

European Programme on High Temperature Reactor Nuclear Physics, Waste and Fuel Cycle Studies

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Abstract: This project represents a collaborative R&D programme in the field of HTR-related nuclear physics, waste and fuel cycle studies (HTR-N) and is funded by the European Commission in the HTR-N and HTR-N1 contracts of the 5th Framework Programme (FP5). The paper illustrates the nuclear physics analyses of first criticality of the Japanese High-Temperature Test Reactor (HTTR) and of the Chinese 10 MWth High-Temperature Reactor (HTR-10). Reasons for discrepancies in the predictions of critical core configuration of HTTR are explained. Further activities deal with different fuel cycles aimed at effective burning of plutonium and minor actinides. First studies, for first generation plutonium (Pu-I) from reprocessed LWR uranium fuel, show that it is possible to reduce the fissile Pu content by up to 90%. Work continues on a symbiotic fuel cycle for LWR and HTR by burning of 2nd generation plutonium (Pu-II) from reprocessed spent LWR MOX fuel in an HTR core still maintaining HTR safety features with regard to nuclear stability and self-acting decay heat removal. Both, block-type and pebble fuel are analysed. Studies on burnable poison and on uncertainties of nuclear data are included into the HTR-N programme. R&D on waste minimization options and experiments and first results on long-term behavior of spent HTR-fuel is reported, too.

I. INTRODUCTION

The European Research and Development (R&D) activities on HTRs concentrate on HTR-related key technologies and innovation potentials with the objective to consolidate and advance modular HTR technology for industrial application in the next decade and to explore new applications like hydrogen production and waste transmutation in the long-term. A collaborative programme on different items like fuel, materials, components, licensing has been established within the European HTR Technology Network (HTR-TN). The HTR-N project and the complementary activities in HTR-N1 deal with High-Temperature Reactor Nuclear Physics, Waste and Fuel cycle studies and include 14 partner organisations.

As the HTR is a promising concept for the next generation of nuclear power reactors and nuclear process heat, the European nuclear community must have analytical tools capable to perform conceptual design studies, industrial calculations (reload calculations and the associated core follow), safety analyses for licensing etc. for new fuel cycles aimed at plutonium and Minor Actinides (MA) transmutation by ultra-high burn-up without multi-reprocessing of the discharged fuel. In addition, it is necessary to identify the HTR-specific waste streams and to find measures for their minimization and treatment. Finally, the disposal characteristics of spent HTR fuel have to be determined with high precision with regard to long-term prediction of the leaching behaviour.

II. CORE AND FUEL CYCLE ANALYSES

Gas-cooled reactors were proposed as early as 1942, the year when the first nuclear pile (CP-1) became critical. Afterwards, some studies were already carried out at ORNL oriented to small HTR units (5-50 MWt). Since more than 40 years, fundamental characteristics of HTR nuclear physics have been investigated and verified in different test and demonstration reactors (DRAGON, AVR, Peach Bottom, FSV, THTR-300) both, for pebble bed and block-type cores [1]. Therefore, a substantial basis for HTR-related core calculations and analytical methods is already available for the High-Enriched Uranium (HEU) fuel cycle which has mainly been applied in these reactors. Especially for economical and proliferation reasons, the Low-Enriched Uranium (LEU) fuel has been chosen for future HTR projects which also differ in the core geometry by annular shaped cross section in contrary to cylindrical cores as formerly applied.

Benchmark calculations and (cold) experiments for (fresh) LEU HTR fuel were undertaken at PSI in the critical facility PROTEUS and can now be continued for the

HTTR and HTR-10 both having a pure LEU core. This provides the unique opportunity to verify existing analytical tools over the full operational temperature range and to determine temperature coefficients and absorber rod worth for the LEU cycle under HTR conditions with an epi-thermal neutron spectrum and increasing burn-up.

In principle, such validation procedures have to be performed for each new fuel cycle over the full burn-up and temperature range especially when introducing innovative approaches for Pu and Minor Actinide (MA) burning. As experiments on such fuel compositions are not yet available for HTR, only code-to-code benchmarks can be undertaken to check the capabilities of the analytical tools, at the time being. Critical experiments in the ASTRA facility on Pu-based fuel are proposed by the Russian Kurchatov Institute under the ISTC programme and would be of utmost importance.

II.A. HTTR analysis

The first benchmark calculations on HTTR dealing with the number of fuel columns necessary to achieve the first (cold) criticality have already been presented e.g. at the HTR 2002 conference. The following tests performed at cold zero power with the HTTR, allows to JAERI [2] to measure isothermal temperature coefficients and control rod efficiency and to propose them as a second phase for the HTTR's start-up core physics benchmark analysis.

As in the first set of the benchmark calculations, similar nuclear data and codes have been used by the European partners in order to estimate the temperature coefficients and the control rod worth. Moreover, the methods that have been developed for this benchmark already have been optimised during the complementary synthesis carried out within the HTR-N contract. They have been used to calculate the temperature coefficient and to model the control rod for evaluating the reactivity worth of the different control rod banks.

As far as first the temperature coefficients are concerned, the results are gathered in the following table. These results correspond to one value of the temperature coefficient at 400 K and are coming from several estimations of the temperature coefficient at different temperature between 280 K and 480 K.

KENO	PANTHER	CRONOS
-14.7 pcm/K	-15.2 pcm/K	-16.2 pcm/K

Table 1: HTTR Temperature coefficients at 400 °K

In order to avoid slowing-down cross section matrix generation at different temperatures no continuous Monte Carlo calculations have been done. Therefore, only multi-

group 3D core calculations have been performed on the basis of diffusion and Monte Carlo transport calculations. Finally, it turns out that the experimental values evaluated from measured control rod positions and calculated control rod worth curve led to a temperature coefficient ranging from -13 to -14 pcm/K. Fig. 1 shows the thermal flux distribution from a CRONOS calculation.

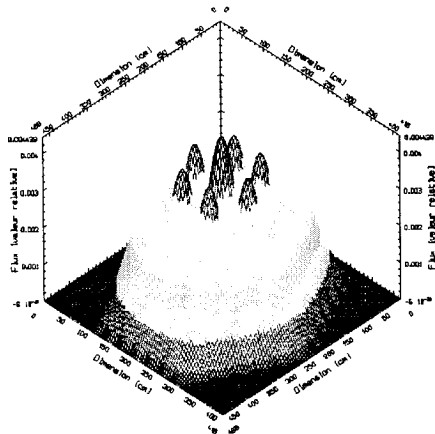


Fig.1. HTR full core thermal flux

Meanwhile, additional investigations have been started in HTR-N1 with regard to the evaluation of the control rod worth in two configurations. First, a configuration corresponding to the reactivity efficiency of the control rod inserted in the reflector only and a second one taking into account all the control rods (core and reflector).

Control position	KENO	TRIPOLI4	CRONOS2	Exp.
178,9 cm	$1.0093 \pm 0,0006$	$1.00117 \pm 0,00024$	1.00020	Crit.
177,6 cm		$0.99972 \pm 0,00038$	0.99840	*
Scram of refl. CRs	$0.9178 \pm 0,0005$	$0,92215 \pm 0,0004$	0.90245	12 %
$\Delta k/k$	9.88 %	8.56 %	10.83 %	± 1.2
Scram of all CRs	$0.6809 \pm 0,0005$	$0.68396 \pm 0,0003$	0.63982	46 %
$\Delta k/k$	47.78 %	46.32 %	56.31 %	± 4.6

Table 2. Analysis of different control rod positions

The results given in table 2 show that quite acceptable values have been obtained with the Monte Carlo calculations on the efficiency of all the control rods inserted in the core and the reflector. These values are comparable to the experimental results. Nevertheless, the diffusion calculations lead to an expected overestimation

of the absorber worth taking into consideration that no equivalence factors have been used in the control rod modelling. This last remark is also true for the estimation of the reactivity worth of the control rod inserted in the reflector. On the contrary, additional investigations need to be carried out for explaining the discrepancies between the Monte Carlo and the experiment when the control rod are only inserted in the reflector.

Taking into account that the control rod inserted in the reflector is a particular characteristic of the HTR core due to the use of pebbles or/and the use of annular core geometry, the HTR benchmark was a good opportunity to initiate a validation of the code and methods with regard to this control rod modelling. The preliminary results indicate that the calculation scheme will have to be improved in the future as far as the control rod modelling is concerned in the diffusion calculations.

II.B. HTR-10 Analysis

Another benchmark available is the Chinese HTR-10 at the INET in Beijing. The HTR-N partners have calculated this pebble bed reactor with their respective code systems as mentioned for the HTR. For the initial loading, the HTR-10 has been loaded with a mixture of fuel and graphite pebbles (57:43% ratio). The cone region at the core bottom is filled with dummy graphite pebbles, only. As shown in figure 2, this core region has been modeled in the Monte-Carlo code TRIPOLI4 by an homogeneous medium.

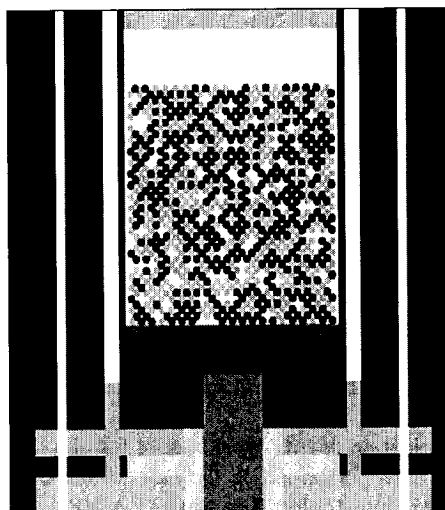


Fig.2. HTR -10 core geometry

For the initial critical core height, the results for the different codes can be summarized as follows:

Critical height in cm	Defined bench.	Revised bench.
Diffusion with VSOP(2D)	124.2 cm	121 cm
Diffusion with VSOP(3D)	126.8 cm	123.3 cm
Diffusion with PANTHER	125.3cm	122.1 cm
Monte Carlo with TRIPOLI		117.4 cm

Table 3. HTR-10 benchmark results

The experimental core level was at 123.1 cm, but the experimental conditions such as core temperature and atmosphere, graphite impurities and densities, deviated from the stated benchmark values. Revised benchmark data have then been proposed. This leads to a good agreement for the diffusion codes with the experimental data. Besides, the Monte Carlo results came from simultaneous point-wise and multigroup cross section calculations in which the pebble bed is explicitly described. As a first step, the self-shielded multigroup cross sections are from a previous transport calculation (172 groups). Additional works are in progress for modelling the coated fuel particles in the Monte Carlo geometry in order to quantify the impact of this modelling related assumption. Furthermore, in the configuration close to the criticality, the Monte-Carlo calculations gave an increase of the reactivity of 0,27% per cm of pebbles loaded in the core.

For the scram reactivity at 20 °C, calculations were done for the ten reflector control rods fully inserted at the critical core height. Calculated values were in a range from 13.61 to 19.31 (%) which still have to be compared to the experiment.

The integral rod worth for one rod has been measured as 1.44 % and has been calculated as the values between 1.31 to 1.48 %. Neutron streaming in the empty control rod holes are a major reason for the deviations.

More detailed analysis have been performed on the asymmetric flux distributions when only one reflector rod was inserted.

II.C. Fuel Cycle Studies for Block-Type Cores

The investigation of fuel cycle studies for block type HTR cores has been performed on the basis of the Gas Turbine Modular Helium-cooled Reactor (GT-MHR) concept [3] for 1st generation plutonium-based fuel cycles. The core depletion calculations are essentially based on a representation in infinite medium of the fuel element treated with the P_{ij} transport method. For the plutonium fuel, preliminary investigations showed that:

- a) the fuel cycle length increases linearly with the mass of plutonium loaded into the core,
- b) there is an optimum for the fuel fed into the core with respect to the discharge burnup which allows best use of plutonium.

It should be noticed that the plutonium balance reaches an optimum with respect to the mass of plutonium loaded into the core. In this study, it has been shown that this optimum is approximately achieved with a total mass of plutonium loaded into the core of 1500 kg. Furthermore, the minor actinides balance increases linearly with the mass of plutonium.

Despite the fact that the moderator temperature coefficient becomes less negative during fuel irradiation, the results have shown that the global core temperature effect keeps negative enough for self-stabilizing, with a fuel management by 1/3 where the average core burnup ranges roughly from 200 and 400 GWd/t between the beginning and the end of cycle.

Further analysis has been performed on the deep-burn HTR-Transmuter approach [4]. In this concept, the plutonium and the minor actinides are located in separate driver and transmutation fuel particles. The concept is based on an optimisation of the physical parameters such as the volumetric packing fraction of the fuel particles and the size of the kernels in order to ensure a better use of the material (fissile and non-fissile). When the equilibrium cycle is reached, all the plutonium discharged from the driver fuel is mixed with the minor actinides coming from the LWR fuel and loaded in the transmuter fuel for an other irradiation. Preliminary results show that the kernels containing plutonium (driver particles) must be as large as possible in order to ensure a better utilisation of the fissile isotopes. Furthermore, there is an optimum with respect to the mass of plutonium loaded into the core. On the opposite, in order to enhance the transmutation of the minor actinides in the transmuter particles, it is necessary to favour the resonances absorptions in the heavy nuclides. This is obtained by reducing the particle size and by increasing the mass of minor actinides loaded into the core. By reducing the size of the kernel, it allows obtaining fast destruction rates. On the other hand, the increase of the mass of minor actinides leads to a spectrum hardening that favour the resonance absorptions.. After all, it has been shown that it was possible to achieve a consumption rate of approximately 65 % of plutonium in driver particles and 35 % of heavy nuclides in transmuter particles in a once-through mode.

A batch-wise fuel load scheme in HTRs has to be combined with burnable poison in a heterogeneous way by mixing burnable particles (small spherical particles made of burnable poison) in the fuel elements [5]. By varying the diameter of the burnable particles and the

number of burnable particles per fuel pebble, it is possible to tailor the reactivity-to-time curve. Further studies will focus on stronger neutron absorbers such as Gd, and on the application of burnable particles for plutonium fuel.

As a very innovative alternative, thin fissile layers have been studied which act as flux converters to generate regions with fast fluxes inside a thermal reactor and thereby improve their incineration capabilities. Block-type HTRs seem to offer good conditions for designing fast islands, and that the typical incineration rates in fast islands are 2-3 times higher than the corresponding rates in thermal reactors.

II.D. Fuel Cycle Studies for Pebble-Bed Cores

NRG has implemented the reference pebble bed reactor ("HTR-MODUL") in their PANTHERMIX code system and has performed some initial studies on the OTTO (Once Through Then Out) loading scheme with UO_2 -fuel (7.8% enriched) and 1st generation (pure) PuO_2 , with 7 grams per pebble of initial heavy metal mass. It was concluded that in the equilibrium state, after 2000 days of operation, 415 (fresh) pebbles are needed per day to maintain criticality. In this state, the maximum power density (i.e. the peak value) in the core is 11.84 MW/m³, the maximum burnup in the core is 77.5 MWd/kg and the maximum (fuel) temperature in the core is 1072.5 K during steady-state operation.

FZJ has concluded a study on continuous reload pebble bed reactors loaded with a mixture of (1st generation) PuO_2 and (U-Th) O_2 , comparing so-called "high Pu burning" and "low residual Pu" scenarios. It was found that the reduction of the average discharge burnup increases the Pu incineration rate but also increases the fraction of residual Pu. Some results for the mass balances for the 2 strategies have been presented in ref [6]. A similar study is being conducted for 2nd generation Pu.

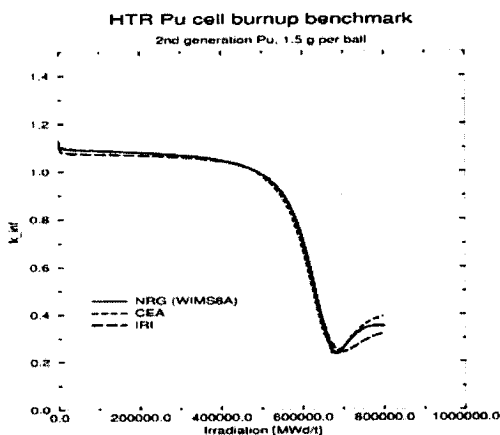


Fig. 3. Results from HTR Pu cell burnup benchmark

As the studies of Pu burning in HTR systems are aiming at very high (>740,000 MWd/t) discharge burnup, additional validation is desirable for the different reactor physics code packages used by the different project partners. Therefore, a HTR Pu cell burnup benchmark has been defined to intercompare the predictive capabilities of these code systems. The benchmark geometry is a standard 6 cm diameter HTR fuel pebble, containing TRISO coated particles in its 5 cm diameter fuel zone. The boundary condition of the outer surface of the pebble is reflective ("white"), hereby effectively establishing an infinite array of pebbles. For simplicity in this set-up no leakage is modeled so that the benchmark is not entirely representative for a full scale pebble bed HTR. However the benchmark is expected to be quite suitable for its intended purpose, viz. intercomparison of the codes and nuclear data used. The diameter of the PuO_2 kernels of the particles is 0.24 mm. The number of coated particles in the pebble is determined by the desired initial amount (grams) of PuO_2 . Three cases have been defined: (1) 1st generation Pu, 1 g per pebble, (2) 1st generation Pu (alternate composition), 1 g per pebble, and (3) 2nd generation Pu, 1.5 g per pebble. The power generated in the pebble is to be kept constant at 1 kW during the burnup from 0 to 800,000 MWd/t. A set of parameters for intercomparison has been defined, to be calculated at designated values of the burnup. The results presently available are very promising, as is indicated in Fig. 3. For the 2nd generation Pu case k_{∞} , as calculated by CEA, NRG (WIMS8A code) and IRI is shown as function of the burnup. Up to about 550 000 MWd/kg the results are quite close together. At higher burnup some deviation occurs, which may be due to differences in the burnup step sizes applied in the calculations. Also note the increase of k_{∞} after about 680 MWd/kg, which is connected to the increase in the amount of higher actinides at that stage of the burnup. UNIFI is performing similar calculation in a different way, using codes coupling MCNP with burnup codes, obtaining comparable results. Further analysis and intercomparison is in progress.

Further studies were performed by IKE for multiple number of fuel element passes through the pebble bed core with continuous reload of fuel elements and various core designs. Cylindrical core design as well as annular cores with graphite pebbles in central column and a solid inner column was considered. The core size, the power, the uranium load and the number of fuel element passes were varied and the corresponding equilibrium state and afterheat distribution for accident investigations were calculated by means of the ZIRKUS system originally developed from FRAMATOME-ANP. The maximum fuel temperature after a loss of coolant accident (DLOCA) was then determined for all variations. The results show the potential for rising the total thermal power using an

annular core design regarding the feature that the maximum fuel temperature after DLOCA remains below 1620 °C without further safety, feasibility and economical constraints.

II.E. Nuclear Data Uncertainties

The calculations of important reactor reactivity parameters and fuel composition as a function of burn-up may be biased due to uncertainties of the available nuclear data. The impact of the uncertainties in the basic data on the calculated quantities depends from type of fuel and from the moderation conditions. Therefore, by means of comparisons of calculated and experimental parameters of benchmarks and sensitivity analyses the quality of currently available nuclear data with reference on HTR specific systems was proven. Additionally, comparisons of calculated parameters for HTR systems performed by Monte Carlo calculations based on the evaluated nuclear data files JEF-2.2 (JEFF-3), ENDF/B-VI and JENDL-3.2 were performed, to investigate the influence of today's most actual nuclear data on criticality. The results of these comparisons showed sufficient good agreement for JEF-2.2 and JENDL-3.2 based calculations for a large variety of moderation ratios. ENDF/B-VI (release 5) based calculations for HTR with UOX fuel showed a substantial under-estimation for low moderation ratios as it is the case for low moderated LWR systems with UOX fuel. Furthermore, sensitivity studies connected with uncertainty analysis were made for the GTMHR with RG-Pu and WG-Pu fuel for zero and high burn-up. The variation of the capture and fission cross sections of Pu isotopes show the largest sensitivity coefficients with reference to reactivity corresponding to the number densities at zero and high burn-up. The influence of minor actinides is lower than 15 pcm/% for RG-Pu and about 5 pcm/% for WG Pu. Uncertainties in multiplication factor due to the calculated sensitivities and covariance data for the regarded heavy nuclides are estimated to about 0.6 % for WG-Pu fuel (both for low and high burn-up). A need for actualised covariance data was found since there are rather few reliable processed data available for thermal systems. Furthermore, the scattering law data and thermal neutron spectra in graphite were investigated based on newer publications [7] since the present available data for thermal neutron scattering in graphite were not actualised for a considerable long time. A comparison of a measured [8] and calculated thermal neutron spectrum in graphite is shown in Fig. 4.

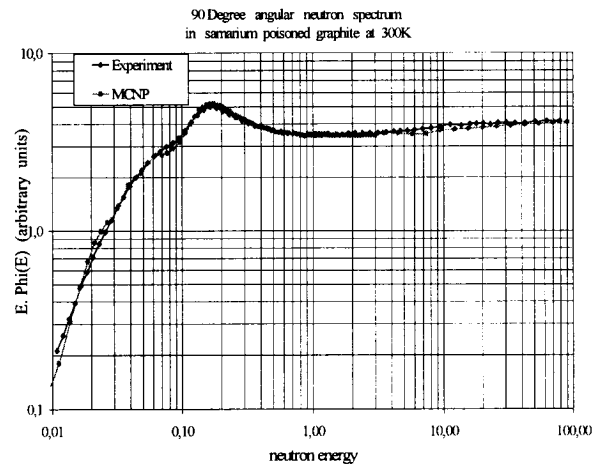


Fig. 4. Measured and calculated spectrum in graphite

II.F Symbiotic LWR / HTR Fuel Cycles

HTRs can accommodate a wide variety of mixtures of fissile and fertile materials without any significant modification of the core design. This flexibility is due to an uncoupling of cooling geometry and the neutronic parameters (moderation ratio or heavy nuclide concentration and distribution). Moreover, thanks to the high resistance of the coated particles, HTR fuels are able to reach very high burn-ups, far beyond the possibilities offered by other fuels (except molten salt reactors) and have excellent long-term disposal behaviour. These features make HTRs especially interesting for closing the nuclear fuel cycle and reducing the plutonium inventory.

Plutonium consumption at high burnups in HTRs has already been tested with encouraging results during the DRAGON project and at Peach Bottom. To maximize plutonium consumption, core studies are being performed on plutonium HTR cores, with emphasis on weapon-grade plutonium and first/second generation civil plutonium consumption. These calculations show that very high burn-up can be achieved from the HTR-specific neutronics standpoint. A representative value of 500 Gwd/t has been used in a recent prospective fuel cycle scenario study for the case of a mixed fleet of LWRs and Pu-burning HTR. First results (see fig. 5) confirm that such a symbiotic reactor fleet involving LWRs and Pu-burning HTRs would be a very effective way to stabilize and slowly decrease the plutonium inventory while providing energy in safe and economical conditions.

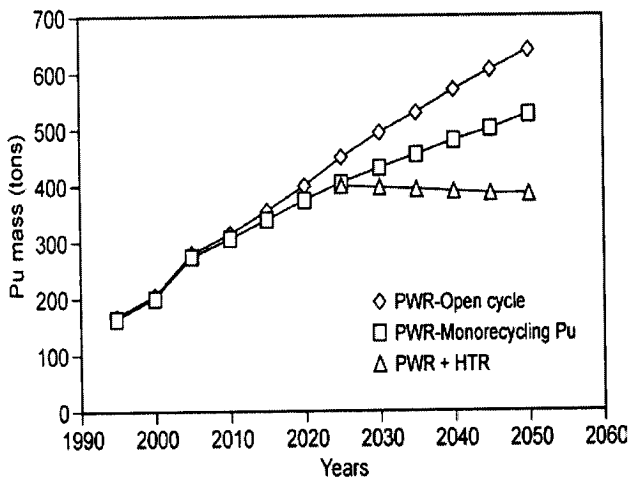


Fig.5. Pu stockpiles for different fuel cycle

III. HTR SPECIFIC WASTES

The objectives of the first phase of the work was to develop an assessment of the wastes arising from HTRs. The purpose was then to predict quantities arising from future designs, compare with waste quantities from operating LWRs, identify where novel features (e.g. direct cycle power units) might give rise to new wastes, and to determine ways in which the waste quantities and treatments can be minimized.

The objectives of the second phase of the work were to review existing methods for the treatment of operational wastes and non carbonaceous decommissioning wastes. Graphite and carbonaceous wastes have long been recognised as a key issue for graphite reactors. Many solutions have been proposed and researched for the disposal of such wastes. However, none of the solutions so far practiced has led to successful decontamination or volume reduction. The experimental objective of the second phase is therefore to determine whether irradiated graphite can be purified by basic thermal and chemical methods.

The objectives of the first phase were obtained by a combination of historical research and analysis. In the case of shutdown HTR reactors in the world, AVR and THTR in Germany, DRAGON in UK, Fort St. Vrain and Peach Bottom in USA, extensive archive searches were put in hand. Where possible, discharges, operating wastes, and decommissioning wastes were extracted and quantified. Good results were obtained for AVR, THTR and DRAGON. However, quantitative results were difficult to obtain for FSV and Peach Bottom. Where quantitative information was available, it was nearly

always expressed in ways that made comparison difficult (differences include, wastes identified on the one hand immediately after shutdown, on the other hand after considerable decay periods, raw wastes versus conditioned waste quantities, masses versus volumes, different waste category definitions for different countries). Nevertheless, simplifying assumptions were used to table as far as possible the wastes from the reactors, and to relate waste quantities to lifetime power output. As a general observation, it was noted that, larger reactors produce proportionally less decommissioning wastes, the longer a reactor operates, the less the amount of decommissioning wastes per output. Operational wastes tend to be proportional to life time operation. Activation analysis was carried out to determine the quantities of spent fuel and operational graphite wastes and decommissioning graphite wastes. The spent fuel wastes were compared, on a unit power output basis, to LWR spent fuels, calculated by previous activation analysis. The results showed spent fuel quantities in favour of HTRs, in some cases by a factor of nearly 2. This increase to a factor of nearly 10 if the graphite surrounds can be satisfactorily separated from the fuel compacts. The report of the work has been completed in draft form. including recommendations for waste minimisation

For the second phase of the work draft reports have been written for review of methods of treatment for operational wastes and treatment of non carbonaceous decommissioning wastes. On the experimental front, equipment at the Juelich research centre has been modified to allow firstly inert thermal treatment of graphite, and secondly pyrolysis of the same graphite. The aim of the former treatment is decontaminate the graphite of the volatiles, e.g. tritium and ^{36}Cl , the aim of the latter is to remove heavy nuclides. The purified graphite which still contains ^{14}C will then be easier to dispose and may even be recycled within the nuclear industry Test programmes been worked out, irradiated graphite samples obtained from the AVR reactor, and first results are being obtained.

IV. LONG-TERM BEHAVIOUR OF DISPOSED SPENT HTR FUEL

Although the behaviour of spent HTR fuel elements in a salt environment has been experimentally investigated for a long time, there is still a considerable lack of reliable experimental data. This is especially true for alternate disposal sites in granite and clay. It can be assumed that no measurable amount of radionuclides will be released from the fuel kernels as long as the coating layers are intact. As only a very small fraction of the coated particles might fail dependent on burn-up, the source term of HTR spent fuel elements will be controlled by the retention

capabilities of the intact particles and their resistance to chemical or mechanical interaction, in the long-term. Fuel kernel and coating dissolution as well as the interaction of the radionuclides with the graphite matrix become the dominant factor. Previous experiments have shown some indications for graphite corrosion that could not be reliably quantified due to the slow reaction.

For allowing long-term predictions, the disposal related R&D in this project finally aims at establishing a geological model and the precise determination of the fundamental reaction parameters in aquatic phase for different disposal sites (salt, granite, clay):

- a) Graphite matrix corrosion rate
- b) Corrosion and lifetime of pyrocarbon and SiC coatings
- c) Fuel kernel dissolution (uranium and thorium-based kernels)

Former performance assessment was mainly governed by the rough resolution of the experimental measurements. The former detection limit of $\sim 4 \mu\text{m/a}$ for corrosion rates of pyrocarbon and silicon carbide coatings e.g. did not allow reliable life time estimations for more than 500 years which are much too conservative.

New corrosion test rigs with autoclaves on different temperature levels have been taken into operation at FZJ. First results for corrosion of graphite powder in the specified brines are shown in fig. 6. Initial corrosion rates fall down to the order of $10^{-8} \text{ g/m}^2\text{day}$ which leads to an expected lifetime of $6 \cdot 10^8$ years. Irradiation effects are simulated by adding H_2O_2 .

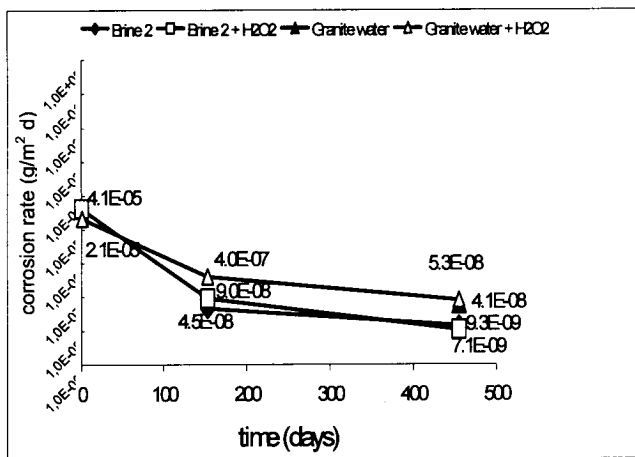


Fig. 6. Graphite oxidation rate in autoclaves (90 °C)

V. CONCLUSIONS

Recent HTR designs take benefit of new core configurations that limit the maximum temperatures in core heat-up accidents without endangering the fission product retention capabilities of the coated-particle fuel. Such an annular core geometry has been realised in the Japanese HTTR for the first time. In addition, this reactor makes use of Low-Enriched Uranium (LEU) fuel which is common for the actual commercial HTR projects but was not applied in former HTR. The pre-calculations of the first criticality showed significant discrepancies that have now been overcome by more precise modelling of the complex geometry of the HTTR fuel element and of streaming effects. The analyses of other HTTR core configurations as well as of first criticality of the HTR-10 pebble bed LEU core show good conformity with experimental data.

The prediction of reactor reactivity behaviour and fuel composition as a function of burn-up largely depends on the quality of available nuclear data. Sensitivity and uncertainty analysis showed large under-estimation for low moderation ratios and that the capture and fission of Pu isotopes has the main influence on reactivity. This means that the isotopes with high sensitivities have comparable small standard deviations. The less well-known minor actinides cross sections have a lower influence on reactivity.

The high burn-up potential of HTR fuel can be used for actinides burning and minimisation of long-lived radiotoxides. Different fuel strategies are analysed also with regard to core physics and nuclear stability. First results show that both types of HTR cores using pebble or block-type fuel can be operated with first generation plutonium from reprocessed LWR uranium fuel. The overall temperature coefficient keeps negative enough to guarantee self-stabilisation of the core in reactivity excursions. The future work will be re-oriented towards second generation plutonium extracted from reprocessed LWR MOX fuel. A symbiosis of LWR and HTR will be capable to reduce the plutonium stockpiles instead of the actual steady increase.

HTR-specific waste streams are quantified to identify waste reduction potentials. The analyses indicate that HTR generate less high level waste due to the ultra-high burn-up, high conversion efficiencies and clean primary circuit. Special investigations will be continued on the management, treatment and disposal or recycling of the discharged contaminated graphite from fuel and reflector structures.

The long-term resistance of directly disposed HTR-fuel is investigated for different geologic conditions. Autoclaves for integral and separate-effect leaching tests are now in

operation providing more exact data for long-term prediction of the behaviour of HTR fuel in a final repository. First experiments on the corrosion of silicon carbide result in a much higher resistance than glass used for vitrified waste. It is expected that spent coated particle fuel is capable for direct disposal.

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