Models for reactor physics calculations for HTR pebble bed modular reactors

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Abstract

For reactor physics calculations for modular pebble bed reactors program systems and nuclear data have to be used to get reliable results for important operational and safety related parameters. For cylindrical or annular core geometry such parameters can be calculated with the modular program system ZIRKUS and additionally with common reactor physics programs based on SN and Monte Carlo theory. The modules of the ZIRKUS system were developed by Framatome ANP GmbH and extensively used for design calculations for the 200 MW e HTR-MODUL and verified by means of experiments and operational data mainly from AVR. The models used in ZIRKUS will be described and discussed for typical design calculations for HTR pebble bed reactors.

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1. Introduction

For the determination of the power distribution and safety related parameters of a HTR pebble bed reactor (Lohnert, 1992) coupled neutron physics, thermal hydraulic and burn-up calculations must be performed to regard all important physical interactions. The most common method for treating the neutron diffusion in HTR is the solution of multigroup diffusion equation for core and reflector with some special treatment of absorbers in the side reflector and the cavity above the pebble bed core. For the preparation of cross-sections in multigroup approximation for the core the double heterogeneity of the fuel and the corresponding self shielding have to be taken into account. The calculation of fuel and moderator temperature which influences the neutron spectra and effective resonance absorption must be calculated by an adequate model taking into account the coolant flow through reflector and pebble bed core. Finally, the circulation of pebbles through the core and the burn-up during their residence in high neutron flux density has to be calculated. The variety of different tasks requires a flexible program system to perform such calculations for the initial cores up to the equilibrium core of a distinct core design or reload strategy.

The modular program system ZIRKUS (Feltes, 1993) developed by Framatome ANP GmbH was designed for the realization of various reactor physics calculations for HTR cores with pebble fuel. The main parts of ZIRKUS are function modules, a general data base and a control module for control of both execution of single modules or sequences of modules. An important feature of ZIRKUS is the data exchange between the modules via a general data base by the mass storage archiving and retrieval system MARS. For every function module, a specific input can be generated and stored before execution. By means of a special command the execution of a single module.

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can be started via the control module. The results of this execution can be put on output device but also on the general database for other function modules. Successive executions of function modules (chains or modular sequences) can be formulated and identified by an arbitrary identifier. ZIRKUS is therefore a flexible tool for calculation of important integral and differential parameters. Furthermore, it can be used to prepare a cross-section database for transient calculations by representing the few group cross-sections as a function of different parameters like fuel temperature, moderator temperature, reflector temperature, Xe-density, H-density (if water ingress is expected) and control rod positions.

2. Description of the reactor physics modules for stationary calculations

The main modules used in ZIRKUS and their tasks are listed in Table 1. They are mainly used for following tasks:

- design of the first core;
- design of the interim (transition) core;
- design of the equilibrium core;
- preparation of cross-section data for transient calculations;
- preparation of afterheat distribution in the core for transient thermal hydraulics calculations.

The principal ZIRKUS flow chart for execution of one reload step is shown in Fig. 1. To realize the pebble flow, the core is radially subdivided into flow channels with individual flow velocities of the pebbles. At every reload step the pebbles will be transported from its present axial fuel zone (burn-up zone) to the corresponding zone below. Every radial channel therefore is subdivided into a number of axial zones with identical volume. The number of zones depends from relative velocity of pebble in the radial channels. The module NIVERM manages all load, reload, transport and withdrawal of pebbles for an arbitrary number of passes through the core. For every burn-up zone and for every fuel cycle there are three isotope vectors, one for the upper limit of the burn-up zone, one for the bottom limit of the burn-up zone and one corresponding to the average values of the burn-up zone. These concentration records are then used for the calculation of cross-sections in the following diffusion module, while the first two isotope vectors pass through the burn-up module for every cycle and are subsequently transferred back to NIVERM. The isotopes contained in a generic cell are burnt with the neutron flux associated to that cell starting from the upper limit of the burn-up zone.

The GGA Code MICROX (Wälti and Koch, 1972) calculates for a two zone model the fast slowing down flux density spectrum and the thermal flux density spectrum. In every spectral zone the code solves the neutron slowing down and the thermalization equation on a detailed energy grid of a lattice cell, where it considers two regions: one representing the fuel grains with coating and binder, and the other representing the moderator. The fluxes in the two regions are then coupled by collision probabilities based on the flat flux.

<table>
<thead>
<tr>
<th>Module</th>
<th>Task</th>
</tr>
</thead>
<tbody>
<tr>
<td>KUGEL</td>
<td>Specification of fresh fuel for every fuel type</td>
</tr>
<tr>
<td>NIVERM</td>
<td>Refueling and transport of pebbles for next time step</td>
</tr>
<tr>
<td>NEWA</td>
<td>Calculation of Dancoff factors for double heterogeneous fuel</td>
</tr>
<tr>
<td>MICROX</td>
<td>Microscopic few group cross-sections for spectral zones</td>
</tr>
<tr>
<td>MAGRIU</td>
<td>Macroscopic cross-sections</td>
</tr>
<tr>
<td>HIBLOCK</td>
<td>Solution of multigroup 2D diffusion equation</td>
</tr>
<tr>
<td>BUCK</td>
<td>Calculation of buckling for spectral zones</td>
</tr>
<tr>
<td>SBURN</td>
<td>Solving of burn-up equations for burn-up zones, calculation of number densities</td>
</tr>
<tr>
<td>NZW</td>
<td>Calculation of afterheat distribution as a function of time after shutdown</td>
</tr>
<tr>
<td>THERMIX/KONVEK</td>
<td>Calculation of fuel and moderator temperatures</td>
</tr>
</tbody>
</table>
approximation. A second level of heterogeneity can be treated, i.e. the fuel region may have grain structure up to two different types of fuel particles. The coupling between the single fuel elements is realised with the Dancoff-Ginsburg factor for grained spheres in a pebble bed calculated by the module MEWA. The MICROX cross-section data are available for three energy ranges. The fast and epithermal energy range is subdivided into 99 fine groups in the energy zone from 14.9 MeV to 0.414 MeV, the thermal energy region is represented by 101 energy points up to 2.38 eV. To take into account the scattering behaviour of the carbon, the fast/thermal energy boundary is chosen to 2.38 eV. For the absorbers in the resonance region (like Th-232, U-238, Pu-240) there are further data with Doppler broadened resonance cross-sections for the resolved resonance range between 0.4 and 3.5 keV, represented by 8500 energy points. For this superfine energy grid the slowing down equation is solved and the corresponding cross-sections are condensed to the group structure of the 99 group library (GAM library). These method allows the consideration of overlapping of resonances from all nuclides present in the fuel zone. The macroscopic and microscopic groups constants are condensed in the spectrum determined each time and for every spectral zone from 93 fast and 101 thermal groups respectively to a few group cross-section set. The spectrum is calculated by means of BN-theory using bucklings from the 2D diffusion calculation to account for the leakage in the corresponding spectral zone. It is proven that this method allows the use of comparable few energy groups for the correct solution of the diffusion equation for the full core if there is an (buckling) iteration between spectral calculation in MICROX and the 2D diffusion code HBLOCK provided that the spectral zones were properly chosen. Therefore, for most calculations only four or seven energy groups were used.

Due to the modular structure of ZIRKUS there is a separate module: MAGRU for calculating the macroscopic cross-sections for the burn-up zones using number densities calculated by the reload module NIVERM and the burn-up module SBURN and the microscopic group constants for the spectral zones calculated by the module MICROX. To regard the streaming effects in the pebble bed core, the diffusion constant is corrected to take into account the void volume between the fuel elements in the core by means of the so called Behrens correction, (Behrens, 1949; Lieberoth and Stojadinovic, 1980). By this correction the diffusion constant is increased by about 8%, for energy values over 0.1 MeV, and by 16% for energy values below 0.1 MeV.

The diffusion code HBLOCK (Lieberoth, 1977) calculates the neutrons flux distribution in the core and the eigenvalue $k_{eff}$. It is a coarse mesh method, characterised as following:

The differential Operator $divD\phi$ is formulated as usual in a five points difference notation. It forms a linear equation system for discrete flux values with nodal points on a R-Z grid of the core fine mesh. Each node can be a vertex of up to for four materials (original equations system and original mesh). By transferring to the coarse mesh equations, the most part of the original equations are in general omitted and auxiliary assumptions are made to enable solvabilit-

![Flow chart for performing coupled neutronics, thermal hydraulics and burn-up calculations by program system ZIRKUS.](image-url)
ity of the remaining equations. The transversal leakage is here determined through a spatial linear interpolation from the respective leakage of each nodal point of the “big lines”. The cavity over the core is simulated with an effective diffusion constant, dependent on the direction. The void is treated as a diffusion zone with vanishing reaction cross-sections. The diffusion cross-sections are synthetic values only dependent on radius and height (Gerwin and Scherer, 1979).

The VORNEK module is used for the calculation of the power density distribution (averaged over all the fuel elements at each point). The distribution of the power density can also be axially smoothed, i.e. discontinuity of the fission cross-sections at the limit of the burn-up zones can be approximated with the method of the least squared errors and as fission cross-sections are assigned at each case the cross-sections values in the middle of the zone. The module does not consider the power delivered by the gamma irradiation in the zones of the graphite reflector: Such heat sources have to be taken into account separately by coupled neutron gamma transport calculations. The BUCK module calculates bucklings according to diffusion approximation for the spectral zones for use in the MICROX code. The SBURN module calculates the burn-up of heavy metal and build up of fission products for the burn-up zones for every reload step. Explicitly regarded in SBURN are 16 actinides and 69 fission products. The remaining absorption due to all not explicitly treated fission products is taken into account by means of four pseudo fission products.

Additionally to the ZIRKUS calculations also 1D and 2D transport calculations by the SN programs ANISN (Engle, 1967) and DORT (Rhoades, 1993) for the homogenisation of effective cross-sections for the reflector with inserted control rods or small absorber spheres (SAS) and 3D Monte Carlo calculations with the programs MOCCA (Lieberoth, 1968, Lieberoth, 1984; Feltes, 1996), KENO (Hollenbach et al., 1999) or MCNP (Briesmeister, 1997) for accurate determination of reactivity effects can be performed. This calculations will be performed outside the ZIRKUS regime, the results, however can be integrated into the data base of ZIRKUS and used from the corresponding modules.

Furthermore, transport effects which cannot be solved by diffusion theory can be treated by means of a coupling of diffusion and transport theory via response matrices (Neumann et al., 1987). Especially the treatment of the neutron streaming in the cavity between the pebble bed core and the upper reflector can be solved accurately by this method (Emendörfer, 1967; Bernnat et al., 1990).

3. Reactor physics modules for instationary calculations

For the simulation of reactivity transients in HTRs the program RZKIND was developed from Framatome, ANP GmbH. The description of the basic physics, the derived equations and method of their solution can be found in (Kindt and Kohtz, 1993). For the verification of the RZKIND code experimental results measured at the AVR high temperature reactor were compared with corresponding calculations (Kindt, 1986).

The KIND codes work on the basis of macroscopic one group cross-sections, which are implemented in the associated libraries as functions of feedback parameters in polynomial representations. The code consists mainly of three parts, which are executed sequentially within a time step:

- neutronic part to calculate the neutron flux density and the power density in the reactor core using
  - one-group neutron diffusion equation;
  - equations for the delayed neutrons in the usual representation with six groups;
  - equations for the fission product poisoning through Xe-135 (two isotopes) and through Sm-149 (chain with six isotopes);
  - special treatment of the decay heat (so called pseudo-emitters);
- thermal hydraulic part to calculate the temperatures of fuel elements (solid material) and coolant gas (helium) in the pebble bed (core) and in the case of RZKIND the temperatures in the reflector regions. The determination of the helium temperatures in the pebble bed assumes isolated channels with forced streaming;
- part for the characterisation of the neutronic/thermal-hydraulic feedback which enter in the form of modified cross-sections into the neutronic part. The main parameters are the fuel and moderator temperatures
Fig. 2. Infinite multiplication factor for HTR pebble fuel cell as a function of U load calculated with the Monte Carlo code MCNP, based on the data evaluations JEF-2.2, JENDL-3.2 and ENDF/B-VI.5.

Fig. 3. Comparison of the infinite multiplication factor calculated by Monte Carlo code MCNP for a HTR cell as a function of U load. The calculations were based on JEF-2.2 (reference value), JENDL-3.2 and ENDF/B-VI.5.
in the pebble bed, the distribution of the xenon concentration in the fuel elements, the distribution of the water steam concentration in the coolant (accidental water ingress) as well as the position of the control and shutdown elements.

The cooling gas passing the pebble bed is treated as flowing in independent radial channels with equal mass flow densities. Furthermore, only convective heat transfer at the side reflector is considered. This calculation model permits flow-rates down to 10% of the nominal mass flow. As an alternative, the thermal hydraulic part can be calculated by THERMIX/KONVEK.

There is a special (1D version of RZKIND) which is able to simulate the detailed transient heat up of fuel in the coated particles. This version can be used for investigation of integrity of fuel during hypothetical fast reactivity transient (Feltes et al., 1993).

All necessary nuclear and geometrical data for RZKIND are strongly related to the results of corresponding pre-calculated ZIRKUS results. These data are stored as polynomial functions of a variety of parameters (fuel and moderator temperature, reflector temperature, xenon concentration, control rod insertion, etc.) in a special prepared library.

4. Thermal hydraulics calculations

The neutron spectrum and the effective cross-section depend on temperature of moderator and fuel. Therefore, the neutronics calculations, also for stationary
cases, must be coupled with thermal hydraulics calculations. In ZIRKUS this can be performed by a very fast running code NECKAR or the 2D program THERMIX/KONVEK. THERMIX/KONVEK (Breitbach, 1979; Petersen, 1984; Verbornern, 1984) is a special tool for thermal hydraulic safety assessment studies for pebble-bed high-temperature reactors (HTR) especially the HTR-MODUL of Framatome ANP (Reutler and Lohnert, 1982). THERMIX/KONVEK simulates fluid flow along with the temperature distribution in fluid and pebble bed/conducting solids. The fluid flow is modelled by the Euler-equations as a steady state flow with no friction. The friction and pressure drop due to the pebble bed is modelled by KTA-rule 3102.3 with an additional term in the momentum equation. The temperature distribution in fluid and solid is modelled with two temperature equations. Fluid and solid are interacting by the additional friction and by heat transfer from pebble bed/solid to fluid. The properties of the fluid are implemented in accordance to KTA-rule 3102.1. In total, the fluid flow and heat transfer in the system is then described by the mass conservation equation, two momentum equations for two directions and two energy equations for fluid and solid. The pressure drop caused by the pebble bed is modelled in accordance to KTA-rule 3102.3. The convective heat transport from pebble and fluid is calculated by heat transfer coefficients defined by KTA-rule 3102.2. The NECKAR module is a very simplified method for calculating the temperature distribution in moderator and fuel for the case of forced convection. For more accurate solutions the THERMIX/KONVEK module is preferred. The thermal hydraulics calculations can be performed also by the code FRECON developed at IKE (Rieger, 1994) and the 3D heat convection module HEATING (Childs, 2000). The afterheat distribution in core due to the irradiation of fuel is calculated by means of the ZIRKUS module NZW. This module calculates this distribution as well as the U-239 and NP-239 number density as a function of time for the simulated
power history (Feltes and Kindt, 1992). The calculation scheme is according to DIN 25458E. It uses data calculated from NIVERM and SBURN as well as data pre-calculated from ORIGEN2 with a special cross-section library condensed over core spectra. The afterheat distribution is used for the transient calculations in THERMIX/KONVEK and RZKIND.

Fig. 6. Neutron flux density as a function of lethargy in a spectral zone of HTR-MODUL.
Fig. 7. Axial profile of the thermal neutron flux in the middle of the core of the HTR-MODUL (equilibrium cycle).
Fig. 8. Radial profile of the thermal neutron flux density at the position of axial power maximum in the HTR-MODUL (equilibrium cycle).
5. Nuclear data libraries

The nuclear data libraries used in MICROX are based partly of ENDF/B-IV or JEF according to the availability of nuclear data evaluations at time of developing of ZIRKUS. Today, new libraries are available from JEF-2.2, JEFF-3, ENDF/B-VI, JENDL-3.2 and JENDL-3.3. The impact of different nuclear libraries on reactivity was investigated for a HTR cell problem. For the cell the U load of a sphere was varied and the infinite multiplication factor was calculated then for the spherical cell with standard coated particles by the MCNP code taking into account the double heterogeneity by a regular cubical grain lattice. The results are shown in Figs. 2–3, respectively. Fig. 2 shows the infinite multiplication factor calculated for the three evaluations as a function of U load. Fig. 3 shows the difference of $k_{\infty}$-infinite for ENDF/B-VI and JENDL-3.2 compared to JEF-2.2. Whilst JENDL-3.2 compares good with JEF-2.2, ENDF/B-VI (revision 5) tends to strong under-estimation for high U load due to the increased U-235 capture cross-section of revision VI.5. This effect was also observed for under-moderated LWR systems. The problem is already under investigation by international nuclear data evaluation groups. For normal load from 7 to 9 g U/pebble, however the U-235 effect is not too distinct so that all three libraries can be applied for HTR calculations. Since with the newer (JEF-2.2) data libraries good results were achieved e.g. for the PROTEUS experiments with HTR fuel, it is planned to use these data for further HTR calculations with ZIRKUS by updating the MICROX libraries.

6. Applications of ZIRKUS for calculating equilibrium cycle

For the design of HTR cores the burn-up, buildup and decay of isotopes, the flow of pebbles, the neutron flux distribution, the power distribution, the temperature distribution of fuel and moderator of mixed fuel...
elements with different burn-up have to be determined for the core under consideration of various reflector and control conditions.

An example of ZIRKUS calculations will be given for the 200 MWe HTR-MODUL, as designed by INTERATOM/SIEMENS/FRAMATOME (Lohnert, 1992). The main data of the reactor are listed in Tables 2 and 3, respectively. The core was subdivided into eight radial channels with different pebble flow velocities. The total number of burn-up zones was 72, the number of spectral zones was 10. The volume of the zones in every radial channel are proportional to the pebble flow velocity in the channel. The complete calculation model regards the pebble bed core, the top and bottom reflector, the cavity between pebble bed and top reflector and the radial side reflector. Due to the rotational symmetry of the core the geometry is treated in 2D cylindrical coordinates ($R, Z$).

![Maximum fuel temperature as a function of time after depressurisation for different recycle passes for HTR-MODUL (equilibrium cycle)](image)

The upper curve represents 5 passes, the lowest curve 15 passes.

| Burn-up zones (1, ..., 72) are the so called burn-up zones with burn-up dependent average number densities. For simulation of the pebble flow the isotopes contained in an axial zone of every radial channel will be shifted after a finite irradiation interval into the axial zone below. In the first axial zone (top of core) of every radial channel fresh fuel together with |}

<table>
<thead>
<tr>
<th>Table 2</th>
<th>Main data of the HTR-MODUL fuel element</th>
</tr>
</thead>
<tbody>
<tr>
<td>Particle radius</td>
<td>0.25 cm</td>
</tr>
<tr>
<td>Thickness of the 1 (coating/density)</td>
<td>0.005 cm/1.05 g/cm³</td>
</tr>
<tr>
<td>Thickness of the 2 (coating/density)</td>
<td>0.004 cm/1.05 g/cm³</td>
</tr>
<tr>
<td>Thickness of the 3 (coating/density) (SiC)</td>
<td>0.0035 cm/3.18 g/cm³</td>
</tr>
<tr>
<td>Thickness of the 4 (coating/density)</td>
<td>0.004 cm/1.90 g/cm³</td>
</tr>
<tr>
<td>External radius of the matrix zone</td>
<td>2.5 cm</td>
</tr>
<tr>
<td>Fuel element radius</td>
<td>3 cm</td>
</tr>
<tr>
<td>Heavy metal content per fuel element</td>
<td>7 g</td>
</tr>
<tr>
<td>U-235 share in uranium</td>
<td>8.0%</td>
</tr>
</tbody>
</table>
Table 3
Main data for the HTR-MODUL

<table>
<thead>
<tr>
<th>Main data</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power</td>
<td>200 MW</td>
</tr>
<tr>
<td>Average power density</td>
<td>3 MW/m³</td>
</tr>
<tr>
<td>Core diameter</td>
<td>3 m</td>
</tr>
<tr>
<td>Average core height</td>
<td>9.4 m</td>
</tr>
<tr>
<td>Primary pressure</td>
<td>60 bar</td>
</tr>
<tr>
<td>Primary coolant temperature (inlet/outlet)</td>
<td>250/700 °C</td>
</tr>
</tbody>
</table>

Nuclear data for the equilibrium core

| Number of radial enrichment zones | 1 |
| Number of fuel element passes   | 15 |
| Heavy metal loading             | 7 g/FE |
| Number of fuel spheres          | 360,000 |
| Enrichment                      | 8 wt.% |
| Target burn-up                  | 80000 MWd/MgU |

Fuel inventory

| Heavy metal      | 2396 kg |
| Fissile material | 107 kg  |

FE, fuel element

irradiated fuel of 1, 2, ..., N−1 passes through the core will be inserted. N is the total number of passes. If N = 1 a once through then out (OTTO) core can be simulated. Results of calculations for the described model are shown in the Figs. 10–12. For calculations up to the equilibrium state some hundreds of module chains according to Fig. 1 were run to reach equilibrium at 80 MWd/kg HM with 7 g heavy metal load per fuel element and 1000 days residence time in core. As examples of the results the flux density distribution is drawn for group 3 (epithermal) of four groups in Fig. 4. The control rod reactivity was calculated by means of several HBLOCK runs with different control rod positions. The result is shown in Fig. 5 together with a Monte Carlo result as a reference solution. In Fig. 6, the typical neutron spectrum in a spectral zone is shown. In Figs. 7–8, the axial and radial thermal flux distributions in the core center and in the axial flux maximum are shown. For HTRs with extended
power there must be an inner fuel free zone if the maximum temperature of fuel under DLOCA conditions will be limited. The PBMR concept with higher power than 200 MWe regards such an inner fuel free column consisting of moderator elements or of solid graphite. Corresponding fuel cycle calculations can be performed with an extended NIVERM version, which regards channel wise reloading of an arbitrary mix of fuel and moderator elements. An example of a typical thermal flux distribution is shown in Fig. 9 with a typical very high thermal flux density in the center of the fuel free region of the core.

7. Safety related calculations

For the 200 MWe MODUL design the maximum fuel temperature after a depressurisation accident DLOCA was analysed for a different number of fuel element passes through the core. For every number of passes from 5 to 15 the equilibrium state was calculated by ZIRKUS. Then the afterheat distribution calculated by module NZW of ZIRKUS was passed to the transient heat conducting program HEATING-7 of the SCALE system. It was conservatively assumed that there was a prompt depressurisation. The propagation of maximum fuel temperature with time is shown in Fig. 10. From this figure, one can see the advantage of the MODUL concept since the maximum fuel temperature even after depressurisation is limited. This maximum temperature, however, is dependent from the number of fuel element passes. In Fig. 11, the maximum fuel temperature after a DLOCA accident as a function of fuel element passes is shown. The larger the number of passes the lower is the maximum temperature. This fact can be used for optimisation since the reactor nominal power can be increased if the number of passes is increased without increase of maximum fuel temperature compared to a reload strategy with fewer passes. In Fig. 12, the peak factor of power distribution is shown as a function of fuel element passes.

![Fig. 12. Correlation between the peak factor of power distribution at nominal conditions of HTR-MODUL (equilibrium cycle) as a function of fuel element passes.](image-url)
8. Transient neutron physics calculations

The Code RZKIND with a nuclear data base prepared by ZIRKUS can be used for a variety of transients. As an example, the hypothetical withdrawal of all control rods (excess reactivity 1.21%) from MODUL reactor in equilibrium state was calculated. Following accident scenario was postulated:

- the rods are withdrawn at maximum withdrawal speed (1 cm/s),
- the first scram signal (relative neutron flux \( \geq 1.2 \)) of the reactor protection system fails
- the second scram signal (average helium outlet temperature \( \geq 750^\circ \text{C} \)) initiates only the trip of the blower.

Furthermore, it was postulated that, although the blower is shutdown after initiation of the scram, the rods do not drop but continue to withdraw. The power response of the core for this beyond-design-basic accident is given in Fig. 13, the corresponding temperature in Fig. 14. The reactor power increases up to ca. 190\% of the initial value, which is the point at which the reactivity insertion rate due to rod withdrawal is compensated by fuel and moderator temperature.
increase due to the strong negative temperature coefficients. As a consequence of the blower trip initiated by the scram after a 2 min period of excess power the fuel element temperatures are still rising and the reactor power drops again to the level of decay heat within further 5 min. The maximum fuel element temperatures resulting from this power excursion reach values of about 1000 °C.

9. Conclusions

The ZIRKUS program system together with the thermal hydraulics module THERMIX/KONVEK and the transient neutronics code RZKIND is able to perform all kind of design calculations for HTR cores with pebble fuel. The modular concept allows the reliable maintenance of the system and implementation of updates. The programs are verified by re-calculation of experiments and AVR operating data as well as by comparison with other calculations and reference solutions. Most of models used in these programs are proven for the usual design calculations. Extended models and the intensified use of Monte Carlo programs for detailed calculation of transport effects will complete the tool for HTR physics calculations.

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