HTR-N

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Document Title

HTR-N

Technical Summary Report of WP4-Specific Wastes

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1 Abstract

One objective of work package 4 within the European Commission co-funded HTR-N project /1/ was to summarize the waste streams arising from HTR reactors and to compare with those wastes from the operation of LWRs. Two different aspects have been considered, i.e. the waste generated during operation and that generated during decommissioning. Provisional data for operational waste had been obtained from the German THTR 300. Decommissioning data were partially available from the American Fort St. Vrain reactor and from decommissioning planning for the German AVR reactor. A general conclusion from these data is, that there are only small differences between HTR and LWR operational wastes, with the exception of waste arising from spent fuel due to the high burn up of HTR fuel. For decommissioning waste, the main difference is the graphite from the reactor core, which poses specific challenges. These challenges are essentially the same as those for all other graphite moderated reactors. Therefore, advanced graphite treatment methods should be developed because the existing approaches and concepts are not optimised with respect to the produced volume and disposal features. Additional decontamination methods (e.g. thermal/chemical treatments) may help to reduce the radiotoxicity of radioactive graphite waste and may even allow recycling for nuclear use. Taking benefit from the analyses on the origins of the waste streams, future HTR designs can be much improved and may even contribute to waste minimisation strategies, in symbiosis with LWR by making use of the high burn-up capabilities.

2 Introduction

HTR differ considerably from LWR with regard to the core design, the fission product retention features of the coated-particle fuel and the ceramic materials used in the core. Therefore, a direct comparison of wastes from HTR and LWR is not possible, in a general sense, but only for some areas as done in this paper. Nevertheless, it provides some indications on specific waste features of HTR and LWR as well as motivations to extend R&D on HTR-related waste issues.

The coolant activities in an HTR core are rather low and also result in a very low dose for the operational staff. This fact has been demonstrated by all HTRs operated, so far. None the less, it is considered necessary to quantify gaseous, liquid and solid discharges from HTR and to find the reasons for their generation as well as options to minimise operational and decommissioning waste. Minimisation and treatment of specific waste streams like contaminated graphite from discharged spent fuel and reflector structures appear as a fundamental challenge for HTR and a potential hurdle for market introduction if not solved satisfactorily.

It has to be mentioned that it turned out to be extremely difficult to retrieve data on operational and decommissioning waste from former HTR projects. Thus, the analysis could only be based on fragmented information on the operational results and some localised measurements in the frame of decommissioning or safe enclosure of those reactors. In addition it has to be kept in mind, that former HTR were very different in design, fuel quality, fuel composition (HEU, Thorium, LEU, Plutonium) as well as in power size and operational periods. New information on HTTR and HTR-10 is not available yet and is of the utmost importance to improve the database. For example, measurements of contamination at AVR compared to prediction differ by a factor of 2-3. Thus, the validation of the waste-related data for HTR is still rather weak and by no means as consistent as it is for LWR.

3 Tasks and Deliverables Within WP4

This Work package examines the waste arising from the whole lifetime of a future HTR project. Information was obtained from existing shutdown HTR and also from assessments of future HTR designs.

3.1 Description of the Tasks

A literature search was undertaken to characterise and quantify the waste arising during operations, maintenance, refuelling and decommissioning. The literature search has been archived and is available for future research.

Analysis by the CEA has characterised and quantified the spent fuel and the activated graphite arising from the whole life of a projected GT-MHR design of block type HTR.

From information sourced from the above the following tasks have been undertaken within WP4:

- Task 4.1 is an assessment of wastes arising from HTR operations, including fuel manufacture.
- Task 4.2 is an assessment of wastes arising from HTR decommissioning, including the potential effects of Wigner energy on stored irradiated graphite.
- Task 4.3 is a comparison of HTR wastes with LWR wastes. The quantities of waste are compared on the basis of the same generated output.
- Task 4.4 is an assessment of how HTR wastes can be minimised.

All the above work scheduled for WP4 of the HTR-N contract has been carried out and the findings are summarised below. Section 4 presents the operational waste findings (Task 4.1) and Section 5 presents the decommissioning waste findings (Task 4.2). Section 6 presents an estimation of graphite waste based on analysis. Graphite wastes arise in HTR as an integral part of the fuel assemblies or pebbles and also from the decommissioning of permanent graphite structures such as reflectors. Thus, Section 6 relates to both Task 4.1 and Task 4.2. The Task 4.3 results are not presented separately but are used in Sections 4 and 5 to put the findings of Tasks 4.1 and 4.2 into perspective. Section 7 presents the key findings for HTR waste minimisation (Task 4.4).

3.2 Deliverables and Support Documents

All the documents available on the SINTER Website that have been produced to complete WP4 are listed in Table 1:

Table 1: Deliverables and Documents of HTR-N, WP4

Deliverable	Title	Document No
4.0.0	Summary Report	HTR-N-04/11-D-4.0.4
4.1.1	Archive Literature Review	HTR-N-02/06-D-4.1.1
4.1.2	Data of Operational Waste	HTR-N-10/03-D-4.0.3
4.2.1	Data of Decommissioning Waste	HTR-N-10/03-D-4.0.3
4.3.1	Comparison with LWR Waste	HTR-N-10/03-D-4.0.3
4.4.1	Waste Minimisation	HTR-N-10/03-D-4.0.3

4 Operational Wastes (Task 4.1)

The Task 4.1 findings are categorised separately below in terms of: fuel manufacture waste, operational waste (except spent fuel) and spent fuel.

4.1 Waste arising from fuel manufacturing

Reliable data on waste arising from fuel manufacturing were only made available for the pilot fabrication of LEU TRISO coated particles in graphite fuel pebbles from NUKEM, formerly HOBEG /2/. This fuel had been proposed for the HTR-Module from Siemens. Data for the production of the THTR fuel elements had not be supplied to the HTR-N project and is not that representative anyway as it was based on BISO particles, Thorium and HEU which is no longer foreseen for future HTR projects, although this cycle produces much less actinides than any other option (see figure 2).

The annual production capacity was 300,000 fuel pebbles. Each fuel pebble contains 6.44 g UO2 with an enrichment of 7.3 %. Assuming a burnup of 100 GWd/t HM and an efficiency factor of 0.44 for the HTR-Module, an energy of 233 MW_ey could be achieved from this annual manufacture capability. The waste amount associated with the production of 300,000 fuel pebbles is given in Table 2.

Table 2: Waste arising from the production of LEU/TRISO HTR fuel elements (FE)

	HTR		LWR
	300000 FE [m³]	1 GW_ey [m³]	1 GW_ey [m³]
	Solid waste		
Compactable	32	139.0	12.7
Compaction Ratio 1:10	*3.2*	*13.9*	
Non Compactable	5	21.7	
	Liquid		
Low activity	2000	8690.5	229

Based upon 1 GWy electrical energy produced, the amounts of solid wastes are somewhat higher from the manufacturing of HTR fuel pebbles than from manufacturing of LWR /3/ fuel. However, this can be neglected in comparison to liquid wastes, that are 30 times larger for HTR than for LWR fuel manufacturing. But it must be taken into account, that these data are derived for a prototype process, which had not been optimised with respect to waste stream, whereas LWR fuel production is a well-established industrial process. Furthermore, the HTR waste predictions are based on waste treatment processes from the 1980's, and will probably be reduced with newer treatment technologies and large scale production. Therefore, future R&D on HTR fuel development should include efforts to reduce the waste streams during fuel manufacture.

4.2 Operational waste (except spent fuel)

Data about operational waste were obtained from the German demonstration THTR in Hamm-Uentrop. The reactor had been operated for a short period, from September 1985 until September 1988. It was loaded with 3.9 metric tons of heavy metal in about 358,000 fuel elements plus 273,000 graphite balls and 44,000 absorber pebbles. Each fuel element contained 0.96 g ^{235}U with an enrichment of 93% and 10.2 g ^{232}Th. The target burn up of 110 GWd/tHM had not been achieved, on average. The efficiency of the electricity conversion is assumed 0.39 for the following calculations. During 427 full operation day the THTR produced a net electrical energy of 330 MWy. which is comparable to modern modular HTR designs. For this reason, THTR has been taken as a reference and not AVR, which had a power output of only 15 MW_e. Unfortunately, similar data from the 330 MW_e FSV was not available. FSV was operated for a longer time than THTR and could provide additional valuable information.

4.2.1 Gaseous discharge

The following Table 3 shows the licensed discharge limits and the measured discharge for exhaust air during the operational period of the German THTR /2/. The emitted activity was mostly 2 orders of magnitude or more lower than the accepted discharge limits.

This activity has been summarized and normalized to a energy of 1 GW_ey in Table 4 in order to assume the same unit power output for THTR and then compare the data with the exhaust air discharge of equivalent 1000 MW_ey LWRs /4/. These data for LWRs had been chosen from 1983 in order to have data with an off gas technology of the 1980's.

The calculations on THTR are only a rough estimation. More robust quantifications would need additional data from THTR operation. Due to the conservative assumptions, any calculation with more precise operation data would lead to a further decrease of the activity per GW_ey because the data include emissions from non-operating days . However, the obtained data are sufficient for the following judgements:

- The THTR had a significant lower release rate for noble gases and ^{131}I than LWR.
- The amount of aerosols and ^{14}C are in the same order of magnitude than LWR.
- Only the release of tritium is much higher than LWR.

Table 3: Exhaust air discharge of the German THTR from 1985 till 1988

	Noble gas	Aerosols	I-131	C-14	H-3
Licensed discharge limits			[Bq/y]		
	6.70E+14	3.70E+08	3.70E+08	1.85E+12	8.10E+12
Measured discharge in			Bq		
1985	2.85E+11	4.76E+06	1.46E+07	2.70E+10	8.40E+09
1986	3.52E+11	1.58E+08	5.83E+06	4.00E+10	1.30E+12
1987	1.32E+11	4.13E+07	1.51E+07	4.60E+10	5.00E+12
1988	2.73E+11	8.97E+07	1.09E+07	2.70E+10	3.80E+12
Total	1.04E+12	2.94E+08	4.64E+07	1.40E+11	1.01E+13

Table 4: Comparison of the exhaust air discharge between THTR and LWR

	Noble gas	Aerosols	I-131	C-14	H-3
			Bq/GW_ey		
THTR	3.16E+12	8.90E+08	1.41E+08	4.24E+11	3.06E+13
BWR	1.18E+14	1.85E+10	3.70E+09	3.70E+11	1.11E+12
PWR	1.18E+14	7.40E+09	1.85E+09	3.70E+11	7.40E+11

The low releases of noble gas and iodine is a clear proof of the retention capabilities of coated particles, despite the fact that a large number of fuel pebbles had been broken by multiple injection of absorber rods into the core. Obviously this did not affect the integrity of the coated particles as such. Concerning aerosol discharge, it has to be taken into account that THTR had to undergo difficult modifications and repair work (fuel discharge, hot gas ducts and inspections) with the primary circuit open which caused some additional aerosol release not typical for an established HTR line. The ^{14}C generation could have been much reduced if the nitrogen content in the primary circuit and in the fresh fuel had been minimised. The rather high tritium discharge is a result of lithium and boron impurities in the reflector structures. This has also been observed for AVR where graphite reflectors contained about 2ppm lithium and boron, each, and some ten ppm in the carbon brick surrounding the reflector. These impurities should be reduced much more in future HTR to attain lower tritium releases. Free boron in absorbers should also be avoided e.g. by the use of coated boron particles.

4.2.2 Liquid waste

The activity of waste water arising during the operation of the THTR has been summarized and normalized to 1 GW_ey in the same manner as it has been done for the exhaust air activity. The obtained data are given in Table 5. The production of tritium is about one order of magnitude higher in comparison to a PWR, from reasons as discussed above. However the production of all other nuclides is 3 orders of magnitude lower compared to BWRs and PWRs. It can be assumed that tritium production will gradually decrease as the burning of initial impurities continues during extended operation.

Table 5: Activity of annual waste water arising from THTR, BWR and LWR per GW_e

	H-3	Other
	Bq/GW_ey	
THTR	2.06E+14	8.18E+07
BWR	5.55E+12	3.15E+10
PWR	3.33E+13	2.78E+10

4.2.3 Solid waste

Table 6 shows the THTR waste amounts, except the spent fuel. These data are a summary of all waste amounts from the THTR beginnings until the year 2000. This period includes waste from test runs before operation as well as waste arising after the shut down where waste from safe enclosure of the reactor is included. Detailed data from the operational period have not been obtained, yet. Therefore, it is not possible to compare it with the waste streams of LWRs.

However it can be noticed that a LWR produces about 10^{14} up to 10^{15} Bq/GW_e waste with volumes of about 350 up to 450 m³ /5/, per year.

Table 6: Total amounts of solid waste of the THTR 300 until the year 2000

Total waste amounts of the THTR until 2000	Mass Mg	Activity Bq
Metal	1.9E+02	3.6E+11
Non-metallic without graphite	2.2E+01	4.6E+13
Graphite	1.2E+02	4.2E+14
Organic	1.7E+01	1.0E+11
Unsorted	6.0E-01	1.3E+07
Sludge	1.9E+03	4.6E+13
Total	2.2E+03	5.2E+14

It is clearly seen that the total amount of solid waste is mainly governed by contaminated graphite which adsorbs most of the activity, in the primary circuit. Better fuel like TRISO particles and lower impurities will improve the situation for future projects but the necessity for specific treatment of graphite waste will remain.

4.3 Spent fuel

The mass and the corresponding volume of discharged spent fuel have been determined for the following HTR designs: THTR 300, Siemens HTR-Module and GT-MHR (with PuO_2 as fuel) to be compared with spent fuel amounts of LWRs. However, it has to be noted that all of these reactors are based on different fuel cyles (HEU, LEU, Pu).

The calculations for the THTR are based on the operational data of the THTR 300, which are already illustrated in section 4.2. The burn-up data had been calculated at FZJ with the code Origen-S 2 developed formerly by HRB GmbH /6/.

Reliable data can also be taken from the HTR-Module design, which was a completely designed and site-independent licensed high temperature pebble bed reactor with a thermal power of 200 MW. An efficiency of 40% is assumed in the design of KWU-Interatom /7/ for the electrical conversion. The reactor was designed for 360,000 fuel pebbles and each fuel pebble contains 7 g uranium as UO_2 with an enrichment of 8%. Burn-up calculation performed done using the code Origen-Juel-II /8/.

Activation analysis was performed by CEA to characterize the spent fuel and graphite waste arising from HTR operation and decommissioning. This evaluation was based on the block type GT-MHR designed for Pu-burning. The fuel was assumed to be weapons grade Pu in the form of PuO_2 kernels of 200µm diameter, grouped within graphite compacts that are inserted into hexagonal graphite fuel assemblies. The targeted burn-up is 640 GWd/tHM, which equates to 770 irradiation days. The graphite and spent fuel analyses are considered separately /9/.

The main radionuclides, in the form of fission products and transuranic elements, were identified from spent fuel activities calculated over an entire fuel cycle. The masses of the main long-lived fission products and transuranic elements predicted for the three different HTR are compared with those determined for LWRs operating with burn-ups of 33 GWd/tHM and 45 GWd/tHM. The comparison is based on the same generated electrical output. The results are presented in Table 7 and Fig. 1 for the long-lived fission products. Differences in the ^{14}C levels between THTR and HTR-Module / GT-MHR are due to more favourable nitrogen content assumptions as compared to the experiences at THTR. The data for the actinides are shown in Fig. 2. In most cases, the amounts of a particular radionuclides within the spent fuel are of a similar order of magnitude for the different reactor designs. There are some differences between LWR and HTR nuclides generation per GW_ey that can be attributed to:

- The calculation method, for example, the activation products are not taken into account for the GT-MHR system (e.g. Nb-94).
- The GT-MHR is loaded with PuO_2 fuel and the THTR with high enriched $(Th,U)O_2$, whereas the LWR system uses low enriched UO_2 fuel. Therefore the ratios dealing with the actinides generated in the two systems present significant differences and the normalized ratios and nuclide comparisons have to be considered cautiously. More realistic comparisons should also include MOX fuel for LWR.
- The actinide data for THTR did not include ^{242}Am, ^{243}Cm and $^{245-248}Cm$.
- Higher thermal efficiencies of HTR which generally result in lower waste generation.

Therefore, the data has to be validated with further calculations and analyses of spent fuel from the GT-MHR and HTR-Module. Additional data for the THTR has not been obtained yet but will be available in 2005. However, Fig. 2 indicates that the spent fuel of HTRs, especially the THTR, will have less minor actinides in the long term, which is reasonable and due to the use of HEU and Thorium as fertile material. Thus, Pu-burners could also gain benefit from using Pu/Th-MOX.

The mass and corresponding volume of annually discharged spent fuels (HLW) is determined for both HTR and LWR (operating with burnups of 33 GWd/tHM and 45 GWd/tHM). The results and key assumptions are presented in Table 8. To enable a direct comparison, the volumes of HLW are determined on a per GW_ey basis.

With the fuel material remaining integral with the graphite of the compacts there is a difference in the annually discharged spent fuel volume per GW_ey between the LWR type and the HTR. The advantage appears to be in favour of GT-MHR, particularly when compared with the LWR at 33 GWd/tHM and the other HTRs, and is mainly due to the high burn-up. This has to be put into a proper perspective as the volume of HTR spent fuel is considered here without moderator block. There is a more significant difference in the volumes of annual spent fuel discharges if the graphite of the HTR fuel compacts is no longer taken into account. Under these circumstances, ratios of 1 to 10 for the volumes of discharged spent fuels between the GT-MHR and the THTR and 1 to 6 for the HTR-Module exist. But this would need the development of specific conditioning or reprocessing techniques.

However, the graphite and the coating will be additional barriers in the case of direct disposal which probably significantly reduces the long term radionuclide mobilization, in a final repository. These protective functions, in the main, also apply to long-term intermediate storage.

Thus, the annual volume of spent fuels arising from an HTR reactor, GT-MHR Pu loaded, is about 3.8 m^3 of HLW per GW_ey (spent fuels without graphite) or 28 m^3 when the graphite of the fuel compacts is also considered to be spent fuel (no separation). It should be emphasized that the predicted advantages in spent fuel waste volume are mainly a consequence of the weapons grade Pu oxide fuel loading assumed in the GT-MHR study. Much of this advantage also applies to the burning of civil plutonium of first and second generation, as other studies within HTR-N have shown. The adoption of a more typical low enrichment U oxide fuel or of a mixed oxide loading would erode much of this advantage.

In any case, HTR being operated as Pu-burners show a significant improvement in the minimization of high-level waste and can help to stabilise or even reduce the Pu inventories when operated in symbiosis with LWR. Thus, there is a strong motivation to foster the development of Pu-based coated particles for Pu-burning.

Table 7: Amount of long-lived radionuclides from different reactor types after a decay time of 5 years

Nuclide	GT-MHR	THTR	HTR Module	LWR-33	LWR-45
			Mass [g/GWy$_e$]		
C-14	9.70E-02	6.62E+01	5.26E-03	3.40E+00	2.39E+00
Se-79	8.63E+01	2.06E+02	1.56E+02	1.56E+02	1.10E+02
Zr-93	1.01E+04	2.32E+04	2.11E+04	2.35E+04	1.65E+04
Nb-94	2.00E-02	4.40E-03	1.90E-02	8.13E+00	5.72E+00
Tc-99	1.61E+04	2.02E+04	2.25E+04	2.73E+04	1.92E+04
Pd-107	1.10E+04	7.34E+02	4.14E+03	7.91E+03	5.56E+03
Sn-126	1.21E+03	5.28E+02	4.68E+02	7.44E+02	5.23E+02
I-129	5.21E+03	5.70E+03	4.57E+03	5.88E+03	4.14E+03
Cs-135	1.36E+04	8.91E+03	7.30E+03	1.23E+04	8.65E+03

Table 8: Main data of the reactor system and spent fuel amount per GW$_e$y

Reactor	LWR 1300	LWR 1300	HTR THTR	HTR HTR-Modul (200 MWth)	GT-MHR
Fuel	UO_2	UO_2	$(Th,U)O_2$ 10:1 93%	UO_2 8%	PuO_2 *
Electric power generation [GW]	1.3	1.3	0.3	0.08	0.28
Discharge burnup [MWd/tHM]	33,000	45,000	110,000	80,000	642,000
Duration of 1 fuel cycle [month]	38	54	continuous ~ 1/3 per yr		28
Load of initial core [tHM]	104	104	4.00	2.52	0.7007
Mass of discharged spent fuel [tHM/GW$_e$y]	25.12	17.67	8.51	11.36	1.07
Volume of discharged spent fuel [m³/GW$_e$y] without compact	50	35	14.20	53.18	3.8
Volume of discharged spent fuel [m³/GW$_e$y] with compact			141.18	300.39	28

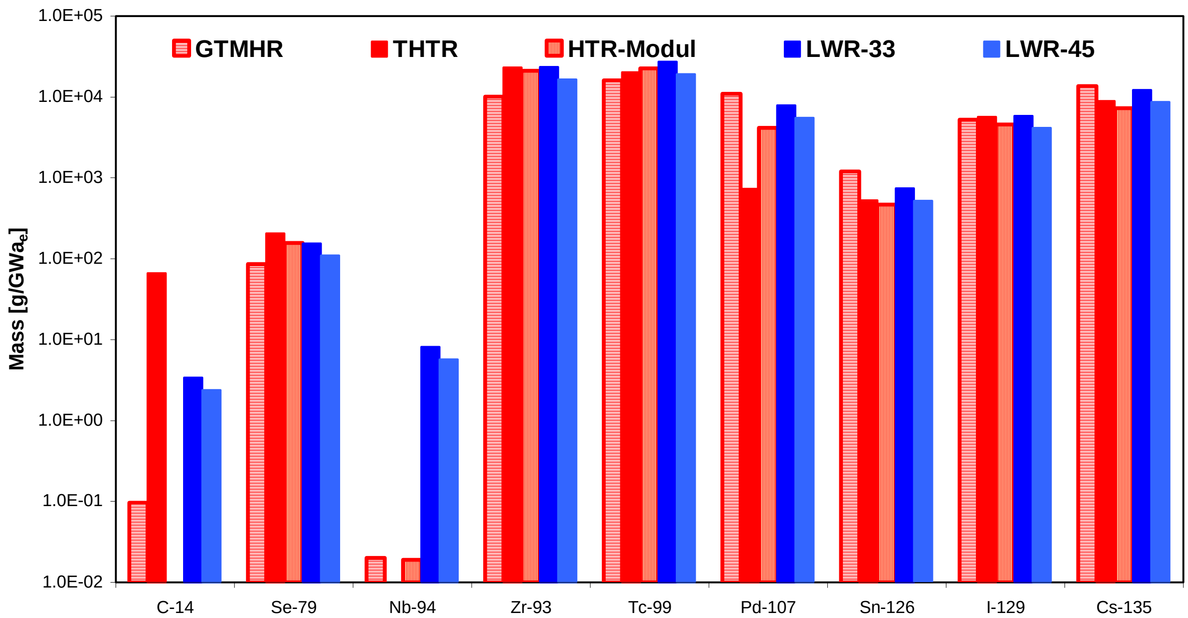

Fig. 1: Comparison of long-lived fission products from different HTRs and LWRs.

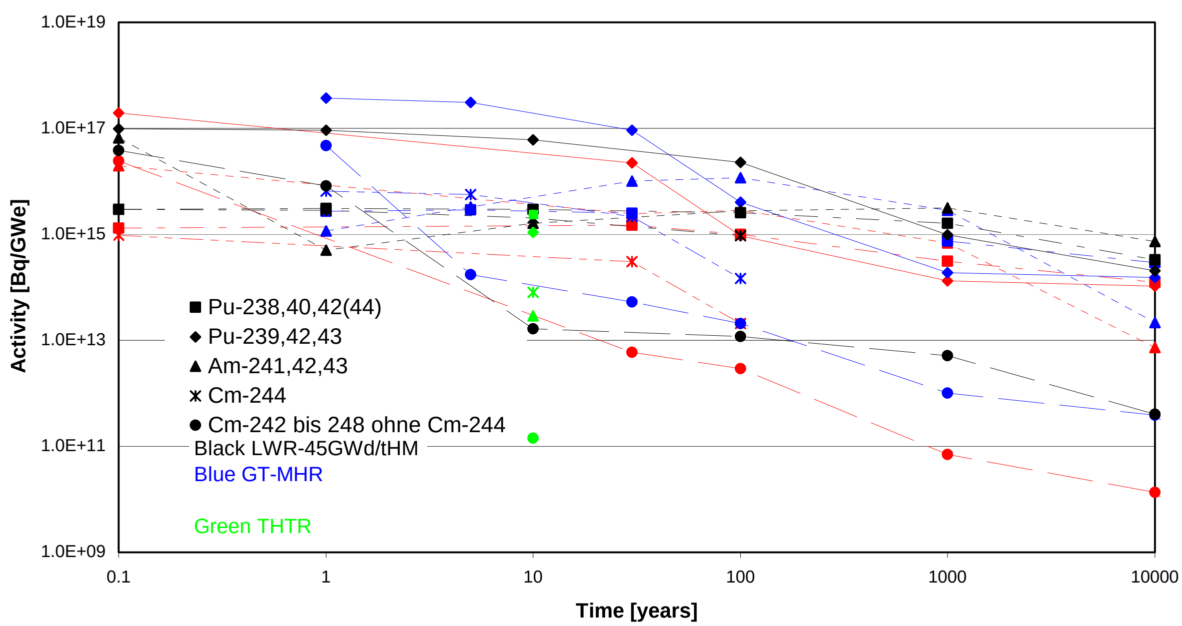

Fig. 2: Comparison of actinides activities from different HTRs and LWR.

5 Decommissioning Waste (Task 4.2)

Data about decommissioning waste had been obtained from the AVR and analysed in more detail. The quantities of solid radioactive waste expected to arise from total dismantling are given in Table 9. Fig. 3 shows the anticipated treatment of all radioactive materials from total dismantling.

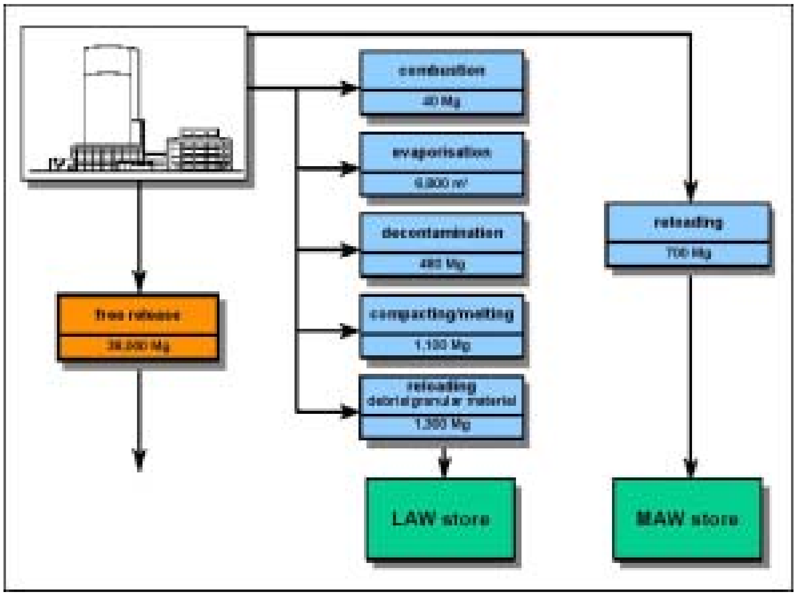

Fig. 3: Treatment concept for radioactive materials from AVR dismantling

The main part of waste, about 38000 Mg, will be below the limits for free release. Most of the radioactive wastes can be treated with well-known techniques, and therefore it is no different to other reactor types. The 3,600Mg of radioactive AVR decommissioning waste compares with approximately 9,000Mg for a 1.3GW$_e$ PWR /10/. The active waste from dismantling a BWR, with a direct power cycle, is approximately double that of the equivalent PWR. Only 230 Mg of ceramic internals (carbon bricks from insulation/shielding and graphite bricks from reflector) need special attention. Containing most of the radioactivity, these ceramic internals are going to be stored in the intermediate-level shielded store until a final repository will be available, or new treatment techniques can be developed. As graphite purification processes are not available yet, there is another provisional option to fill the whole pressure vessel with grout and then to remove the whole vessel including all internals for separate storage and future treatment.

Table 9: Quantities of solid radioactive wastes arising from AVR dismantling

Type of solid waste	**Masses in Mg**
ceramic internals	230
metallic residual materials and waste	1,500
debris / granular materials	1,280
non-combustible solid waste	200
combustible solid waste	40
secondary waste	370
total	approx. 3,600

A wide range of ideas and options for the treatment of irradiated graphite from shut down reactors have been proposed. They comprise, among others:

- mechanical methods (e.g. cutting, crushing) to optimise the packaging factor of typical storage containers,
- thermal methods (e.g. incineration) to drastically reduce the volume of materials (ashes) to be stored. ^{14}C would be released to the atmosphere, representing the principal objection to incineration due to lack of public perception. Processes for the recovery of ^{14}C from the CO_2 off gas, e.g. with washing in a calcium hydroxide solution and using the carbonate formed as backfill material in a repository have therefore been evaluated,
- pyrolysis, where steam reacts with the organic material to separate a ^{14}C rich gaseous fraction from the rest of radionuclides, which would be fixed in a carbon char, chemical treatment with solvents or gases to remove adsorbed radionuclides in order to improve the storage properties, or
- recycling of irradiated graphite, e.g. to be used as additive for container or repository backfill material. Reuse for fabrication of fuel matrix or moderator/ reflector blocks is another option.

Wigner energy accumulation ('stored energy') occurs in graphite under neutron irradiation because atoms are displaced from their normal lattice position into configurations of higher potential energy /11/. Some simultaneous thermal and irradiation annealing takes place, but there is a net energy gain which is a function both of irradiation time and temperature. A proportion of the stored energy can be released if the graphite is heated to about 50°K above its irradiation temperature, although a temperature in excess of 2000°K is required before all the energy is annealed. Therefore, the sudden release of Wigner energy due to a temperature raise above critical limits during any treatment or storage must be considered. However, due to the high operating and reactor coolant inlet temperatures, Wigner energy release is generally considered to be less important for HTR graphite than it is for other gas-cooled reactors.

6 Estimation of graphite waste for future HTR PuO_2 burner systems

This waste stream is an additional one for HTR that is not applicable to LWR systems. Therefore, the amount of radioactive graphite waste has been calculated /9/ for the GT-MHR, as an example. The results should be regarded only as orders of magnitude, not as precise or absolute values of nuclide quantities in an HTR reactor.

A total mass of 727 t has been assumed for the calculation bases of the GT-MHR core design, which are divided into 700 t graphite blocks and 27 t of graphite in the fuel compacts.

The assumed impurities of the graphite are a representative average of levels found in actual nuclear graphite types.

Over the complete HTR lifecycle, the volume of graphite requiring non-surface disposal comprises:

– The 148 m^3 of permanent reflector graphite removed during decommissioning.
– The 405 m^3 of graphite discharged as an integral part of the fuel compacts. This volume is derived assuming ⅓ of the 15 m^3 of compact graphite in the core is discharged every 280 days and this continues for 60 years of operation.

With 60 years of operation and a net electrical output of 0.278 GW, the volume of non-surface disposal graphite, normalised to 1 GWy, is 33 m^3/GW_ey.

The CEA analysis showed that, in the case of the French disposal criteria, the dominant long lived nuclides were ^{14}C and ^{36}Cl. Fig. 4 shows the activities of these nuclides in the different locations of the GT-MHR type core with respect to the French surface disposal limits. These two nuclides can arise from the activation of nitrogen and chlorine impurities, respectively. It therefore follows that the nitrogen and chlorine content of the graphite needs to be reduced, as far as possible. Also the nitrogen content in the helium coolant needs to be kept as low as possible to avoid ^{14}C from contamination. The purging of the pressure vessel during start-up and the implementation of an efficient coolant purification and make-up system were identified as key areas for controlling and removing air ingress. Attention has also to be put on the 'free' uranium contamination of the fuel matrix as it may cause more fission product release than high performance coated-particle fuel.

Actual experience of treatment and removal of graphite from shut down HTR is limited to the Fort St. Vrain plant /12/ in USA. This reactor has been fully decommissioned with the graphite sent to intermediate disposal facilities. Data on activity mapping at FSV should be retrieved for further analyses and predictions for future HTR. As with most other graphite-moderated reactors (MAGNOX, AGR) final solutions for graphite disposal or purification and reuse still have to be elaborated.

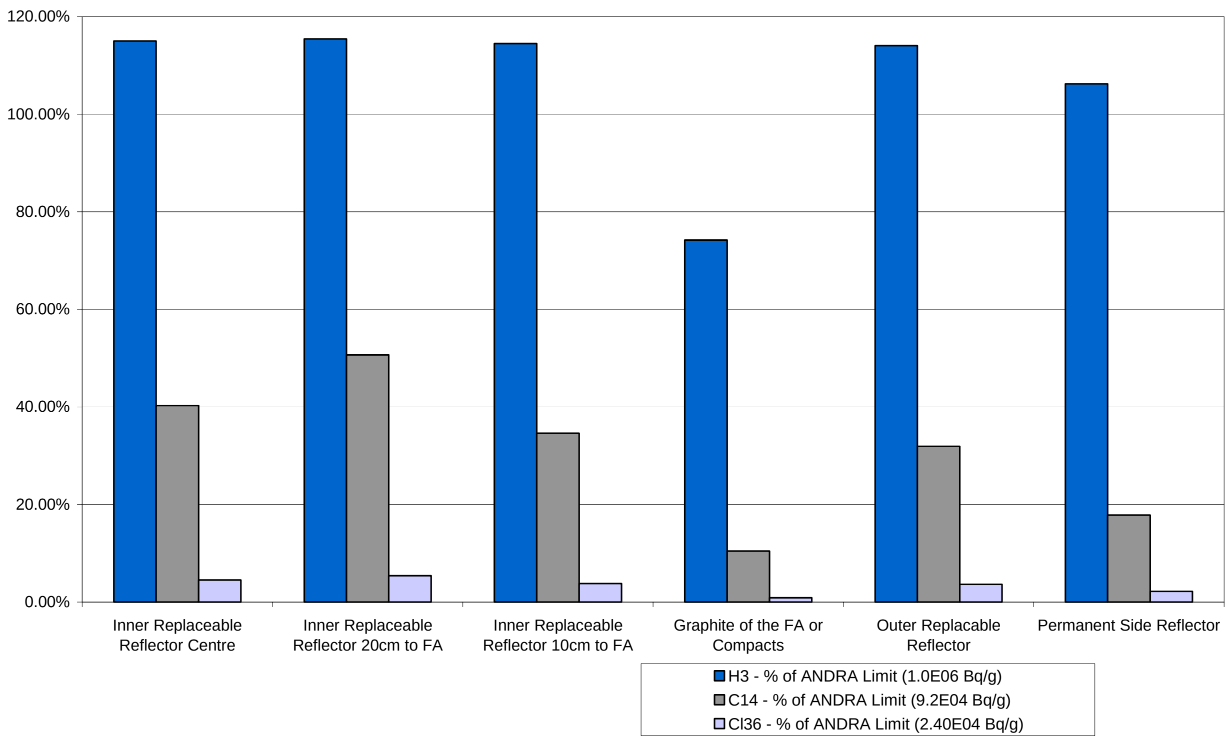

Fig. 4: Key nuclide activities with respect to French surface disposal limits at different graphite locations in the GT-MHR after 3 years irradiation.

7 Waste Minimisation (Task 4.4)

Waste minimisation has been considered in terms of graphite, ex-pressure vessel components and power circuit contamination.

Graphite

From the findings summarised in Section 6 there would be a reduction in the volume of graphite requiring non-surface disposal if the concentration of chlorine and nitrogen impurities were reduced in the pre-irradiated graphite. Air ingress to the He coolant provides another source of nitrogen and this should be kept as low as possible. Most of the activity from tritium, which as can be seen from Fig. 4 could potentially prevent the surface disposal of graphite, would have decayed after several decades due to its 12 year half-life. There is, thus, a benefit in having a period of safestore before final graphite disposal. The tritium activity level can also be reduced by decreasing the levels of lithium impurities initially present within the graphite. Potential treatments for irradiated graphite are summarised in Section 5.

Ex-Pressure Vessel Components

By analogy with the UK gas reactors, it was found that the activation of components and structures immediately ex vessel, i.e. the high temperature section of boilers and gas ducts, was significantly reduced by the presence of substantial neutron shields. The same would be relevant to the first stages of the HTR power circuit.

Power Circuit Component

Potential sources of contamination in HTR were identified as:

– Graphite or carbonaceous deposits carried over from the reactor.
– Tritium released from the reactor.
– Radioactive silver deposits, caused by fuel fission products.

The possible mechanisms for reducing contamination were identified as cyclones or filtration for the graphite and gas purification for the tritium.

8 Conclusions

With regard to sustainable developments, the waste issue nowadays receives much more attention than it got in the past when the focus was, more or less exclusively, put on the technological development of the reactor systems and high conversion fuel cycles. Waste minimisation during operation and decommissioning has to be built and designed into future reactors, from the very beginning and will be part of the licensing, in many countries.

Analyses of gaseous, liquid and solid operational waste discharges of former HTR prototypes and demonstration reactors showed that tritium, ^{14}C and ^{36}Cl. are the main effluents which in part even exceeded those of LWRs. These releases were mainly due to lithium, boron, chlorine and nitrogen impurities in the graphite and in carbon brick structures. The excessive generation of tritium and ^{14}C can be reduced by improved manufacture processes and operational provisions. This also applies to the uranium contamination of the fuel matrix and structural material.

Most activity is concentrated in the graphite and carbon brick structures as measurements at AVR have shown. This also comprises other fission products which are adsorbed in the porous graphite, too. With regard to the large amount of carbonaceous waste from HTR, there is a strong need for development of purification and conditioning processes for contaminated graphite and carbon brick including potential recycling. This might even be a precondition for a broad application and market introduction of HTR.

Further increase of burn-up is a strong incentive for waste reduction. Special fuel for plutonium burning needs to be developed to stabilise or reduce plutonium stockpiles and proliferation issues. Ultra-high burn-up of Pu in HTR in a once-through mode is feasible and is equivalent to multiple recycling of MOX in LWR which poses considerable challenges.

Some fundamental discussions are still necessary on how to evaluate the assumed good qualities of confinement of TRISO particles for the spent fuel in a repository. It has to be clarified whether the coating and the matrix and the resulting low heat densities can be taken as a benefit for disposal and storage and as a substitute for other protective measures (e.g. vitrification, cooling) or whether it has to be evaluated as additional waste and as a parasitic volume.

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