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SELECTION OF BENCHMARK CASES FOR MECHANICAL FAILURE PREDICTION

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Selection of Benchmark Cases for Mechanical Failure Prediction

Contents:

Abstract:

It is recommended to use irradiation tests HFR-K3/1, HFR-K3/3, FRJ2-K15/2 and HFR-EU1/3 for comparative mechanical failure predictions. Models and materials data should be based on the EU data book “Selection of properties and models for the HTR coated particle” from December 2003[1], as far as possible and meaningful. Fuel and irradiation data are given here.

HFR-K3 fuel has been manufactured, irradiated and heated. There was no manufacturing defect, no in-pile failure and no particle failure during 1600°C heating (HFR-K3/1).

FRJ2-K15 fuel has been manufactured and irradiated. There was no manufacturing defect and no in-pile failure despite high burnups and high irradiation temperatures. Core heatup simulation heating is planned at EU-JRC-ITU in Karlsruhe.

HFR-EU1 fuel with spheres from AVR 21/2 (type GLE 4/2) is planned for irradiation in Petten to very high burnup.

[1] Abram et al., Selection of properties and models for the HTR coated particle, EU-HTR-F-WP3 Deliverable No 12, A collaboration of work performed by BNFL, CEA, FRAMATOME-ANP and FZJ, Document Number: HTR-F-0312-D-3.2.1, December 2003

I. Selection of benchmark cases

To test computational tools for coated particle mechanical failure predictions, the following tests have been selected for benchmarking:

		Irr. time	Burnup	Fluence	Temperature		Comment
Test	sphere	days	%FIMA	$10^{25}m^{-2}$, E>16 fJ	Surface (°C)	Centre (°C)	
HFR-K3	1	359	7.5	4.0	1020	1200	Irr+PIE
HFR-K3	3	359	10.6	5.9	700	920	Irr+PIE
FRJ2-K15	2	682	15.3	0.2	980	1150	Irr
HFR-EU1	3	600	20	6.0	900	1100	planned

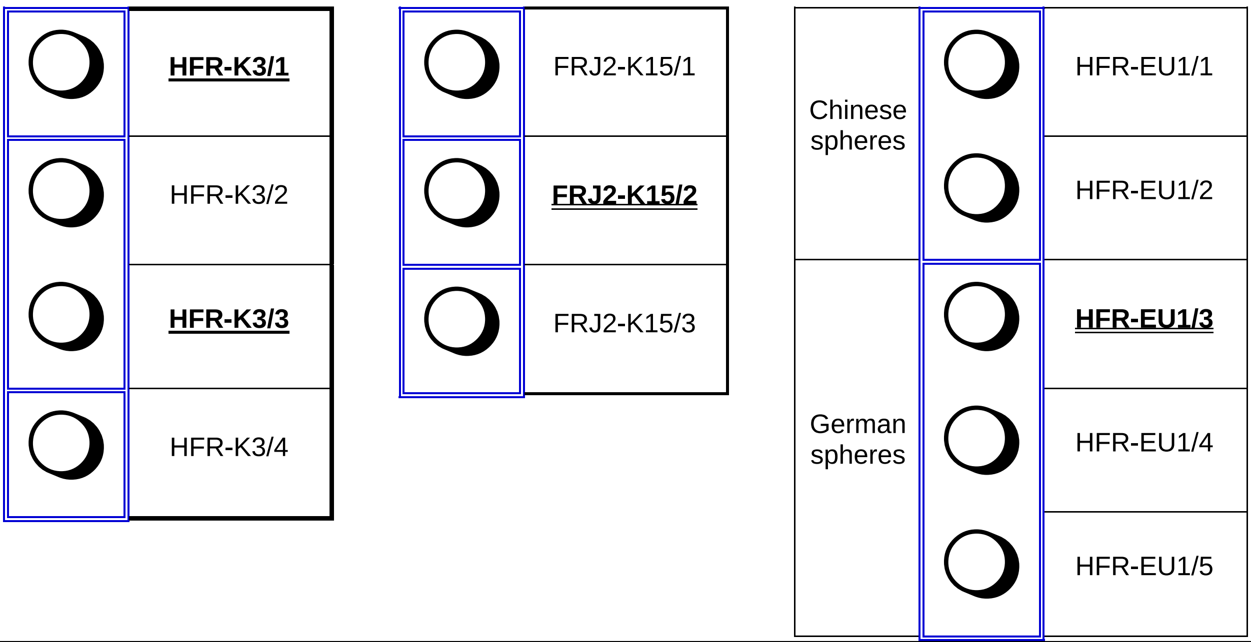
Irr= irradiation completed,
PIE= post irradiation examinations including heating tests completed.

HFR-K3 spherical fuel elements contained UO_2 LTI Triso particles from the LEU Phase 1 coating batch EUO 2308, a pre-runner of the large scale AVR 19 GLE-3 production with 24,600 spheres.

FRJ2-K15 spherical fuel elements were taken from large scale AVR 21 GLE-4 production with 20,500 spheres (later designated AVR 21/1 and GLE-4/1).

HFR-EU1 spherical fuel elements are taken from large scale AVR 21/2 GLE-4/2 production with 14,000 spheres and are planned to be irradiated in the High Flux Reactor Petten to above 20% FIMA.

Irradiation positions of the benchmark cases are:



Sphere 1 with 7.5% FIMA and sphere 3 with 10.6% FIMA have been selected. No defect or failure in either fuel element.	Sphere 2 with 15.3 % FIMA had neither defects nor failures. All three elements are planned for future heating tests.	Sphere HFR-EU1/3 has been selected for benchmarking. Irradiation is planned in HFR Petten in the EU programmes.

Particle data are:

Test	HFR-K3	FRJ2-K15	HFR-EU1
Coated particle batch	EUO 2308	HT 354-383	HT 385-393 HT 395-404 HT 406-423
Kernel composition	LEU UO_2	LEU UO_2	LEU UO_2
Enrichment [U-235 wt.%]	9.82	16.76	16.76/ 16.67
Kernel diameter [μm]	497 ± 14.1	501 ± 10.8	502.2 ± 10.6
Buffer layer thickness [μm]	94 ±10.3	92 ±14.3	94.3 ±13.0
iPyC layer thickness [μm]	41 ± 4.0	38 ±3.4	40.6 ±3.7
SiC layer thickness [μm]	36 ± 1.7	33 ± 1.9	35.9 ± 2.2
oPyC layer thickness [μm]	40 ±2.2	41 ± 3.8	39.8 ± 3.3
Kernel density [g/cm^3]	10.81	10.85	10.86
Buffer density [g/cm^3]	1.00	1.013	1.012
iPyC density [g/cm^3]	~ 1.9	~ 1.9	1.87
SiC density [g/cm^3]	3.20	3.20	3.20
oPyC density [g/cm^3]	1.88	1.88	1.87
iPyC Anisotropy BAF	1.053	1.029	1.02
oPyC Anisotropy BAF	1.019	1.020	1.02

All four cases suggested for benchmark calculations comprise complete spherical fuel elements with somewhere between 9,000 and 16,000 particles. The experimentally obtained results will therefore be statistically significant.

II. Background on German high quality TRISO fuel testing

The German LEU-TRISO fuel design was the culmination of German pebble fuel development that started in 1972 and continued to 1996. The development of LEU UO_2 TRISO fuel started in 1981 and continued until 1988 when fuel spheres for the HTR-Modul Proof Tests were manufactured. The different LEU-TRISO fuel types manufactured and tested during this period are shown below.

Characteristic	**Pre – 1985 Production**			**Post – 1985 Production**	
Year of Manufacture	1981	1981	1983	1985	1988
Designation	GLE 3	LEU Phase 1	GLE 4	GLE 4/2	Proof Test
Matrix Material	A3-27	A3-27	A3-27	A3-3	A3-3
Irradiation Test Designation	AVR 19	HFR-K3, FRJ2-K13 HFR-P4 SL-P1 FRJ2-P27	AVR 21-1 FRJ2-K15	AVR 21-2, HFR-EU1, HFR-EU1bis	HFR-K5, HFR-K6, [HFR-EU2 in GA compacts]
Number of fuel spheres manufactured	24,600	100	20,500	14,000	200

The symbols used in the Irradiation Test Designation row have the following meanings:

1. The first three letters describe the reactor in which the test was done:
   AVR = Arbeitsgemeinschaft Versuchsreaktor in Juelich
   HFR = High Flux Reactor in Petten
   FRJ2 = DIDO reactor in Juelich
   SL = Siloe reactor in Grenoble
2. The next group of symbols describe the irradiation sample type and test number. In the case of AVR irradiations the reload number is used i.e. AVR 19 means that the fuel spheres made up the 19th partial reload of the reactor. In other tests the letter K designates a full sized fuel sphere, the letter P designates coated particles in any other form i.e. small spheres, compacts or coupons, and the number is the test number. Thus FRJ2-P27 means irradiation test number 27 performed on compacted coated particles in the DIDO reactor in Juelich.
3. Doubly underlined tests are earmarked for benchmark calculations, whereby HFR-K3 and FRJ2-K15 are tests that have completed irradiation and PIE, while HFR-EU1 is a test in preparation.

Fuel spheres intended for AVR irradiation testing were manufactured in large numbers with the purpose of bulk testing in a reactor environment. Small fuel sphere lots were specially manufactured for the German Phase 1 irradiation test programme and for the Proof Test for the HTR-Modul.

Manufacturing details for the different LEU-TRISO fuel types below.

Characteristic	Pre – 1985 Production			Post – 1985 Production	
Designation	GLE 3	LEU Phase 1	GLE 4/1	GLE 4-2 (remaining)	Proof Test
Kernel Diameter (μm) Kernel Density ($g.cm^{-3}$)	500 10.80	497 10.81	501 10.85	502.2 10.86	508 10.72
Coating Thickness (μm) Buffer Layer Inner PyC Layer SiC Layer Outer PyC layer	 93 38 35 40	 94 41 36 40	 92 38 33 41	 92.3 40.6 35.9 39.6	 102 39 36 38
Coating Density ($g.cm^{-3}$) Buffer Layer Inner PyC Layer SiC Layer Outer PyC Layer	 1.01 1.86 3.19 1.89	 1.00 ~1.9 3.20 1.88	 1.013 ~ 1.9 3.20 1.88	 1.012 1.87 3.2 1.87	 1.02 1.92 3.20 1.92
Loading Heavy Metal (g/FE) ^{235}U (g/FE) Enrichment (% ^{235}U) no coated particles/FE	 10 1 9.82 16 400	 10 1 9.82 16 400	 6 1 16.76 9 560	 6 1 16.76 9 560	 9.4 1 10.6 14 580
Free Uranium Fraction ($x 10^{-6}$)	50.7	35	43.2	7.8	13.5
Number of kernel sets	5	1	2	5	1
Number of Coating Batches	65	1	54	29	8
Coating Batch Size	5 kg	5 kg	3 kg	3 kg	5 kg
Coated Particle Batch Designation	HT232-245	EUO 2308	HT354-383	HT385-393 HT395-404 HT406-423	EUO 2358-2365
Number of spherical fuel element lots	14	-	11	8	-

Note: In German terminology a set is a combination of various production batches; e.g. all 29 coating batches from the GLE 4-2 production were combined into coating sets BP-S3, BP-S4, and BP-S5 for which extensive quality control measurements were performed on a per-set basis.

Although a number of different LEU-TRISO fuel types were manufactured and tested in Germany the following characteristics of the fuel design remained unchanged:

- A graphite matrix, which is a mixture of natural graphite and electro-graphite (synthetic graphite) in the ratio 4:1 with a resinous binder.
- Spheres that are cold pressed under quasi-isostatic conditions.

- Spheres consisting of an inner fuel-containing region of 50 mm diameter with coated fuel particles evenly distributed in the graphite matrix surrounded by a fuel free region of thickness 5 mm of the same graphite matrix material as the fuel region. These are nominal figures. To ascertain that no single particle can be found in the fuel free zone, the thickness is 6.2 mm on average.
- Spheres heat-treated at a temperature of 1950°C without complete graphitisation of the binder coke.
- Coated particle design. Although the enrichment and coated particle loading were changed the coated particle design was never changed.

Two different graphite matrix types, A3-3 and A3-27, were used for the manufacture of LEU-TRISO fuel spheres. In the German fuel development programme preceding the LEU-TRISO programme, type A3-3 graphite matrix was used extensively for fuel sphere manufacture. However, it was replaced by type A3-27 during the initial stage of the LEU-TRISO programme. The programme later reverted back to type A3-3 due to environmental health concerns about the manufacturing process of type A3-27. The results summarised in this report cover fuel spheres manufactured from both graphite types For type A3-27 production, two chemically pure components phenol and hexamethylenetetramine were mixed with natural flake graphite and graphitised coke powder at 130°C and then ground to form a resinated powder, which was used for fuel sphere pressing. Further hardening of the binder occurred in the first stages of the fuel sphere heat treatment. This process resulted in a graphite matrix whose properties had narrower bandwidths than the same properties for type A3-3. For type A3-3 production a pre-fabricated phenolic resin was dissolved in methanol to form a binder that was then added to a mixture of natural flake graphite and graphitised coke powder, kneaded, dried, ground and used for sphere pressing.

Composition of Graphite Matrix Types after Final Heat Treatment

Component	**A3-3**	**A3-27**
Natural graphite	72%	71.2%
Graphitised coke	18%	17.8%
Non graphitised binder coke	10%	11%

German experience with LEU-TRISO fuel types can be classified into three categories. The first part of the German development programme consisted of the so-called Phase 1 irradiation test series during which fuel spheres, small spheres, compacts, and coupons, all containing coated particles from the same EUO 2308 coated particle batch, were irradiated under a variety of conditions. The intention with these irradiation tests was to provide an operation envelope for a number of different reactor types designed around LEU-TRISO fuel:

- Power production via steam raising with 700-750°C helium exit temperature,
- Power production via direct cycle gas turbines with 850-900°C, and
- Process heat generation at 950-1000°C.

Phase 1 tests were mostly conducted under isothermal conditions i.e. at fixed and constant temperatures and demonstrated the suitability of the UO_2 Triso particle and the spherical fuel element for all three types of applications.

The second part of the development programme was aimed at extending the Phase 1 database and to provide a Proof Test irradiation specifically for the HTR-Modul reactor design. During this irradiation test fuel spheres were subjected to temperature cycles similar to HTR-Modul normal operation thermal cycles as well as thermal transients expected during frequent operational events.

The third part of the programme was aimed at bulk testing of fuel spheres under actual conditions in the AVR. Mass produced fuel spheres of similar designs as fuel spheres used during Phase 1 and Proof test programmes were irradiated in the AVR to full design burn-up. Fuel spheres were removed from the reactor at different burn-up stages and subjected to ex-pile heating tests. The main purpose was to identify and to quantify all mechanisms that could possibly contribute to the release of fission products from fuel elements. As a result of the negligible uranium contamination of TRISO coated particles manufactured by NUKEM and the excellent containment properties of intact coated particles for most fission products, the most important fission product release source is the release from failed coated particles. The most important contributing factors are:

- Production processes;
- Irradiation temperature;
- Burn-up;
- Irradiation by fast neutrons;
- Degree of acceleration (irradiation test time vs. real operating time); and
- Temperature transients.

The failed particle fraction resulting from production processes can be measured through quality control measurements performed during fuel sphere production whereby the burn-leach test of Triso particles in spheres is most sensitive and informative. Data as published in 1990 are:

Evaluation of free U and defective SiC layers in LEU Triso fuel elements[2]

Designation of fuel element (FE) population	AVR 19	AVR 21	AVR 21-2	LEU PHASE I	Proof test fuel
Production year	1981	1983	1985	1981	1988
Number of FE lots	14	11	8	–	–
Number of FEs produced	24,600	20,500	14,000	<100	< 200
Type of fuel	LEU UO_2	LEU UO_2	LEU UO_2	LEU UO_2	LEU UO_2
^{235}U enrichment	9.8%	16.7%	16.7%	9.8%	10.6%
Coating batch size	5 kg	3 kg	3 kg	5 kg	5 kg
Number of coating batches	65	54	29	1	8
Number of particle sets	4	2	3	–	1
Number of particles/FE	16,400	9,560	9,560	16,400	14,600
One particle uranium equivalent (in parts per million)	61	105	105	61	68
Evaluation of free uranium from burn-leach measurements					
Mean value (in parts per million)	50.7	43.2	7.8	35.0	13.5
Number of FEs tested in burn-leach	70	55	40	5	10
Number of FEs with 0 particle defects	31	42	38	3	8
Number of FEs with 1 particle defect	26	8	1	1	1
Number of FEs with 2 particle defects	9	2	1	1	1
Number of FEs with 3 particle defects	4	2	0	0	0
Number of FEs with 4 particle defects	0	0	0	0	0
Number of FEs with 5 particle defects	0	0	0	0	0
Number of FEs with 6 particle defects	0	1	0	0	0
Number of FEs with $\geq$ 7 particle defects	0	0	0	0	0

2 Nabielek et al., Nucl Eng Des 121 (1990), 199.

Failure during irradiation can be determined from irradiation data, and failure due to temperature transients can be determined using post-irradiation heating experiments.

If uranium contamination of coated particles and matrix graphite is negligible, the only remaining sources of fission product release are defective coated particles and diffusion from intact coated particles. Phase 1 Irradiation tests were designed to investigate both of these release mechanisms as a function of temperature.

The system of irradiation testing is shown below:

RDD program	**Old LEUs 1972-1976**	**HEU Program for PNP and HHT 1977-1981**			**LEU Program 1982-1993**
Coated Particle	**UO_2 TRISO UO_2 BISO**	**Variant 1 $(Th,U)O_2$ BISO**	**Variant 2 $(Th,U)O_2$ Triso**	**Variant 3 UC_xO_y Triso+ ThO_2 Triso**	**UO_2 TRISO**
<u>Test Goal</u>					
Particle Performance	**HFR-M5 DR-S6**	**BR2-P24**	**BR2-P25**	**BR2-P23**	**HFR-P4 SL-1**
FP Transport in Intact Particles	**DR-S4**	**FRJ2-P22**	**FRJ2-P23**	**FRJ2-P24**	**FRJ2-P27**
Release from Kernel	**-**	**FRJ2-P25**	**FRJ2-P25**	**FRJ2-P25**	**FRJ2-P28**
Chemical Effects	**FRJ2-P16**	**-**	**-**	**HFR-P3**	**HFR-P5**
Fuel Element Tests					
Integral performance in spherical FE	**DR-K5**	**HFR-K1**	**R2-K12 R2-K13**	**R2-K12**	**HFR-K3**
FIssion product transport in spherical FE	**-**	**-**	**FRJ1-K11**	**FRJ2-K10**	**FRJ2-K13 FRJ2-K15**
Large Scale Demonstration	**AVR 6**	**AVR 14**	**AVR 15**	**AVR 13**	**AVR 19 AVR 21 AVR 21-2**
Proof Tests					**HFR-K5 HFR-K6**
High Burnup					**HFR-EU1**
High Temp. for Hydrogen Prod					**HFR-EU1bis**

The 1993 summary of test results is:

Experiment	Fuel	Specimen number and type	No of particles	Irr. time (fpd)	Surf./ centre temperature (°C)	Burnup (% FIMA)	Fluence (E>16fJ) 10^25 m^-2	End of irradiation R/B Kr 85m	End of irradiation Fract. rel. Cs 137
R2-K12/1	(Th,U)O2 HEU	1 fuel element	10,960	308	950 / 1100	11.1	5.6	3E-07	2E-05
R2-K12/2	(Th,U)O2 HEU	1 fuel element	10,960	308	1120 / 1280	12.4	6.9	2E-07	1E-05
R2-K13/1	(Th,U)O2 HEU	1 fuel element	19,780	517	960 / 1170	10.2	8.5	7E-08	1E-05
R2-K13/4	(Th,U)O2 HEU	1 fuel element	19,780	517	750 / 980	9.8	6.8	5E-08	3E-06
BR2-P25	(Th,U)O2 HEU	12 small spheres	17,880	350	1018 / 1043	13.9 to 15.6	6.2 to 8.1	1E-06	4E-04
HFR-P4 /1	UO2 LEU	12 small spheres	19,600	351	915 / 940	11.1 to 14.7	5.5 to 8.0	8E-08	5E-06
HFR-P4/2	UO2 LEU	12 small spheres	19,600	351	920 / 945	9.6 to 14.9	5.5 to 8.0	8E-08	5E-06
HFR-P4/3	UO2 LEU	12 small spheres	19,600	351	1050 / 1075	9.9 to 14.0	5.5 to 8.0	9E-08	2E-05
SL-P1	UO2 LEU	12 small spheres	19,600	330	780 / 800	8.6 to 11.3	5.0 to 6.7	1E-06	5E-06
HFR-K3/1	UO2 LEU	1 fuel element	16,400	359	1020 / 1200	7.5	4.0	2E-07	9E-06
HFR-K3/2	UO2 LEU	1 fuel element	16,400	359	700 / 920	10.0	5.8	2E-07	2E-05
HFR-K3/3	UO2 LEU	1 fuel element	16,400	359	700 / 920	10.6	5.9	2E-07	2E-05
HFR-K3/4	UO2 LEU	1 fuel element	16,400	359	1020 / 1220	9.0	4.9	3E-07	1E-05
FRJ2-K13/1	UO2 LEU	1 fuel element	16,400	396	985 / 1125	7.5	0.2	2E-08	2E-05
FRJ2-K13/2	UO2 LEU	1 fuel element	16,400	396	990 / 1150	8.0	0.2	2E-09	2E-05
FRJ2-K13/3	UO2 LEU	1 fuel element	16,400	396	990 / 1150	7.9	0.2	7E-09	6E-06
FRJ2-K13/4	UO2 LEU	1 fuel element	16,400	396	980 / 1120	7.6	0.2	7E-09	6E-06
FRJ2-K15/1	UO2 LEU	1 fuel element	9,600	651	800 / 970	14.1	0.2	1E-06	<1E-5
FRJ2-K15/2	UO2 LEU	1 fuel element	9,600	651	980 / 1150	15.3	0.2	5E-09	<1E-5
FRJ2-K15/3	UO2 LEU	1 fuel element	9,600	651	800 / 990	14.7	0.1	3E-09	<1E-5
FRJ2-P27/1	UO2 LEU	3 compacts	7,340	232	880 to 1080	7.6	1.4	2E-06	2E-05
FRJ2-P27/2	UO2 LEU	3 compacts	7,340	232	1220 to 1320	8.0	1.7	1E-05	1E-04
FRJ2-P27/3	UO2 LEU	3 compacts	7,340	232	1080 to 1130	7.6	1.3	1E-07	1E-05
HFR-K5/1	UO2 LEU	1 fuel element	14,600	396	cycled	4.9	1.9	5E-08	still
HFR-K5/2	UO2 LEU	1 fuel element	14,600	396	cycled	6.6	2.7	7E-08	under
HFR-K5/3	UO2 LEU	1 fuel element	14,600	396	cycled	6.8	2.9	7E-08	irradi-
HFR-K5/4	UO2 LEU	1 fuel element	14,600	396	cycled	6.5	2.6	2E-07	ation
HFR-K6/1	UO2 LEU	1 fuel element	14,600	634	cycled	7.2	3.2	1E-07	not
HFR-K6/2	UO2 LEU	1 fuel element	14,600	634	cycled	9.3	4.6	4E-07	yet
HFR-K6/3	UO2 LEU	1 fuel element	14,600	634	cycled	9.7	4.8	4E-07	measured
HFR-K6/4	UO2 LEU	1 fuel element	14,600	634	cycled	9.2	4.5	9E-07	

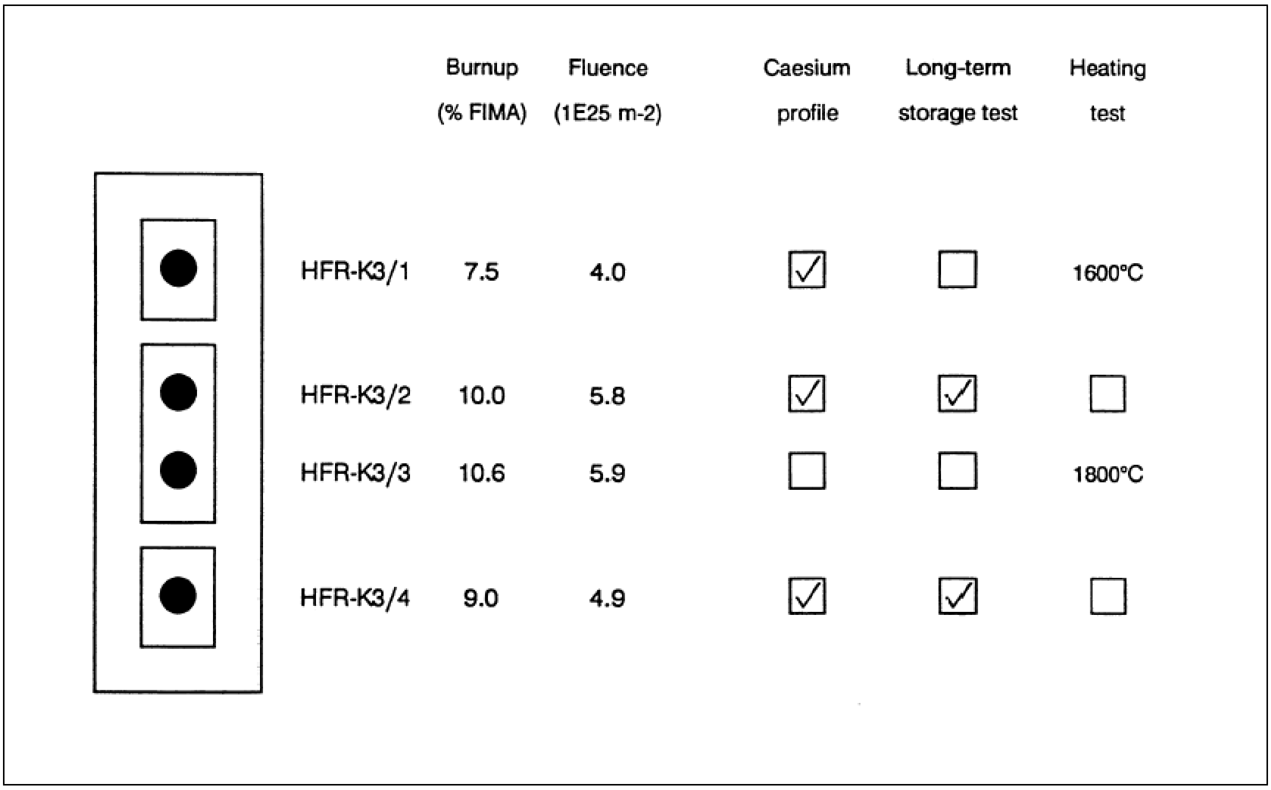
A more recent update on irradiation data:

Test	Sample Number	Irradiation Time (efpd)	Centre Temperature (°C)	Burn-up (%FIMA)	Fast Neutron Dose 10^{25} m^{-2} E>16fJ	R/B			
						BOL		EOL	
						Kr 85m	Kr 88	Kr 85m	Kr 88
HFR-P4	1	351	940-1008	11.1-14.7	5.5-8.0	$3.5x10^{-9}$	$3.6x10^{-9}$	$8.0x10^{-8}$	$6.2x10^{-8}$
	3		1010-1082	9.9-14.7	5.5-8.0	$3.6x10^{-9}$	$2.5x10^{-9}$	$8.0x10^{-9}$	$4.7x10^{-8}$
SL-P1	-****	330	743-794	8.6-11.3	5.0-6.8	$5.8x10^{-7}$	$9.7x10^{-7}$	$1.2x10^{-6}$	$1.0x10^{-6}$
HFR-K3	1	359	1200	7.5	4.0	$1.0x10^{-9}$	$1.2x10^{-9}$	$1.0x10^{-7}$	$1.7x10^{-7}$
	2		920	10.0	5.8	$9.0x10^{-10}$	$9.0x10^{-10}$	$1.0x10^{-7}$	$1.6x10^{-7}$
	3		920	10.6	5.9				
	4		1220	9.0	4.9	$2.0x10^{-9}$	2x10-9	$3.0x10^{-7}$	$2.8x10^{-7}$
FRJ2-K13	1	396	1125	7.5	0.2	$2.0x10^{-9}$	$1.5x10^{-9}$	$2.0x10^{-8}$	$1.4x10^{-8}$
	2		1150	8.0	0.2				
	3		1150	7.9	0.2	$8.0x10^{-10}$	$6.1x10^{-10}$	$7.0x10^{-9}$	$5.4x10^{-9}$
	4		1120	7.6	0.2				
FRJ2-P27	1*	232	880-1080	7.2-8.4	1.4	$1.0x10^{-6}$	$4.0x10^{-6}$	$1.6x10^{-6}$	$9.0x10^{-7}$
	2***		1220-1320	8.0-9.0	1.7	$8.6x10^{-7}$	$4.0x10^{-6}$	$1.0x10^{-5}$	$4.0x10^{-6}$
	3		1080-1130	7.2-8.1	1.3	$2.0x10^{-8}$	$2.2x10^{-8}$	$1.2x10^{-7}$	$1.0x10^{-7}$
FRJ2-K15	1	682	970	14.1	0.2	$2.0x10^{-10}$	-	$1.0x10^{-8}$	-
	2		1150	15.3	0.2	$2.47x10^{-10}$	-	$5.0x10^{-9}$	-
	3		990	14.7	0.1	$2.0x10^{-10}$	-	$3.0x10^{-9}$	-
HFR-K5	1	565	923	7.94	4.0	$4.76x10^{-10}$	$3.67x10^{-10}$	$1.64x10^{-7}$	$1.35x10^{-7}$
	2		909	10.18	5.8	$2.69x10^{-10}$	$1.47x10^{-10}$	$3.11x10^{-7}$	$2.56 x10^{-7}$
	3		903	10.48	5.9				
	4*		921	9.43	4.9	$2.51x10^{-7}$	$1.95x10^{-7}$	$3.5x10^{-7}$	$2.9x10^{-7}$
HFR-K6	1	634	1090	8.43	3.2	$5.0x10^{-10}$	$4.95x10^{-10}$	$1.52x10^{-7}$	$1.73x10^{-7}$
	2		1130	10.81	4.6	$3.0x10^{-10}$	$2.07x10^{-10}$	$4.38x10^{-7}$	$3.44x10^{-7}$
	3		1140	11.10	4.8				
	4**		1130	10.04	4.5	$4.5x10^{-7}$	$3.66x10^{-7}$	$1.22x10^{-6}$	$9.7x10^{-7}$

* One manufacturing defect, no additional failures during irradiation
** Two manufacturing defects, no additional failures during irradiation
*** Three manufacturing defects, no additional failures during irradiation
**** Five manufacturing defects, no additional failures during irradiation

III. The HFR-K3 irradiation test

Four fuel spheres were irradiated for 359 efpd. The rig contained three irradiation capsules and the fuel spheres were arranged one to a capsule for the top and bottom capsules with two fuel spheres in the centre capsule, as shown below:

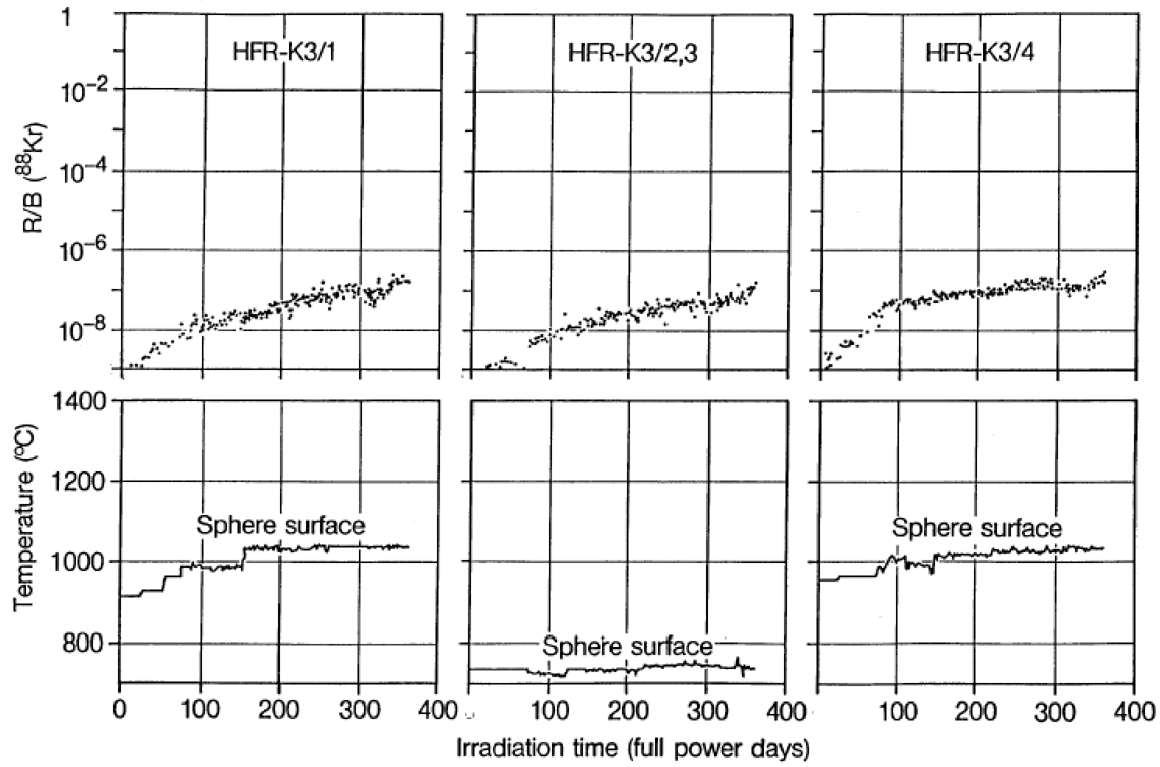


Summary of HFR-K3 Irradiation Results

Sphere Number	Centre Temperature (°C)	Burn-up (%FIMA)	Fast Neutron Dose E>0.1 MeV (x 10^{25} m^{-2})	R/B BOL Kr 85m	R/B BOL Kr 88	R/B EOL Kr 85m	R/B EOL Kr 88
HFR-K3/1	1200	7.5	4.0	1.0×10^{-9}	-	2.0×10^{-7}	-
HFR-K3/2	920	10.0	5.8	9.0×10^{-10}	-	1.0×10^{-7}	-
HFR-K3/3	920	10.6	5.9	9.0×10^{-10}	-	1.0×10^{-7}	-
HFR-K3/4	1220	9.0	4.9	2.0×10^{-9}	-	3.0×10^{-7}	-

HFR-K3 releases of Cs-134, Cs-137, and Ag-110m that were:

Sphere Number	Fractional Release		
	Cs-134	Cs-137	Ag-110m
HFR-K3/1	2.7×10^{-6}	9.1×10^{-6}	2.2×10^{-3}
HFR-K3/2	4.1×10^{-6}	1.7×10^{-5}	4.5×10^{-4}
HFR-K3/3	1.9×10^{-6}	1.7×10^{-5}	1.6×10^{-4}
HFR-K3/4	3.2×10^{-6}	1.4×10^{-5}	1.8×10^{-2}



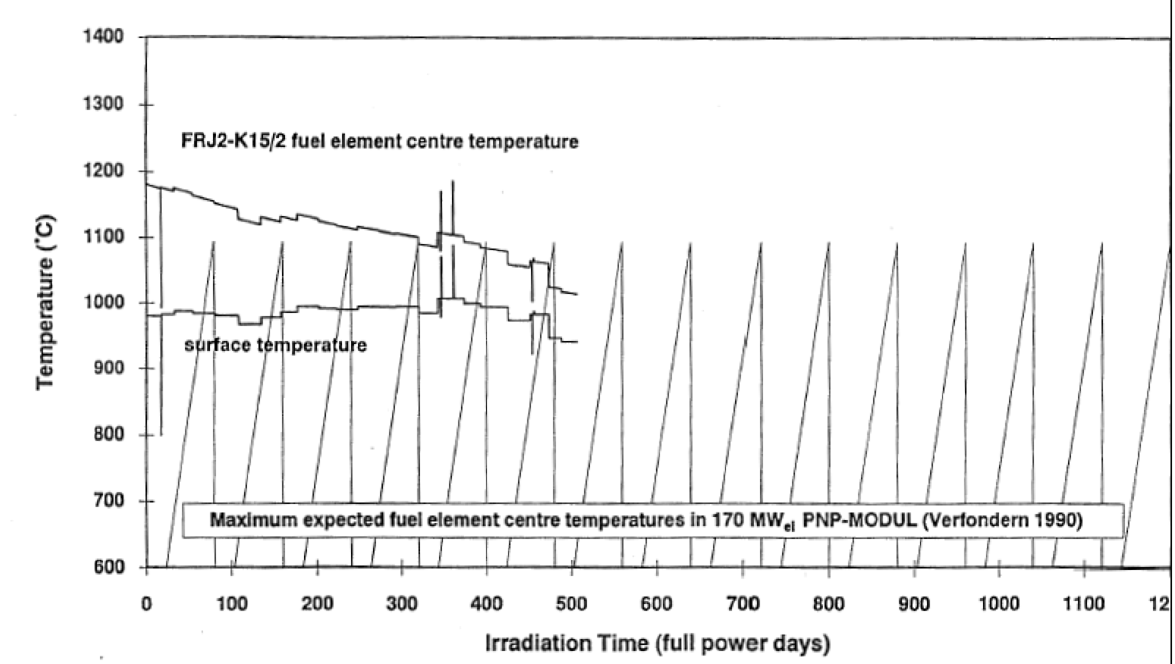
Fission gas release and surface temperatures of spherical fuel elements in HFR-K3

IV. The FRJ2-K15 irradiation test

FRJ2-K15 is an MTR irradiation experiment to accompany testing in AVR. A further aim was to investigate the effect of temperature transients on R/B values of irradiated spheres at burn-up values of 10% FIMA and 13% FIMA. Three fuel spheres from the production charge AVR 21-1 were irradiated in three separate capsules in a single irradiation rig in the FRJ2 reactor at Juelich for a period of 682 efpd. During the eighth irradiation cycle the fuel spheres were subjected to a +150°C temperature transient. The fuel sphere temperatures were adjusted to a value of 1100°C at the surface and kept at this value for 11 hours to simulate the first portion of a pressurized loss of forced circulation (PLOFC) event.

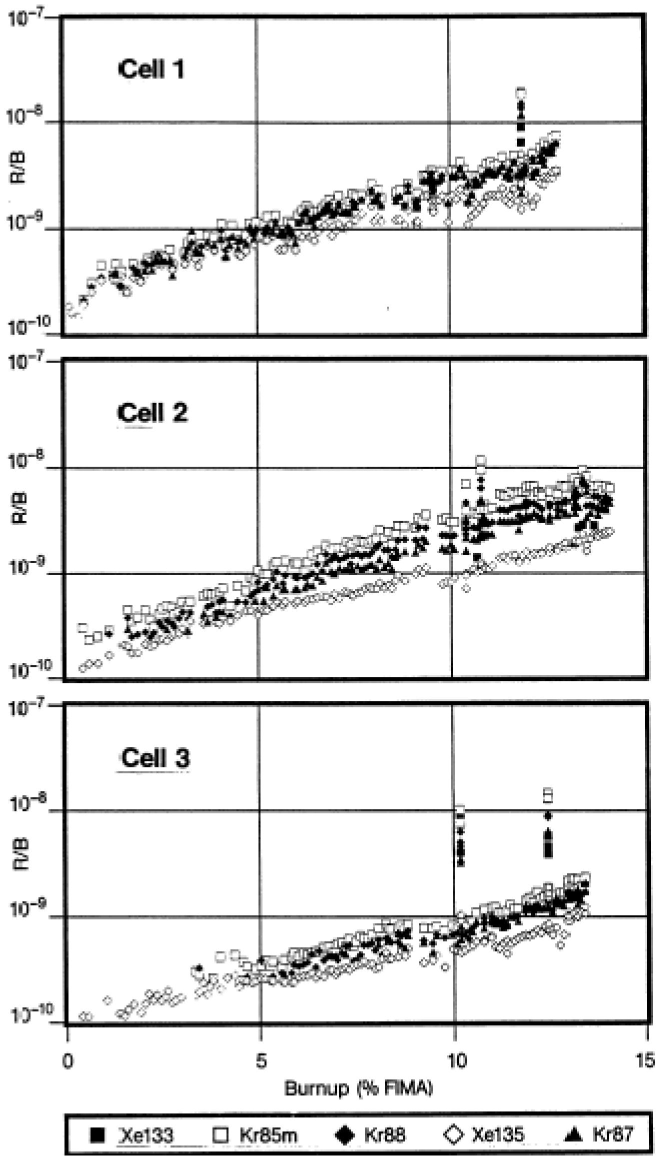
Summary of FRJ2-K15 Irradiation Results

Sphere Number	Temperature (°C)	Burn-up (%FIMA)	Fast Neutron Dose E>0.1 MeV (x 10^{25} m^{-2})	R/B			
				BOL		EOL	
				Kr 85m	Kr 88	Kr 85m	Kr 88
FRJ2-K15/1	970	14.1	0.2	2.0 x 10^{-10}	-	1.0 x 10^{-8}	-
FRJ2-K15/2	1150	15.3	0.2	2.47 x 10^{-10}	-	5.0 x 10^{-9}	-
FRJ2-K15/3	990	14.7	0.1	2.0 x 10^{-10}	-	3.0 x 10^{-9}	-



Centre and surface temperatures during irradiation of FRJ2-K15 (status November 1989) in comparison to temperatures in a process heat MODUL

FRJ2-K15 status from November 1989:



Fission gas release as a function of burnup in FRJ2-K15. R/B spikes visible during the +150°C temperature transients still do not reach the level of failure on one particle which would be at R/B around $1x10^{-6}$.

FRJ2-K15 Heating Test Results: Due to the closing down of the German HTR programme, by the time the irradiation ended in 1993, no heating tests were performed on the irradiated fuel spheres from this test. However, the new European HTR programme has provided for the shipment of all these spheres and the necessary KÜFA heating equipment to JRC Karlsruhe ITU, where extensive heating tests are planned.

V. The HFR-EU1 irradiation test

The main objective of the HFR-EU1 irradiation test[3] is the demonstration of the feasibility of high burnup for existing high-quality fuel. It will include in particular:

- the irradiation to the 200 GWd/t range to be reached within 2 years;
- the evaluation of fuel performance at such ultra-high burnup to explore the real limits of the existing coated particle that have formerly only been designed for operational conditions of the HTR MODUL and for direct cycle and process heat plant;
- the extension of the existing data base for the metallic fission product release, particularly the silver isotope Ag-110m for an improved assessment of the particle choice for the direct cycle gas turbine HTR concept;
- the post-irradiation demonstration of fission product retention beyond 1600 °C are the main subjects of investigation.

The main objective of the irradiation of the Chinese fuel elements in the HFR-EU1 test is the demonstration of the feasibility of high burnup beyond 100 GWd/t, in particular:

- the irradiation up to the 160 GWd/t burnup range, to be reached within 2 years;
- the evaluation of fuel performance at such ultra-high burnup to explore the real limits of the coated particle system that have been designed for operational conditions of the HTR-10;
- the post-irradiation demonstration of fission product retention beyond 1600 °C are the main subjects of investigation.

Three spherical fuel elements of German production with 60 mm outer diameter, type AVR GLE4/2 (AVR reload 21-2) will be irradiated in one of the two independent capsules of the HFR-EU1 test. The enrichment in U-235 is 16.7 %.

During the AVR reactor operation, GLE-4 fuel elements have reached a calculated average burnup of 8.6 % FIMA with a maximum burnup of approx. 21 % FIMA. Since no peculiarities in the gas release measurements have been observed, a high potential of LEU-TRISO particles to withstand consistently extreme high fuel burn-up can be expected.

German particles and spheres are characterized by:

Coated Particle	
Particle batch	HT385-393 HT395-404 HT406-423
Kernel composition	UO_2
Kernel diameter [µm]	502.2
Enrichment [U-235 wt.%]	16.76 and 16.67, respectively
Coating thicknesses [µm]:	
buffer	92.3
inner PyC	40.6
SiC	35.9
outer PyC	39.6
Fuel Element	
Heavy metal loading [g/FE]	6.0
U-235 contents [g/FE]	1.00 ± 1 %
Number of coated particles per FE	9560
Volume packing fraction [%]	6.2
Defective SiC layers [U/U_{tot}]	7.8×10^{-6}
Matrix graphite grade	A3-3
Matrix density [kg/m³]	1750

[3] Conrad et al., Test Specification for the Irradiation Experiment HFR-EU1, EU-HTR-F-WP2 report Document Number: HTR-F-01/07-D-2.1.1 from April 1991.

Chinese particles and spheres are characterized by:

Coated Particle	
Particle batch	V000802
Kernel composition	UO_2
Kernel diameter [μm]	490.3
Enrichment [U-235 wt.%]	17.08 %
Coating thicknesses [μm]:	
buffer	97.7
inner PyC	42.0
SiC	37.8
outer PyC	40.8
Fuel Element	
Heavy metal loading [g/FE]	5.02
U-235 contents [g/FE]	0.858
Number of coated particles per FE	8500
Volume packing fraction [%]	5.0
Defective SiC layers [U/U_{tot}]	$2.3x10^{-7}$
Matrix graphite grade	A3-3
Matrix density [kg/m³]	1760

The irradiation experiment will be performed in the High Flux Reactor (HFR) Petten. The irradiation rig of the HFR-EU1 experiment will consist of two separated, fully instrumented and monitored capsules for individual control of fission gas release. The irradiation shall be conducted until a maximum burnup of approximately 20 % FIMA in the highest loaded GLE-4 fuel element is achieved, whereby the highest loaded INET fuel element shall have about 15 % FIMA. The fast neutron fluence of the highest loaded fuel element shall not exceed $6x10^{25}$ m^{-2} (E>16 fJ). Below 18 % FIMA of the highest loaded GLE-4 fuel element, the irradiation shall be discontinued at the end of the actual cycle if a significant enhanced ^{85m}Kr release is observed as a result of massive failure of coated particles. Beyond 18 % FIMA, however, the irradiation test shall be continued, until the final burnup of 20 % FIMA in the highest loaded GLE-4 FE is reached, irrespective of a potentially massive failure of coated particles late in the test. The occurrence of particle failure should initiate an assessment of the consequences of later PIE heating tests of this fuel for the KÜFA facility. The irradiation duration for the two INET fuel elements shall follow the scheme of GLE-4 spherical fuel elements.

Irradiation targets of HFR-EU1 in comparison to respective operating conditions of reference test HFR-K6 and various HTR designs are:

Parameter	HTR-Module	HFR-K6	HFR-EU1 GLE-4	HFR-EU1 INET	HTR-10	GT-MHR	PBMR
Fuel	UO_2 LEU-TRISO	UO_2 LEU-TRISO	UO_2 LEU-TRISO	UO_2 LEU-TRISO	UO_2 LEU-TRISO	UCO TRISO	UO_2 LEU-TRISO
Max. Burnup [% FIMA]	9.8	9.7	20	15	8.6	(20) fissile (6) fertile	9.8
Max. Fluence 1E25m^{-2} (E>0.1 MeV)	2.4	4.8	6	4.6		4.0	2.4
Max. Temperatures: Gas outlet [°C] Fuel surface [°C] Fuel centre [°C]	 750 926 1130	 - 650/850 800(1) 1000(1) 1200(2)	 - 950(4) 1100	 - 880 1000	 700/900 831 864	 850 (1250)	 900 1000 1100
Maximum fission power per FE [kW]	1.6	1.5(3) - 2.7(3)	2.3	1.95	0.52	<1.5	3.4/4.5(6)
Maximum fission power per particle [mW]	150	<200	<240	<210	<250	30(5) (< 400 at BOI)	250/300(6)
Irradiation time efpd		634	560 - 600	560 - 600	1116	(834)	900
Simulation of number of passes	17	17	-	-	-	-	10

(1) 1/3 and 2/3 temperature of cycle, simulation of one full pass of FE in the HTR-Module core

(2) Simulation of operational temperature transient of 5 hours at beginning, half and end-of irradiation

(3) Maximum power at beginning of irradiation

(4) Increases in order to keep the central temperature constant

(5) Average value

(6) Equilibrium core / first core

Predicted irradiation characteristics of fuel elements of HFR-EU1 test

Fuel element type	GLE-4			INET	
DESIGNATION	**HFR-EU1/5**	**HFR-EU1/4**	**HFR-EU1/3**	**HFR-EU1/2**	**HFR-EU1/1**
Temperature central of FE [°C]	1100	1100	1100	1000	1000
Temperature at surface, North pole [°C]	950	930	930	870	890
Fission power per FE [W]	1950	2150	2300	1750	1450
Fission power per particle [mW]	204	225	241	206	171
Burnup [GWd/t]	170	190	200	155	125
Fast fluence [m^{-2}](E>16 fJ)	$5.1x10^{25}$	$5.7x10^{25}$	$6x10^{25}$	$4.6x10^{25}$	$3.8x10^{25}$
Axial position with respect to centre-line core [mm]	-140	-70	Centre-line core	+137	+207

VI: Some early attempts in benchmark calculation testing from IAEA CRP6

Benchmarking is deemed an important step for the validation and verification of computer models[4]. HTR fuel performance codes have been developed in the past and regained attention in the recent years with increasing interest in the development of advanced fuel technology. Future HTRs will have to be designed for higher temperatures and higher burnups. Respective computer tools will be essential in the process of optimization of the fuel design.

Many participants either possess a fuel performance code or have recently started the development of a model. Most of them are at the moment basically concentrating on the conditions during irradiation/normal reactor operation, and may be later extended to accidental conditions. The benchmark exercises will help compare the quality of the models against experimental data and also, of course, against each other, thus being an ideal support for further development and/or refinement. The list of existing HTR fuel performance codes is as follows:

France	ATLAS
Germany	CONVOL and PANAMA
Japan	JAERI model for short-lived gas release
Russia	GOLT-V1
South Africa	NOBLEG
United Kingdom	STRESS3
United States	PARFUME and an MIT model

First calculations within the frame of the benchmark for normal operating conditions have been conducted. Focus was on the prediction of fuel performance during the irradiation experiment HFR-EU1. The five codes CONVOL, PANAMA, GOLT-V1, STRESS-3, and PARFUME have been applied to assess coated particle behaviour for this test at very high burnups. In addition, the ATLAS code has been applied to the conditions of the HFR-P4 experiment.

The HFR-EU1 is a high burnup irradiation experiment in HFR Petten. Three spherical fuel elements from the latest German production plus two fuel spheres from recent Chinese production in separate capsules are to be irradiated to a maximum burnup of 20% FIMA. Assumptions for the predictive calculations were:

Irradiation time:	600 efpd
Irradiation temperature:	950 °C surface, 1100 °C centre
Maximum burnup :	20% FIMA
Maximum fast fluence :	$6x10^{25}$ m^{-2}, E> 16 fJ

Some of the results are summarized below:

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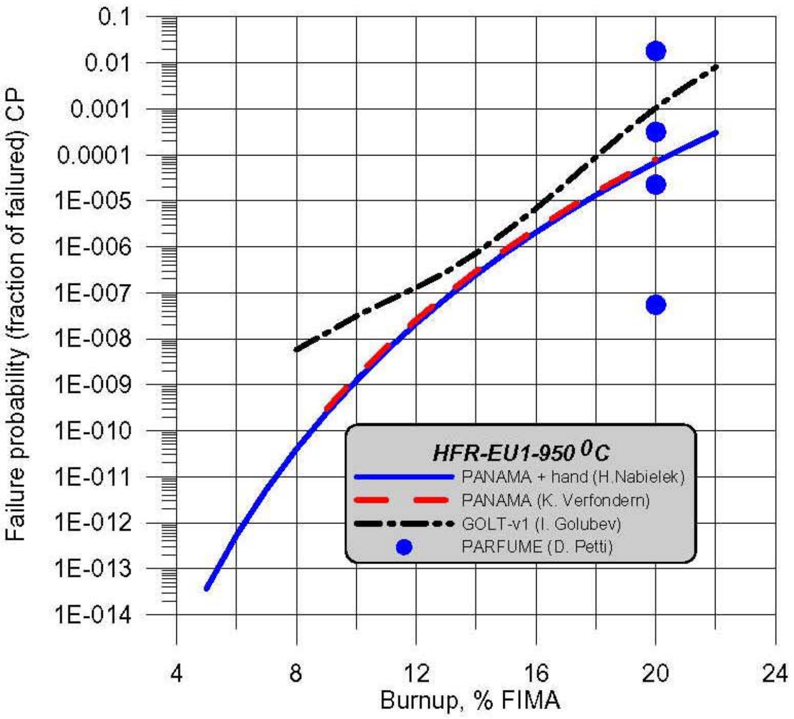
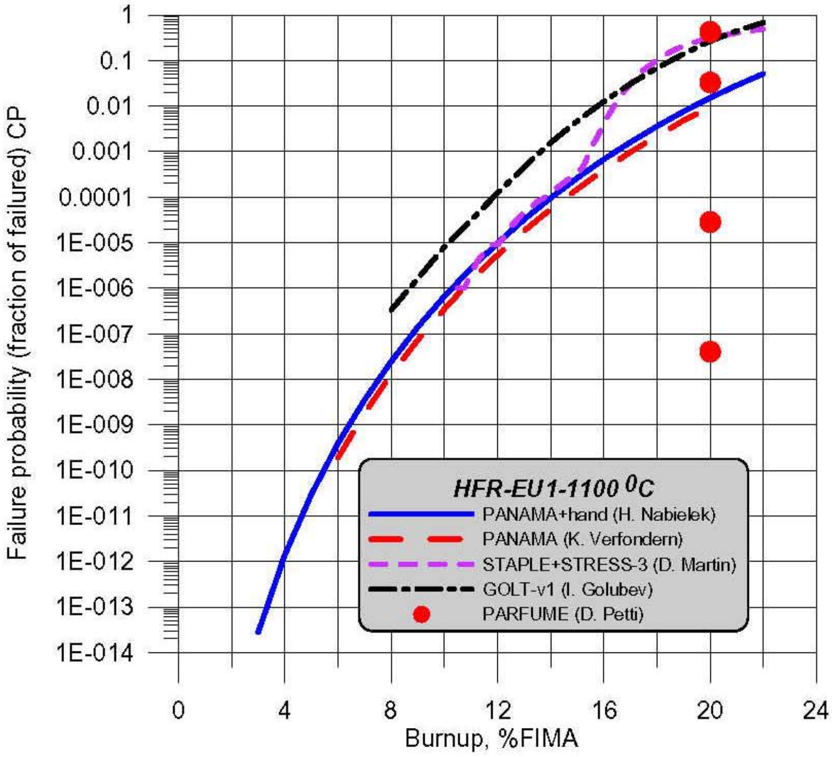
4 Phélip et al., TRISO Particle Fuel Performance Code Benchmarking Activities Performed under the IAEA 6th Coordinated Research Programme on Advances in HTR Fuel Technology, Technical Meeting on “Current Status and Future Prospects of Gas Reactor Fuels”, held at IAEA, Vienna, 7-9 June 2004.





Prediction of coated particle failure probability for HFR-EU1 irradiation test with different computer models

The PANAMA code represents a simple thin shell pressure vessel with SiC as the only layer considered. Assuming an initial SiC strength of 834 MPa and an initial Weibull modulus of 8.02, PANAMA predicts the first particle in a German sphere to fail (which is about equivalent to reaching a failure fraction level of 10^{-4}) at a burnup between 14 %FIMA (T_{irr} = 1100 °C) and 20 %FIMA (T_{irr} = 950 °C). The failure probability of particles in the Chinese fuel elements is predicted to be somewhat lower due to a smaller kernel diameter and thicker buffer and SiC layers.

The Russian GOLT-V1 code includes the modelling of both SiC and pyrocarbon layers assuming temperature and fluence dependence of PyC parameters as well as an irradiation induced dimensional change. Calculations for the HFR-EU1 test were conducted for different sets of Weibull parameters. The results revealing a strong dependence on material properties show the first particle to fail between 14 and 16 %FIMA (Weibull parameter comparable to PANAMA calculation), and at > 10 %FIMA for the cases of lower SiC strength data.

Calculations with the UK code STRESS3 in connection with the statistical code STAPLE consider the particle kernel and all layers of the TRISO coating. The effect of kernel swelling with burnup is taken into account as well as the variability in the layer thicknesses. A first STRESS3 calculation with mean particle specifications indicates that the fracture stress of the SiC layer of assumed 392 MPa will be reached in the burnup range of about 21-24 %FIMA depending on the assumed swelling rate. Applying the STAPLE code with statistical variations in the layer thicknesses, involving some 10^6 STRESS3 computer runs, results in a failure fraction exceeding the level of 10^{-4} (or 1 failed particle) near 14 %FIMA. A significant improvement of this value could be achieved, if the large variability of the buffer layer thickness would be reduced.

The US code PARFUME includes two modes for a particle to fail, apart from the traditional pressure vessel failure, also an SiC failure caused by irradiation-induced shrinkage cracking in the IPyC layer is considered. Furthermore there are two options of calculating CO pressure; with the two irradiation temperatures and two sets of Weibull data (873 MPa, 8.02 and 409 MPa, 6.0), a total of 8 predictive calculations have been conducted. The internal pressures based on the two CO pressure options were calculated to be about 100 and 25 MPa for T_{irr} = 1100 °C, and 60 and 15 MPa for T_{irr} = 950 °C, respectively. Internal gas pressure dominated in most cases the total failure probability. Predicted failure fractions of 20 %FIMA burnup particles vary in a wide range between 4×10^{-8} and 0.43 because of differences in gas pressure and SiC strength.