OPERATIONAL EXPERIENCE WITH HTR-FUEL IN THE AVR EXPERIMENTAL POWER STATION

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The 15 MW experimental power station with HTR reactor, operated by Arbeitsgemeinschaft Versuchsreaktor (AVR) in Jülich, FRG, went into operation in 1967.

One of its main tasks is to test different kinds of fuel elements and to demonstrate in how far the concept of the pebble-bed reactor permits a safe and reliable operation at high gas temperatures.

CONCISE DESCRIPTION OF THE REACTOR CONCEPT

The reactor core consists of a bed of about 100,000 sperical fuel elements, each 6 cm in diameter, which con-

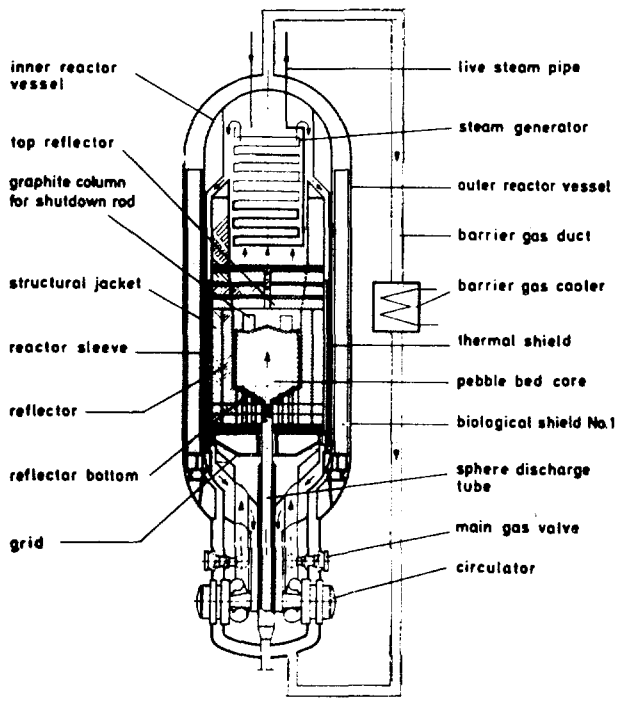

Fig. 1: Vertical Cross Section of AVR Reactor

tain heavy metal in small coated particles of a size of about 0.5 mm. The steam generator is arranged on top of the core within the reactor vessel (fig. 1). It is shielded against radiation from the core by a graphite reflector and by two other layers of carbon bricks. The coolant is helium. This has a pressure of 10 bars. It is circulated by two blowers situated in the lower part of the reactor vessel. The average gas outlet-temperature has been increased at first successively to the design value of 850 ^{o}C, and lateron in 1974, to 950 ^{o}C.

The inner reactor vessel is concentrically surrounded by a second reactor vessel. The gap is filled with helium, the pressure of which is somewhat higher than that of the coolant-gas pressure. In addition there is a first biologic shielding between the cylindric parts of the reactor vessels. The double-vessel system is sourrounded by a containment and by a concrete tower with a wall thickness of 1.5 m. The side reflector has four perpendicularly pierced ledges extending into the core. Each boring contains a shut-down rod.

The rods are inserted from below into their borings. They are coupled via a pinion with a heavier counter rod. The coupling is situated between pinion and driving motor. If one opens this coupling, the heavier counter rod will fall down, thus shifting the absorber component via the pinion into the core region.

The addition and removal as well as the circulation of fuel balls occurs continuously during reactor operation; thus no surplus reactivity needs to be loaded for compensation of burn-up. The fuel balls are removed by a pipe. It has a length of about 15 m, an inner diameter of 500 mm, and ends into a slowly rotating slotted disc. Hereafter follow the instruments to singulize the fuel element flow, to extract the damaged fuel elements, to collect the scrap, and to retransport them either into the core or into the extraction lock.

600 elements are removed from the core per day. The burn-up of each element is measured, those 60 with the highest burn-up are separated via the sluice-system. The

other 540 fuel balls, along with 60 fresh ones, are transported upwards back into the core. Thus, during its lifetime, a fuel ball passes the core 10 times on the average. The burn-up of sluiced elements is beyond 15 % fima (for elements of 6 g heavy metal).

The steam generator consists of four parallel Benson-systems with water injection for temperature control. The injection is performed outside the reactor vessel. Each system, up to this injection, consists of five parallel pipe strings. The superheater unit following the injection system consists of 10 parallel pipe strings each. This means the steam generator consists of 60 pipe strings individually leading into and out of the reactor vessel. The pipes of the steam generator are of ferritic steel as usual in conventional boiler construction.

The generated steam has a temperature of 505 ^{o}C and a pressure of 72 bars.

The coolant gas is circulated by two blowers. They are situated in the lower part of the reactor vessel. Each blower consists of an asynchronous motor and of a fan wheel arranged on the motor shaft in overhung position. The bearings are oil-lubricated. The drive is performed via a frequency transformer. The rotational speed can be chosen between 400 and 4,400 rpm. Labyrinths and a blocking gas-system prevent an inflow of both oil into the coolant-gas circuit and coolant gas into the motor casing.

A gas purification plant extracts the gaseous impurities from the coolant.

Fig. 2 shows some design data of the power station.

Thermal power	46 MW
Gross electrical output	15 MW
Maximum fuel element surface temperature	1117° C
Maximum fuel temperature	1224° C
Helium pressure	10,9 bar
Helium flow rate	13 kg/sec
Core inlet helium temperature	270° C
Core outlet helium temperature	950° C
Main steam flow rate	56 t/h
Main steam pressure	72 bar
Main steam temperature	505° C

Fig. 2: Design Data

FUEL ELEMENT TESTING IN A V R

The testing of fuel elements in AVR is characterized by some features which cannot be achieved with usual radiation tests in material test reactors.

- Large numbers of fuel elements are tested in AVR which were fabricated in series.
- The testing occurs under operation conditions of a nuclear power plant. They comprehend mehcanical stresses of the feed system, chemical reciprocal effects of cooling-gas impurities, change of temperature and power as a follow of fuel elements running through the core for several times and the load change of the power plant. The hot gas temperature of 950 ^{o}C permits to perform specific fuel element testing with regard to nuclear process heat.
- The large number of introduced fuel elements permits to perform post-irradiation investigation on a large number of removed fuel elements. Thus it is possible to make statistically safe statements on property variations.

The basis of fuel element development is the isostatically pressed graphite fuel ball of an outer diameter of 60 mm and an inner fuel zone of 50 mm, the latter having coated fuel particles embedded.

Two projects of fuel element development have been followed up in AVR up to now:

- For AVR itself and for THTR 300, the testing of pressed fuel elements started in 1969 with high enriched uranium (HEU) fuel with an inventory of 50,000 fuel elements on the whole, and (Th,U)C_2 as fuel material. The carbidic fuel was replaced in 1971 by fuel elements with oxide fuel (Th,U)O_2. On the whole, 100,000 fuel elements with (Th,U)O_2 were introduced, including those 15,000 fuel elements from original THTR production. The coating of all particles consists of a pyrocarbon coating only.

- The second project aims at succeeding THTR 300 systems, especially at HTRs for process heat production. For this additional requirement, coated particles have an additional layer of SiC (Triso particles). For succeeding systems of THTR 300 it is planned to use fuel of low enriched uranium (LEU). Therefore fuel elements with SiC-coated particles are tested in AVR, on the one hand containing $(Th,U)O_2$, on the other hand containing UO_2.

Parallel to testing fuel element behaviour under reactor operation conditions, fuel elements removed from the reactor are used or made available for different tests of temporary or ultimate storage, or reprocessing. These tests will be presented in a different lecture of this conference /5/.

The table of fig. 3 (from /1/) gives a survey on the different kinds of fuel element charges introduced into AVR up to now.

RESULTS OF FUEL ELEMENT TESTING

As long as the fuel element is inside the reactor, no information is available concerning the individual fuel element on processes which might occur during its stay in the reactor. Only when it is extracted from the reactor, only then a statement can be made. We therefore distinguish between two procedures of fuel element control:

- integral measurements (noble fission gas, solid fission products in the coolant gas), giving information on activity release of all fuel elements in the core.
- measurements on an individual fuel element after extraction from the reactor which are mostly performed in the Hot Cells Laboratory of KFA Nuclear Research Center Jülich.

In the following are described some characteristic operational measurements and laboratory investigations, showing up the most essential results.

Table 3: Spherical Fuel Elements Manufactured by NUKEM/HOBEG for AVR Reloads

Isostatic pressing has been used for reload 3 through to 21

AVR RELOAD		FUEL PARTICLES			FUEL ELEMENT			
Manufacturer's Designation	Operator's Designation	Coated Particle Type		Uranium Enrichment	Fuel Element Graphite	Firing Temperature	Number of Fuel Elements	Insertion Date
0[a]	UCC	$(Th,U)C_2$	HTI BISO	93 %	Graphite	1450°C	30,155	Jul '66
1	T	$(Th,U)C_2$	HTI BISO	93 %	Graphite	1450°C	7,510	Oct '68
3	GK	$(Th,U)C_2$	HTI BISO	93 %	A3-3[b]	1800°C	17,770	Apr '69
4	GK	$(Th,U)C_2$	HTI BISO	93 %	A3-3	1800°C	6,210	Jul '70
5-1	GK	$(Th,U)C_2$	HTI BISO	93 %	A3-3	1800°C	25,970	Nov '70
5-2	GO 1	$(Th,U)O_2$	HTI BISO	92 %	A3-3	1800°C	20,825	Dec 71
6-1	GO 1	$(Th,U)O_2$	HTI BISO	92 %	A3-3	1900/ 1800°C	11,000	Oct '73
6-2	GLE 1	UO_2 UO_2	LTI BISO LTI BISO	15 % 0,7 %	A3-3	1900°C	2,446	Dec '73
7	GO 1	$(Th,U)O_2$	HTI BISO	93 %	A3-3	1900°C	7,840	Jan '73
6-1	GFB 1	UO_2 ThO_2	LTI BISO LTI BISO	93 % -	A3-3	1900°C	1,440	May '74
8-2	GFB 2	UO_2 ThO_2	LTI TRISO LTI BISO	93 % -	A3-3	1900°C	1,610	May '74
9	THTR 1	$(Th,U)O_2$	HTI BISO	93 %	A3-3	1950°C	5,145	Sep '74
10	THTR 2	$(Th,U)O_2$	HTI BISO	93 %	A3-3	1950°C	10,000	Dec '74
11	THTR 2	$(Th,U)O_2$	HTI BISO	93 %	A3-3	1950C	5,000	Dec '74
12	GO 1	$(Th,U)O_2$	HTI BISO	93 %	A3-3	1950°	11,325	Mar '76
13-1	GFB 3	UC_2 ThO_2	LTI TRISO LTI BISO	90 % -	A3-27[b]	1950°C	6,077	Dec '77
13-2	GFB 4	UC_2 ThO_2	LTI TRISO LTI BISO	90 % -	A3-27	1950/ 1800°C	5,861	Jul '80
13-3	GFB 5	UCO_2 ThO_2	LTI TRISO LTI TRISO	92 % -	A3-27	1950°C	5,354	Dec '77
14	GO 1	$(Th,U)O_2$	HTI BISO	93 %	A3-27	1950°C	9,930	Nov '76
15	GO 2	$(Th,U)O_2$	LTI TRISO	93 %	A3-27	1950°C	6,087	Feb '81
18	GO 3	$(Th,U)O_2$	HTI BISO[d]	93 %	A3-27	1950°C	11,547	Jul '81
19	GLE 3	UO_2	LTI TRISO	10 %	A3-27	1950°C	24,615	Jul '82
20	GO 2	$(Th,U)O_2$	LTI TRISO	93 %	A3-27	1950°C	11,854	Planned 1985
21	GLE 4	UO_2	LTI TRISO	16,7 %	A3-27	1950°C	ca.20,000	Planned 1984

Legend

a) First core manufactured by Union Carbide
b) A3-3 and A3-27 are NUKEM manufactured matrix graphites
c) ThO_2 kernels with fission product gettering additives
d) Heavy metal contamination $\cdot 10^{-4}$

HTI high temperature isotropic pyrocarbon
LTI low temperature isotropic pyrocarbon
BISO pyrocarbon only coating
TRISO pyrocarbon/silicon carbide/pyrocarbon coating

Fig. 3

RELEASE OF FISSION NOBLE GAS

The content of noble fission gas is continuously controlled in the circuit and immediately indicates the failure of particles in a fuel element charge. Typical noble gas activities in the primary circuit (about 1.600 m^3 Helium) under full load and hot-gas temperatures of 950 ^{o}C are:

30 - 50 Ci using fuel elements of the older design with carbidic fuel and Biso particles,

< 20 Ci using younger fuel elements with oxide fuel and Triso particles to some extent (the value is presently close to 19 Ci).

Activity release is therewith practically determined by only heavy-metal contamination of the fuel element matrix, whereas contamination is a consequence of fuel element fabrication. With Biso particles the uranium-contamination fraction is 10^{-4} to 10^{-3}, with modern Triso particles < 10^{-4}.

The highest noble-gas activity measured up to now was 160 Ci in 1976 when 2,500 elements of a special charge showed up particle damage to a greater extent. It was possible to prevent another activity increase, for during their circulation in the reactor, the damaged fuel balls were extracted much faster than another damage of particles increased. A continued reactor operation, however, would have been possible with ten times the maximum noble-gas activity measured up to now.

When, as described before, a fuel element charge indicates the begin of a particle damage, the routine reactor-operation control-system immediately indicates an increase of noble gas activity. But one does not know yet which is the damaged charge. Sample elements have to be removed therefore and studied individually. For that purpose an appropriate procedure has been developed by KFA by which the fuel balls are individually heated up to 1,250 ^{o}C and their Kr-85 release is measured. From the result one can clearly distinguish between normal or increased noble-gas release.

In order to characterize different fuel element charges, 220 radiated AVR fuel balls were subjected to this annealing test up to now.

RELEASE OF SOLID FISSION PRODUCTS

Solid fission products are measured in both hot and cold gas. Beyond that, the 5 mm thick fuel-free zones of removed fuel elements are investigated as random samples. Long-living isotopes of cesium, silver and strontium are of particular interest. They mostly are deposited on graphite surfaces and therefore will be transported along with graphite dust which is present in coolant gas. A partial vapourization of these isotopes occurs only at hot places of the core.

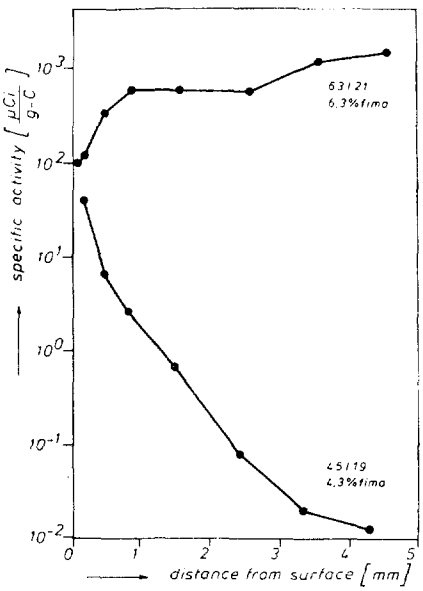

Fig. 4: Cs-137 in the Fuel-free Zone

Of special interest is the activity distribution in the fuel-free zone of the fuel balls (Fig. 4). The concentrations at the surface result from condensation of steam fractions in colder core zones, for during their last period in the core, the removed fuel balls had been situated in the lower cold zone. Concentrations in somewhat deeper layers of the fuel ballsgive information on the fuel-element release behaviour.

The table of Fig. 5 shows, as an example, the temperature dependence of the release of solid fission products

Hot Gas Temperature	770°C	850°C	950°C
Sr 90	0,1	0,1	12,8
Ag 110m	2,5	8,1	39
J 131	8,8	19,8	63
Cs 134	0,6	0,7	4,3
Cs 137	0,5	0,6	2,2

Fig. 5: Annual Release for Solid Fission Products [Ci/a]

for 1973/74. Especially strontium, deposited on finest particles of dust, requires special precautions when performing works of maintenance and repair on components of the primary circuit.

FUEL ELEMENT CORROSION

It is impossible, during reactor operation, to avoid small amounts of water and carbon dioxide being mixed with coolant gas. Normally their concentrations are around 1 to 3 µbar for water, and 5 to 30 µbar for CO_2, and they are negligible for fuel element corrosion. Occasionally, for example when the primary circuit has been opened after shut-down, higherlevels of impurities occur. The hot gas temperature for succeeding operation is then limited that much that fuel element corrosion cannot exceed a permissible limit - 1 weight-% during fuel element lifetime. Radiated fuel elements for post-investigation are additionally controlled by random samples on their weight. The life-time of a fuel ball, which generally is about 5 years, results an average loss of

appr. 0.7 g = 0.35 % for older elements with carbidic fuel,

appr. 0.5 g = 0.25 % for younger elements with oxide fuel.

In order to keep these values as low as possible, much attention is paid to the corrosion-resistant quality of the fuel element matrix which above all depends on the degree of graphite purity. Appropriate fabrication measures guarantee a good corrosion-resistance of fresh fuel elements. During reactor operation, however, further impurities could arise at the surface, in particular those which have a catalytic effect on carbon oxidation.

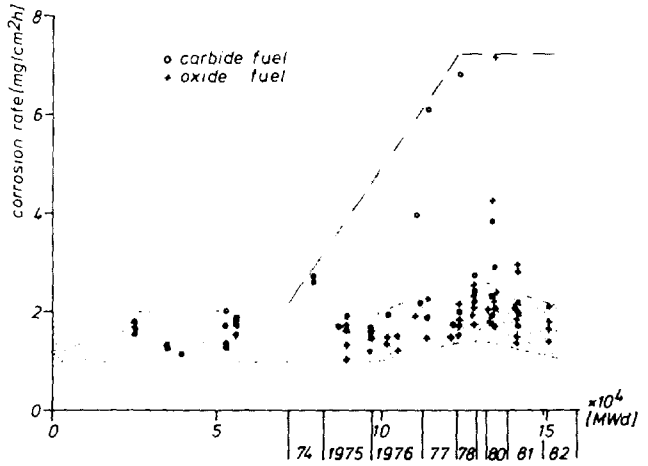

Fig. 6: Corrosion Rate versus Time

Figure 6 shows the corrosion rates (at 1000 °C and 1 % water steam in helium), measured on burnt fuel elements in function of reactor work performed. Starting 1976, about 2 years after increasing the hot-gas temperature to 950 °C, fuel elements with carbidic fuel are found sometimes showing-up a corrosion rate of up to three times an increase. This is explained by the catalysis of increasingly available fission products, especially strontium and barium. After 1978, fuel balls with oxidic fuel show up increased corrosion rates, too. The origin were not released fission products. Moreover, tracers of metal and iron and other metals were found on their surfaces, presumably as a follow of disturbed thermo-couple coatings. In springtime of 1978, more than 20 t of water entered the shut-down reactor, because of a damaged steam generator tube. That's why humidity reached the interior of the thermo-couple coatings and made them burst by swelling up the filling material, which is magnesium oxide /4/.

We sincerely hope the corrosion rates of burnt fuel elements to have gone back in future to the normal measure, for all fuel balls with carbidic fuel have meanwhile been removed from the reactor; also those with metal tracers, as mentioned before, will no longer be in the circuit in a few years.

METALLOGRAPHIC POST-INVESTIGATIONS

The metallographic investigations of irradiated fuel balls at KFA delivers extremely valuable results for the evaluation of fuel element behaviour. Slices are made through the fuel element and, above all, through the coated particles which permit an investigation by metallographic methods. The result makes statements on graphite-matrix behaviour, different coatings and fuels, and on their physical and chemical reciprocal effects. 150 radiated AVR fuel elements were metallographically investigated up to now.

VARIATION OF MECHANICAL PROPERTIES AND VARIATION OF DIMENSIONS

After having passed the fuel cycle for several times on their way through the reactor core, the fuel balls are mechanically stressed, mainly by the components of fuel element charging, in some cases they are even damaged. Damaged fuel balls are separated by the scrap-separator and collected in the scrap can. The separated fuel balls indicate mostly small exfoliations of the fuel-free graphite coating. Also fuel balls are randomly found splitted into several pieces. The fuel element charging system has been transporting 2 million fuel balls up to day. The scrap rate was about 0.5×10^{-4}, a value without influence on reactor operation. Most essentially it turned out that particle coatings remained undamaged, although fuel element damage had occurred, and there was no additional release of fission products.

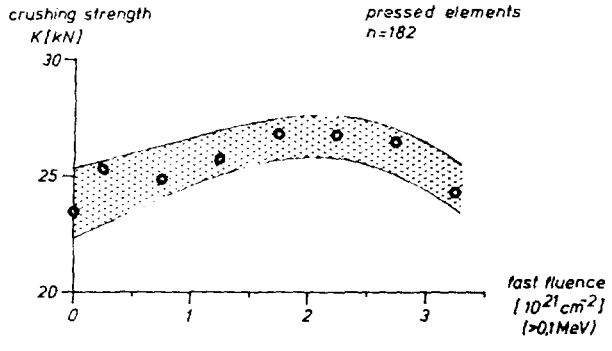

Figure 7:

Crushing Strenght versus Fast Neutron Fluence

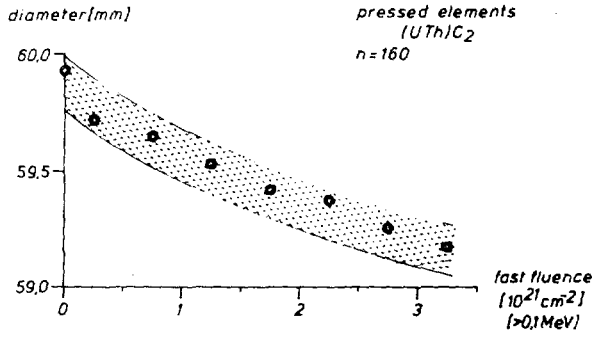

Figure 8:

Sphere Diameter versus Fast Neutron Fluence

The scrap stability of fuel balls, however, varied under influence of the fluence of fast neutrons. It is determined by means of a crushing test in KFA in the frame of the post-irradiation program. Figure 7 shows the results for pressed fuel elements.

Figure 8 gives corresponding results for the variation of dimensions of pressed fuel elements with carbidic fuel.

SUMMARIZING EVALUATION OF THE RESULTS

- Pressed fuel elements with highly enriched $(UTh)C_2$ with Biso coating show up a good retention capability for fission products up to hot-gas temperatures of 900 °C. No particle damages arise up to highest burn-ups. At higher temperatures mainly strontium is released which not only impedes repair works on the primary system, but also lowers corrosion resistance of fuel element graphite.
- Pressed fuel elements with highly enriched $(ThU)O_2$ and Biso coating shop up an excellent behaviour without particle damage, even at hog-gas temperatures of 950 °C. In particular all 15,000 fuel elements introduced since 1974 from the production for 300 MWe THTR confirm all results as expected.
- Special attention is paid to fuel elements with LEU-TRISO particles that are being tested since 1982. Their excellent retention behaviour at hot-gas temperatures up to 950 °C is confirmed by the presently especially low coolant-gas activity. Because of the low burn-up of these elements it is still too early to make conclusive statements.

TEMPORARY AND ULTIMATE STORAGE OF IRRADIATED AVR FUEL ELEMENTS

120,000 burnt fuel elements were removed from AVR up to now and deposited in wet and dry stores in KFA. In future, however, these and the future ones are supposed to be deposited along with those of THTR 300 in cast-iron transportation and storage casks until their transportation to the ultimate store in a salt dome will be possible /5/.

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