Water ingress scenario analyses
of a thorium fuelled HTR

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Nomenclature

\( \alpha \)  Coupling coefficient
\( \beta \)  Fraction of non prompt neutrons
\( \Lambda \)  Neutron generation time
\( \lambda \)  Decay constant
\( \nu \)  Average number of neutrons released per fission
\( \Phi \)  Neutron flux
\( \Phi_V \)  Volume flow
\( \rho \)  Reactivity
\( \Sigma_a \)  Macroscopic absorption cross section
\( \Sigma_f \)  Macroscopic fission cross section
\( r \)  Position vector
\( A \)  Separation surface
\( c \)  Precursor concentration
\( c_p \)  Specific heat
\( D \)  Diffusion coefficient
\( d \)  Separation distance
\( E \)  Internal energy
\( F \)  Force
\( f \)  Fanning friction factor
\( G \)  Mass flow
\( h \)  Heat transfer coefficient
\( h \)  Specific enthalpy
\( J \)  Net neutron current density
\( j \)  Neutron current density
\( k_{eff} \)  Effective multiplication factor
\( L \)  Length
\( l \)  Neutron lifetime
\( M \)  Mass
Nomenclature

\( n \) Neutron concentration
\( P \) Generated power
\( p \) Impuls
\( p \) Pressure
\( Q \) Heat
\( r \) Fission rate
\( T \) Temperature
\( t \) time
\( V \) Volume
\( v \) Velocity
\( W \) Water mass
\( \text{Nu} \) Nusselt number
\( \text{Pr} \) Prandtl number
\( \text{Re} \) Reynolds number
Abstract

A thorium fuelled pebble bed reactor is a promising nuclear reactor design for the next generation of reactors. One of the features of the next generation pebble bed reactors is that they are designed to be inherently safe. The use of thorium makes the reactor more sustainable compared to uranium-235 fuelled designs. It is important for the design process of the reactor to be able to describe the dynamic behaviour in case of slight operating condition changes and accident scenarios. The HTR is loaded with graphite pebbles which contain TRISO elements. These TRISO elements contain the fuel of the reactor and are tested to be able to contain all the radioactive fission fragments for temperatures up to 1600°C. Temperatures of the TRISO elements should thus stay below this maximum safe temperature during any accident.

To describe the dynamic behaviour of the reactor, a coupled dynamic model has been developed. This model couples a simple nodal thermodynamics model with a point kinetics model. The conditions inside the very tall reactor core of the HTR-PM design vary greatly over height. The regular point kinetics model is not able to provide a sufficient accurate description. This is why the point kinetics model is extended to account for multiple zones in the core. Steady state calculations with this model do describe the reactor as expected and in agreement to reference models. Dynamic behaviour seems physically realistic as well. This dynamic model can be used as a first tool in investigating the dynamic behaviour of the core. Specific situations can be described in more detail with more advanced models.

Special attention has been given to the water ingress scenario. During such an accident water from the secondary steam loop leaks into the primary coolant loop of the reactor. The extra moderating effect of the water in the core can result in increased reactivity in the core. It was discovered that only a very small amount of the water ingressed in the primary loop actually resides in the core itself. During a design basis accident where only a maximum of 600kg of water enters the primary loop, the maximum fuel temperature never exceeded the failure temperature of the TRISO elements. During a beyond design accident where the water ingress can potentially accumulate to 2500kg of water, the TRISO failure temperature was exceeded in some situations. This was the case when the water was not extracted out of the system. With the coolant pump still enabled, the maximum allowable temperatures were reached at a total mass of 842kg. With the cooling pumps disabled these temperatures were reached at an ingress of 1176kg.
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Chapter 1

Introduction

The world has an ever growing demand for energy. This consequently results in a need for reliable and efficient energy sources. Currently the main sources of energy are fossil fuels [2]. Extracting energy out of fossil fuels is relatively easy and safe to do, hence its popularity. However extracting energy from fossil fuels is not without drawbacks. During the process of burning fossil fuels, a lot of emission gases and potentially ash and sludge are produced as well. And maybe more importantly, fossil fuels are a limited resource. It is important to have other, preferably more sustainable energy sources by the time the fossil fuel resources become limited.

Nuclear energy can potentially provide this energy source. Nuclear energy is generated by the fission of atoms. This is possible because certain heavy atoms tend to split after absorbing an extra neutron. These atoms split into multiple smaller atoms and neutrons. The split parts of the original atom, called fission fragments, gain a certain amount of kinetic energy. These fission fragments will bump into other materials and transfer their thermal energy to their surroundings. This heat can then be transferred to a coolant which transports the energy to the energy transformer where electricity is generated.

One very common element used for nuclear fission is uranium-235. This atom consists of 92 protons and 143 neutrons. Uranium-235 has a high probability to undergo a fission reaction after absorbing a neutron. This fission will result in the release of thermal energy and on average 2.5 neutrons. These released neutrons can then initiate a fission reaction in another U-235 atom to have a similar return. This way the fission chain reaction can continue. This chain reaction is displayed in Figure 1.1. The goal of a nuclear reactor is to use this mechanism and extract as much energy as possible from the fuel elements in a safe and controlled manner. In such a critical reactor the amount of neutrons and the energy generated stay constant over time.
CHAPTER 1. INTRODUCTION

Figure 1.1: Schematic representation of the U-235 fission reaction.

1.1 Generation IV reactors

As of March 2013 there are 434 operational nuclear reactors in the world [2]. One of the biggest concerns with nuclear power is the high amount of energy that can be generated in a small amount of time. Under normal conditions, the energy is generated in a controlled manner. During an accident however, a lot of heat can be produced in a short period of time via a super critical chain reaction. In case such an accident occurs, the chain reaction should be shut down first. After shutdown, heat is still produced by decay of fission products. Even after shutdown the core has to be cooled to dispose this heat. Safely shutting down the reactor in a timely fashion and cooling the reactor sufficiently can be troublesome as has become apparent by several accidents in the past. Especially the accidents of Three Mile Island, Chernobyl and Fukushima have shocked the world with their impact. In the case of Chernobyl the chain reaction got out of control, while during the accidents in Fukushima and Three Mile Island the decay heat could not be disposed of quickly enough.

Another concern with nuclear energy is the production of nuclear waste. During nuclear fission, fuel is burned. In this process radioactive fission products are created. At some point in time the fuel is depleted and needs to be replaced by new fuel. The spent fuel still contains the radioactive fission products. These fission products will keep decaying and can be a threat for humans for many thousands of years after the use of the fuel in the reactor. These waste products thus have to be handled with great care.

To reduce the disadvantages of present nuclear power plants, the current designs have to be improved. Reactors can especially be designed more sustainable and safer. A group called the generation IV international forum or GIF is in charge of the development of generation IV reactors which should improve on these aspects compared to the current generation III reactors. GIF has selected six promising
1.2 VHTR

The Very High Temperature Reactor is one of the generation IV designs. The name of the reactor betrays its main characteristic: the reactor operates at very high temperatures. There are multiple designs of this reactor type with outlet temperatures ranging from about 750°C to 1000°C. HTR designs consist of a core region which is surrounded by reflectors. The fuel is located in the core and is different from other reactor types because the fuel is combined with the moderator in the fuel containers. The fuel elements come in the shape of TRISO particles which are contained in pebbles or prismatic blocks. The coolant is a non reactive and neutronically inert gas like helium. A schematic side view of such an HTR reactor is shown in Figure 1.3.

One of the main goals of the HTR is to improve on the safety aspects of generation III reactors. Generation IV reactors should be inherently safe. This means that the reactor should be able to handle any accident scenario by passive means. In case of an accident, a situation where damage is done to the environment should never occur; all heat should be disposed of naturally without the need of any non inherent cooling mechanisms.

To accomplish this inherent safety, the HTR is designed to be able to sustain very high temperatures. The negative reactivity temperature feedback of the reactor en-
CHAPTER 1. INTRODUCTION

Figure 1.3: Schematic side view of an HTR displaying the core(1), reflector(2), steam generator(5) and helium pump(9). [18]

...sures that the fission chain reaction remains under control. The decay heat can be disposed of to the environment in a timely manner such that the decay heat does not overheat the fuel of to the environment. This heat is disposed of without the need active cooling.

One of the current HTR designs in development is the Chinese HTR-PM [17]. The idea behind this reactor is that it consists of multiple smaller reactors instead of one big one so that the power produced per reactor is limited. Another advantage of such a modular concept is that it provides the option to design the total power production of the nuclear site by selecting the corresponding number of reactors. An industry-scale prototype reactor is currently under construction which possesses two such reactors. The small reactors will generate a lot less power than a typical light-water reactor; one HTR-PM reactor is designed to produce 250MW of thermal power under normal operating conditions. This enables the core to dissipate its heat without the need of active cooling in case of an accident.

The HTR-PM is fuelled with TRISO or tristructural-isotropic particles encapsulated in graphite pebbles. These pebbles have a diameter of 6cm. One such pebble can
1.3. THORIUM FUEL CYCLE

contain about 10,000 TRISO particles. The TRISO particles consist of four protective layers which contain a fuel ball. The layers of the TRISO particle are outside to inside are shown in Figure 1.4.

![Figure 1.4: Schematic overview of the fuel of the HTR-PM. TRISO particles contain the fuel elements. Fuel pebbles contain about 10,000 of these TRISO particles. [12]](image)

These TRISO particles can withstand temperatures of up to 1600 degrees Celsius without releasing the contained radioactive fission products. To ensure that this temperature is never reached, the reactor is designed with a negative temperature feedback and a pathway for decay heat removal by passive means. Because of this negative temperature feedback, the number of fission reactions will decrease as the temperature in the reactor increases. Fewer reactions lead to a lower power production and will counteract further temperature increase. The decay of fission products still keeps producing heat. When the heat disposal of the reactor is equal to the heat generated, a steady state temperature will be reached. By choosing the design parameters, this accident steady state temperature should be lower than the TRISO failure temperature of 1600 degrees Celsius.

1.3 Thorium fuel cycle

To improve the sustainability of the nuclear reactor design, the use of thorium containing fuel elements is proposed. The use of thorium instead of uranium-235 yields several advantages: thorium yields less radioactive waste and is 3-4 times more abundant in nature [7]. Thorium-232 itself can not sustain a nuclear fission reaction. The thorium needs to be converted first into fissile U-233. This conversion is done by irradiating the thorium with neutrons. Thorium-232 captures a neutron after which it decays twice via β-decay to U-233 as is displayed in decay diagram (1.1). This conversion process is called breeding.

\[ ^{232}_{90} \text{Th} + n \rightarrow ^{233}_{90} \text{Th} \xrightarrow{\beta^-} ^{233}_{91} \text{Pa} + \xrightarrow{\beta^-} ^{234}_{92} \text{U} \] (1.1)
An HTR-PM design incorporating thorium loaded fuel elements is proposed by Wols et al. [16]. This design, based on the HTR-PM design, uses a driver zone which is surrounded by a breeder zone. Th fuel inserted in the driver zone contains 90% Th-232 and 10% U-233, while pebbles inserted in the breeder zone contain Th-232 only. In the breeder zone the Th-232 is partly converted into U-233. Later the uranium in these irradiated breeder pebbles can be processed into the driver pebbles to generate neutrons to breed more U-233.

1.4 Water ingress scenario

The HTR consists out of two cooling fluid loops. Through the primary loop helium is heated up by the reactor core after which it flows to a heat exchanger. In this heat exchanger energy is transferred to water in the secondary loop. The water in the secondary loop will vaporize and drive a turbine to generate electricity. A schematic representation of these loops and the position of the heat exchanger to the reactor core are displayed in Figure 1.3.

Water is a good moderator and therefore is used in several other reactor designs as moderator as well as a coolant. During a water ingress scenario, water from the secondary loop leaks into the primary helium loop of the reactor. This can happen when one or more pipes in the heat exchanger break.

Such a leakage can cause an extra moderating effect in the reactor core due to water, increasing the reactivity of the core. This increase in reactivity can potentially lead to dangerously high temperature levels above the failure temperature of 1600 degrees Celsius. In this case the designed negative temperature feedback and heat disposal are not strong enough to compensate for this temperature increase.

1.5 Outline

The inherent safety of a uranium fuelled HTR design has already been researched in previous studies. The goal of this thesis is to be able to give an accurate description of the inherent safety of a thorium based HTR reactor. To describe the reactor, a simple dynamic model has been developed. The model is also able to describe the water distribution in the reactor in case of a water ingress scenario. The dynamic model is applied to a reference reactor design proposed by Wols. Relevant thermodynamics and neutronics are coupled. The used equations in the model are explained in Chapter 2. The developed numerical model and the relevant state equations are described in Chapter 3. Finally the validation and the results of the model are shown in Chapter 4.

The model should be able to give a clear indication of the reaction times of the reactor to certain condition changes. The power generated by the reactor and the temperature of the fuel elements should also be closely monitored. Special attention should be given to the water ingress scenario. It is investigated if the inherent safety features of the reference design can guarantee the safety of the fuel elements. These fuel elements are considered safe up to temperatures of 1600°C.
Chapter 2

Theory

The main goal of the project is to monitor the temperature of the fuel pebbles of an HTR during condition changes and in particular in case of a water ingress scenario. This research will be carried out by creating a numerical model that can describe the temperature of the fuel pebbles at different heights in the core. The model should include all important physical phenomena that influence these temperatures. The core temperature is influenced in two ways. The temperature will increase because of the heat generated in the core while it is in operation. At the same time heat is transported away through heat exchange with the colder environment and coolant.

2.1 Point kinetics

The power generated in the core, and thus the heat produced here, is directly related to the number of fission reactions that occur. Every fission gives fission fragments a certain velocity after which the fission fragments will transfer their kinetic energy into heat. The number of fission reactions is again proportional to the free neutron density in the core. The probability of a neutron to cause fission is given by the macroscopic fission cross section, $\Sigma_f$. The fission rate, $r$, that describes the number of fissions that occur per volume per second is related to the macroscopic fission cross section by equation (2.1). Where $\Phi$ declares the neutron flux.

$$r = \Phi \Sigma_f$$

(2.1)

The one-speed neutron diffusion equation gives a good approximation of the changes in the neutron density of the core. This diffusion equation relates the change of the neutron flux to the changes in neutron production and neutron loss rates as is presented in equation (2.2) [13].

$$\frac{1}{\nu} \frac{d\phi (r, t)}{dt} = (F_p - M) \Phi (r, t) + S_d (r, t) + \Gamma (r, t)$$

(2.2)

Where $F_p \Phi$ represents the prompt neutrons created by fission. Not all neutrons are released instantly after the fission; some neutrons are released after a delay. The fraction of these delayed neutrons is denoted by $\beta$. The amount of prompt neutrons that is released by a fission reaction is thus equal to the total number of fission reactions minus the fraction of delayed neutrons. The total number of neutrons
created by fission is equal to the number of fissions times the average amount of neutrons released per fission $\nu$. The number of prompt neutrons created this way is given in equation (2.3).

$$F_p \Phi (r, t) = (1 - \beta) \nu \Sigma_f \Phi (r, t) \quad (2.3)$$

The term $S_d$ in equation (2.2) represents the amount delayed neutrons created. The delayed neutrons are subdivided over the precursor groups $k$. Each group denoting neutrons with similar decay times. The production term for delayed neutrons is simplified as in equation (2.4) with $c$ the precursor group concentration.

$$S_d (r, t) = \sum_k \lambda_k c_k (r, t) \quad (2.4)$$

$M \Phi$ represents the rate of neutron loss. Assuming that the only way to lose neutrons is for them to get absorbed in the core, this will result in:

$$M \Phi = \Sigma_a \Phi (r, t) \quad (2.5)$$

$\Gamma (r)$ describes the movement of neutrons through the reactor space as in the diffusion approximation:

$$\Gamma = \nabla^2 D \Phi. \quad (2.6)$$

Considering a point reactor with uniform parameters. Integrating equation (2.2) over the total volume of the reactor results in equation (2.8). The diffusion term in the equation cancels out because it is assumed that all neutrons stay within the core.

$$\int_V \frac{1}{v} \frac{d \Phi (r, t)}{d t} dV = \int_V \left( (1 - \beta) \nu \Sigma_f \Phi (r, t) - \Sigma_a \Phi (r, t) + \sum_k \lambda_k c_k (r, t) + \Gamma (r, t) \right) dV \quad (2.7)$$

$$\frac{1}{v} \frac{d \phi (t)}{d t} = (1 - \beta) \nu \Sigma_f \Phi (t) - \Sigma_a \Phi (t) + \sum_k \lambda_k c_k (t) \quad (2.8)$$

Using the derivation as is shown in Appendix A, this results in the well known point kinetics equations. With 6 precursor groups these equations are:

$$\dot{n} (t) = \frac{\rho (t) - \beta}{\Lambda} n (t) + \sum_{k=1}^{6} \lambda_k c_k (t) \quad (2.9)$$

with

$$\dot{c}_k (t) = -\lambda_k c_k (t) + \frac{\beta_k}{\Lambda} n (t), \quad k = 1, .., 6 \quad (2.10)$$
2.2. Reactivity

2.1.1 Nodal point kinetics

To get a more precise description of the neutron kinetics in the core, more measurement points have to be taken into consideration. This is done by dividing the core into multiple core zones each with uniform properties. In case that the reactor is divided into multiple zones, the one group diffusion equation should be integrated over the corresponding volumes of each one of these core zones as proposed by Zheng et al. [4]. For node $i$ this yields the result as in equation (2.11).

\[
\int_{V_i} \frac{1}{v} \frac{d\Phi_i (r,t)}{dt} dV = \int_{V_i} \left( (1 - \beta) \nu \Sigma_f \Phi_i (r,t) - \Sigma_a \Phi_i (r,t) + \sum_k \lambda_k c_k (r,t) + \Gamma (r,t) \right) dV
\]

\[
= \frac{1}{v_i} \frac{d\Phi_i (t)}{dt} = (1 - \beta) \nu \Sigma_f \Phi_i (t) - \Sigma_a \Phi_i (t) + \sum_k \lambda_k c_{i,k} (t) + \int_{V_i} \nabla^2 D \Phi_i (r,t) dV
\]

(2.11)

It is assumed that a particular core zone can only exchange neutrons with other core zones. There is thus no neutron exchange with other parts of the reactor. For a zone $i$ adjacent to zones $j$, the neutron transfer between the zones can be described as in equation (2.12).

\[
\int_{V_i} \nabla^2 D \Phi_i (r,t) dV = \int_{S_i} \nabla D \Phi_i (t) \hat{n} dS
\]

\[
= \sum_j A_{ij} \frac{D_j \Phi_j - D_i \Phi_i}{d_{ij}} = \sum_j A_{ij} \frac{D_j v_j n_j - D_i v_i n_i}{d_{ij}}
\]

(2.12)

Where $A_{ij}$ is the area between zones $i$ and $j$.

Using the same substitutions as in (2.9), will give the nodal point kinetics equations for core zone $i$ in (2.13) and (2.14).

\[
\dot{n}_i (t) = \frac{\rho_i (t) - \beta}{\Lambda} n_i (t) + \sum_{k=1}^6 \lambda_k c_{k,i} (t) + \sum_j A_{ij} \frac{D_j v_j n_j - D_i v_i n_i}{d_{ij}}
\]

(2.13)

\[
\dot{c}_{k,i} (t) = -\lambda_k c_{k,i} (t) + \frac{\beta_k}{\Lambda} n_i (t), \quad k = 1, \ldots, 6
\]

(2.14)

2.2 Reactivity

In the point kinetics equations the reactivity $\rho$ is of great importance. This variable is related to the effective multiplication factor $k$ by equation (2.15). The effective multiplication factor is a measure of the total neutron balance between two fission generations. This factor relates the amount of neutrons in one generation to the amount of neutrons in the next generation as in equation (2.16) [5].

\[
\rho = \frac{k_{eff} - 1}{k_{eff}}
\]

(2.15)
CHAPTER 2. THEORY

\[ k_{\text{eff}} = \frac{\text{#neutrons current generation}}{\text{#neutrons preceding generation}} \]  (2.16)

The effective multiplication factor and also the reactivity of the reactor are temperature dependent. Corresponding cross sections vary with temperature. It is a realistic assumption that other variables in the nodal point kinetics equation (2.13) are also temperature dependent. In the model all these temperature dependencies are lumped together in the reactivity temperature feedback. To guarantee the reliability of a model based on the nodal kinetics equations it is thus of great importance to have a good grip on the reactivity feedback. The relation between the reactivity and temperature is calculated with already developed models. Such a reference model is developed by Wols [16]. This model incorporates CSAS, SCALE 6.0 and DALTON codes to calculate the properties of a reactor in critical steady state configurations.

All other core parameters in the nodal point kinetics equations are calculated for the operating critical configuration. They are responsible for the relation between the prompt and delayed neutrons as well as the relations between different core zones. These relations are thus assumed to be constant for varying temperatures. The temperature dependencies of the neutron density distribution in equation eqrefnodalpointkinetics are all lumped together in the reactivity feedback this way.

The reactivity can be calculated by varying one of the temperatures of the fuel, moderator or reflector while keeping the other sections at temperatures corresponding to the original control configuration. This way the influences of the temperatures of the separate sections on the reactivity can be calculated.

Because the reactivity coefficients of the separate sections are assumed to be independent of each other, a total reactivity coefficient can be calculated based on several separate reactivity relations as in equation (2.17) [9].

\[ \rho_{\text{tot}} = \rho_{\text{fuel}} + \rho_{\text{moderator}} + \rho_{\text{reflector}} \]  (2.17)

### 2.3 Coupling coefficients

In the nodal neutron kinetics equations (2.13), the neutron flux between different core zones is taken into account. This neutron flux can be divided in a part for incoming neutrons and a part for outgoing neutrons through the surface \( A_{ij} \) as in equation (2.18). Conversions as are described in Appendix A are applied.

\[
\sum_j A_{ij} \frac{D_j v_j n_j - D_i v_i n_i}{d_{ij}} = \sum_j \frac{D_j A_{ij} n_j}{A_j v_j \Sigma_{f,j} d_{ij}} - D_i n_i \sum_j \frac{A_{ij}}{d_{ij} A_j v_j \Sigma_{f,j}}
\]  (2.18)

In this simplistic model, only the neutron density, \( n \), is assumed to be varying with temperature. All the remaining parameters can be grouped together for every core zone. These grouped constants are called the coupling coefficients \( \alpha \). These coupling
coefficients are then responsible for the neutron distribution in the core in case the whole reactor is held at the same temperature.

Considering several core zones stacked on top of each other. In this case every core zone can only exchange neutrons with a maximum of two adjacent zones; one above and one below. Every core zone thus has one or two coupling coefficients. These coupling coefficients are called $\alpha_{\text{up}}$ for the upward flowing neutrons and $\alpha_{\text{down}}$ for the downward flowing neutrons. This model is very similar to a 1D heated rod with heat dissipating from the ends of the rod. It is expected that for constant neutron production over all the core zones, or constant reactivity, the highest neutron density will be found in the middle core zone.

Between two core zones $i$ and $i+1$ a neutron current density, $j_+$, will flow from core zone $i$ to core zone $i+1$. Another current will flow in the opposite direction, $j_-$. A net neutron current, $J_{i,i+1}$, between these two zones can then be denoted by equation (2.19).

$$J_{i,i+1} = j_+ - j_- = \alpha_{\text{down},i+1} n_{i+1} - \alpha_{\text{up},i} n_i \quad (2.19)$$

### 2.3.1 Neutron loss

Knowing the coupling coefficients does not completely describe the neutron distribution. The coupling coefficients account for the neutron flow in the reactor core in the same way heat conduction accounts for the heat flow in a heated rod. The temperature profile only becomes apparent when the heat loss at the ends of the rod is specified. In the same way the reactor loses neutrons to the top and bottom part of the core. The top and bottom part of the core are capped of with a reflector. The goal of this reflector is to reduce the neutron loss. Neutron loss is not reduced to zero however.

A derivation of the diffusion theory expression of both upward and downward neutron currents is formulated by Duderstadt and Hamilton [15]. The results for the upward current density, $j_+$, and the downward current density, $j_-$, are shown in equations (2.20) and (2.21).

$$j_+ (0) = \frac{1}{4} \phi (0) - \frac{1}{2} D \frac{d \phi (0)}{d x} \quad (2.20)$$

$$j_- (0) = \frac{1}{4} \phi (0) + \frac{1}{2} D \frac{d \phi (0)}{d x} \quad (2.21)$$

Discretizing these equations to account for lumped core zones with uniform properties results in equations (2.22) and (2.23).

$$j_+ (i) = \frac{1}{4} \phi (i) - \frac{1}{2} D_i \frac{\phi (i + 1) - \phi (i)}{d} \quad (2.22)$$

$$j_- (i) = \frac{1}{4} \phi (i) - \frac{1}{2} D_i \frac{\phi (i - 1) - \phi (i)}{d} \quad (2.23)$$
The number of neutrons lost to the top reflector can now be calculated by calculating the upward neutron flow in the upper most core zone and the downward neutron flow from the top reflector to this zone. The difference is the number of neutrons that are never reflected and thus lost. To exclude the reflector from further calculations in the model, it is assumed that the fraction of neutrons lost to the reflector is constant related to the number of neutrons that flow to the reflector. This means that the neutron loss to the reflectors is only dependent on the neutron densities in the top and bottom core zones and not on the neutron densities in the reflectors.

\[
J_{\text{top,refl}} = j_+\text{(top)} - j_-\text{(refl)} = j_+(\text{top}) - j_+(\text{top}) \ast \text{loss} = j_+(\text{top})(1 - \text{loss}) \quad (2.24)
\]

During calculations of the reactivity, the neutron loss to the reflectors surrounding the core has already been taken into account. To avoid subtracting the lost neutrons to the top and bottom reflector twice in the calculations, a term has to be introduced to compensate for the lost neutrons to the top and bottom reflector. The amount of neutrons lost to the top and bottom reflector is equal to the number of neutrons leaving the core zones minus the number of neutron reflected into the core zones in equation (2.18).

These neutrons are then reinserted in the core in the model. They are distributed proportional to the neutron densities in every core zone.

### 2.4 Thermodynamics

Because of the fission reactions that occur in the core, heat is generated. This heat has to be removed from the core or otherwise temperatures will keep rising. The heat is transferred to the reflector and the coolant fluid via conduction. Then the heat is transferred away via convection in the coolant flow. Heat transfer through radiation has been neglected.

To simply describe the thermodynamic behaviour of the reactor, the power plant is nodalized; the reactor is divided into several nodes. Every node has uniform properties and behaves uniform by changing conditions. The interactions between the nodes can then be modelled with simple relationships. The heat flow between nodes is based on the conservation laws of energy, mass and momentum (2.25)(2.26)(2.27) as proposed by Li et al. [9].

\[
\dot{M} = G_{in} - G_{out} \quad (2.25)
\]

\[
\dot{E} = G_{in}h - G_{out}h + Q + P \quad (2.26)
\]

\[
\dot{p} = M_{in}v_{in} - M_{out}v_{out} + F \quad (2.27)
\]

The flow of the coolant through the reactor can be described by the conservation law of mass (2.25). The simplification of no mass build up in the reactor is assumed. The output mass flow in the vessel can then be related to its input mass flow according to:
2.4. THERMODYNAMICS

\[ \dot{M} = 0 = G_{in} - G_{out} \]
\[ G_{out} = G_{in}. \]

The change of temperature in a node can be described by the law of conservation of energy. Temperature and heat are related by the specific heat.

\[ \dot{E} = M c_p \dot{T} \quad (2.28) \]

Now a relation for the temperature can be derived:

\[ \dot{T} V \rho c_p = G h_{in} - G h_{out} + P + Q. \quad (2.29) \]

Where the enthalpy can be related to temperature according to:

\[ \dot{h} = c_p \dot{T} + \frac{d}{dt}(pV), \quad (2.30) \]

and for constant volume and pressure this simplifies to:

\[ \dot{h} = c_p \dot{T}. \quad (2.31) \]

Newton’s law of cooling can describe the heat conduction between different nodes with a temperature difference \( \Delta T \) as in equation (2.32).

\[ \frac{Q}{A} = h \Delta T \quad (2.32) \]

with heat transfer coefficient \( h \):

\[ h = \frac{\lambda}{D}. \quad (2.33) \]

The heat transfer coefficient is dependent on the substance and the temperature of the object the heat is transferred through. The relation of the heat transfer coefficient to temperature can be empirically estimated and implemented in the model.

An empirical relationship for the Nusselt number, \( Nu \), of the fuel pebbles is proposed by Fenech [6] in equation (2.34).

\[ Nu = \frac{0.037 (Re)_{0.8} Pr}{1 + 2.443 (Re)^{-0.8} (Pr^{2/3} - 1)} \quad (2.34) \]

By substituting equations (2.31) and (2.32) in equation (2.29), a differential equation for the temperature of a node is obtained:

\[ \dot{T} = \frac{1}{V \rho c_p} (G c_p \Delta T + h A \Delta T + W) \quad (2.35) \]

The pressure drop can be described as in equation (2.36) [11].

\[ \Delta p = \frac{1}{2} \rho V^2 \left( \frac{4 f L}{D} \right) \quad (2.36) \]
2.5 Water ingress

During a water ingress scenario, water from the secondary loop flows into the primary coolant loop. According to Lohnert [10], water ingress in the primary loop poses two dangers. One problem is that the power output of the reactor can increase to intolerable levels due to the moderating effect of the water. The other is that the steam together with hot graphite can be converted into potentially dangerous hydrogen gas. The corrosion of graphite is neglected in this research. Instead the focus will be on the increase of temperature related to a water ingress scenario.

![Figure 2.1: Schematic representation of the secondary water loop. [1]](image)

The primary loop is connected to the secondary loop by a heat exchanger. The loops are schematically presented in Figure 2.1.

In case of a guillotine break of one of the steam generator tubes, water will flow into the primary circuit with 6kg/s out of either side of the cut tube. The resulting ingress rate is thus 12kg/s [10]. After 10 seconds the safety measurements of the reactor will be activated. During such a design basis accident, the pump is shut down, a helium purification facility extracts the water out of the coolant gas of the system and the primary loop is disconnected from the secondary loop to prevent more water from ingressing into the primary loop. After the loops are separated, a maximum of 600 kg of steam has penetrated in the coolant loop.

It is possible for a situation to occur where only a few or none of the active safety measurements will kick in to reduce the impact of the ingressed water. Such an accident is denoted a beyond design basis accident. A total of 4000kg of water and steam is located in the secondary loop. During a beyond design basis accident, a maximum of 2500kg of this water mass can potentially vaporize and ingress into the primary loop [10].

After the water has ingressed in the primary loop, the water will spread through the loop alongside the helium flow. The speed with which the water will spread through the system can be simulated by assuming that the reactor can be divided into several perfectly mixed vessels. The change of the mass of water, $\dot{M}$, in each of
2.5. WATER INGRESS

these vessels can then be calculated according to:

\[ \dot{M}_i = \phi_{V,i-1} \rho_{w,i-1} - \phi_{V,i} \rho_{w,i}, \]  

(2.37)

where \( \rho_{w,i} \) is the water density in vessel \( i \), while \( \phi_{V,i} \) is the volume flow in vessel \( i \).

The volume flow \( \phi_V \) is derived from the constant mass flow and temperature dependent densities of the helium and water as in equation (2.38).

\[ \phi_V = \frac{G}{\rho} = \frac{G_w}{\rho_w} + \frac{G_{he}}{\rho_{he}} = \frac{G}{M_W + M_{he}} \left( \frac{M_W}{\rho_W} + \frac{M_{he}}{\rho_{he}} \right) \]  

(2.38)

The extra reactivity coefficient as a result of water ingress can be included in the total reactivity coefficient:

\[ \rho_{tot} = \rho_{fuel} + \rho_{moderator} + \rho_{reflector} + \rho_{water}. \]  

(2.39)

2.5.1 Water density

Once the water is ingressed in the reactor, it is subject to conditions with a pressure of 7MPa to 10MPa and temperatures above 900°C. Under these conditions the water will be in its gas phase.

This water vapour has a temperature dependent density. The relation between this density and the temperature of the vapour is given by equation (2.40) [8].

\[ \rho_w = 2.165 \cdot 10^{-3} \rho/T \]  

(2.40)
Chapter 3

Numerical Model

To understand how the physical relations described in the previous chapter influence each other, a dynamic model has been developed. Such a dynamic model is able to model reactor behaviour over time to certain parameter changes. To simplify the model, the reactor is divided into several nodes which all have uniform properties. The model relies on a set of state equations which define the change in state variables of the chosen nodes. These differential equations can then later be evaluated to simulate reactor behaviour over time.

3.1 Reference reactor

A thorium breeder reactor based on the HTR-PM is used as a reference for the model. All dimensions of the HTR-PM are assumed while fuel properties of a thorium breeder reactor are implemented. The core is 11m in height and 3m in diameter. A long core like this helps dissipate heat in case of an accident because of the larger surface area. The remainder of the dimensions of the reactor can be found in Appendix B.

3.1.1 Breeder reactor

The reference reactor is a breeder reactor. In the reactor thorium-232 is bred into fissile uranium-233. In the reference reactor a distinction is made between a breeder zone and a driver zone. The driver zone is located in the centre of the reactor and generates an excess of neutrons. The excess neutrons flow to the breeder zone where there is a need for extra neutrons to be able to breed. The driver zone is dimensioned as a cylinder with a radius of 0.9 meters. The breeder zone is a hollow cylinder with an inner radius of 0.9 meters and an outer radius of 1.5 meters which encapsulates the driver zone.

Making use of the model of Wols, the multiplication factor of this reactor design can be calculated for varying conditions. This way relations between the temperatures of the fuel, moderator and reflector and the multiplication factor are obtained. The resulting relations are plotted in Figure 3.1.
3.2 Nodal model

A nodalization of the reactor is proposed by Sanchez [14]. The core is split up in the nodes: lower plenum, lower header, riser, upper header, downcomer, core, reflector and outlet header. All these nodes are modelled as uniform vessels. This nodalization scheme is represented in Figure 3.2.

The helium mass flow can be calculated in every node according to the law of conservation of mass (2.25). The helium circulates through the nodes one by one. The only deviation is a leakage in the lower header because the outlet for the pebbles is located over there [1]. The temperature of the helium rises according to the added heat in equation (2.31). Heat transfer by conduction is possible between: riser and reflector, reflector and core, core and helium. This conductive heat transfer is calculated via equation (2.32).
3.2. NODAL MODEL

3.2.1 Core zones

The core is the most interesting node because here the temperatures will reach the highest values and the fuel elements should be protected against temperatures higher than 1600°C. To get a more detailed temperature profile of the core, the core is split into multiple core zones which all behave as nodes on their own. The result is that the temperatures of all these zones can be calculated separately.

Figure 3.2: Reactor divided in several lumped nodes with corresponding mass flow and heat conduction between the nodes.
The $N$ core zones are selected such that they are stacked on top of each other as in Figure 3.3. For convenience, the coolant, reflector and riser are separated in the same manner.

A vertical stack of core zones is selected because the biggest variation in temperature is expected in this direction. Radial properties are assumed uniform in this research. A total of 11 core zones are chosen. All core zones have equal dimensions of 1m in height and 1.5m in radius. Choosing equally sized core zones simplifies calculations later on. The dimensions chosen here also simplify comparisons with the model by Wols. The core zones cover the heights of the reactor as in Table 3.1.

<table>
<thead>
<tr>
<th>core zone</th>
<th>height (m)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>10-11</td>
</tr>
<tr>
<td>2</td>
<td>9-10</td>
</tr>
<tr>
<td>3</td>
<td>8-9</td>
</tr>
<tr>
<td>4</td>
<td>7-8</td>
</tr>
<tr>
<td>5</td>
<td>6-7</td>
</tr>
<tr>
<td>6</td>
<td>5-6</td>
</tr>
<tr>
<td>7</td>
<td>4-5</td>
</tr>
<tr>
<td>8</td>
<td>3-4</td>
</tr>
<tr>
<td>9</td>
<td>2-3</td>
</tr>
<tr>
<td>10</td>
<td>1-2</td>
</tr>
<tr>
<td>11</td>
<td>0-1</td>
</tr>
</tbody>
</table>

Table 3.1: Core zone heights
3.3 Water ingress

The primary circuit of the reactor through which the cooling helium flows is connected to a secondary steam loop via a heat exchanger. A schematic representation of these loops is shown in Figure 2.1.

The ingressed water does not significantly influence any physical properties outside the core except for the flow speed of the coolant. Once the water is inserted into the core, the reactivity will be influenced because of the extra moderating effect. Because the most important parameter is the amount of water that is located in the core, the model addresses the amount of water in every core zone separately as well as the amount of water in the primary loop before and after the core. The collection of vessels located before the core are denoted as the Inlet Header while the vessels after the core are denoted as the Outlet Header as in Figure 3.4.

![Figure 3.4: Schematic representation of the primary loop of the chosen water ingress model.](image)

The Inlet and Outlet Headers are assumed to be perfectly mixed vessels. The volume of the Inlet Header corresponds to the combined volumes of the lower plenum, lower header and upper header. The volume of the Outlet Header is equal to the volume of the outlet header of the nodal model.

Water and helium will flow from the secondary loop into the Inlet Header. From there the water flows alongside the helium through the core and Outlet Header back to the steam generator according to equation (2.37). From there the water and helium can be directed to the helium purification facility where part of the water is extracted out of the helium. If the flow is not directed to the purification facility, it will be recirculated into the inlet header. Both situations can be simulated.
The mass flow of the water in every node is calculated according to equation (2.37).

The extra moderating effect of the water is calculated with the model of Wols. The multiplication factor is plotted against the water density in the whole core in Figure 3.5. It can be seen that at first the multiplication factor rises as more water is ingressed due to the extra moderating effect of the water. At about 20kg/m$^3$ more water actually has a negative effect on the effective multiplication factor. Too many neutrons are captured by the water at this point.

The water ingress effect for different temperatures is displayed in Figure 3.6. A higher temperature still has a negative effect on the multiplication factor.

![Figure 3.5: Effective multiplication factor plotted as a function of the water density in the core at a uniform core temperature of 1100K.](image-url)
3.4 State equations

All relevant physical phenomena can be described by a set of state equations. These state equations are differential equations that describe the temperatures of all the nodes of the reactor, neutron and precursor densities of all the core zones as well as the water vapour densities of the relevant nodes. The temperature state equations are given by equation (2.35). The temperatures of the 11 core zones, lower plenum, riser, upper header, coolant, reflector and outlet header are described this way. The state equations for the neutron and precursor densities are given by equations (2.13) and (2.14) for every core zone \( i \) and every precursor group \( k \). The water masses in the core zones as well as in the Inlet and Outlet Headers are determined by state equation 2.37. All these equation together result in the system displayed in matrix (3.1).
\[
\begin{bmatrix}
\dot{T}_{lp} \\
\dot{T}_{ol} \\
\dot{T}_{c,i} \\
\dot{T}_{refl,i} \\
\dot{T}_{r,i} \\
\dot{T}_{cl,i} \\
\dot{\dot{n}}_i \\
\dot{c}_{k,i} \\
\dot{M}_{IH} \\
\dot{M}_{i} \\
\dot{M}_{OH}
\end{bmatrix} =
\begin{bmatrix}
\frac{1}{\frac{\rho_{lp}}{c_p}}G_{lp}c_p(T_{il} - T_{lp}) \\
\frac{1}{\frac{\rho_{ol}}{c_p}}(G_{ol}c_p(T_{cl,end} - T_{ol})) \\
\frac{1}{\frac{\rho_{c,i}}{c_p}}(G_{c,i}(T_{c,i-1} - T_{c,i}) + P_i + hA(T_{refl,i} - T_{c,i})) \\
\frac{1}{\frac{\rho_{refl,i}}{c_p}}(hA(T_{c,i} - T_{refl,i}) + hA(T_{r,i} - T_{refl,i})) \\
\frac{1}{\frac{\rho_{r,i}}{c_p}}(hA(T_{refl,i} - T_{r,i})) \\
\frac{1}{\frac{\rho_{cl,i}}{c_p}}(G_{r,c,i}(T_{cl,i-1} - T_{cl,i}) + hA(T_{c,i} - T_{cl,i})) \\
\frac{n_i(\rho - \beta)}{\lambda} + \sum_j \frac{A_{ji}}{\lambda_{di,j}} \left( \frac{D_{ji}n_j}{\nu_j\Sigma_{j,j}} - \frac{D_{ji}n_j}{\nu_j\Sigma_{j,j}} \right) + \sum_{k=1}^{6} \lambda_k \dot{c}_{k,i} \\
-\lambda_k \dot{c}_{k,i}(t) + \frac{\beta_{k,i}}{\lambda}n_i, \quad k = 1, 6 \\
W_{in} + \int \rho_{w,OH}\phi_{V,OH} - \rho_{w,IH}\phi_{V,IH} \\
\rho_{w,i} - \rho_{w,i-1}\phi_{V,i} - \rho_{w,i}\phi_{V,i} \\
\rho_{w,\text{end}}\phi_{V,\text{end}} - \rho_{w,OH}\phi_{V,OH}
\end{bmatrix}
\]
Chapter 4

Results

In this chapter the developed dynamic model is used to simulate the thorium based HTR-PM under several conditions. First it is tested if the model is able to reach a steady state critical condition. Once this condition is found, the results of the model can be compared to results of other models that simulate these critical configurations to validate the dynamic model.

This dynamic model is also used to simulate the time response to condition changes starting from the steady state situation in the reactor. This way an estimation of the inherent safety and reaction times to condition changes can be made.

4.1 Steady state condition

The dynamic model has incorporated negative temperature feedback coefficients. If the feedback coefficients are strong enough and the initial conditions are well chosen, a steady temperature will be met after some amount of time. To find this steady state condition, initial conditions of the HTR-PM were inserted in the model. Once temperature fluctuations were minimized to a tenth of a degree, a steady situation was assumed. In this chapter, results are shown which describe different steady state situations corresponding to different model parameters.

4.2 Power distribution

The feasibility of the power distribution of the dynamic model is tested. The power distribution can easily be compared to other models and can give an estimation of the reliability of the dynamic model. The power generated is different for all the core zones. This is a result of the temperature dependent reactivity and the coupling coefficients which describe the neutron distribution over the core zones. Also because the pebbles are circulated through the core, the specifications of these fuel pebbles vary with height as they burn up. The results of the influences by the neutron currents between the core zones and the temperature dependent reactivity coefficients will be independently calculated and compared to the model of Wols where possible.

4.2.1 Neutron current

In case the core temperature is the same for all core zones, the power profile is defined by the coupling coefficients. The coupling coefficients are calculated as in
equation (2.18). With the loss terms to the reflectors as in equation (2.24). The calculated loss terms are:

<table>
<thead>
<tr>
<th>Reflector</th>
<th>Loss</th>
</tr>
</thead>
<tbody>
<tr>
<td>Top reflector</td>
<td>0.4316</td>
</tr>
<tr>
<td>Bottom reflector</td>
<td>0.1696</td>
</tr>
</tbody>
</table>

The fraction of neutrons lost to the top reflector is thus significantly higher than the fraction lost to the bottom reflector. This result can be explained by the differing properties of the top and bottom reflector. The bottom reflector is there only to reflect neutrons as best as possible. The top reflector has the same goal, but it is also the inlet for the helium gas to the core. This top reflector is more porous to be able to distribute the helium in the best possible way. Because of the extra porosity, the reflecting capabilities are not as adequate.

The neutron currents of the dynamic model can now be compared to the neutron currents between the core zones by Wols without temperature feedback. The neutron currents in the model of Wols can be calculated according to equation (2.22), while the neutron currents of the dynamic model are directly related to the coupling coefficients. The resulting normalized neutron currents are shown and compared in Figure 4.1. Both models show a similar pattern of neutrons flowing from the middle of the core to the top and bottom. The neutron current in the bottom of the core is also slightly lower than the current in the top of the core. Though the models show similar patterns, the results vary slightly from each other with a relative root mean squared error of 0.09.
The power distribution is not dependent on temperature. A comparison is made between the dynamic model and the model of Wols. The power profile as a result of the coupling coefficients is displayed in Figure 4.2. The relative power of all the core zones is displayed for the dynamic model and the model of Wols. As expected, the highest concentration of neutrons can be found in the middle core zones. It can also be noted that the neutron concentrations in top core zones are lower than in the bottom core zones. This is a result of the higher loss to the top reflector. The relative root mean squared error between the power profiles is 0.03. Differences in fuel have been taken into account in the model of Wols while this is not the case in the dynamic model. Typically the fuel is more depleted the lower it is situated in the core. This means that the power output is slightly higher in the top parts of the core compared to the bottom parts of the core. This may explain some of the observed deviations.
4.2.2 Reactivity

The power distribution can also be calculated depending on the reactivity which can vary over the height of the core. In this case the neutron loss to the top and bottom reflectors are not taken into account. The reactivity will be individually calculated for all core zones. Neutron exchange between core zones is still possible. Because the reactivity is temperature dependent, the power distribution will also be temperature dependent. The power distribution obtained this way is shown in Figure 4.3. The corresponding temperature profile is displayed in Figure 4.4. The graphs match up in that the highest energy production is found in the top core zones. These top zones are the colder zones because cold helium is inserted there. Also, in the lower core zones the power production is comparatively low. This results in less temperature difference between the core zones over there.
4.2. POWER DISTRIBUTION

![Bar chart showing power distribution as a result of temperature dependent reactivity.](image1)

**Figure 4.3:** Power distribution as a result of the temperature dependent reactivity.

![Bar chart showing temperature profile core as a result of temperature dependent reactivity.](image2)

**Figure 4.4:** Temperature profile core as a result of temperature dependent reactivity.
4.2.3 Combined neutron current and temperature dependent reactivity

The effects of the temperature dependent reactivity and the neutron current through the core zones as a result of neutron loss to the top and bottom reflectors can also be combined.

The normalized neutron currents are compared to the model of Wols with a temperature feedback in Figure 4.5. The currents transport neutrons away from the middle of the core as in the simulation without temperature feedback. This time the current to the top cores is significantly higher than the current to the bottom of the core for both models. This is the result of the higher neutron density in the top of the core. The relative MSE between the models is 0.10. This rMSE is slightly higher than the rMSE between the models without temperature feedback.

![Figure 4.5: Power distribution in the core under steady state conditions. Temperature dependencies have been taken into account.](image)

Combining both influences on the neutron distribution results in the power distribution as shown in Figure 4.6. This power profile has the highest power output in node 3. This node covers the part of the core which is situated between 8 and 9 meters in height. This power profile can be compared to the power profile of the temperature dependent model of Wols. Both models match up quite nicely in that they have the same form. The highest neutron density is also found in node 3 in the model of Wols. The models have a relative rMSE of 0.01.
4.3 Water ingress

In this section the results of the water ingress model will be shown. It is important that the simple ingress model still captures all the relevant physics. This is why the model will first be tested under certain simplified conditions.

4.3.1 Constant density

To get a clear view of what is actually going on in the model, the density is assumed constant at first. Normally the density of water is temperature dependent. At first it is also assumed that the water will stay in the coolant loop instead of gradually being extracted out of the coolant by the helium purification facility. This way the distribution of a certain amount of water in the primary loop can be described. Water is inserted in the reactor with a speed of 12kg/s for 50 seconds from the 10 second mark. The resulting water distribution throughout the reactor is shown in Figure 4.7. The water will distribute itself over the Inlet Header, Outlet Header and the core zones according to their respective volumes. The total water mass in the primary loop will stay at 600kg because water is not extracted out of the coolant loop. Most of the water will stay in the nodes that are included in the Inlet Header because this vessel has the biggest volume. It should be concluded that for every kilogram of water ingressed only a small portion is actually located in the core where it is most dangerous.

Figure 4.6: Power distribution as a result of both temperature dependent reactivity and neutron loss to the top and bottom reflector.
A close up of the water distribution over the core zones is shown in Figure 4.8. Only the core zones 1, 6 and 11 are included to keep the image clear. It can be seen that the water will first enter the top zone after which it flows into the lower zones. After the ingress has stopped, the water is evenly divided over the core zones. The velocity with which the cooling fluid flows through the core is approximately 72 m$^3$/s while the volumes of the core zones are 2.76 m$^3$. This explains the short time in which the water distributes itself over the core zones.

The same calculations can be executed in case the water is immediately extracted out of the loop once it has run through the core. In this situation, a steady state will be reached where the water ingress in a core zone is equal to the water flow that transports water out of this zone. A maximum amount of water per core zone will thus be reached once this equilibrium is met. The result is shown in Figure 4.9. As can be seen, the maximum amount of water in the core zones is very low in case the purification facility is engaged. After the water ingress has stopped, the water is drained from the core zones in a few seconds.
4.3. WATER INGRESS

Figure 4.8: Water distribution in the core zones 1, 6, 11. A total of 600kg of water is ingressed in the reactor in 50 seconds.

Figure 4.9: Water distribution in the core zones 1, 6, 11. Water is ingressed at a rate of 12kg/s from the 10 second mark for 50 seconds.
4.3.2 Temperature dependent density

A more realistic model incorporates the temperature dependency of the density of water as is declared by equation (2.40). In this case the water will flow faster as the core temperature increases because of the expanding water vapour. The resulting distribution in case the water is extracted out of the system is displayed in Figure 4.10. Most of the water mass is found in the top core zone. This is explained by the fact that this core zone is the best cooled zone. This lower temperature results in a higher density and slower flow speed. More water can be stored in this volume this way.

![Image of Figure 4.10: Water distribution in the core zones 1, 6, 11. Water is ingressed at a rate of 12kg/s from the 10 second mark for 50 seconds. The water vapour is compressible.]

Now the case that the water is not extracted out of the coolant gas is looked at. This case results in the water distribution described by Figure 4.11. In this scenario, most of the water is found in the top core zone as well. The models with a temperature dependent water density are more realistic than the models which neglect this density. At the same time these models have the water distributed unevenly over the core zones, which possibly results in a more dangerous situation in the core zones with more water as will be shown in Section 4.5. The models where the water is not extracted out of the helium generally result in more water in the core zones compared to the models where the water is extracted.
4.4 Dynamic model predictions on condition changes

Now that the dynamic model is tested on reliability, it is time to let the model calculate some scenarios. In the following scenarios, the temperature of the coolant, pressure, coolant flow and the insertion of control rods are simulated. The main goal is to get a grip on what time scales and temperature scales are associated with certain condition changes. Similar research has been done by Rodriguez for one core zone [14]. In this dynamic model a more accurate description of the core is obtained because of the more detailed core modelling in multiple zones. In the following subsections a change in one of the conditions is simulated at a time. The change will be applied after 10 seconds of steady state operation.

4.4.1 Change in inlet temperature

The inlet temperature is the temperature of the helium when it flows from the heat exchanger to the inlet header. Under normal conditions, the inlet temperature of the helium is 523K. Changing the inlet temperature will change the cooling capabilities of the reactor. As a result high temperatures or a high reactivity increase can potentially be obtained. In Figure 4.12 an increase of the inlet temperature by 50% is simulated. The total power generated by the core did decrease by 53% as a result. It can be seen that the higher zones experience a temperature increase because of the reduced cooling. At the same time the lower zones experience a temperature decrease because of the lower energy production as a result of a lower reactivity. A new steady state condition is approximately reached at the 400 second mark. In the end the reactor is in no danger because of the inherent negative feedback.
In the same manner a 50% lower inlet temperature is simulated. This results in Figure 4.13. An opposite effect is achieved. In the top core zones the temperature is lowered and as a result the reactivity has increased. Because of this reactivity increase, 58% more power is put out and the temperature of the cooling helium is increased accordingly. This then results in a temperature increase of the lower core zones. This situation is more dangerous than an increase of the inlet temperature because temperatures can get too high. The reaction time is in the order of 100 seconds.

This is an interesting result when comparing to the results of Rodriguez [14]. In the dynamic model of Rodriguez it is shown that a decrease of the inlet temperature will result in a decrease of the core temperature as well. In the dynamic model of this research it is actually found that the overall temperature of the core increases due to the reactivity increase in the top part of the reactor.
4.4. DYNAMIC MODEL PREDICTIONS ON CONDITION CHANGES

4.4.2 Change in pressure

Under normal circumstances the pressure of the inlet helium is 7MPa. In this section changes of the pressure are simulated. The pressure can increase up to 70MPa before a pressure release valve is opened in the reactor [10]. An increase by a factor of 10 in the inlet helium pressure is displayed in Figure 4.14 while a decrease of 50% in the inlet helium pressure is shown in Figure 4.15. It should be noted that the helium flow is not changed as a result of this inlet pressure adjustment. As was also discovered by Rodriguez, changes in inlet helium pressure have negligible effects on the dynamic model.
Figure 4.14: An increase of the inlet helium pressure by a factor of 10 is simulated after the 10 second mark for 11 core zones.

Figure 4.15: A decrease of the inlet helium pressure by 50% is simulated after the 10 second mark for 11 core zones.
4.4.3 Change in coolant flow

The coolant flow is regulated by a pump which is located near the heat exchanger. The regular coolant flow imposed by this pump is 96kg/s. In this section changes in this helium inlet flow are investigated. An increase of the helium flow is especially interesting because this has potential influence on the water ingress scenario. In the case helium is pumped around and extra water is ingressed at the same time, the total input flow is actually increased. This can be simulated by a higher helium input flow. A decrease in helium input flow can simulate the failing of the pump. An increase of the input helium flow by 100% is simulated in Figure 4.16. The power of the reactor increased with 98% in the process. The effect on the core temperature is minimal again just as in the model of Rodriguez. Even a decrease of the helium flow by a factor of 10 as is shown in Figure 4.17 has a very small impact on the steady state temperature. In this case the power output decreased by 83%.

In case of a helium flow increase, the power output will increase because of the higher reactivity in the top core zones. The opposite happens in case of a helium flow decrease. Another effect is the response time to condition changes. Increasing the helium flow distributes the heat faster over the core while a lower helium inlet flow will increase the time it takes to adjust to a new power profile. This can be seen as in Figure 4.16 the new steady state situation is reached quite fast compared to the situation in Figure 4.17.

Figure 4.16: An increase of the inlet helium flow by 100% is simulated after the 10 second mark for 11 core zones.
4.4.4 Change in reactivity by control rods

The control rods have the ability to absorb neutrons out of the reactor. By insertion of the control rods more neutrons can be absorbed while less neutrons are absorbed by pulling the control rods back out. As can be seen in Figure 4.18, a decrease of the reactivity by 0.5$ yields a respectable result. The temperature of the hottest core zone dropped by 100K in a time span of about 250 seconds while the power of the reactor decreased by 10%.
4.5 Water ingress

In this section it will be investigated if a water ingress scenario can cause the temperatures in the core to reach the dangerous range above 1600°C. First the design basis accident is tested on safety. In this accident the secondary loop is separated from the primary loop after a maximum of 600kg of water has ingressed. The temperatures are investigated under the conditions of a failing purification facility, a failing pump or both.

Beyond design basis accidents are also investigated. During these situations a lot more water can enter the primary loop. A total of 2500kg can flow from the secondary loop in the primary loop in case these loops are not separated properly. Research has been done to find the total amount of water that can ingress in the core while still guaranteeing the safety of the reactor. Because the ingress rate of the water is considered constant, the amount of water ingressed is proportional to the time it takes to separate the primary and secondary loop.

The effect of differences in reactivity due to pressure changes are neglected because the influence of pressure changes on the core temperatures are minimal. The effects of the insertion of control rods and the decrease of inlet coolant temperature are also neglected. This because these effects will only influence the core temperature in a positive way; the temperatures will drop. A less favourable situation is investigated this way.
4.5.1 Design basis accident

During the design basis accident the purification facility is engaged and the pump is disabled after the water has ingressed. After 50 seconds a maximum of 600kg has ingressed into the primary loop. The water is extracted out of the primary loop after it has passed through the core once. The resulting temperature increase of the core zones is displayed in Figure 4.19.

The maximum temperature achieved is 1250K. This temperature is well below the maximum safe temperature of 1873K. With all safety mechanisms working, a water ingress scenario can thus be controlled nicely. In about 100 seconds after the water ingress has finished, the maximum temperature has been reached after which the reactor will start to stabilize.

Figure 4.19: A total of 600 kg of water has ingressed into the primary loop. The purification facility is working while the pumps are disabled. The dotted line represents a temperature of 1873K.

A situation where it is not possible to disable the helium pump is shown in Figure 4.20. The maximum temperature achieved is 1230K. The core also quickly cools down again in about 100 seconds. The maximum temperature is actually lower when the pump is enabled. This is probably a result of the faster water drainage.
4.5. WATER INGRESS

Figure 4.20: A total of 600 kg of water has ingressed into the primary loop. The purification facility is working while the pumps are not disabled. The dotted line represents a temperature of 1873K.

In case the helium purification facility is not working, the ingressed water will stay in the primary loop. As is seen in the previous section, this results in a significantly higher amount of water in the core. The situation where the pump is disabled is shown in Figure 4.21. When the pumps are disabled, a maximum temperature of 1465K is reached. This temperature is not attained in the lowest core zone as was the case with the purification facility enabled but in core zone 8. Apparently the amount of water in the lowest core is significantly lower than in the core zones just above it. This results in a higher reactivity in these core zones which in the end obtain a higher temperature despite having a better cooling environment.

The situation where the pump is not disabled is shown in Figure 4.22. This situation results in the highest maximum temperature of all the cases. This temperature reaches 1655K.

It can also be noted that maximum temperatures with the purification facility off are actually higher for the situation with the pumps not disabled as opposed to the situation with the purification facility on. This can be explained by result obtained in Section 4.4.
Figure 4.21: A total of 600 kg of water has ingressed into the primary loop. The purification facility is not working while the pumps are disabled. The dotted line represents a temperature of 1873K.

It can be concluded that during a design basis accident the core is never in danger of reaching temperatures that exceed the maximum safe temperature of the fuel elements. Also the temperature increase difference between a system where the purification facility is working and a system where the purification facility is not working is significant.
4.5. WATER INGRESS

Figure 4.22: A total of 600 kg of water has ingressed into the primary loop. The purification facility is not working while the pumps are not disabled. The dotted line represents a temperature of 1873K.

4.5.2 Beyond design basis accident

During a beyond design basis accident the primary loop and secondary loop are not closed off from each other in the timely manner of a design basis accident. This means that more water can ingress in the primary loop than during a design basis accident. During such an accident it is also still possible for the pump to fail. The maximum temperatures attained during different levels of ingressed water mass are recorded for the different possible situations. This way the maximum amount of water that can ingress in the core while maintaining the safety conditions can be obtained. A conception of the reaction time is also acquired this way. In case the water is extracted out of the system immediately, a steady state situation with a maximum amount of water in each core zone is attained that is related to the ingress speed. The figures 4.19 and 4.20 represent the maximum temperatures in that situation already.

In Figure 4.23 the maximum temperatures attained for different amounts of ingressed water are plotted in case the purification facility is not engaged. The more water that is inserted in the primary loop the higher the maximum core temperature gets. With the helium pump still enabled, higher temperatures are reached for the different levels of water ingress. This is the result of the extra power generated in the top core zones 1 and 2.
Figure 4.23: Different quantities of water are ingressed in the primary loop. The purification facility is not working. The dotted line represents a temperature of 1873K.

In the situation with the pump still enabled, the maximum safe temperature of 1873K is reached with a total of 842kg ingressed water after 70 seconds. The maximum safe temperature in the situation that the pump is disabled is reached at a total of 1176kg after 98 seconds.

In both cases, temperatures are higher than the maximum safe temperature before the maximum possible amount of 2500kg has ingressed and thus pose a threat to the inherent safety of this reactor design.
Chapter 5

Conclusion

A more sustainable reactor design has been proposed by Wols which is based on the HTR-PM design and is fuelled by uranium-233 bred out of thorium-232. To design a safe reactor, it is important to predict its behaviour under normal operating conditions and also during possible accidents.

The fuel elements are stored in TRISO elements which are held themselves in fuel pebbles. These TRISO elements can contain the radioactive fission products in conditions with a maximum temperature of 1600°C. It is thus important that the fuel temperature will always remain below this temperature during potential accidents.

5.1 Dynamic model

A neutron kinetics model is a simple model that can describe basic neutronics behaviour of the reactor. Because the HTR-PM is designed with a tall reactor core, significant different conditions can be attained at different heights in the reactor. This is why the basic neutron kinetics model has been expanded to account for multiple regions in the core. The dynamic model includes 11 core zones which are evenly divided over the axis of the reactor. Temperature dependent characteristics of the reactor have all been lumped in a temperature feedback of the reactivity coefficient to account for temperature dependent behaviour of the reactor.

The temperature increase of the core as a result of the produced power and the heat distribution throughout the reactor has been modelled by a simple thermodynamics model. This model considers the reactor as a collection of nodes with each of these nodes having uniform conditions.

The neutronics model and the thermodynamic model are coupled together. Both models can be described by a single system of differential state equations.

The first question that arises is the reliability of the developed model. In a static environment the dynamic model can be compared to the model of Wols. Results of the neutron currents through the core zones show good similarities. Also the power production throughout the core once the temperature feedback of the thermodynamic model has been included matches the result of Wols.

The validation of the dynamic behaviour of the reactor is more difficult, because there is no reference model to compare to. For now it can only be stated that reaction times to condition changes seem reasonable.
Most notable result of the dynamic model was that extra cooling of the core via a lower inlet temperature does not actually always make each fuel element cooler. The top of the core will take advantage of the cooler fluid and reduce in temperature. This results in a higher power output in this part of the core which will increase the temperature of the cooling fluid. At standard coolant flow velocities this results in higher temperatures in the bottom of the core.

At first this result may seem strange, because as the cooling fluid flows down the core, it rises in temperature to the standard input helium temperature from where the helium is expected to warm up as in the standard configuration. It was calculated that the total power output with this lower inlet helium temperature was 58% higher than in the normal configuration. This extra power is apparently enough to rise the cooling helium to extra high temperatures. This high power output can be explained by the extra reactivity in the top cores where the temperature is actually lowered. Extra power is also generated in the lower core zones because of neutrons originating from the top core zones cause fission reactions in these lower zones.

5.2 Water ingress

During a water ingress scenario water vapour leaks from the secondary loop into the primary cooling loop of the reactor. Normally the primary cooling loop contains helium which does not influence the neutronics of the reactor. Water does have a moderating effect and this can lead to dangerous situations.

A model has been developed to describe the ingress rate and distribution through the core of the ingressed water during such an accident.

The amount of water in the core is determined by the ingress rate of the water if the water is extracted out of the primary loop directly after it has passed the core. In this case only about 1kg of water is located in every core zone at every point in time. During such an accident a maximum temperature increase of about 150K has been encountered. A maximum temperature of 1250K has been reached this way. This temperature is well below the safety boundary of the TRISO elements.

The situation where the water is not directly extracted out of the system has also been considered. In this case all water that has ever ingressed the primary loop will keep circulating there. This results in a water build up in the primary loop. During a design basis accident the primary and secondary loop are disconnected in a timely manner. This way only 600kg of water has ingress. This resulted in a maximum core temperature of 1655K. Although these temperatures and associating power outputs are probably not desired, these temperatures do not endanger the TRISO elements yet.
5.2. WATER INGRESS

During a beyond design basis accident, the water ingress initially goes unnoticed and can keep flowing in the primary loop for a longer amount of time before the ingress is halted. It is investigated what the maximum amount of ingressed water is at which the fuel elements reach their maximum safe temperature. In case the core is still cooled, the bottom of the core reaches TRISO failure temperatures at an ingress of 842kg. In case the cooling is disabled, this temperature is reached at a total water mass of 1176kg. Because previous research showed that a maximum of 2500kg water can potentially ingress, both situations are a potential threat to the inherent safety of the thorium HTR design.
Chapter 6

Recommendations

6.1 Dynamic model

The developed dynamic model has already been compared to other numerical models in steady state operating conditions. These comparisons did increase the confidence in the model. Although the dynamic behaviour of the model did seem reasonable, the behaviour of the model is not validated properly yet. The results of the model should thus be used with great caution. The new steady state situations can be validated against other static models with input conditions corresponding to the new steady state situations to better determine their reliability.

6.2 Water ingress model

The water ingress model is an extreme simplification of the actual water ingress process. This is justified because the main objective of this model is to give an impression of the amount of water mass in every core zone at every point in time. The model performs especially poor in the prediction of the time dependent distribution of the water. The time dependency of this water mass is primarily dependent of the water ingress rate. The poor time dependent behaviour is not a major let down because the flow velocity of the water is very high which results in very fast distribution of the water mass. Nevertheless, to get a more reliable time dependent behaviour as a result of water ingress in the core a more advanced model should be developed to calculate the water mass in every core zone. This is especially true for slower flow speeds, because in that case the water distribution over the zones vary more significantly.

6.3 Helium purification facility

In this research only two options for the helium purification facility are investigated. One situation modelled a perfect purification facility where all the water was extracted out of the primary loop after a single pass through the core. In the other situation no water was ever extracted out of the primary loop. It is probably not realistic for the helium purification facility to extract all of the water out of the coolant gas after one pass through the core. To get a more realistic approximation
of the design basis accident, the amount of water that the purification facility can extract out of the loop at maximum speed should be investigated. It is expected that the result will be a medium between the two situations tested in this thesis.

6.4 Radial temperature variations

The radial temperature variation in the reactor core is completely neglected in this research. In this research the average properties of the core zones are taken into account. This while these properties possibly vary quite notably. The core will be hotter in the inside than on the outside of the core because heat is transferred to the outside. This temperature difference is further amplified because the core is divided in a breeder and a driver zone and the highest power output is found in the central driver zone. Also differences in helium flow can potentially have an impact on the radial temperature profile. This radial temperature variation can have an impact on the inherent safety of the reactor because higher temperatures can be attained locally. A better safety analysis can thus be obtained if this radial temperature profile is incorporated in the model.
Appendix A

Derivations

In this appendix the derivation of the point kinetics equation out of the one-group diffusion equation is shown as well as the parameter conversion of the coupling coefficients.

A.1 Point kinetics

The equation shown in (A.1) was derived from the one-speed neutron diffusion equation in Chapter 2.

\[
\frac{1}{v} \frac{d\phi(t)}{dt} = (1 - \beta) \nu \Sigma_f \Phi(t) - \sum \lambda_k c_k(t) + \sum \nu \Sigma_f \Phi(t) - \Sigma_a \Phi(t) + \sum \lambda_k c_k(t) \tag{A.1}
\]

Substituting the neutron density \( n \) as in equation (A.2) results in equation (A.3).

\[
\Phi(t) = n(t) v
\]

\[
\frac{dn(t)}{dt} = (1 - \beta) \nu \Sigma_f v n(t) - \Sigma_a v n(t) + \sum \lambda_k c_k(t) \tag{A.3}
\]

Substituting the average neutron lifetime, \( l \), in the equation:

\[
l = \frac{1}{\nu \Sigma_a},
\]

\[
\frac{dn(t)}{dt} = (1 - \beta) \frac{\nu \Sigma_f}{l \Sigma_a} n(t) - \frac{1}{l} n(t) + \sum \lambda_k c_k(t). \tag{A.5}
\]

Now substituting the multiplication factor \( k \):

\[
k = \frac{\nu \Sigma_f}{\Sigma_a},
\]

\[
\frac{dn(t)}{dt} = (1 - \beta) k \frac{1}{l} n(t) - \frac{1}{l} n(t) + \sum \lambda_k c_k(t). \tag{A.7}
\]
APPENDIX A. DERIVATIONS

And with the neutron generation time $\Lambda$:

$$\Lambda = \frac{l}{K}, \quad (A.8)$$

$$\frac{dn(t)}{dt} = (1 - \beta) \frac{1}{\Lambda} n(t) - \frac{1}{k\Lambda} n(t) + \sum_k \lambda_k c_k(t), \quad (A.9)$$

$$\frac{dn(t)}{dt} = \frac{n(t)}{k\Lambda} (k(1 - \beta) - 1) + \sum_k \lambda_k c_k(t). \quad (A.10)$$

With the reactivity:

$$\rho = \frac{k - 1}{k}, \quad (A.11)$$

this results in the point kinetics equation in equation (A.12)

$$\frac{dn(t)}{dt} = \frac{n(t)}{\Lambda} (\rho - \beta) + \sum_k \lambda_k c_k(t). \quad (A.12)$$

A.2 Coupling coefficients

The coupling coefficients between zone $i$ and zones $j$ are given as in equation (2.12):

$$\sum_j A_{ij} \frac{D_j v_j n_j - D_i v_i n_i}{d_{ij}}. \quad (A.13)$$

Substituting the neutron lifetime (A.4) results in:

$$\sum_j A_{ij} \frac{D_j v_j n_j}{\Sigma a,j} - \frac{D_i v_i n_i}{\Sigma a,i} \quad (A.14)$$

Continuing by substituting the multiplication factor (A.6):

$$\sum_j A_{ij} \left( \frac{D_j n_j k_j}{\nu_j \Sigma f,j} - \frac{D_i n_i k_i}{\nu_i \Sigma f,i} \right). \quad (A.15)$$

And with the neutron generation time (A.8) this results in:

$$\sum_j A_{ij} \left( \frac{D_j n_j}{\nu_j \Sigma f,j \Lambda_j} - \frac{D_i n_i}{\nu_i \Sigma f,i \Lambda_i} \right). \quad (A.16)$$
Appendix B

HTR-PM dimensions

As described by Rodriguez [14].

<table>
<thead>
<tr>
<th>Design parameter</th>
<th>Design value</th>
<th>Units</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power</td>
<td>2x250</td>
<td>MW</td>
</tr>
<tr>
<td>Operation lifetime</td>
<td>60</td>
<td>years</td>
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<tr>
<td>Load factor</td>
<td>85</td>
<td>%</td>
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**Fuel elements**

<table>
<thead>
<tr>
<th>Design parameter</th>
<th>Design value</th>
<th>Units</th>
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</thead>
<tbody>
<tr>
<td>Diameter of the fuel elements</td>
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<td>cm</td>
</tr>
<tr>
<td>Nuclear fuel</td>
<td>UO₂</td>
<td></td>
</tr>
<tr>
<td>Heavy metal loading</td>
<td>7</td>
<td>g</td>
</tr>
<tr>
<td>(^{235}\text{U}) enrichment</td>
<td>8.8</td>
<td>%</td>
</tr>
<tr>
<td>Neutron generation time</td>
<td>0.0011</td>
<td>s</td>
</tr>
<tr>
<td>Number of fuel pebbles</td>
<td>520000</td>
<td></td>
</tr>
<tr>
<td>Number of graphite pebbles</td>
<td>225530</td>
<td></td>
</tr>
<tr>
<td>Average discharge burn-up</td>
<td>80000</td>
<td>MWd/tU</td>
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**Core design parameters**

<table>
<thead>
<tr>
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<th>Design value</th>
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</thead>
<tbody>
<tr>
<td>Active core diameter</td>
<td>300</td>
<td>cm</td>
</tr>
<tr>
<td>Average height of active core</td>
<td>1100</td>
<td>cm</td>
</tr>
<tr>
<td>Average core power density</td>
<td>3.22</td>
<td>MW/m^3</td>
</tr>
<tr>
<td>Average output power per fuel pebble</td>
<td>0.881</td>
<td>kW</td>
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<tr>
<td>Neutron leakage from the core</td>
<td>15.08</td>
<td>%</td>
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**Reactivity control**

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<tbody>
<tr>
<td>Number of control rods</td>
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<td></td>
</tr>
<tr>
<td>Number of absorber pebble units</td>
<td>18</td>
<td></td>
</tr>
<tr>
<td>Worth of control rods</td>
<td>5.25</td>
<td>%</td>
</tr>
<tr>
<td>Worth of absorber pebble units</td>
<td>11.32</td>
<td>%</td>
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</table>
## APPENDIX B. HTR-PM DIMENSIONS

<table>
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<th>Graphite reflector</th>
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<tbody>
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<td>Height of the graphite structure</td>
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<td>Nominal diameter</td>
<td>55 cm</td>
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<td>Height of core cavity</td>
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<table>
<thead>
<tr>
<th>Coolant</th>
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<tbody>
<tr>
<td>Primary helium pressure</td>
<td>7 MPa</td>
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<td>Reactor outlet temperature</td>
<td>750 °C</td>
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<tr>
<td>Reactor inlet temperature</td>
<td>250 °C</td>
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<tr>
<td>Primary helium flow rate</td>
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<td>Leakage ratio (Lower header to outlet header)</td>
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<td>Maximum fuel temperature (normal operation)</td>
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<td>Maximum fuel temperature (accidents)</td>
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<table>
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<td>Height</td>
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<td>Wall thickness</td>
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<table>
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<tr>
<td>Feed water temperature</td>
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<tr>
<td>Main steam pressure at turbine inlet</td>
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</tr>
<tr>
<td>Main steam temperature at turbine inlet</td>
<td>535 - 567 °C</td>
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<tr>
<td>Generator power</td>
<td>210 MW</td>
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<tr>
<td>Electrical efficiency</td>
<td>42 %</td>
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</tbody>
</table>


