TREATMENT AND DISPOSAL OF THE RADIOACTIVE GRAPHITE WASTE OF HIGH-TEMPERATURE GAS-COOLED REACTOR SPENT FUEL

LI JUNFENG

(INET Tsinghua University, Beijing 100084, P. R. China)

1. INTRODUCTION

High-temperature gas-cooled reactors (HTGRs) represent one of the Gen IV reactors in the future market, with efficient generation of energy and the supply of process heat at high temperature utilised in many industrial processes [1, 2]. HTGR development has been carried out within China's National High Technology Research and Development Program. The first industrial demonstration HTGR of 200 MWe is under construction in Shandong Province China.

HTGRs use ceramic-coated fuel particles that are strong and highly resistant to irradiation. Graphite is used as moderator and helium is used as coolant. The fuel particles and the graphite block in which they are imbedded can withstand very high temperature (up to ~1600℃).

Graphite waste presents as the fuel element components of HTGR with up to 95% of the whole element beside the graphite blocks in the core [3]. For example, a 200 MWe reactor could discharge about 90,000 fuel elements with 17 tonnes irradiated graphite included each year.

The core of the HTGR in China consists of a pebble bed with spherical fuel elements. The UO_2 fuel kernel particles (0.5mm diameter) (triple-coated isotropic fuel particles) are coated by several layers including inner buffer layer with less dense pyrocarbon, dense pyro-carbon, SiC layer and outer layer of dense pyro-carbon, which can prevent the leaking of fission products (Fig. 1). Spherical fuel elements (60mm diameter) consist of a 50mm diameter inner zone and 5mm thick shell of fuel free zone [3]. The inner zone contains about 8300 triple-coated isotropic fuel particles of 0.92mm in diameter dispersed in the graphite matrix.

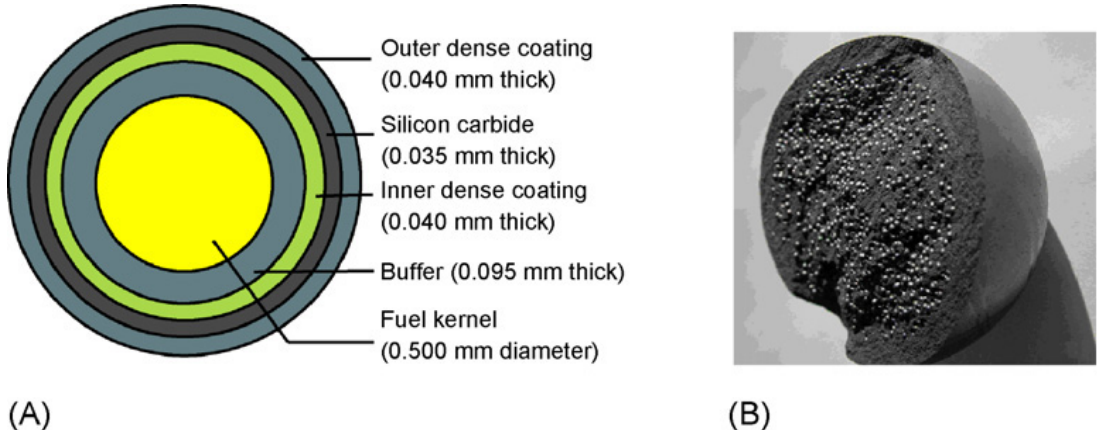

FIG. 1. Structures of the coated particle and fuel element of HTR-10 in China:

(A) cross-section of the coated particle; (B) cross-section of the fuel element.

The spherical fuel elements of HTGR contain a large amount of the graphite matrix as a structural material. The irradiated graphite from the element's matrix could contribute a major part of the carbonaceous waste with a small fraction of the other carbide and pyro-carbon, and the two kinds of carbonaceous waste should be treated separately due to their different radioactivity.

Other helium-cooled reactor designs (*e.g.* MHR) use hexagonal fuel elements[4]. A large number of fuel particles are mixed with a carbon-based mastic to make fuel compacts. The mixture is pressed into the shape of a short right-circular cylinder and sintered at a high temperature to drive off all the volatile components. The compacts are then loaded into predrilled holes in a machined graphite block to make the fuel element. The fuel elements are hexagonal in shape and contain additional holes to accept control rods and provide a path for flowing coolant.

HTR spent fuel processing was studied at laboratory scale and then at pilot scale in the 1960s and 1970s in the USA (Oak Ridge National Laboratory) and until the early 1980s in Germany (Forschungszentrum Jülich GmbH) [5-7]. The first objective of these studies was the recovery of the uranium 233 in the U–Th cycle. This work essentially concerned the head-end process to recover the uranium and thorium oxide kernels by eliminating the successive layers of carbon and silicon carbide (SiC). After this step, the uranium and the thorium were recovered and separated from the fission products by the THOREX process (dissolution of the oxide by HNO_3 and HF followed by extraction of the thorium and uranium by TBP) and used to fabricate fresh oxide.

2. RADIOACTIVE ISOTOPES IN GRAPHITE MATRIX OF HTGR SPENT FUEL ELEMENTS

Radioactive isotopes in graphite matrix which require the most consideration are mainly ^{14}C and ^{3}H from the spent fuel elements. ^{14}C was formed by the neutron reaction of ^{14}N adsorbed from air and of natural component ^{13}C in graphite during the radiation in the core. The generated ^{14}C level was estimated to be 2.96×10^{11} $Bq.m^{-3}$ of the graphite material matrix, which is the upper limit of ^{14}C for class C LLW by the U.S. Nuclear Regulatory Commission (NRC) [10,11], such as Davis' calculation with a total ^{14}C radioactivity of below 1.11×10^{11} $Bq.m^{-3}$ on the presumption of 30 ppm nitrogen impurity in graphite and natural component ^{13}C with a burnup of 100 GWd/MTHM [11]. This result is relatively conservative since the adsorbed nitrogen reported was about 6~12 ppm in the graphite matrix implying that the actual overall ^{14}C level may be lower by a factor of 2 to 5, and the technologies through the improved storage in non-nitrogen-containing inertising atmospheres like argon could significantly reduce the nitrogen impurity.

^{3}H was generated though the neutron irradiation of ^{3}Li impurity in graphite, the radioactivity of which could be approximately 5.55×10^{11} $Bq.m^{-1}$ for the case of 0.005 ppm Li impurity under a burnup of 100 GWd/MTHM, which is far below 1.48×10^{12} $Bq.m^{-3}$ the limit for class A LLW by the U.S. NRC [10]. Therefore, the wastes from the graphite matrix of spent fuel elements have the potential to be classified as LLW, when the proper technology is employed to avoid further contamination of the fission products.

The relatively long half-life of ^{14}C (5730 years)makes it a potential long-term health hazard. Combustion of the graphite could convert any ^{14}C present into $^{14}CO_2$ which could potentially pose a β radiation exposure hazard to the general public if released into the atmosphere.

3. OPTIONS FOR TREATING HTGR FUEL [10]

There are several possible options to put HTGR spent fuel into a form that will be acceptable for repository disposal, The preferred option depends on the waste acceptance criteria for the repository, availability of LLW disposal for graphite, overall economics, and overall risks. Frequently, the minimum processing that yields an acceptable waste form also gives the lowest costs as well as the simplest process and the least risk. The diagram in Fig. 2 shows the options that could be used.

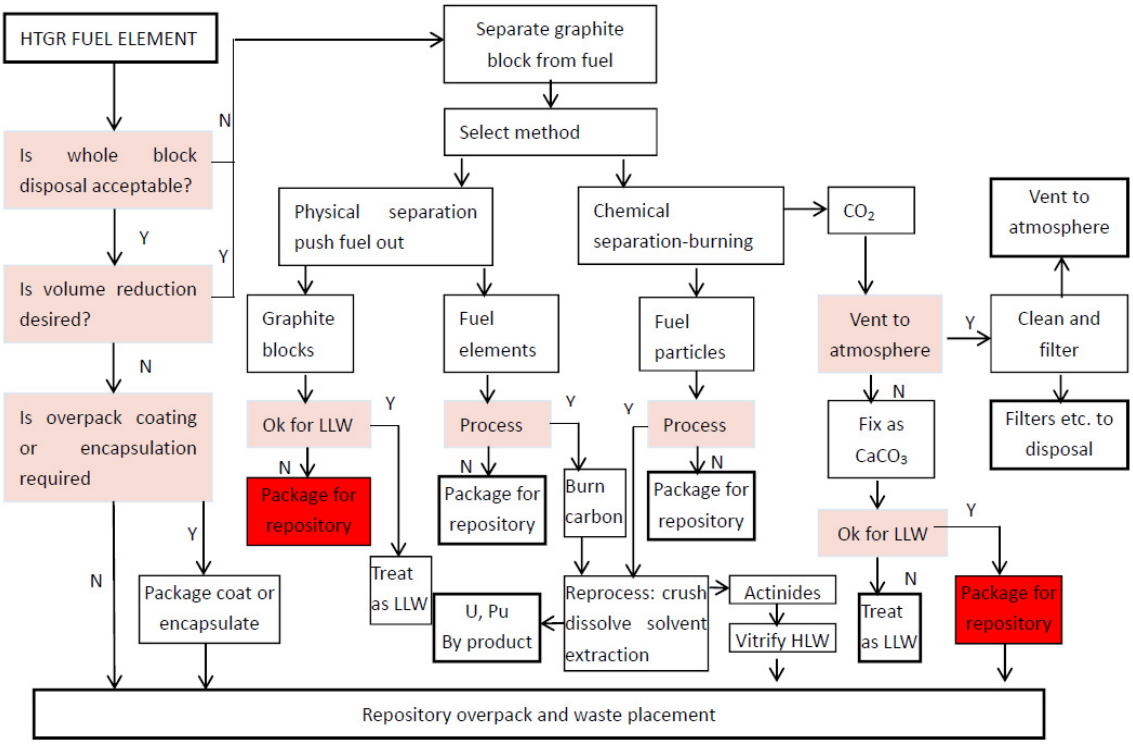

FIG. 2. Options for processing of HTGR spent fuel for repository disposal

3.1 Whole-block Disposal

If whole block HTGR spent fuel disposal is acceptable. The HTGR spent fuel can be disposed of in the repository after suitable packaging. This leads to the next question, Is overpack, coating or encapsulation required? The answer to this question determines if direct disposal of the HTGR spent fuel block is allowed. If direct disposal is acceptable, the spent fuel can be placed in waste canisters as is proposed for LWR spent fuel.

Repository waste canister size is limited by spent fuel decay heat load. If too much spent fuel is put into a canister, the waste or nearby rock will overheat. HTGR spent fuel has a lower decay heat load; therefore, larger, more economical waste packages might be an option for the direct spent fuel disposal scenario. The optimum canister size is determined by handling and economic considerations within the limits imposed by the heat load.

If direct disposal is not allowed, the option exists to "overpack, coat, or encapsulate" whole HTGR spent fuel blocks to improve the disposal performance of the waste before packaging.

3.2 Disposal with Prior Removal of Graphite

If whole block spent fuel is unacceptable for repository emplacement, the first processing option is to separate the spent fuel assembly into (1) the spent fuel matrix material and (2)

the carbon from the graphite block. The separation process can be a physical or mechanical process that yields graphite wastes or a chemical process such as combustion where the graphite is oxidized to carbon dioxide. After separation, spent fuel processing and carbon processing must be considered separately.

Recently in France [12] and USA, for block-type HTGR spent fuel elements, scientists adopted mechanical extraction (by pushing or coring operations) to separate nuclear compacts from the spent fuel block to get rid of the most graphite without damaging the particle fuel coatings. They considered the mechanical extraction method to be a very promising option for the block-type elements of HTGR after the demonstration of its feasibility [4, 8]. For spherical fuel elements, the graphite particles were separated by applying voltage discharges through a fluid in a suitable reaction vessel [13] which is appropriate for a hot-cell environment and for industrially relevant material streams; an alternative electrochemical method with salt solution is user-friendly without any damage to the coated fuel particles, but needs more work to verify its feasibility [14].

For prismatic fuel elements, the mechanical separation process can be coring, for spherical fuel elements, the process can be lathing.

For disposal of spent fuel, several options are available that depend upon whether the fuel is obtained by mechanical separation or coated particles obtained by combustion. For prismatic fuel elements, the simplest option is to package the fuel rods for disposal at the repository. The other option for mechanically separated fuel is to combust and chemically process the fuel to final products consisting of fissile-fertile by product and a suitable waste form for the fission products and actinides. If separation has been by combusting the fuel element, the coated particles can be reprocessed or the particles can be packaged for disposal at the repository. Various overpack, coating, or encapsulation technologies can be considered to produce an acceptable waste form.

For carbon processing, there are two different materials (CO_2 or graphite) to be disposed depending on whether combusting or mechanical separation has been used. Depending on the fuel design and performance and upon the separation process efficiency, some fission products could be associated with the carbon waste stream. If the fission product contamination is low enough, the carbon waste can be treated as LLW. (It has been assumed that release of CO_2 to the environment is not an acceptable option although it is depicted on the option diagram.) If the carbon stream contains significant quantities of long lived radionuclides, it may be required to go to the HLW repository.

If the carbon waste stream must go to the repository, it must meet certain minimum requirements, if the carbon is in the form of carbon dioxide, it can be reacted with calcium hydroxide or barium hydroxide to yield stable calcium carbonate (limestone) or barium carbonate. A carbonate waste form going to a repository would be significantly different from spent fuel; in particular, its heat generation rate would be very low. This would allow the carbon waste form to be emplaced in suitable disposal facilities without

the need for significant dispersion of heat. In contrast, typical spent fuel with its much higher decay heat is disposed of in small canisters to allow conduction of decay heat to the rock while not overheating the waste.

This method requires the separation of the fuel from the graphite, by physical means or by combustion. In either case, disposal of the remaining graphite or the "fixed" ^{14}C remains a problem that can be solved technically. However, the option is considerably more expensive and may not be more desirable from an environmental point-of-view than the whole block option. Some development work is required to implement this option.

3.3 Disposal with dissolution of spent fuel

Chemical processing of the spent fuel matrix includes the option of conventional reprocessing with recovery of uranium, plutonium, and/or thorium. If the waste from reprocessing is converted into glass, as has been proposed for HTGR reprocessing plants, the waste should meet all acceptance criteria. The repository is currently designed for glass waste forms from nuclear fuel reprocessing plants. Other waste forms must be evaluated on a case-by-case basis.

In chemical processing, fuel reprocessing is carried out to recover the remaining fissile and fertile value in the spent fuel. The minor actinides (such as Pu, Np, Am, and Cm) are also recovered and managed separately from the high-level fission product waste, It was assumed that transuranic and other actinide wastes would be incorporated into borosilicate glass.

This option requires separation of the fuel and solvent extraction to separate waste and fuel components of the spent fuel. Probably the only time this option will be used is when it is desirable to recover the fuel for recycling. While it is technically feasible to utilise this option, substantial development must be done before facilities can be designed and operated.

4. DISINTEGRATION OF HTGR SPENT FUEL

4.1 Mechanical methods

The technologies developed were primarily grinding of the prismatic elements or pebbles and incineration of the crushed graphite. Different types of crushers were tested: jaw crushers, hammer crushers, grinding mills, ball mills, etc. Fig. 3 shows the device used at Oak Ridge National Laboratories to grind compacts.

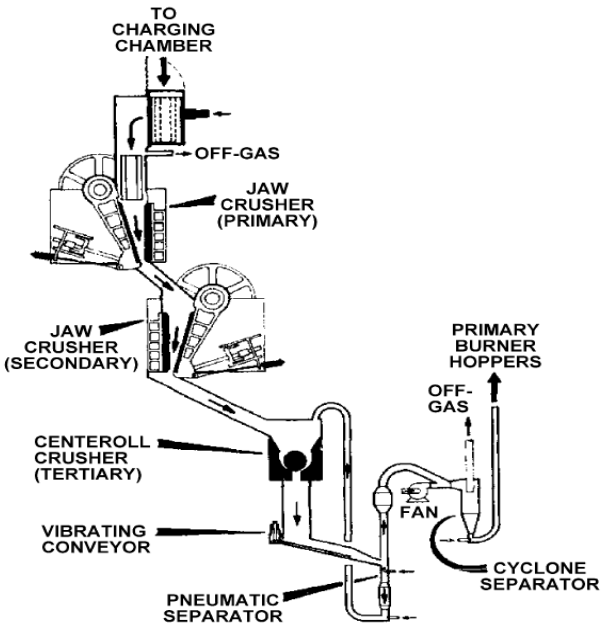

FIG. 3. Schematic of the ORNL grinding technology

Crushing or grinding techniques can thus be used for this operation. Some of these had previously been developed up to pilot scale and their limitations are known: difficulties in obtaining a very high recovery yield, significant dust generation, etc.

4.2 Combustion

Combustion was carried out mainly in a fluidised bed under flowing oxygen with no specific precautions for ^{14}C release. A demonstration test was performed at the FZJ research center (Germany) within the framework of the Joint Fuel Reprocessing Demonstration (JFPD) project: about 11,000 fresh pebbles (2.2 Mt of materials) were processed by grinding and combustion at a rate of about 14 $kg.h^{-1}$.

Overall, the processes studied in United States and Germany combining grinding and combustion techniques followed by dissolution in aqueous acid and separation of U and Th by the THOREX process, provided recovery yields of the order of 95%. Considering the potential radiotoxicity of the ultimate wasteform, higher recovery yields are required today. Taking into account the additional constraint of limiting the activity of the off-gas releases, it is clear that these processes do not meet the performance criteria.

4.3 Electrochemical Method

Two techniques (acoustic waves and pulsed currents) have been considered to overcome these problems, but only the latter is currently being tested. This process [5] has been used by the CEA to clean equipment items. Solids are exposed to high voltage, high intensity pulses between 200 and 500 kV, and discharge currents from 10 to 20 kA. As the pulses are of short duration (about 2 μs), the implemented energy is a few kJ. Under

these conditions, in the presence of water, an electric arc is created near the solid and propagates preferentially through it. The energy (10–100 J.cm^{-1}) is deposited in few microseconds in discharge holes less than 10μm in diameter, resulting in a temperature increase of up to 10,000K together with a pressures rise of 1010 Pa. These extreme conditions immediately split the solid and reduce it to fine fragments. If the electric arc does not penetrate the solid material, the electrical energy is transformed into mechanical energy that propagates in the water in the form of a shock wave provoking a similar disassembly effect. The principle of the technique shown in Fig. 4.

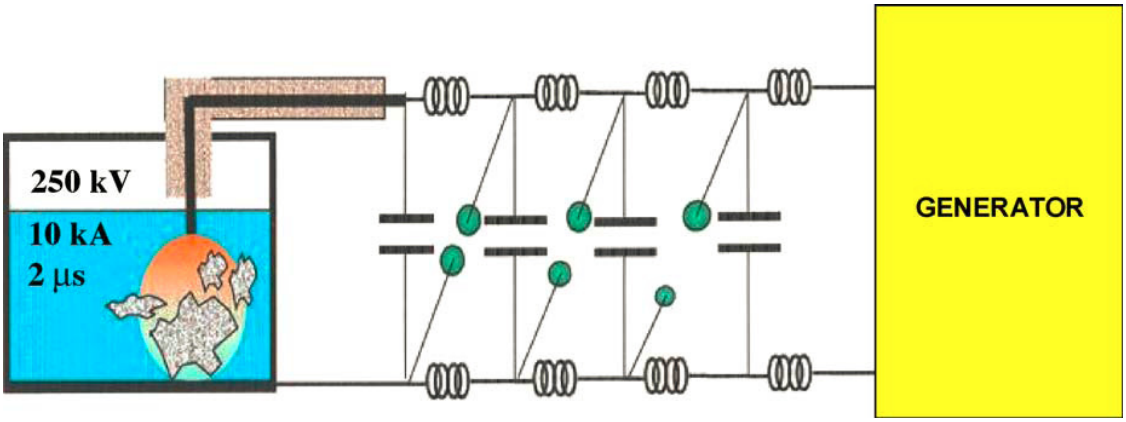

FIG. 4. Principle of graphite disassembly by pulsed currents

Electrochemical method with salt as electrolyte has been studied to disintegrate the graphite matrix from the simulative HTGR fuel elements[14]. Ammonium nitrate was experimentally chosen as the appropriate electrolyte. The volume average diameter of disintegrated graphite fragments is about 100μm and the maximal value is less than 900μm. After disintegration, the weight of graphite is found to increase by about 20% without the release of a large amount of CO_2, probably owing to the partial oxidation to graphite in electrochemical process. The work indicates that the improved electrochemical method has the potential to reduce the secondary nuclear waste and is a promising option to disintegrate graphite matrix from high temperature gas-cooled reactor spent fuel elements in the head-end of reprocessing.

The maximum diameter of graphite fragments is less than 900μm, which is less than the diameter (920μm) of coated fuel particles. This is very useful for the further separation of the disintegrated graphite fragments from the coated fuel particles. 230 g graphite fragments were collected after disintegration of a 194 g graphite element. The weight increased by 18.6% probably owing to the partial oxidation to graphite in electrochemical process. The experimental results show that the improved electrochemical method with nitrate as electrolyte is a very promising option to disintegrate the graphite matrix from HTGR spent fuel elements.

The potential using electrochemical method for fragmenting the graphite compacts and

separating the TRISO coating from the kernels is illustrated in Fig. 5.

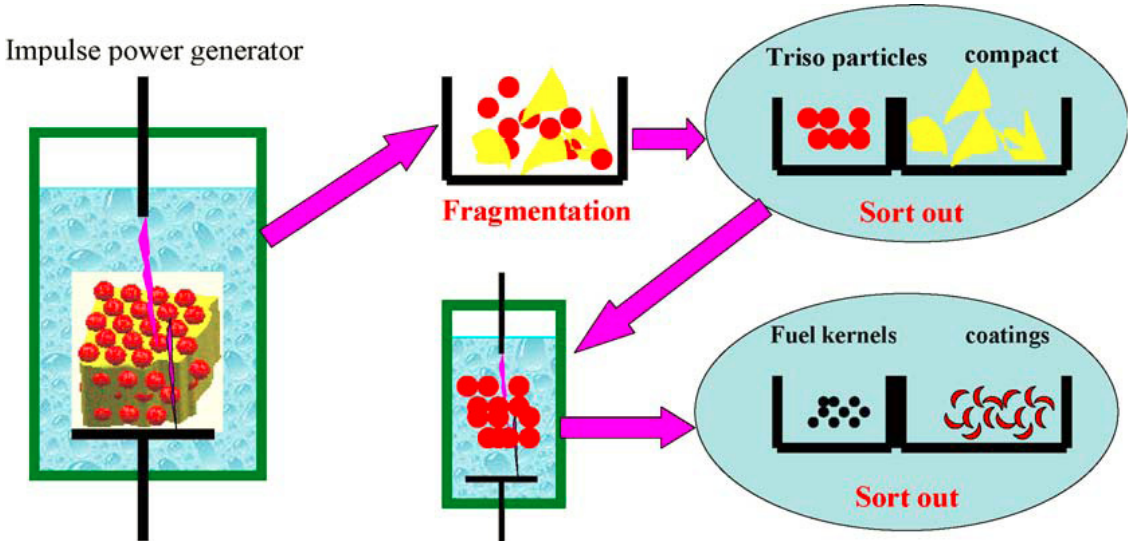

Fig. 5. Potential use of pulsed currents to fragment graphite from compacts and TRISO particle layers

Further studies will be undertaken on removal of the coating from the TRISO particles, on optimizing the parameters and assessing the safety of the technology. As the process is implemented under water, techniques for separating the graphite from the TRISO particles by flotation are examined to avoid any change of media. The relative density of the graphite is 1.8 compared with about five for the particles, so the process should be suitable for this type of separation. The requirement will essentially be quantitative recovery of particles without the graphite fraction.

5. Silicon Carbide Shell Removal

Treatment methods, such as combustion and electrochemical, generally leave the spherical SiC coating intact. The inner carbon layers and fuel kernel are consequently unharmed. The SiC coating may be breached by grinding to expose the inner carbon and fuel kernel. Further treatment methods may then recover the fuel while leaving the empty SiC shell fragment as a waste. Grinding and milling can be followed by a solid-solid separation as an option prior to aqueous dissolution, or dissolution may follow directly.

Once the carbon layer has been removed, chemical methods to remove the SiC layer may also be implemented. Dry fluorination in which the processing time is adjusted to a period sufficiently long to permit removal of SiC coating but short enough to prevent attack of the heavy metal kernel has been studied[16].

Breaking the layers also releases a fraction of the fission products (the internal pressure of the particles exceeds 50 bars) and probably causes partial crumbling of the kernels. Off-gas treatment is therefore required to trap the volatile species and other fission products desorbed during this operation. Oxidizing processes therefore seem preferable to

mechanical ones to obtain maximum desorption of the gaseous or volatile fission products in the off gases under the effect of the temperature.

Despite the extensive research work done in the past on TRISO fuel processing, no comprehensive process is available today that meets the criteria imposed for recycling the actinides and for managing the waste and off-gas releases.

Innovative techniques for mechanical extraction of the compacts, disassembly of the graphite by pulsed currents, and de-coating of the particles by oxidizing thermal processes to reach the fissile compound and allow the industrial processing of this type of fuel, are under development.

6. TREATMENT AND DISPOSAL OF IRRADIATED GRAPHITE

Regarding the characteristics of irradiated graphite, it is of value to recycle carbon resource and reuse it into new cycles, since the quantity of radioactive waste to be disposed of in deep repository would be greatly reduced when there are not enough repositories available. Basically, three options may have the potential for the treatment and disposal of irradiated graphite, including the disposal through graphite gasification, the direct disposal without any chemical reaction involved and the process of the recycle and the decontamination of radioactive nuclides. The disposal mode of the waste depends on the type of different irradiated graphite wastes. Therefore an appropriate process should be determined after a comprehensive examination of the irradiated graphite.

6.1 Conditioning/disposal through graphite gasification

Two processes could be applied for the gasification of graphite, incineration in the atmosphere of oxygen and gasification with steam [17,18].

Incineration is of great advantages for the combustible radioactive waste [19], because it could achieve a large volume minimisation coefficient leaving only a small quantity of residual ashes, eliminate the safety problem with the conversion of waste into inorganic ashes and also possibly recycle some valuable heavy metals such as U and Pu. Subsequent processes could be employed to deal with the gaseous products. Gases without radioactive ^{14}C could be released into the atmosphere after a segregation of aerosol; while for gases with the presence of ^{14}C, separation of ^{14}C before the release into the atmosphere should be implemented, otherwise gases should totally be immobilized into carbonate by the reaction with $Ca(OH)_2$ or $Ba(OH)_2$ [20]. Thus, the incineration process is of great advantages to cope with the graphite waste without the existence of ^{14}C, and could obtain the maximal volume minimization coefficient after the release of gases products. For example, radioactive graphite waste generated from the graphite tools, such as graphite crucibles and molds sometime contaminated by U and Pu, could be treated by the incineration method cooperated with the segregation of radioactive aerosol.

In the instance of graphite waste contaminated by various kinds of radionuclides, especially ^{14}C, the gases product could not simply be released, i.e., the treatment of graphite waste from reactor cores. Moreover, the final volume of the carbonate after the solidification could be largely increased by a factor of six [21]. Therefore, the option of graphite incineration is supposed to be cooperated with the separation of ^{14}C when dealing with the graphite waste of large quantities in the case of ^{14}C contamination.

Another method for the graphite gasification is based on the steam pyrolysis. Mason [18, 22] indicated in the patent that the gasification of radioactive graphite could be implemented *in situ* or outside of the decommissioned reactors, and the graphite at the temperature 250~900 °C could be transformed into H_2 and CO with final products of H_2O and CO_2 after further oxidation. It removed the Wigner energy in graphite and achieved an efficient separation of graphite from radionuclides in a slowly controlled process. When it is adopted to condition the graphite from reactor cores, the gases could not be released into the atmosphere due to the ^{14}C content. Thus, it could constrain the application as the incineration process. In addition, many researchers have been investigating the subsequent procedures to transform the gaseous products into the solid of carbon with the objective of no gaseous emissions, which requires some complicated mechanical and chemical engineering equipments [23] but could fundamentally eliminate the potential influences on human health and the environment.

It is difficult and expensive to handle the separation of ^{14}C from large quantities of off-gas, and the gasification process seems only advantageous for the graphite waste without ^{14}C included with no cheap procedure available at present.

6.2 Direct disposal of graphite

The radioactive waste could be ultimately disposed of in near surface sites or deep geological repositories. The near surface disposal could accommodate the lightly contaminated waste, i.e., low-level radioactive wastes (LLW), and high-level radioactive waste (HLW) should be disposed of in deep geological repositories [24] for the permanent removal from the environment as solid graphite, an option which is presently unpopular with the public. Some disadvantages could present themselves for the direct disposal of radioactive waste, because this method requires a large capacity of the repository in the case of great amounts of accumulating waste to be disposed of and terminates the graphite as the final waste. The other method, dumping low-level radioactive graphite into the ocean, was once to be recognised as a most acceptable option to dispose of the radioactive graphite waste. Unfortunately, the option of dumping radioactive waste into sea has been prohibited due to the historical dumping of radioactive waste adjacent to shores which led to public opposition [23].

Graphite is already a solid material, and no large volume minimization could be expected with the disposal as solid graphite waste, which is still of great advantage compared with

the great increase of volume for the final carbonate. The disposal of graphite in its original form has the special advantages of avoiding the diffusion of radioactive ^{14}C to the environment, and of keeping the volume of radioactive waste constant or even providing a little reduction. The graphite material from the reactor cores of graphite moderated reactors being decommissioned are characterised by large volume and high radioactivity, and several countries are likely to choose direct disposal in a repository after proper coating or encapsulation [25], especially for the deep geological repositories. Japanese scientists have carried out corresponding investigations which demonstrated that the cost of disposal could be significantly reduced through some judicious cutting actions of the graphite blocks. Attention should also be paid to the segregation of graphite material from the steel due to the probable galvanic coupling between steel and graphite under aerobic conditions, which could be simply overcome by the introduction of moderator of cement between the two materials or employing the nonmetallic material instead of the steel [26].

To obtain irradiated graphite as the LLW, it is of paramount importance to protect the graphite material from the nuclear contamination, because it is much more difficult and costly to condition the highly contaminated wastes. For example, various measurements should be carried out to assist the graphite waste from HTGR spent fuel elements to be classified as less contaminated as possible, from the manufacture process of fuel elements and the retrieval procedure of graphite waste from spent fuel elements [7].

6.3 Decontamination

In High Temperature Reactors, three kinds of graphite have to be examined for decontamination : fuel blocks, permanent reflectors and replaceable reflectors. Graphite from fuel blocks, activated in the bulk by neutrons, cannot be decontaminated. Graphite from replaceable reflectors could be decontaminated after to be retired from the reactor. This decontaminated graphite could be reused in the reactor, but the main goal of the decontamination operation is the reduction of the graphite waste inventory. Graphite from permanent reflectors could be decontaminated several times. The first ones (frequency is to define) would be maintenance operations. These operations must absolutely preserve the functionality of the material. The last one would have the same main objective than the replaceable reflectors graphite decontamination operation: reduction of the graphite waste inventory. The choice of the decontamination process depends on the contamination characteristics: either the contamination is on the surface of the graphite, or the contamination has migrated into the graphite because of his high porosity. In the first case, processes leading to the graphite surface erosion could be chosen. The erosion phenomenon can be chemical or physical. In the case of migration of contamination into the material, decontamination with an electrokinetic process could be envisaged. This kind of process is already used for the dechlorination of civil engineering structures like bridges. It allows the extraction of mobile radionuclides from porous media, with an

electric field.

7. CONCLUSIONS

The main purpose of separation of the graphite from fuel element is to allow the graphite to be disposed as LLW.

No comprehensive process is available today that meets the criteria imposed for recycling the actinides and for managing the waste and off-gas releases.

For China, the most direct, perhaps the most satisfactory, and certainly the least expensive option is to dispose the fuel as whole blocks.

For separation of the fuel from the graphite by physical means or by combustion, disposal of the remaining graphite or the "fixed" ^{14}C remains a problem, and the total volume of waste will increase.

Disposal with dissolution of fuel will be used when it is desirable to recover the fuel for recycling

REFERENCES

[1] VERFONDERN K. and VON LENSA W., "Past and Present Research in Europe on the Production of Nuclear Hydrogen with HTGR"; Progress in Nuclear Energy, **47**, *pp.* 472-483 (2005)

[2] ONUKI K., INAGAKI Y., HINO R. and TACHIBANA Y., "Research and Development on Nuclear Hydrogen Production Using HTGR at JAERI"; Progress in Nuclear Energy, **47**, *pp.* 496-503 (2005)

[3] TANG C.H., TANG Y.P., ZHU J.G., ZOU Y.W., LI, J.H. and NI X.J., "Design and Manufacture of the Fuel Element for the 10 MW High Temperature Gas-Cooled Reactor"; Nuclear Engineering and Design, **218**, *pp.* 91-102 (2002)

[4] MASSON M., GRANDJEAN S., LACQUEMENT J., BOURG S., DELAUZON J.MM and LACOMBE J., "Block-Type HGTR Spent Fuel Processing: CEA Investigation Program and Initial Results", Nuclear Engineering and Design, **236**, *pp.* 516-525 (2006)

[5] NOTZ K.J., "An Overview of the HTGR Fuel Cycle"; Oak Ridge National Laboratory Report ORNL-TM-4747 (1976)

[6] EUROPEAN COMMISSION, "Thorium as a Waste Management Option" , EUR 19142 Report (2000)

[7] YOUNG D.T., "Fluidized Combustion of Beds of Large, Dense Particles in Reprocessing HTGR Fuel, General Atomics ReportGA-A13174 t (1977)

[8] DELAGE F., LATGE C. and THOUVENOT P.,"Overview of Graphite Treatment Processes in High Temperature Reactors"; 2nd International Topical Meeting on High Temperature Reactor Technology, Beijing, China (2004)

[9] DEL CUL G.D., SPENCER B.B., FORSBERG C.W., COLLINS E.D. and RICKMAN, W.S., "Triso-Coated Fuel Processing to Support High-Temperature Gas-Cooled Reactors"; UT-Battelle, Oak Ridge National Laboratory Report ORNL/TM-2002/156 (2002)

[10] LOTTS A.L., BOND W.D., FORSBERG C.W., GLASS R.W., HARRINGTON, F.E., MICHAELS G.E., *et al.,* "Options for Treating High-Temperature Gas-Cooled Reactor Fuel for Repository Disposal"; Oak Ridge National Laboratory Report ORNL/TM-12027, UC-522,-810,-811 (1992)

[11] DAVIS W.J., "Carbon-14 Production in Nuclear Reactors,"; Management of Low-Level Radioactive Wastes"; **1**, *pp.* 151-191 (1979)

[12] GRENECHE D. and SZYMCZAK W.J., "The Areva HTR Fuel Cycle: An Analysis of Technical Issues and Potential Industrial Solutions"; Nuclear Engineering and Design, **236**, *pp.* 635-642 (2006)

[13] FUTTERER M.A., "Head-End Process for the Reprocessing of Reactor Core Material"; The Netherlands, WO 2006/087360Al (2006)

[14] TIAN L.F., WEN M.F., LI L.Y. and CHEN J., "Disintegration of Graphite Matrix from the Simulative High Temperature Gas-Cooled Reactor Fuel Element by Electrochemical Method"; Electrochimica Acta, **54**, *pp.* 7313-7317 (2009)

[15] ANDRES, U., "Parameters of Disintegration of Rocks by Electrical Pulses"; Powder Technol. **58**, *pp.* 265–269 (1989)

[16] MAEDA M., and Yagi E., "Feasibility Study for Removal of SiC Coating in HTGR-Type Fuel by F2-Fluorination"; Japan Atomic Energy Research Institute Report JAERLM7060 (1977)

[17] INTERNATIONAL ATOMIC ENERGY AGENCY, "The Principles of Radioactive Waste Management"; IAEA Vienna (1995)

[18] MASON J.B. and BRADBURY, D., "Pyrolysis and Its Potential Use in Nuclear Graphite Disposal"; Nuclear Energy, **39**, *pp.* 305-310 (2000)

[19] LUO S.G., "Incineration Treatment of Radioactive Wastes"; Nuclear Techniques, **13**, *pp.* 1-8 (1990 in Chinese).

[20] BRAUN H., GUTOWSKI H., BONKA H. and GRUNDLEN D., "Plant for Retention of ^{14}C in Reprocessing Plants for LWR Fuel Elements"; 17th DOE Nuclear Air Cleaning Conference, *pp.* 381-399 (1983)

[21] FORSBERG C.W., DEL CUL G.D., SPENCER B.B., and COLLINS E.D., "A

New Repository Waste Form: Graphite–Carbon High-Level Waste"; International HLW Management Conference, Las Vegas, Nevada, USA (2003)

[22] MASON J.B. and BRADBURY D., "Process for the Treatment of Radioactive Graphite", US Patent 20020064251A1 (2002)

[23] McENANEY B, WICKHAM A.J., and DUBOURG M., "Planning for Disposal of Irradiated Graphite: Issues for the New Generation of HTRs"; Basic Studies in the Field of High-Temperature Engineering, OECD Paris, France, *pp.* 281-291 ISBN 92-64-19796-6 (2002)

[24] RINGWOOD A.E., "Disposal of High-Level Nuclear Wastes: A Geological Perspective"; Mineralogical Magazine, **49**, *pp.* 159-176 (1985)

[25] BRINKMANN H.U., DUWE R., GANSER B., MEHNER A.W. and REBMANN A., "Contributions Towards the Development of a Packaging Concept for the Final Disposal of Spent HTGR Pebble Bed Fuel"; Nuclear Engineering and Design, **118**, *pp.* 107-113 (1990)

[26] WISE M., "Management of UKAEA Graphite Liabilities"; Proceedings of IAEA Technical Committee Meeting on Nuclear Graphite Waste Management, Manchester, UK, IAEA Vienna CD-ROM 01-00120 (2001)