twice the default reflood rates and \( m = 0.5 \) is half the default reflood rates.

Using the no reflood scenario as a reference, the relocation of the fuel rods began at about 2000 s and the core was completely devoid of water at about 4400 s. As it can be seen from Graph 5.1 above, when the reflood time was delayed initially, the total hydrogen produced also risen. This is likely due to the thermal shock introduced by the onset of reflood. A later reflood implies that the fuel rods would be hotter, and the temperature difference between the water entering the system and the fuel rods would be greater. As such, greater mechanical damage could be inflicted to the fuel rods, which might have resulted in cladding failure and led to more areas being exposed to steam. Another possible reason is that a hotter core implies that at the onset of reflood, the oxidation process would be faster as the oxidation rate is proportional to temperature. Therefore, this would have led to more hydrogen produced.

As the reflood time was further delayed, the hydrogen produced reached a peak or plateau and began to fall. At this point, a significant portion of the fuel rods have melted. Hence, reflooding would no longer be quenching oxidation reactions of the zircaloy cladding and the above argument would not apply anymore. Since most of the hydrogen was produced by the oxidation of the zircaloy cladding (see Section 5.2), reflooding when most of the fuel rods have melted would involve the oxidation of the magma instead of the fuel rods, resulting in less hydrogen produced (see Section 5.2). Therefore, the dip in hydrogen produced was expected.
There is a curious increase in hydrogen produced at 5000 s. This was due to the magma heating up, which led to faster oxidation reactions and more hydrogen production. Subsequently, reflood at 5500 s and 6000 s produced less hydrogen than at 5000 s due to the upper layer of the magma cooling down, which slowed down oxidation reactions. This cooling down was due to heat being conducted away by the lower head walls. This can be seen in Figure 5.4 by observing the topmost layer.

(a) Graph of temperature field at 4500 s
(b) Graph of temperature field at 5000 s
(c) Graph of temperature field at 5500 s
(d) Graph of temperature field at 6000 s

Figure 5.4: Graphs of temperature field at different times
As the reflood time approached the maximum possible value, the amount of hydrogen produced was minimal as the core was unable to spend much time oxidising since the vessel was rupturing. This eventually lined up with the results from the scenario with no reflood, meaning that the results were consistent.

For different reflood rates, the graphs seemed to converge initially for early reflood times. This was likely due to most of the zircaloy cladding still heating up, therefore they were not at their critical temperature during reflood. This implies that after reflood, time was still required for the zircaloy to heat up to the critical temperature to begin oxidation. Hence, the reflood rate did not significantly affect the hydrogen produced.

At later reflood times, the graphs began to diverge and showed the effect of different reflood rates. For higher reflood rates, more hydrogen was produced. Since a higher reflood rate came with more water, this was likely the cause of the increase in hydrogen produced. The reverse is true for lower reflood rates.

Eventually, the graphs converged again when reflood began to deal with the oxidation of magma at about 3500 s. Since the amount of oxidation possible was limited by the amount of area exposed to steam, changing the reflood rates made little difference to the amount of hydrogen produced. Therefore, the graphs generally converged.
5.4.3 Vessel Rupture Time

Looking at the default reflood rates only, the rupture time generally remained relatively constant but began to consistently decline at about 4500 s. For a fixed amount of water, the amount of heat it can remove is the same, regardless of the time which it is introduced into the vessel. This explains why the rupture time remained relatively constant. However, in the later part of the graph, reflood eventually became ineffective as the lower head got hotter. At later refloods, the lower head would be closer to its rupture temperature, thereby limiting the amount of time the water can remove heat before the vessel finally ruptured. Hence, this results in a fall in rupture time when reflood is further delayed.

For higher reflood rates, the rupture time was generally later. This is probably due to the presence of more water being introduced into the vessel at higher rates, which helped to cool it down further and delayed rupture. The reverse is true for lower reflood rates. However, the graphs eventually converged for the same reason explained earlier. The lower head was hotter and closer to its rupture temperature, therefore no matter how high or low the reflood rates were, the amount of heat removed was limited by the amount of time available prior to vessel rupture. Therefore, changing the rates did not significantly affect the rupture time.
5.5 Reliability

To ensure that the codes are working correctly, ASTEC compares its results with validation tests and experiments. Generally for ASTEC V2.0 (which is the version in use in this project), the results for early phase core degradation are pretty accurate prior to reflood. For example, by comparison with the QUENCH-03 validation test, it can be seen in Figure 5.5 that the oxidation kinetics used by ASTEC for zircaloy are fairly reliable [13].

![Figure 5.5: Total hydrogen produced from QUENCH-03 [13]](image)

However, ASTEC’s reflood model seems to underestimate the amount of hydrogen produced and in particular for late reflood, the experimentally observed hydrogen peaks were absent. These are observed when ASTEC results were compared with the validation test CORA-13 [14]. Therefore, it seems that the values of the total hydrogen collected in this project are underestimations of the true amount that should be produced. Work on the reflood of severely damaged cores is still ongoing for ASTEC. The latest version (V2.1) provides a new framework in dealing with the reflood of degraded cores [15], which may produce better results.

As it does not seem like experiments have been done with a focus on varying reflood times, it is not possible to determine if the trends obtained in this project are reflective of the phenomenon in the real world. As can be seen in Figure 5.6 below, these are all the experiments or validation tests done for reflood as of 2007. There is room for more research and perhaps in the future, it would be possible to have real experimental figures to ascertain if changing various reflood parameters will have a significant impact on the amount of hydrogen produced. Thereby, suggesting or providing an optimal set of parameters for reflood to occur.
Figure 5.6: The reflood map. Consolidation of experiments done on reflood [16]
6 Conclusion

To an extent, ASTEC’s simulations seem to be reflective of real world phenomena as some of the observations discerned from experimental results and theoretical discourse could be used to explain the simulated results. For results across different reflood times, it also seems that the results were self-consistent, in that the results eventually led to the scenario with no reflood and that unexplainable fluctuations that cannot be rationalized as uncertainty were mostly absent.

The results collected from ASTEC clearly demonstrated that varying reflood parameters does affect the amount of hydrogen produced and the time at which the vessel ruptures. In fact, it seems that reflood at 4500 s, where the water in the lower plenum had just been completely evaporated, is the optimal time to reflood, since the amount of hydrogen produced was minimal but the vessel rupture time was comparable with earlier refloods. This seems to be true across different rates as well, but the applicability across a wider range of rates needs to be verified with more data. Furthermore, it is also not known how adjusting the duration will affect these results since that is not completed in this project due to time constrains. Another possible future direction which this project can take is to look at how these results would apply to other types of vessels, as the evolution of core degradation is heavily dependent on core geometry. Therefore, it can be seen that there is much room to explore and learn about the reflood of degraded nuclear cores.
References


Appendix A: General Algorithm of the Oxidation Process for Zircaloy

When directly exposed to the steam flow, the oxidation of a zircaloy wall occurs when the following conditions are fulfilled simultaneously:

- temperature of the wall exceeds 700 K
- there is steam available in the cooling fluid (no full starvation)

For the zircaloy wall that is not directly exposed to the steam flow, oxidation is possible if the above conditions are met and an additional condition is satisfied:

- the wall is cracked

For the control rod guide tube, internal oxidation is suspended when other chemical reactions are occurring on the same face of the tube. Therefore, in addition to the three conditions listed above, two additional conditions need to be satisfied:

- the Ag-Zr interaction have not begun
- the Fe-Zr interaction have not begun

Figure A.1 is a flow chart of the zircaloy oxidation model. The process is the same for steel oxidation, except that the assumptions and initial conditions are different.
Verify whether oxidation can proceed based on the conditions listed previously. If yes, continue. Otherwise, terminate.

Define the following terms:
- $\Delta t = Dt$
- $t_{\text{beg}} = t$
- $t_{\text{end}} = t + Dt$
- $MO_{av} =$ oxygen mass available for oxidation, with $MO_{av} = \infty$ for the first call (unlimited steam condition)

Verify if oxidation occurs on both sides

- Calculate the evolution of the layers (sub-time step $\Delta t$ can be changed if a layer disappears)
- Modify the content of the cladding layers according to the oxidation process
- Modify the geometry of the cladding
- $MO_{av} = MO_{av} - MO_{\text{fixed}}$, where $MO_{\text{fixed}} =$ oxygen mass fixed during $\Delta t$ (consumed for oxidation)
- $t_{\text{beg}} = t_{\text{beg}} + \Delta t$

If $t_{\text{beg}} < t_{\text{end}}$, return to the previous section and repeat the process. Otherwise, proceed to the next section.

- Calculate the energy released by oxidation
- Calculate the "enthalpic power" $WH$ due to appearance and disappearance of materials:

$$WH = M_{\text{new}}H_{\text{new}} - M_{\text{old}}H_{\text{old}}$$

Figure A.1: Flow chart of the zircaloy oxidation model [7]
Appendix B: Varying Reflood Duration

For this section, the duration in which reflood occurred was varied. Reflood began at 300 s, which is the default. The duration was varied by adding 1000 s when the reflood rate was at 100 kg/s. More specifically, the specific time which the reflood rates slowed to 20 kg/s is changed by 1000 s between each data point. All data points have simulations that converged.

Graph B.1: Graph of total hydrogen produced against reflood duration

Graph B.2: Graph of vessel rupture time against reflood duration
Generally, it seems like a longer duration leads to more hydrogen being produced and a later rupture time. The increase in hydrogen was most likely due to the increase in the amount of water entering the core from the extended duration. The longer duration also allowed longer exposure to water (and by extension, steam) which promoted more oxidation. The increase in vessel rupture time was probably a consequence of the core being cooled down for a longer period of time, which helped to delay the vessel from reaching its rupture temperature.

For further increases in duration, the hydrogen produced will probably plateau as there is a limited amount of zircaloy and steel that can be oxidized. The vessel rupture time will probably continue to increase indefinitely if the given mass flow rate is sufficient to remove heat faster or at least equal to the rate that the core produces heat.
Appendix C: Varying Reflood Rate

For this section, the reflood rates were varied. Adjusting the reflood rates is slightly more complicated as scaling up the rates will lead to an increase in volume of water, but keeping the volume constant will result in a shorter duration of reflood. These scenarios are considered separately in the next two subsections.

C.1 With Constant Volume

To ease calculations, several changes were made to the default values. Reflood continued to begin at 300 s. The total volume to be kept constant was chosen arbitrarily as 20000 kg. The reflood rates were kept constant throughout the reflood duration and it was assumed that the reflood rates increased linearly from one value to another. Therefore, the time evolution of the reflood rate in this subsection would look like:

![Figure C.1: How reflood rate is assumed to change with time](image)

$r$ was the rate varied for each simulation run. $x$ was taken to be the 300 s, where reflood began. The value of $y$ changed for each simulation run, and was dependent on $r$ as the volume was kept constant.
There is a slight downward trend for the hydrogen produced but the graph contained too many fluctuations to be conclusive. More data at smaller time steps are required to determine if the fluctuations are a result of uncertainties or due to phenomena that were occurring in the core. Data can also be collected for other reflood times to check if the trends are consistent with different initial conditions.

Graph C.3: Graph of vessel rupture time against reflood rate

Graph C.2: Graph of total hydrogen produced against reflood rate
The rupture times seem to be relatively constant, which suggest that a fixed amount of water can only afford a fixed amount of cooling, and that is independent of the speed at which it is pumped into the system. In this case, the rupture time is expected to remain constant at even higher reflood rates.

**C.2 With Constant Duration**

Reflood times at 300 s and 1000 s were examined. The reflood rates were all scaled by the same factor $k$ from the default rates, implying that the assumption made in the previous section need not apply here.

For the hydrogen graph produced by reflood at 300 s, it can be seen that initially when the flow rates were increased, more hydrogen was produced. This is likely due to the fact that there were more water (hence, more steam) introduced in the system which allowed for more oxidation to occur. There is some fluctuation after the initial increase and due to the insufficient data, it is difficult to tell if those fluctuations are part of the uncertainty or due to actual phenomena. However, the amount of hydrogen produced is expected to reach a plateau since there is a limited amount of zircaloy and steel available for oxidation.
For the rupture time graph of both reflood times, it can be seen that initial increase in reflood rates helped to extend the vessel rupture time as the coolant was able to remove heat at a higher rate than before. Subsequently, the values eventually plateaued, indicating that higher rates no longer improved heat removal. This was probably due to the water not having enough time to absorb the heat before being driven away from the core. Therefore, this seems to indicate that there is a limit to which increasing the reflood rates can help cool down the core.
Appendix D: The Code Used

Most of the input deck remained unchanged from the default given by ASTEC. Only the reflood section was edited and additional codes were added for data collection. The reflood portion has been marked out.

```plaintext
(namedata="tcsr MCP")
(compute=GETENV(computer))
(compile=GETENV(compiler))

!=================================================================
!
!                ICARE STAND-ALONE CALCULATION
!                    ------------------------
!
!               900 MWe Vessel
!
!   Strong pressures conditions (Break) + reflooding
!
!=================================================================

(TINI = 0. )
(TEND = 20000. )

TITL "ASTEC V2 - TCSRCP"

!=================================================================
!
! SEQUENCE DEFINITION
!=================================================================

STRU SEQUENCE
    TINI (TINI) TIMA (TEND) TSTOP 'TRUP'
    STRU MACR DTFI 1. MINI 1. MAXI 10. END
END

!=================================================================
!
! SAVE DATA BASE
!=================================================================

STRU SAVE
    FILE (namedata/"_"/compute/".bin")
    FREQ 500.
    STOP (namedata/"_"/compute/".stop")
END

!=================================================================
!
! CALCULATION OPTIONS
!=================================================================

STRU CALC_OPT
    SC1 MODULIST 'ICARE' TERM
    STRU ICARE
        MITS 0.1 MATS 10.
```
STRU MESSAGES CONVERGE 'YES' END
END

!=================================================================
! CONNECTIONS
!=================================================================

STRU CONNECTI
NAME 'CORE_OUT' TYPE 'BREAK'
FROM 'VESSEL' TO 'USER'
STAT 'ON' FP 'YES'
MACR 'VAP1' MACR 'VAP2' MACR 'VAP3'
MACR 'VAP4' MACR 'VAP5' MACR 'VAP6' MACR 'VBYP'
MACR 'LIQ1' MACR 'LIQ2' MACR 'LIQ3'
MACR 'LIQ4' MACR 'LIQ5' MACR 'LIQ6' MACR 'LBYP'
SR1 P (TINI) 90.0D5 130. 90.0D5 160. 45.0D5 190. 4.0D5 (TEND) 4.0D5
TERM
END

STRU CONNECTI
NAME 'DC_OUT' TYPE 'BREAK'
FROM 'VESSEL' TO 'USER'
STAT 'ON'
MACR 'VDCO' MACR 'LDCO'
SR1 P (TINI) 90.0D5 130. 90.0D5 160. 45.0D5 190. 3.80D5 (TEND) 3.80D5
TERM
END

STRU CONNECTI
NAME 'BOUND' TYPE 'BCTI'
FROM 'VESSEL' TO 'USER'
STAT 'ON'
MACR 'INSUL' FACE 'EXTERNAL'
INST (TINI)
SR1 Z 0. 3.66 TERM
SR1 TIMP 320. 320. TERM
SR1 H 20. 20. TERM
END

(X = 300.)
(K = 1.)
(M = 1.)

STRU CONNECTI
NAME 'DC_IN' TYPE 'SOURCE'
FROM 'USER' TO 'VESSEL'
STAT 'ON'
rel_elev 1.
MACR 'LDCO'
STRU SOURCE
TYPE 'FLUID' SPEC 'WATER' UNIT 'kg/s-W'
SR1 FLOW (TINI) 0. 0.
(X-20.) 0. 0.
(X) (K*100.) (K*100.D0*596.D3)

Reflood
(X+M*700.) (K*20.) (K*20.D0*596.D3)
(X+M*1700.) 0. 0.
(TEND) 0. 0. TERM

END
END

!=================================================================
! POWER MANAGEMENT
!=================================================================

STRU FP_HEAT
  FUELMASS 82250.D0
  FUELMATE 'UO2'
  XFP (238./(238.+2.*16.))
  SRG FP 'U' 1.0 TERM
  SR1 Powe (TINI) 30.D6 (TEND) 30.D6 TERM
  TIME (TINI)
  SRG FRAC 'U' 1.0 TERM
  TIME (TEND)
  SRG FRAC 'U' 1.0 TERM
END

!=================================================================
! PHYSICAL OPTIONS
!=================================================================

STRU PHYSICAL
  SC1 FP 'U' TERM
  SRG FP_PHFR 'U' 1.000D0 TERM
  SIG FP_SPMD 'U' 2 TERM
END

!=================================================================
! VESSEL DEFINITION
!=================================================================

(pres = 90.D5)
(tgas = 577.)
(tliq = 573.)

STRU VESSEL

!--- options icare

STRU OPTI ARRA SQUARE ENBA NO PITC 1.26D-2 END
STRU NUME EPSI 1.D-3 END

!--- meshing

STRU DISC
  SR1 AXIA 0. 0.4 0.8 1.2 1.5 1.8 2.1 2.4 2.8 3.2 3.66 TERM
  SR1 RADI 0.0 0.25 0.50 0.75 1.0 1.25 1.52 1.671 2.30 TERM

SC1 ROD FUEL1 CLAD1 TERM
SC1 ROD CROD1 GAIN1 GTUB1 TERM
SC1 ROD FUEL2 CLAD2 TERM
SC1 ROD CROD2 GAIN2 GTUB2 TERM
SC1 ROD FUEL3 CLAD3 TERM
SC1 ROD CROD3 GAIN3 GTUB3 TERM
SC1 ROD FUEL4 CLAD4 TERM
SC1 ROD CROD4 GAIN4 GTUB4 TERM
SC1 ROD FUEL5 CLAD5 TERM
SC1 ROD CROD5 GAIN5 GTUB5 TERM
SC1 ROD FUEL6 CLAD6 TERM
SC1 ROD CROD6 GAIN6 GTUB6 TERM
SC1 GRID GRID1 TERM
SC1 SHRO BAFFLE TERM
SC1 SHRO BARREL SHIELD TERM
SC1 SHRO VESSEL INSUL TERM
STRU LOWE
   ZMIN -2.6
   ZMAX 0.0
   R 2.5
END
END

!--- physical objects definition
STRU MACR
   NAME FUEL1 TYPG CYLINDER
   WEIG 1121 R 0.125
   ZMIN 0.0 ZMAX 3.66
   DINT 0.0 DEXT 0.0094
   MATE UO2 MASF 1.0
   TGEO 'REAL'
   PORO 0.0628
   T 300.C
END
STRU GEOP BURN 0. END
STRU MACR
   NAME CLAD1 TYPG CYLINDER
   WEIG 1121 R 0.125
   ZMIN 0.0 ZMAX 3.66
   DINT 0.00958 DEXT 0.0109
   MATE ZR MASF 1.0
   TGEO 'REAL'
   T 300.C
END
STRU MACR
   NAME GRID1 TYPG GRID
   WEIG 1121 R 0.125
   MATE NI MASF 1.
   Z 0.5 PERI 0.05
   HEIG 0.04 THIC 0.0002
   T 300.C
END
STRU MACR NAME FUEL2 COPY FUEL1 WEIG 3364 R 0.375 END
STRU MACR NAME CLAD2 COPY CLAD1 WEIG 3364 R 0.375 END
STRU MACR NAME FUEL3 COPY FUEL1 WEIG 5606 R 0.625 END
STRU MACR NAME CLAD3 COPY CLAD1 WEIG 5606 R 0.625 END
STRU MACR NAME FUEL4 COPY FUEL1 WEIG 7849 R 0.875 END
STRU MACR NAME CLAD4 COPY CLAD1 WEIG 7849 R 0.875 END
STRU MACR NAME FUEL5 COPY FUEL1 WEIG 10091 R 1.125 END
STRU MACR NAME CLAD5 COPY CLAD1 WEIG 10091 R 1.125 END
STRU MACR NAME FUEL6 COPY FUEL1 WEIG 13417 R 1.375 END
STRU MACR NAME CLAD6 COPY CLAD1 WEIG 13417 R 1.375 END
STRU MACR NAME CROD1 TYPG CYLINDER
WEIG 31 R 0.125
ZMIN 0.0 ZMAX 3.66
DINT 0.0 DEXT 0.00866
MATE 'AIC' MASF 1.0
TGEO 'REAL'
T 300.C
END
STRU MACR NAME GAIN1 TYPG CYLINDER
WEIG 31 R 0.125
ZMIN 0.0 ZMAX 3.66
DINT 0.00866 DEXT 0.0097
MATE 'STEEL' MASF 1.0
TGEO 'REAL'
T 300.C
END
STRU MACR NAME GTUB1 TYPG CYLINDER
WEIG 31 R 0.125
ZMIN 0.0 ZMAX 3.66
DINT 0.01265 DEXT 0.01346
MATE 2R MASF 1.0
TGEO 'REAL'
T 300.C
END
STRU MACR NAME CROD2 COPY CROD1 WEIG 93 R 0.375 END
STRU MACR NAME GAIN2 COPY GAIN1 WEIG 93 R 0.375 END
STRU MACR NAME GTUB2 COPY GTUB1 WEIG 93 R 0.375 END
STRU MACR NAME CROD3 COPY CROD1 WEIG 156 R 0.625 END
STRU MACR NAME GAIN3 COPY GAIN1 WEIG 156 R 0.625 END
STRU MACR NAME GTUB3 COPY GTUB1 WEIG 156 R 0.625 END
STRU MACR NAME CROD4 COPY CROD1 WEIG 218 R 0.875 END
STRU MACR NAME GAIN4 COPY GAIN1 WEIG 218 R 0.875 END
STRU MACR NAME GTUB4 COPY GTUB1 WEIG 218 R 0.875 END
STRU MACR NAME CROD5 COPY CROD1 WEIG 281 R 1.125 END
STRU MACR NAME GAIN5 COPY GAIN1 WEIG 281 R 1.125 END
STRU MACR NAME GTUB5 COPY GTUB1 WEIG 281 R 1.125 END
STRU MACR NAME CROD6 COPY CROD1 WEIG 373 R 1.375 END
STRU MACR NAME GAIN6 COPY GAIN1 WEIG 373 R 1.375 END
STRU MACR NAME GTUB6 COPY GTUB1 WEIG 373 R 1.375 END
STRU MACR NAME BAFFLE TYPG CYLINDER
ZMIN 0.0 ZMAX 3.66
DINT 3.04 DEXT 3.09
MATE 'STEEL' MASF 1.0
TGEO 'REAL'
T 300.C
END
STRU MACR NAME BARREL TYPG CYLINDER
    ZMIN 0. ZMAX 3.66
    DINT 3.34 DEXT 3.44
    MATE STEEL MASF 1.0
    TGEO 'REAL'
    T 300.C
END
STRU MACR NAME SHIELD TYPG CYLINDER
    ZMIN 0. ZMAX 3.66
    DINT 3.44 DEXT 3.51
    MATE STEEL MASF 1.0
    TGEO 'REAL'
    T 300.C
END
STRU MACR NAME VESSEL TYPG CYLINDER
    ZMIN 0. ZMAX 3.66
    DINT 3.987 DEXT 4.387
    MATE STEEL MASF 1.0
    TGEO 'REAL'
    T 300.C
END
STRU MACR NAME INSUL TYPG CYLINDER
    ZMIN 0. ZMAX 3.66
    DINT 4.387 DEXT 4.6
    MATE NID MASF 1.0
    TGEO 'REAL'
    T 300.C
END
STRU MACR NAME BED1 TYPG DEBRIS
    DINT 0.0 DEXT (2.*0.25)
    ZMIN 0. ZMAX 3.66
STRU DEBR
    FMAS 1.0D0
STRU PART
    SR1 PERC 1.0D0 TERM
    SR1 DEXT 1.0D-4 TERM
    SR1 DINT 0.0D0 TERM
END
    MATE STEEL MASF 1.0D0
END
    SR1 T 0. 3.66 548. 548. TERM
RESI YES
END
STRU MACR NAME BED2 COPY BED1 DINT (2.*0.25) DEXT (2.*0.50) END
STRU MACR NAME BED3 COPY BED1 DINT (2.*0.50) DEXT (2.*0.75) END
STRU MACR NAME BED4 COPY BED1 DINT (2.*0.75) DEXT (2.*1.) END
STRU MACR NAME BED5 COPY BED1 DINT (2.*1.) DEXT (2.*1.25) END
STRU MACR NAME BED6 COPY BED1 DINT (2.*1.25) DEXT (2.*1.52) END
STRU MACR NAME MAGMA1 TYPG MAGMA
    DINT 0.0 DEXT (2.*0.25)
    ZMIN 0. ZMAX 3.66
    MATE ZRO2 MASF 1.0D0
    SR1 T 0. 3.66 548. 548. TERM
STRU MACR
NAME LOWER TYPG LOWERPLE
ZMIN -2.5 ZMAX 0.
DINT 3.987 DEXT 4.387
MESAS 15 MESAC 1 MESR 3
MATE STEEL MASF 1. T 300.C
STRU LSTRUCT
ZMIN -1.79 ZMAX 0. NTUB 75
DINT 0.0 DEXT 0.1542 PITCH 0.34
MATE STEEL MASF 1. T 300.C
END
END
STRU MACR NAME VAP0 TYPG FLUIDGAS LOWE YES
ZMIN -2.5 ZMAX 0.
GAS H2O PP (pres)
T (tgas) P (pres)
END
STRU MACR NAME VAP1 TYPG FLUIDGAS R 0.125
ZMIN 0. ZMAX 3.66
GAS H2O PP (pres)
T (tgas) P (pres)
END
STRU MACR NAME VAP2 COPY VAP1 R 0.375 END
STRU MACR NAME VAP3 COPY VAP1 R 0.625 END
STRU MACR NAME VAP4 COPY VAP1 R 0.875 END
STRU MACR NAME VAP5 COPY VAP1 R 1.125 END
STRU MACR NAME VAP6 COPY VAP1 R 1.375 END
STRU MACR NAME VDC0 COPY VAP1 R 1.755 END
STRU MACR NAME LIQ0 TYPG FLUIDLIQ LOWE YES VELO ONCE
ZMIN -2.5 ZMAX 0.
P (pres) T (tliq) LEVEL 0.
END
STRU MACR NAME LIQ1 TYPG FLUIDLIQ R 0.125
ZMIN 0. ZMAX 3.66
P (pres) T (tliq) LEVEL 3.66
END
STRU MACR NAME LIQ2 COPY LIQ1 R 0.375 END
STRU MACR NAME LIQ3 COPY LIQ1 R 0.625 END
STRU MACR NAME LIQ4 COPY LIQ1 R 0.875 END
STRU MACR NAME LIQ5 COPY LIQ1 R 1.125 END
STRU MACR NAME LIQ6 COPY LIQ1 R 1.375 END
STRU MACR NAME LBYP COPY LIQ1 R 1.545 END
STRU MACR NAME LDCO COPY LIQ1 R 1.755 END

!--- material properties

STRU MATE NAME NID
TBEG 1748.D0 TEND 1755.D0 HEAT 2.D06 TYPE L
SR1 CP 373. 3000.
     350. 350. TERM
SR1 CDS 293. 1000. 1573.
     0.1 0.2 0.3 TERM
SR1 EMS 273. 1773.
     0.8 0.8 TERM
SR1 ROS 273. 1773.
     780. 780. TERM
SR1 ROL 1755. 3000.
     7700. 7700. TERM
SR1 CDL 1755. 3000.
     50. 70. TERM
SR1 EML 1773. 3000.
     0.8 0.8 TERM
END

!--- Fission products transport and release
STRU FPEVOL
PHYS 'NORELEAS'
SR1 PROF 0.     0.
     0.183  0.565
     0.549  1.084
     0.915  1.277
     1.281  1.286
     1.647  1.235
     2.013  1.15
     2.379  1.077
     2.745  0.976
     3.111  0.777
     3.477  0.571
     3.66  0. TERM
SC1 EMIT  'FUEL1' 'FUEL2' 'FUEL3' 'FUEL4' 'FUEL5' 'FUEL6' TERM
SC1 BARR  'CLAD1' 'CLAD2' 'CLAD3' 'CLAD4' 'CLAD5' 'CLAD6' TERM
SR1 FACT 1.2572  1.2127  1.1469  1.0596  0.951   0.8198 TERM
END

!--- Thermal exchanges
STRU GAP PGAS 35.D5 HE 1. H2 0. H2O 0.
FACE EXTERNAL FACE INTERNAL
MACRGEN ROD_FUEL MACRGEN ROD_FUEL END
STRU GAP PGAS 70.D5 HE 0. H2 0. H2O 1.
FACE EXTERNAL FACE INTERNAL
MACRGEN ROD_CTRL    MACRGEN ROD_CTRL    END
STRU GAP  FACE EXTERNAL    FACE INTERNAL
MACRGEN SHROUD MACRGEN SHROUD END
STRU GAP  FACE EXTERNAL    FACE INTERNAL
MACR CLAD1 MACR GRID1 END
STRU CONVLOWE FLUI VAP0 FLUI LIQ0 END
STRU CONV CONF BUNDLE ZMIN 0. ZMAX 3.66
FLUIGEN GASPHASE MACRGEN ROD FACE EXTERNAL END
STRU CONV CONF BUNDLE ZMIN 0. ZMAX 3.66
FLUIGEN LIQPHASE MACRGEN ROD FACE EXTERNAL END
STRU CONV CONF PIPE ZMIN 0. ZMAX 3.66
FLUIGEN GASPHASE MACRGEN SHROUD FACE INTERNAL END
STRU CONV CONF PIPE ZMIN 0. ZMAX 3.66
FLUIGEN LIQPHASE MACRGEN SHROUD FACE INTERNAL END
STRU CONV CONF PIPE ZMIN 0. ZMAX 3.66
FLUIGEN GASPHASE MACRGEN SHROUD FACE EXTERNAL END
STRU CONV CONF PIPE ZMIN 0. ZMAX 3.66
FLUIGEN LIQPHASE MACRGEN SHROUD FACE EXTERNAL END
STRU CONV
CONF BUNDLE FLUI VAP1
DROD 0.0109
SC1 MACR GRID1 TERM
SC1 FACE INTERNAL TERM
END
STRU CONV
CONF BUNDLE FLUI LIQ1
DROD 0.0109
SC1 MACR GRID1 TERM
SC1 FACE INTERNAL TERM
END
STRU COND MACRGEN ROD MACRGEN ROD
FACE UP FACE DOWN END
STRU COND MACRGEN ROD MACRGEN ROD
FACE EXTERNAL FACE INTERNAL END
STRU COND MACRGEN SHROUD MACRGEN SHROUD
FACE UP FACE DOWN END
STRU COND MACRGEN SHROUD MACRGEN SHROUD
FACE EXTERNAL FACE INTERNAL END
STRU COND
    MACR LOWER MACR LOWER
END

STRU COND
    MACR VESSEL FACE DOWN    MACR LOWER FACE UP
END

STRU EXCHLOWE END

!--- Radiation

!--- Radiative exchanges in upper and lower cavities
STRU RADCAV
    NCUT 3   ZOUT 3.66
    ZMIN 0.  ZMAX 3.66
    RMIN 0.  RMAX 2.30
END

!--- Radiation in the core
STRU RADR
    ABSG YES GAS  HE
    PGAS 35.D5 HE 1.  H2 0.  H2O 0.
    PITC 1.26D-2
    ZMIN 0.  ZMAX 3.66
END

!--- Chemistry
STRU ZROX FLUIGEN GASPHASE MACRGEN ROD_FUEL
    PHYS URBANIC ZMIN 0. ZMAX 3.66
    FACE EXTERNAL STAT ALWAYS
    FACE INTERNAL STAT CRACKED
END

STRU ZROX FLUIGEN GASPHASE MACRGEN ROD_CTRL
    PHYS URBANIC ZMIN 0. ZMAX 3.66
    FACE EXTERNAL STAT ALWAYS
    FACE INTERNAL STAT ALWAYS
END
STRU ZROX FLUI VAP1 MACR GRID1
    PHYS URBANIC FACE INTERNAL STAT ALWAYS
END
STRU ZRGR MACR CLAD1 GRID GRID1 MTAB NI END
STRU UZRL MACRGEN ROD_FUEL MACRGEN ROD_FUEL SOLU LIQUIDUS END
STRU FEOX META STEEL ZMIN 0. ZMAX 3.66
    FLUIGEN GASPHASE MACRGEN ROD_CTRL FACE EXTERNAL END

!--- Degradation
STRU DECA
    MACRGEN ROD_CTRL  DIRE EXTERNAL
    MACRGEN ROD_CTRL  SC1 FACE EXTERNAL TERM
BLOC YES TYPE CHEMICAL
END

STRU DECA
MACRGEN ROD_FUEL DIRE EXTERNAL
MACRGEN ROD_FUEL SC1 FACE EXTERNAL TERM
BLOC YES
END

STRU DECA
SC1 FROM GRID1 TERM
SC1 TOWA CLAD1 TERM
SC1 FACE EXTERNAL TERM
DIRE INTERNAL
END

STRU DECALOWE END

STRU CREE INTS FUEL1 EXTS CLAD1 FLUI VAP1
PROO 30.D5 TROO 298.
EPMX 0.25D0
LOPL 4.0D-6 UPPL 8.0D-6 CRAC 0.25D0 END

STRU CREE INTS FUEL2 EXTS CLAD2 FLUI VAP2
PROO 30.D5 TROO 298.
EPMX 0.25D0
LOPL 4.0D-6 UPPL 8.0D-6 CRAC 0.25D0 END

STRU CREE INTS FUEL3 EXTS CLAD3 FLUI VAP3
PROO 30.D5 TROO 298.
EPMX 0.25D0
LOPL 4.0D-6 UPPL 8.0D-6 CRAC 0.25D0 END

STRU CREE INTS FUEL4 EXTS CLAD4 FLUI VAP4
PROO 30.D5 TROO 298.
EPMX 0.25D0
LOPL 4.0D-6 UPPL 8.0D-6 CRAC 0.25D0 END

STRU CREE INTS FUEL5 EXTS CLAD5 FLUI VAP5
PROO 30.D5 TROO 298.
EPMX 0.25D0
LOPL 4.0D-6 UPPL 8.0D-6 CRAC 0.25D0 END

STRU CREE INTS FUEL6 EXTS CLAD6 FLUI VAP6
PROO 30.D5 TROO 298.
EPMX 0.25D0
LOPL 4.0D-6 UPPL 8.0D-6 CRAC 0.25D0 END

STRU INTE
MACR CLAD1 MACR CLAD2 MACR CLAD3
MACR CLAD4 MACR CLAD5 MACR CLAD6
MATE ZRO2

STRU STAT
ISTA WHATEVER
FSTA DISLOCAT
STRU RULE
  VARI TEMPERAT VALU 2250.D0 CRIT GREATER
END
STRU RULE
  MATE ZRO2
  VARI THICKNES VALU 300.D-6 CRIT LOWER
END
END
STRU INTE
  MACR CLAD1 MACR CLAD2 MACR CLAD3
  MACR CLAD4 MACR CLAD5 MACR CLAD6
  MACR GTUB1 MACR GTUB2 MACR GTUB3
  MACR GTUB4 MACR GTUB5 MACR GTUB6
STRU STAT
  ISTA WHATEVER
  FSTA DISLOCAT
  STRU RULE
    VARI TEMPERAT VALU 2500.D0 CRIT GREATER
END
END
STRU INTE
  MACR FUEL1 MACR FUEL2 MACR FUEL3
  MACR FUEL4 MACR FUEL5 MACR FUEL6
  MATE UO2
STRU STAT
  ISTA COMPACT
  FSTA DISLOCAT
  STRU RULE
    VARI TEMPERAT VALU 1500.D0 CRIT GREATER
END
END
STRU STAT
  ISTA CRACKED
  FSTA DISLOCAT
  STRU RULE
    VARI TEMPERAT VALU 1500.D0 CRIT GREATER
END
END
STRU CCYL MACRGEN ROD MACRGEN ROD END
STRU CCYL CONT 0.0 MACR 'BARREL' MACR 'SHIELD' END
STRU CCYL CONT 0.0 MACR 'VESSEL' MACR 'INSUL' END

--- Physics in the debris bed
--- Debris generation from fuel rods and claddings
STRU CREADEBR
MACR FUEL1 MACR FUEL2 MACR FUEL3 MACR FUEL4 MACR FUEL5 MACR FUEL6

BLOC NO
ZMIN 0. ZMAX 3.66
STRU PART
   SR1 PERC 1.0 TERM
   SR1 DEXT 2.0D-3 TERM
   SR1 DINT 0.0D0 TERM
END

STRU CREADEBR
MACR CLAD1 MACR CLAD2 MACR CLAD3 MACR CLAD4 MACR CLAD5 MACR CLAD6

ZMIN 0. ZMAX 3.66
BLOC NO
STRU PART
   SR1 PERC 1.0 TERM
   SR1 DEXT 2.D-4 TERM
   SR1 DINT 0.D0 TERM
END

STRU CREADEBR
MACR CLAD1 MACR CLAD2 MACR CLAD3 MACR CLAD4 MACR CLAD5 MACR CLAD6

ZMIN 0. ZMAX 3.66
BLOC NO
STRU PART
   SR1 PERC 1.0 TERM
   SR1 DEXT 2.0D-3 TERM
   SR1 DINT 0.0D0 TERM
END

STRU OPTIDEBR
   SR1 TEMP 300. 3300. TERM
   SR1 CGAS 0.01 0.01 TERM
END

STRU POOL
   ZMIN 0.D0 ZMAX 4.D0
   EFFC 10. CRUP STABLE
END

STRU MOVEMAG DTMI 1.0 LEAK 'YES' END

STRU DECA
MACRGEN DEBRIS   DIRE EXTERNAL
MACRGEN MAGMA    SC1 FACE INTERNAL TERM TYPE CHEMICAL
END
STRU DECA
MACRGEN ROD      DIRE EXTERNAL
MACRGEN MAGMA    SC1 FACE INTERNAL TERM TYPE CHEMICAL
END
STRU DECA
MACRGEN SHROUD   DIRE INTERNAL
MACRGEN MAGMA    SC1 FACE INTERNAL TERM TYPE CHEMICAL
END
STRU DECA
MACRGEN SHROUD   DIRE EXTERNAL
MACRGEN MAGMA    SC1 FACE INTERNAL TERM TYPE CHEMICAL
END

STRU DECA
MACRGEN DEBRIS   DIRE EXTERNAL
MACRGEN MAGMA    SC1 FACE INTERNAL TERM TYPE CHEMICAL
END
STRU DECA
MACRGEN ROD      DIRE EXTERNAL
MACRGEN MAGMA    SC1 FACE INTERNAL TERM TYPE CHEMICAL
END
STRU DECA
MACRGEN SHROUD   DIRE INTERNAL
MACRGEN MAGMA    SC1 FACE INTERNAL TERM TYPE CHEMICAL
END
STRU DECA
MACRGEN SHROUD   DIRE EXTERNAL
MACRGEN MAGMA    SC1 FACE INTERNAL TERM TYPE CHEMICAL
END
!--- Conduction in the debris bed
STRU COND MACRGEN MAGMA MACRGEN ALL END

!--- Convection between debris and fluid and magma
STRU CONV MACRGEN DEBRIS FLUIGEN MAGPHASE END
STRU CONV MACRGEN DEBRIS FLUIGEN GASPHASE END
STRU CONV MACRGEN DEBRIS FLUIGEN LIQPHASE END

!--- Creation of debris
STRU COLLDEBR
STRU PORO
  PMIN 0.7
  MACR BED1 MACR BED2 MACR BED3 MACR BED4 MACR BED5 MACR BED6
END
END

STRU STATDEBR
STUC 0.1
MACR BED1 MACR BED2 MACR BED3 MACR BED4 MACR BED5 MACR BED6
STRU STAT
  ISTA FIXED
  FSTA FREE
STRU RULE
  VARI POROSITY
  VALU 0.7
  CRIT GREATER
END
END
END

!--- Chemics in debris bed
!--- Oxidation of the debris
STRU ZROX
  FLUIGEN GASPHASE MACRGEN DEBRIS
  FACE EXTERNAL STAT ALWAYS
END

!--- Chemics in magma
!--- Oxidation of U-Zr-O in the magma components
STRU UZOXMAG
  ZMIN 0. ZMAX 3.66 TBEG 2098.D0
END

!--- Slump of magma into the lower plenum
STRU SLUMP QSMAX 10.D0 END

!--- Corium fragmentation in the lower plenum
STRU FRAGLOWE DHOLE 0.085D0 NHOLE 6 END

!--- Corium phase separation in the lower plenum
STRU SEPALOWE TAU 10. SEPA_ACT 2 END

!--- Rupture of the lower head
STRU RUPTURE STRU OEUF END END

!--- Radiative exchange between core and liquid in the lower plenum
STRU RADL MLIM 10.D0 END

END

!================================================================
! TOOL MAINTENANCE
!=================================================================
CALL "maint_icare.tool"

!================================================================
! VISU
!=================================================================
CALL "power_copy.visu"
CALL "lower_head_corium_mass_copy.visu"
CALL "lower_head_oeuf_copy.visu"
CALL "core_h2_copy.visu"
CALL "core_h2_mass_copy.visu"
CALL "core_magma_saturation_copy.visu"
CALL "core_debris_porosity_copy.visu"
CALL "core_temperature_field_copy.visu"
CALL "core_water_level_copy.visu"
CALL "menu.visu"
CALL "save.visu"

!
! GRAPHS
!

STRU VISU
FILE "evol.vis"
TITL "Mass of Steam"
SFIL "mass_of_steam"
FREQ 100.
STRU VARI
    NAME 'msteam'
    PATH "BASE:'GENERAL'1:'MSTEAM'1"
END
END

STRU VISU
FILE "evol.vis"
TITL "Mass of Water"
SFIL "mass_of_water"
FREQ 100.
STRU VARI
    NAME 'mwater'
    PATH "BASE:'GENERAL'1:'MWATER'1"
END
END

STRU VISU
FILE "evol.vis"
TITL "Fraction of unoxidised Zr"
SFIL "unoxizr"
FREQ 100.
STRU VARI
  NAME 'unoxidised_zr'
  PATH "BASE:'VESSEL'1:'GENERAL'1:'UNOX_ZR'1"
END
END

STRU VISU
  FILE "evol.vis"
  TITL "Fraction of unoxidised Fe"
  SFIL "unoxife"
  FREQ 100.
  STRU VARI
    NAME 'unoxidised_fe'
    PATH "BASE:'VESSEL'1:'GENERAL'1:'UNOX_FE'1"
  END
END

STRU VISU
  FILE "autosave.vis"
END

!
! RAW DATA
!

STRU PLOT
  FILE "raw.plot"
  FREQ 10.
  STRU VARI
    NAME 'H2_T'
    PATH "BASE:'VESSEL'1:'GENERAL'1:'PRODH2'1"
  END
  STRU VARI
    NAME 'H2_T'
    PATH "BASE:'VESSEL'1:'GENERAL'1:'PRODHZ'1"
  END
  STRU VARI
    NAME 'HF_T'
    PATH "BASE:'VESSEL'1:'GENERAL'1:'PRODHF'1"
  END
  STRU VARI
    NAME 'H2_I'
    PATH "BASE:'VESSEL'1:'GENERAL'1:'PRODH2IN'1"
  END
  STRU VARI
    NAME 'H2_I'
    PATH "BASE:'VESSEL'1:'GENERAL'1:'PRODHZIN'1"
  END
  STRU VARI
    NAME 'HF_I'
    PATH "BASE:'VESSEL'1:'GENERAL'1:'PRODHFIN'1"
  END
  STRU VARI
    NAME 'M_STEAM'
PATH "BASE:'GENERAL'1:'MSTEAM'1"
END
STRU VARI
  NAME 'M_WATER'
  PATH "BASE:'GENERAL'1:'MWATER'1"
END
STRU VARI
  NAME 'R_FE'
  PATH "BASE:'VESSEL'1:'GENERAL'1:'UNOX_FE'1"
END
STRU VARI
  NAME 'R_ZR'
  PATH "BASE:'VESSEL'1:'GENERAL'1:'UNOX_ZR'1"
END
STRU PLOT
  FILE "rawp.plot"
  FREQ 10.
STRU VARI
  NAME 'PZRO2'
  PATH "BASE:'VESSEL'1:'GENERAL'1:'PZRO2'1"
END
STRU VARI
  NAME 'PZROXD'
  PATH "BASE:'VESSEL'1:'GENERAL'1:'PZROXD'1"
END
STRU VARI
  NAME 'PFEOX'
  PATH "BASE:'VESSEL'1:'GENERAL'1:'PFEOX'1"
END
STRU VARI
  NAME 'PUZOXM'
  PATH "BASE:'VESSEL'1:'GENERAL'1:'PUZOXM'1"
END
STRU VARI
  NAME 'PUZOXL'
  PATH "BASE:'VESSEL'1:'GENERAL'1:'PUZOXL'1"
END
STRU VARI
  NAME 'NEUTRON'
  PATH "BASE:'VESSEL'1:'GENERAL'1:'POWENT'1"
END
STRU VARI
  NAME 'FP'
  PATH "BASE:'VESSEL'1:'GENERAL'1:'POWEF_TO'1"
END
END

!=================================================================
! END
!=================================================================
END