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Calculation of the Fission Product Release for the HTR-10 based on its Operation History

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Abstract – *Since the first criticality of the HTR-10 test reactor in 2000, a rather complex operation history was performed. As the HTR-10 is the only pebble bed reactor in operation today delivering experimental data for HTR simulation codes, an attempt was made to simulate the whole reactor operation up to the presence. Special emphasis was put on the fission product release behaviour as it is an important safety aspect of such a reactor. The operation history has to be simulated with respect to the neutronics, fluid mechanics and depletion to get a detailed knowledge about the time-dependent nuclide inventory. In this paper we report about such a simulation with VSOP 99/11 and our new fission product release code STACY. While STACY (Source Term Analysis Code System) so far was able to calculate the fission product release rates in case of an equilibrium core and during transients, it now can also be applied to running-in-phases. This coupling demonstrates a first step towards an HCP Prototype. Based on the published power histogram of the HTR-10 and additional information about the fuel loading and shuffling, a coupled neutronics, fluid dynamics and depletion calculation was performed. Special emphasis was put on the complex fuel-shuffling scheme within both VSOP and STACY. The simulations have shown that the HTR-10 up to now generated about 2580 MWd while reshuffling the core about 2.3 times. Within this paper, STACY results for the equilibrium core will be compared with FRESCO-II results being published by INET. Compared to these release rates, which are based on a few user defined life histories, in this new approach the fission product release rates of Ag-110m, Cs-137, Sr-90 and I-131 have been simulated for about 4000 tracer pebbles with STACY. For the calculation of the HTR-10 operation history time-dependent release rates are being presented as well.*

I. INTRODUCTION

The 10 MW pebble bed Helium cooled high temperature reactor, called HTR-10, has been built at the Institute of Nuclear Energy Technology of Tsinghua University. Table 1 gives an overview of some main geometrical data and operation parameters at fuel load and Table 2 shows some characteristic values of the fuel element design.

In the framework of an IAEA benchmark (CRP-5) [1] calculations with the system code VSOP 99/05 were performed in Jülich to investigate the initial criticality of the HTR-10. The minimal filling level in the atmosphere of air that is needed for the first criticality was calculated. Within the CRP-5, aspects such as the filling level at which the first criticality is attained under air atmosphere are being studied with pre-defined temperatures.

Now the simulation of the running-in phase up to the current state was done with the FZJ system code VSOP 99/09 [2] to prepare the calculation of the fission product release.

Table 1: Geometry and parameters at full load

Core diameter	150.0 cm
Average core height	197.0 cm
Number of control rods	10
Thermal power	10 MW
Helium mass flow	4.3 kg/s
Helium inlet temperature	250°C
Helium outlet temperature	700°C

Table 2: Core content

	Cylindrical core volume	Cone volume
Fuel fraction	57 %	0 %
Dummy fraction	43 %	100 %
Heavy metal	5 g/sphere	
U235-Enrichment	17.0 w.-%	
Number of coated	83000	
Number of fuel	27000	
Average number of	5	
Average burn up	80000 MWd/t_{HM}	
Average power	2 MW/m^3	
Average residence	1080 d	

As a part of the Archer project, the fission product release rates of main nuclides of interest are investigated. For this purpose, the before mentioned VSOP model has been extended which allows to model the operating history of the HTR-10.

II. MODELLING OF THE HTR-10 IN VSOP

As a first step within the simulation chain, the operating history of the reactor up to the current core status needs to be simulated. Not all details regarding the current so-called "running-in phase" are known. For this reason, the running-in phase being presented in this paper is a possible running-in phase which takes the boundary conditions being published by INET [3] into account as far as possible. Boundary conditions for which any data could be found in literature have been resolved by reasonable assumptions.

II.A. Condensation of the operation history

The HTR-10, being operated at INET at the Tsinghua University in Beijing is a test reactor with a nominal power of 10 MW_{th}. Many tests have been conducted to demonstrate different safety aspects of HTR. The first criticality was achieved on December 1, 2000 [4]. Up to now, the reactor was operated at different power levels for different periods of time including shutdown phases. Table 3 shows the year-averaged energies.

Table 3: Condensed operating history of HTR-10

Year	Operation [d]	Average [MWd]	Average [MW]	EFPD [d]
2003	106	258.9	2.44	25.89
2004	168	708.5	4.22	70.85
2005	149	821.4	5.51	82.14
2006	97	532.0	5.48	53.20
2007	49	182.6	3.73	18.26
total	569	2503.4	21.38	250.34

A better approximation of the operation history is given in Fig. 1. Fig. 1 has been digitalized in a first step. In a second step the resulting curve has been condensed to form a power histogram, containing time steps with a constant power level (see Fig. 2). The resulting power histogram can be used within simulations.

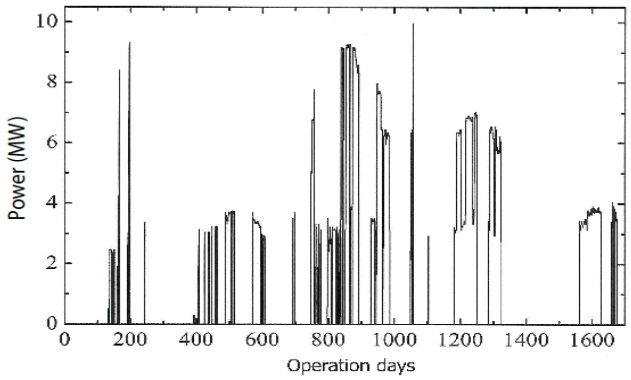

Fig.1: Power as a function of time of HTR-10 [3]

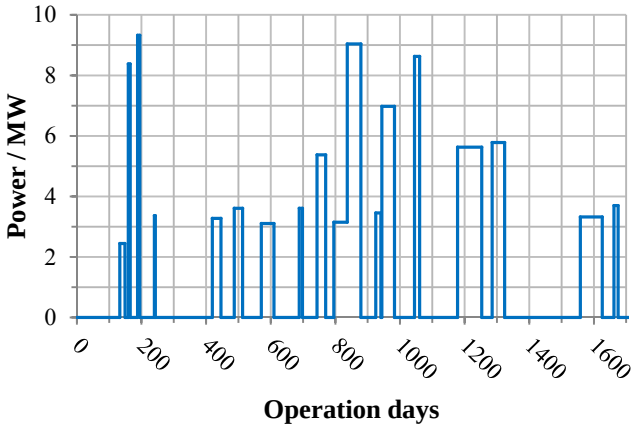

Fig. 2: Condensed power histogram of HTR-10

II.B. Control rod calibration

For the running–in phase, a correct modelling of the control rods is necessary to calculate the different core status at the different power levels including the zero power phases. As a first step, the control rods have been modelled and calibrated. Due to limitations of VSOP with respect to the geometry, the control rods cannot be modelled individually. The "grey curtain approach" is applied instead. It is assumed that the position for each control rod is equal and the rods are smeared out along the azimuthal direction. The control rods need to be calibrated in the two dimensional model in such way, that the effect of a movement of all control rods is equal to a detailed description of the rods. The calibration is performed by adjusting the nuclide densities of e.g. boron in case all control rods are totally inserted in case of the equilibrium core. Fig. 3 shows the results of the VSOP calculations for several control rods positions after calibration in comparison to NEA benchmark results.

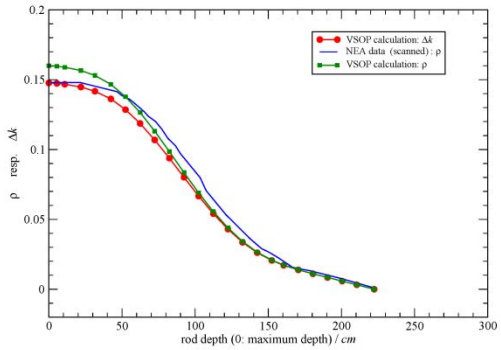

Fig. 3: Comparison of control rod worth

It can be seen, that the results are in good agreement. This means that the grey curtain approach can be used to simulate the different core states of the HTR-10.

II.C. Fuel shuffling

In [5] the ideal running-in phase without any shutdown phases is discussed. Induced by the relative low height to diameter ratio of the HTR-10 core, the flow pattern cannot be modelled as a cylinder where pebbles at different radii move with equal velocity through the core. The flow velocity can be adapted in VSOP by the following procedure. The pebble flow is modelled according to the stream tube model. For this reason, the pebble bed's volume is divided in so-called channels (stream tubes), which are limited by flow curves and or a reflector surface in the radial direction (see Fig. 4). Each of these channels is subdivided in regions of equal volume. The continuous pebble flow is being approximated by a step-wise moving of the complete content of a region to the region below. Thus, the number of regions per channel influences the speed with which fuel flows through a channel. Each of the regions contains in general one or several batches, where each batch represents a specific fuel type at a specific (burnup) state. For each region belonging to the same channel, the number of batches is equal. Batches have no distinct position within a region; they are being treated as a homogenous mixture.

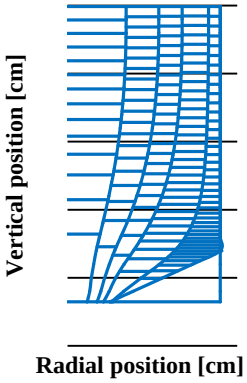

Fig. 4: Fuel channels, divided into regions of equal volume in case of HTR-10 (not drawn to scale

In case of the equilibrium core, the fuel scheme is regular (see Fig 5). Per definition, the way of shuffling the fuel is constant and not changed.

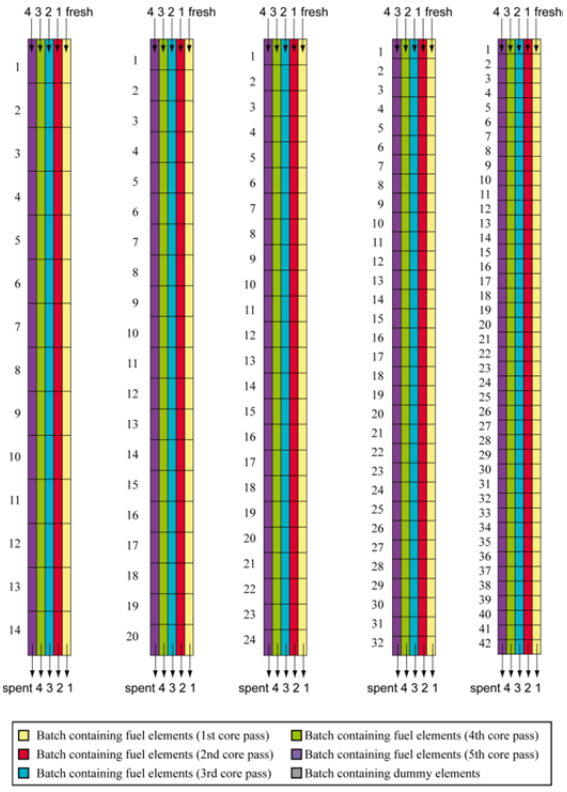

Fig. 5: Exemplary fuel shuffling scheme for a time step during the running-in phase

Before reaching the equilibrium core, the fuel shuffling scheme is rather irregular. An exemplary fuel shuffling scheme during the running-in phase of HTR-10 which demonstrates the core content at a specific time point and all movements crossing the core boundary at the upper and lower end of the pebble bed is displayed in Fig. 6. It can be seen, that the lower part of the innermost fuel channel contains fuel elements which are about to finish the first, second or third core pass. No dummy elements, which are part of the initial core are present in this channel. In the outermost fuel channel on the other hand, dummy elements have not been discharged yet.

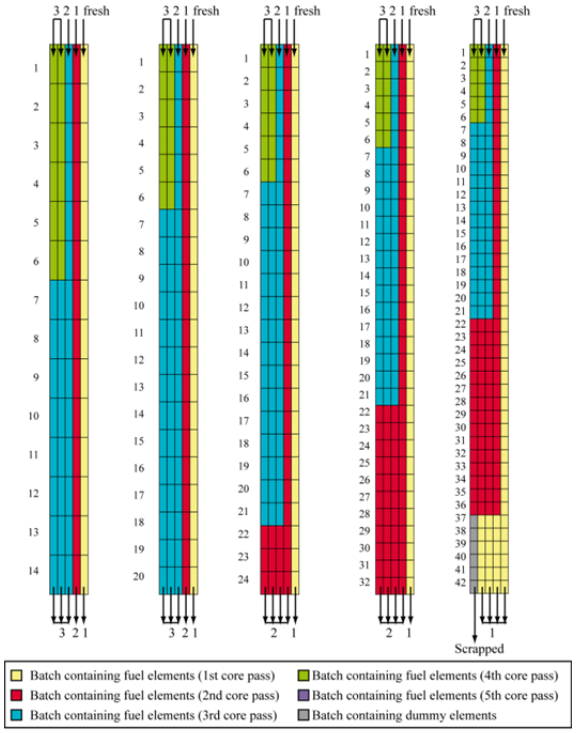

Fig. 6: Fuel shuffling scheme during a certain time step during the running-in phase

Due to the different number of axial regions (14/20/24/32/42) for the different fuel channels, an individual fuel shuffling scheme exists for almost each time step during the running-in phase.

II.D. Fluid dynamics

To simulate the power histogram as shown in Fig. 2, the nominal flow rate and the coolant in- and outlet temperature have to be varied. They are functions of the power level at which the reactor is being operated (part-load diagram), and be needed for the fluid dynamics calculations with the fluid dynamics module of VSOP called THERMIX. Due to the many similarities between the HTR-Module and the HTR-10, it has been assumed that the part-load diagram of the HTR-Module is (at least) similar to the one of the HTR-10 (as detailed boundary conditions are not published for the HTR-10). For this reason, the part-load diagram of the HTR-Module [6] has been used to set the missing parameters of the fluid dynamics calculations. It is assumed that the flow rate of the coolant is varied linearly with the power level at which the reactor is operated. At the same time while varying the power level, the in- and outlet temperature of the coolant is adjusted between the boundaries defined by the part-load diagram of the HTR-Module.

II.E. Results from the VSOP calculation

For a given fuel element configuration and a given core loading the multiplication factor k_{eff} is determined only by the position of the control rods and the shuffling frequency of the fuel. Both values can be used in VSOP only in a discrete form. The variable “DELDAY” describes a large time step in which the fuel is circulated twice. In a continuously operating reactor k_{eff} (control rods in constant position) can be set precisely by varying “DELDAY”. VSOP simulation show, that the average residence time in a region is equal to 8.76 days in case of the equilibrium core. In the case of real running-in of the HTR-10 there are many different power levels including zero loads. So “DELDAY” has to be adjusted to the given time frame. Calculated reactivity values are displayed in Fig. 7.

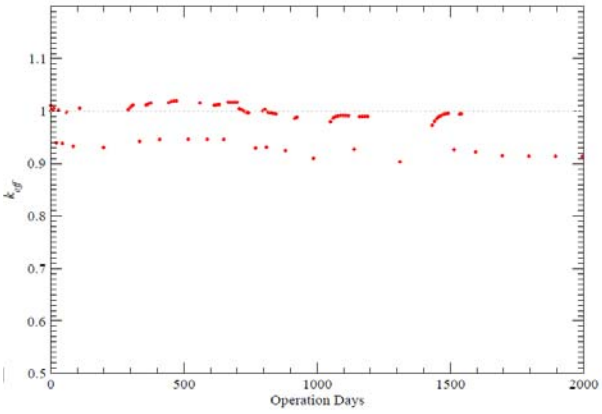

Fig. 7: Calculated reactivity values from VSOP during operation and zero load

An exact adjustment on $k_{eff} = 1{,}0$ is not possible. A technical alternative to compensate the discrepancy would be possible by moving the control rods. This approach, however, does not reflect reality, as it merely a problem of numerical discretization would be corrected. Therefore, the control rods are not moved. However Fig. 7 shows that all values of k_{eff} calculated by VSOP account within the control range of the rods. In addition,

VSOP calculation show that the HTR-10 is far from reaching the equilibrium core.

II.F. Comparison of VSOP Results with Serpent

Comparisons have been conducted with a Serpent Model of the HTR-10. Serpent [7] is a Monte-Carlo neutronics code including burnup calculation which has been developed by the VTT Technical Research Center in Finland. Control rod worth values have been determined for the HTR-10 completely filled up with the mixture of fuel and dummy elements under helium atmosphere at operating pressure. Within the Serpent model, the Spheres are placed according to a regular lattice. By using a Monte-Carlo method, it is being determined which sphere is a fuel element and which is not. It is checked whether the average filling factor of 61 % and the fuel fractions in both core zones are achieved. In comparison to the VSOP model, the fuel pile at the upper end of the pebble bed is being modelled with an inclination of 20 degrees. VSOP results for the initial core were in good agreement with those from Serpent.

III. FISSION PRODUCT RELEASE

III.A. Introduction to STACY

Main target of the paper is the simulation of fission product release rates, based on the core states (presented by e.g. neutron flux distributions for the different time points) being discussed in chapter II. The fission product release rates of some nuclides which are of radiological importance have been calculated by using the Source Term Analysis Code System (STACY). First, STACY will be introduced and key improvements made with respect to its predecessor FRESCO-II will be discussed.

STACY incorporates all functionalities of legacy codes such as FRESCO-I FRESCO-II and PANAMA. After the modernization and merging of these codes to one single code module, the module has been extended towards spatially resolved full-core calculations. STACY is able to perform calculations for a vast number of single tracer fuel elements automatically. Due to the fact that a large number of tracer fuel elements are taken into account, the code considers spatial distributions for fuel temperatures and nuclide inventories within the core. Routines have been implemented which read the necessary data from a VSOP-output for example.

The first version of STACY was able to calculate the fission product release rates in case of an equilibrium core and during transients starting from this equilibrium core. The fission product release rate is being determined by simulating the release rates of a large number of individual, so called “tracer pebbles”. Great care is taken to position these tracer elements in such way, that the fission product release rates of these spheres are representative for the complete pebble bed.

An important aspect of the fission product release calculation is a correct description of the nuclide inventory, which is treated in STACY in more detail. Different from FRESCO-II, STACY uses the FZJ burnup core TNT (Topological Nuclide Transformation), which is comparable to the burnup code ORIGEN, to determine the time-depending inventory of the differing fission products of interest. FRESCO-II calculates fractional releases where the release fraction is related to the fuel element inventory at the end of the irradiation phase. The inventory between start and end of the irradiation phase is treated in a rudimentary way by using an analytical equation which assumes e.g. a constant birth rate of the nuclide during the irradiation phase. In case of an equilibrium core this equation is a good approximation for long-living nuclides. In case of short-living nuclides or in case of a running-in phase, a correct description of the time-depending nuclide inventories is necessary. An analytical equation cannot describe the time-depending inventories during a running-in phase with varying power levels and shutdown phases at all. In case of a spatial resolved fission product release calculation for the whole core, an individual burnup calculation for each pebble being studied. Neutron fluxes being calculated by VSOP are used to perform these burnup calculations.

STACY is coupled via both the input and output file of the multi-physics system code VSOP. For the calculation of the fission product release rates for the equilibrium core, it is sufficient to transfer solely one set of e.g. spatial broad group fluxes and fluid temperatures.

The simulation of the running-in phase as discussed in this paper poses some additional challenges in comparison to the simulation of the equilibrium core. Due to the fact that these spatial distributions vary over time during a running-in phase, these data sets have to be calculated for each individual time step by VSOP. To handle this larger amount of data, the data structure of STACY has been reorganized and extended. For each shuffling time step, an individual data object is initialized and filled with data which is extracted automatically from the VSOP output file. In addition, dummy elements can be considered.

Even if VSOP results form an important part of the STACY input, a number of differences between the codes exist. One of the main differences between the STACY and VSOP is that STACY shuffles

individual tracer pebbles instead of batches. In VSOP, normally, the properties of the batches are mixed at the end of each core pass in case of a HTR with MEDUL fuel (multiple core pass) strategy. Batches of all fuel channels which get unloaded at the lower end of the core are mixed and average batches are determined (see Fig. 8). Fig. 8 shows the development of the burnup in VSOP for the first two core passes. Due to the mixing, properties like the nuclide vector and burnup are homogenized, which leads to a loss of information.

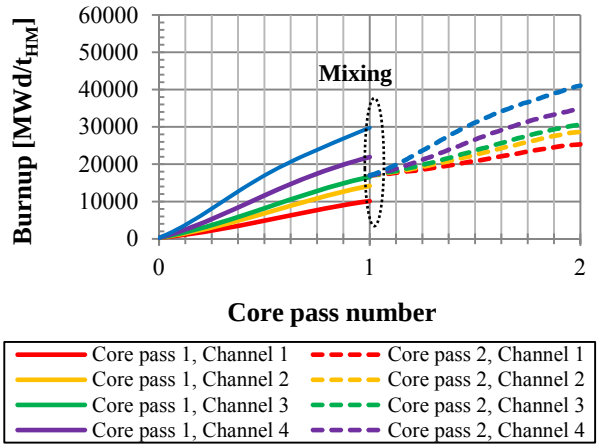

Fig. 8: Burnup as a function of the core pass number for the first two core passes

In case of a particle failure fraction or fission product release calculation it is not trivial to determine an average batch (representative fuel element) which behaves like the sum of the batches which are part of the mixture. It would be necessary to mix key properties like the fission product concentration profile along the fuel element radius and the number of failed coated particles. In addition, it is necessary to take multiple fuel element histories into account to achieve best-estimate values for fission product release rates.

For this reason, the fuel elements are represented by so-called tracer pebbles, which have a distinct position within the core in STACY. These pebbles are shuffled through the core along flow lines. These curves correspond to the limiting curves of the flow channels. By interpolation, intermediate flow curves are calculated in between the user-defined flow curves. In contrast to VSOP, the tracer pebbles are not mixed with other pebbles at the end of each core pass. Instead the burnup is checked and compared with the burnup target. If the target has not been reached, the pebble is reintroduced into the core. The new radial position of the tracer element is being determined by a random number generator.

Rather exceptional cases are taken into account, which can influence the overall fission product release perceivably. Due to the statistical mixing of batches after each core pass in the VSOP-calculation it is not possible to simulate the fission product release of one specific batch.

In case of an equilibrium core, only one set of fuel shuffling instructions (which define from where to what position fuel is being moved) have to be defined (see Fig. 5). Due to the complexity and irregularity of the fuel shuffling scheme during the running-in phase of the HTR-10, it is necessary to read all these individual sets of instructions and to process them.

Nonetheless, some connection has to be established between the batches and the tracer pebbles. The tracer pebbles being used in STACY are loaded in a similar way as the batches. The user-defined number of tracer pebbles is divided over the different batches according to the initial batch loading scheme. Each tracer pebble is linked to a batch which allows using the fuel shuffling instructions from VSOP without any modification. The tracer pebbles are moved almost continuously. The time step width used for the fission product release calculation is smaller than the residence time of a tracer pebble in a region. It is assumed that the tracer pebbles which are located at an equal vertical position before a fuel shuffling step are located at another, but equal position after the fuel shuffling step.

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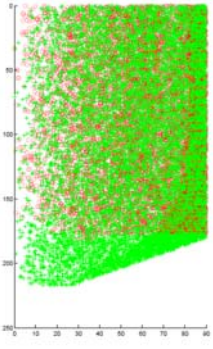

Fig. 9: Tracer elements in initial core of HTR-10

In the lower part of initial core, there are solely dummy elements (green). In this picture, the three dimensional positions of the pebbles are condensed to a two dimensional model. By doing so, the density of tracer pebbles is higher at larger radii.

An important aspect of the fission product release calculation is the correct description of the fuel management, because the life history of each individual tracer pebble depends mainly on the time-depending position through the active core. After the fuel shuffling code module in STACY has been extended to enable fuel shuffling during the running-in phase, a verification process has been conducted to verify the implementation. As an example, the content of the core is being displayed for two time steps in Fig. 10.

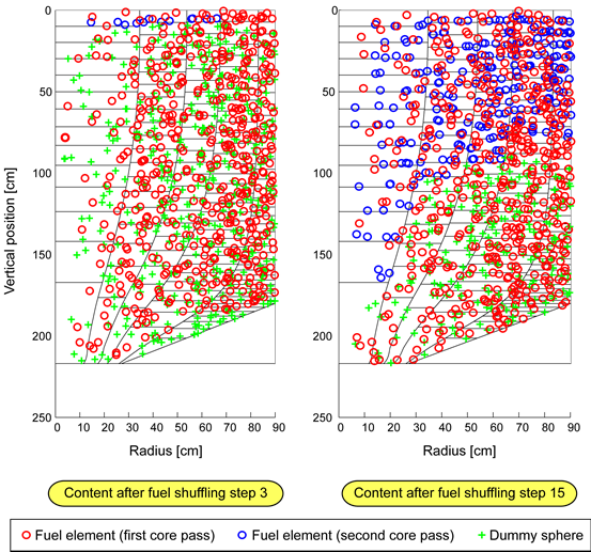

Fig. 10: Tracer elements at two different core status of HTR-10

In comparison to the calculation itself, which is based on 4000 fuel elements, only 1000 fuel elements are being displayed. Fig. 15 shows the core content after time step 3, at which fuel elements are reintroduced to the core for the first time (second core pass = blue circle). Due to the limited number of spheres and the randomness added by the Monte-Carlo approach, no fuel elements are added to the fourth and fifth fuel channel. As another example, the core content after 15 fuel shuffling steps is displayed. During the first two fuel shuffling steps a mixture of fresh fuel elements (red circle) and dummy spheres (green "+"-symbol) is added to the core. After these two fuel shuffling steps, no dummy spheres are added to the core. The first channel contains 14 regions. Due to this, after 15 fuel shuffling steps, the dummy spheres have reached the lowest region of fuel channel one.

These features enable to calculate the fission product release during the operating history of the HTR-10 and to describe the current state of the fuel within the reactor.

III.B. Input parameters of FP release calculation

In this section, the application of STACY to the HTR-10 is presented. First of all, an overview of input parameters which used for the fission product release calculations will be presented.

A main parameter within the fission product release calculation is the particle failure fraction distribution, due to the high retention level of intact coated particles. In STACY the coated particle failure fraction can be calculated after the PANAMA-model or the so-called "Trumpet Curve". The PANAMA-model considers the coated particle as a pressure vessel and calculates the failure fractions based on a rudimentary physical model. The trumpet curve on the other hand has been derived by Siemens Interatom for the HTR-Module and models the failure fraction as function of the fuel temperature and the burnup level (see Fig. 11).

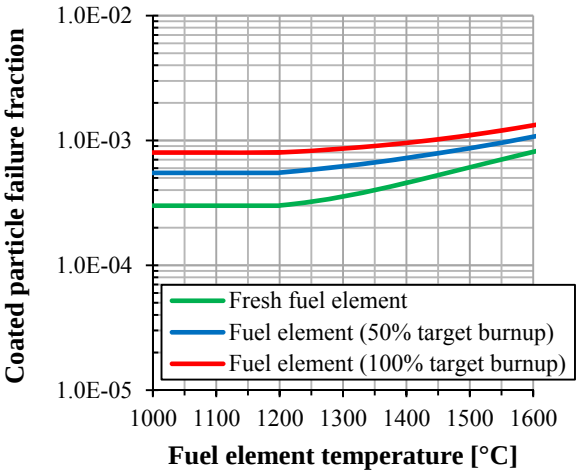

Fig. 11: So-called "Trumpet Curve" describing the coated particle failure fraction as a function of the fuel temperature and the burnup level

The curve does not take into account, how long the fuel is exposed to which temperatures. As soon as a certain temperature has been exceeded the corresponding failure fraction is assumed. For the HTR-10 under normal operating conditions, the fuel temperatures stay well below 1200°C. For these fuel temperatures, the particle failure fraction is only depending on the burnup level. The trumpet curve assumes that the failure fraction is increasing linearly with the burnup level. For this reason, the trumpet curve has been modified to fit with INET

specifications (see Fig. 11). A fresh fuel element has a coated particle defect fraction of 3.0×10^{-4} and a spent fuel element has a defect fraction of 8.0×10^{-4}.

Table 4: Diffusion coefficients [8],
Γ: Fast neutron fluence

	Cesium		Strontium	
	D_0 [m^2/s]	Q [kJ/mol]	D_0 [m^2/s]	Q [kJ/mol]
UO_2	5.6×10^{-8} 5.2×10^{-4}	209 362	2.2×10^{-3}	488
PyC	6.3×10^{-8}	222	2.3×10^{-6}	197
SiC	$5.5\times10^{-4}\cdot e^{\Gamma/5}$ 1.6×10^{-2}	125 514	1.2×10^{-9} 1.8×10^{6}	205 791

An excerpt of the diffusion coefficients being used are being displayed in table 4. These are the diffusion coefficients being proposed by the IAEA for fuel elements which comply with the quality of German fuel elements.

As earlier mentioned, the fuel elements are shuffled through the core for which the spatial neutron flux distributions have been calculated by VSOP. The time depending neutron fluxes for in case of four broad group fluxes is displayed for a random tracer pebble of the equilibrium core in Fig. 12.

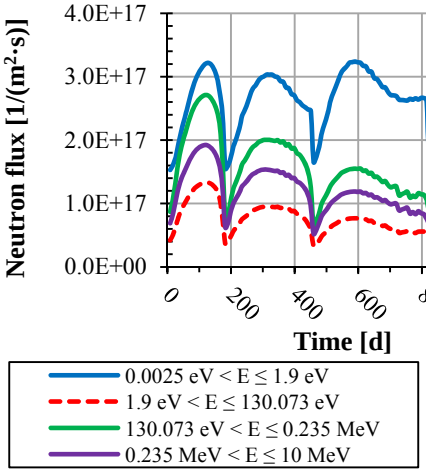

Fig. 12: Broad group neutron fluxes for a randomly picked fuel element of the HTR-10 equilibrium core

This randomly selected fuel element passes the core via fuel channel 2, 4, 5, 1 and 3 before achieving its final burnup (see Fig. 13). It can be seen that the residence time in the outermost fuel channel (during the third core pass) is longer due to the lower pebble flow velocity. The highest I-131 is achieved within the innermost channel due to the higher neutron fluxes.

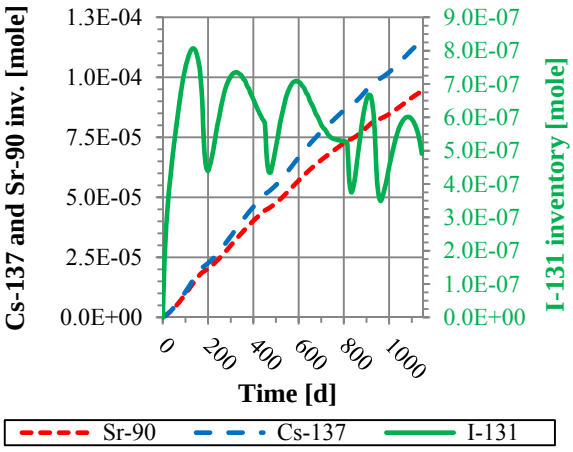

Fig. 13: Cs-137, Sr-90 and I-131 inventory a randomly picked fuel element of the HTR-10 equilibrium core

In comparison to the broad group fluxes for a fuel element within the equilibrium core, a randomly picked fuel element which is part of the initial core is exposed to neutron fluxes which look very different (see Fig. 14).

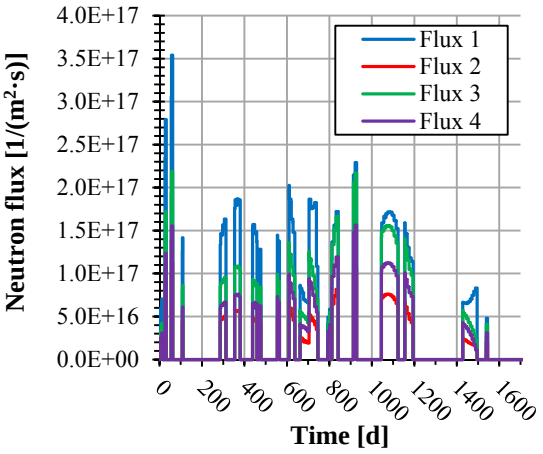

Fig. 14: Broad group fluxes of one HTR-10 fuel element during the running-in phase (see figure Fig. 10 for the energy boundaries of the broad group neutron fluxes)

The burnup calculations are performed by using TNT. The inventories of Sr-90, Cs-137 and I-131 are displayed in Fig. 15. The long-lived nuclides Sr-90 and Cs-137 do only decay to a negligible amount during the shutdown phases. The short-lived nuclide I-131 decays in some longer shutdown phases completely. Due to the time step widths chosen, the exponential shape of the time depending inventories cannot be recognized.

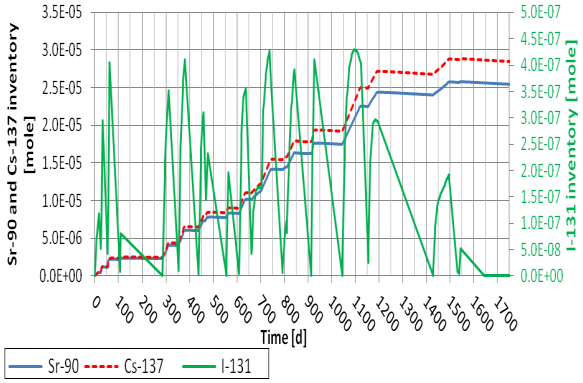

Fig. 15: Sr-90, Cs-137 and I-131 inventory of one HTR-10 fuel element during the running-in phase

Another important aspect of the fission product release calculation is the central fuel temperatures (see Fig. 16). During each core pass, the central fuel temperature increases gradually. The power level at which the reactor is being operated has only a minor effect on the fuel temperature.

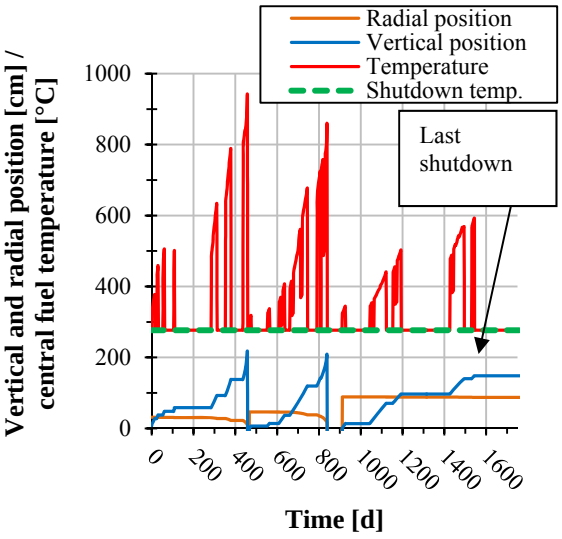

Fig. 16: Position and central fuel temperature of a random tracer fuel element during the running-in phase of the HTR-10

A constant fuel temperature is used for all core zones during shutdown phases (see Fig. 16). This is necessary because THERMIX calculations do not converge during a shutdown phase. This constant temperature does not take into account that the fuel temperature during shutdown phases is both depending on the position of the fuel element and the time-depending decay power. Nevertheless, during shutdown phases, fuel temperatures decrease rapidly and are rather low. For this temperature range, the diffusion coefficients of the nuclides of interest do not depend heavily on the "exact" fuel temperature. Thus, a constant temperature is a good approximation for these operating phases.

It can be seen that the fuel elements do not move during shutdown phases. The radial position decreases gradually at the end of each core pass. In case of the equilibrium core the residence time within the second fuel channel is shorter in comparison to the residence time within the first fuel channel because the core is being shuffled continuously. In case of the running-in phase, the residence time is shorter in the second fuel channel than in the first fuel channel due to longer shutdown phases at the beginning of reactor operation. After being discharged at the end of the second core pass, the tracer element resides for approximately 50 days in the fuel handling system (out-of-core box), because after discharge, the core is shutdown.

A fission product release calculation is only conducted for the fuel elements. The deposition on and diffusion into dummy elements and reflector surfaces is neglected. Combining the FP release results from all tracer-elements results in the fission product release rates of the complete core during the running-in phase.

III.C. Equilibrium core

Combining the results from all tracer elements results in the fission product release rates of the complete core during the running-in phase. A fission product release calculation is only conducted for the fuel elements. The deposition on and diffusion into dummy elements is neglected due to unknown boundary conditions. The dummy elements are shuffled by the fuel management to have a correct representation of the content of each region. This allows for a correct scaling of the fission product release rates from the individual tracer pebbles to overall core release rates.

The tracer pebbles are gradually being introduced to the core at a constant rate in case of an equilibrium core calculation. Due to this, the number of tracer pebbles increases steadily until tracer pebbles are discharged, because they have achieved the burnup target. By doing so, the number of tracer pebbles for which a fission product calculation has to be performed is lower than having a constant number of pebbles within the core from the very beginning. This is only needed to simulate a running-in phase, which will be shown later. In case of a model where each fuel channel has an equal number of regions, equilibrium conditions are attained as soon as the first tracer elements are discharged. At this moment, discharged fuel elements have fulfilled a complete life history and each core zone contains a mixture of fuel elements at a different stage during their life history.

In case of the HTR-10, where each fuel channel contains a different number of regions, it is more difficult to predict when equilibrium conditions from the fission product release point of view have been attained. Due to the higher residence time, it can be assured that the burnup target is attained after the nominal number of core passes in the outermost fuel channel. Thus, even a fuel element which goes only through the outermost channel has fulfilled a complete life history. Other life histories are shorter.

In former studies only a few fuel elements have been analyzed. The fuel elements are exposed to maximum fuel temperatures. Results from these individual calculations have been extrapolated to derive specific release rates for the whole active core. In comparison to the INET calculations, in the new STACY approach, the spatial distributions of e.g. fuel temperatures and nuclide inventories are taken into account instead of maximum values only.

No precise data are available regarding the boundary conditions being used for the fission product release calculation being published in [9]. The paper does not mention e.g. the diffusion coefficients applied and the uranium contamination of the different coated particle layers on one hand and the graphite grains and pores (graphite grain boundaries) on the other hand. For this reason, a detailed comparison between the results in published in [9] and results calculated by using STACY cannot be made. Next, no spatial temperature distribution is given, which does not allow for a reproduction of the calculations performed.

The fission product release of long-lived nuclides is mainly affected by taking the fuel temperature and nuclide inventory distribution into account. The STACY approach results in lower fission product release rates for long-living nuclides (see Table 5). Especially the fission product release rates of Cs-137 and Ag-110m are about two magnitudes lower.

Table 5: Fission product release rates in the HTR-10 equilibrium core (STACY calculation)

	Cs-137	Sr-90	Ag-110m	I-131
	[Bq/(MW$_t$·h)]	[Bq/(MW$_t$·h)]	[Bq/(MW$_t$·h)]	[Bq/(MW$_t$·h)]
INET	8.9 x10^{3}	2.5 x10^{-1}	1.5 x10^{2}	4.9 x10^{6}
FZJ	5.2 x10^{1}	8.7 x10^{-2}	2.5 x10^{-1}	2.2 x10^{7}
INET / FZJ	1.7 x10^{2}	2.9 x10^{0}	6.0 x10^{2}	2.2 x10^{-1}

In addition to the spatially resolved calculation, there is a second effect which affects the fission product release calculation of short-lived nuclides. Due to the simplified calculation of the nuclide inventory in FRESCO-II, the equilibrium inventory of a short-lived nuclide is achieved after a few days. From this time point, the inventory of a short-lived nuclide is constant. In case of reactor with a MEDUL fuelling strategy, each fuel element is shuffled through the core and is exposed to a time-depending neutron flux. Due to this, the neutron flux increases at the beginning of each core pass and the rate at which the nuclide is being produced is higher than the rate at which the nuclide is being removed due to decay. At the end of each core pass, the neutron flux decreases and the removal rate is higher than the production rate. Thus, the inventory is periodic and the maximum inventory is located in the middle part of the core. Neglecting this influences the fission product release.

III.D. Running-in phase

As an important part of the fission product release calculation, the time-depending nuclide inventories are calculated while taking the time-depending neutron fluxes for each tracer pebble into account. These tracer pebbles are shuffled through the core according to the fuel management schemes being defined in the VSOP input file. Combining all results leads to the fission product release rate during the running-in phase. The Cs-137 release rate during the operation history is displayed in Fig. 17. It can be seen, that the release rate is lower than the value of the equilibrium core.

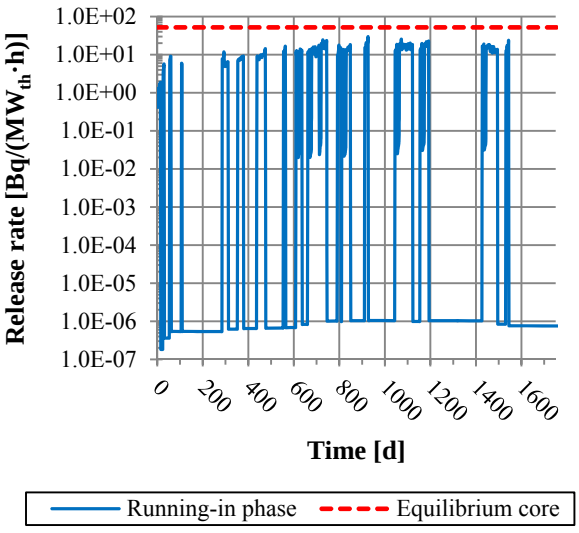

Fig. 17: Cs-137 release during the running-in phase

The time depending release rates of Sr-90 and Ag-110m are comparable. The I-131 release rate during the operation history is displayed in Fig. 18. It can be seen that after a restart of the reactor after a shutdown phase, the release rate is close to the equilibrium release rate after a short time delay. After a shutdown, the release rate decreases rapidly as expected.

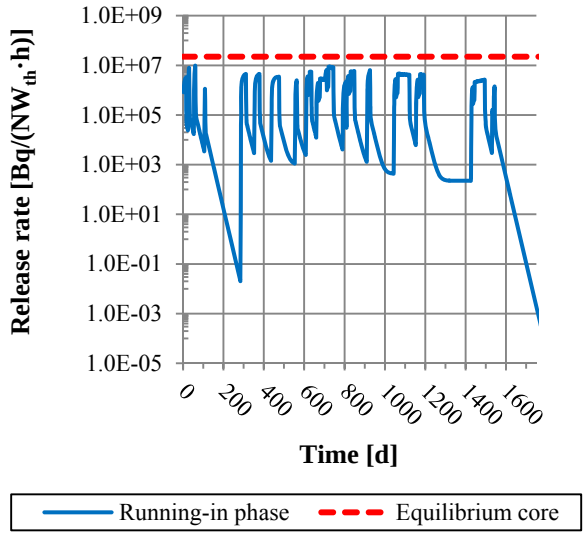

Fig. 18: I-131 release during the running-in phase

Due to its half-life, the release rate of I-131 varies less than the release rate of long-living nuclides like Cs-137 (see Fig. 17).

IV. CONCLUSION

Within this work, the fission product release of the HTR-10 over its operation time was simulated. The fuel temperature distribution and the neutron flux distribution which serve as input to the STACY calculation are generated by a coupled neutronics and fluid dynamics calculation with the FZJ system code VSOP 99/09. Within this calculation, the power histogram published by INET and additional assumptions based on the HTR-Modul were taken as further input.

The STACY code, which calculates the fission product release of an HTR core by taking a large number of so called tracer pebbles into account developed, has been extended for this study in such a way, that all core states can be treated. This extended version has been used to calculate the fission product release rates of main nuclides of interest within the HTR-10 for the running-in phase.

Fission product release rates have been published by INET only for the HTR-10 equilibrium core. Only a comparison can be conducted for the equilibrium core. The release rate of Sr-90 is about a factor of 3, the rate of Cs-137 about a factor of 170, the rate of I-131 about a factor of 5 and the release rate of Ag-110m about two magnitudes lower. During the actual operation phase, the fission product release rates are comparable to those of the equilibrium core as long as the reactor is operated. After shutdown, the release rates decrease rapidly.

Measurements comparable to the VAMPYR experiments conducted within the AVR are required to validate the overall release results of the HTR-10.

V. ACKNOWLEGMENT

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