

6. Evaluation of Post irradiation Heat-up Tests

The highest average central fuel temperature evaluated during the detailed irradiation test evaluation in paragraph 5 was 1 289 °C. During postulated accident events, fuel temperatures are expected to rise to 1 600 °C for design base accidents and up to 1 800 °C for beyond design base accidents. The ^{110m}Ag source term during accident events is only of secondary importance due to the relative small inventory in the core, but the release of ^{111}Ag may be significant. It is therefore important to understand the range of applicability of diffusion coefficients derived in paragraph 5.7 and to ensure that release behaviour during temperature transients do not differ to such an extent as to warrant different diffusion coefficients.

Behaviour of fuel spheres under high-temperature transients, as expected during loss of forced coolant events, are best studied through heat-up tests of irradiated fuel spheres in the famous KÜFA-instrument [49].

6.1 The KÜFA Instrument

The KÜFA-instrument is an ingenious device developed to analyse fission product release from a fuel sphere during post-irradiation heat-up testing. KÜFA is a German acronym for ‘Cold finger apparatus’ which refers to the water-cooled plates inserted into a furnace as illustrated in Figure 18.

A test sphere is inserted into the KÜFA furnace placed in a hot cell and heated to the desired temperature. Noble gas fission products are removed by a sweeping gas (helium at 111 kPa at 30 litres/hour), captured in liquid nitrogen traps outside the hot cell, and measured by a NaI detector. Metallic fission products released from the fuel sphere plate out on the water-cooled condensation plate and can be removed from the furnace while heating continues. Condensation plates are removed from the KÜFA instrument at the desired rate and analysed by gamma spectrometry, and then leached for liquid scintillation analyses to determine beta-emitters quantitatively.

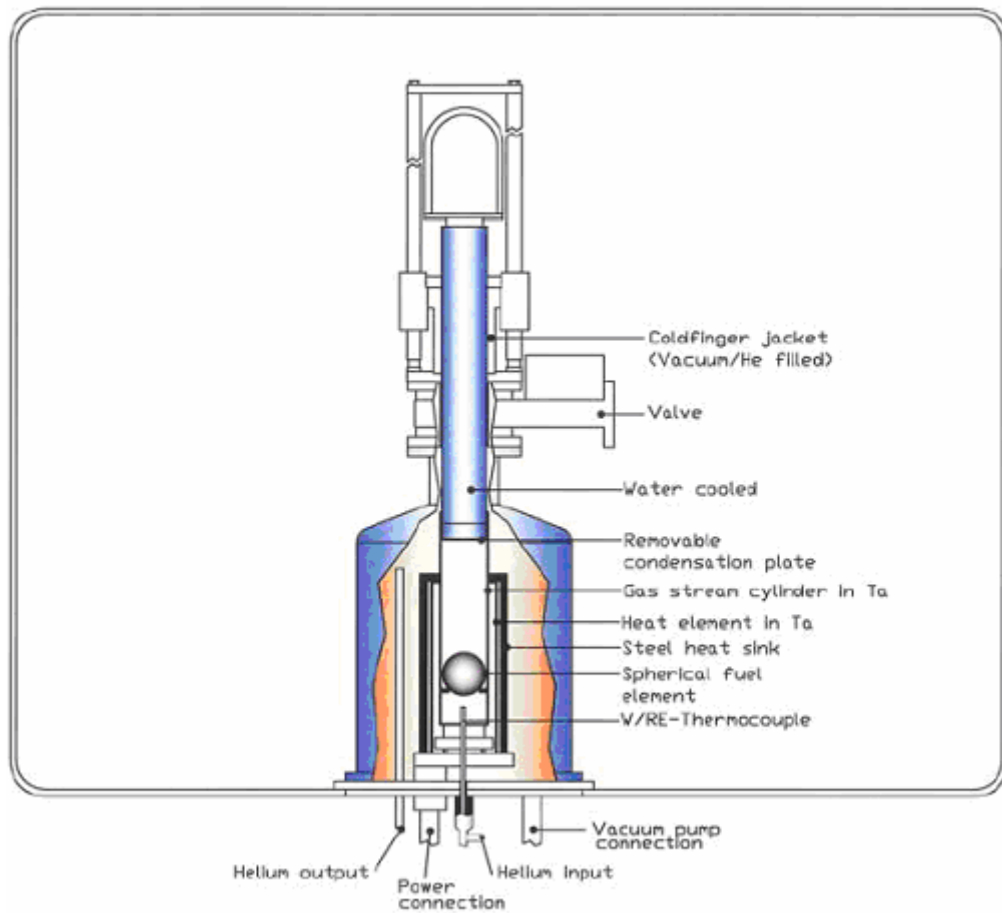


Figure 18: KÜFA-instrument used for Heat-up Testing [49]

6.2 Heat-up Tests

The first investigations into the behaviour of fuel spheres and coated particle compacts during post-irradiation heating have been performed by Schenk [50]. Only heat-up tests of complete spheres of sufficient quality and applicable fuel design are considered for this evaluation. During the German fuel development programme, a total of twenty-nine high-quality fuel spheres underwent post-irradiation heat-up investigations. They were two spheres from each of HFR-K3 (1 and 3) and FRJ2-K13 (2 and 4) respectively, and twenty-five spheres taken from the AVR. Not all heating tests have detailed data available, and only a selection of these tests is evaluated below.

Previously, fuel performance and ^{137}Cs release behaviour of HFR-K3/1 and /3, FRJ2-K13/2 and /4, as well as AVR 76/18 were evaluated [51]. Fuel failure and caesium release were modelled according to the Martin-Goodin-Nabielek model. It was recommended at the time that the study be expanded to include $^{110\text{m}}\text{Ag}$ and ^{85}Kr but this was never done.

Further work was done by Nabielek *et al.* on coated particles at temperatures up to 2 500 °C [52]. Their main conclusion was that at temperatures above 1 900 °C, thermal decomposition occurs very rapidly and all fission products including gases are released. For lower temperatures up to 1 700 °C, the SiC decomposition rates were negligible and coated particles preserve their ability to retain fission products. At temperatures above 1 700 °C, SiC becomes highly permeable to caesium, strontium and silver although gaseous fission products are still retained by the outer PyC layer. Further IMGA studies performed on coated particles from deconsolidated fuel spheres showed that at temperatures of 1 800 °C, very large fractions of caesium and other metallic fission products are released without necessarily observing fission gases release [53].

6.2.1 HFR-K3

The HFR-K3 irradiation test has been described in detail in paragraph 5.1. Two of the test spheres, 1 and 3 were selected for post-irradiation heat-up testing. The first sphere in the HFR-K3 experiment underwent post-irradiation heat-up testing at 1 600 °C for 500 hours. The $^{110\text{m}}\text{Ag}$ fractional release during the experiment is shown in Figure 19. The measured release curve appears very flat, as if a non-diffusion process has occurred. It was exactly this type of behaviour that gave rise to the MVR theory [8]. The IAEA current diffusion coefficient overestimates the release significantly, while the newly-derived diffusion coefficient is much closer. The strange non-diffusion curve is not replicated by either calculated curve, but at least the new diffusion coefficient describes measured release behaviour much better than currently accepted diffusion parameters.

The third test sphere of the HFR-K3 test underwent post-irradiation heat-up testing at 1 800 °C for 100 hours. After heat-up testing the test sphere was deconsolidated and caesium inventories of coated particles were measured with IMGA [53]. It was found that about 50% of coated particles analysed showed release of about 80% of ^{137}Cs inventory while the rest showed release of about 40% of ^{137}Cs inventory. Even though fission gas release suggests a modest fuel failure fraction, the majority of ‘unfailed’ particles released their metallic fission products. By using a ‘silver retention failure rate’ of 50% at the end of irradiation, the release curves of Figure 20 can be drawn.

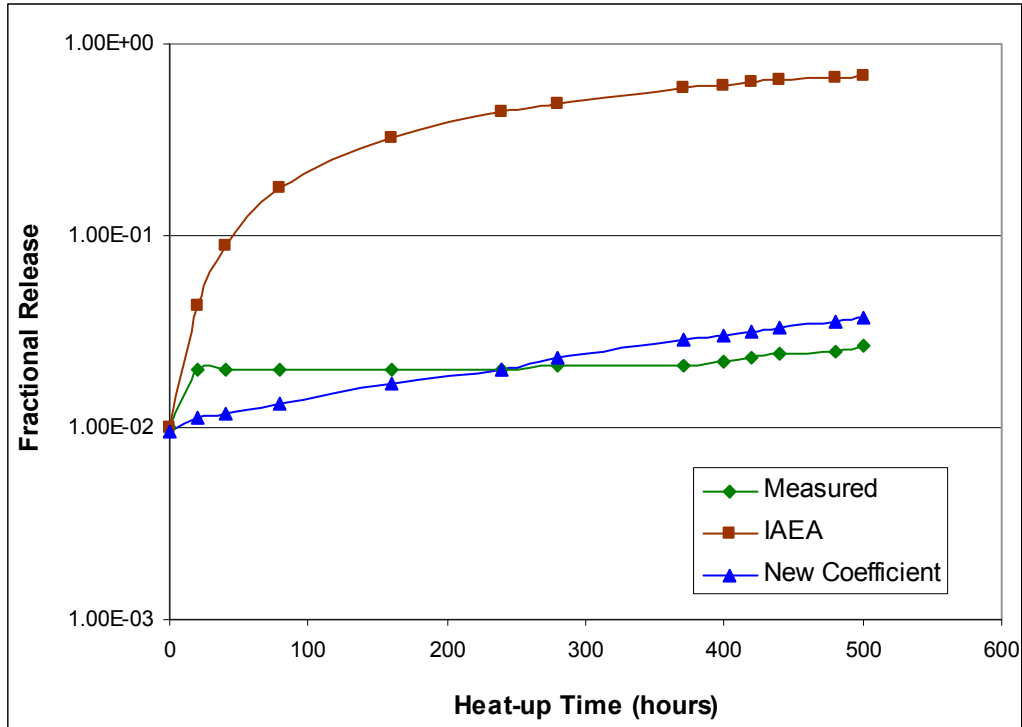


Figure 19: Silver Release during Heat-up of HFR-K3/1

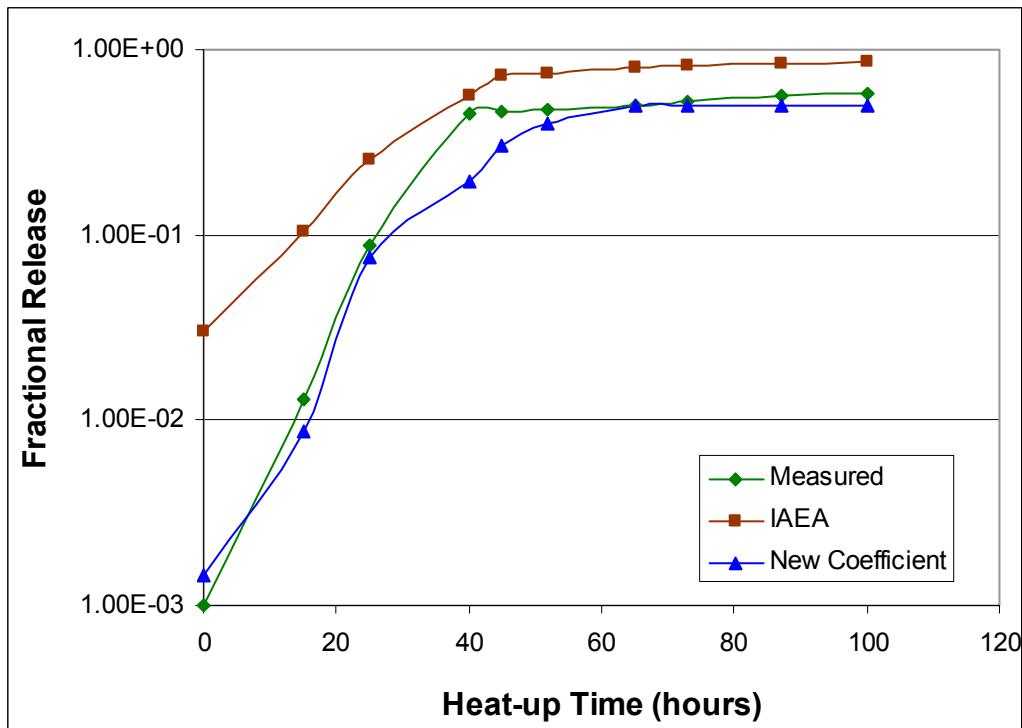


Figure 20: Silver Release during Heat-up of HFR-K3/3

6.2.2 FRJ2-K13

The FRJ2-K13 irradiation test was described in detail in paragraph 5.2. Two of the test spheres, 2 and 4, were selected for post-irradiation heat-up testing. The second sphere in the FRJ2-K13 irradiation test was heated for 138 hours at 1 600 °C. According to ^{85}Kr release during the heat-up test, no coated particles appeared to have failed. In Figure 21, the measured and calculated $^{110\text{m}}\text{Ag}$ fractional releases are plotted. Fractional release was calculated with both the existing IAEA diffusion coefficient and the newly-derived diffusion coefficient. Similar to the HFR-K3/1 1 600 °C heat-up test, the IAEA diffusion coefficient produces a very high fractional release. The new diffusion coefficient also over-predicts fractional release, but at the end of heating produces a final fractional release very close to the measured value.

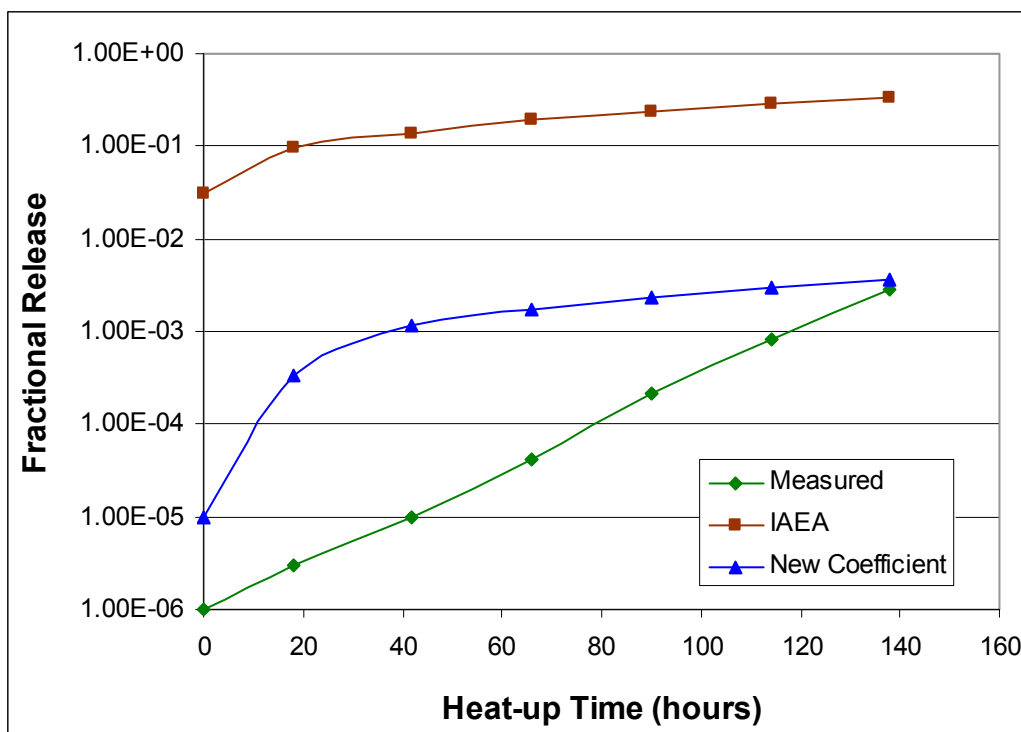


Figure 21: Silver Release during Heat-up of FRJ2-K13/2

The fourth sphere of the FRJ2-K13 test was subjected to 1 600 °C heating for 138 hours and then a further 100 hours at 1 800 °C. The 1 600 °C heating period produces results very similar to the 1 600 °C heating tests performed on the HFR-K3/1 and FRJ2-K13/2 test spheres.

The existing IAEA diffusion coefficient significantly over-predicts release for heating times less than about 140 hours, while the newly-derived diffusion coefficient also over-predicts release, but to a lesser extent. During the 1 800 °C heating period, the ^{85}Kr release increased a hundred-fold and caesium release by three orders of magnitude. Complete silver release was measured after 71 hours of heating at 1 800 °C. This curve could only be reproduced with the new diffusion curve if 100% failure (in terms of silver retention ability) is assumed. Fractional release curves are shown as before in Figure 22.

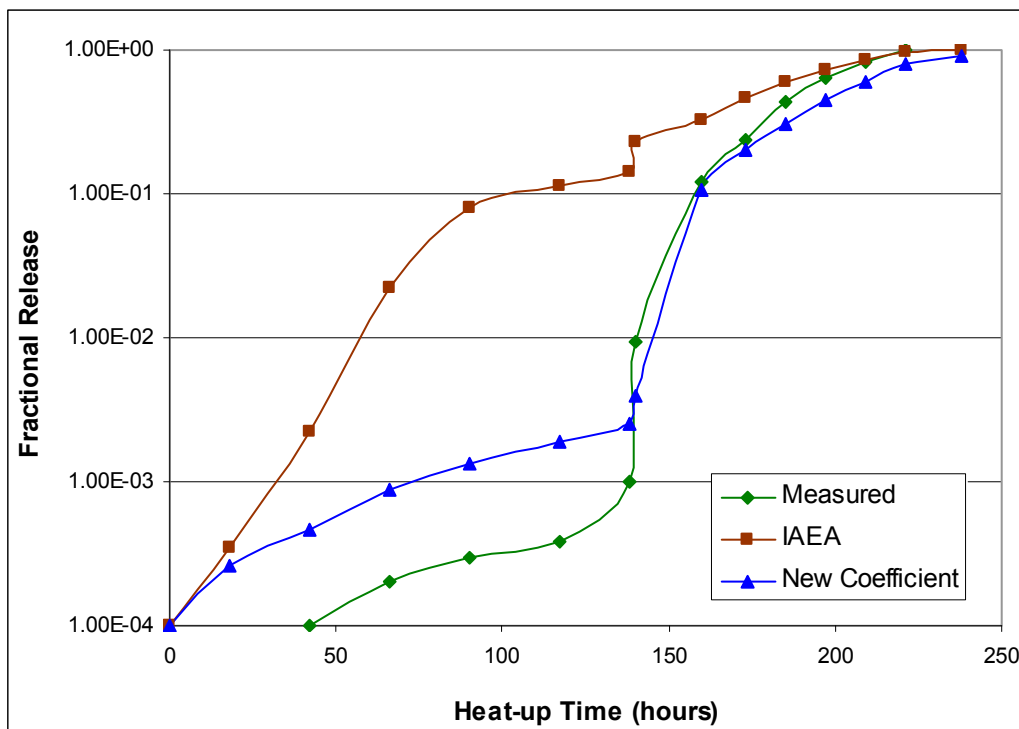


Figure 22: Silver Release during Heat-up of FRJ2-K13/4

6.2.3 AVR 74/11

Fuel sphere AVR 74/11 was irradiated in the AVR for approximately 850 full power days where it achieved a burn-up of 6.2% FIMA and a fast fluence of $1.6 \times 10^{25} \text{ m}^{-2}$ [25]. It contained 16 400 LEU-TRISO particles. After irradiation in the AVR, the fuel sphere was subjected to heat-up testing at 1 700 °C for 180 hours. ^{85}Kr fractional release measurements show no failure during irradiation or subsequent heating for the first 83 hours. After 89 hours of heating the ^{85}Kr fractional release measurement suggests a coated particle failure fraction of up to 9×10^{-4} .

The heat-up test's evaluation is presented in Figure 23. Even for the 1 700 °C case the current IAEA diffusion coefficient over-predicts the silver fractional release by an order of magnitude. The best estimate of the newly-derived diffusion coefficient is about a factor of three too low at the end of the heat-up test that still falls within the previously accepted uncertainty limits for ^{111}Ag release during loss of forced cooling accidents. To test this, the calculation was also performed with the upper limit of the newly-derived diffusion coefficient. The upper limit calculation matches the end of irradiation fractional release, which means that for this specific heat-up test, the newly-derived diffusion coefficient is still feasible.

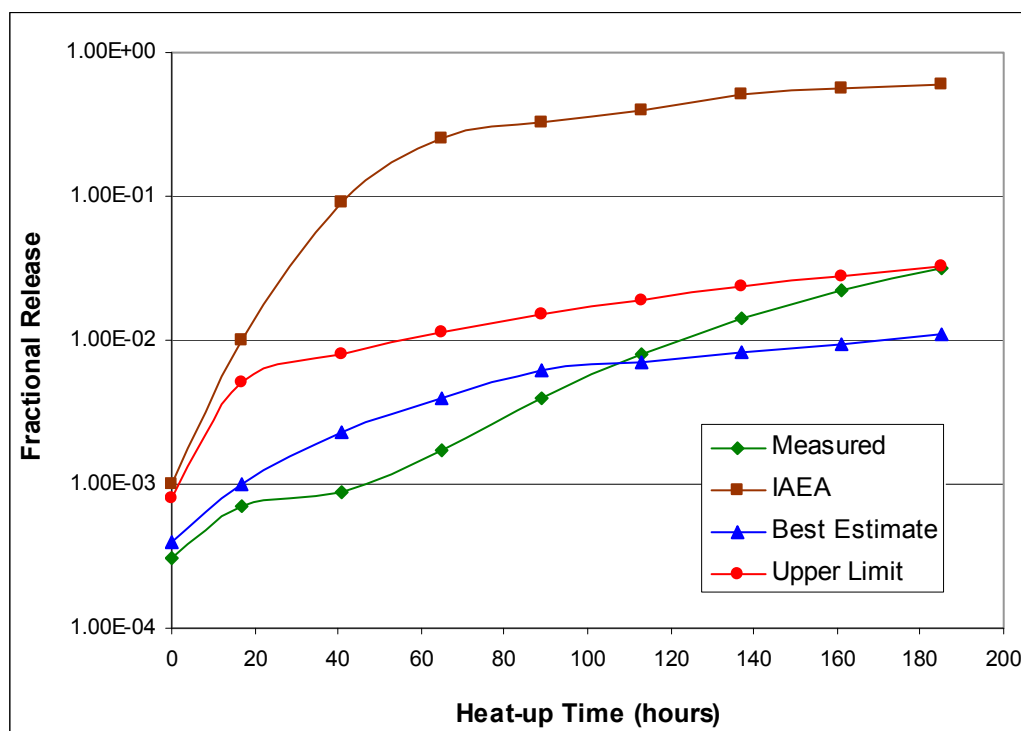


Figure 23: Silver Release during Heat-up of AVR 74/11

6.2.4 AVR 71/22

Fuel sphere AVR 71/22 was irradiated in the AVR for approximately 480 full power days where it achieved a burn-up of 3.5% FIMA and a fast fluence of $0.9 \times 10^{25} \text{ m}^{-2}$ [25]. It contained 16 400 LEU-TRISO particles. After irradiation in the AVR, the fuel sphere was subjected to heat-up testing at 1 600 °C for 500 hours. ^{85}Kr fractional release measurements show no failure during irradiation or subsequent heating.

The heat-up test's evaluation is presented in Figure 24. This is a relatively low burn-up fuel sphere and both diffusion models over-predicted fractional release significantly. This could be due to the fact that the irradiation history of the fuel sphere is not modelled correctly or that silver transport through the coating layers of a coated particle is more dependent on irradiation time and total burn-up than currently accepted. An irradiation history as recommended in the latest literature was used [55], but it could be that this fuel sphere spent most of its time in cooler regions in the core, and this sphere is then more representative of the average AVR core.

The newly-derived diffusion coefficient over-estimates measured fractional release consistently by an order of a magnitude and the IAEA diffusion coefficient by three orders of magnitude. In order to investigate whether this could be a burn-up induced phenomena, higher burn-up spheres heated up to similar temperature regimes have to be evaluated.

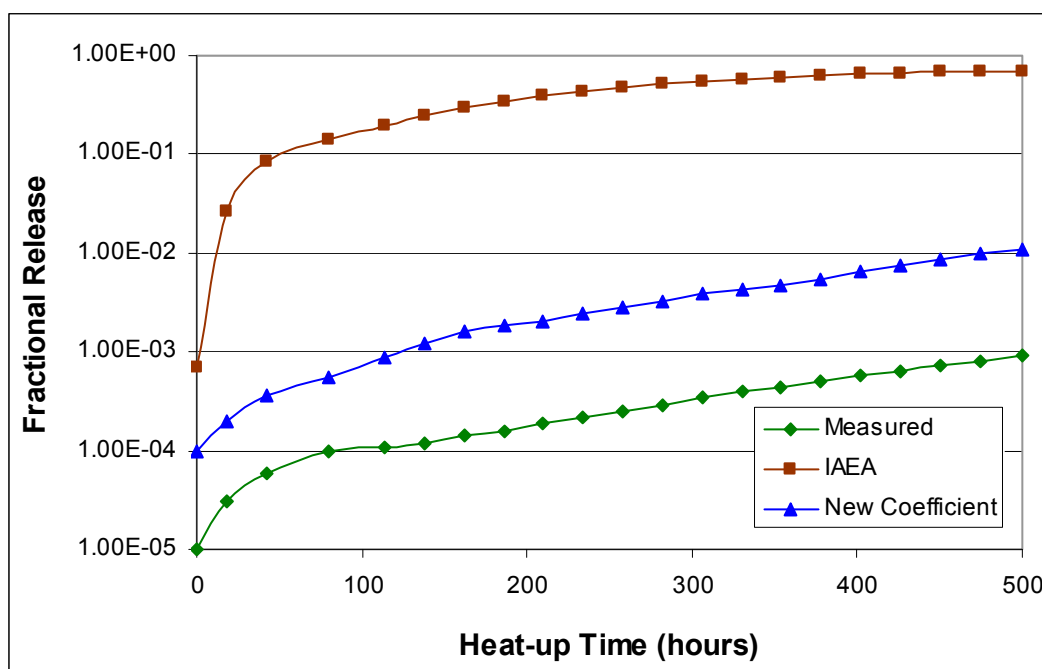


Figure 24: Silver Release during Heat-up of AVR 71/22

6.2.5 AVR 82/9

Fuel sphere AVR 82/9 was irradiated in the AVR for approximately 1 300 full power days where it achieved a burn-up of 8.9% FIMA and a fast fluence of $2.3 \times 10^{25} \text{ m}^{-2}$ [25]. It contained 16 400 LEU-TRISO particles. After irradiation in AVR, the fuel sphere was subjected to heat-up testing at 1 600 °C for 500 hours. ^{85}Kr fractional release measurements show no failure during post-irradiation heating.

The heat-up test's evaluation is presented in Figure 25. This is a relatively high burn-up fuel sphere and the IAEA diffusion coefficient over-predicts the fractional release significantly. The newly-derived diffusion coefficient calculation matches the measured fractional release very well. When comparing this evaluation with AVR 71/22, it appears as if the diffusion coefficient is irradiation-dependent. Since the newly-derived diffusion coefficient is based on high burn-up irradiation test results, it may be that silver release during the early part of a fuel sphere's irradiation life is over-estimated.

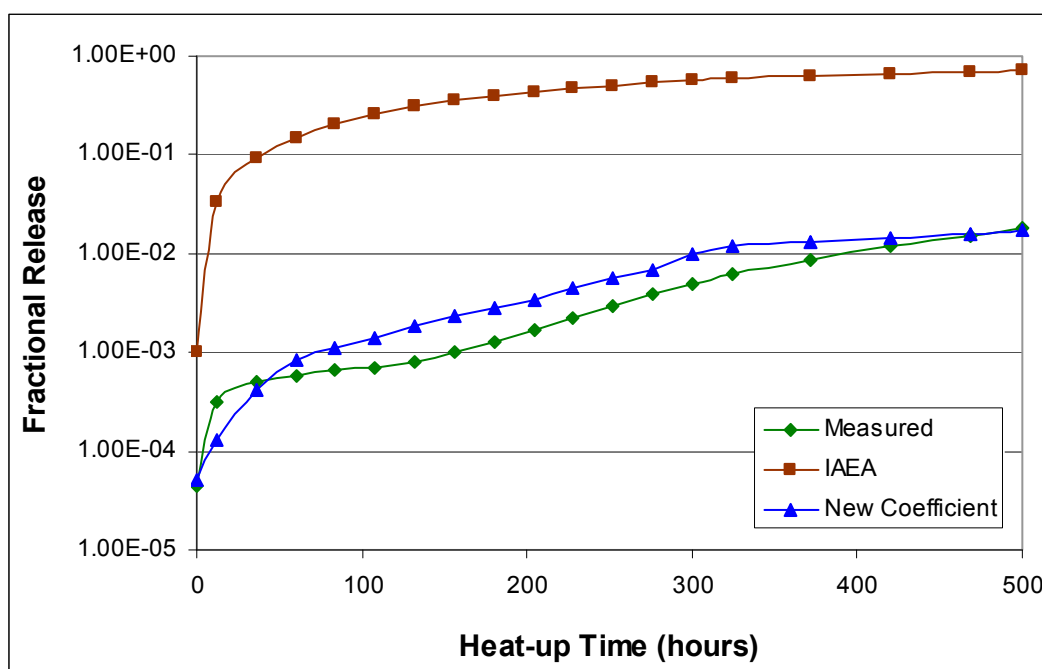


Figure 25: Silver Release during Isothermal Heating of AVR 82/9

6.2.6 AVR 90/5

After evaluating two 1 800 °C heat-up tests (HFR-K3/3, FRJ2-K13/4), which showed that for temperatures above 1 700 °C, metallic fission product release is dominated by SiC layer degradation and not by transport mechanisms through materials, and the only 1 700 °C (AVR-74/11) and several 1 600 °C heat-up tests at various burn-ups (HFR-K3/1, FRJ2-K13/2, AVR-71/22 and AVR-82/9), that showed that the new diffusion coefficient is applicable for temperatures up to 1 600 °C and possibly 1 700 °C, it is time to evaluate a complete loss of forced coolant (LOFC) accident test with the new diffusion coefficient.

Fuel sphere AVR 90/5 was irradiated in the AVR for approximately 1 400 full power days where it achieved a burn-up of 9.2% FIMA and a fast fluence of $2.5 \times 10^{25} \text{ m}^{-2}$ [28]. It contained 16 400 LEU-TRISO particles. After irradiation in the AVR, the fuel sphere was subjected to heat-up testing to simulate a 1 620 °C loss of forced coolant accident. The temperature was raised to 1 620 °C for 30 hours and then gradually cooled to 1 145 °C over a period of 270 hours. ^{85}Kr fractional release measurements show no failure during irradiation or subsequent heating. The heat-up test's evaluation is presented in Figure 26. The newly-derived diffusion coefficient calculation matches measured fractional release very well. This demonstrates that the new diffusion model will be quite sufficient to model possible ^{111}Ag releases during a loss of forced coolant accident.

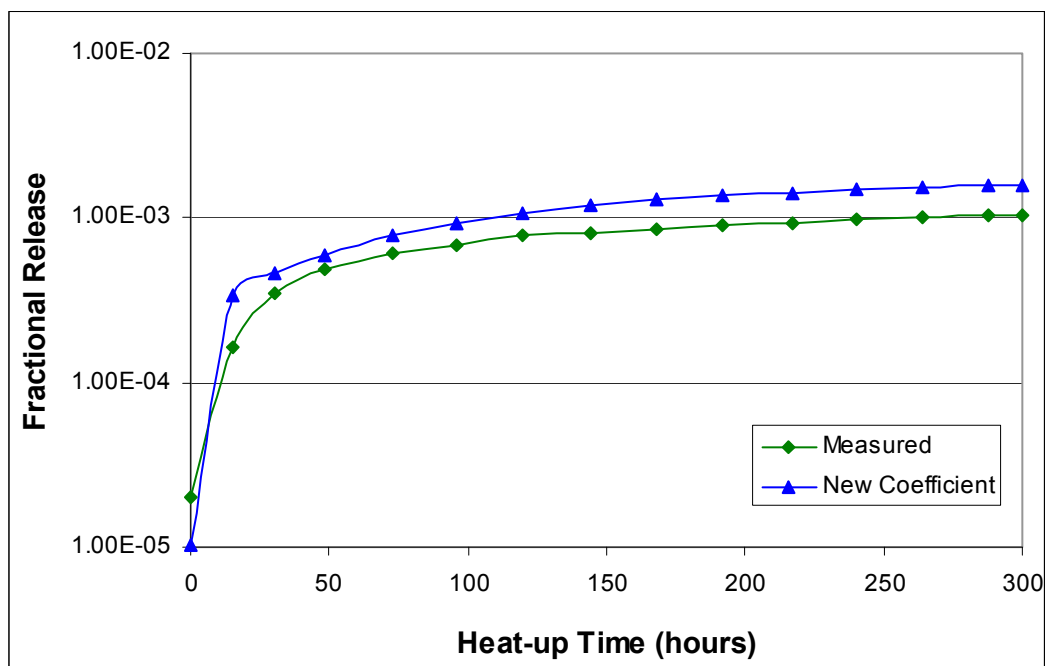


Figure 26: Silver Release LOFC Simulation of AVR 90/5

6.3 Discussion of Heat-up Tests

From the post-irradiation heat-up tests evaluated it is clear that the newly-derived diffusion coefficient may be used up to temperatures of 1 600 °C with confidence. Results for 1 600 °C heating tests tend to be slightly conservative which is acceptable for core analyses. Three high burn-up test spheres as well as spheres from AVR operation of burn-ups between 3.5% and 8.9% FIMA were evaluated. There appear to be some burn-up dependence on the transport of silver through SiC at 1 600 °C.

Silver release during the only 1 700 °C heating test evaluated appears to obey the new diffusion coefficient, however, it is difficult to judge based on only one test. All tests performed above 1 700 °C (1 800 °C) show massive silver release and can only be modelled by assuming that the vast majority of coated particles fail in terms of metallic fission product retention.

This is supported by an evaluation performed on IMGA results from HFR-K3/3 [54]. In this study it was attempted to explain the observed caesium inventories in each coated particle after irradiation and heat-up testing. Very high diffusion coefficients were suggested, which in practice for silver transport means completely permeable coated particle layers. This failure fraction (in terms of metallic fission product retention) cannot be measured by ^{85}Kr release and therefore existing fuel failure curves are not applicable for temperatures above 1 700 °C for metallic fission product release calculations. Unless future tests can prove the contrary, the retention of metallic fission products cannot be modelled by any other means than to accept complete release due to SiC degradation for temperatures above 1 700 °C.

7. Application in PBMR Core Analyses

It is important to understand how the new calculation model affects PBMR core analyses. The new model and parameters derived during this study are compared with current diffusion and alternative MVR models for a sample PBMR core design.

7.1 PBMR Core Model and Analyses

The core design is modelled by reactor analyses to present flow channels consisting of axial layers to form a number of core regions. A typical PBMR core is modelled with several (4 - 10) fuel sphere flow channels, each channel divided into axial layers (10 - 20), to yield 80 to 150 core regions. These core regions are used by core neutronics, thermohydraulic and fission product release codes to model core parameters. Each region is calculated separately to supply thermohydraulic (gas and fuel surface temperatures) and neutronic data (fast and thermal neutron fluxes as well as neutron capture cross sections) to the fission product release analyst.

These data sets are calculated by Very Superior Old Program (VSOP) and Monte Carlo N-particle Transport Code (MCNP) analyses and supplied through an interface document (e.g. [56]) to the fission product release analyst. Fuel passes through the core several times (4 – 16 times) to flatten the axial power profile. Core regions sizes are selected so that a fuel sphere takes an equal amount of time to travel through any core region. A typical PBMR core geometry is presented in Figure 27.

The model used to describe the fuel is explained in paragraph 1.1 and the models used to describe the transport process are explained in paragraphs 2.1 and 2.2 for diffusion and MVR respectively. Diffusion calculations were performed with the software product FIPREX [57] to determine the long-lived fission product release from fuel elements. The software product performs all pre-calculation data manipulation, creates GETTER input files and executes GETTER software. FIPREX then performs post-GETTER calculation data manipulation, delivering fission product release values (inventories, release rates and accumulated releases). The complete FIPREX model is available in the FIPREX theory description [58].

MVR calculations were performed with the VBIST software [8]. All input data (fuel parameters, temperatures, neutronics such as cross section and neutron fluxes, reactor parameters etc., from sources, e.g. [2], [10], [11], [27] and [56]) were entered into the input files. Best estimate calculations are performed with best available values.

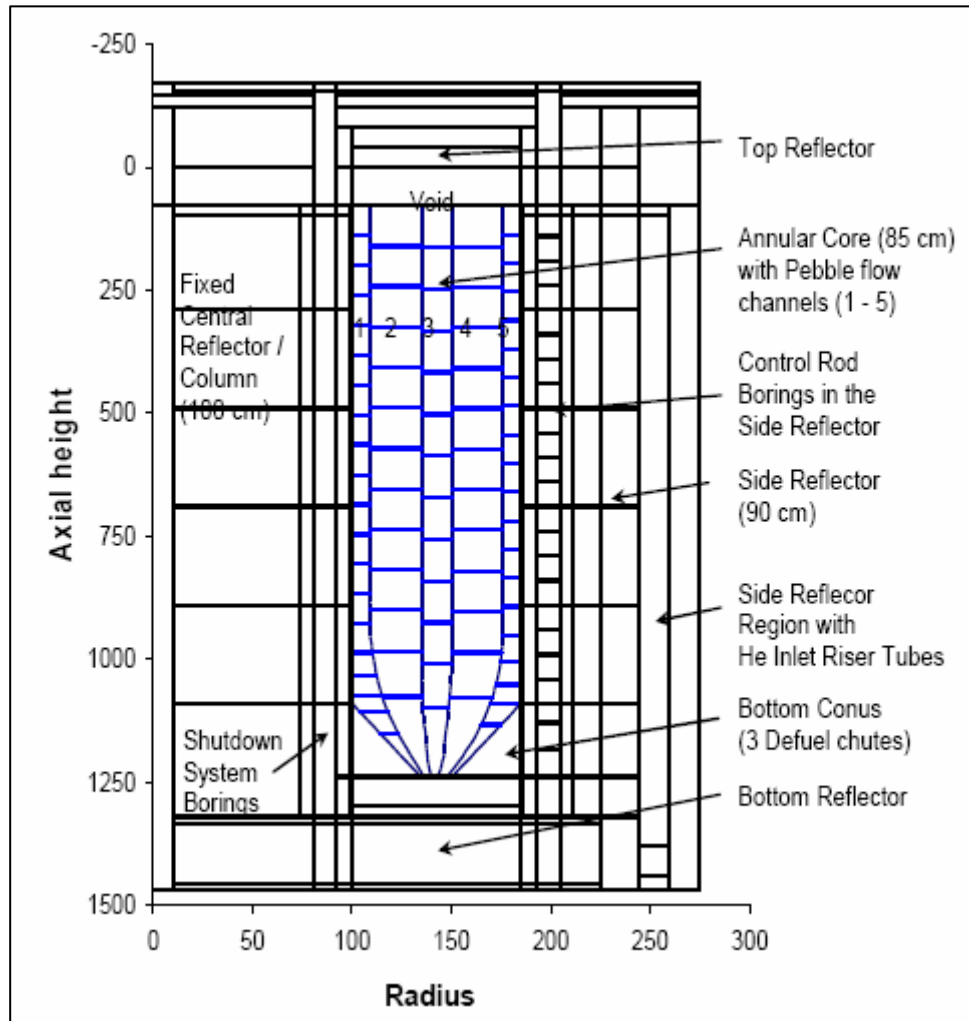


Figure 27: Sample PBMR VSOP Calculation Model Layout

Design values are determined through sensitivity analyses performed on all uncertain parameters. Design calculations presented in paragraph 7.2 are based on a stacked uncertainty analysis where all uncertain parameters are simply set to their design parameters and the calculation is performed. The same thermohydraulic and neutronic data set is used for all cases of each core analysed. GETTER calculates fission product release from one sphere for each time step throughout its residence time in the core. A fuel element spends one time step in each core region. Each fuel element's release history contributes to the total core release, so that the total core release is the average of all the core region releases over all time steps of the irradiation histories of all fuel elements. Furthermore, since the fission products in question are all long-lived, the fission product inventory, and therefore the release from a core region, is dependent on the irradiation history of the fuel elements in that core region.

This implies that GETTER must calculate release histories of all fuel elements in the core for all possible irradiation histories, since irradiation and temperature conditions in each core channel is different from all others. GETTER runs for all 400 000 fuel elements in the PBMR core will take an exorbitant amount of time. Furthermore, calculation of Depressurized Loss of Forced Cooling (DLOFC)-type accidents would be close to impossible since accidents can occur at any time step in a specific fuel element's irradiation history. Calculating such an accident would then require in the order of 35 million GETTER calculations to be performed. It is impossible to perform Monte Carlo-type parametric studies on this large number for all applicable nuclides.

To overcome these problems, the PBMR core's fission product release is calculated by selecting a representative set of fuel elements. This representative set is generated by randomly selecting a large number (~10 000) of individual fuel elements from the core. The average of releases from this selection is assumed to be a good representation of the calculated release from all the fuel elements. Analysing a selection twice the size of the original selection and comparing the average release rates easily verifies this assumption. The size of the selection depends on the accuracy required and the computation time available. The random fuel element set is selected by assigning a random core pass history based on the probability of a fuel element passing through the core in a specific core channel. The probability of a fuel element passing through the core in a specific core channel is dependent on the size of the core channel and the speed at which fuel elements pass through the core channel. Therefore, a specified number of fuel elements are selected with core pass histories randomly selected from the weighted core channel probability.

7.2 Silver Release from a PBMR Core

A sample PBMR core was analysed at 400 MW power and an outlet temperature of 900 °C. The same core design was also analysed at 300 MW and 500 MW power and outlet temperatures of 750 °C and 950 °C respectively. Steady state core release rates (atoms per second) and average fractional releases (for best estimate calculations) for the three cases are presented in Table 36, Table 37 and Table 38 for the different calculation models under consideration. These results are for illustration of the effect of different model assumptions only and should not be used for any other purpose or compared to any other analysis performed elsewhere. The 'Original German' model refers to the legacy model [13] inherited with the original GETTER software and utilizes IAEA Tecdoc 978 [2] parameters and values. This analysis of a 400 MW PBMR core has been reported in [59].

The ‘MVR’ results are taken from the original Olivier calculation described in [8]. The ‘First Estimate’ model is based on the material data evaluation of Chapter 3 and the irradiation test evaluation of paragraph 4.3.1. The ‘Detailed Evaluation’ analysis is based on the final detailed evaluation of all applicable German fuel sphere irradiation tests described in paragraph 5. MVR analyses have not been performed for the 500 MW case, but are expected to predict considerably lower release rates than any of the diffusion models. Since the 500 MW study was only a conceptual study [60], only best estimate analyses were performed.

Table 36: Comparison of Calculated ^{110m}Ag Releases from a 400 MW PBMR Core

	Fractional Release	Best Estimate	Design Limit
Original German	8.90×10^{-4}	8.12×10^{11}	1.22×10^{13}
MVR	2.60×10^{-3}	3.24×10^{12}	1.50×10^{13}
First Estimate	9.73×10^{-4}	9.18×10^{11}	8.95×10^{12}
Detailed Evaluation	6.03×10^{-4}	4.30×10^{11}	3.90×10^{12}

Table 37: Comparison of Calculated ^{110m}Ag Releases from a 300 MW PBMR Core

	Fractional Release	Best Estimate	Design Limit
Original German	6.86×10^{-5}	1.30×10^{10}	2.84×10^{10}
MVR	3.02×10^{-4}	8.10×10^{10}	4.80×10^{11}
First Estimate	1.67×10^{-4}	3.45×10^{10}	2.67×10^{11}
Detailed Evaluation	7.24×10^{-5}	1.35×10^{10}	7.23×10^{10}

Table 38: Comparison of Calculated ^{110m}Ag Releases from a 500 MW PBMR Core

Atoms/s	Fractional Release	Best Estimate
Original German	6.46×10^{-3}	1.90×10^{13}
First Estimate	2.07×10^{-3}	2.59×10^{12}
Detailed Evaluation	2.29×10^{-3}	3.17×10^{12}

7.3 Effect on PBMR Core Analyses Discussion

As can be seen from the tables in paragraph 7.2, differences between the considered calculation models are relatively small. For the 400 MW design, the biggest difference between core release rates is only a factor of seven for best estimate and a factor of four for design limit analyses (between MVR and Detailed evaluation in both cases). There is almost no difference between the diffusion models. Considering that uncertainty factors of up to 20 have been suggested and used in the past [21], the differences are acceptable. The main reason for this is that the fuel temperatures for this specific core design and an outlet gas temperature of 900 °C lie primarily between 900 °C and 1 100 °C. As can be seen in Figure 28 the three considered diffusion coefficients converge in this temperature regime and the differences in silver transport rated through SiC is small.

For the 300 MW case the differences are similarly very small, but in this case it is because the fuel temperatures are simply too low. Fuel temperatures of a 300 MW power and 750 °C coolant outlet design lie between 700 °C and 900 °C and silver transport through the SiC layers is too slow. Releases at these temperatures are generally from matrix material contamination and failed particles. The controlling transport mechanism in this case is diffusion through the matrix material.

The 500 MW core, on the other hand, has significant differences between the different calculation models. The original German parameters predict a higher release rate compared to the two diffusion coefficients derived in this study. Fuel temperatures are high enough that silver transport through SiC dominates the source term.

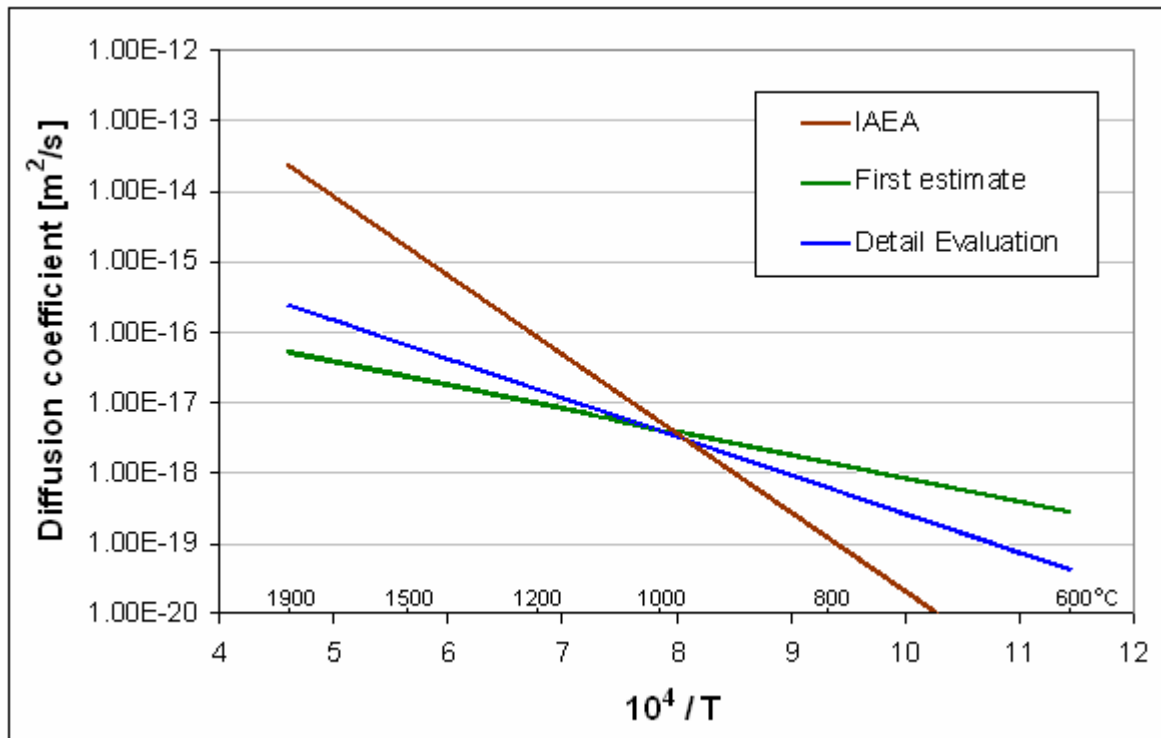


Figure 28: Comparison of the Three Considered Diffusion Coefficients

8. Conclusions

Release of silver fission and activation products from fuel contamination and failed particles are only of importance at low temperatures when the radiological impact of ^{110m}Ag on maintenance is small ($< 700\text{ }^\circ\text{C}$). At higher temperatures, intact coated particles release dominates, with the SiC layer being the main retarding layer. Modelling of silver transport through SiC is therefore most important in the analyses of silver release from a planned high-temperature reactor. Material tests and irradiation experiments on coated particles have not been able to clearly identify the exact transport mechanism. Historically the approach was to use ‘effective diffusion coefficients’ in a diffusion model, but this appears to lead to overestimates at very high temperatures ($> 1\ 100\text{ }^\circ\text{C}$) and possible underestimates in the critical $800\text{ }^\circ\text{C} - 1\ 100\text{ }^\circ\text{C}$ region.

Other mechanisms have been proposed, the most promising being Molecular Vapour Transport Release (MVR). From a modelling perspective, the best way forward appeared to ignore the exact mechanism and focused on deriving an ‘effective transport model’ based on all irradiation test (integral effects) and material experiment (separate effect) data. Even though transport through the SiC layer has been identified as the dominating phenomenon, any other transport process has to be quantified as best as possible to ensure that analyses of irradiation tests would provide the best possible SiC transport model and parameters.

Available material tests that would investigate each phenomenon separately were evaluated and current models and parameters used were confirmed or updated. Simplified sorption isotherms for silver from graphitic matrix material was derived for the first time from the limited material tests by using a caesium benchmark for the same test conditions. Matrix material diffusion coefficients were confirmed for best estimate analyses and design limit coefficients have been suggested based on diffusion coefficients for original and irradiated matrix material. Diffusion coefficients for fuel kernel and PyC layers were confirmed. Due to rapid transport through these materials, the recoil effects from UO_2 fuel and contaminations in PyC layers and matrix materials are negligible and are only modelled for the sake of completeness.

Material test data for silver transport in SiC show erratic results which lead to both high and extremely low diffusion coefficients or new contradicting mechanisms. An accurate PBMR model to analyse silver release from a high-temperature core that could be defended using all available test results was required. Two calculation models were identified and first estimate evaluations of the applicable German irradiation tests were performed for both MVR (12

tests) and diffusion (31 tests) models. It was shown that a carefully selected set of diffusion constants can in principle simulate irradiation tests results. Diffusion theory was shown to remain a viable option to model transport of silver.

Based on first estimate results and all material test evaluations, detailed analyses of all applicable irradiation tests were performed. Diffusion coefficients for silver in SiC were derived for both best estimate predictions and design limit analyses. It was found that the newly derived diffusion coefficients for each test were below the current IAEA recommended diffusion coefficient line, which in turn is based on experimental work performed on fuel manufactured before 1978. Detailed evaluations performed during this study are primarily based on fuel manufactured after 1980. The differences between the existing IAEA diffusion coefficient and the newly derived diffusion coefficients could be due to improvements in SiC manufacture that enhanced silver retention.

Heat-up tests of fuel elements irradiated during fuel tests and fuel elements from AVR operation were evaluated in order to justify the range of temperatures where the newly derived diffusion coefficients are applicable. It was found that the new diffusion coefficients are valid for temperatures from 800 °C up to 1 600 °C and may be used for accident analyses up to 1 700 °C. Above 1 700 °C it appears that major particle failure (with respect to silver retention) occurs and the new models should not be used at temperatures above 1 700 °C.

Existing and newly derived diffusion coefficients as well as the MVR model were applied to a sample PBMR core model at 300 MW, 400 MW and 500 MW power with reactor outlet coolant gas temperatures of 750 °C, 900 °C and 950 °C respectively. It was found that only under the high temperature and power conditions are there significant differences between the release rates of ^{110m}Ag calculated by the different models.

9. Recommendations

It is recommended that the transport parameters suggested in Chapter 3 and the diffusion model and diffusion coefficients derived in Chapter 5 be used for future core analyses until new data from fuel qualification programs become available. The applicable temperature range for these new diffusion coefficients lie between 800 °C and 1 700 °C. For temperatures above 1 700 °C coated particle failure due to SiC degradation become the dominant source term contributor and diffusion coefficients cannot be effectively evaluated. For temperatures below 800 °C, transport through PyC layers is slow enough that it does not matter which diffusion coefficient is used for SiC.

Differences between MVR and the new diffusion-based calculation model are relatively small and will be reduced even further if sorption is included in the MVR model. The MVR model must not be discarded though. In future it may be shown through fuel characterization efforts, that silver is transported by a physical process that could be modelled by MVR. The MVR model can then be further developed and used in reactor analyses. The main problem with the MVR model is absence of any reported nanotubes in polycrystalline SiC making the relatively good correlation with the experimental data fortuitous. In the meantime, the diffusion model and parameters derived in this study should be used.

Important lessons regarding the design, execution and evaluation of fuel irradiation tests were learned during this study:

- a. The irradiation conditions (temperature, neutron fluxes and burn-up) are important considerations to ensure that realistic core conditions are achieved, and also to produce a range of conditions to evaluate fuel performance through the whole envelope of expected core conditions. It is therefore recommended that fuel spheres are irradiated at temperatures not exceeding 1 200 °C. However, it is also important to evaluate fuel performance at lower temperatures. Therefore irradiation temperatures in the range of 900 -1 100 °C should be considered for some later fuel irradiation tests.
- b. The irradiation rig design and placement of irradiation monitors and thermocouples are very important. In all irradiation tests, thermocouple failure is a major problem that influences the accuracy of measurements and even ended some tests prematurely. Only the highest quality thermocouples should be used and the test should be designed to ensure that thermocouple failure remains a minimum.

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- c. At the end of the German fuel programme, data were not recorded rigorously and a lot of important information from irradiation tests was lost. Similarly, older progress reports and data sets were not stored properly and a lot of information disappeared. For PBMR's own irradiation programme it is vital that all data from irradiation tests are recorded at the end of each irradiation period and evaluated. In this way, discrepancies can be identified and resolved while the current personnel involved are still available.
 - d. Especially for lower-temperature irradiation tests (mean fuel temperature $< 1\ 100\ ^\circ\text{C}$) the natural uranium, thorium and silver contamination in the irradiation rig materials dominate the measured $^{110\text{m}}\text{Ag}$ activity on rig materials. Determination of transport parameters and evaluation of fuel performance then becomes very complicated and uncertain. Only in one irradiation facility was the natural silver contamination measured for only one of the rig materials. It is therefore critical that all the rig materials are assayed for their uranium, thorium and silver content, as well as any other contamination that might influence the irradiation test results.
 - e. The natural silver contamination of fuel matrix material is currently not specified. It appears that the natural silver contamination could be the dominant source term for the PBMR core under lower power and temperature conditions. For pre-1980 fuel, German references suggest 2.7 ng/g as a maximum silver contamination level for A3-27 matrix material. For post-1980 fuel a lower value 0.8 ng/g is suggested. For future best estimate analyses it is recommended to use the higher value of 2.7 ng/g. As design limit the actual detection limit results measured on A3-3 matrix material should be used up until such time the actual silver contamination level on PBMR materials can be measured.

Mean fuel irradiation temperatures should not exceed $1\ 200\ ^\circ\text{C}$ as subsequent heat-up tests to $1\ 600\ ^\circ\text{C}$ may release significant amounts of silver. Should mean fuel temperatures remain below $1\ 200\ ^\circ\text{C}$, $1\ 600\ ^\circ\text{C}$ accident events appear feasible. However, accident temperatures above $1\ 700\ ^\circ\text{C}$ are not acceptable. All heat-up tests above $1\ 700\ ^\circ\text{C}$ show massive SiC degradation failure regarding silver retention. Only one $1\ 700\ ^\circ\text{C}$ heat-up test was evaluated in this study, so it cannot be commented if $1\ 700\ ^\circ\text{C}$ is an acceptable accident temperature at this stage and should be investigated during PBMR's own fuel qualification programme.