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Model improvements in ATTICA-DORT/ TORT concerningwater ingress

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Summary

This document describes results of calculations with the coupled code system TORT-TD/ATTICA3D for an assumed accident scenario with water ingress. The calculations have been carried out for a reference plant design oriented at the Chinese HTR-PM [1]. In order to be able to perform the neutron dynamics calculations, appropriate nuclear cross section data had to be prepared for the TORT-TD code that take into account the presence of water. This has been done by means of the code system ZIRKUS of IKE [1] in which the NEA spectral code MICROX2 is embedded [3]. Initial and boundary conditions for the accident scenario with water ingress have been defined taking into account previous published literature [4][5]. The consequences of the water ingress are mainly analysed with respect to two major effects: The initial fast power increase due to the addition of moderator and a long term heat-up phase with corrosion due to the decay heat of the reactor. The results of the TORT-TD/ATTICA3D calculations are presented. Concerning the initial power increase the results are compared with those of TINTE calculations available from literature. Here a good agreement was obtained. Concerning the long-term effect of corrosion over up to 80 hours, only a moderate attack of the fuel elements was found. The maximum burn-off of the fuel sphere graphite matrix material is only 2.7%. An anticipated transient without scram case was also analysed with the same boundary conditions. Here the power increase was about 15 %, and presented together with the change of decay heat which is shown to be negligible. With the system code SPECTRA which simulated the transient behaviour more realistic boundary conditions were determined. As shown in chapter 5 the initially assumed increase in steam flow rate was somewhat conservative. The analyses were repeated with the new boundary conditions and showed that the increase of power is even less.

Approval

Date	By
13/03/2015	Michael FUTTERER
16/03/2015	Norbert KOHTZ
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1 Introduction

The aim of this task is to establish a state-of-the-art analysis chain for the safety relevant analysis of water ingress events into a High Temperature Gas-cooled Reactor (HTGR). The computational analysis of water ingress to the reactor core of an HTGR has not been improved in the last decades because the prevailing HTGR designs did not use steam generators for power conversion. Recent developments, however, aim again at producing steam for generation of electricity or supply of process heat.

IKE is further developing the three-dimensional thermal hydraulic code ATTICA-3D (Advanced Thermal hydraulic Tool for In-vessel and Core Analyses), see [9]. ATTICA-3D is designed to model heat and mass transport in HTRs with pebble or block fuel elements. It allows the calculation of concentrations for an arbitrary number of gas components, taking into account convection and diffusion as well as sources/sinks from chemical reactions.

A basic model of carbon chemistry (reactions with oxygen and steam) was implemented in ATTICA3D. Also the interface to neutronics was enhanced to transfer an additional parameter, the hydrogen density in either steam or hydrogen. The coupling with the transient neutron transport code TORT-TD (developed by GRS); see [7] [8], was implemented in the framework of a nationally sponsored project (German Federal Ministry of Economics and Technology Förderkennzeichen/support code1501382), see [9].

Within this task the ATTICA-3D/TORT-TD code combination shall be used for calculation of water ingress scenarios with particular emphasis on:

- Reactivity effects (heterogeneous temperature model of the fuel element is implemented for spheres, work for block type fuel elements is ongoing)
- carbon chemistry

In D22.23 there is detailed description of the models used in both ATTICA3D and TORT-TD. This deliverable is available to ARCHER partners on the online platform.

The present report describes first results of calculations with the coupled code system TORT-TD/ATTICA3D for an assumed accident scenario with water ingress carried out for a reference plant design oriented at the Chinese HTR-PM [1]. The preparation of nuclear cross section data that take into account the presence of water by means of the code system ZIRKUS is described in Chapter 2. Initial and boundary conditions for the accident scenario with water ingress are given in Chapter 3. Chapter 4 presents results of the TORT-TD/ATTICA3D calculations. The consequences of the water ingress are analysed with respect to the initial fast power increase due to the addition of moderator and a long term heat-up phase with corrosion due to the decay heat of the reactor. This analysis is completed by taking more realistic boundary conditions determined with SPECTRA which models the whole primary circuit including safety measures.

2 Generation of cross sections for water ingress

To be able to simulate a water ingress both neutronically and thermal hydraulically, it is necessary to have the appropriate cross sections (XS) available. Otherwise no feedback can come from the introduction of additional moderator. To generate these XS for the reference system selected: the Chinese HTR-PM [1], the 2-d steady-state and long-term transient HTR simulation system ZIRKUS [2] is used. With ZIRKUS, we have the possibility to produce XS for the equilibrium core, with respect to a number of variation parameters. Due to the lack of possibility to simulate transients that demand real modelling of reactor kinetics (including of neutron precursor equations), the XS generated with ZIRKUS are afterwards extracted from the output files and transferred in a format that our transient time-dependent neutron code TORT-TD can work with, i.e. the NEMTAB format.

For XS generation with ZIRKUS, a geometry is used that comprises 399 material zones, 144 burn up zones for the core region, and 24 spectral zones (zones with different temperatures). Note that the core region (from zone 1 to 144) is sub-divided in eight so called flow channels. This is necessary to account for the constant reload and discharge of pebble fuel elements during operation and also to account for the fact that like in an hour glass, the pebbles in the most inner channel have a shorter residence time in the core compared to the pebbles in the most outer channel. The difference in residence time of a pebble can be found in the differing number of zones in each channel. While the innermost channel has 16 axial subdivisions, the most outer channel has 24 axial subdivisions. When one fuel reload step is executed, the number densities of the upper zone, e.g. zone 1 is passed to zone 2 at the end of a burn-up step. The same applies to all other zones. One can now see that after 16 reload steps the innermost fuel pebbles will be discharged or recycled, while the pebbles in the outermost channel still have to undergo 8 more reload steps until reaching the discharge tube. This is how ZIRKUS accounts for the different flow velocities of each channel. The fact, that the outermost channel takes about 1.5 times longer than the innermost channel was determined by experiments.

13 neutron groups are used for the computation and the generation of XS, and it is this elevated number of neutron groups that allows for omitting the buckling dependency for neutronics which otherwise, with a smaller number of groups, should have been taken into account.

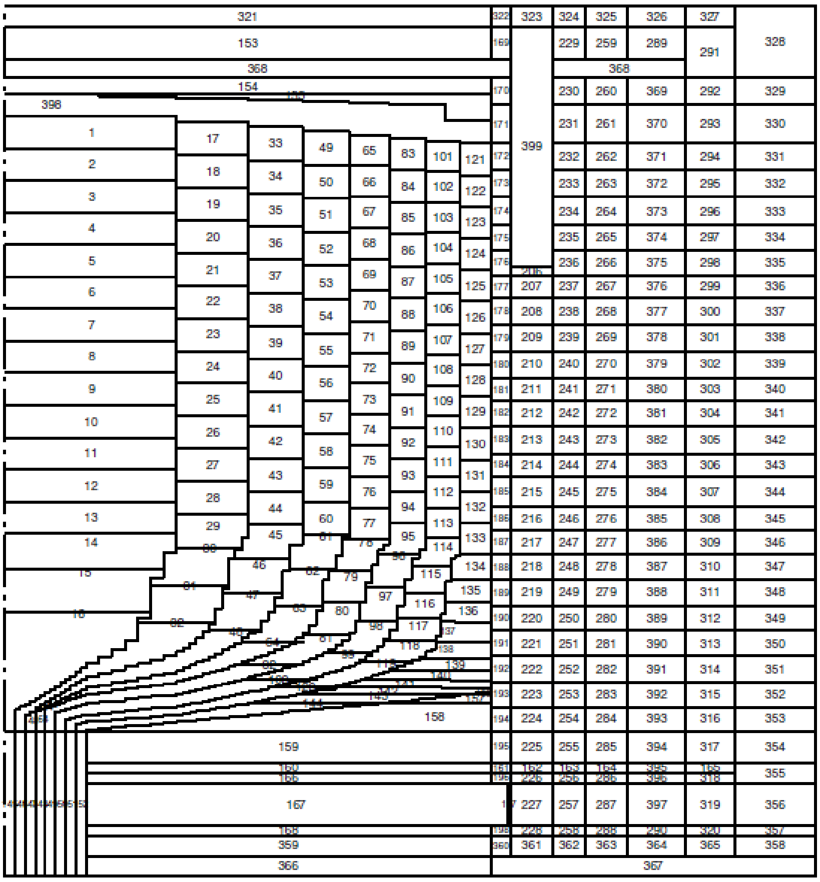

Figure 1 : Geometry of the HTR-PM in the ZIRKUS HTR simulation system

For a TORT-TD calculation, the spatial meshes are condensed with respect to a simplified geometry for a transient calculation. This reduces computational time. For the TORT-TD geometry only 232 material zones are used. However the number of burn-up zones and also, since burn-up and spectral zones are roughly the the same, are increased to 154. Note, that in the TORT-TD geometry the flow channels introduced above are dissolved. The reason for this lies within the fact, that for a transient incident or accident, the recycling of the fuel is assumed to be turned off. This goes hand in hand with the safety regulations that would prohibit business-as-usual operation during an incident or accident.
For the transient calculation the initial number of 13 neutron groups will be condensed to only 7 neutron groups using a condensation routine within the ZIRKUS code. The condensed XS were also tested in the condensed ZIRKUS geometry for consistency. For the control rod zone (zone 399 in ZIRKUS, zone 192 in TORT-TD) the XS were manipulated manually in order to have spatially resolved control rods instead of the 2-d grey curtain assumption made in ZIRKUS. Here, a MCNP [6] calculation for the control rod was done, and the σ_a and σ_{tot} had to be modified such that MCNP and TORT-TD had the same increase in reactivity for the equilibrium and totally withdrawn position of the control rods (about 0.4% reactivity increase per spatially resolved control rod, 1.2 % reactivity increase when the whole control rod bank is fully withdrawn).

Figure 2 : TORT-TD geometry for the HTR-PM

The variation parameters that are used for the XS generation are:

- Fuel temperature
- Moderator temperature
- Xenon density
- Hydrogen content

The range of parameters that are possible are presented in Table 1. Some constellations within these variations are physically irrelevant, e.g. moderator temperatures lower than the fuel temperatures, but they are nevertheless provided since it would demand modifications of the XS generation with ZIRKUS treatment of the latter.

Table 1 : Range of XS with supporting points

Variation parameter	Unit	Number of supporting points	Supporting points
Fuel temperature	K	10	300, 500, 700, 900, 1100, 1300, 1500, 1700, 1900, 2100
Moderator temperature	K	7	300, 600, 900, 1200, 1500, 1800, 2100
Buckling	-	0	-
Xenon density	$(\text{barn}^* \text{cm})^{-1}$	3	0.0, 2.0E-11, 8.0E-10
Water content	kg	3	0, 200, 800

What might seem strange is the "water content" in kg ranging from 0 to 800 kg; since in the text above, the variation parameter is introduced as the hydrogen density. The reason for this is that ZIRKUS adds a certain H_2O-density homogeneously to each of the core zones. Corrosion processes are not implemented, so the steam will not react with the hot graphite to produce hydrogen and either CO or CO_2.

In ATTICA3D however these corrosive processes are implemented. Since there would be two moderating media, namely hydrogen and steam, this would also demand for a further variation parameter. To circumvent this fact, only the hydrogen density is transferred from ATTICA3D to TORT-TD. The fraction of hydrogen contained in steam, i.e. about 2/18, the relative atomic weight of hydrogen in H_2O, is calculated for each core zone and added to the hydrogen content of the respective zone. Proceeding like that one obtains the increase of moderation, with a little detour over the hydrogen density, but for the benefit of saving of an additional variation parameter that would further increase computational time. In the water ingress scenario and results chapter, the reader will find that the power increase and the following increase in fuel and moderator temperature will feed-back neutronically and will shut the reactor down before the onset of a significant hydrogen production. This justifies the use of the ZIRKUS method to account for steam content within the core.

With the average temperature distribution from ZIRKUS which is coupled to the 2-d thermal hydraulic code THERMIX/KONVEK, see [2][1], the spectral groups are a weighted. The detailed comparison of the codes THERMIX and ATTICA3D applied for HTR-PM problem geometry can be found in [11].

For demonstration purposes the solid temperature at two different radial positions (deviations of radii due to application of different discretisation methods) is depicted in Figure 3.

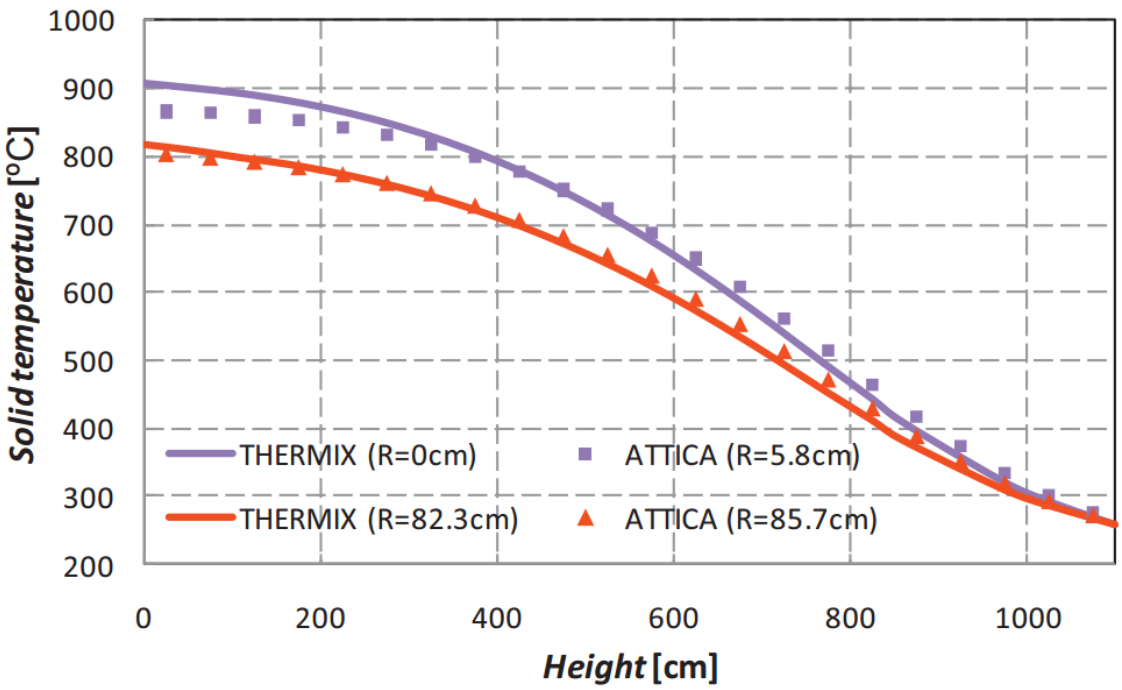

Figure 3 : Axial solid temperature distribution comparison for the HTR-PM at different radial positions, from [11]

The increase of reactivity for the HTR-PM in nominal operational conditions with 0, 200 and 800 kg in the core is displayed in

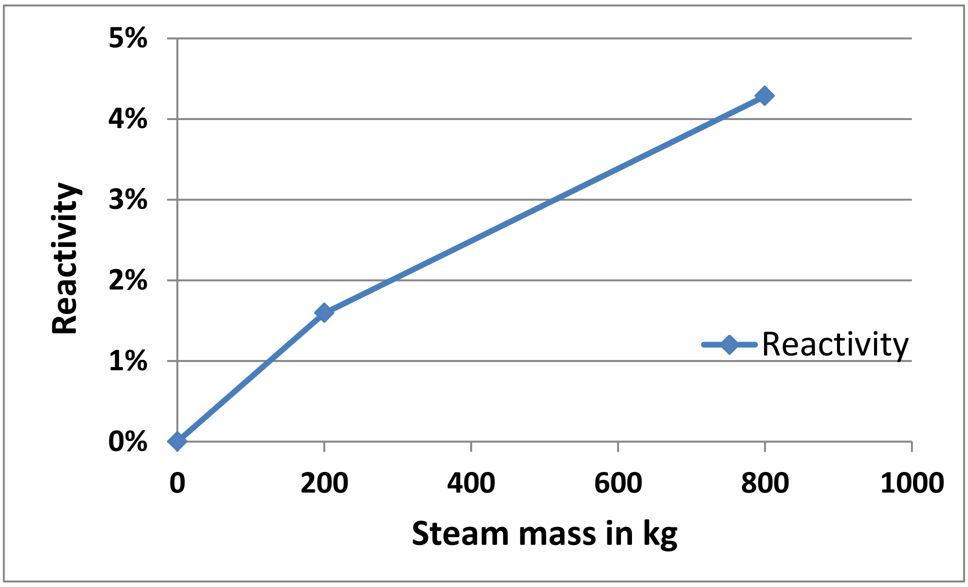

Figure 4 : Reactivity increase with steam content in the core

3 Accident scenarios for the water ingress

For the reference plant – the HTR-PM - the scenario to be calculated is a common 2F break of a single steam generator tube. Since ATTICA3D does only simulate processes inside the reactor pressure vessel, important parameters and boundary conditions of the accident for the beginning have to be taken from other analysis in literature. Here references [4] and [10] prove to be helpful. In these publications the water ingress is also analysed and source terms for water are determined with the help of whole primary circuit (and also

secondary circuit) simulations. The design basis accident will be taken as starting point. This scenario considers a double-end break of one steam generator tube that is detected and, subsequently, the steam generator will be isolated with the two steam generator isolation valves and drained with the steam generator emergency draining system allowing approximately 600 kg to enter the primary circuit, see [4].
As design basis accident (DBA) a 2 F break of one of the steam generator tubes is assumed. Since the pressure on the secondary side is much higher (139 bar) than the pressure in the primary circuit (70 bar), water/steam will pour into the primary circuit. As a consequence, the total pressure will immediately rise in the primary circuit. Analysis of Zheng et al in [4] determined the pressure rise to be 4 bar due to ingress of steam and water. Corrosion of graphite in a steam atmosphere will produce corrosion products like CO, CO_2 and H_2. These additional gas components will gradually lead to a further increase in pressure, but in the range of several hours. After the set point of the pressure relief valve is reached (79 bar), the valve will open reducing the primary pressure slightly below the operational pressure.
The boundary conditions for the anticipated transient without SCRAM are are presented in the next section in comparison to the conditions of the DBA, .

3.1 Assumptions and sequence of events for the DBA and ATWS

The sequence of events is taken from the water ingress analysis described in [4]. Some additional assumptions had to be made, e.g. the amount of steam that reaches the core.

The first scenario to be examined is the case when a steam generator tube ruptures, steam enters and the blower flaps close to end coolant flow completely. The steam generator is quickly isolated by the steam generator isolation valves. Flow restrictors, an engineered safety feature, should prevent flow rates that are too high from entering the reactor. For the transient the steam generator isolation is assumed to work, effectively separating the primary side from the secondary. The different boundary conditions with respect to time are [4]:

Table 2: Boundary conditions for the water ingress scenarios

Time	**Event**	**Difference to design basis accident**
	Design basis accident	Anticipated transient without SCRAM
Initial situation	Reactor at 105 % of nominal power rating → 262.5 MW_{th}	Reactor at 100 %
0 – 10 sec	break of one steam generator tube, approx. 600 kg of steam/water pour into the primary circuit Primary pressure increase (70 → 74 bars), pressure increase over time including pressure relief taken from [4]	
10 – 40 sec	humidity detector actuates blower shutdown linearly decreasing blower power control rods are inserted over 55 seconds	control rods not moved
40 sec – 80 hrs	onset of pressurised loss of forced cooling case including corrosion of graphite in steam atmosphere short term effect: power increase due to improved moderation long term effect: weakening of fuel and load-carrying graphite due to corrosive attack	
at 23 hrs	• pressure relief valve opens (set point 79 bar) to reduce pressure to 70 bar	

Additional boundary conditions for the TORT-TD/ATTICA3D simulation:
The pressure increase is a direct consequence of the increase of the primary inventory. Here, steam has the major contribution. During the course of the transient, the appearance of multiple gas components due to corrosion of steam will lead to a further increase in pressure. Since TORT-TD/ATTICA^{3D} can only simulate processes within the reactor pressure vessel and in such a way that reference pressure is an input parameter of the user, the pressure increase was approximated by using a time dependent pressure

boundary condition. Since the major part of the flow will only occur inside the core (hot gas rises in the centre, transfers the heat to the top reflector graphite structure, and flows down on the outer core close to the side reflector), the outer helium risers are simulated as blocked (very high friction due to very small hydraulic diameter). This approach can underestimate corrosion in the side reflector. But comparison calculation showed that no significant corrosion occurs there anyway, nevertheless the time step size of the solution including corroding side reflector became unbearably small (less than 10^{-10} seconds).

The pressure in the primary circuit is taken from the TINTE analysis by Zheng et al in [4] and is depicted in Figure 5

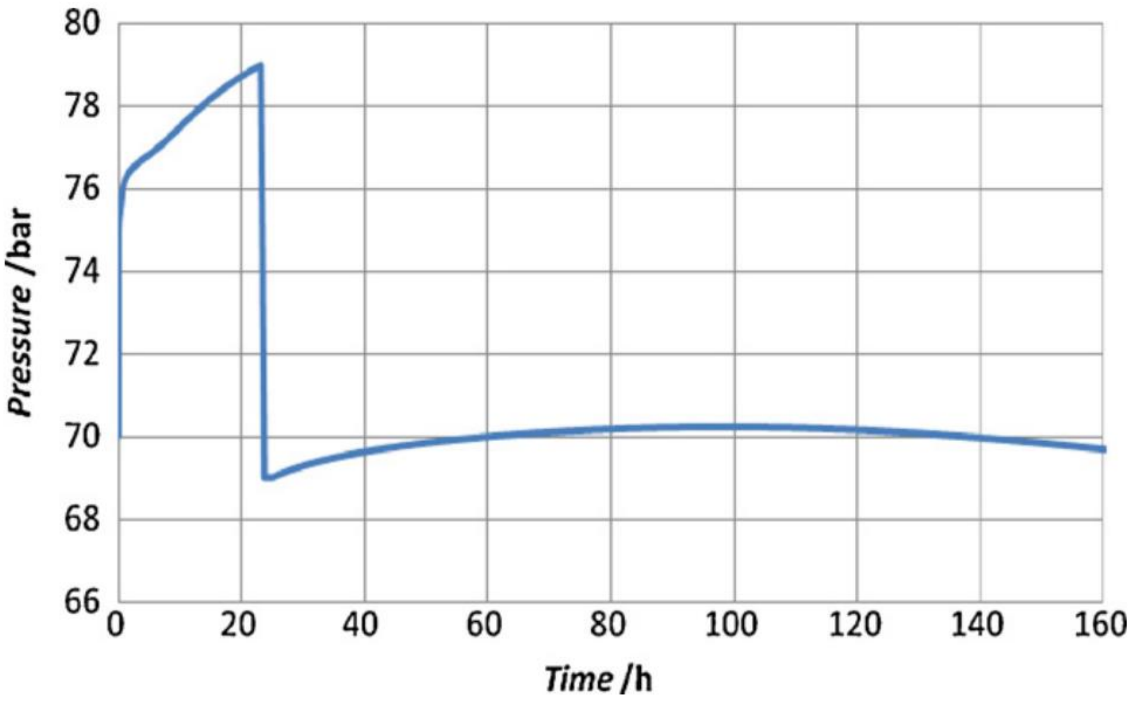

Figure 5 : Primary pressure during the design basis accident and the anticipated transient without scram (ATWS), from Zheng et. al [4]

Additional assumptions:

- The steam needs about 3 seconds from the steam generator to the reactor pressure vessel, so the steam flow in ATTICA3D will be started after 3 seconds of the transient.
- The steam mass flow increases parabolically from 0 to 5.44 kg/s within 7 seconds, and decreases linearly after the blower stops (like the helium mass flow), see Figure 6.

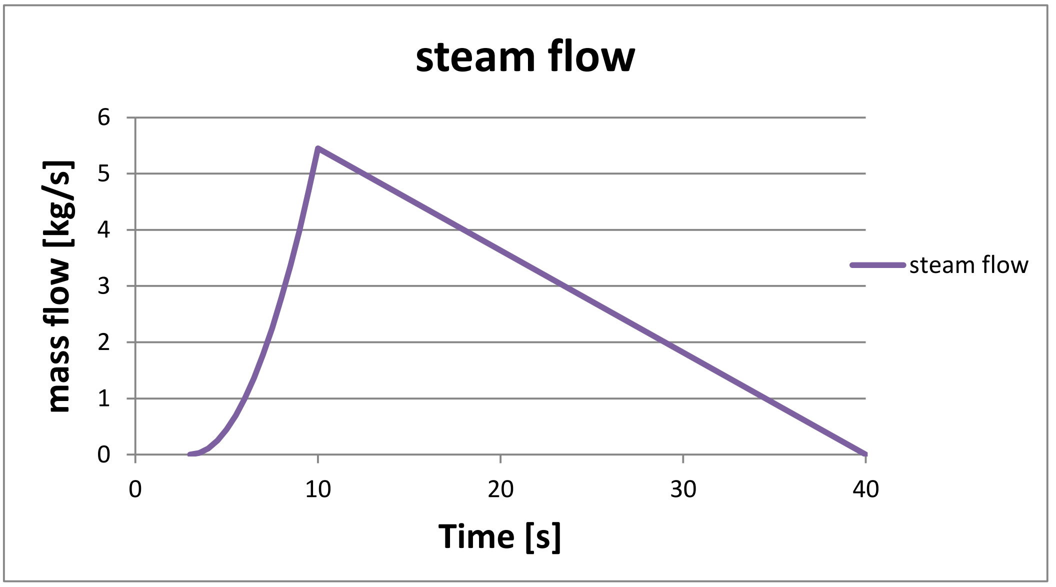

Figure 6 : Assumed steam flow rate over time for the design basis accident and the anticipated transient without scram

- The relative steam concentration at the outlet is increased to 0.0545 within 10 seconds. This steam concentration will be « sucked in » from the outside and is assumed to decrease linearly over 500,000 seconds in order to account for reduction of steam partial pressure. (This assumption will only hold as long as there is no total steam consumption balance implemented in ATTICA3D, i.e. the amount of steam that is available for chemical reaction will decrease with increasing time. If the 600 kg that entered the primary circuit due to a break of a single steam generator tube is consumed the steam partial pressure on the in- and outlet should be reduced to zero.)

Conditions at t = 0 seconds, i.e. initial conditions for the water ingress are presented in Figure 7

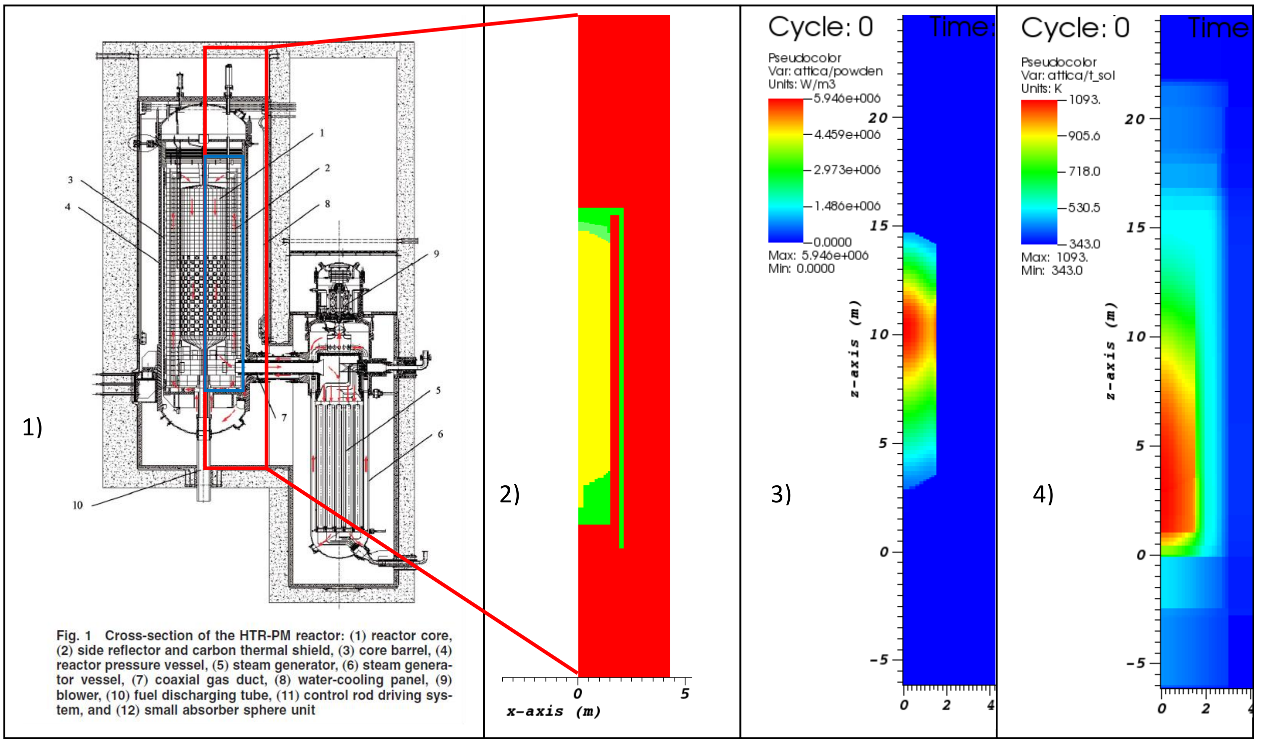

Figure 7 : 1) Geometry of the HTR-PM with the computational domains of TORT-TD in blue and ATTICA3D in red frame, 2) flow path of helium with core (yellow), 3) power distribution 4) solid temperature distribution

4 Results for the water ingress

The dynamics of water ingress accidents necessitate a simulation system that can not only model processes within the reactor pressure vessel or the primary circuit but must also cover the secondary circuit. As TORT-TD/ATTICA^{3D} lacks these abilities, use of pre-calculated boundary conditions is made. The dynamics of water pouring into the primary circuit of an HTR following steam generator tube rupture was investigated in [10]. Additionally, taking findings of [10], water ingress scenarios including neutronic response were investigated in [4]. The results produced with TORT-TD/ATTICA^{3D} for two water ingress scenarios – one with Scram, one anticipated transient without Scram (ATWS) – are compared to results of INET.

Figure 8 shows the increase of power in case of the design basis accident where humidity is detected and appropriate measures, i. e. insertion of all control rods, are taken. The increase of total power amounts to 5 % after 10 seconds. The first five seconds are needed by the steam reach the core region in significant amounts (to have an impact). Then power increases until 10 seconds, when the control rods are inserted to end the power increase. The control rods are inserted over 55 seconds. The corresponding average fuel and moderator temperatures for the DBA case are shown in Figure 9

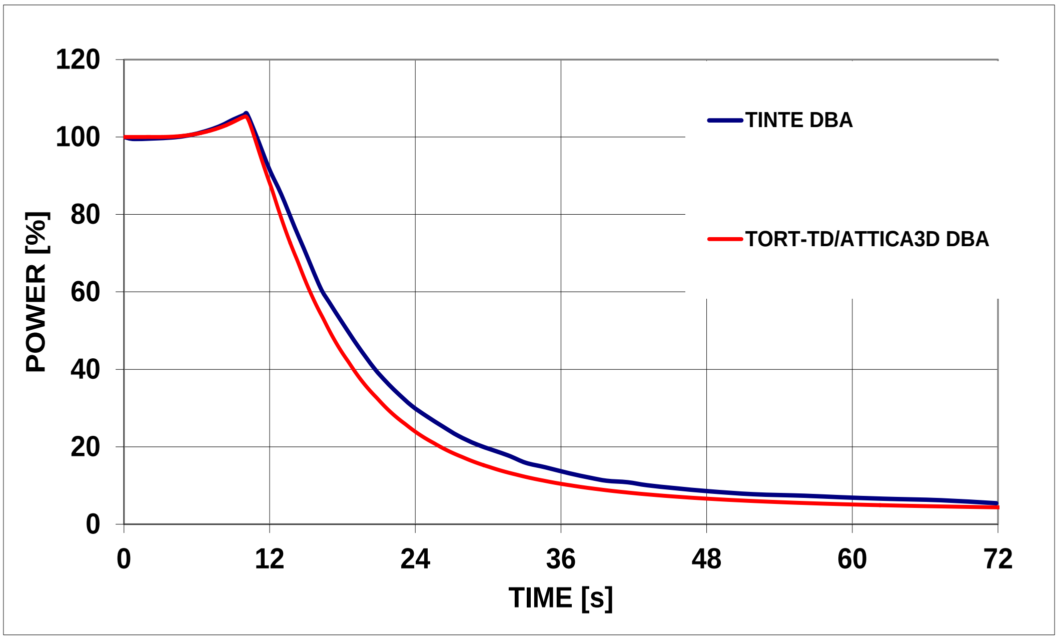

Figure 8: Short term power increase for the HTR-PM as a consequence of water ingress

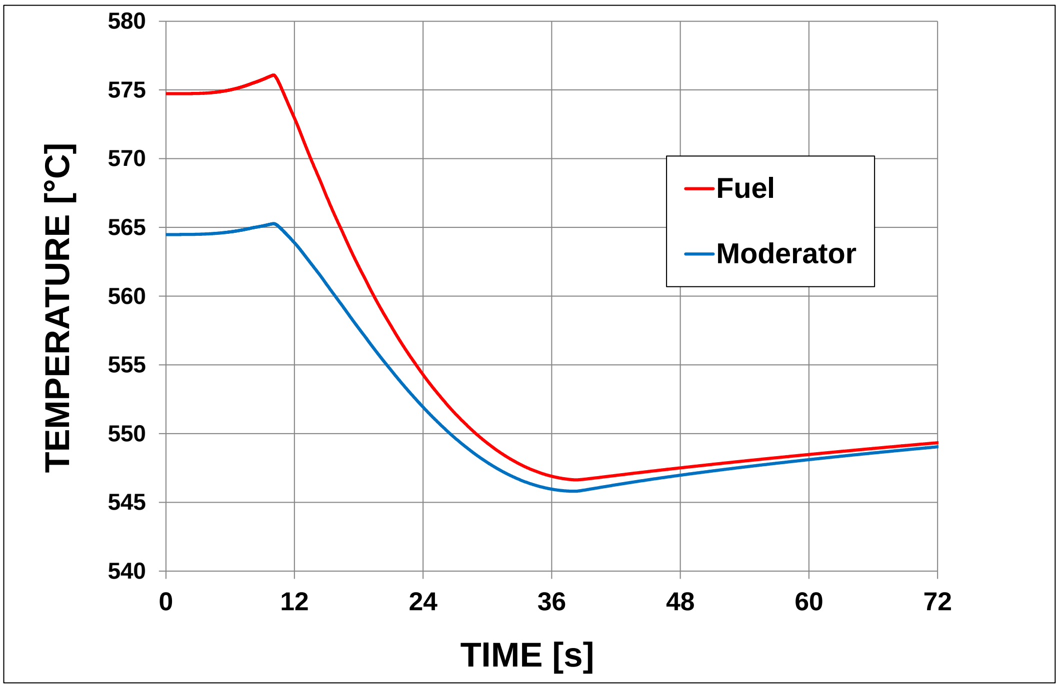

Figure 9: Short time effect on the average fuel and moderator temperature for the design basis water ingress accident, results from the TORT-TD/ATTICA^{3D} solution

After the steam is detected by the humidity sensors, the blower is turned off and the rods are inserted over 55 seconds ending the power increase for the design basis accident. The decay heat distribution of 105% of nominal conditions is taken from the ZIRKUS analysis. The short power increase in the range of seconds will not change decay heat distribution significantly, and therefore will be taken as decay heat source for the long-term corrosion effects.
The long-time behaviour is characterised best by a pressurised loss of forced cooling case and is presented in Figure 10.

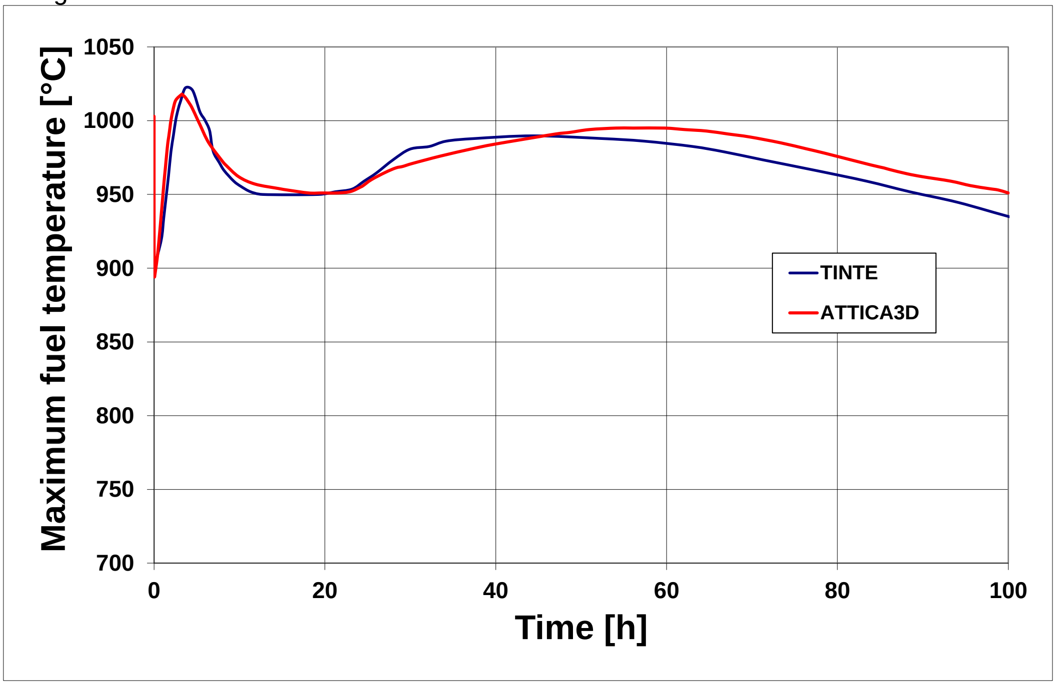

Figure 10: Maximum fuel temperatures for a design basis water ingress accident, TINTE results from [4].

In the beginning of the transient, the steam enters. However with the temperature distribution of nominal operation, even though at a higher level, the upper part of the reactor where the cold gas (250°C) enters is far below temperatures where corrosion takes place. Only at the outlet, where the solid temperatures reach (780°C) there is a minor corrosion attack. But the major part of corrosion due to steam in the core will take place in the course of hours. The reason is that the core first needs to heat up with the decay heat, so that temperatures are reached where corrosion becomes significant.

As mentioned above, the flow pattern in the water ingress design basis accident resembles the pressurised loss of forced cooling accident. After the blower stops, the bottom core and reflector are the hottest parts. The gas still under pressure heats up in the bottom and flows up in the central core. When the gas reaches cold top reflector (250°C in the beginning of the transient), it transfers its heat to the top reflector. The cooler gas flows downwards next to the side reflector in the outer part of the core.

In Figure 11 the flow patterns in steady state conditions and also after 11,800 seconds are shown. Here, the above described effect is visible.

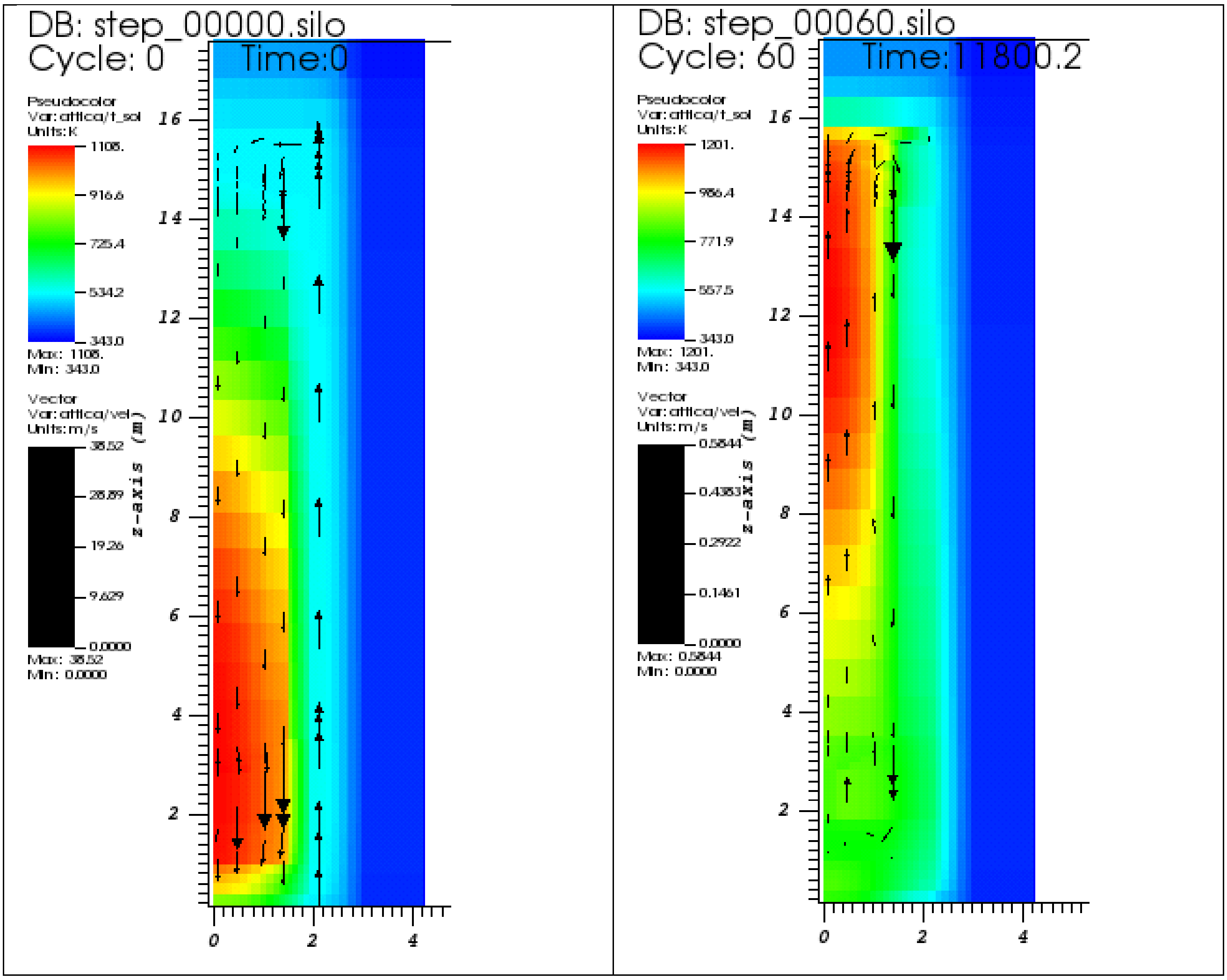

Figure 11 : Temperature distribution and velocities at t = 0 seconds (left) and after 11,800 seconds (right)

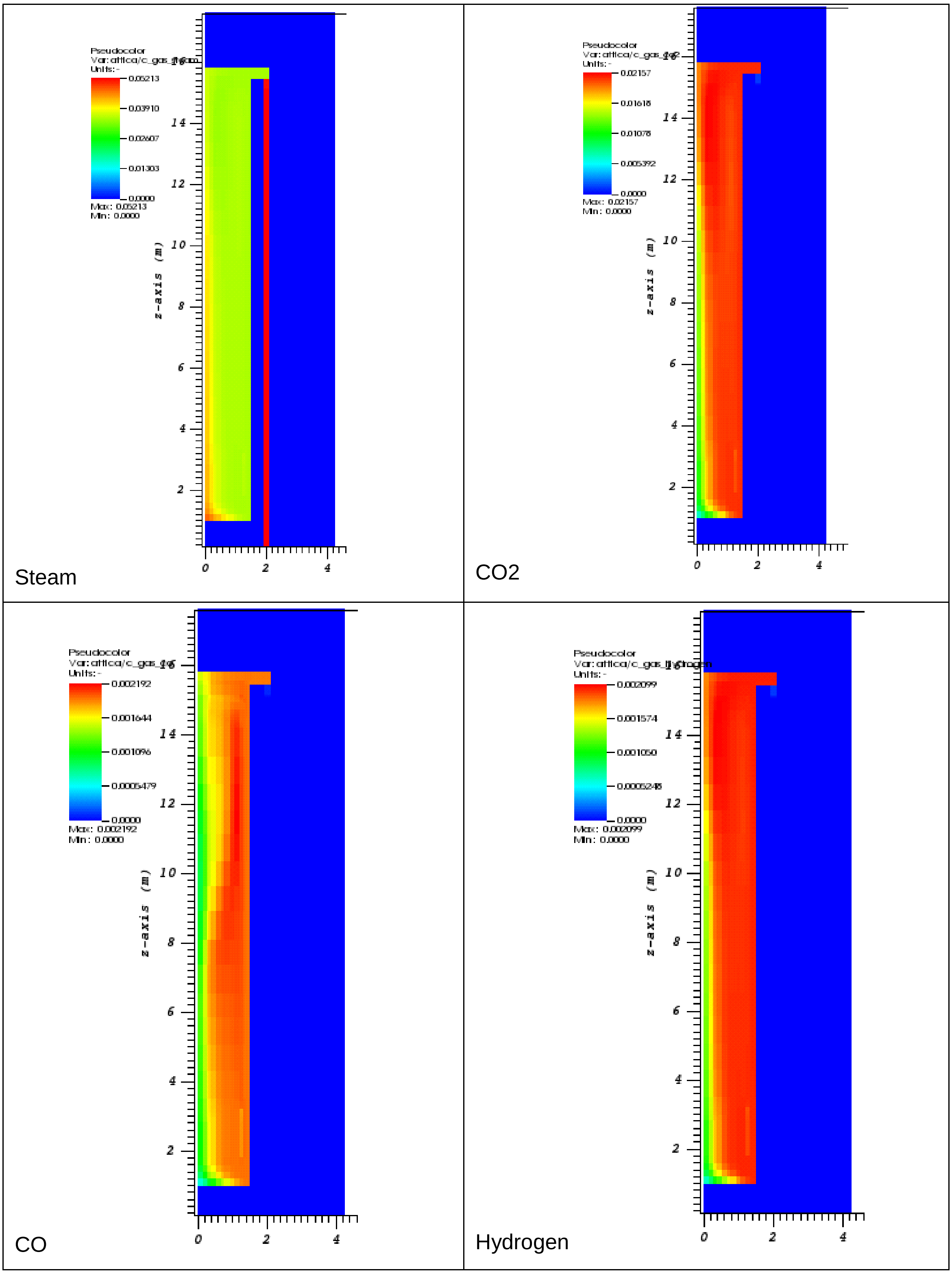

Figure 12: Steam and corrosion product distribution after 11,800 seconds (3.3 hours)

In Figure 12 above, the distribution of corrosion gases is presented. Here, it is to be noted, that the CO_2 is about five times higher than the corresponding CO concentration.

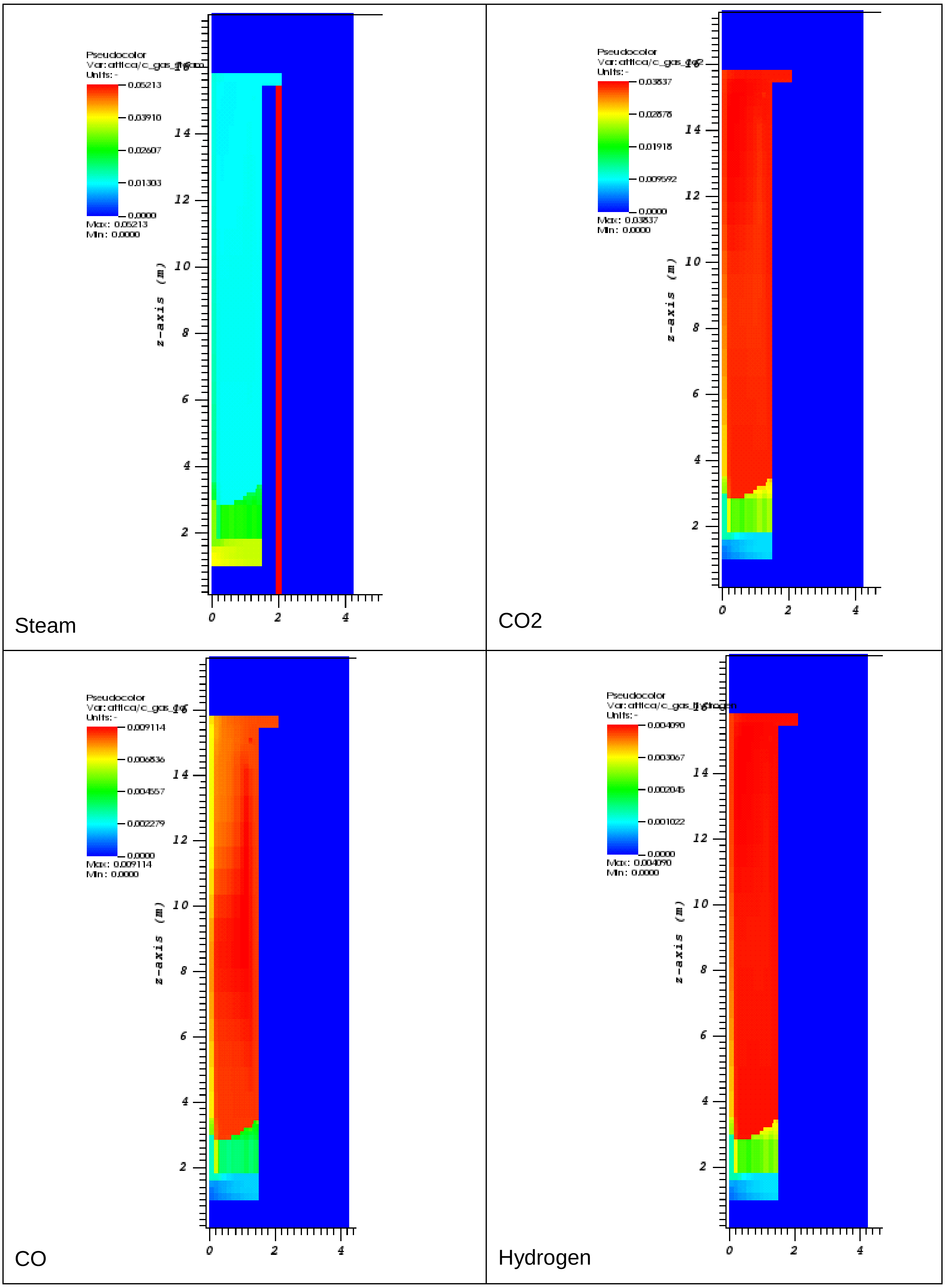

Figure 13 : Steam and corrosion gas distribution after 108,000 seconds (30 hours)

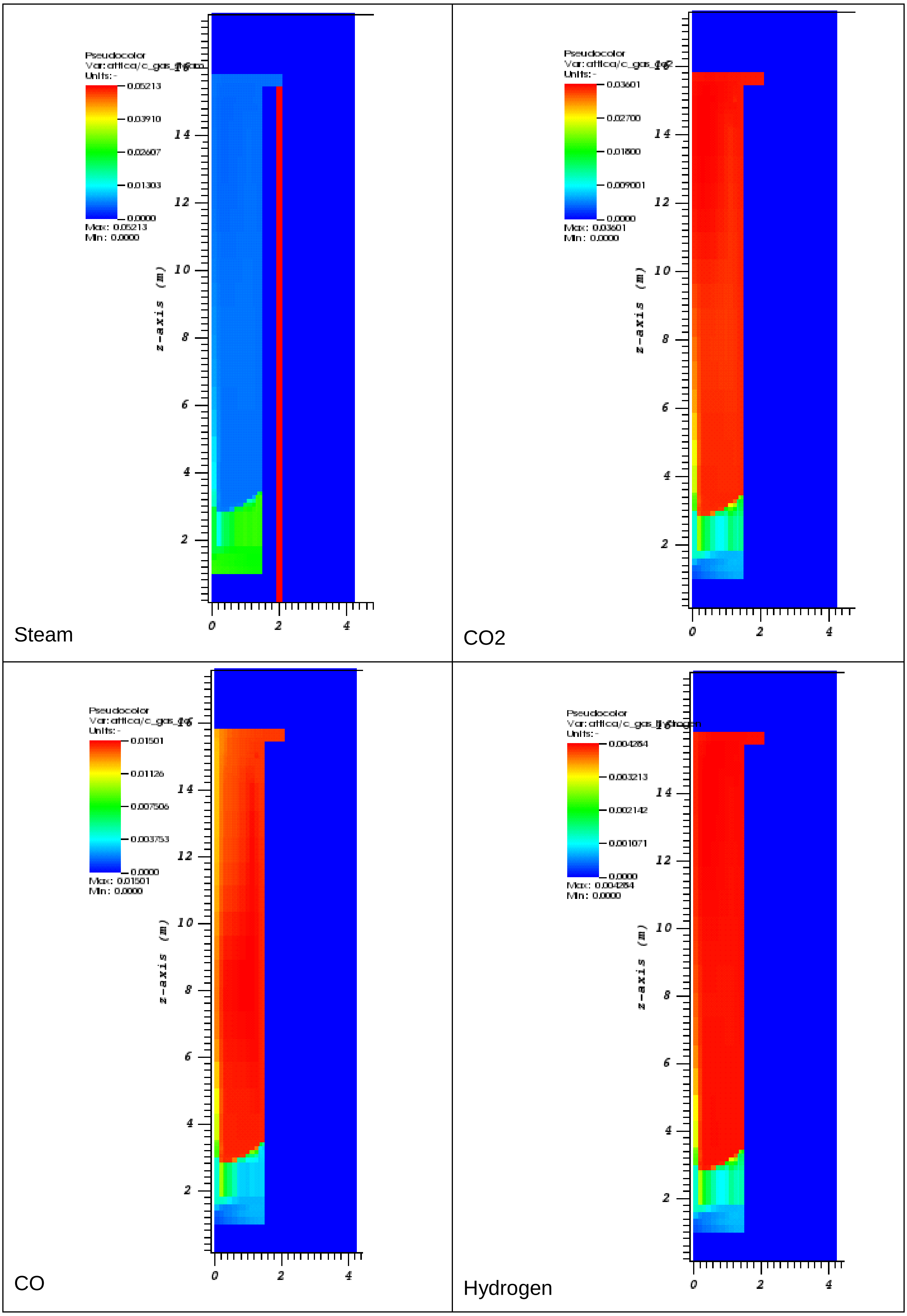

Figure 14: Steam and corrosion gas distribution after 216,000 seconds (60 hours)

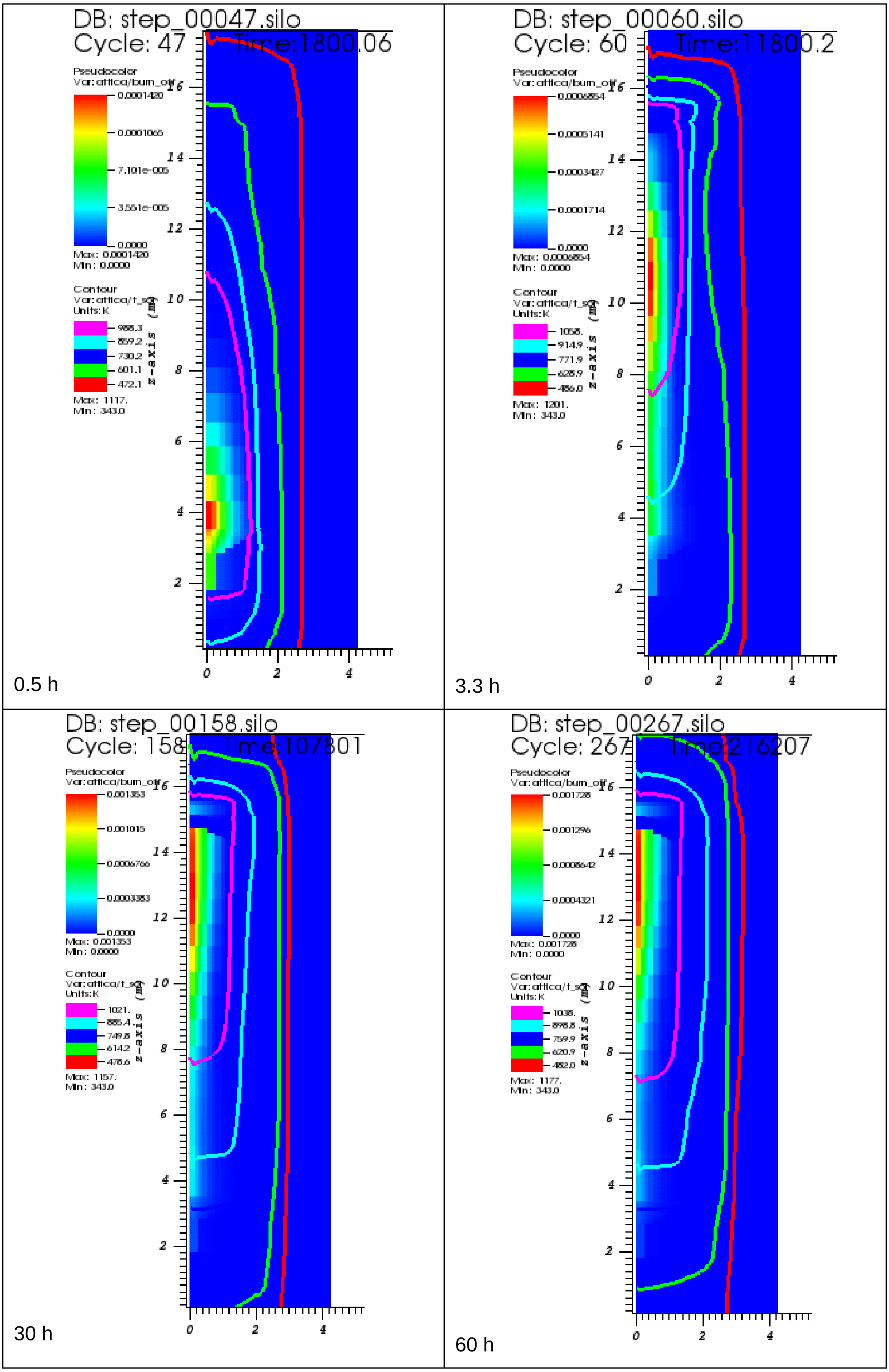

Figure 15: Relative burn-off of graphite after 0.5, 3.3, 30 and 60 hours

At the outlet, there is a concentration of steam present that corresponds to the 4 bar partial pressure out of 74 bar total pressure in the start of the transient. This steam concentration reduces linearly to 0 over 500,000 seconds.
The steam flow is corresponding to the gas flow of the helium. Since the core heats up and the temperature maximum shifts from bottom to top, the first centre of corrosion can be found in the lower part. With the re-distribution of the solid temperature and the cooling down of the lower part, the steam can now enter the core region and attack fuel elements. The centre of corrosion slowly shifts upwards and follows the temperature maximum.
The centre of corrosion starts at the bottom reflector where in steady state conditions the temperatures are high enough that corrosion can take place. This can be seen in Figure 16 and Figure 17, where 0 metre is the lower end of the pebble bed; 11 metres being the top of the core.

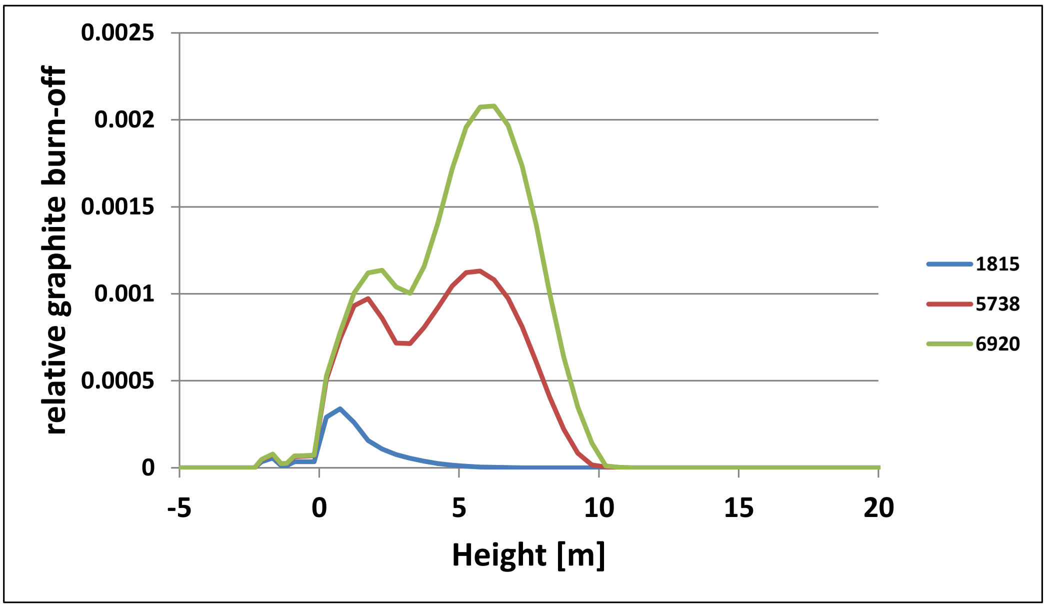

Figure 16: Relative graphite burn-off after 1,815 seconds, 5738 seconds and 6,920 seconds in the core zone at the centre line (at radius = 0.058 m)

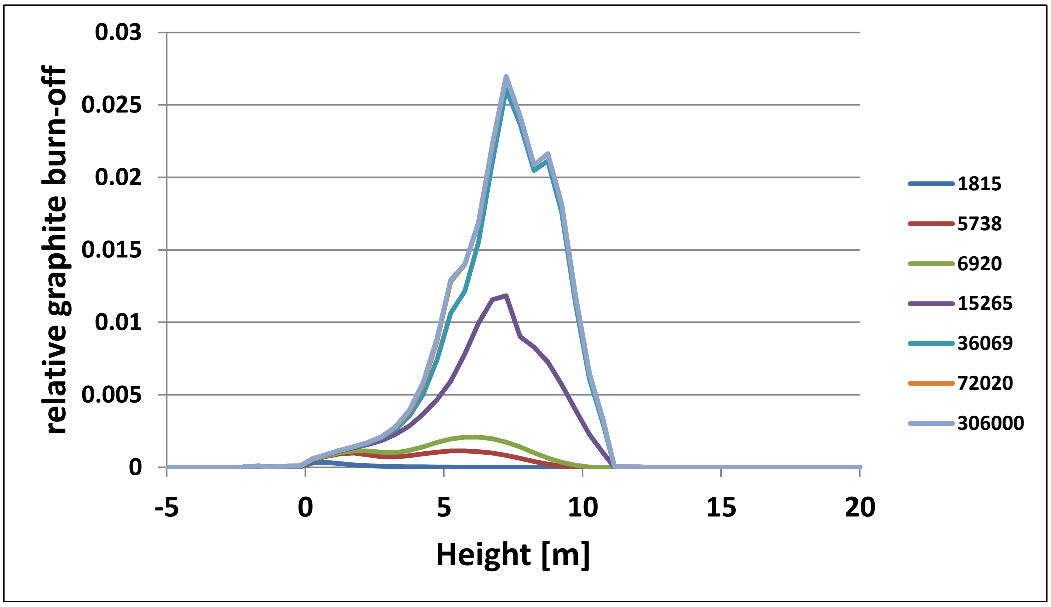

Figure 17: Relative graphite burn-off after 1,815 seconds, 5738 seconds, 6,920 seconds, 15,265 seconds, 36,069 seconds, 72,020 seconds and 306,000 seconds in the core zone at the centre line (at radius = 0.058 m)

Figure 18 shows the power increase as a consequence of water ingress without the insertion of control rods. In this case, the increase in power is stopped by the fuel temperature effect, see Figure 19. In the TINTE result the power level increases a bit faster and hence, presumably fuel and moderator temperatures increase more rapidly which could explain why the maximum value of TORT-TD/ATTICA^{3D} lags around one second behind. Also one has to keep in mind that for our calculation assumptions for the increase of the steam at the inlet has to be made. Unlike the TINTE code, the steam transport has to be simulated by changing boundary conditions.

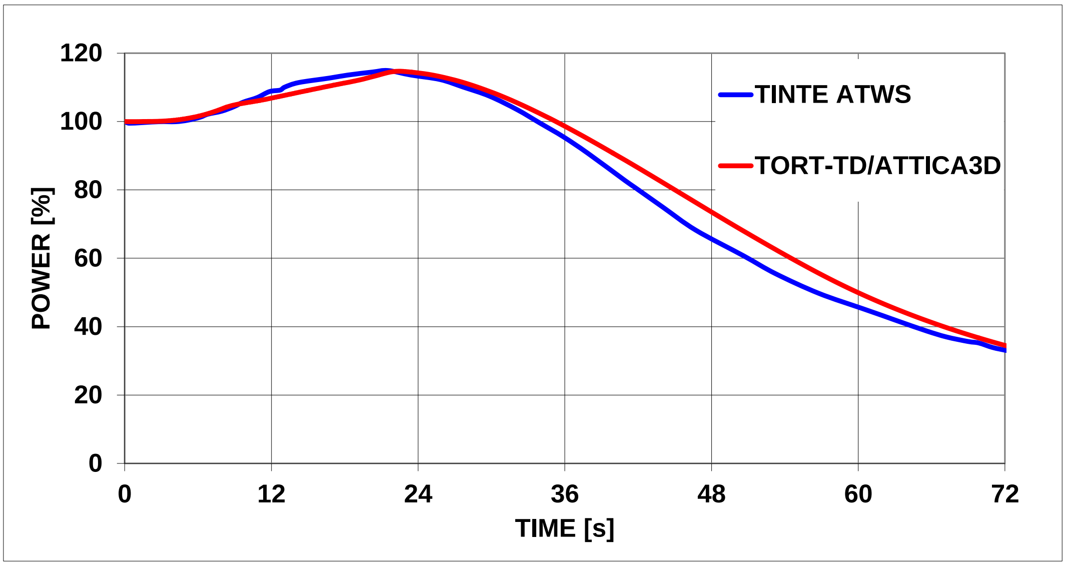

Figure 18: Short-term power increase in an anticipated transient without SCRAM case. The reactor is only shutdown by means of the fuel and moderator temperature effect.

Figure 19 below shows the total power again, but this time together with the decay heat which obviously does not change due to the short-term power increase. Also the fuel and moderator temperatures are presented which obviously start to rise as soon as the power increases.

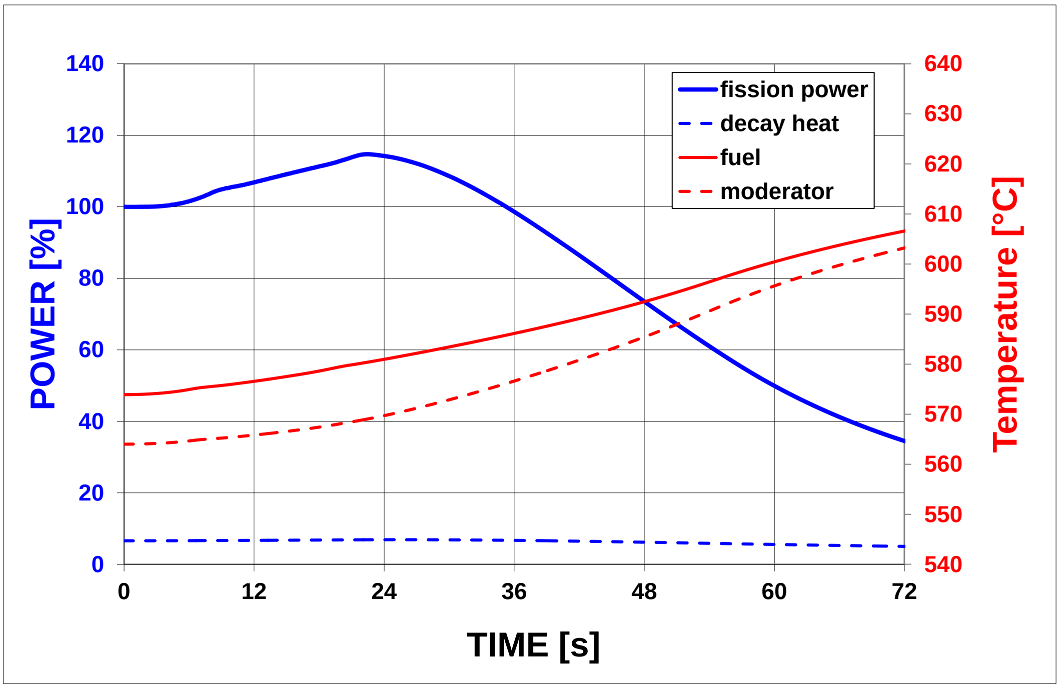

Figure 19: Fission and decay power (left axis), fuel and moderator (right axis) for the anticipated transient without Scram case.

4.1 Discussion of results

Design basis accident:
The relative graphite burn-off is a quantity that is normalised on the initial amount of graphite. When after 306,000 seconds the burn-off is 0.027, it means that only 2.7% of the graphite is corroded. Although the outer matrix graphite shell without fuel already comprises 42% of the fuel sphere volume this is not the percentage of graphite that can be lost without reaching the first particles. In a pebble bed, the gas can only flow in the porous region. The fuel pebbles are always subjected to anisotropic corrosion since the part that is perpendicular facing a flow channel will be stronger corroded than if the flow is parallel to the pebble surface. This fact can be accounted for by using a shape factor. In [4], it is stated that using a shape factor of 6 the admissible loss of graphite from the outer shell is 7%.
The results of ATTICA^{3D} show a maximum corrosion within the core region of 2.7%. It is therefore far away from the fuel region and now coated particle will be attacked by the steam atmosphere.

The HTR-PM is designed with two helium purification systems to remove chemicals such as H_2O, O_2, CO, CO_2, N_2, H_2, CH_4 etc within the coolant; one in operation, the other one serves as back-up if the first purification system fails.
For mitigation measures of the case of water ingress in the HTR-PM there is a specially designed stand-by accident helium purification system line connected to both systems, to remove chemical impurities. This purification system has special water seperators and has a throughput of 100% of the primary circuit inventory per hour. This would further decrease the steam and hydrogen content in the reactor for long-term corrosion effects. But it is not included in the safety related systems of the HTR-PM and is assumed **not** to work to give a conservative estimate.
Therefore, the corrosion attack of the core will be less if the helium purification system with the water separators operates as designed.

Anticipated transient without SCRAM:
For the ATWS case, the boundary conditions were taken from [4][5]. Like in the DBA scenario, the rise of partial pressure was subject to assumption. Furthermore, since TORT-TD/ATTICA3D only models the primary circuit until the gas outlet, feedback of the loss of heat sink was not accounted for. Due to this the gas inlet temperatures will rise quickly because the steam generator will not remove heat and is itself emergency-drained. With increasing inlet temperature, the moderator will be passively heated in the upper part of the reactor and - with knowledge of the negative moderator effect - will lead to a reactivity decrease. This applies also to the DBA scenario.

To account for the feedback of increasing inlet temperature and more realistic increase of the steam partial pressure, the boundary conditions are taken from a SPECTRA calcuation presented in D22.22 of the ARCHER project [12] and are presented in the next chapter.

5 Improvements for water ingress scenario

The results from the code system TORT-TD/ATTICA3D in the previous deliverable D22.24 were produced assuming boundary conditions outside the reactor pressure vessel. I.e the increase of pressure due to the break in the steam generator tube and the subsequent ingress of either steam or water into the primary circuit were determined by Zheng et al. [4]. However, the change of partial pressure of water, especially in case of long-term corrosion case, could not be predicted. Hence the assumption was taken that the partial pressure will linearly decrease to zero within 500,000 seconds after the initiation of the accident. This approach was also used for the ATWS case in the previous chapter that is newly presented

To be able to account for the decrease of steam partial pressure more realistically, it was decided to use the transient boundary conditions at the in- and outlet determined with the system code SPECTRA. For SPECTRA, options for water ingress were implemented within the course of the ARCHER project. Also important mechanisms like the first and secondary pressure relief system of the primary system to limit the maximum pressure possible within the RPV, are modeled to account for the reduction of the helium/steam/corrosion product inventory in case of an opening. Additionally the effect of steam generator drainage system and blower flaps are modeled. The results and models for the water ingress with SPECTRA are presented in the ARCHER deliverable D22.22 [12].
These delivered more appropriate boundary conditions for the case of a steam generator tube rupture (SGTR). For the SGTR there are two major differences for the location of break, the break in the lower part where the water is still liquid, and in the upper part, where there is only steam. Additionally, some sub-cases

were analysed with respect to safety actions available. These counter-measures are emergency drainage of the steam generator and closing of the blower flaps to prevent a formation of a natural convection loop through core and steam generator. The analysed cases with respect to break locations are

Bottom: Water plenum
1) Single tube rupture, emergency drainage system works, blower flaps close (Bot-all)
2) Single tube rupture, emergency drainage system fails, blower flaps close (Bot-NoDrain)
3) Single tube rupture, emergency drainage system fails, blower flaps open (Bot-NoDrain-NoFlaps)

Top: Steam plenum
4) Single tube rupture, emergency drainage system works, blower flaps close (Top-all)
5) Single tube rupture, emergency drainage system fails, blower flaps close (Top-NoDrain)
6) Single tube rupture, emergency drainage system fails, blower flaps open (Top-NoDrain-NoFlaps)

In all cases, the SGTR occurs at t = 0 seconds, the humidity sensor detects humidity at t = 10 seconds and the reactor is scrammed with a 2 seconds delay at t = 12 seconds together with the stop of blower which will decrease rotation speed until the time of blower flaps closing at t = 42 seconds.. The steam drainage system is initiated at t = 20 seconds and lasts till t = 63 seconds.

The upper six cases are compared with repect to the steam density entering the core region to find the most severe case of steam ingress. The mass flows with respect to the scenario is presented in Figure 20.

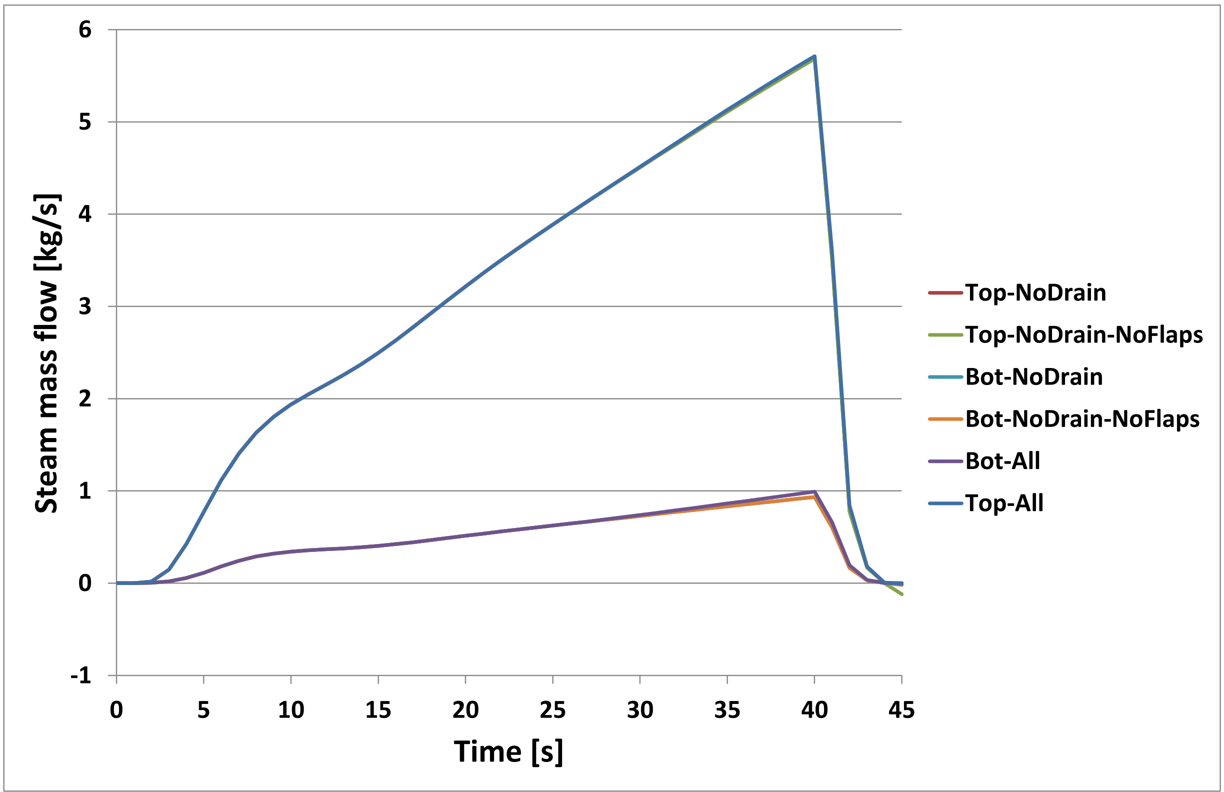

Figure 20: Different steam mass flows at the coaxial gas duct for case 1) - 6)

It is obvious that at the top location the steam mass flow at the RPV inlet is much higher. The reason for this is that if the break occurs in the lower part, there is a protective water layer hindering the steam from directly entering the primary circuit. Instead the water level drops and the liquid water is evaporated by the residual heat within the primary side of the steam generator. When comparing these values to the evolution presented in Figure 6 it is obvious that the steam flow rate was somewhat overestimated. Especially the increase of steam mass flow is very different. While the steam mass flow reached approximately the same value of 5.5 kg/s the time until this happens was assumed to be within the first 10 seconds, whereas Figure 20 shows this takes about 40 seconds to reach this flow rate.
Also, the inlet temperatures are increasing. This is shown in Figure 21. Compared to the analyses in the previous chapter the temperatures increase by 50 K for the case 1) and is still around 30 K for all other cases. This will heat up the moderator and therefore will have a negative effect on the reactivity.

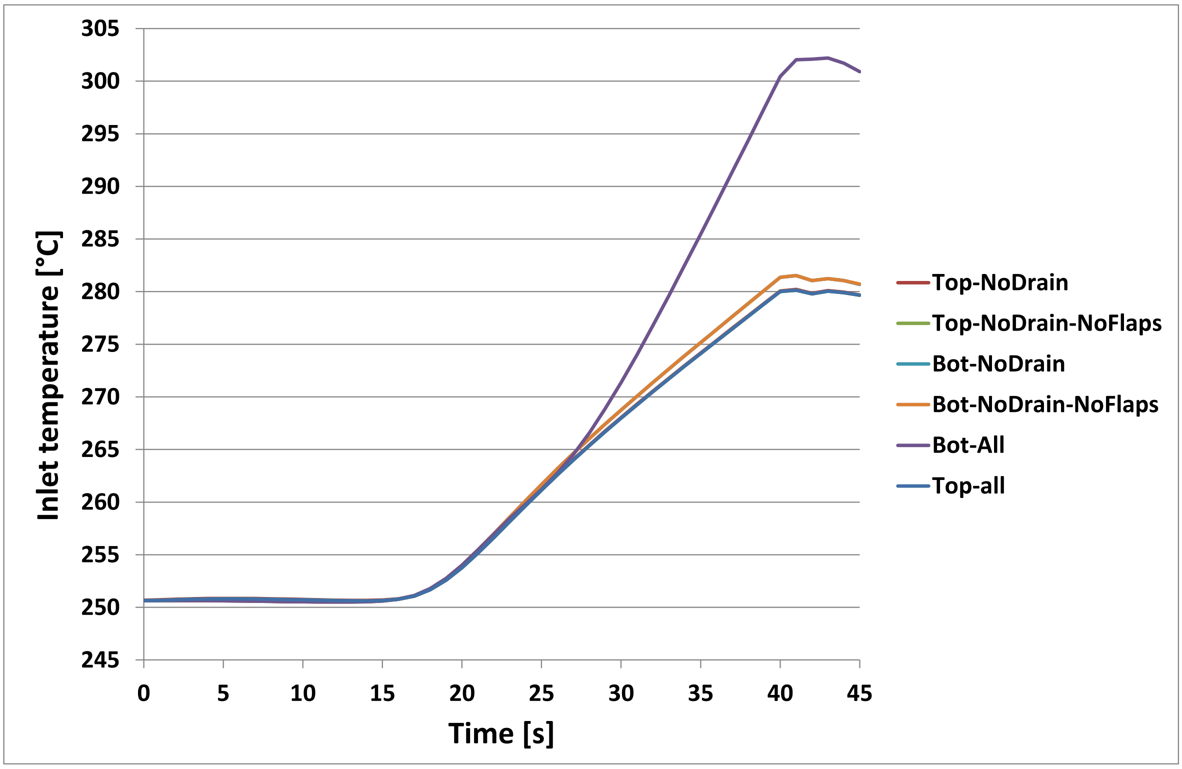

Figure 21: Increase of inlet temperatures at the coaxial gas duct within the first 45 seconds for case 1) – 6)

For the coupled analysis with the above shown boundary conditions the cases 1) and 4) are selected. For both cases a DBA and a ATWS are simulated. Since the steam mass flow rate increases less and the inlet temperature increase like shown above, the resulting power increase will be somewhat lower. The DBA where control rods are inserted the power excursion is stopped by absorbers. In the ATWS case only the feedback from boundary conditions will determine the behaviour.

Case 1 (Bot-all) ATWS

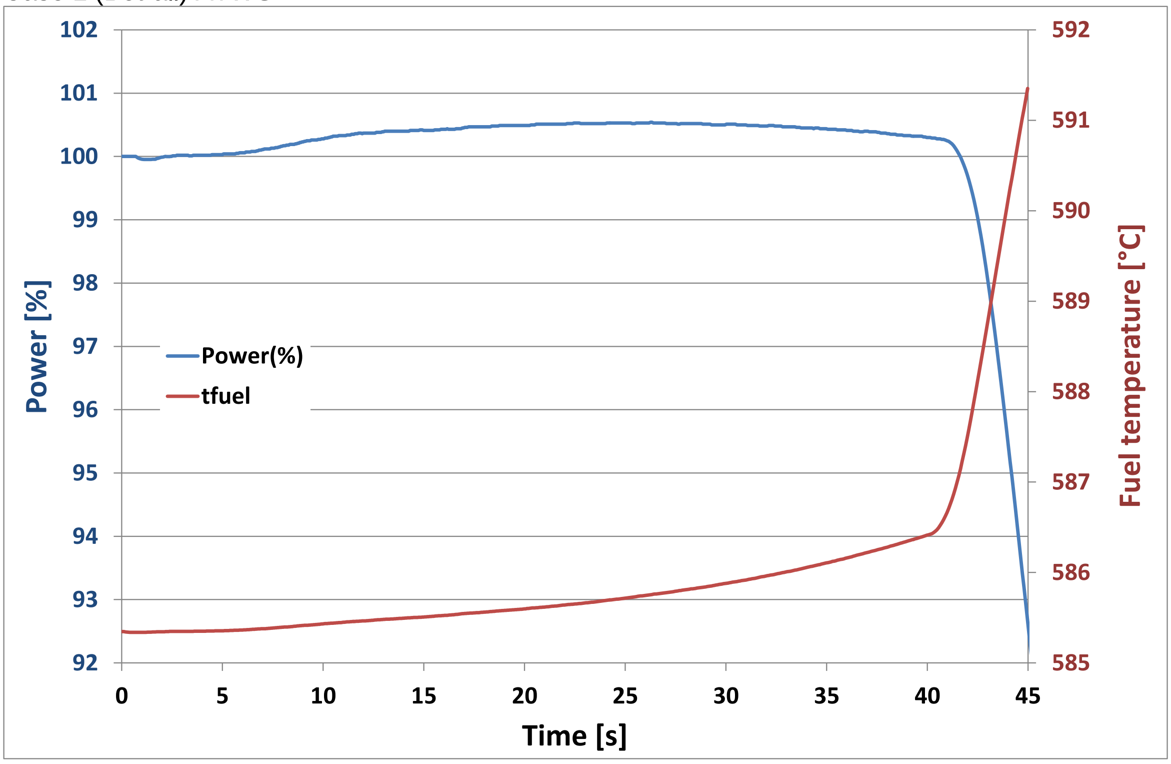

Figure 22: Left axis: Power increase, right axis: Increase of average fuel temperature of case 1)

Case 1 (Bot-all) DBA

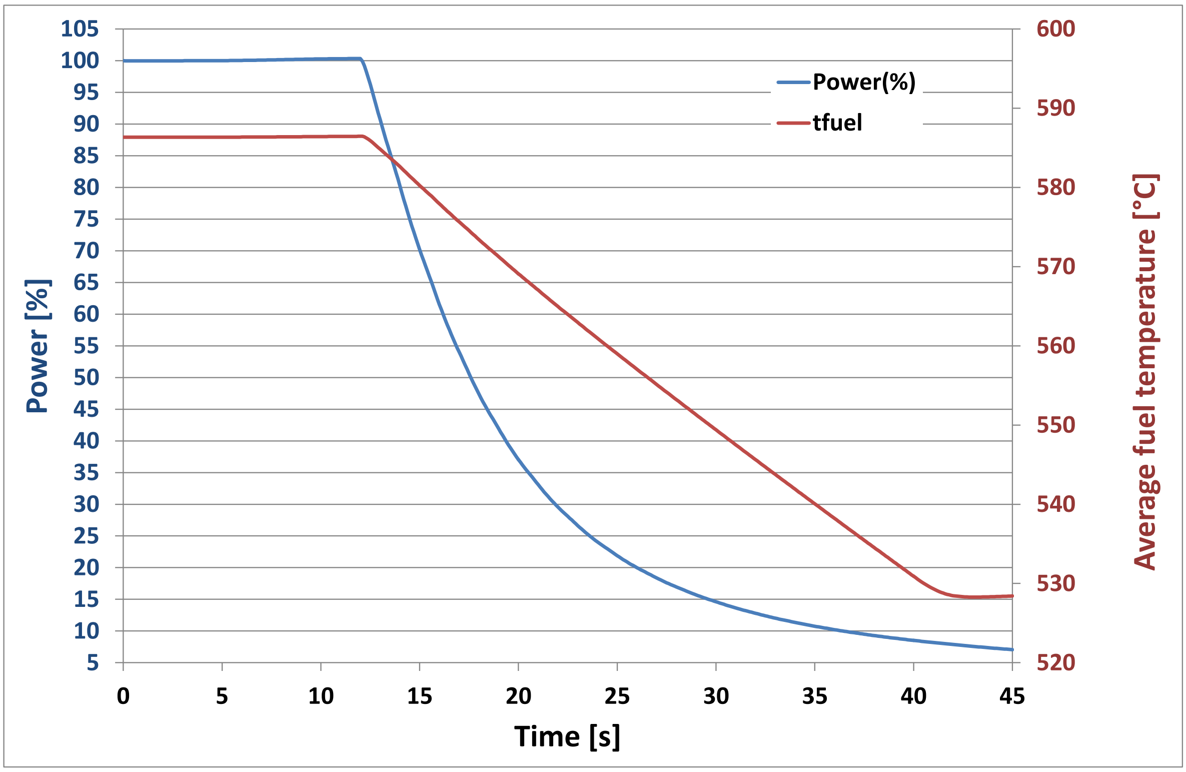

Figure 23: Left axis: Power evolution, right axis: average fuel temperatures for the design basis accident where rods are inserted at t = 12 seconds

In Figure 22 and Figure 23, the evolution of power and the average fuel temperatures are shown. Of course, in the DBA the power and the fuel temperature drop instantaneously after start of rod dropping. For the ATWS case, the power increases by only 0.54 % after 25 seconds. The increase in the fuel temperature is somewhat small and approx. 1 K until the stop of the mass flow when a pressurised loss of forced cooling case starts and the core heats up with decay heat.

Case 4) (Top-all) ATWS

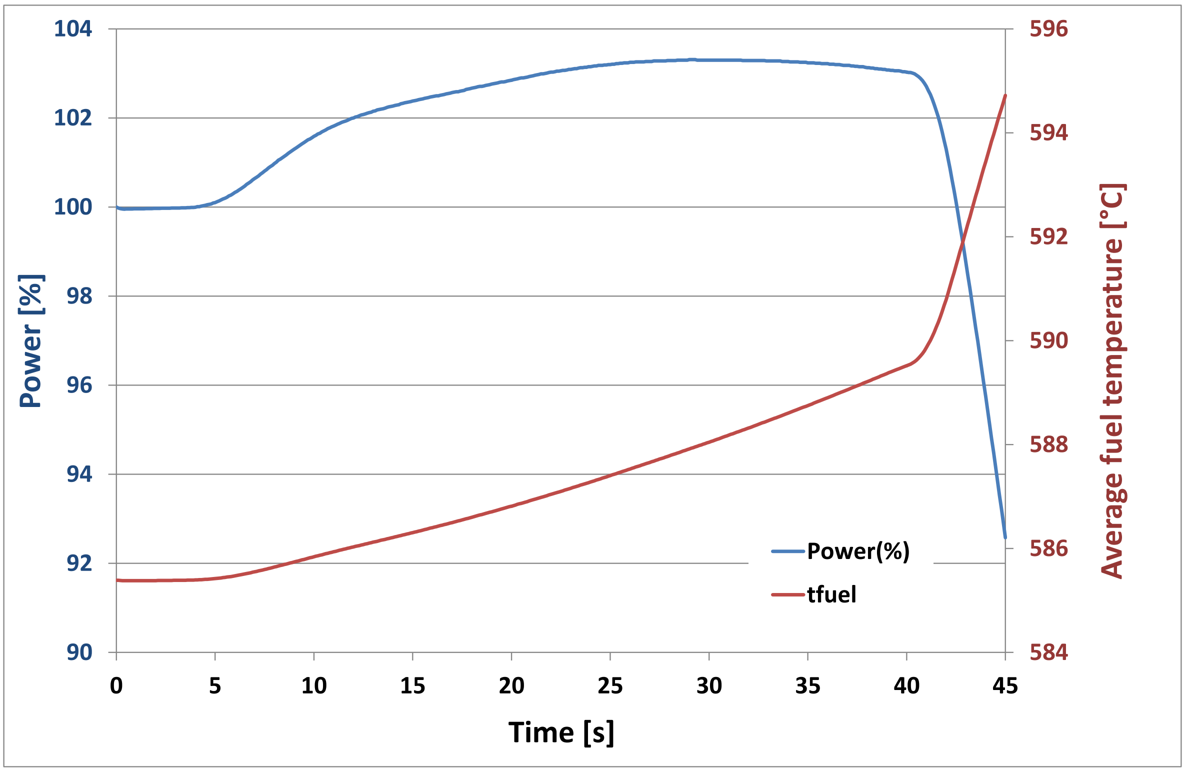

Figure 24: Left axis: evolution of power, right axis: evolution of average fuel temperatures for case 4)

Case 4) (Top-all) DBA

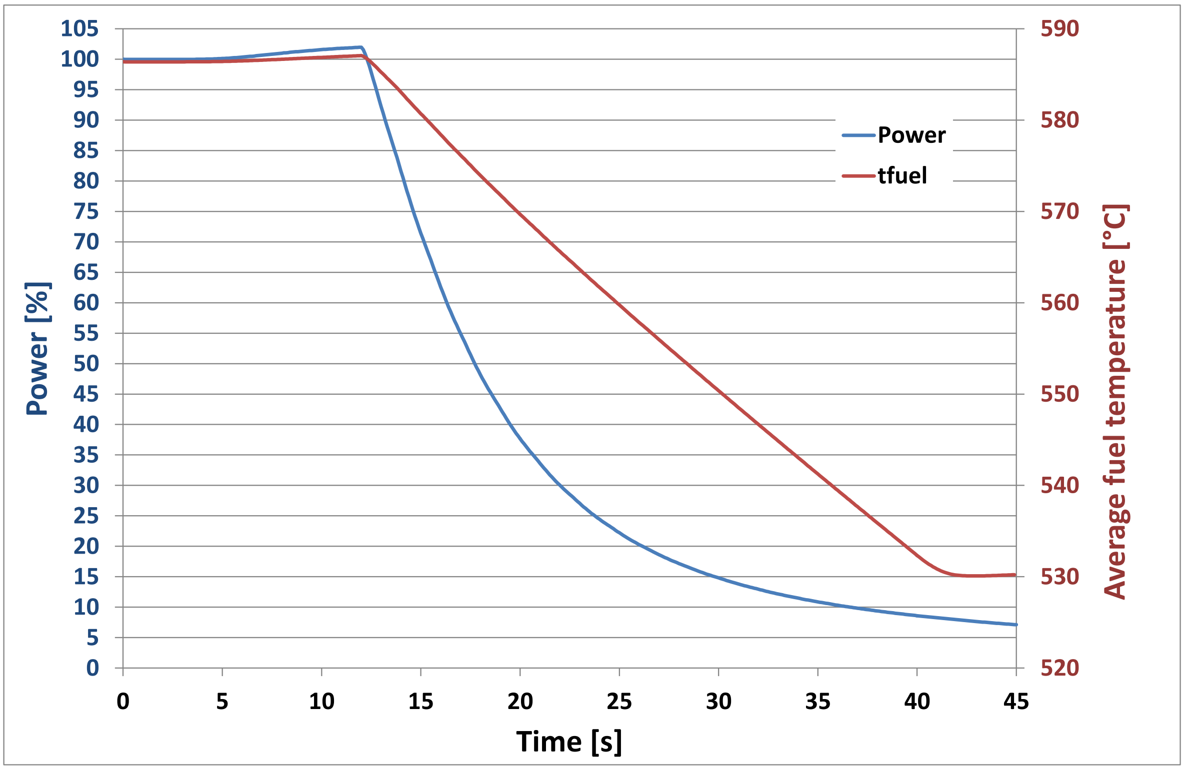

Figure 25: Left axis: Power evolution, right axis: evolution of average fuel temperature for case 4)

In Figure 24 and Figure 25, the power and temperature evolution is shown for the ATWS and the DBA cases. Like in case 1) the DBA shows decreasing temperatures and power after rod insertion is activated. In the ATWS case, the increase of power amounts to 3.31 % and the average fuel temperatures increase by 4 K. At 42 seconds, when the blower stops the power decreases and the core heats up with decay heat.

Remark: For the DBA cases of case 1) and 4) the rods are inserted over 30 seconds (12 – 42 seconds) and not over 50 seconds. But this does not matter in this case, since the emergency rod drop speed is 1 m/s and take less than 5 seconds for the emergency insertion.

5.1 Long term corrosion

Unlike expected, the boundary conditions delivered by SPECTRA for the long-term corrosion imposed programming changes in our code ATTICA3D, since generally the direction of mass flow changes multiple times within the cases considered. This led to unbearably long calculation times, and therefore has to be left out in this deliverable, although it was planned.

6 Summary and outlook

In the first chapter an introduction to XS generation and the procedural steps to obtain these are described. After description of the considered cases, the scenario to be calculated was described. The earlier generated XS were then also tested and compared to results of a water ingress calculation with TINTE done by Zheng et al [4]. Here a good agreement with the TINTE results from the paper could be obtained. After presentation of the short-term results a long-term analysis including corrosion was presented. As a finding it could be shown that corrosion over up to 80 hours will lead to a moderate attack of the fuel elements. The maximum burn-off of graphite is only 2.7%. But it is only after 7% of burn-off that the first coated particles will be subject to corrosion (attack by carbon monoxide).

New in this deliverable is the comparison of the anticipated transient without scram with the TINTE code. Here comparable power increase could be obtained. Besides the fission power evolution the decay heat production is also shown but is negligible since the transient lasts only less than a minute. Within this time no significant increase of decay heat can be detected.

With the boundary conditions obtained from the SPECTRA primary circuit calculation, new coupled calculations were performed. The power increase for the cases is less than predicted with SPECTRA. The maximum power increase for a break in the steam plenum was 3.31 %, whereas the power increase for the break in the water plenum yielded only 0.54 % of power increase. These results do not suggest inadmissibly high temperatures due to power excursion following increase of moderator within the pebble bed.

The relatively small increase of steam flow rate compared to the previous analysis and the increase of the coolant inlet temperatures can explain a much smaller power excursion than before.
Unless the possible steam flow rate is not restricted to such small values as presented in chapter 5, the long-term corrosion seems to be the more important issue.

7 References

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8 Annexes

Annex 1 – Document approval by beneficiaries' internal QA

Annex 2 – Title 2

8.1 Annex 1: Document approval by beneficiaries' internal QA

Fill involved beneficiaries as appropriate (mandatory for Milestones and Deliverables, but optional for other document type)

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