JAERI-M

8063

THE USE OF LOW ENRICHED URANIUM FUEL CYCLE IN HIGH TEMPERATURE GAS-COOLED REACTORS

January 1979

GENFIC Working Group No.1

日 本 原 子 力 研 究 所
Japan Atomic Energy Research Institute

The Use of Low Enriched Uranium Fuel Cycle in High Temperature Gas-Cooled Reactors

Japan Atomic Energy Research Institute,
GENFIC Working Group No.1*

(Received December 27, 1978)

In Japan, research and development of the experimental multi-purpose VHTR (Very High Temperature gas-cooled Reactor) have been carried out with aiming at application of the reactor to process heat. The experimental reactor employs the LEU fuel cycle because the cycle is compatible with the LWR-FBR line calling for Pu recycle in Japan.

The present paper begins with a brief review of the status of research and development of the experimental VHTR. On the basis of the experience gained from these works and the above philosophy of fuel cycle, core models are constructed for commercial high temperature gas-cooled reactors with thermal output of 3000 MW, and the material balances of the reactor and the fuel cycle system are studied.

Two core models are assumed for comparison; one is a VHTR with an outlet temperature of 1000 °C for process heat application and the other a steam cycle high temperature gas-cooled reactor (HTGR) with an outlet temperature of 750 °C. Material balances are estimated for each model, which are uranium requirements during the life of 30 years, separative work, plutonium productions, generations of fission products (including ^{3}H and ^{14}C). The results are compared with each other and with LWR data. The comparison indicates that the savings of separative work and uranium requirements are larger in VHTR than in HTGR and that uranium requirements and plutonium productions per unit electric output are smaller in these HTR's than in LWR's.

On the principle that the uranium and plutonium recovered from the spent fuels of the HTR's should be recycled to LWR's or FBR's, the integrated system of the fuel cycle consisting of fabrication of fuels, reprocessing and waste management are studied. Material flows of uranium, plutonium, fission products and radioactive waste are then estimated following the flowsheet of the system. In addition, the status and R & D requirements of each process are surveyed and some processes are also assessed from the viewpoints of proliferation resistance and environmental impact. Especially the head-end step of HTR fuel reprocessing is assessed in more detail and covers a possible way of reducing carbon wastes.

Keywords: Low Enriched Uranium, Fuel Cycle, Plutonium, HTGR, VHTR Reactor, Material Balance, Fuel Fabrication, Reprocessing Waste, INFCE, Fission Products.

*Working Group No.1 of Generic Evaluation of Nuclear Fuel Integrated Cycle

Group leader	Junichi Shimokawa	(Head, Div. of Nuclear Fuel Research)
Sub-leaders	Shojiro Matsuura	(Div. of JPDR)
	Koreyuki Shiba	(Div. of Nuclear Fuel Research)
Members	Takehiko Yasuno	(Div. of Power Reactor Projects)
	Yoshiaki Miyamoto	(ditto)
	Ryuiti Shindo	(ditto)
	Yoshihiko Kaneko	(Div. of Reactor Engineering)
	Kazumi Iwamoto	(Div. of Nuclear Fuel Research)
	Takeshi Tsujino	(ditto)
	Mitsuru Maeda	(ditto)
	Kunio Araki	(Div. of Environmental Safety Research)

高温ガス炉における低濃縮ウラン燃料サイクルの利用

日本原子力研究所
核燃料サイクル評価プロジェクト第1ワーキンググループ*

（1978年12月27日受理）

日本では高温ガス炉のプロセスヒートへの適用性に着目し，多目的高温ガス実験炉の研究開発を進めてきた。この多目的高温ガス実験炉では低濃縮ウラン燃料サイクルを採用している。これはプルトニウム・リサイクルを前提とする日本の軽水炉 — 高速増殖炉路線との整合性を重視したからである。

本論文では，はじめに多目的高温ガス実験炉の研究開発の現状について紹介した。ついで，多目的高温ガス実験炉で採られた上述の基本的考え方を踏襲しかつ従来の研究成果を基礎として，熱出力 3,000 MW の商用高温ガス炉を設計し，燃料物質収支の特性および燃料サイクルシステムについて考察した。

その炉心については，比較のため2種類のモデルを設定した。一方は冷却材炉出口温度 750 ℃の発電用高温ガス炉（HTGR）であり，他方は出口温度 1000 ℃ のプロセスヒート用高温ガス炉（VHTR）である。各炉心モデルにおける 物質収支—30年の寿命期間中のウラン必要量，濃縮分離作業量，プルトニウム生成量，核分裂生成物（^{3}H ^{14}Cも含む）生成量 —を計算し，VHTRとHTGRの相互比較および軽水炉との対照を行った。その結果 VHTR の方が HTGR よりもウラン資源の節約，濃縮分離作業量の低減化に優れていること，また単位発電量当りのウラン所要量およびプルトニウムの生成量では高温ガス炉（VHTR および HTGR）の方が軽水炉より著しく小さいことが明らかとなった。

高温ガス炉の使用済燃料から回収されるウランとプルトニウムは軽水炉または高速増殖炉で再利用されるべきであるという基本方針の下に，燃料の製造，再処理，廃棄物処理処分を包含する核燃料サイクルのトータルシステムを構築した。このシステムのフローシートに基いて，ウラン，プルトニウム，核分裂生成物，放射性廃棄物などの物質の移動量を評価した。さらに，各プロセスにおける技術の現状と問題点及びR＆D項目について核拡散防止性，安全性などの見地からも検討した。特に高温ガス炉燃料特有の再処理のヘッドエンドプロセスについてはより詳しく検討し，炭素廃棄物を低減化する方策にも言及した。

* 核燃料サイクル評価プロジェクト第1ワーキンググループ
グループリーダー　下川純一（燃料工学部長）
グループリーダー補佐　松浦祥次郎（動力試験炉部）　柴　是行（燃料工学部）
グループ員　安野武彦（動力炉開発・安全性研究管理部）　宮本喜晟（動力炉開発・安全性研究管理部）　新藤隆一（動力炉開発・安全性研究管理部）　金子義彦（原子炉工学部）　岩本多実（燃料工学部）　辻野　毅（燃料工学部）　前田　充（燃料工学部）　荒木邦夫（環境安全研究部）

Contents

目　　次

1. Introduction

A high temperature gas-cooled reactor (HTR) is known to have many advantages such as uranium resource saving, low environmental impact and high inherent safety. Especially in Japan, attention was given to a possibility of its application to process heat. A development program of very high temperature gas-cooled reactor (VHTR) was started in Japan Atomic Energy Research Institute (JAERI) in 1969. Since that time, design, research and development of the experimental multi-purpose VHTR have been carried out in cooperation with other research institutions.

A low enriched uranium (LEU) fuel cycle was chosen for the VHTR because the cycle was compatible with the LWR-FBR line calling for Pu recycle in Japan. Moreover, the production of Pu, which is characteristic of the LEU fuel cycle, should be assessed from the viewpoint of efficient utilization of Pu, because the Pu recycle is an integral part of saving nuclear resources.

The present paper begins with a brief review of the status of research and development of the experimental VHTR in Japan. On the basis of the experience gained from these works, assessment is made of commercial HTR's. Material balance with fuel burnup is calculated for the two core models; one is HTGR for steam cycle and the other VHTR for process heat application. A typical flow chart of the fuel cycle is proposed; each section of the fuel cycle presents features of process, option, proliferation resistance and research and development work requirements. The results of assessment of commercial HTR's are compared with those for LWR.

2. Research and development works for the experimental VHTR

Research and development have continued, aiming to construct the experimental multi-purpose VHTR by about 1985. Design works of the experimental VHTR plant have proceeded for past several years. Preliminary and conceptual designs were finished in 1972 and 1976 respectively. Now more detailed design and safety analysis of the plant are continuing. Table 2-1 shows design conditions of the conceptual design of the experimental VHTR. The essential problems of

1. Introduction

A high temperature gas-cooled reactor (HTR) is known to have many advantages such as uranium resource saving, low environmental impact and high inherent safety. Especially in Japan, attention was given to a possibility of its application to process heat. A development program of very high temperature gas-cooled reactor (VHTR) was started in Japan Atomic Energy Research Institute (JAERI) in 1969. Since that time, design, research and development of the experimental multi-purpose VHTR have been carried out in cooperation with other research institutions.

A low enriched uranium (LEU) fuel cycle was chosen for the VHTR because the cycle was compatible with the LWR-FBR line calling for Pu recycle in Japan. Moreover, the production of Pu, which is characteristic of the LEU fuel cycle, should be assessed from the viewpoint of efficient utilization of Pu, because the Pu recycle is an integral part of saving nuclear resources.

The present paper begins with a brief review of the status of research and development of the experimental VHTR in Japan. On the basis of the experience gained from these works, assessment is made of commercial HTR's. Material balance with fuel burnup is calculated for the two core models; one is HTGR for steam cycle and the other VHTR for process heat application. A typical flow chart of the fuel cycle is proposed; each section of the fuel cycle presents features of process, option, proliferation resistance and research and development work requirements. The results of assessment of commercial HTR's are compared with those for LWR.

2. Research and development works for the experimental VHTR

Research and development have continued, aiming to construct the experimental multi-purpose VHTR by about 1985. Design works of the experimental VHTR plant have proceeded for past several years. Preliminary and conceptual designs were finished in 1972 and 1976 respectively. Now more detailed design and safety analysis of the plant are continuing. Table 2-1 shows design conditions of the conceptual design of the experimental VHTR. The essential problems of

Table 2-1 Design Conditions of the Experimental Multi-purpose VHTR

	Main items	Basic design conditions
1	Reactor thermal output (nominal)	500 MW
2	Reactor outlet coolant temperature	1,000°C
3	Reactor inlet coolant temperature	400°C
4	Fuel	UO_2 kernel, coated fuel particle, graphite matrix, dispersion-type
5	Fuel element type	Pin-in-block type
6	Direction of coolant flow	Downward-flow through the core
7	Pressure vessel	Steel
8	Number of primary coolant circuits	2
9	Heat transmission	Indirect (Adoption of IHX)
10	Primary coolant pressure	40 kg/cm^2G
11	Secondary coolant pressure	Higher than that of the primary coolant
12	Components in the secondary circuit	Heat removal components Heat utilizing components
13	Containment	Reactor and primary cooling system are adopted in the containment building.

VHTR must be solved through design, construction, and operation of the experimental multi-purpose VHTR.

The research and development works have extended to such areas as fuel, graphite materials, structural metals, fluid dynamics-heat transfer, reactor physics, reactor engineering and irradiation techniques. Two out-of-pile helium gas loops operating at 1,000 °C have contributed to high temperature technology in Japan Atomic Energy Research Institute. An in-pile helium gas loop, Oarai Gas Loop No-1 (OGL-1), of which the operating temperature is 1,000 °C, was installed in Japan Material Test Reactor (JMTR), and has played an important role in high temperature irradiation tests since 1977. And construction of HENDEL facility will start this year. HENDEL (standing for High Temperature-Helium Engineering Demonstration Loop) is used for tests of the large components of the experimental VHTR.

These experiences will raise the technical potentials for a commercial VHTR.

3. Models of commercial VHTR and HTGR for evaluation

3.1 Design basis for commercial reactor core

In order to evaluate fuel material balance of the commercial reactor core of 3,000 MWt output using the low enriched uranium fuel, two core models are provided: one is the multi-purpose commercial very high temperature gas cooled reactor (VHTR) and the other is the steam cycle high temperature gas cooled reactor (HTGR). The core models shown in Table 3-1 have been chosen to fit the design basis as follows.

(1) The experimental VHTR core is extrapolated into the commercial core to meet the requirement of 3,000 MWt output with design features kept same.

- Low enriched uranium oxide is used for the fuel and the particle is TRISO coating with 600 μm kernel diameter.
- A fuel element is a prismatic fuel of pin-in-block type.
- Coolant is controlled by each flow control assembly which is located at the inlet to a seven-fuel-column region; it flows downward in the core.
- Access for refuelling is through an array of penetrations of the reactor vessel in off-load operation.
- Two control rods are arranged in each center column of a

VHTR must be solved through design, construction, and operation of the experimental multi-purpose VHTR.

The research and development works have extended to such areas as fuel, graphite materials, structural metals, fluid dynamics-heat transfer, reactor physics, reactor engineering and irradiation techniques. Two out-of-pile helium gas loops operating at 1,000 °C have contributed to high temperature technology in Japan Atomic Energy Research Institute. An in-pile helium gas loop, Oarai Gas Loop No-1 (OGL-1), of which the operating temperature is 1,000 °C, was installed in Japan Material Test Reactor (JMTR), and has played an important role in high temperature irradiation tests since 1977. And construction of HENDEL facility will start this year. HENDEL (standing for High Temperature-Helium Engineering Demonstration Loop) is used for tests of the large components of the experimental VHTR.

These experiences will raise the technical potentials for a commercial VHTR.

3. Models of commercial VHTR and HTGR for evaluation

3.1 Design basis for commercial reactor core

In order to evaluate fuel material balance of the commercial reactor core of 3,000 MWt output using the low enriched uranium fuel, two core models are provided: one is the multi-purpose commercial very high temperature gas cooled reactor (VHTR) and the other is the steam cycle high temperature gas cooled reactor (HTGR). The core models shown in Table 3-1 have been chosen to fit the design basis as follows.

(1) The experimental VHTR core is extrapolated into the commercial core to meet the requirement of 3,000 MWt output with design features kept same.

- Low enriched uranium oxide is used for the fuel and the particle is TRISO coating with 600 μm kernel diameter.
- A fuel element is a prismatic fuel of pin-in-block type.
- Coolant is controlled by each flow control assembly which is located at the inlet to a seven-fuel-column region; it flows downward in the core.
- Access for refuelling is through an array of penetrations of the reactor vessel in off-load operation.
- Two control rods are arranged in each center column of a

Table 3-1 Main Parameters of Low-enriched Uranium Core

Items	Designs	VHTR	HTGR
1. General parameters			
Thermal output	(MW)	3,000	3,000
Electric power generation	(MW)		(1,200)
Reactor outlet coolant temp.	(°C)	1,000	750
Reactor inlet coolant temp.	(°C)	400	400
Coolant pressure	(kg/cm^2)	40	40
2. Core			
Equivalent core diameter	(m)	10.3	8.4
Core height	(m)	6	6
Number of fuel elements in a fuel column		7	7
Power density	(MW/m^3)	6	9
Specific power	(MW/T)	81.8	122.6
3. Fuel element			
Type of fuel element		hex-block hollow pin	hex-block hollow pin
Width across flats	(mm)	410	410
Height	(mm)	860	860
Fuel pin inner diameter	(mm)	20	20
Fuel pin outer diameter	(mm)	42	42
Pin hole diameter	(mm)	50	50
4. Fuel parameters			
Fuel inventory	(T_{HM})**	36.7	24.5
Enrichment	(w/o)	6.0	6.5
Fuel burnup	(MWD/T)	62,800	62,300
Conversion ratio(BOL*/Average)		0.38/0.55	0.36/0.52
Residence time	(Y)	3	2
Plant factor	(%)	70	70
Plant life	(Y)	30	30

* Beginning of Life ** Ton of heavy metals

seven-fuel-column region and these drive mechanisms are in the reactor penetrations above the core.

(2) The maximum fuel temperature is 1,400 °C in normal operation and a monolithic fuel pin is usable.

(3) As target, the fuel burnup per unit enrichment and the natural uranium requirements of VHTR and HTGR are non inferior to those of LWR.

3.2 Considerations in choosing main parameters

(1) Reactor coolant temperature

In the VHTR system, the helium coolant temperature at reactor outlet is an important parameter, which is closely related to nuclear process heat application. The outlet temperature of 1,000 °C is recommended to meet the requirement of iron making process using an intermediate heat exchanger loop. In the HTGR system, a survey of the past design studies shows the outlet temperature to be in the range of 760~790 °C. The outlet temperature of 750 °C is adopted to clarify the difference between HTGR and VHTR cores.

The increase of the inlet temperature brings about high flow rate and improves the heat transfer performance in the core. At the same time, the pressure loss in the loop decreases the plant efficiency and there also is a certain limit to the circulator. The inlet temperature should be considered comprehensively as above, but the temperature of 400 °C is chosen as in the experimental VHTR.

(2) Fuel element type and power density

Many types of hexagonal block are possible for prismatic fuel. The pin-in-block type fuel can be used for the commercial reactors, as shown in Fig.3-1.

As a result of the surveys using an exponential axial power distribution to lower fuel temperature, the hollow pin type fuel has attained a core average power density of about 6 MW/m^3 and the annular (tubular) pin type fuel about 8 MW/m^3 in the VHTR system. In addition, to attain fuel residence time of more than 3 years, the hollow pin with power density 6 MW/m^3 is chosen.

In the HTGR system, the fuel configuration is identical with the VHTR's and the power density is 9 MW/m^3.

(3) Fuel charge and discharge

The atomic ratio of carbon to uranium, Nc/Nu, in the cores is

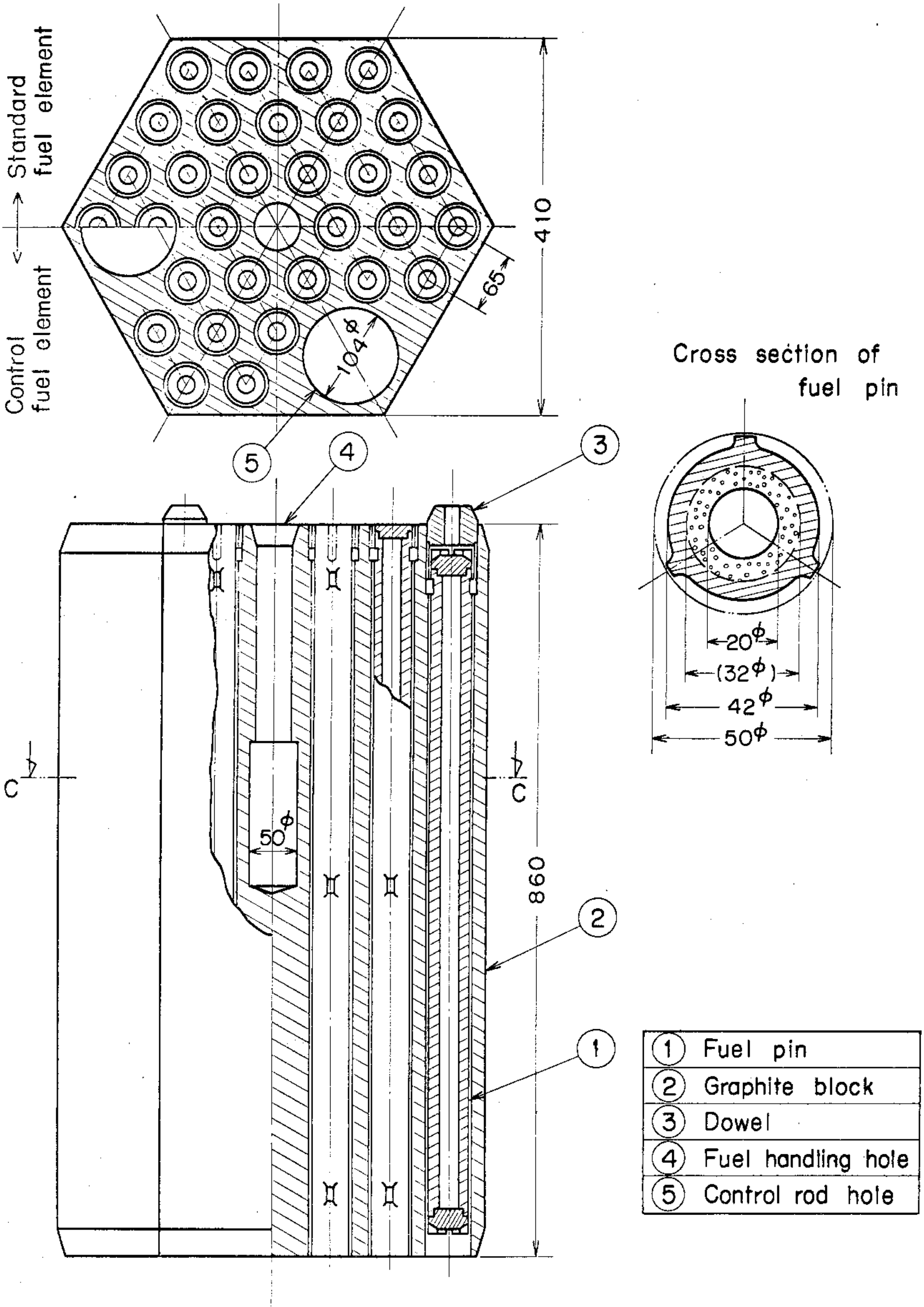

Fig. 3-1 Fuel Element Concept for Commercial VHTR and HTGR

selected to be about 350 to attain similar fuel burnup and natural uranium requirement to those of LWR and also to have as long a fuel residence time as possible.

The fuel charge and discharge program is determined from considerations of age peaking factor and fuel charge rate.

(4) Reactivity control

Reactivity control of LEU-HTR is not so easy because of the necessity of high initial excess reactivity owing to low conversion ratio.

The reactivity for fuel burnup, therefore, is suppressed mainly by burnable poison rods encaged in the fuel elements and control rods are used principally to regulate the reactivity variation due to temperature rise and Xe & Sm buildup.

3.3 Calculation of material balance

Calculations were made to evaluate the material balance of HTR's using the fresh low enriched uranium fuel.

Characteristics of the VHTR and the HTGR described previously are shown in Table 3-2. Features in material balance of these reactors are:

- Similar fuel burnup and conversion ratio.
- Larger fuel core inventory in VHTR (about 50 %) but similar fuel charge rate.
- Lower cumulative requirement of natural uranium in VHTR (about 10 %).
- Lower cumulative requirement of separative works in VHTR (about 10 %).
- Similar fissile plutonium production.
- Lower radioactivity in the spent fuels from VHTR (about 30 %).

As compared with LWR's characteristics of LEU-HTR's are:

- Smaller cumulative fuel loading (about half).
- Similar cumulative requirement of natural uranium per reactor but lower requirement per unit electric output (more than about 10 %).
- Higher cumulative requirement of separative works per reactor (about 30~45 %) but similar requirement per unit electric output.
- Smaller cumulative production of fissile plutonium (about 40 %).
- Higher radioactivity of spent fuels per metal ton (about 2~3 times).

From material balance studies, it is concluded that lower power density (decrease of neutron flux level and increase of fuel residence time) is advantageous in design of LEU-HTR.

Table 3-2 Specification of Fuel Material Balance in LEU-HTR

Items \ Designs	VHTR	HTGR
Fuel loading (T_{HM})		
Initial core	36.7	24.5
Annual reload	12.2	12.2
Fuel discharge (annual) (T_{HM})		
Uranium	11.2	11.2
(fissile nuclide content, w/o)	(1.4)	(1.6)
Plutonium	0.157	0.162
(fissile nuclide content, w/o)	(60)	(65)
Natural uranium requirement* (T_{HM})		
Initial core	454	330
Annual reload	151	165
30-Year cumulation	4,682	5,115
Separative work requirement* (T-SWU)		
Initial core	367	271
Annual reload	122	136
30-Year cumulation	3,783	4,215
Radioactivity in annually discharged fuels (Ci/T_{HM})	(Actinides) / (FP)	(Actinides) / (FP)
Cooling time 0 Y	1.0×10^8 / 4.2×10^8	1.5×10^8 / 6.2×10^8
0.5 Y	3.8×10^5 / 9.8×10^6	3.4×10^5 / 1.3×10^7
1 Y	3.4×10^5 / 5.4×10^6	3.1×10^5 / 6.5×10^6
5 Y	2.6×10^5 / 1.0×10^6	2.4×10^5 / 1.1×10^6
10 Y	2.1×10^5 / 6.8×10^5	1.9×10^5 / 7.0×10^5

* Tail enrichment --- 0.25 %

4. Characteristics of the HTR fuel cycle

The fuel elements of HTR's are simple in material but complex in structure. This gives rise to variety of options at the back-ends. A typical flow chart of the fuel cycle for the LEU-HTR's is shown in Fig.4-1. The basic idea is that the plutonium and uranium recovered from spent fuels of the HTR's are recycled to LWR or FBR's.

Three sections below are for fabrication, reprocessing and waste management of the fuel cycle, respectively; features of process, options (if any), proliferation resistance, safety, research and development work requirements are described. Emphasis is placed on characteristics of the HTR fuel cycle, because many back-end technologies are common to LWR's. In addition, the material balances are estimated for the VHTR, unless otherwise stated, since there is little difference in annual charge and discharge of fuels between the VHTR and the HTGR. Another section is for compositions of the spent fuels.

The integrated system of the fuel cycle is schematically shown in Fig.4-2 with the material balances estimated.

4.1 Fabrication

(1) Mass flow

A mass flow of the fabrication process is shown schematically in Fig.4-3; an initial charge and an annual make-up of the coated particle fuel with low enriched uranium for the VHTR and HTGR are described. No essential difference is seen in the annual amount of the make-up fuel between both the reactors.

(2) Technological feasibility

(i) Fuel kernel: The UO_2 kernel with 600 μm diameter can be produced by conventional techniques such as sol-gel process and powder agglomeration. Improvement of the kernel is necessary, however, to suppress fission product release and amoeba effect.

(ii) Coated particle: The kernel is coated with pyrolytic carbon and silicon carbide to complete the TRISO particle. The fuel particles will be used at a nominal maximum temperature of 1400 °C, which is appreciably higher than the temperature 1250 °C for power generation HTGR. The rise in temperature requires further irradiation tests to assess performance of the particles. In the TRISO coating, the use

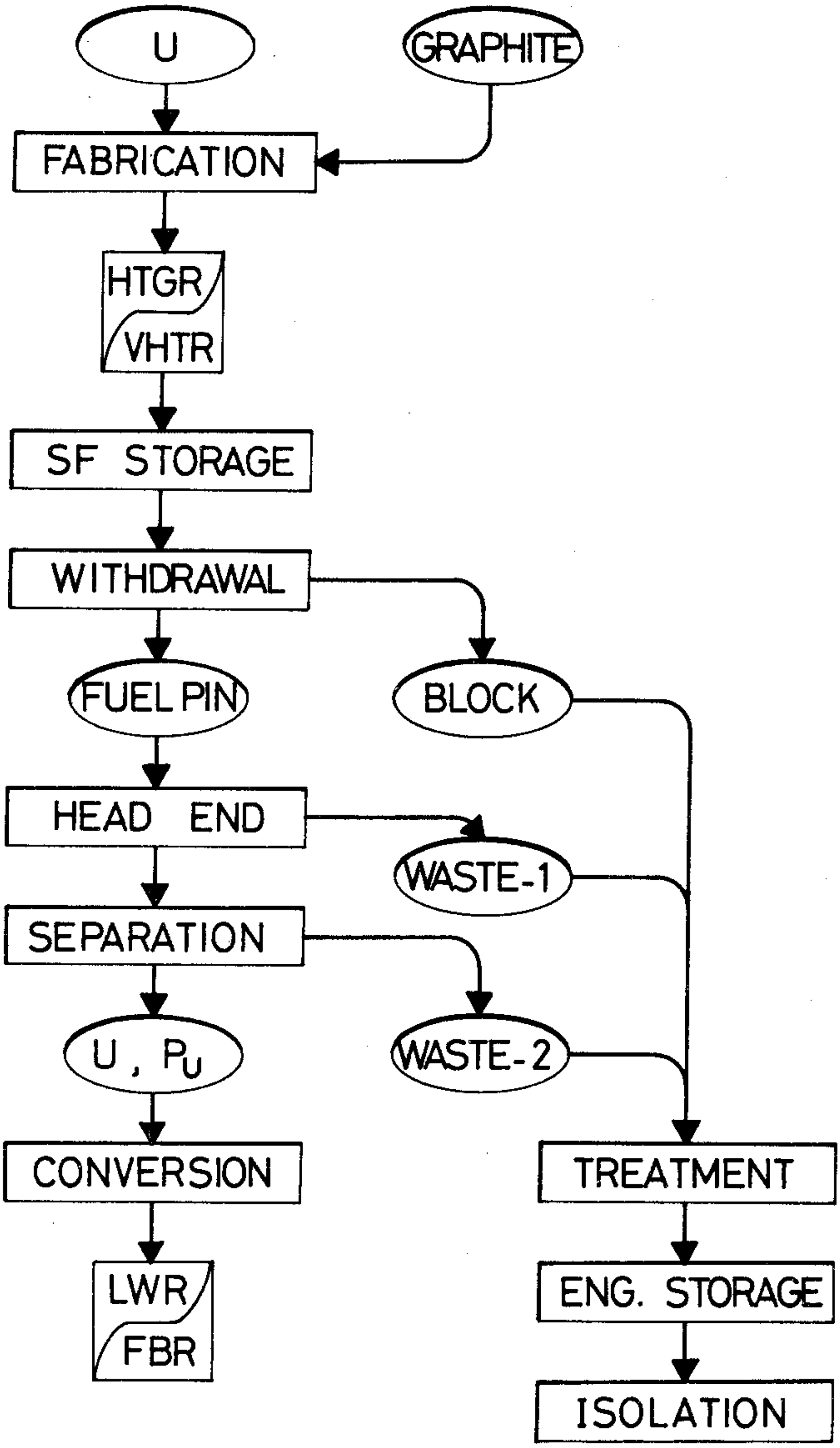

Fig. 4-1 Typical Flow Chart of LEU-HTR Fuel Cycle

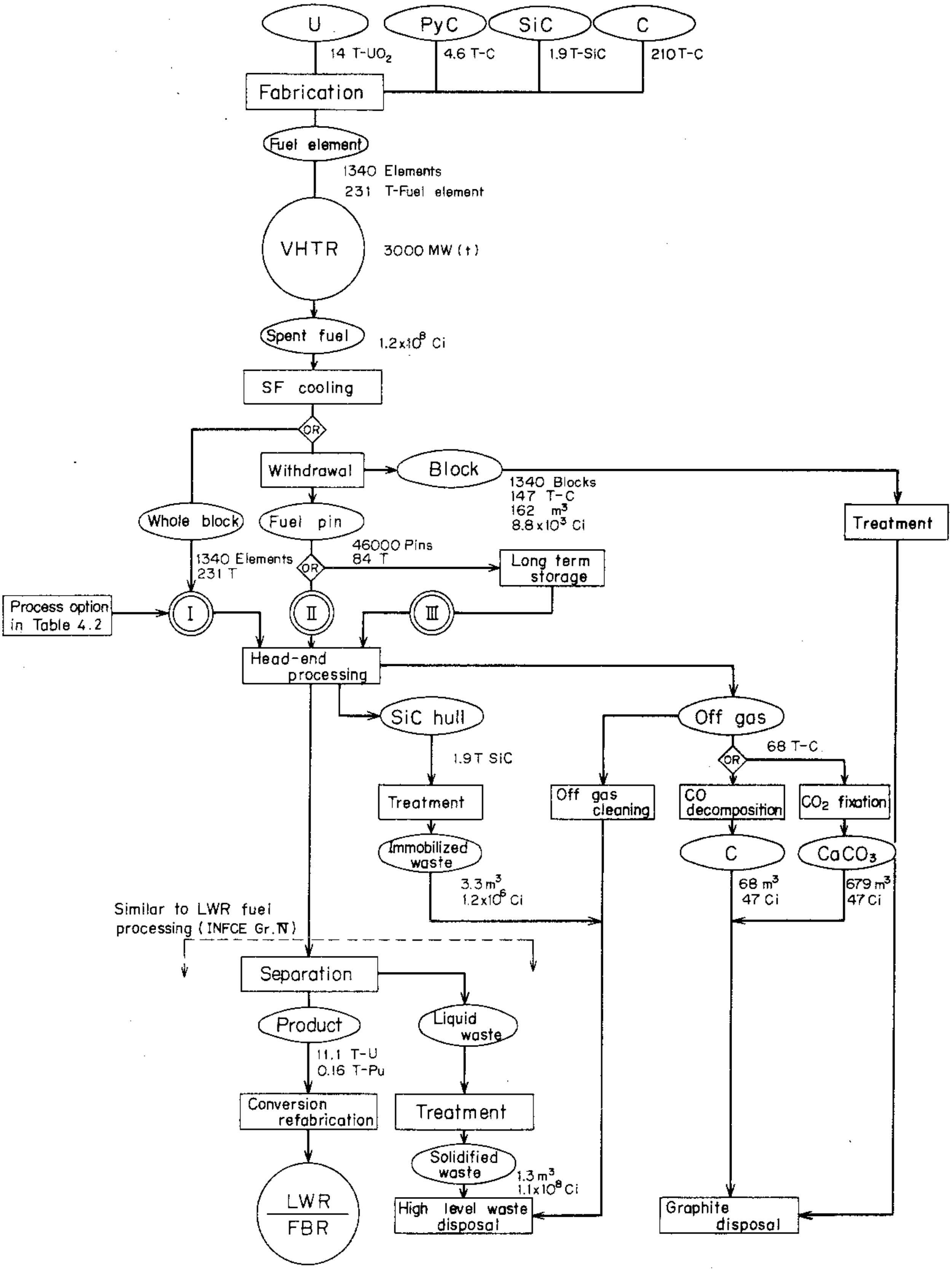

Fig. 4-2 Fuel Cycle Flowsheet and Material Balance for 3000 MW(t) VHTR

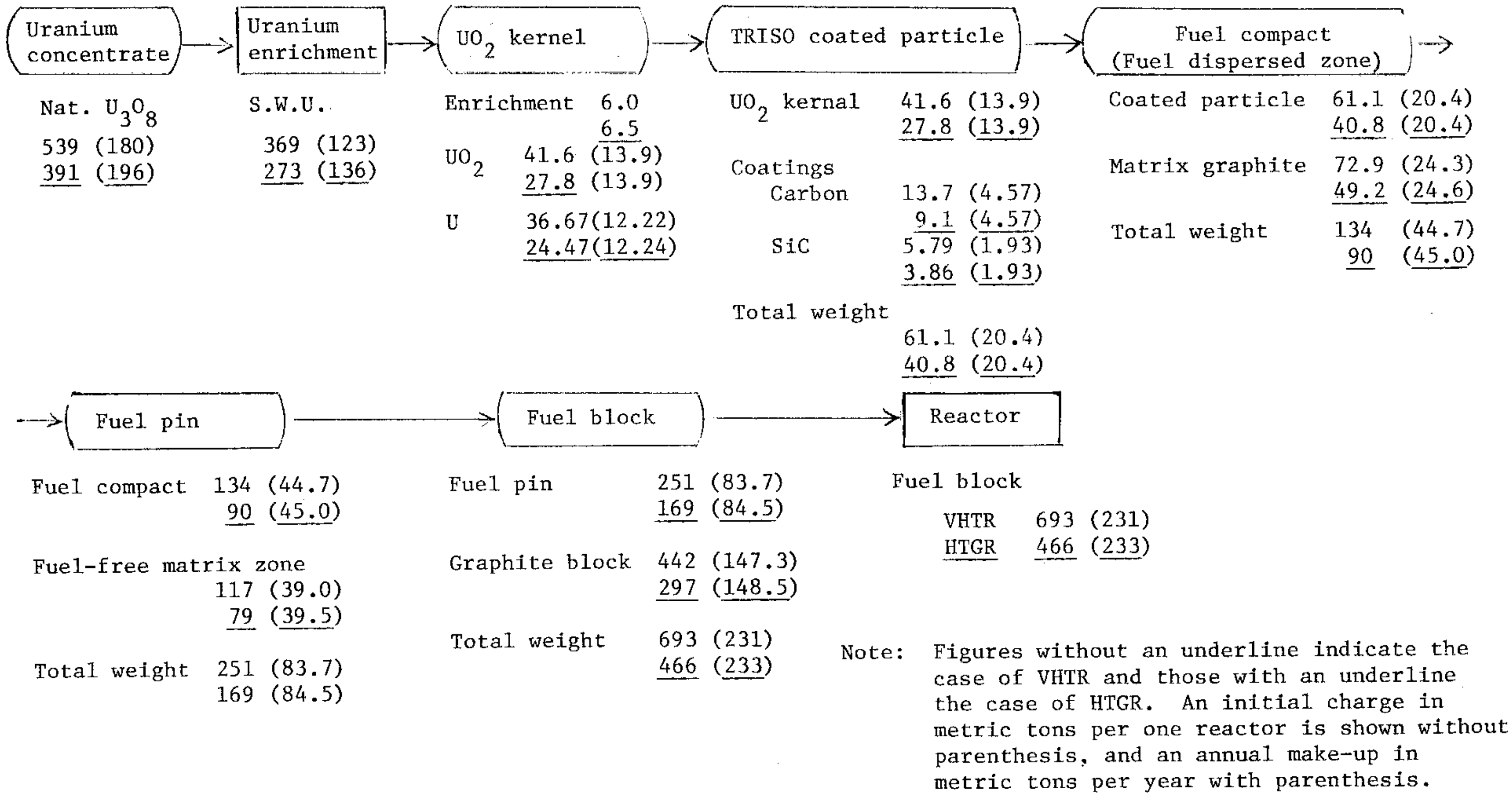

Fig. 4-3 Mass Flow of Fabrication Process for 3,000 MWt VHTR and HTGR

of zirconium carbide in place of the silicon carbide may improve thermal stability of the coating. However, an extensive work including irradiation is required to realize this improvement.

(iii) Fuel pin: A directly cooled monolithic fuel pin with no gap between the compact (fuel dispersed zone) and the graphite sleeve (fuel-free matrix) may be produced by isostatical rubber pressing or extrusion of the mixture of particle, graphite powder and binder, either followed by carbonization, degassing and mechanical finish. Fabrication experiences of this type of fuel pins are none yet. Also required are further studies on the corrosion resistance and the irradiation stability.

(iv) Fuel block: The fuel pins are inserted into a prismatic graphite block to complete the fuel block. Much study is required to choose graphite material which endures the high temperature in the reactor; an isostatically pressed isotropic graphite commercially available in Japan is one of the materials being investigated. The establishment of NDT techniques for testing internal imperfections and cracks of the graphite material is required to ensure the internal integrity.

4.2 The characteristics of HTR spent fuels

The HTR's which make use of the fuel element of graphite both as structural material and moderator, release a substantial amount of graphite waste (See Table 4-1). As to the composition of the spent fuels, the HTGR and VHTR do not make a great difference. The concentrations of transuranium and fission products in the spent fuels are higher in the HTR's owing to the higher burnup than in LWR's while the total inventories per reactor are comparable to each other. On the other hand, the use of graphite leads to the production of a signicicant amount of ^{14}C. This is an outstanding feature of the fuel cycle.

In selecting a process at the back-ends, it is essential to take into consideration the distribution of radionuclides in the fuel elements as well as their total amounts. The distribution is mainly caused by failure of coated particles and has a decisive effect on the process choice.

Table 4-1 Annual Discharge and Compositions of Spent Fuels

Items \ Designs		VHTR	HTGR
Reactor performance			
Thermal output	(MW)	3,000	3,000
Fuel burnup	(MWD/T)	62,800	62,300
Fuel residence time	(Y)	3	2
Annual discharge			
Fuel pin	(m^3/Y)	42.4 (84)*	42.5 (85)*
Block	(m^3/Y)	84.2 (147)	84.8 (149)
Total	(m^3/Y)	126.6 (231)	127.3 (234)
Heavy metal	(T/Y)	11.38	11.39
Composition of HM**			
U	(kg/T_{HM})	984	984
Pu	(kg/T_{HM})	14.8	15.0
Am - 241	(kg/T_{HM})	0.164	0.123
Cm - 242	(kg/T_{HM})	0.0145	0.0097
Composition of radionuclide**			
H-3 (fission)	(Ci/T_{HM})	9.7×10^{2}	9.7×10^{2}
H-3 (from Li)	($Ci/T_{Gr.}$)	2.4×10^{1}	2.5×10^{1}
C-14	($Ci/T_{Gr.}$)	5.6×10^{-1}	5.3×10^{-1}
Kr-85	(Ci/T_{HM})	2.0×10^{4}	2.1×10^{4}
I-129	(Ci/T_{HM})	7.5×10^{-2}	7.4×10^{-2}
I-131	(Ci/T_{HM})	3.7×10^{-1}	5.6×10^{-1}
Sr-90, Y-90	(Ci/T_{HM})	1.5×10^{5}	1.6×10^{5}
Zr-95	(Ci/T_{HM})	6.1×10^{5}	9.2×10^{5}
Nb-95	(Ci/T_{HM})	1.2×10^{6}	1.8×10^{6}
Ru-103, Rh-103	(Ci/T_{HM})	1.4×10^{5}	2.1×10^{5}
Ru-106, Rh-106	(Ci/T_{HM})	8.8×10^{5}	9.8×10^{5}
Cs-137	(Ci/T_{HM})	2.2×10^{5}	2.2×10^{5}
Total FP's	(Ci/T_{HM})	9.8×10^{6}	1.3×10^{7}

* Annual discharge in terms of T/Y.

** Cooling time 180 days.

4.3 Reprocessing

(1) Technology status and possible options

Because of the unique configuration of the graphite type fuel, the back-end of HTR-U fuel cycle requires different technology from LWR's; i.e. handling and processing of graphite in huge volume and coated particle fuel. For the LEU-HTR, the "Purex process" developed for U-Pu thermal reactors, is usable, once the fuel is dissolved after decladding. Therefore, technological problems are mainly in such as fuel storage, transportation and head-end of reprocessing.

The head-end processing, which presents most technically difficult problem, has been developed to a cold-pilot stage in U.S.A.[1] and FRG,[2] and the demonstration with hot pilot plants is now awaited. The procedure consisits of mechanical crushing of the whole fuel block, burning the crushed powder with air to remove the graphite matrix as CO_2, and breaking the SiC coated particle if necessary, followed by reburning the decladded fuel and its dissolution. Head-end off-gas, mostly CO_2, is planned to be purified and finally solidified as $CaCO_3$ in control of the ^{14}C release.

In order to mitigate the burden in handling fuel element with large volume and generation of large amount of graphite waste, there are some processing alternatives. Possible options for VHTR "Pin-in-Block" type fuel are:

(i) Dismantling of the fuel block, followed by processing of the withdrawn fuel pins.

(ii) Dismantling of the fuel block and long-term storage, followed by delayed reprocessing.

In graphite removal from the fuel pins and off-gas fixation, other options are possible than burning-solidifying as $CaCO_3$. Although such alternatives as physical[3] and electrochemical disintegrating[4] concepts without burning were proposed, their development efforts were discontinued before their technical feasibilities were fully confirmed, probably because of the process instabilities and the generation of highly contaminated graphite waste. A feasibility study is under way on another alternative process in Japan.[10] Detailed description of the process is given below.

These processes are summarized in Table 4.2, showing their technical statuses and development needs. Their technical features of safety and proliferation resistance are also studied preliminarily.

Table 4-2 Major Options of Backend of HTR Fuel Cycle and their Characteristics

Option	Process description	Technology status	Preliminary evaluation of[a)] Proliferation resistance	Preliminary evaluation of[a)] Environmental and safety	Required R & D
I	Burn whole block with air. Effluent CO_2 is fixed as $CaCO_3$.	Cold pilot test	+ b)	— c)	- Simplification of crushing step. - $^{14}CO_2$ fixation technology.
II	Burn fuel pin alone with CO_2 after dismantling fuel block. Formed CO is decomposed to carbon and CO_2, which is recycled for burning.	Laboratory scale research	+ b)	— c)	- Disassembling technology of fuel block. - Reactor development for graphite gasification with CO_2. - Confinement technology of decomposed carbon fines.
III	Dismantle fuel block to make long-term storage of fuel pin alone, which will be reprocessed later.	Conceptual stage	+ b)	— c)	- Disassembling technology of fuel block. - Safety evaluation on long-term storage of spent fuels.

a) Compared with the current reprocessing technology for LWR fuel. Mark "+" and "-" indicate positive and negative effect, respectively.

b) High inaccessibility due to large volume and difficult material modificability due to the fuel configuration.

c) More generation of graphite waste with ^{14}C and process complexity in head-end.

Table A2-1 in Appendix shows preliminary evaluations on these characteristics including the results for other options.

(2) Head-end process proposed and the material balance

The head-end process studied in Japan was selected so as to minimize the environmental release of radioactive effluents, to reduce the radioactive waste volume and to reduce mechanical handling. A conceptual diagram of the process is shown in Fig.4-4. In this process, fuel pins mechanically separated from the graphite block are burned catalytically with CO_2 to give CO, which is then decomposed to CO_2 and solid carbon so that the volume of graphite waste containing ^{14}C is minimized. The CO_2 formed is recycled without its environmental release. Removal of SiC coating is made by pneumatic jet grinding. The remaining PyC-coated kernel is planned to be reburned for removing volatile nuclides such as ^{3}H, ^{131}I, ^{129}I and ^{85}Kr.

Based on the spent fuel composition of VHTR, a material balance in the process flowsheet was calculated preliminarily, showing the characteristics of environmental safety, waste generation and proliferation resistance in reprocessing of HTR fuel. The distribution of nuclides in the head-end step is assumed, referring to the data reported for LWR fuel[5~7] or estimated for HTGR fuel.[8,9] Table A2-2 summarizes these assumptions.

The material balances are summarized in Table 4-3. Compared with the LWR fuel processing, the HTR one features (A) generation of graphite waste, in large volume, of low or medium level, (B) load in off-gas cleaning step, because of the large volume of carbon, along with semi-volatiles and particulates and (C) generation of different types of solid waste like SiC hull and chemical reagents from off-gas traps, both highly contaminated with insolubles, particulates and semi-volatiles. There thus are the needs for development of waste treatment technology, radioactive powder handling and confinement. Material balances for extended cooling times of 10 and 50 years are shown Table A2-3 (A) and (B) for reference.

(3) Preliminary evaluation of HTR fuel reprocessing

(i) Environment and safety aspects

Heavy load in the off-gas cleaning step is inevitable as described above. However, sufficient decontamination can be achieved at reasonable cost if the techniques currently studied are completed. ^{3}H confinement will be easier in HTR, while control of semi-volatiles will require further development. Volume reductions of the spent fuel and graphite waste for disposal, possible in the proposed fuel design and head-end process of Table 4-2, will improve the safety and environment aspects.

Although the risk of fire and explosion will increase in the above head-end operation, the cells and equipments for radioactivity confinement will provide safety barriers. Less powder handling and endothermic graphite removal reaction, which are characteristics of the proposed flowsheet, may be advantageous; while great care is necessary for CO control.

(ii) Proliferation resistance features

In Table 4-4, characteristics of HTR fuel in proliferation resistance are summarized for major head-end steps of reprocessing. HTR spent fuel elements possess high proliferation resistance because of the high gamma radiation in the course of access and of the voluminous fuel elements coupled with the complex fuel configuration. Since refined technology is needed for the head-end processing, it will be difficult to extract fissile materials from the spent fuels. Similar arguments also apply to dismantled fuel pins and even to coated particle fuel.

In the separation step, proliferation problems are similar to LWR fuel reprocessing.

(iii) Technical features in long term storage

Design concept of "Pin-in-Block" type fuel has such advantage as reducing the volume of spent fuels in long term storage-delayed reprocessing. Table 4-5 shows the effect of storage time on the amounts of fissile materials and the activities of major fission products.

After 50 years fissile Pu is reduced by 30 % due to the decay of ^{241}Pu to ^{241}Am. Such nuclides as Zr, Nb and Ru, which are troublesome in the solvent extraction, have decayed to negligible levels after ten years, though the gross activity of fission products is not reduced by more than a factor of 100. Radioactivities of off-gases are reduced by half after ten years, except ^{14}C and ^{129}I. Therefore, very simple process is usable in the separation step, though technical difficulties in the head-end step will be hardly mitigated. The decay of ^{238}Pu and ^{241}Pu will also reduce somewhat occupational dose in the refabrication.

This stowaway cycle must be evaluated in details, considering

Table 4-3 Material Banalce of Head-end Process for Spent Fuel Annually Dishcarged from 3000 MW(t) VHTR after 0.5 Year Cooling.

Composition \ Stream[d)]		Feed SF	Dissolved solution	Dismantled graphite	Decomposed graphite	SiC hull	Waste from off gas cleaning	Release to environment
		①	②	③	④	⑤	⑥	⑦
Actinide	kg	11,400	11,343	—a)	—b)	45.6	11.4	1.14×10^{-4}
	Ci	4.29×10^{6}	4.27×10^{6}	—a)	47b)	1.71×10^{4}	4.29×10^{3}	4.28×10^{-3}
Fission product	kg	790	622	—a)	—b)	7.9	35	125
	Ci	1.16×10^{8}	1.11×10^{8}	—a)	—b)	1.16×10^{6}	4.30×10^{6}	2.35×10^{3}
Graphite	kg	215,170	0	147,300	67,870	—	0	0
Others	kg	3,570 (SiC, O_2)	—	—	—	1,930 (SiC)	—c)	1,640 (O_2)
Total mass	kg	230,930	11,965	147,300	67,870	1,984	46.4c)	1,765
Total radio-activities	Ci	1.16×10^{8}	1.15×10^{8}	(3.7×10^{3})a)	47b)	1.18×10^{6}	4.30×10^{6}	2.35×10^{3}
Major composition								
Actinide								
U total	kg	11,220	11,164	—	—b)	44.88	11.22	1.12×10^{-4}
U fissile	kg	157	156.2	—	—b)	0.628	0.157	1.57×10^{-6}
Pu total	kg	160	159.2	—	—b)	0.64	0.16	1.6×10^{-7}
Pu fissile	kg	95	94.53	—	—b)	0.38	9.5×10^{-2}	9.5×10^{-8}
Np total	kg	9.41	9.40	—	—b)	3.8×10^{-2}	9.4×10^{-3}	9.4×10^{-9}
Am total	kg	6.05	6.04	—	—b)	2.4×10^{-2}	6.1×10^{-3}	6.1×10^{-9}
Cm total	kg	1.15	1.14	—	—b)	4.6×10^{-3}	1.2×10^{-3}	1.2×10^{-9}
Volatile FP's								
H-3	Ci	1.67×10^{4}	1.31×10^{2}	3.59×10^{3}	—	—	1.3×10^{4}	(1.31×10^{2})
C-14	Ci	1.30×10^{2}	—	8.26×10^{1}	4.67×10^{1}	—	—	4.72×10^{-1}
Kr-85	Ci	2.22×10^{5}	—	—	—	—	2.20×10^{5}	2.22×10^{3}
I-129	Ci	8.55×10^{-1}	—	—	—	—	8.55×10^{-1}	8.55×10^{-4}
I-131	Ci	4.27×10^{0}	—	—	—	—	4.27×10^{0}	4.27×10^{-3}
Non-volatile FP's				—				
Sr-90, Y-90	Ci	3.42×10^{6}	3.38×10^{6}	—	—b)	3.42×10^{4}	4.06×10^{6} (Semi volatiles)	4.06×10^{-1} (Semi volatiles)
Zr-95	Ci	6.94×10^{6}	6.86×10^{6}	—	—b)	6.94×10^{4}		
Nb-95	Ci	1.35×10^{7}	1.34×10^{7}	—	—b)	1.35×10^{5}		
Ru-103, Rh-103	Ci	3.32×10^{6}	2.95×10^{6}	—	—b)	3.32×10^{4}	7.12×10^{4} (Particulates)	7.12×10^{-2} (Particulates)
Ru-106, Rh-106	Ci	2.0×10^{7}	1.78×10^{7}	—	—b)	2.0×10^{5}		
Cs-137	Ci	2.48×10^{6}	2.21×10^{6}	—	—b)	2.48×10^{4}		

a) Activities due to impurities other than N_2 and Li are not included.

b) Calculated assuming that decomposed graphite is not contaminated with actinides and FP's.

c) Chemicals and corrosion products are not included.

d) Circled numbers correspond to streams shown in Fig. 4-4.

Table 4-4 Characteristics of Proliferation Resistance of HTR Fuel

Characteristics on proliferation resistance	Related parameter of component	Component unit in HTR head-end			
		Spent fuel	Dismantled fuel pin	Coated particle	Dissolved solution
Material accessibility (for 0.5 year cooling)	Number per kg-$Pu_{fissile}$	14 Element	480 Pin	——	——
	Volume per kg-$Pu_{fissile}$	1.5 m^3 b)	0.6 m^3 b)	78 ℓ c)	330 ℓ
	Activity per component	87×10^3 Ci/Element	2.5×10^3 Ci/Pin	16×10^3 Ci/1-cpf c)	3.7×10^3 Ci/1-solution
Material convertibility	Technical difficulty to recover Pu a)	A	A	B	C

a) A identifies material that requires a large facility of shielded cell with remote operation to obtain purified Pu.

B identifies material that requires remotely operated engineering equipment in/out of shielded cell to obtain purified Pu.

C identifies material that requires remotely operated laboratory scale equipment in/out of shielded cell to obtain purified Pu.

b) Estimated without considering void fraction in packaging.

c) Estimated assuming void fraction of 0.35.

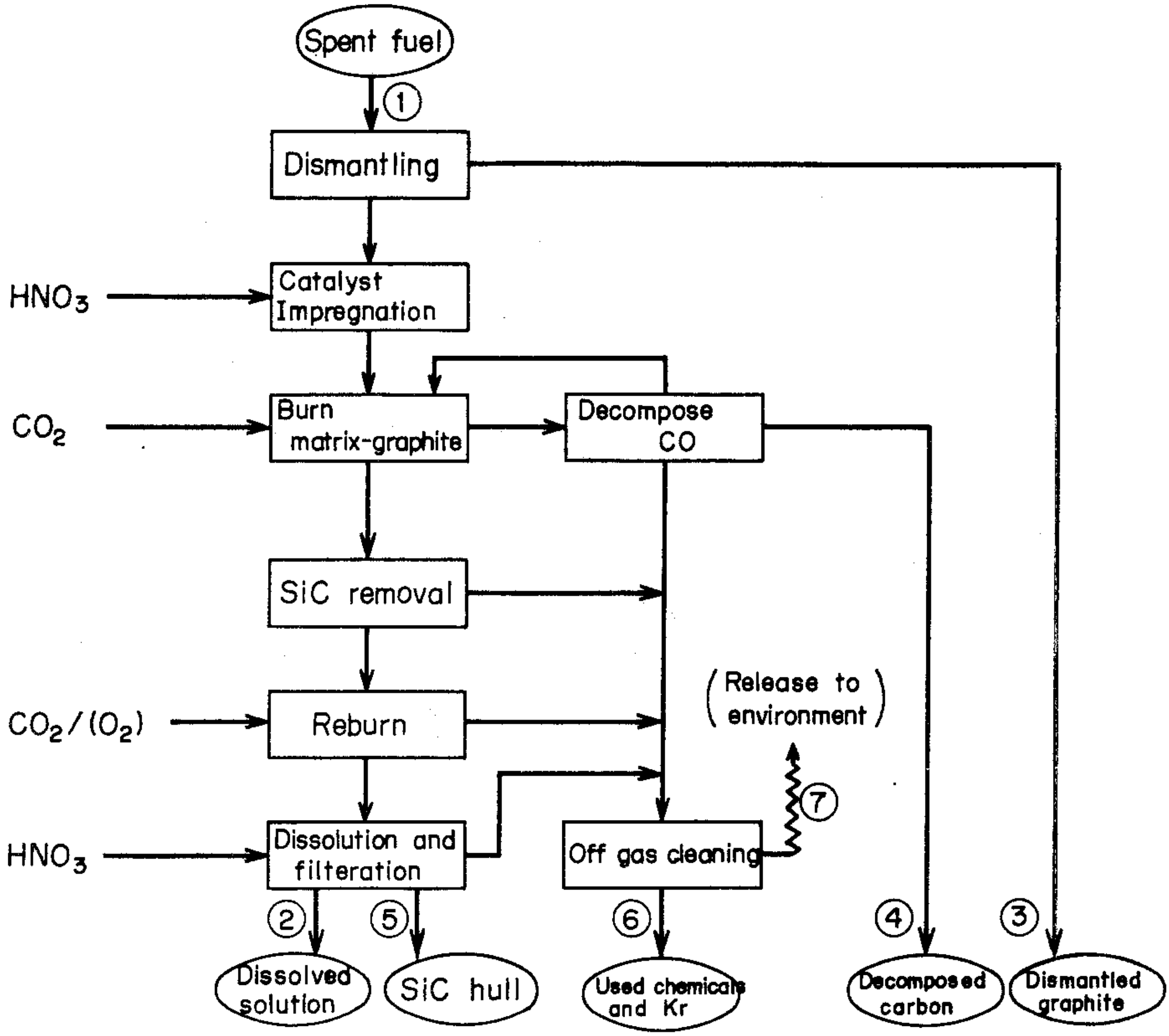

Fig. 4-4 Proposed Flowsheet for Head-end Processing of VHTR Spent Fuel (Option II)

Table 4-5 Fissile Amounts and Radioactivities as a Function of Cooling Time in the Spent Fuel Annually Discharged from 3000 MW(t) VHTR.

Composition and radioactivity in spent fuel	Cooling time		
	0.5 y	10 y	50 y
Total fissile U	157 kg	157 kg	157 kg
Total fissile Pu	95	83	66
Total radioactivities of FP's	1.1×10^{8} Ci	7.8×10^{6} Ci	2.8×10^{6} Ci
^{90}Sr, ^{90}Y	3.4×10^{6}	2.7×10^{6}	1.1×10^{6}
^{95}Zr	6.9×10^{6}	7.8×10^{-10}	—
^{95}Nb	1.4×10^{7}	1.7×10^{-9}	—
^{103}Ru, ^{103m}Rh	3.3×10^{6}	1.9×10^{-20}	—
^{106}Ru, ^{106}Rh	2.0×10^{7}	2.9×10^{4}	3.2×10^{-8}
^{137}Cs	2.5×10^{6}	2.0×10^{6}	8.0×10^{5}

other factors like resource availability and waste disposal policy.

4.4 Waste management

A major research subject in waste management of HTR is the treatment of graphite waste and silicon-carbide hull from spent fuels. The amounts and volumes of wastes are estimated below.

In the management of graphite waste, there are three possible options as follows:

Option I: Carbon dioxide generated by burning the graphite waste with oxygen is immobilized as calcium carbonate. Assuming that the weight percent of calcium carbonate in the solid concrete is 40 w% to obtain the compression strength exceeding 150 kg/cm^2, 1 ton of graphite waste finally becomes 21 tons and 10 M^3 of the solid concrete with specific density 2.0. Hence the amount of solid waste from graphite blocks is estimated to be 251 T/T_{HM} and 126 M^3/T_{HM}.

For fuel pins, the amount of the solid concrete is estimated as 116 T/T_{HM} and 56 M^3/T_{HM}. The activity of the concrete is estimated as 34 mCi/M^3 due to C-14.

Option II: The graphite in fuel pins is converted into carbon as shown in Sect.4.3-(2). The calculated amount of the recovered carbon is about 6 T/T_{HM} and its volume is about 4 M^3/T_{HM} with compressed bulk density 1.5. The radiation level of the carbon waste is estimated to be 1.2 Ci/M^3 due to C-14.

Option III: After withdrawing of fuel pins, the residual graphite block with β,γ-emitters is enclosed in a canister by welding, though the graphite itself is chemically stable. The weight of the graphite is estimated to be 12 T/T_{HM} and its volume is 13 M^3/T_{HM}. The volume of blocks may be reduced to 7 M^3/T_{HM} when compacted the most. The induced radioactivity of the graphite waste is estimated to be about 60 Ci/M^3 with major nuclides of Fe-55, Co-60, Cs-134 and Ag-110 m.

There is little difference in management of high level radioactive liquid waste such as solidification, engineering storage and geologic isolation between HTR and LWR fuel cycles with LEU.

Assuming that the recovery of nuclear fuels such as Pu and U is 99.5 w%, the major components of the high level waste including hulls are estimated as shown in Table 4-3. There is no significant change from 0.5 to 50 years in the weight of the vitrified waste with borosilicate glass as shown in Table 4-6. However, the level of its activity decreases from 8×10^7 to 2×10^6 Ci/M^3 during the cooling time from 0.5 to 50 years. The activity of the vitrified high level waste from HTGR fuel cycle is estimated to be 9×10^8 Ci/M^3 for 0.5 years cooling time; there is a difference in radiation level between HTGR and VHTR only in short cooling times as 0.5 years.

The silicon carbide such as hulls is also vitrified with borosilicate glass, and its weight, volume and activity are shown in Table 4-6. The volume of hulls (SiC) is about ten times smaller per unit electric power than that of hulls (Zr-alloy) in LWR.

The amounts of semi-volatile radioactive elements and particulates estimated are shown in Table 4-6. These elements are mainly Cs and Ru and partly trans Pu, which are absorbed in the silica gel, hepa-filter and other chemical reagents. Assuming the 1000-fold increase of weight by bitumization and the resulting solid with density 1.0, the activity of the product is calculated to be 57 and 1 Ci/1 for 0.5 and 50 years of cooling time, respectively.

Volatile elements must be fractionated into such as Kr and I stored for cooling. The estimated amounts of the groups are shown in Table 4-6.

Table 4-6 Estimation of High Level Radio-active Waste and Other Wastes from VHTR Fuel Cycle

Wastes \ Cooling time(y)	0.5	10	50
1. Vitrified HLW			
weight kg/T_{HM}	316	322	330
volume M^3/T_{HM}	0.11	0.11	0.11
activity Ci/M^3	7.7×10^7	5.7×10^6	2.1×10^6
Major wastes (total weight kg/T_{HM})	(63)	(64)	(66)
FP oxide kg/T_{HM}	56	56	57
α-waste oxide*1 kg/T_{HM}	2	3	4
Fuel oxide kg/T_{HM}	5	5	5
2. Vitrified SiC hulls			
weight kg/T_{HM}	816	816	816
volume M^3/T_{HM}	3	3	3
activity Ci/M^3	3×10^4	3×10^3	9×10^2
Major wastes (total weight kg/T_{HM})	(163)	(163)	(163)
SiC hulls kg/T_{HM}	158	158	158
FP oxide etc kg/T_{HM}	5	5	5
3. Semi-volatile and particulates			
weight*2 kg/T_{HM}	6	6	6
activity Ci/T_{HM}	3×10^5	1×10^4	7×10^3
4. Volatile radioactive FP			
weight*3 kg/T_{HM}	12	12	11
activity Ci/T_{HM}	2×10^4	1×10^4	8×10^2
5. Non radioactive volatile waste, weight*4 kg/T_{HM}	11	11	11

*1 Np, Am, and Cm are included.

*2 Major elements are Cs, Ru and a small amount of transplutonium.

*3 Major elements are Kr and I.

*4 Major elements are Br and Xe.

Acknowledgments

The authors would like to thank Establishment Director Mr. Noboru Amano for his encouragements.

Reference

(1) Burch, W.D. et al. : IAEA-CN-36/14 (1977).

(2) Kaiser, G. et al. : IAEA-CN-36/112 (1977).

(3) Brooks, L.H. et al. : GAMD-9260 (1969).

(4) Beone, G. et al. : Kerntechnik, vol.15, p.270 (1974).

(5) Finney, B.C. et al. : ORNL-TM 4901 (1975).

(6) U.S. ERDA : ERDA 76-43, vol.2 (1976).

(7) McKay, H.A.C. et al. : EUR-5801 (1977).

(8) Lin, K.H. : ORNL-TM 5096 (1976).

(9) Pence, D.T. : GA-A 13919 (1976).

(10) Nomura, S., Iwamoto, T. and Tsujino, T. : IAEA-CN-36/168 (1977).

Acknowledgments

The authors would like to thank Establishment Director Mr. Noboru Amano for his encouragements.

Reference

(1) Burch, W.D. et al. : IAEA-CN-36/14 (1977).

(2) Kaiser, G. et al. : IAEA-CN-36/112 (1977).

(3) Brooks, L.H. et al. : GAMD-9260 (1969).

(4) Beone, G. et al. : Kerntechnik, vol.15, p.270 (1974).

(5) Finney, B.C. et al. : ORNL-TM 4901 (1975).

(6) U.S. ERDA : ERDA 76-43, vol.2 (1976).

(7) McKay, H.A.C. et al. : EUR-5801 (1977).

(8) Lin, K.H. : ORNL-TM 5096 (1976).

(9) Pence, D.T. : GA-A 13919 (1976).

(10) Nomura, S., Iwamoto, T. and Tsujino, T. : IAEA-CN-36/168 (1977).

Appendix

A1 Fuel material balance

A1.1 Material balance of once through fuel cycle in LEU-HTR

The detailed material balances of once through fuel cycle in LEU-HTR are described.

The items and the results analyzed are as follows;

1) Fuel material balance of HTR
   (Table A1.1-1, Fig. A1.1-1)
2) Difference of fuel material balance between HTR and LWR
   (Table A1.1-2)
3) Build up and decay of actinide in spent fuels of HTR
   (Table A1.1-3, A1.1-4)
4) Radioactivity of actinide in spent fuels of HTR
   (Table A1.1-5, A1.1-6, Fig. A1.1-2, A1.1-3, A1.1-4)
5) Radioactivity of fission products in spent fuels of HTR
   (Table A1.1-7, A1.1-8, Fig. A1.1-5)
6) Difference of radioactivity in spent fuels between reprocessing and non-reprocessing
   (Table A1.1-9)
7) Radioactivity of ^{3}H and ^{14}C in spent fuel elements of HTR
   (Table A1.1-10)

The above (1) and (2) were estimated with the multi-group point burnup code, DELIGHT-4* and the (3)~(6) with the analysis code of build up and decay of nuclides, DCHAIN.**

---

* R. Shindo, M. Hirano ; JAERI-M 8097 (1979).

** K. Tasaka ; JAERI 1250 (1977).

Table A1.1-1 Fuel Material Balance of LEU-HTR

HTR type		VHTR	HTGR
Thermal power	(MW)	3000	3000
Power density	(W/cm^3)	6.0	9.0
Specific power	(w/g)	81.8	122.6
Fuel enrichment	(w/o)	6.0	6.5
Fuel burnup*1	(MWD/T)	62,800	62,300
Fuel dwelling time*2	(Yr)	3.0	2.0
Initial fuel loading	(ton HM)	36.67	24.47
U^{235}	(ton)	2.2_0	1.5_4
U^{238}	(ton)	34.4_7	22.8_8
Fuel reloading (annual)	(ton HM)	12.22	12.23
Uranium	(ton)	12.2_2	12.2_3
U^{235}	(ton)	0.73_4	0.79_5
U^{238}	(ton)	11.49	11.44
Fissile nuclide content	(w/o)	6.0	6.5
Fuel discharge (annual)	(ton HM)	11.3_8	11.3_9
Uranium	(ton)	11.2_2	11.2_3
U^{235}	(kg)	157	180
U^{236}	(kg)	90	101
U^{238}	(ton)	10.9_7	10.9_5
Fissile nuclide content	(w/o)	1.4	1.6
Plutonium	(kg)	157	162
Pu^{239}		64	72
Pu^{240}		41	41
Pu^{241}		31	31
Pu^{242}		21	18
Fissile nuclide content	(w/o)	60	64

*1 Operational margin of reactivity --- 0.01 Δk_{eff}.

*2 Plant factor --- 70 %.

Table A1.1-2 Difference of Fuel Material Balance between HTR and LWR

Reactor Type	VHTR	HTGR	BWR	PWR
Thermal power (MW)	3000	3000	3000	3000
Thermal efficiency (%)	—	40	34	33
Specific power (w/g)	81.8	122.6	28	38
Burnup (MWD/T)				
Initial loading fuel	55,800	51,900	17,000	22,600
Reload fuel	62,800	62,300	27,500	32,600
Uranium loading (ton HM)				
Initial loading (enrich. %)	36.6_7 (6.0)	24.4_7 (6.5)	107.1 (2.03)	78.9 (2.26)
Annual reloading (enrich. %)	12.2_2 (6.0)	12.2_3 (6.5)	28.2 (2.73)	23.2 (3.21)
Natural uranium requirements* (ton)				
Initial loading	454	330	411	342
Annual reloading	151	165	151	148
30 years	4682	5115	4639	4514
Separative work unit (ton-swu)				
Initial loading	367	271	209	187
Annual reloading	122	136	91	97
30 years	3783	4215	2848	3000
Fissile plutonium production (ton)				
Annual	0.095	0.103	0.166	0.162
30 years	2.85	3.09	5.16_2	4.99_4
30 year natural uranium requirement (ton/GWe)	(2080/3121)**	4263	4548	4560
30 year separative work (ton-swu/GWe)	(1681/2522)**	3513	2792	3030

* Tail enrichment --- 0.25 %.

** Right; type of mainly process heat generation --- thermal effective efficiency 70 %.
Left; type of mainly electric power generation --- thermal effective efficiency 50 %.

*** Plant factor 70 %.

Table A1.1−3 Buildup and Decay of Actinide in Spent Fuels Annually Discharged from VHTR Core

Cooling time	0	30 d	100 d	0.5 y	1.0 y	2 y	3 y	5 y	10 y	20 y	50 y	100 y	10^3 y	10^4 y	10^5 y	10^6 y
Uranium																
U−232	8.0 (−7)	9.0 (−7)	1.1 (−6)	1.4 (−6)	1.9 (−6)	2.8 (−6)	3.5 (−6)	4.4 (−6)	5.3 (−6)	5.2 (−6)	4.0 (−6)	2.5 (−6)	5.2 (−10)	7.5 (−47)	0.0	0.0
U−234	0.014	0.016	0.021	0.027	0.039	0.065	0.091	0.142	0.266	0.499	1.096	1.821	3.28	3.20	2.48	0.20
U−235	135.0	135.0	135.0	135.0	135.0	135.0	135.0	135.0	135.0	135.0	135.0	135.0	135.0	135.0	134.9	134.8
U−236	97.3	97.3	97.3	97.3	97.3	97.3	97.3	97.3	97.3	97.3	97.5	97.7	101.4	123.9	137.9	134.3
U−237	0.248	0.011	9.6 (−6)	1.0 (−6)	9.9 (−7)	9.4 (−7)	8.9 (−7)	8.0 (−7)	6.1 (−7)	3.6 (−7)	7.5 (−8)	5.4 (−9)	3.2 (−12)	1.6 (−12)	2.0 (−15)	1.4 (−44)
U−238	11016.0	11016.0	11016.0	11016.0	11016.0	11016.0	11016.0	11016.0	11016.0	11016.0	11016.0	11016.0	11016.0	11015.9	11015.8	11014.3
U−239	0.017	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total	11248.5	11248.2	11248.2	11248.2	11248.2	11248.3	11248.3	11248.3	11248.5	11248.8	11249.5	11250.4	11255.6	11278.0	11291.1	11283.6
Neptunium																
Np−236	5.2 (−8)	7.3 (−18)	7.5 (−41)	6.0 (−68)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Np−237	9.16	9.40	9.41	9.41	9.41	9.41	9.42	9.44	9.53	9.81	11.01	13.26	35.12	42.55	41.37	30.90
Np−238	0.035_3	1.8 (−6)	1.6 (−16)	2.4 (−28)	1.7 (−54)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Np−239	2.39	3.5 (−4)	3.3 (−6)	3.3 (−6)	3.3 (−6)	3.3 (−6)	3.3 (−6)	3.3 (−6)	3.3 (−6)	3.3 (−6)	3.3 (−6)	3.3 (−6)	3.0 (−6)	1.4 (−6)	5.4 (−10)	4.2 (−44)
Np−240	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total	11.59	9.40	9.41	9.41	9.41	9.41	9.42	9.44	9.53	9.81	11.01	13.26	35.12	42.55	41.37	30.90
Plutonium																
Pu−236	5.4 (−6)	5.4 (−6)	5.1 (−6)	4.8 (−6)	4.3 (−6)	3.4 (−6)	2.6 (−6)	1.6 (−6)	4.8 (−7)	4.2 (−8)	2.8 (−11)	1.5 (−16)	0.0	0.0	0.0	0.0
Pu−238	2.93	3.00	3.08	3.14	3.22	3.25	3.24	3.19	3.06	2.83	2.22	1.48	0.001_2	2.3 (−22)	0.0	0.0
Pu−239	67.01	69.42	69.42	69.42	69.42	69.42	69.41	69.41	69.40	69.39	69.34	69.26	67.82	54.28	4.40	3.4 (−11)
Pu−240	40.80	40.80	40.81	40.82	40.83	40.86	40.89	40.94	41.05	41.20	41.36	41.25	37.53	14.53	0.001_1	6.9 (−45)
Pu−241	32.77	32.63	32.30	31.92	31.09	29.50	27.99	25.20	19.38	11.46	2.37	0.171	1.0 (−4)	5.1 (−5)	6.3 (−8)	4.4 (−37)
Pu−242	24.16	24.16	24.16	24.16	24.16	24.16	24.16	24.16	24.16	24.16	24.16	24.16	24.12	23.73	20.12	3.88
Pu−243	5.9 (−3)	1.8 (−46)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total	167.68	170.02	169.77	169.46	168.72	167.19	165.70	162.90	157.06	149.04	139.45	136.33	129.47	92.53	24.52	3.88
Americium																
Am−241	1.03	1.17	1.50	1.88	2.70	4.29	5.79	8.56	14.30	21.94	29.80	29.71	7.65	0.001_9	2.3 (−6)	1.6 (−35)
Am−242m	0.014_3	0.014_3	0.014_3	0.014_2	0.014_2	0.014_2	0.014_1	0.014_0	0.013_6	0.0130_0	0.011_4	0.009_0	1.5 (−4)	2.2 (−22)	0.0	0.0
Am−242	2.8 (−3)	1.7 (−7)	1.7 (−7)	1.7 (−7)	1.7 (−7)	1.7 (−7)	1.7 (−7)	1.7 (−7)	1.6 (−7)	1.6 (−7)	1.4 (−7)	1.1 (−7)	1.8 (−9)	2.6 (−27)	0.0	0.0
Am−243	4.15	4.15	4.15	4.15	4.15	4.15	4.15	4.15	4.15	4.15	4.14	4.12	3.81	1.736	6.75 (−4)	5.3 (−38)
Am−244	3.2 (−3)	1.1 (−24)	9.3 (−75)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total	5.20	5.34	5.67	6.05	6.87	8.46	9.96	12.73	18.46	26.10	33.95	33.84	11.46	1.738	6.77 (−4)	1.61 (−35)
Curium																
Cm−242	0.358	0.317	0.236	0.166	0.076_3	0.016_2	0.003_5	1.9 (−4)	3.4 (−5)	3.2 (−5)	2.8 (−5)	2.2 (−5)	3.7 (−7)	5.4 (−25)	0.0	0.0
Cm−243	0.007_7	0.007_7	0.007_7	0.007_7	0.007_6	0.007_4	0.007_3	0.006_9	0.006_2	0.005_0	0.002_6	0.0008_9	3.0 (−12)	0.0	0.0	0.0
Cm−244	0.910	0.910	0.903	0.895	0.878	0.844	0.811	0.750	0.616	0.415	0.127	0.0177	7.0 (−18)	0.0	0.0	0.0
Cm−245	0.077_6	0.077_6	0.077_6	0.077_6	0.077_6	0.077_6	0.077_6	0.077_6	0.077_5	0.077_4	0.077_3	0.0770	0.0720	0.0368	4.47 (−5)	3.2 (−34)
Total	1.353	1.313	1.224	1.147	1.040	0.945	0.900	0.835	0.700	0.498	0.207	0.0957	0.0720	0.0368	4.47 (−5)	3.15 (−34)
Total	11434.3	11434.3	11434.3	11434.3	11434.3	11434.3	11434.3	11434.3	11434.2	11434.2	11434.1	11433.9	11431.7	11414.9	11357.0	11318.4

* (a ± b) means $a \times 10^{\pm b}$ ** Unit ; Kg *** Annually discharged fuel …… 11.4 ton HM

Table A1.1－4 Buildup and Decay of Actinide in Spent Fuels Annually Discharged from HTGR Core.

Cooling time	0	30 d	100 d	0.5 y	1.0 y	2 y	3 y	5 y	10 y	20 y	50 y	100 y	10^3 y	10^4 y	10^5 y	10^6 y
Uranium																
U−232	5.4(−7)	6.5(−7)	9.0(−7)	1.2(−6)	1.7(−6)	2.7(−6)	3.4(−6)	4.3(−6)	5.3(−6)	5.2(−6)	4.0(−6)	2.5(−6)	5.1(−10)	7.5(−47)	0.0	0.0
U−234	0.008	0.010	0.014	0.019	0.030	0.052	0.074	0.118	0.224	0.424	0.936	1.557	2.81	2.74	2.13	0.17
U−235	159.9	159.9	159.9	159.9	159.9	159.9	159.9	159.9	159.9	159.9	159.9	159.9	159.9	159.9	159.9	159.8
U−236	104.0	104.0	104.0	104.0	104.0	104.0	104.0	104.0	104.0	104.1	104.2	104.4	107.8	128.9	141.9	138.3
U−237	0.383	0.018	1.4(−5)	9.5(−7)	9.2(−7)	8.8(−7)	8.3(−7)	7.5(−7)	5.7(−7)	3.4(−7)	7.0(−8)	5.1(−9)	2.0(−12)	1.0(−12)	1.3(−15)	8.8(−45)
U−238	11048.8	11048.8	11048.8	11048.8	11048.8	11048.8	11048.8	11048.8	11048.8	11048.8	11048.8	11048.8	11048.8	11048.8	11048.6	11047.1
U−239	0.024	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total	11313.1	11312.7	11312.7	11312.7	11312.7	11312.8	11312.8	11312.8	11313.0	11313.2	11313.8	11314.7	11319.3	11340.3	11352.6	11345.3
Neptunium																
Np−236	7.9(−8)	1.1(−17)	1.1(−40)	9.2(−68)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Np−237	9.64	10.01	10.02	10.02	10.03	10.03	10.04	10.06	10.13	10.39	11.50	13.58	33.82	40.69	39.54	29.54
Np−238	0.048_9	2.4(−6)	2.3(−16)	3.4(−28)	2.3(−54)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Np−239	3.44	4.9(−4)	2.4(−6)	2.4(−6)	2.4(−6)	2.4(−6)	2.4(−6)	2.4(−6)	2.4(−6)	2.4(−6)	2.4(−6)	2.4(−6)	2.2(−6)	1.0(−6)	3.9(−10)	3.1(−44)
Np−240	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total	13.13	10.01	10.02	10.02	10.03	10.03	10.04	10.06	10.13	10.39	11.50	13.58	33.82	40.69	39.54	29.54
Plutonium																
Pu−236	5.7(−6)	5.6(−6)	5.3(−6)	5.0(−6)	4.5(−6)	3.5(−6)	2.7(−6)	1.7(−6)	5.0(−7)	44(−8)	3.0(−11)	1.5(−16)	0.0	0.0	0.0	0.0
Pu−238	2.56	2.63	2.68	2.72	277	2.79	2.78	2.73	2.63	2.42	1.90	1.27	9.9(−4)	1.4(−22)	0.0	0.0
Pu−239	79.17	82.63	82.63	82.63	82.63	82.62	82.62	82.62	82.61	82.59	82.53	82.42	80.56	63.67	5.07	3.9(−11)
Pu−240	38.37	38.37	38.37	38.38	38.38	38.40	38.42	38.45	38.52	38.62	38.69	38.56	35.08	13.58	0.0010	6.5(−45)
Pu−241	30.66	30.53	30.22	29.86	29.09	27.60	26.19	23.58	18.13	10.72	2.22	0.160	6.6(−5)	3.3(−5)	4.0(−8)	2.8(−37)
Pu−242	18.95	18.95	18.95	18.95	18.95	18.95	18.95	18.95	18.95	18.95	18.95	18.95	18.92	18.61	15.78	3.04
Pu−243	6.7(−3)	2.0(−46)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total	169.71	173.10	172.85	172.54	171.83	170.37	168.96	166.34	160.84	153.30	144.28	141.36	134.56	95.86	20.86	3.04
Americium																
Am−241	0.629	0.761	1.07	1.42	2.20	3.68	5.08	7.68	13.05	20.20	27.57	27.51	7.08	0.0012_0	1.4(−6)	1.0(−35)
Am−242m	0.008_6	0.008_6	0.008_6	0.008_6	0.008_6	0.008_6	0.008_5	0.008_4	0.008_3	0.007_9	0.006_9	0.005_5	9.0(−5)	1.3(−22)	0.0	0.0
Am−242	2.4(−3)	1.0(−7)	1.0(−7)	1.0(−7)	1.0(−7)	1.0(−7)	1.0(−7)	1.0(−7)	9.9(−8)	9.5(−8)	8.3(−8)	6.6(−8)	1.1(−9)	1.6(−27)	0.0	0.0
Am−243	3.03	3.04	3.04	3.04	3.04	3.03	3.03	3.03	3.03	3.03	3.02	3.01	2.78	1.269	4.93(−4)	3.9(−38)
Am−244	3.3(−3)	1.2(−24)	9.8(−75)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total	3.67	3.81	4.11	4.47	5.24	6.72	8.13	10.72	16.09	23.24	30.60	30.52	9.86	1.270	4.95(−4)	1.02(−35)
Curium																
Cm−242	0.241	0.214	0.159	0.112	0.051_5	0.010_9	0.002_3	1.2(−4)	2.0(−5)	2.0(−5)	1.7(−5)	1.4(−5)	2.2(−7)	3.3(−25)	0.0	0.0
Cm−243	0.004_8	0.004_8	0.004_8	0.004_8	0.004_7	0.004_6	0.004_5	0.004_3	0.003_9	0.003_1	0.001_6	0.0005_5	1.9(−12)	0.0	0.0	0.0
Cm−244	0.616	0.618	0.613	0.608	0.596	0.573	0.551	0.509	0.418	0.282	0.086_4	0.0120	4.7(−18)	0.0	0.0	0.0
Cm−245	0.0491	0.049_1	0.049_1	0.049_0	0.049_0	0.049_0	0.049_0	0.049_0	0.049_0	0.049_0	0.048_9	0.0487	0.0455	0.0233	2.83(−5)	2.0(−34)
Total	0.912	0.886	0.826	0.774	0.701	0.637	0.607	0.563	0.471	0.334	0.136_9	0.0613	0.0455	0.0233	2.83(−5)	1.99(−34)
Total	11500.5	11500.5	11500.5	11500.5	11500.5	11500.5	11500.5	11500.5	11500.5	11500.5	11500.3	11500.2	11497.6	11478.2	11413.0	11377.9

* (a±b) means $a \times 10^{\pm b}$ ** Unit; Kg *** Annually discharged fuel …… 11.4 ton HM

Table A 1.1−5　Radioactivitiy of Actinide in Spent Fuels Annually Discharged from VHTR Core.

Cooling time	0 d	30d	100d	0.5y	1.0y	2y	3y	5y	10y	20y	50y	100y	10^3 y	10^4 y	10^5 y	10^6 y
Uranium																
U−232	1.67 −2	1.89 −2	2.39 −2	2.95 −2	4.07 −2	5.93 −2	7.37 −2	9.30 −2	1.12 −1	1.10 −1	8.36 −2	5.22 −2	1.08 −5	1.58 −42	0.0	0.0
U−234	8.97 −2	1.02 −1	1.30 −1	1.65 −1	2.43 −1	4.02 −1	5.61 −1	8.77 −1	1.64 0	3.09 0	6.79 0	1.13 1	2.03 1	1.98 1	1.54 1	1.23 0
U−235	2.89 −1	2.89 −1	2.89 −1	2.89 −1	2.89 −1	2.89 −1	2.89 −1	2.89 −1	2.89 −1	2.89 −1	2.89 −1	2.89 −1	2.89 −1	2.89 −1	2.89 −1	2.89 −1
U−236	6.17 0	6.17 0	6.17 0	6.17 0	6.17 0	6.17 0	6.17 0	6.17 0	6.17 0	6.18 0	6.18 0	6.20 0	6.43 0	7.86 0	8.75 0	8.52 0
U−237	2.02 7	9.29 5	7.85 2	8.28 1	8.06 1	7.64 1	7.25 1	6.53 1	5.02 1	2.97 1	6.14 0	4.44 −1	2.60 −4	1.33 −4	1.62 −7	1.14 −36
U−238	3.67 0	3.67 0	3.67 0	3.67 0	3.67 0	3.67 0	3.67 0	3.67 0	3.67 0	3.67 0	3.67 0	3.67 0	3.67 0	3.67 0	3.67 0	3.67 0
U−239	5.56 8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0.	0.0	0.0	0.0	0.0	0.0
Total, α	1.60 1	1.03 1	1.03 1	1.03 1	1.04 1	1.06 1	1.08 1	1.11 1	1.19 1	1.33 1	1.70 1	2.15 1	3.07 1	3.16 1	2.81 1	1.37 1
Total, β	5.76 8	9.29 5	7.85 2	8.28 1	8.06 1	7.64 1	7.25 1	6.53 1	5.02 1	2.97 1	6.14 0	4.44 −1	2.60 −4	1.33 −4	1.62 −7	1.14 −36
Neptunium																
Np−236	3.12 1	4.39 −9	4.52 −32	3.65 −59	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Np−237	6.46 0	6.63 0	6.64 0	6.64 0	6.64 0	6.64 0	6.65 0	6.66 0	6.72 0	6.92 0	7.77 0	9.36 0	2.48 1	3.00 1	2.92 1	2.18 1
Np−238	9.22 6	4.62 2	4.27 −8	6.36 −20	4.39 −46	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Np−239	5.56 8	8.12 4	7.69 2	7.69 2	7.69 2	7.69 2	7.69 2	7.69 2	7.69 2	7.68 2	7.66 2	7.63 2	7.05 2	3.22 2	1.25 −1	9.88 −36
Np−240	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total, α	6.46 0	6.63 0	6.64 0	6.64 0	6.64 0	6.64 0	6.65 0	6.66 0	6.72 0	6.92 0	7.77 0	9.36 0	2.48 1	3.00 1	2.92 1	2.18 1
Total, β	5.65 8	8.16 4	7.69 2	7.69 2	7.69 2	7.69 2	7.69 2	7.69 2	7.69 2	7.68 2	7.66 2	7.63 2	7.05 2	3.22 2	1.25 −1	0.0
Plutonium																
Pu−236	2.90 0	2.85 0	2.72 0	2.58 0	2.28 0	1.79 0	1.40 0	8.63 −1	2.55 −1	2.24 −2	1.51 −5	7.84 −11	0.0	0.0	0.0	0.0
Pu−238	5.12 4	5.25 4	5.38 4	5.49 4	5.63 4	5.68 4	5.66 4	5.57 4	5.36 4	4.94 4	3.88 4	2.60 4	2.10 1	4.10 −18	0.0	0.0
Pu−239	4.11 3	4.26 3	4.26 3	4.26 3	4.26 3	4.26 3	4.26 3	4.26 3	4.26 3	4.26 3	4.26 3	4.25 3	4.16 3	3.33 3	2.70 2	2.09 −9
Pu−240	9.25 3	9.25 3	9.25 3	9.25 3	9.25 3	9.26 3	9.27 3	9.28 3	9.30 3	9.34 3	9.37 3	9.35 3	8.50 3	3.29 3	2.50 −1	1.57 −42
Pu−241	3.69 6	3.67 6	3.63 6	3.59 6	3.50 6	3.32 6	3.15 6	2.83 6	2.18 6	1.29 6	2.66 5	1.93 4	1.13 1	5.79 0	7.04 −3	4.96 −32
Pu−242	9.43 1	9.43 1	9.43 1	9.43 1	9.43 1	9.43 1	9.43 1	9.43 1	9.43 1	9.43 1	9.43 1	9.43 1	9.41 1	9.26 1	7.85 1	1.51 1
Pu−243	1.52 7	4.56 −37	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total, α	1.53 7	6.61 4	6.74 4	6.85 4	6.99 4	7.04 4	7.02 4	6.94 4	6.72 4	6.31 4	5.25 4	3.97 4	1.28 4	6.72 3	3.49 2	1.51 1
Total, β	3.69 6	3.67 6	3.63 6	3.59 6	3.50 6	3.32 6	3.15 6	2.83 6	2.18 6	1.29 6	2.66 5	1.93 4	1.13 1	5.79 0	7.04 −3	4.96 −32
Americium																
Am−241	3.34 3	3.80 3	4.86 3	6.09 3	8.77 3	1.39 4	1.88 4	2.78 4	4.64 4	7.11 4	9.66 4	9.63 4	2.48 4	6.12 0	7.40 −3	5.22 −32
Am−242m	1.39 2	1.39 2	1.39 2	1.39 2	1.38 2	1.38 2	1.37 2	1.36 2	1.33 2	1.27 2	1.11 2	8.80 1	1.45 0	2.11 −18	0.0	0.0
Am−242	2.25 6	1.39 2	1.39 2	1.39 2	1.38 2	1.38 2	1.37 2	1.36 2	1.33 2	1.27 2	1.11 2	8.80 1	1.45 0	2.11 −18	0.0	0.0
Am−243	7.68 2	7.70 2	7.69 2	7.69 2	7.69 2	7.69 2	7.69 2	7.69 2	7.69 2	7.68 2	7.66 2	7.63 2	7.05 2	3.22 2	1.25 −1	9.88 −36
Am−244	4.05 6	1.40 −15	1.19 −65	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total, α	4.05 6	4.57 3	5.63 3	6.86 3	9.54 3	1.47 4	1.95 4	2.85 4	4.71 4	7.19 4	9.74 4	9.71 4	2.55 4	3.28 2	1.32 −1	5.22 −32
Total, β	2.25 6	1.39 2	1.39 2	1.39 2	1.38 2	1.38 2	1.37 2	1.36 2	1.33 2	1.27 2	1.11 2	8.80 1	1.45 0	2.11 −18	0.0	0.0
Total, I.T.	1.39 2	1.39 2	1.39 2	1.39 2	1.38 2	1.38 2	1.37 2	1.36 2	1.33 2	1.27 2	1.11 2	8.80 1	1.45 0	2.11 −18	0.0	0.0
Curium																
Cm−242	1.19 6	1.05 6	7.80 5	5.49 5	2.53 5	5.36 4	1.15 4	6.23 2	1.12 2	1.07 2	9.32 1	7.42 1	1.22 0	1.78 −18	0.0	0.0
Cm−243	3.56 2	3.55 2	3.54 2	3.52 2	3.48 2	3.41 2	3.34 2	3.20 2	2.87 2	2.31 2	1.20 2	4.08 1	1.37 −7	0.0	0.0	0.0
Cm−244	7.58 4	7.59 4	7.53 4	7.46 4	7.32 4	7.03 4	6.76 4	6.25 4	5.13 4	3.46 4	1.06 4	1.48 3	5.83 −13	0.0	0.0	0.0
Cm−245	1.22 1	1.22 1	1.22 1	1.22 1	1.22 1	1.22 1	1.22 1	1.22 1	1.22 1	1.22 1	1.21 1	1.21 1	1.13 1	5.78 0	7.03 −3	4.95 −32
Total, α	1.26 6	1.13 6	8.56 5	6.24 5	3.26 5	1.24 5	7.94 4	6.35 4	5.17 4	3.50 4	1.08 4	1.61 3	1.25 1	5.78 0	7.03 −3	4.95 −32
Total α	2.06 7	1.20 6	9.29 5	7.00 5	4.06 5	2.09 5	1.69 5	1.61 5	1.66 5	1.70 5	1.61 5	1.38 5	3.84 4	7.11 3	4.06 2	5.06 1
Total β	1.15 9	4.68 6	3.64 6	3.59 6	3.50 6	3.32 6	3.15 6	2.84 6	2.18 6	1.29 6	2.67 5	2.01 4	7.18 2	3.27 2	1.32 −1	4.96 −32
Total I.T.	1.39 2	1.39 2	1.39 2	1.39 2	1.38 2	1.38 2	1.37 2	1.36 2	1.33 2	1.27 2	1.11 2	8.80 1	1.45 0	2.11 −18	0.0	0.0

* (a ± b) means $a \times 10^{\pm b}$　　** Unit ; Curie　　*** Annually discharged fuel …… 11.4 ton HM

Table A1.1−6 Radioactivity of Actinide in Spent Fuels Annually Discharged from HTGR Core.

Cooling time	0 d	30 d	100 d	0.5 y	1.0 y	2 y	3 y	5 y	10 y	20 y	50 y	100 y	10^3 y	10^4 y	10^5 y	10^6 y
Uranium																
U−232	1.12 −2	1.36 −2	1.88 −2	2.46 −2	3.64 −2	5.58 −2	7.08 −2	9.10 −2	1.11 −1	1.10 −1	8.33 −2	5.20 −2	1.08 −5	1.57 −42	0.0	0.0
U−234	5.15 −2	6.21 −2	8.71 −2	1.17 −1	1.85 −1	3.21 −1	4.58 −1	7.28 −1	1.39 0	2.63 0	5.80 0	9.64 0	1.74 1	1.69 1	1.32 1	1.05 0
U−235	3.43 −1	3.43 −1	3.43 −1	3.43 −1	3.43 −1	3.43 −1	3.43 −1	3.43 −1	3.43 −1	3.43 −1	3.43 −1	3.43 −1	3.43 −1	3.43 −1	3.43 −1	3.43 −1
U−236	6.60 0	6.60 0	6.60 0	6.60 0	6.60 0	6.60 0	6.60 0	6.60 0	6.60 0	6.60 0	6.61 0	6.62 0	6.84 0	8.18 0	9.00 0	8.77 0
U−237	3.13 7	1.44 6	1.16 3	7.76 1	7.54 1	7.15 1	6.79 1	6.11 1	4.70 1	2.78 1	5.74 0	4.15 −1	1.65 −4	8.42 −5	1.02 −7	7.21 −37
U−238	3.68 0	3.68 0	3.68 0	3.68 0	3.68 0	3.68 0	3.68 0	3.68 0	3.68 0	3.68 0	3.68 0	3.68 0	3.68 0	3.68 0	3.68 0	3.68 0
U−239	7.99 8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total, α	1.60 1	1.07 1	1.07 1	1.08 1	1.08 1	1.10 1	1.12 1	1.14 1	1.21 1	1.34 1	1.65 1	2.03 1	2.82 1	2.91 1	2.62 1	1.38 1
Total, β	8.30 8	1.44 6	1.16 3	7.76 1	7.54 1	7.15 1	6.79 1	6.11 1	4.70 1	2.78 1	5.74 0	4.15 −1	1.65 −4	8.42 −5	1.02 −7	7.21 −37
Neptunium																
Np−236	4.77 1	6.71 −9	6.90 −32	5.58 −59	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Np−237	6.80 0	7.06 0	7.07 0	7.07 0	7.07 0	7.08 0	7.08 0	7.09 0	7.15 0	7.33 0	8.12 0	9.58 0	2.39 1	2.87 1	2.79 1	2.08 1
Np−238	1.28 7	6.39 2	5.91 −8	8.82 −20	6.09 −46	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Np−239	7.99 8	1.16 5	5.62 2	5.62 2	5.62 2	5.62 2	5.62 2	5.62 2	5.62 2	5.61 2	5.60 2	5.57 2	5.15 2	2.35 2	9.14 −2	7.22 −36
Np−240	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total, α	6.80 0	7.06 0	7.07 0	7.07 0	7.07 0	7.08 0	7.08 0	7.09 0	7.15 0	7.33 0	8.12 0	9.58 0	2.39 1	2.87 1	2.79 1	2.08 1
Total, β	8.12 8	1.17 5	5.62 2	5.62 2	5.62 2	5.62 2	5.62 2	5.62 2	5.62 2	5.61 2	5.60 2	5.57 2	5.15 2	2.35 2	9.14 −2	0.0
Plutonium																
Pu−236	3.01 0	2.97 0	2.84 0	2.68 0	2.38 0	1.86 0	1.46 0	8.99 −1	2.66 −1	2.33 −2	1.57 −5	8.17 −11	0.0	0.0	0.0	0.0
Pu−238	4.47 4	4.60 4	4.69 4	4.76 4	4.85 4	4.88 4	4.85 4	4.78 4	4.59 4	4.24 4	3.33 4	2.22 4	1.73 1	2.48 −18	0.0	0.0
Pu−239	4.86 3	5.07 3	5.07 3	5.07 3	5.07 3	5.07 3	5.07 3	5.07 3	5.07 3	5.07 3	5.06 3	5.06 3	4.94 3	3.91 3	3.11 2	2.41 −9
Pu−240	8.69 3	8.69 3	8.69 3	8.70 3	8.70 3	8.70 3	8.71 3	8.71 3	8.73 3	8.75 3	8.77 3	8.74 3	7.95 3	3.08 3	2.33 −1	1.46 −42
Pu−241	3.45 6	3.43 6	3.40 6	3.36 6	3.27 6	3.10 6	2.95 6	2.65 6	2.04 6	1.21 6	2.49 5	1.80 4	7.16 0	3.66 0	4.45 −3	3.14 −32
Pu−242	7.39 1	7.39 1	7.39 1	7.39 1	7.39 1	7.39 1	7.39 1	7.39 1	7.39 1	7.39 1	7.39 1	7.39 1	7.38 1	7.26 1	6.16 1	1.19 1
Pu−243	1.72 7	5.18 −37	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total, α	1.73 7	5.99 4	6.08 4	6.15 4	6.23 4	6.26 4	6.24 4	6.17 4	5.98 4	5.63 4	4.72 4	3.61 4	1.30 4	7.06 3	3.73 2	1.19 1
Total, β	3.45 6	3.43 6	3.40 6	3.36 6	3.27 6	3.10 6	2.95 6	2.65 6	2.04 6	1.21 6	2.49 5	1.80 4	7.16 0	3.66 0	4.45 −3	3.14 −32
Americium																
Am−241	2.04 3	2.47 3	3.46 3	4.61 3	7.12 3	1.19 4	1.65 4	2.49 4	4.23 4	6.55 4	8.94 4	8.92 4	2.30 4	3.88 0	4.68 −3	3.30 −32
Am−242m	8.41 1	8.40 1	8.40 1	8.39 1	8.37 1	8.33 1	8.29 1	8.22 1	8.03 1	7.67 1	6.69 1	5.33 1	8.77 −1	1.28 −18	0.0	0.0
Am−242	1.97 6	8.40 1	8.40 1	8.39 1	8.37 1	8.33 1	8.29 1	8.22 1	8.03 1	7.67 1	6.69 1	5.33 1	8.77 −1	1.28 −18	0.0	0.0
Am−243	5.61 2	5.62 2	5.62 2	5.62 2	5.62 2	5.62 2	5.62 2	5.62 2	5.62 2	5.61 2	5.60 2	5.57 2	5.15 2	2.35 2	9.14 −2	7.22 −36
Am−244	4.23 6	1.47 −15	1.24 −65	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total, α	4.24 6	3.03 3	4.02 3	5.18 3	7.68 3	1.25 4	1.70 4	2.54 4	4.29 4	6.61 4	8.99 4	8.97 4	2.35 4	2.39 2	9.60 −2	3.30 −32
Total, β	1.97 6	8.40 1	8.40 1	8.39 1	8.37 1	8.33 1	8.29 1	8.22 1	8.03 1	7.67 1	6.69 1	5.33 1	8.77 −1	1.28 −18	0.0	0.0
Total, I.T.	8.41 1	8.40 1	8.40 1	8.39 1	8.37 1	8.33 1	8.29 1	8.22 1	8.03 1	7.67 1	6.69 1	5.33 1	8.77 −1	1.28 −18	0.0	0.0
Curium																
Cm−242	7.99 5	7.09 5	5.27 5	3.71 5	1.71 5	3.62 4	7.72 3	4.13 2	6.78 1	6.46 1	5.64 1	4.49 1	7.38 −1	1.08 −18	0.0	0.0
Cm−243	2.22 2	2.21 2	2.21 2	2.20 2	2.17 2	2.12 2	2.08 2	1.99 2	1.79 2	1.44 2	7.51 1	2.54 1	8.56 −8	0.0	0.0	0.0
Cm−244	5.14 4	5.15 4	5.11 4	5.06 4	4.97 4	4.77 4	4.59 4	4.24 4	3.48 4	2.35 4	7.20 3	1.00 3	3.96 −13	0.0	0.0	0.0
Cm−245	7.70 0	7.70 0	7.70 0	7.70 0	7.70 0	7.70 0	7.70 0	7.70 0	7.70 0	7.69 0	7.68 0	7.65 0	7.15 0	3.65 0	4.44 −3	3.13 −32
Total, α	8.51 5	7.61 5	5.78 5	4.22 5	2.21 5	8.42 4	5.38 4	4.30 4	3.51 4	2.37 4	7.34 3	1.08 3	7.89 0	3.65 0	4.44 −3	3.13 −32
Total α	2.24 7	8.24 5	6.43 5	4.88 5	2.91 5	1.59 5	1.33 5	1.30 5	1.38 5	1.46 5	1.44 5	1.27 5	3.65 4	7.36 3	4.27 2	4.65 1
Total β	1.65 9	4.99 6	3.40 6	3.36 6	3.27 6	3.11 6	2.95 6	2.65 6	2.04 6	1.21 6	2.50 5	1.86 4	5.23 2	2.39 2	9.58 −2	3.14 −32
Total I.T.	8.41 1	8.40 1	8.40 1	8.39 1	8.37 1	8.33 1	8.29 1	8.22 1	8.03 1	7.67 1	6.69 1	5.33 1	8.77 −1	1.28 −18	0.0	0.0

* (a ± b) means $a \times 10^{\pm b}$ ** Unit ; Curie *** Annually discharged fuel …… 11.4 ton HM

Table A1.1−7 Radioactivity of Fission Products in Spent Fuels Annually Discharged from VHTR Core.

Nuclide \ Cooling time (Yr)	0	0.5	1	5	10	50	100	500	10^3	10^4	10^5	10^6
Se^{79}	7.23+0 6.34−1	7.23+0 6.34−1	7.23+0 6.34−1	7.23+0 6.34−1	7.22+0 6.34−1	7.22+0 6.33−1	7.22+0 6.33−1	7.19+0 6.30−1	7.15+0 6.27−1	6.49+0 5.70−1	2.49+0 2.18−1	1.68−4 1.47−5
Kr^{85}	2.29+5 2.01+4	2.22+5 1.95+4	2.15+5 1.89+4	1.66+5 1.46+4	1.20+5 1.06+4	9.14+3 8.01+2	3.64+2 3.19+1	2.30−9 2.02−10	2.32−23 2.03−24	0.0 0.0		
Rb^{87}	4.41+5 3.87+4	4.38+5 3.84+4	4.35+5 3.82+4	4.12+5 3.61+4	3.84+5 3.37+4	2.01+5 1.93+4	1.10+5 9.67+3	4.29+2 3.76+1	4.17−1 3.66−2	2.50−55 2.19−56	0.0 0.0	
Sr^{89}	2.87+7 2.48+6	2.34+6 2.05+5	1.94+5 1.70+4	4.25−4 3.73−5	6.41−15 5.62−16	0.0 0.0						
Sr^{90}	1.74+6 1.52+5	1.71+6 1.50+5	1.69+6 1.49+5	1.54+6 1.35+5	1.36+6 1.20+5	5.23+5 4.58+4	1.57+5 1.38+4	1.07+1 9.35−1	6.55−5 5.74−6	0.0 0.0		
Y^{90}	1.73+6 1.52+5	1.71+6 1.50+5	1.69+6 1.49+5	1.54+6 1.35+5	1.37+6 1.20+5	5.23+5 4.59+4	1.57+5 1.38+4	1.07+1 9.35−1	6.55−5 5.76−6	0.0 0.0		
Y^{91}	3.54+7 3.11+6	4.15+6 3.64+5	4.83+5 4.23+4	1.59−2 1.40−3	7.13−12 6.25−13	0.0 0.0						
Zr^{93}	6.59+1 5.78+0	6.60+1 5.79+0	6.60+1 5.79+0	6.60+1 5.79+0	6.60+1 5.79+0	6.59+1 5.78+0	6.59+1 5.78+0	6.59+1 5.78+0	6.59+1 5.78+0	6.55+1 5.74+0	6.13+1 5.38+0	3.18+1 2.79+0
Zr^{95}	4.79+7 4.20+6	6.94+6 6.09+5	1.01+6 8.83+4	1.94−1 1.70−2	7.83−10 6.87−11	0.0 0.0						
Nb^{93m}	1.68+1 1.47+0	1.80+1 1.57+0	1.91+1 1.67+0	2.71+1 2.38+0	3.51+1 3.08+0	5.91+1 5.18+0	6.24+1 5.47+0	6.26+1 5.49+0	6.26+1 5.49+0	6.22+1 5.46+0	5.82+1 5.11+0	3.02+1 2.65+0
Nb^{94}	1.75−1 1.53−2	1.75−1 1.53−2	1.75−1 1.53−2	1.75−1 1.53−2	1.75−1 1.53−2	1.74−1 1.53−2	1.74−1 1.53−2	1.72−1 1.51−2	1.69−1 1.48−2	1.24−1 1.08−2	5.45−3 4.78−4	1.52−16 1.33−17
Nb^{95}	4.79+7 4.20+6	1.35+7 1.18+6	2.13+6 1.87+5	4.18−1 3.66−2	1.69−9 1.48−10	0.0 0.0						
Tc^{99}	3.28+2 2.87+1	3.30+2 2.89+1	3.30+2 2.89+1	3.30+2 2.89+1	3.30+2 2.89+1	3.30+2 2.89+1	3.29+2 2.89+1	3.29+2 2.89+1	3.29+2 2.88+1	3.19+2 2.80+1	2.38+2 2.09+1	1.27+1 1.11+0
Ru^{103}	3.98+7 3.49+6	1.66+6 1.46+5	6.91+4 6.06+3	6.13−7 5.38−8	9.43−21 8.27−22	0.0 0.0						
Ru^{106}	1.41+7 1.24+6	1.00+7 8.78+5	7.10+6 6.23+5	4.53+5 3.97+4	1.45+4 1.27+3	1.62−8 1.43−9	1.87−23 1.64−24	0.0 0.0				
Rh^{103m}	3.94+7 3.46+6	1.64+6 1.44+5	6.85+4 6.01+3	6.07−7 5.33−8	9.34−21 8.20−22	0.0 0.0						
Rh^{106}	1.69+7 1.49+6	1.00+7 8.78+5	7.10+6 6.23+5	4.53+5 3.97+4	1.45+4 1.27+3	1.62−8 1.43−9	1.87−23 1.64−24	0.0 0.0				
Pd^{107}	2.63+0 2.31−1	2.63+0 2.31−1	2.63+0 2.31−1	2.63+0 2.31−1	2.63+0 2.31−1	2.63+0 2.31−1	2.63+0 2.31−1	2.63+0 2.31−1	2.63+0 2.31−1	2.63+0 2.31−1	2.60+0 2.28−1	2.38+0 2.09−1
Sb^{125}	1.97+5 1.72+4	1.74+5 1.53+4	1.53+5 1.35+4	5.55+4 4.87+3	1.56+4 1.37+3	6.01−1 5.27−2	1.83−6 1.60−7	1.33−50 1.17−51	0.0 0.0			
Te^{125m}	3.77+4 3.30+3	3.81+4 3.34+3	3.41+4 2.99+3	1.24+4 1.09+3	3.48+3 3.05+2	1.34−1 1.18−2	4.08−7 3.57−8	2.97−51 2.61−52	0.0 0.0			
I^{129}	8.46−1 7.42−2	8.55−1 7.50−2	8.55−1 7.50−2	8.55−1 7.50−2	8.55−1 7.50−2	8.55−1 7.50−2	8.55−1 7.50−2	8.55−1 7.50−2	8.55−1 7.50−2	8.55−1 7.50−2	8.51−1 7.47−2	8.19−1 7.18−2
I^{131}	2.68+7 2.35+6	4.27+0 3.74−1	6.58−7 5.77−8	1.89−61 1.66−62	0.0 0.0							
Cs^{134}	4.43+6 3.88+5	3.74+6 3.28+5	3.16+6 2.77+5	8.22+5 7.21+4	1.53+5 1.34+4	2.16−1 1.90−2	1.05−8 9.25−10	3.39−67 2.98−68	0.0 0.0			
Cs^{135}	7.59+0 6.66−1	7.62+0 6.68−1	7.62+0 6.68−1	7.62+0 6.68−1	7.62+0 6.68−1	7.62+0 6.68−1	7.62+0 6.68−1	7.62+0 6.68−1	7.62+0 6.68−1	7.59+0 6.66−1	7.39+0 6.48−1	5.63+0 4.94−1
Cs^{137}	2.51+6 2.20+5	2.48+6 2.18+5	2.45+6 2.15+5	2.24+6 1.96+5	1.99+6 1.75+5	7.96+5 6.98+4	2.52+5 2.21+4	2.58+1 2.27+0	2.66−4 2.33−5	0.0 0.0		
Ba^{137m}	2.35+6 2.06+5	2.32+6 2.03+5	2.29+6 2.01+5	2.09+6 1.83+5	1.86+6 1.64+5	7.44+5 6.53+4	2.36+5 2.07+4	2.42+1 2.12+0	2.49−4 2.18−5	0.0 0.0		
Ce^{141}	4.77+7 4.19+6	9.83+5 8.62+4	2.01+4 1.76+3	6.04−10 5.30−11	7.61−27 6.67−28	0.0 0.0						
Ce^{144}	3.35+7 2.94+6	2.15+7 1.88+6	1.38+7 1.21+6	3.91+5 3.43+4	4.56+3 4.00+2	1.57−12 1.37−13	7.32−32 6.42−33	0.0 0.0				
Pr^{144}	3.39+7 2.97+6	2.15+7 1.88+6	1.38+7 1.21+6	3.91+5 3.43+4	4.56+3 4.00+2	1.57−12 1.37−13	7.32−32 6.42−33	0.0 0.0				
Pm^{147}	3.92+6 3.44+5	3.62+6 3.17+5	3.17+6 2.78+5	1.10+6 9.65+4	2.93+5 2.57+4	7.48+0 6.56−1	1.35−5 1.19−6	1.56−51 1.37−52	0.0 0.0			
Pm^{148m}	5.61+6 4.92+5	2.66+5 2.34+4	1.26+4 1.11+3	3.18−7 2.79−8	1.80−20 1.58−21	0.0 0.0						
Sm^{151}	3.93+3 3.45+2	4.08+3 3.58+2	4.07+3 3.57+2	3.95+3 3.46+2	3.80+3 3.34+2	2.82+3 2.48+2	1.94+3 1.71+2	9.84+1 8.63+0	2.36+0 2.07−1	1.67−29 1.46−30	0.0 0.0	
Eu^{154}	2.66+5 2.33+4	2.54+5 2.23+4	2.43+5 2.14+4	1.71+5 1.50+4	1.09+5 9.59+3	3.12+3 2.74+2	3.66+1 3.21+0	1.30−14 1.14−15	6.38−34 5.60−35	0.0 0.0		
Eu^{155}	1.76+5 1.55+4	1.64+5 1.44+4	1.53+5 1.34+4	8.81+4 7.73+3	4.40+4 3.86+3	1.71+2 1.50+1	1.67−1 1.46−2	1.33−25 1.16−26	9.98−56 8.75−57	0.0 0.0		
Tb^{158}	8.87−1 7.78−2	8.86−1 7.77−2	8.86−1 7.77−2	8.84−1 7.76−2	8.82−1 7.74−2	8.66−1 7.60−2	8.46−1 7.43−2	7.04−1 6.17−2	5.58−1 4.90−2	8.70−3 7.63−4	7.33−21 6.43−22	0.0 0.0
Total F. P.	4.75+9 4.17+8	1.12+8 9.83+6	6.16+7 5.41+6	1.19+7 1.05+6	7.75+6 6.80+5	2.82+6 2.48+5	9.16+5 8.04+4	1.11+3 9.76+1	5.09+2 4.47+1	4.93+2 4.32+1	3.86+2 3.39+1	8.35+1 7.33+0

* Unit : Curie

** (a± b) means $a \times 10^{\pm b}$

*** Upper······for total discharged fuels (11.4 ton H.M.)

Lower······for unit discharged fuel (1 ton H.M.)

Table A1.1—8 Radioactivity of Fission Products in Spent Fuels Annually Discharged from HTGR Core.

Nuclide \ Cooling time (Yr)	0	0.5	1	5	10	50	100	500	10^3	10^4	10^5	10^6
Se^{79}	7.49+0 6.57−1	7.49+0 6.57−1	7.49+0 6.57−1	7.49+0 6.57−1	7.49+0 6.57−1	7.48+0 6.57−1	7.48+0 6.56−1	7.45+0 6.53−1	7.41+0 6.50−1	6.73+0 5.90−1	2.58+0 2.26−1	1.74−4 1.52−5
Kr^{85}	2.41+5 2.12+4	2.34+5 2.05+4	2.26+5 1.99+4	1.75+5 1.53+4	1.27+5 1.11+4	9.61+3 8.43+2	3.83+2 3.36+1	2.43−9 2.13−10	2.44−23 2.14−24	0.0 0.0		
Rb^{87}	4.59+5 4.03+4	4.56+5 4.00+4	4.53+5 3.97+4	4.29+5 3.76+4	4.00+5 3.51+4	2.30+5 2.01+4	1.15+5 1.01+4	4.46+2 3.92+1	4.34−1 3.81−2	2.60−55 2.28−56	0.0 0.0	
Sr^{89}	4.40+7 3.86+6	3.65+6 3.20+5	3.03+5 2.65+4	6.63−4 5.82−5	1.00−14 8.77−16	0.0 0.0						
Sr^{90}	1.81+6 1.59+5	1.79+6 1.57+5	1.77+6 1.55+5	1.61+6 1.41+5	1.43+6 1.25+5	5.47+5 4.79+4	1.65+5 1.44+4	1.11+1 9.78−1	6.85−5 6.01−6	0.0 0.0		
Y^{90}	1.81+6 1.58+5	1.79+6 1.57+5	1.77+6 1.55+5	1.61+6 1.41+5	1.43+6 1.25+5	5.47+5 4.80+4	1.65+5 1.44+4	1.11+1 9.78−1	6.85−5 6.01−6	0.0 0.0		
Y^{91}	5.49+7 4.81+6	6.43+6 5.64+5	7.48+5 6.56+4	2.47−2 2.17−3	1.10−11 9.69−13	0.0 0.0						
Zr^{93}	6.74+1 5.91+0	6.74+1 5.92+0	6.74+1 5.92+0	6.74+1 5.92+0	6.74+1 5.92+0	6.74+1 5.92+0	6.74+1 5.92+0	6.74+1 5.91+0	6.74+1 5.91+0	6.69+1 5.87+0	6.27+1 5.50+0	3.25+1 2.85+0
Zr^{95}	7.25+7 6.36+6	1.05+7 9.22+5	1.52+6 1.34+5	2.93−1 2.57−2	1.19−9 1.04−10	0.0 0.0						
Nb^{93m}	1.80+1 1.58+0	1.92+1 1.68+0	2.03+1 1.78+0	2.84+1 2.49+0	3.64+1 3.20+0	6.05+1 5.30+0	6.38+1 5.59+0	6.40+1 5.62+0	6.40+1 5.62+0	6.36+1 5.58+0	5.96+1 5.22+0	3.09+1 2.71+0
Nb^{94}	1.87−1 1.64−2	1.87−1 1.64−2	1.87−1 1.64−2	1.87−1 1.64−2	1.87−1 1.64−2	1.86−1 1.64−2	1.86−1 1.63−2	1.84−1 1.61−2	1.80−1 1.58−2	1.32−1 1.16−2	5.82−3 5.11−4	1.62−16 1.42−17
Nb^{95}	7.22+7 6.33+6	2.04+7 1.79+6	3.22+6 2.83+5	6.32−1 5.55−2	2.56−9 2.24−10	0.0 0.0						
Tc^{99}	3.27+2 2.87+1	3.30+2 2.90+1	3.30+2 2.90+1	3.30+2 2.90+1	3.30+2 2.90+1	3.30+2 2.90+1	3.30+2 2.90+1	3.30+2 2.89+1	3.29+2 2.89+1	3.20+2 2.80+1	2.38+2 2.09+1	1.27+1 1.12+0
Ru^{103}	5.79+7 5.08+6	2.42+6 2.12+5	1.01+5 8.83+3	8.92−7 7.83−8	1.37−20 1.20−21	0.0 0.0						
Ru^{106}	1.58+7 1.38+6	1.12+7 9.81+5	7.93+6 6.95+5	5.06+5 4.43+4	1.62+4 1.42+3	1.81−8 1.59−9	2.09−23 1.83−24	0.0 0.0				
Rh^{103m}	5.37+7 5.03+6	2.39+6 2.10+5	9.97+4 8.75+3	8.84−7 7.76−8	1.36−20 1.19−21	0.0 0.0						
Rh^{106}	2.11+7 1.85+6	1.12+7 9.81+5	7.93+6 6.95+5	5.06+5 4.43+4	1.62+4 1.42+3	1.81+8 1.59+9	2.09−23 1.83−24	0.0 0.0				
Pd^{107}	2.40+0 2.11−1	2.40+0 2.11−1	2.40+0 2.11−1	2.40+0 2.11−1	2.40+0 2.11−1	2.40+0 2.11−1	2.40+0 2.11−1	2.40+0 2.11−1	2.40+0 2.11−1	2.40+0 2.10−1	2.38+0 2.08−1	2.17+0 1.91−1
Sb^{125}	2.04+5 1.79+4	1.81+5 1.59+4	1.60+5 1.40+4	5.78+4 5.07+3	1.62+4 1.42+3	6.26−1 5.49−2	1.90−6 1.67−7	1.39−50 1.22−51	0.0 0.0			
Te^{125m}	3.67+4 3.22+3	3.94+4 3.46+3	3.55+4 3.11+3	1.29+4 1.13+3	3.62+3 3.17+2	1.40−1 1.22−2	4.24−7 3.72−8	3.09−51 2.71−52	0.0 0.0			
I^{129}	8.27−1 7.25−2	8.39−1 7.36−2	8.39−1 7.36−2	8.39−1 7.36−2	8.39−1 7.36−2	8.39−1 7.36−2	8.39−1 7.36−2	8.39−1 7.36−2	8.39−1 7.36−2	8.39−1 7.36−2	8.36−1 7.33−2	8.04−1 7.05−2
I^{131}	3.98+7 3.49+6	6.33+0 5.56−1	7.97−7 6.99−8	2.80−61 2.46−62	0.0 0.0							
Cs^{134}	4.54+6 3.99+5	3.84+6 3.37+5	3.25+6 2.85+5	8.44+5 7.40+4	1.57+5 1.38+4	2.22−1 1.95−2	1.08−8 9.50−10	3.48−67 3.06−68	0.0 0.0			
Cs^{135}	5.77+0 5.06−1	5.80+0 5.09−1	5.80+0 5.09−1	5.80+0 5.09−1	5.80+0 5.09−1	5.80+0 5.09−1	5.80+0 5.09−1	5.80+0 5.09−1	5.80+0 5.09−1	5.78+0 5.07−1	5.63+0 4.94−1	4.29+0 3.76−1
Cs^{137}	2.52+6 2.21+5	2.49+6 2.19+5	2.47+6 2.16+5	2.25+6 1.97+5	2.01+6 1.76+5	8.00+5 7.02+4	2.54+5 2.23+4	2.60+1 2.28+0	2.67−4 2.35−5	0.0 0.0		
Ba^{137m}	2.37+6 2.08+5	2.33+6 2.05+5	2.31+6 2.02+5	2.10+6 1.84+5	1.87+6 1.64+5	7.48+5 6.56+4	2.37+5 2.08+4	2.43+1 2.13+0	2.50−4 2.19−5	0.0 0.0		
Ce^{141}	7.17+7 6.29+6	1.48+6 1.29+5	3.02+4 2.65+3	9.08−10 7.96−11	1.14−26 1.00−27	0.0 0.0						
Ce^{144}	4.29+7 3.76+6	2.75+7 2.41+6	1.76+7 1.55+6	5.00+5 4.39+4	5.84+3 5.12+2	2.00−12 1.76−13	9.37−32 8.22−33	0.0 0.0				
Pr^{144}	4.37+7 3.83+6	2.75+7 2.41+6	1.76+7 1.55+6	5.00+5 4.39+4	5.84+3 5.12+2	2.00−12 1.76−13	9.37−32 8.22−33	0.0 0.0				
Pm^{147}	4.26+6 3.73+5	4.01+6 3.52+5	3.51+6 3.08+5	1.22+6 1.07+5	3.25+5 2.85+4	8.28+0 7.27−1	1.50−5 1.32−6	1.73−51 1.52−52	0.0 0.0			
Pm^{148m}	8.56+6 7.51+5	4.06+5 3.56+4	1.93+4 1.69+3	4.85−7 4.25−8	2.74−20 2.41−21	0.0 0.0						
Sm^{151}	4.04+3 3.54+2	4.28+3 3.75+2	4.26+3 3.74+2	4.13+3 3.63+2	3.98+3 3.49+2	2.96+3 2.59+2	2.04+3 1.79+2	1.03+2 9.04+0	2.47+0 2.17−1	1.74−29 1.53−30	0.0 0.0	
Eu^{154}	2.57+5 2.25+4	2.45+5 2.15+4	2.35+5 2.06+4	1.64+5 1.44+4	1.05+5 9.25+3	3.01+3 2.64+2	3.53+1 3.09+0	1.26−14 1.10−15	6.15−34 5.40−35	0.0 0.0		
Eu^{155}	1.70+5 1.49+4	1.59+5 1.39+4	1.48+5 1.30+4	8.50+4 7.45+3	4.25+4 3.73+3	1.65+2 1.45+1	1.61−1 1.41−2	1.28−25 1.12−26	9.62−56 8.44−57	0.0 0.0		
Tb^{158}	8.26−1 7.25−2	8.26−1 7.25−2	8.26−1 7.24−2	8.24−1 7.23−2	8.22−1 7.21−2	8.07−1 7.08−2	7.89−1 6.92−2	6.56−1 5.75−2	5.20−1 4.56−2	8.11−3 7.11−4	6.83−21 6.00−22	0.0 0.0
Total F.P.	7.09+9 6.22+8	1.44+8 1.26+7	7.37+7 6.47+6	1.26+7 1.10+6	7.96+6 6.98+5	2.89+6 2.53+5	9.38+5 8.23+4	1.14+3 9.97+1	5.10+2 4.48+1	4.94+2 4.33+1	3.87+2 3.39+1	8.34+1 7.31+0

* Units: Curie *** Upper…… for total discharged fuels (11.4 ton H.M.)

**(a ± b) means $a\times10^{\pm b}$ Lower…… For unit discharged fuel (1 ton H.M.)

Table A1.1-9 Radioactivity of Fission Products and Actinide in Spent Fuels and Wastes Annually Discharged from VHTR and HTGR Cores

(i) For VHTR Core

Cooling time		(0 day)	(180 day) 0 year	0.5	1.0	5	10	50	100	500	10^3	10^4	10^5	10^6
Actinide	A	1.16_8+9	4.29_9+6	3.90_8+6	3.68_2+6	2.92_5+6	2.29_1+6	4.21_1+5	1.57_9+5	6.87_4+4	3.90_6+4	7.43_9+3	4.06_2+2	5.06_4+1
	B	—	6.48_9+5	3.49_2+5	2.10_4+5	8.11_7+4	6.79_5+4	2.13_6+4	1.04_9+4	4.85_8+3	3.10_0+3	8.84_1+2	3.16_3+1	6.01_0+0
F.P.	A	4.74_9+9	1.13_4+8	6.19_9+7	4.31_9+7	1.10_2+7	7.59_2+6	2.79_0+6	9.06_5+5	1.10_8+3	5.09_5+2	4.93_0+2	3.86_2+2	8.35_3+1
	B	—	1.13_1+8	6.17_8+7	4.29_8+7	1.08_5+7	7.47_6+6	2.78_1+6	9.06_2+5	1.10_8+3	5.09_4+2	4.93_0+2	3.86_1+2	8.34_5+1
Total	A	5.91_7+9	1.17_7+8	6.59_0+7	4.68_7+7	1.39_4+7	9.88_4+6	3.21_1+6	1.06_4+6	6.98_5+4	3.95_7+4	7.93_2+3	7.92_4+2	1.34_2+2
	B	—	1.13_8+8	6.21_3+7	4.31_9+7	1.09_4+7	7.54_4+6	2.80_3+6	9.16_7+5	5.96_6+3	3.60_9+3	1.37_7+3	4.17_7+2	8.94_6+1

(ii) For HTGR Core

Cooling time		(0 day)	(180 day) 0 year	0.5	1.0	5	10	50	100	500	10^3	10^4	10^5	10^6
Actinide	A	1.67_0+9	3.85_4+6	3.56_7+6	3.39_0+6	2.71_6+6	2.12_6+6	3.87_9+5	1.45_0+5	6.44_3+4	3.70_3+4	7.59_6+3	4.27_3+2	4.65_5+1
	B	—	4.41_9+5	2.39_4+5	1.45_5+5	5.77_7+4	4.83_2+4	1.52_8+4	7.68_5+3	3.64_4+3	2.31_3+3	6.49_2+2	2.56_1+1	6.02_9+0
F.P.	A	7.08_8+9	1.45_5+8	7.42_2+7	5.02_2+7	1.15_4+7	7.79_0+6	2.85_6+6	9.28_1+5	1.13_2+3	5.10_3+2	4.93_8+2	3.86_9+2	8.33_8+1
	B	—	1.45_2+8	7.39_9+7	5.00_0+7	1.13_7+7	7.66_8+6	2.84_6+6	9.27_7+5	1.13_2+3	5.10_3+2	4.93_7+2	3.86_9+2	8.33_0+1
Total	A	8.75_9+9	1.49_3+8	7.77_9+7	5.36_1+7	1.42_5+7	9.91_6+6	3.24_4+6	1.07_3+6	6.55_7+4	3.75_4+4	8.09_0+3	8.14_3+2	1.29_9+2
	B	—	1.45_7+8	7.42_3+7	5.01_4+7	1.14_3+7	7.71_6+6	2.86_2+6	9.35_4+5	4.77_6+3	2.82_3+3	1.14_3+3	4.12_5+2	8.93_3+1

* Unit : Curie ** $(a \pm b)$ means $a \times 10^{\pm b}$

*** Total discharged fuel : 11.4 ton H.M.

**** A : spent fuels B : wastes released in reprocessing after 180 days cooling, which consist of 0.1 % of U, 0.3 % of Pu and all the FP excluding 10 % of I and 100 % of Br, Kr and Xe.

Table A1.1-10 Radioactivity of ^{3}H and ^{14}C in Fuel Elements and Reactor Cores of HTR's

		Cooling time (Yr)	0	0.5	1	5	10	50	10^2	10^3	10^4	10^5
VHTR	^{3}H	In fuel (fission) (Curie/Year)*	1.13_5+4	1.10_3+4	1.07_2+4	8.55_9+3	6.45_5+3	6.76_5+2	4.03_4+1	0.0		
		(Curie/ton HM)	9.95_6+2	9.67_5+2	9.40_4+2	7.50_8+2	5.66_2+2	5.93_4+1	3.53_9+0	0.0		
		In coolant He (^{3}He) (Curie/kg)	8.46_7+2	8.23_1+2	8.00_3+2	6.38_8+2	4.81_7+2	5.04_9+1	3.01_1+0	0.0		
		In graphite (Imp. ^{6}Li)** (Curie/ton)	2.51_4+1	2.44_4+1	2.37_6+1	1.89_6+1	1.43_0+1	1.49_9+0	8.93_8-2	0.0		
	^{14}C	In graphite (^{13}C) (Curie/ton)	6.15_8-2	6.15_7-2	6.15_7-2	6.15_4-2	6.15_0-2	6.12_0-2	6.08_2-2	5.43_7-2	1.77_3-2	2.40_6-7
		In graphite (^{14}N)*** (Curie/ton)	5.00_3-1	5.00_3-1	5.00_3-1	5.00_0-1	4.99_7-1	4.97_2-1	4.94_1-1	4.41_8-1	4.44_0-1	1.95_5-6
HTGR	^{3}H	In fuel (fission) (Curie/Year)*	1.13_8+4	1.10_6+4	1.07_5+4	8.58_4+3	6.47_3+3	6.78_4+2	4.04_5+1	0.0		
		(Curie/ton HM)	9.98_2+2	9.70_2+2	9.43_0+2	7.53_0+2	5.67_8+2	5.95_1+1	3.54_8+0	0.0		
		In coolant He (^{3}He) (Curie/kg)	8.75_1+2	8.50_8+2	8.27_2+2	6.60_2+2	4.97_9+2	5.21_8+1	3.11_2+0	0.0		
		In graphite (Imp. ^{6}Li)** (Curie/ton)	2.53_1+1	2.46_0+1	2.39_2+1	1.90_9+1	1.44_0+1	1.50_9+0	8.99_8-2	0.0		
	^{14}C	In graphite (^{13}C) (Curie/ton)	5.78_3-2	5.78_3-2	5.78_2-2	5.77_9-2	5.77_6-2	5.74_7-2	5.71_1-2	5.10_6-2	1.66_5-2	2.25_9-7
		In graphite (^{14}N)*** (Curie/ton)	4.71_6-1	4.71_6-1	4.71_6-1	4.71_3-1	4.71_1-1	4.68_7-1	4.65_8-1	4.16_4-1	1.35_8-1	1.84_3-6

* Total discharged fuels --- 11.4 ton , ** Content of impurity, ^{6}Li in graphite --- 0.1 ppm., *** Content of impurity, ^{14}N in graphite --- 30 ppm.

(Note)

(1) (a ± b) means $a \times 10^{\pm b}$ (2) Charged graphite in core --- VHTR 564 ton, HTGR 378 ton

(3) Helium weight of 1 m^3 --- 1.75 kg

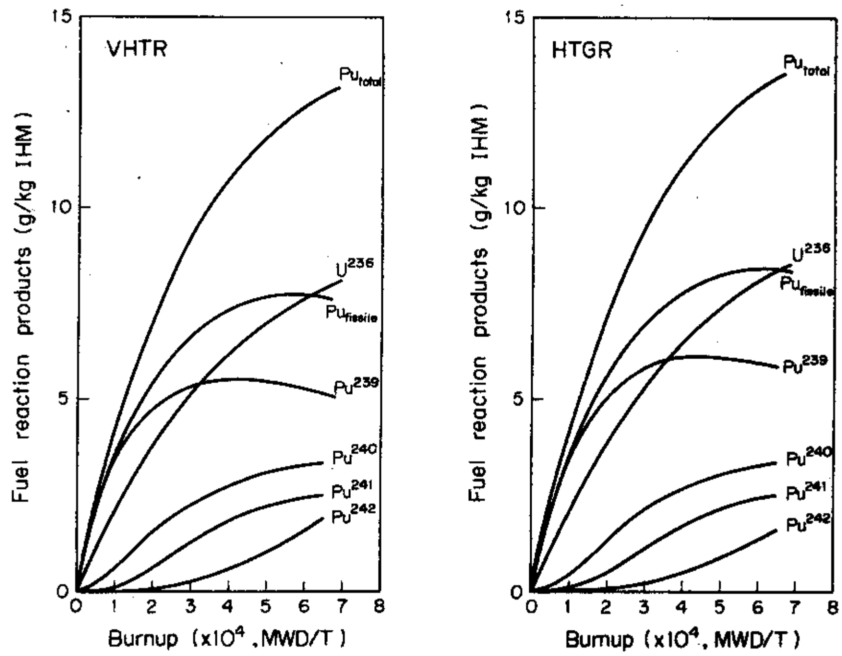

Fig.A1.1-1 Build up of Reaction Products with Fuel Burnup in HTR Cores

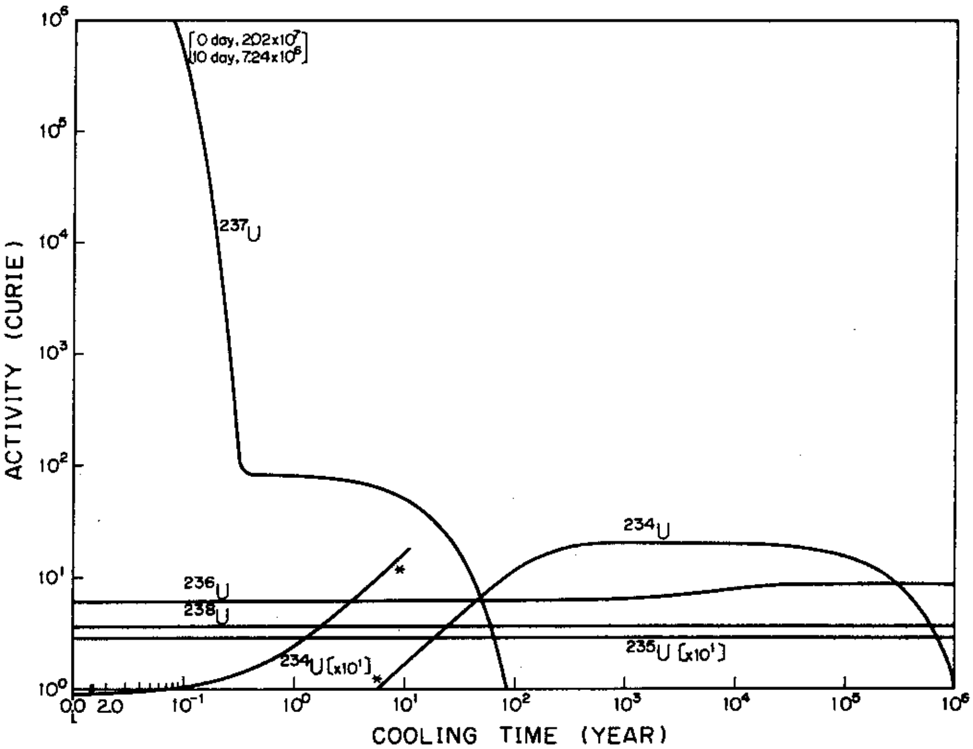

Fig.A1.1-2 Activity of Uranium Isotopes in the Spent Fuels Annually Discharged from VHTR Core.

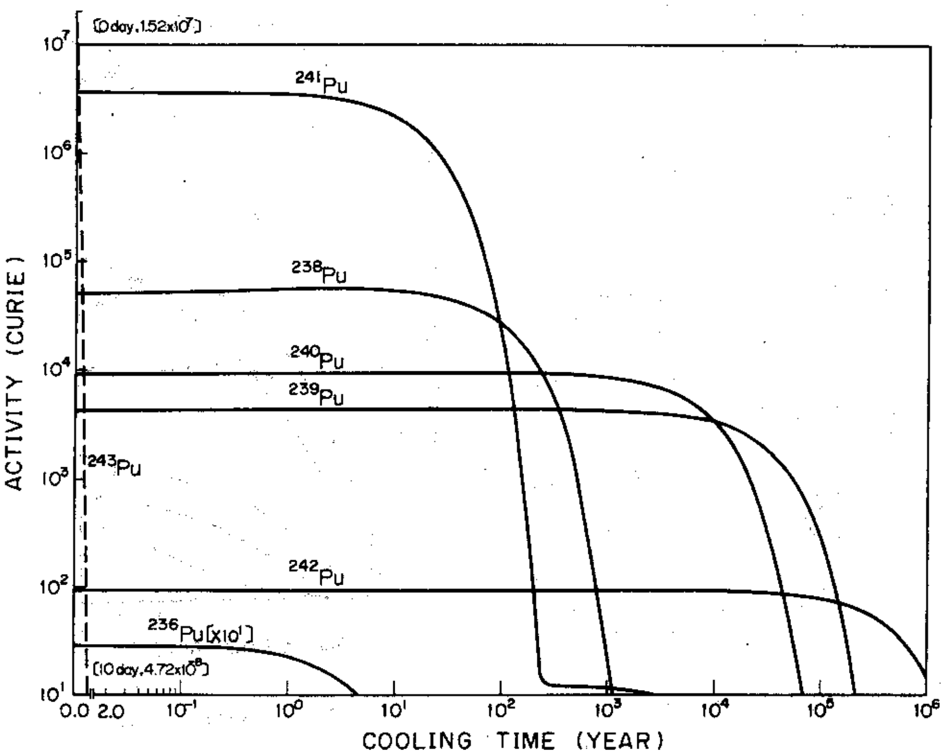

Fig.Al.1-3 Activity of Plutonium Isotopes in the Spent Fuels Annually Discharged from VHTR Core.

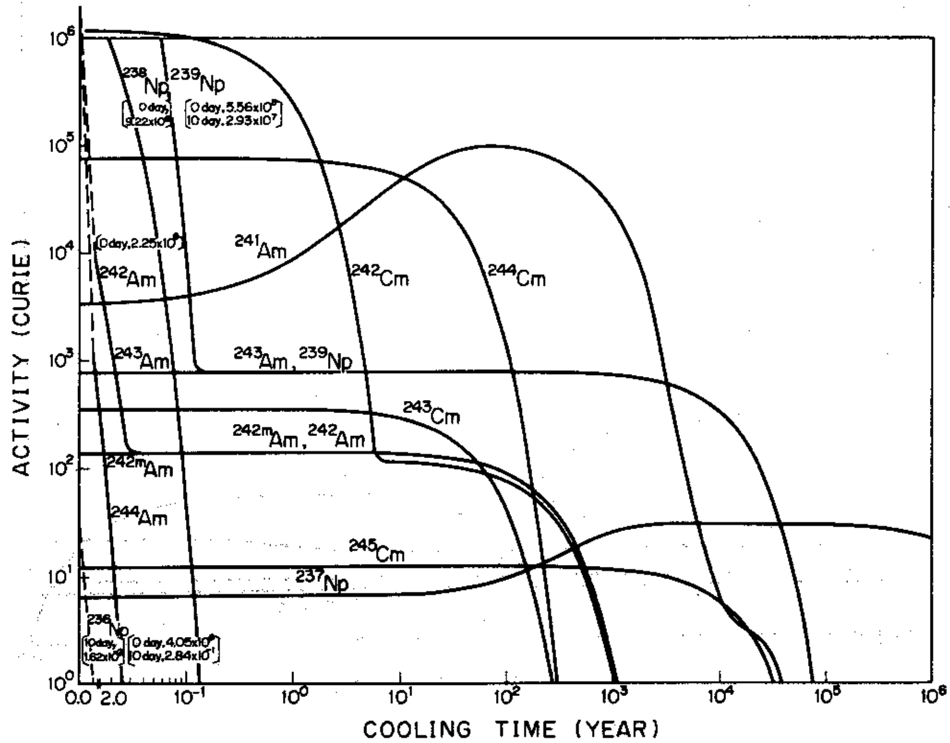

Fig.Al.1-4 Activity of Neptunium, Americium and Curium Isotopes in the Spent Fuels Annually Discharged from VHTR Core.

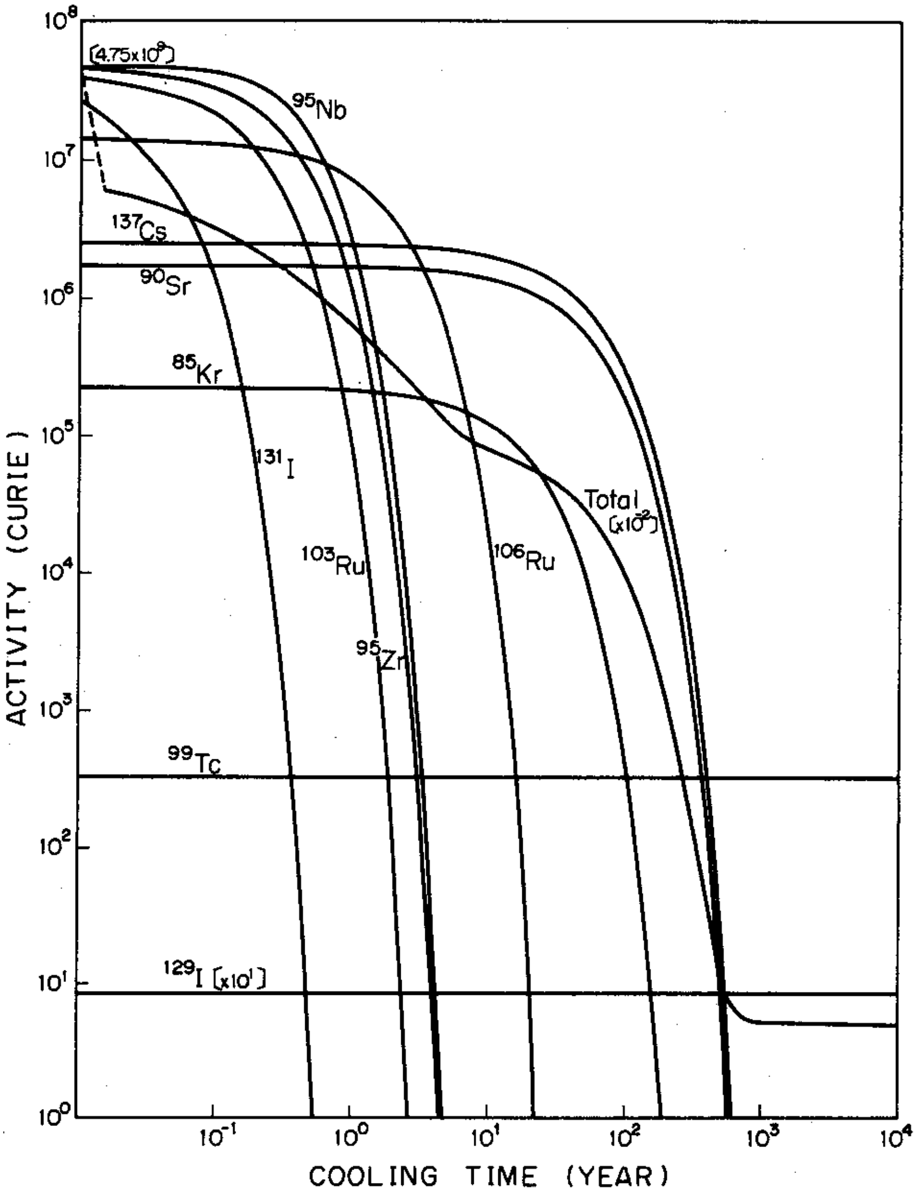

Fig.A1.1-5 Activity of Main Fission Products in the Spent Fuels Annually Discharged from VHTR Core.

A1.2 Fuel material balance of the mixed uranium and plutonium fuel use in HTR

The fuel material balances were studied to evaluate possibilities of once through plutonium use in VHTR.

The conditions of the analysis are as follows;

1) N_c/N_{HM} 400 ~ 600
2) Concentration of fissile nuclide in fuel (w/o) 2 ~ 6
3) Enrichment of base uranium Natural
4) Plutonium composition ($^{239}Pu/^{240}Pu/^{241}Pu/^{242}Pu$) (w/o)
   Type A* 59/24/13/4
   Type B** 35/30/15/20

(* From LWR spent fuels, ** From HTR spent fuels)

Some results are shown in Table A1.2-1 and Fig.A1.2-1. The main features of the once-through mixed fuel cycle (Pu/U-VHTR) may be summarized as follows;

1) The Pu/U-VHTR cycle leads to higher N_c/N_{HM} and shorter fuel dwelling time than the LEU cycle.
2) The use of the plutonium discharged from LWR requires fissile enrichments over 4 w%, N_c/N_{HM} exceeding 500 and 3~4 batches refueling for VHTR to achieve the current burnup level of LWR fuels. However, the achievement seems to be difficult with the plutonium discharged from HTR.
3) The Pu/U-VHTR cycle has some advantages such as no need of separative work and smaller, initial excess reactivity and natural uranium requirements (about 1/10 of LEU cycle), while it requires more plutonium which corresponds to one extracted from the annually discharged fuels of 3~4 LWR's.

It is concluded that the continuous refueling appears to be essential to the positive use of Pu/U in HTR.

Table A1.2-1 Fuel Material Balance in Plutonium Fueled VHTR Cores

N_c/N_u		500	600
Thermal power	(MW)	3000	3000
Power density	(W/cm^3)	6	6
Specific power	(W/g)	117	141
Fissile nuclide conc.	(w/o)	6	6
Fuel burnup	(MWD/T)	62500	72200
Fuel dwelling time	(Yr)	2.1	2.0
Conversion ratio		0.54/0.63	0.49/0.62
Initial fuel loading	(t HM)	25.6_4	21.2_8
Natural uranium	(t HM)	23.7_4	19.7_0
Plutonium**	(t HM)		
(Fissile)		1.3_7	1.1_4
(Total)		1.9_0	1.5_8
Fuel reloading	(t HM)	12.2_1	10.6_4
Natural uranium	(t HM)	11.3_0	9.8_5
Plutonium	(t HM)		
(Fissile)		0.65_2	0.57_0
(Total)		0.90_5	0.79_0
Fuel discharge	(t HM)		
Uranium	(t HM)	10.8_2	9.3_9
(fissile nuclide conc.	w/o)	(0.37)***	(0.29)***
Plutonium	(t HM)		
(Fissile)		0.28_1	0.16_8
(Total)		0.57_9	0.42_6
Natural uranium requirement			
Initial core	(t HM)	23.7_4	19.7_0
Annual reload	(t HM)	11.3_0	9.8_5
30 year cum.	(t HM)	351	305
Plutonium requirement**			
Initial core	(t HM)	1.90(1.37)***	1.58(1.14)***
Annual reload	(t HM)	0.91(0.65)***	0.79(0.57)***
30 year cum.	(t HM)	28.3(20.2)***	24.5(17.7)***

* Plant factor --- 70 %

** Plutonium from LWR spent fuels

*** Fissile plutonium

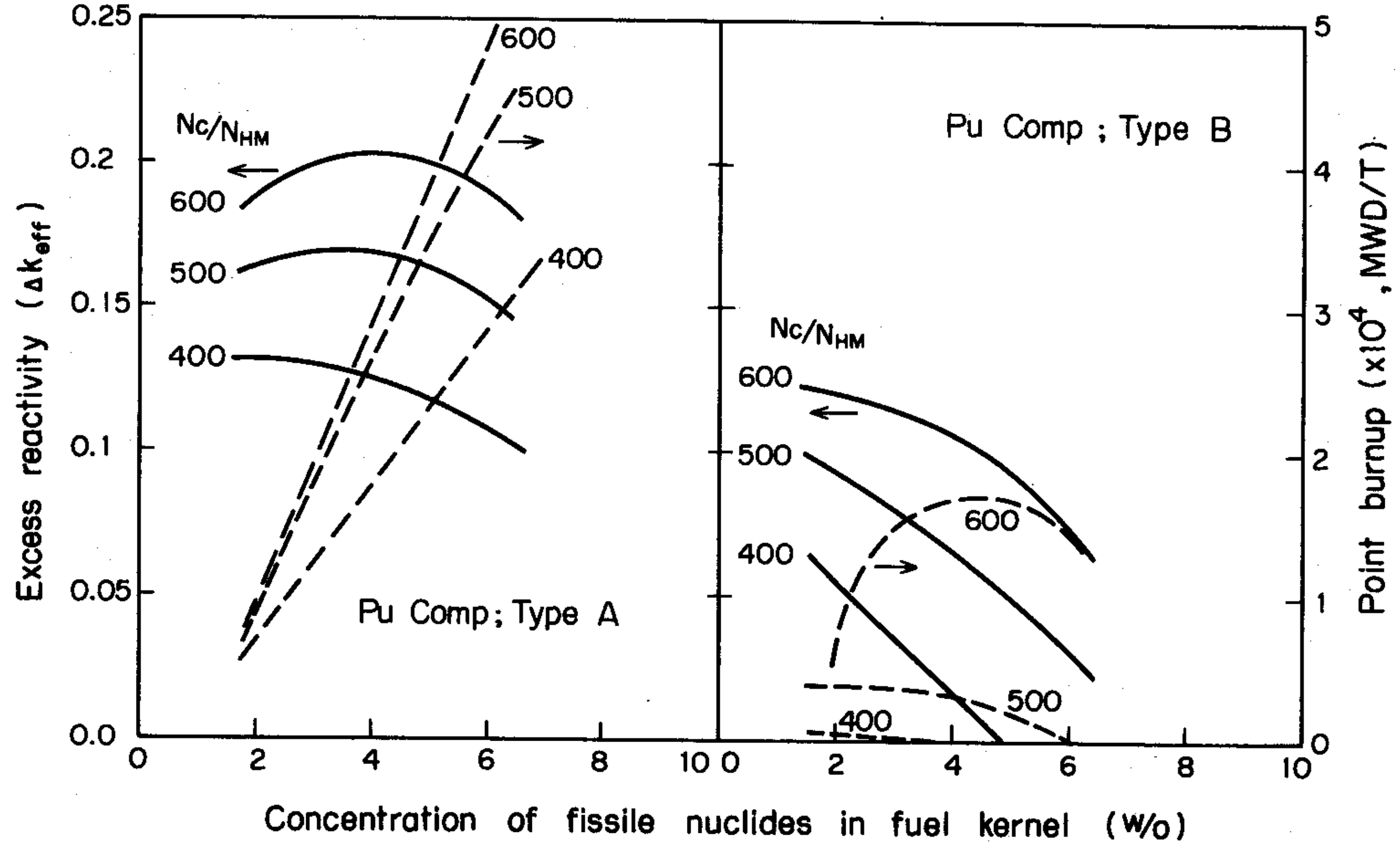

Fig.A1.2-1 Excess Reactivity and Point Burnup for Plutonium Fueled VHTR Cores.

A2 Fuel reprocessing

For the discussion on chapter 4.3, following tables are preliminarily prepared;

(1) Table A2-1 Process or operation option in transfer, storage and reprocessing of spent HTR fuel.

(2) Table A2-2 Distribution of radioactive nuclides assumed for their material balance in VHTR head-end processing.

(3) Table A2-3 Material balance of head-end process of spent fuel annually discharged from 3000 MW(t) VHTR for longer cooling time, i.e. 10 years and 50 years.

Table A2-1 Process or Operation Option in Transfer, Storage and Reprocessing of Spent HTR Fuel

Fuel cycle step	Process/operation option	Preliminary evaluation of			Technology status	Required R & D
		Technical feasibility a)	Proliferation resistance	Safety		
Transport	Transport of whole block	NF	- Difficult accessibility and material modificability, due to radioactivities, and voluminous and complex fuel configuration	Higher risk due to larger transport capacity	- Conceptual design of cask and transport system - Under safety analysis	- Realization and testing - Safety analysis and evaluation
	Transport of disassembled fuel with reduced volume	FA	- Less difficult than the above option especially in material modificability	- More stringent control of cooling and criticality required - More stringent confinement of radioactivity required	- Conceptual study	- Compact on-site head-end process - Cask selection
Storage	Storage of whole block	NF	- Difficult accessibility and material modificability due to radioactivity, and voluminous and complex fuel configuration	(- Larger storage area required)	- Conceptual design - Under safety analysis	- Demonstration - Safety analysis
	Storage of disassembled fuel with reduced volume	FA	- Less difficult than the above option especially in material modificability	- More stringent control of cooling and criticality, required	- Conceptual study	- Design of storage system - Simple disassembling method
Reprocessing (Headend step only)	Partial burn by disassembling fuel block	FA	- Easier accessibility due to smaller volume - Difficult facility modificability	- Smaller waste volume	- Conceptual study	- Demonstration of disassembling with irradiated fuel block - Safety disposal method of disassembled fuel graphite
	Whole block burn with air/O_2 and roll crushing of SiC coated particle	NF	- Difficult accessibility due to larger fuel volume and radioactivity - Difficult facility modificability	- Larger waste volume as $CaCO_3$ - More powder handling and mechanical operation required, causing process unstability - Difficult reaction control due to exothermic reaction	- Cold pilot scale testing	- Demonstration of operability and maintenability - Development of off gas treatment - Safety analysis of voluminous waste
	Whole block burn with CO_2 and jet grinding of SiC coating	ID	- Difficult accessibility due to larger fuel volume and radioactivity - Difficult facility modificability	- Smaller waste volume - Less powder handling and mechanical operation with a merit of improved process stability	- Laboratory scale study	- Improvement of burning rate - Optimization of grinding conditions - R & D of burner and jet grinder - R & D of off gas treatment process

a) Note NF : Technology in advanced development stage, realized within 10~15 years
FA : Need further analysis in comparison with the advanced technology
ID : Insufficient data for the evaluation; R/D required

Table A2-2 Distribution of Radioactive Nuclides Assumed for their Material Balance in VHTR Head-end Processing

Nuclides \ Distribution	Head-end processing				Dissolution and purification	
	Volatilization fraction to off gas system	Collected as off gas waste	Released to environment as airborn waste	With dismantled graphite	With SiC hulls	As HLW g)
Volatile fission products						
^{3}H	1.00 b)	0.99 b)	0.01 b)	1.00 c)	0	0
^{14}C	1.00 b)	0.99 b)	0.01 b)	1.00 c)	0	0
^{85}Kr	1.00	0.99	0.01	0	0	0
^{129}I	1.00	0.999	10^{-3}	0	0	0
^{131}I	1.00	0.999	10^{-3}	0	0	0
Semi volatile fission products d)	0.10	0.10	10^{-8}	0	0.01 f)	0.89
Particulates e)	0.001	0.001	10^{-9}	0	—	—
Non volatile fission products	—	—	—	0	0.01 f)	0.989
Uranium total	0.001	0.001	10^{-8}	0	0.005 f)	0.005
Plutonium total	0.001	0.001	10^{-9}	0	0.005 f)	0.01
Other actinides	0.001	0.001	10^{-9}	0	0.005 f)	0.995

a) Values are fraction of total amount in spent fuel except ^{3}H and ^{14}C.
b) Fraction of total amounts in sleeve, compact and coated particle.
c) Fraction of total amounts in block graphite.
d) Includes nuclides of Cs, Ru, Sb and Ce only.
e) Refers airborn nuclides of non-volatile fission products and actinides.
f) Refers insoluble residues
g) Assumes zero release of ILW and LLW.

Table A2-3(A) Material Balance of Head-end Process for Spent Fuel Annually Discharged from 3000 MW(t) VHTR after 10 Year Cooling

Stream[d)] / Composition		Feed SF ①	Dissolved solution ②	Dismantled graphite ③	Decomposed graphite ④	SiC hull ⑤	Waste from off gas cleaning ⑥	Release to environment ⑦
Actinide	kg	11,400	11,343	— a)	— b)	45.6	11.4	1.14×10^{-4}
	Ci	2.35×10^{6}	2.34×10^{6}	— a)	47 b)	9.4×10^{3}	2.35×10^{3}	2.25×10^{-3}
Fission product	kg	790	622	— a)	— b)	7.9	35	.125
	Ci	7.76×10^{6}	7.32×10^{6}	— a)	— b)	7.62×10^{4}	3.5×10^{5}	1.2×10^{3}
Graphite	kg	215,170	—	147,300	67,870	—	—	—
Others	kg	3,570 (SiC, O_2)	—	—	—	1,930 (SiC)	—	1,640 (O_2)
Total mass	kg	230,930	11,965	147,300	67,870	1,984	40.4	1,760
Total radio-activities	Ci	1.01×10^{7}	9.66×10^{6}	2.11×10^{3}	47	8.56×10^{4}	3.50×10^{5}	1.2×10^{3}
Major composition								
Actinide								
U total	kg	11,220	11,164	—	— b)	44.9	11.22	1.12×10^{-4}
U fissile	kg	157	156.2	—	— b)	0.628	0.157	1.57×10^{-6}
Pu total	kg	148.5	146.1	—	— b)	0.578	0.147	1.49×10^{-7}
Pu fissile	kg	83.4	83	—	— b)	0.334	0.0834	8.34×10^{-8}
Np total	kg	9.35	9.48	—	— b)	3.8×10^{-2}	9.5×10^{-3}	9.5×10^{-9}
Am total	kg	18.48	18.39	—	— b)	7.4×10^{-2}	1.85×10^{-2}	1.8×10^{-8}
Cm total	kg	0.7	0.697	—	— b)	2.8×10^{-3}	7×10^{-4}	7×10^{-10}
Volatile FP's								
H-3	Ci	9.76×10^{3}	7.66×10^{1}	2.1×10^{3}		—	7.58×10^{3}	(7.66×10^{1})
C-14	Ci	1.30×10^{2}	—	8.25×10^{1}	4.66×10^{1}	—	—	4.71×10^{-1}
Kr-85	Ci	1.20×10^{5}	—	—	—	—	1.19×10^{5}	1.2×10^{3}
I-129	Ci	8.55×10^{-1}	—	—	—	—	8.55×10^{-1}	8.55×10^{-4}
I-131	Ci	—	—	—	—	—	—	—
Non-volatile FP's								
Sr-90, Y-90	Ci	2.73×10^{6}	2.7×10^{6}	—	— b)	2.73×10^{4}		
Zr-95	Ci	7.83×10^{-10}	7.74×10^{-10}	—	— b)	7.83×10^{-12}	2.18×10^{5}	2.18×10^{-2}
Nb-95	Ci	1.69×10^{-9}	1.67×10^{-9}	—	— b)	1.69×10^{-11}	(Semi volatiles)	(Semi volatiles)
Ru-103, Rh-103	Ci	1.88×10^{-20}	1.67×10^{-20}	—	— b)	1.88×10^{-22}		
Ru-106, Rh-106	Ci	1.45×10^{4}	1.29×10^{4}	—	— b)	1.45×10^{2}	5.44×10^{3}	5.44×10^{-3}
Cs-137	Ci	1.99×10^{6}	1.77×10^{6}	—	— b)	1.99×10^{4}	(Particulates)	(Particulates)

a) Activities due to impurities other than N_2 and Li are not included.
b) Calculated assuming that decomposed graphite is not contaminated with actinides and FP's.
c) Chemicals and corrosion products are not included.
d) Circled numbers correspond to streams shown in Fig.4-4.

Table A2-3(B) Material Balance of Head-end Process for Spent Fuel Annually Discharged from 3000 MW(t) VHTR after 50 Year Cooling

Composition \ Stream d)	Unit	Feed SF ①	Dissolved solution ②	Dismantled graphite ③	Decomposed graphite ④	SiC hull ⑤	Waste from off gas cleaning ⑥	Resease to environment ⑦
Actinide	kg	11,400	11,343	—a)	—b)	45.6	11.4	1.12×10^{-4}
	Ci	4.27×10^{5}	4.25×10^{5}	—a)	47 b)	1.71×10^{3}	4.27×10^{2}	4.27×10^{-4}
Fission product	kg	790	622	—a)	—b)	7.9	35	125
	Ci	2.82×10^{6}	2.70×10^{6}	(2.21×10^{2})a	—b)	1.84×10^{4}	9.15×10^{4}	9.99×10^{1}
Graphite	kg	215,170	—	147,300	67,870	—	0	0
Others	kg	3,570 (SiC, O_2)	—	—	—	1,930 (SiC)	—	1,640 (O_2)
Total mass	kg	230,930	11,965	147,300	67,870	1,984	46.4	1,760
Total radio-activities	Ci	3.23×10^{6}	3.18×10^{6}	(2.21×10^{2})a)	—	2.01×10^{4}	9.19×10^{4}	9.99×10^{1}
Major composition								
Actinide								
U total	kg	11,220	11,164	—	—b)	44.8	11.22	1.12×10^{-4}
U fissile	kg	157	156.2	—	—b)	0.628	0.157	1.57×10^{-6}
Pu total	kg	131	130.3	—	—b)	0.524	0.131	1.31×10^{-7}
Pu fissile	kg	66.4	66.1	—	—b)	0.266	0.064	6.64×10^{-8}
Np total	kg	11.01	10.95	—	—b)	4.4×10^{-2}	1.1×10^{-2}	1.1×10^{-8}
Am total	kg	33.95	33.78	—	—b)	1.4×10^{-1}	3.4×10^{-2}	3.4×10^{-8}
Cm total	kg	0.21	0.20	—	—b)	8.4×10^{-4}	2.1×10^{-4}	2.1×10^{-10}
Volatile FP's								
H-3	Ci	1.02×10^{3}	8.03	2.2×10^{2}	—	—	7.95×10^{2}	(8.03)
C-14	Ci	1.29×10^{2}	—	8.21×10^{1}	4.64×10^{1}	—	—	4.96×10^{-1}
Kr-85	Ci	9.14×10^{3}	—	—	—	—	9.05×10^{3}	9.14×10^{1}
I-129	Ci	8.55×10^{-1}	—	—	—	—	8.55×10^{-1}	8.55×10^{-4}
I-131	Ci	—	—	—	—	—	—	—
Non-volatile FP's								
Sr-90, Y-90	Ci	1.05×10^{6}	—	—	—b)	1.05×10^{4}	7.96×10^{4} (Semi volatiles)	7.96×10^{-3} (Semi volatiles)
Zr-95	Ci	—	—	—	—b)	—		
Nb-95	Ci	—	—	—	—b)	—		
Ru-103, Rh-103	Ci	—	—	—	—b)	—	2.01×10^{3} (Particulates)	2.01×10^{-3} (Particulates)
Ru-106, Rh-106	Ci	3.24×10^{-8}	2.88×10^{-8}	—	—b)	3.24×10^{-10}		
Cs-137	Ci	7.96×10^{5}	7.08×10^{5}	—	—b)	7.96×10^{3}		

a) Activities due to impurities other than N_2 and Li are not included.
b) Calculated assuming that decomposed graphite is not contaminated with actinides and FP's.
c) Chemicals and corrosion products are not included.
d) Circled numbers correspond to streams shown in Fig.4-4.

A3 Waste management

For reference, the amounts of volatile fission products such as Kr, Br, Xe and I in spent fuels were estimated as a function of cooling time, as shown in Table A3-1.

Table A3-1 Volatile Waste from VHTR Fuel Cycle

Time / Item / Element	0.5 year				10 year				50 year			
	Atom Number $(12.2T_{HM})^{-1}$	Weight kg/T_{HM}	Activity $Ci/12.2T_{HM}$	Activity Ci/T_{HM}	Atom Number $(12.2T_{HM})^{-1}$	Weight kg/T_{HM}	Activity $Ci/12.2T_{HM}$	Activity Ci/T_{HM}	Atom Number $(12.2T_{HM})^{-1}$	Weight kg/T_{HM}	Activity $Ci/12.2T_{HM}$	Activity Ci/T_{HM}
Kr-78	0.211E+14		0.0	–	0.211E+14		0.0	–	0.211E+14		0.0	–
Kr-80	0.379E+20		0.0	–	0.379E+20		0.0	–	0.379E+20		0.0	–
Kr-81	0.470E+15		0.133E-08	–	0.470E+15		0.133E-08	–	0.470E+15		0.133E-08	–
Kr-82	0.273E+23	3.04×10^{-4}	0.0	–	0.273E+23	3.04×10^{-4}	0.0	–	0.273E+23	3.04×10^{-4}	0.0	–
Kr-83m	0.406E+16		0.133E+02	9.26×10^{-1}	0.105E+04		0.295E-11	–	0.108E-47		0.303E-64	–
Kr-83	0.867E+25	9.80×10^{-2}	0.0	–	0.867E+25	9.80×10^{-2}	0.0	–	0.867E+25	9.80×10^{-2}	0.0	–
Kr-84	0.156E+26	1.78×10^{-1}	0.0	–	0.156E+26	1.78×10^{-1}	0.0	–	0.156E+26	1.78×10^{-1}	0.0	–
Kr-85	0.402E+25	4.65×10^{-2}	0.222E+06	1.82×10^{4}	0.218E+25	2.52×10^{-2}	0.120E+06	9.84×10^{3}	0.165E+24	1.91×10^{-3}	0.914E+04	7.49×10^{2}
Kr-86	0.289E+26	3.38×10^{-1}	0.0	–	0.289E+26	3.38×10^{-1}	0.0	–	0.289E+26	3.38×10^{-1}	0.0	–
		6.61×10^{-1}		1.82×10^{4}		6.40×10^{-1}		9.84×10^{3}		6.16×10^{-1}		7.49×10^{2}
Br-79	0.132E+20		0.0		0.934E+20		0.0	–	0.431E+21		0.0	–
Br-81	0.370E+25	4.08×10^{-2}	0.0		0.370E+25	4.08×10^{-2}	0.0	–	0.370E+25	4.08×10^{-2}	0.0	–
Br-82	0.429E-17		0.631E-33		0.0		0.0	–	0.0		0.0	–
		4.08×10^{-2}		~0		4.08×10^{-2}		0.0		4.08×10^{-2}		0.0
I-125	0.494E+15		0.179E-02	–	0.157E-02		0.571E-20	–	0.0		0.0	–
I-126	0.313E+14		0.521E-03	–	0.0		0.0	–	0.0		0.0	–
I-127	0.626E+25	1.08×10^{-1}	0.0	–	0.635E+25	1.10×10^{-1}	0.0	–	0.635E+25	1.10×10^{-1}	0.0	–
I-129	0.230E+26	4.04×10^{-1}	0.855E+00	7.01×10^{-2}	0.230E+26	4.04×10^{-1}	0.855E-00	7.01×10^{-2}	0.230E+26	4.04×10^{-1}	0.855E+00	7.01×10^{-2}
I-131	0.159E+18		0.427E+01	3.50×10^{-1}	0.0		0.0	–	0.0		0.0	–
I-132	0.213E+06		0.485E-09	–	0.0		0.0	–	0.0		0.0	–
I-133	0.954E-40		0.239E-55	–	0.0		0.0	–	0.0		0.0	–
		5.12×10^{-1}		4.20×10^{-1}		5.14×10^{-1}		7.01×10^{-2}		5.14×10^{-1}		7.01×10^{-2}
Xe-126	0.107E+20		0.0	–	0.107E+20		0.0	–	0.107E+20		0.0	–
Xe-127	0.587E+12		0.350E-05	–	0.120E-16		0.715E-34	–	0.0		0.0	–
Xe-128	0.302E+24	5.26×10^{-3}	0.0	–	0.302E+24	5.26×10^{-3}	0.0	–	0.302E+24	5.26×10^{-3}	0.0	–
Xe-129m	0.172E+09		0.465E-08	–	0.131E-08		0.830E-26	–	0.0		0.0	–
Xe-129	0.162E+19		0.0	–	0.111E+20		0.0	–	0.510E+20		0.0	–
Xe-130	0.894E+24	1.58×10^{-2}	0.0	–	0.894E+24	1.58×10^{-2}	0.0	–	0.894E+24	1.58×10^{-2}	0.0	–
Xe-131m	0.267E-21		0.463E-37	–	0.0		0.0	–	0.0		0.0	–
Xe-131	0.473E+26	8.44×10^{-1}	0.0	–	0.473E+26	8.44×10^{-1}	0.0	–	0.473E+26	8.44×10^{-1}	0.0	–
Xe-132	0.112E+27	2.01×10^{0}	0.0	–	0.112E+27	2.01×10^{0}	0.0	–	0.112E+27	2.01×10^{0}	0.0	–
Xe-133m	0.134E-01		0.128E-17	–	0.0		0.0	–	0.0		0.0	–
Xe-133	0.635E+14		0.261E-02	–	0.0		0.0	–	0.0		0.0	–
Xe-134	0.150E+27	2.74×10^{0}	0.0	–	0.150E+27	2.74×10^{0}	0.0	–	0.150E+27	2.74×10^{0}	0.0	–
Xe-136	0.245E+27	4.54×10^{0}	0.0	–	0.245E+27	4.54×10^{0}	0.0	–	0.245E+27	4.54×10^{0}	0.0	–
		1.01×10^{1}		~0		1.01×10^{1}		0.0		1.01×10^{1}		0.0
Total		1.13×10^{1}		1.82×10^{4}		1.13×10^{1}		9.84×10^{3}		1.13×10		7.49×10^{2}