THE IMPACT OF FUEL CYCLE OPTIONS ON THE SPACE REQUIREMENTS OF A HLW REPOSITORY

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Because of increasing concerns regarding global warming and the longevity of oil and gas reserves, the importance of nuclear energy as a major source of sustainable energy is gaining recognition worldwide. To make nuclear energy truly sustainable, it is necessary to ensure not only the sustainability of the fuel supply but also the sustained availability of waste repositories, especially those for high-level radioactive waste (HLW). From this perspective, the effort to maximize the waste loading density in a given repository is important for easing repository capacity problems. In most cases, the loading of a repository is controlled by the decay heat of the emplaced waste. In this paper, a comparison of the decay heat characteristics of HLW is made among the various fuel cycle options. It is suggested that, for a future fast breeder reactor (FBR) cycle, the removal and burning of minor actinides (MA) would significantly reduce the heat load in waste and would allow for a reduction of repository size by half.

KEYWORDS : Repository, HLW, Fuel Cycle Options, FBR Cycle, Decay Heat, Minor Actinides, Sustainability

1. THE HLW REPOSITORY IS A VERY PRECIOUS NATIONAL ASSET

The importance of nuclear power as a major source of sustainable energy is gaining recognition worldwide, because of increasing concerns about the longevity of oil and gas reserves and, especially, concerns about global climate change due to carbon dioxide emissions resulting from the use of fossil fuels. To make nuclear energy truly sustainable, it is necessary to ensure not only the sustainability of the fuel supply but also the sustained availability of waste repositories, especially those for high-level radioactive waste (HLW). Due to political sensitivities and the complex social implications involved, it is unlikely that any country can afford to build many HLW repositories over the next century. From this perspective, a HLW repository will be a very precious asset for any society. Therefore, it seems that the effort to maximize the utilization efficiency of a given repository is mandatory in countries where large-scale nuclear power programs exist or are being launched.

2. COUNTRIES WITH LARGE-SCALE NUCLEAR POWER PROGRAMS TEND TO CHOOSE THE CLOSED FUEL CYCLE OPTION

In February 2006, the United States Department of Energy (USDOE) expressed its intention to explore a closed fuel cycle option during the announcement of the Global Nuclear Energy Partnership (GNEP) program.[1] In the USA, the closed fuel cycle option was abandoned in 1977 by the Carter administration to avoid proliferation risks associated with civilian utilization of plutonium. Since then, the once-through (or direct disposal) policy has been firmly maintained for almost three decades. In July 2002, the Yucca Mountain site was approved as the first US HLW repository, but its capacity is limited to the spent fuel arising to around 2010. Under rising expectations of a "Nuclear Renaissance", concerns about the HLW disposal problem have grown and become widely recognized. It is predicted that, if the USA were to continue the direct disposal policy, 4 to 20 repositories on the Yucca Mountain scale would be required by the end of this century, depending on how much the installed generating capacity increases.[2] In order to overcome this problem, the USA has begun to reconsider spent fuel reprocessing as an inevitable means of bringing about a significant improvement of the waste loading density at the Yucca Mountain site by reducing both the volume and heat load of the waste.

Table 1 summarizes the fuel cycle policies chosen by the countries with large-scale nuclear power programs. The five countries in the upper rows are those whose current installed capacity exceeds 20 GWe. The three countries in the lower rows are those whose capacity is

Table 1. Fuel Cycle Policies Chosen by Countries with Large-scale Nuclear Power Programs (>20 GWe)

Country	N. of NPP	Installed Capacity(GWe)	Fuel Cycle Policy	Fast Reactor Program
USA	103	99.2	Reconsidering Recycle (GNEP)	ABR development (GNEP)
France	58	63.4	Closed cycle	Phenix in operation GEN-IV prototype by 2020
Japan	55	47.8	Closed cycle	Monju prepairing for restart FaCT started (prototype by 2025)
Russia	31	21.7	Closed cycle	BN600 in operation BN800 under construction (-2012)
Germany	17	20.3	Nucl. phase-out	-
Korea	20	16.8 26.1 (2015)	Wait & see	Design study in progress (KALIMER)
China	9	6.6 40 (2020)	Closed cycle	CEFR under construction (-2008) Design of CPFR in progress
India	15	3.0 20.9 (2020)	Closed cycle	PFBR under construction (-2010) 4 commercial FBRs by 2020

Installed capacity as of end of 2005

expected to reach or exceed 20 GWe by 2020. The latter are all Asian countries. It is indicated that most of these countries are inclined to choose the closed fuel cycle policy. Easing HLW repository capacity problems is becoming an important factor driving these countries to choose a closed fuel cycle, as exemplified by changes in the US policy. It is also worthwhile to note that these countries are also engaged in various levels and approaches in fast reactor development that will eventually lead to complete recycling. Although the number of countries engaged in the closed fuel cycle is small, many of them have relatively large-scale nuclear power programs. As a result, nearly 45 % of the world's nuclear power generation is attributed to these countries. If the USA actually returns to the closed fuel cycle, this share will reach or could even exceed 80%.

3. HEAT SOURCES IN HLW AND THE REPOSITORY CONCEPT

There are three major design parameters that govern the optimization of the waste loading density in a repository: the physical dimensions of the waste packages, the mechanical strength of the host rock and the decay heat. However, except for a case in which the mechanical strength of the host rock is extremely low, the loading density is essentially determined by the thermal design. Therefore, the decay heat at the time of disposal is of primary interest in the consideration of the waste loading density. The decay heat of HLW varies depending on factors such as the fuel cycle options (direct disposal or reprocessing), the fuel type (UO_2 or mixed oxide), the burnup level, the reprocessing scheme, and the duration of surface storage.

Major heat-generating radionuclides in the waste are listed in Table 2. Among the fission products (FP), Cs-137 and Sr-90 are the dominant heat sources for a period lasting from several years to hundreds of years after spent fuel discharge. The decay heat from Pu-240 and Pu-238 increases as the fuel burnup increases, and it

Table 2. Major Heat Sources and their Half-Lives

Fission Products	
	Sr-90 (28.8 y)
	Cs-137 (30 y)
Actinides	
	Pu-238 (87.7 y)
	Pu-240 (6560 y)
	Am-241 (432 y) ← Pu-241 (14.4 y)
	Cm-244 (18.1 y)

constitutes an important fraction of the heat source and affects the thermal design of the repository for direct disposal. It should be noted that Pu-238 is one of the most intense heat sources and is used for thermo-electric generators. Am-241 is another important heat source. It is produced by the decay of Pu-241 with a half-life of 14.4 years. In the design of a repository, Am-241, with a half-life of 443 years, requires special attention because once it accumulates in waste, its decay heat does not readily dissipate in any practically conceivable timeframe for storage. Cm-244 is also an important heat source in the waste, but its relatively short half-life mitigates thermal problems in geological disposal.

The fundamental concept of HLW disposal in Japan is depicted in Fig. 1.[3] The repository is to be placed far below the water table to take the advantage of a reducing environment at these depths. The main features of this concept are common to those countries involved in HLW disposal programs, except for the USA where the Yucca Mountain repository is placed above the water table in a region with very low annual precipitation. As Japan adopts the closed fuel cycle policy, HLW represents vitrified waste that is generated by reprocessing. Vitrified waste canisters encased in carbon-steel overpacks are to be placed in cavities excavated in a suitable host rock. The overpack is an engineered barrier which is designed to prevent contact between the vitrified waste and groundwater for at least 1,000 years. The space between the overpack and the host rock is filled with a buffer material, which is a mixture of bentonite and sand. The buffer material has the important role of limiting groundwater penetration to the overpack, thus limiting corrosion and extending the lifetime of the overpack. It also retards migration of any radionuclides released due to gradual dissolution of the vitrified waste. To prevent the degradation of the buffer's function due to mineral alteration at elevated temperatures, the maximum temperature of the buffer material should remain below a certain value. In the present Japanese design, this temperature limit is set at 100 °C. For the direct disposal case, a vitrified waste canister, as shown in Fig.1, is replaced with a spent fuel canister, allowing for some dimensional adjustments while maintaining other features.

4. LARGER DECAY HEAT FORCES DIRECT DISPOSAL TO USE A DISPOSAL SPACE THAT IS SEVERAL TIMES LARGER

A comparison of the decay heat of the HLW normalized to the unit ton of uranium was made between the direct disposal and reprocessing cases in Figs. 2 and 3. In the Japanese design, the waste is assumed to be emplaced after cooling for 30 to 50 years. Due to the additional heat from plutonium and americium, the decay heat in the direct disposal case is approximately 60 % higher than that in the reprocessing case with similar cooling times of approximately 50 years.

A preliminary calculation was made on buffer material temperatures for the direct disposal case based on the disposal concept established in the H12 report[3], which focused on the disposal of vitrified HLW generated from reprocessing. The configuration of spent fuel packages for use in temperature calculations was borrowed from the SKB design because a waste package for direct disposal has not been designed in Japan, where the closed fuel cycle policy is employed. Temperature calculations were made for the following three cases:

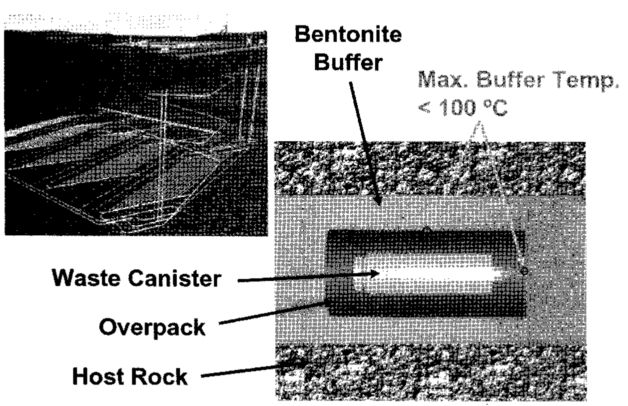

Fig. 1. Concept for HLW Geological Disposal in Japan

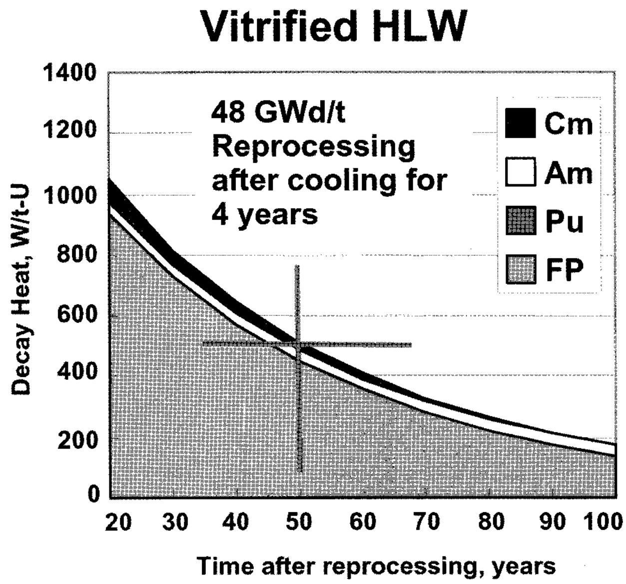

Fig. 2. Decay Heat Characteristics of Vitrified HLW

Case 1 : Allocated disposal area normalized to the unit ton of uranium (A_d) is identical to that of the reference case for vitrified HLW disposal (A_0), i.e., $A_d = A_0$.
Case 2 : $A_d = 2 \times A_0$.
Case 3 : $A_d = 4 \times A_0$.

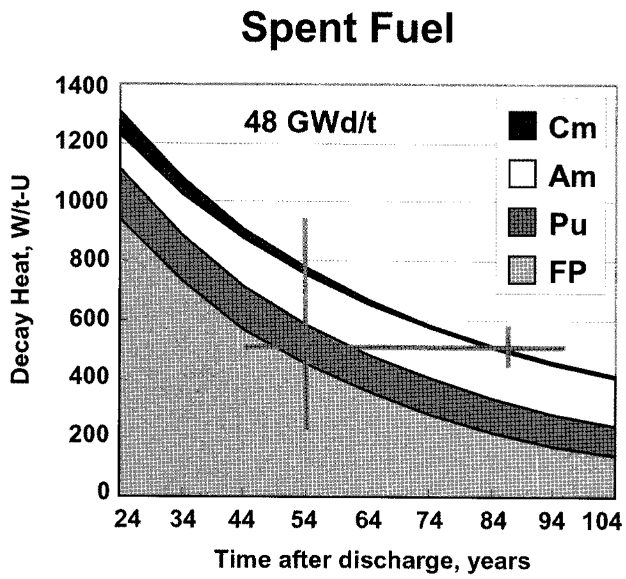

Fig. 3. Decay Heat Characteristics of Spent Fuel

The variation of the calculated maximum temperatures in the buffer material with time is shown in Fig. 4. In Fig. 5, the peak values are plotted against the allocated disposal area. This figure indicates that, in order to hold the peak temperature of the buffer material below 100 °C, the direct disposal option requires a disposal area that is nearly four times larger than that for the reprocessing option. Similar observations regarding the disposal space requirements for the two options have been made elsewhere in the world. These are summarized in Table 3.[4,5,6] This table suggests that the reprocessing option allows for several times denser waste package loading in a repository compared to the direct disposal option.

Considering that many variations exist in repository design which involves numerous different decision-making factors, the loading density may be, in some cases, a second or third priority. This is especially true if a country has ample available land for repositories or if nuclear power generation in a country is either limited in scale or scheduled to be phased out in the near future. However, this analysis implies that, at least for countries with large-scale nuclear power programs, the direct

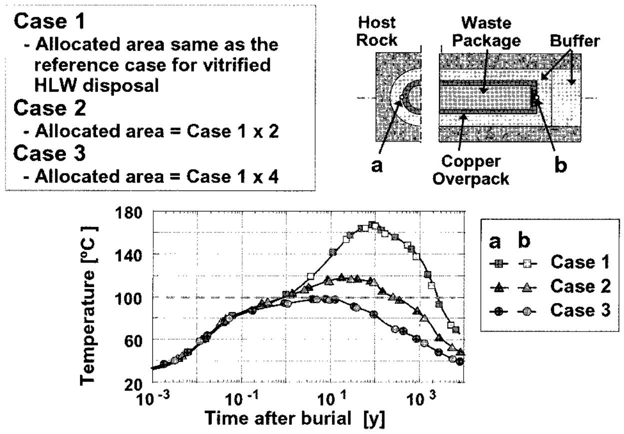

Fig. 4. Variation of the Maximum Temperatures in Buffer Material Calculated for the Direct Disposal Case Based on H12 Conditions

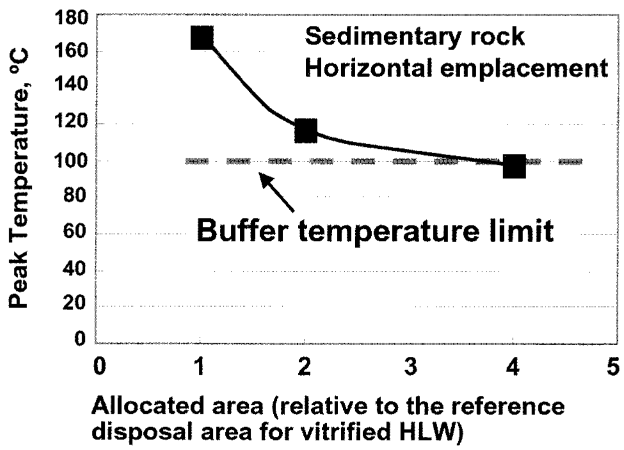

Fig. 5. Peak Temperatures of Buffer Material Evaluated for the Direct Disposal Cases with Variable Disposal Area

Table 3. Improvement in Waste Loading Density for the Reprocessing Case Versus the Direct Disposal Case

Study	Improvement factor for waste loading density
JAEC study (Japan)	1.5 - 2.3 (a)
JAEA study (Japan)	4
ANDRA Dossier 2005 (France)	3.25 (b)
ONDRAF SAFIR-2 (Belgium)	6
ANL/AFCI (USA)	5.7 (c)

(a) Constraints from other factors than thermal design are involved
(b) Disposal of MOX or HLW from MOX reprocessing is included
(c) Case for 99.9% removal of U, Pu, Am and Cm

disposal option would not be a favorable option in ensuring the sustained availability of HLW repositories.

5. HOW WILL HIGH DECAY HEAT FROM MA ACCUMULATION IN LWR-MOX AFFECT REPOSITORY SPACE REQUIREMENTS?

In the current light-water reactor (LWR) fuel cycle, the reprocessing option is usually combined with the use of plutonium in the form of mixed-oxide (MOX) fuel. That the decay heat of spent MOX fuel is significantly higher than that of standard (uranium) spent fuel with the same burnup owing to the enhanced accumulation of higher plutonium isotopes and minor actinides (MAs). Consequently, when disposing of spent MOX fuel directly, the number of fuel assemblies to be loaded in a waste package must be reduced so that the total heat load per package is maintained at the same level as the waste package for standard fuel. For instance, in the ANDRA design, only one spent fuel assembly is loadable in a package for MOX fuel, whereas four are loadable for standard UO_2 fuel.[5] Therefore, spent MOX fuel requires much larger disposal space than standard fuel in the same tonnage.

Likewise, HLW generated by reprocessing spent MOX fuel has significantly higher decay heat than standard HLW if it is vitrified separately from the latter as indicated in a comparison between Figs. 6 and 7. Here, in these and the following figures, the decay heat is normalized to units of electricity generation for a fair comparison between the cases with different fuel burnup rates. The dashed curve with a cross on the figures represents the decay heat of standard HLW for ease of comparison. If reprocessing is delayed for spent MOX fuel, the increase of the decay heat is remarkable owing to increased accumulation of Am-241, as shown in Fig. 8.

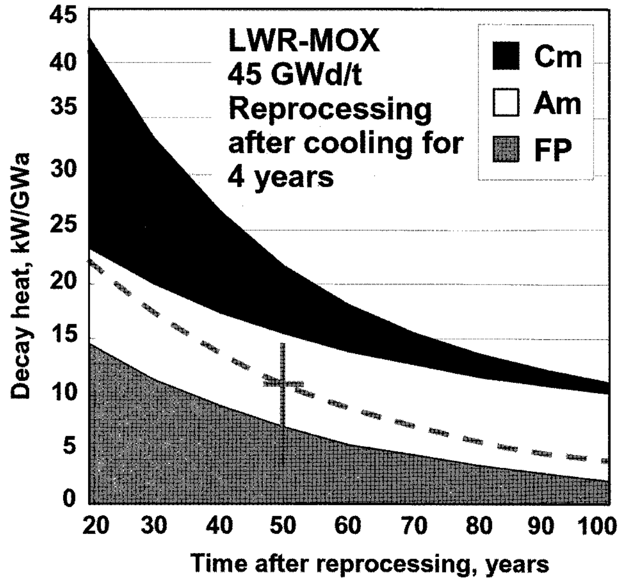

Fig. 7. Decay Heat of Vitrified HLW from the Early Reprocessing of LWR-MOX

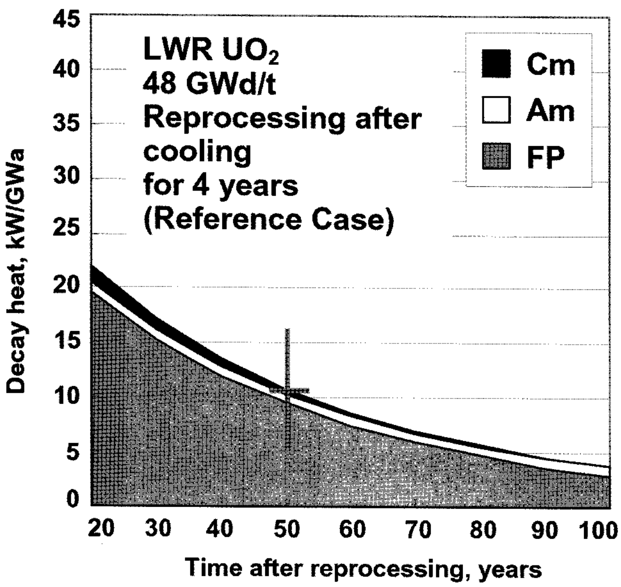

Fig. 6. Decay Heat of Standard Vitrified HLW from LWR Fuel Reprocessing

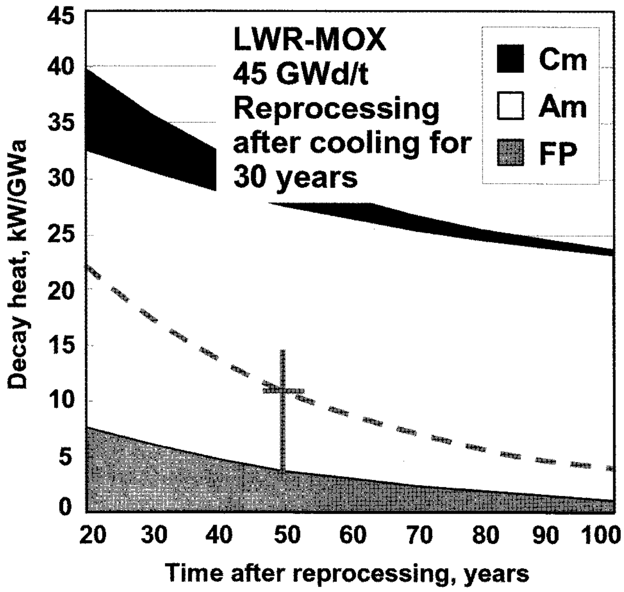

Fig. 8. Decay Heat of Vitrified HLW from the Delayed Reprocessing of LWR-MOX

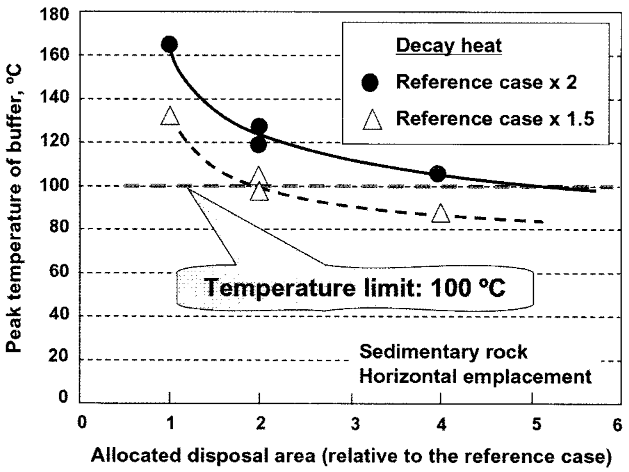

Fig. 9. Peak Temperatures of Buffer Material Evaluated for Variable Decay Heat of HLW and Disposal Area

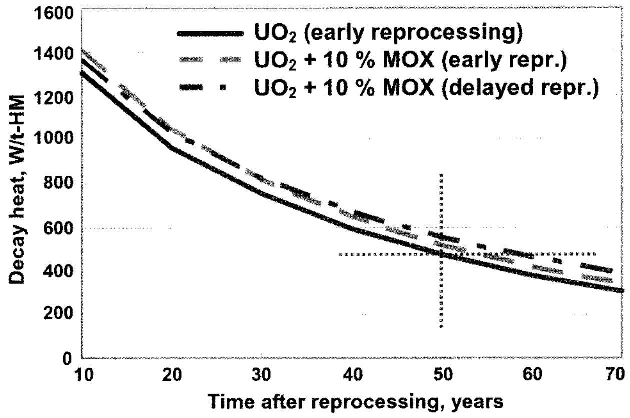

Fig. 10. Decay Heat of the "Mixed Waste"

In Fig. 9, peak temperatures of the buffer material are evaluated with changes in the decay heat of the waste and the allocated disposal space. When the decay heat is increased by 50 %, doubling of the disposal space is necessary to hold the buffer temperature below 100 °C. For a 100% increase in the decay heat, the disposal space must be five times larger. This analysis suggests that the high decay heat of HLW originating from MOX reprocessing would have a very large impact on the repository space requirements. However, this is not necessarily true in a real-case scenario. In the actual reprocessing and MOX recycling scheme in the LWR cycle, the share of MOX fuel in the entire fuel supply is usually limited to below 15 % due to limitations imposed by the plutonium balance. This represents a viable opportunity for high-level liquid waste (HLLW) from MOX reprocessing to be mixed in a storage tank with much larger amounts of HLLW from uranium fuel reprocessing. The decay heat of "mixed waste" containing 10 % MOX reprocessing waste is compared with that of standard HLW in Fig. 10. As shown in the figure, the increase in the decay heat of mixed waste is insignificant even for the delayed reprocessing. Thus, the necessity of a larger disposal space appears to be avoidable when simply extending the cooling period for ten years before burial. A larger disposal space would inevitably be required if, for some reason, HLW from MOX reprocessing were to be vitrified separately from or in higher proportions in a mixture with standard HLW.

6. IN FUTURE FBR CYCLE WITH MA RECOVERY AND BURNING CAPABILITIES, THE REPOSITORY SPACE WILL BE REDUCED BY HALF.

A simplified material balance for the LWR once-through cycle and the future fast breeder reactor (FBR) cycle is schematically shown in Fig. 11. In terms of both maximizing utilization efficiency of uranium resources and minimizing waste, the FBR cycle is superior to the LWR once-through cycle and is ideal as a next-generation option to realize truly sustainable nuclear energy.

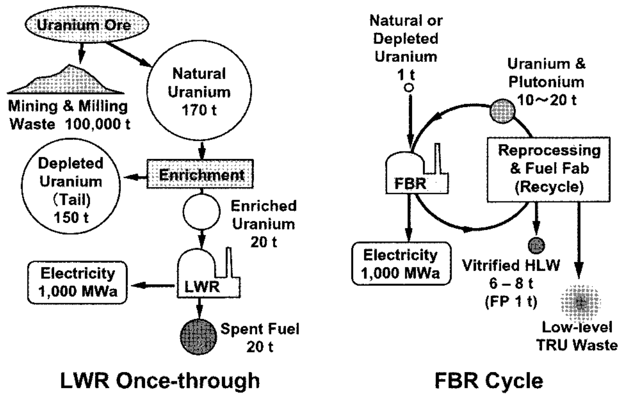

Fig. 11. Comparison of Material Balance Between the LWR Once-Through and the FBR Cycle

As both the plutonium content and the burnup rate in FBR fuel are higher compared to those in LWR MOX fuel, close attention should be paid to the accumulation of MAs from a waste management perspective. In a FBR system, there are two types of fuel, core fuel and blanket fuel. The former is needed to produce energy by fission and the latter to produce plutonium for use as new fuel by transmuting non-fissile U-238. As weapons-grade plutonium is produced in blanket fuel, reprocessing of blanket fuel alone is not desirable from a non-proliferation perspective. Therefore, it is reasonable to assume a reprocessing scheme in which both core and blanket fuels are mixed prior to dissolution to avoid the production of weapons-grade plutonium. Figure12 shows the decay heat of typical HLW from the FBR cycle in which the mixture of the core and blanket fuels is processed by conventional reprocessing method without the capability to recover MAs. It is interesting to note that, when normalized to units of electricity generation, the decay heat of HLW from the FBR cycle with 50-year cooling is nearly equal to that of standard HLW from the LWR cycle. However, the decay heat from FP in HLW from the FBR cycle is significantly lower than that in standard HLW from the LWR cycle. This is partly because the FBR system has higher thermal efficiency compared to the LWR system and partly because the yield of Sr-90 during Pu-239 fission is smaller than that during U-235 fission. On the other hand, an increase in the decay heat from MAs is clearly observed because the FBR system uses fuel with a high plutonium content. However, this increase is noticeably lower when compared with HLW originating from LWR-MOX reprocessing, owing to the superior MA burning capability in the FBR system. In any event, Fig. 12 suggests that, despite the high plutonium content in FBR fuel, the space requirement for HLW disposal per unit of electricity generation for the FBR cycle would be nearly equivalent to that for standard HLW from the LWR cycle even without removal of MAs.

Since the contribution from MAs to total decay heat is relatively large in HLW from the FBR cycle, removal of MAs will directly ease the thermal design of the repository and thus improve the waste loading density. Decay heat is shown in Fig. 13 for a case in which 90 % of MAs are removed from HLW. Recently, more detailed analyses have been conducted on fuel cycle options and their effects on repository space requirements. Preliminary results indicate that the FBR cycle with MA recovery and burning will improve the waste loading density by a factor of up to 2.5 in comparison with the case of standard HLW from the LWR cycle. Therefore, the introduction of a MA recovery and burning scheme in a future FBR cycle is truly beneficial for ensuring the sustained availability of a HLW repository.

It should be noted that MA removal does not help reduce disposal space requirements at all for standard HLW from the LWR cycle, as is evident from Fig. 6. On the other hand, MA removal is beneficial for HLW from LWR-MOX reprocessing if there is an option to burn the separated MAs effectively. As discussed in many earlier papers, the LWR system itself is not an efficient tool to burn MAs. Therefore, the benefit of MA removal will be realized only when MA burning in the FBR system becomes feasible.

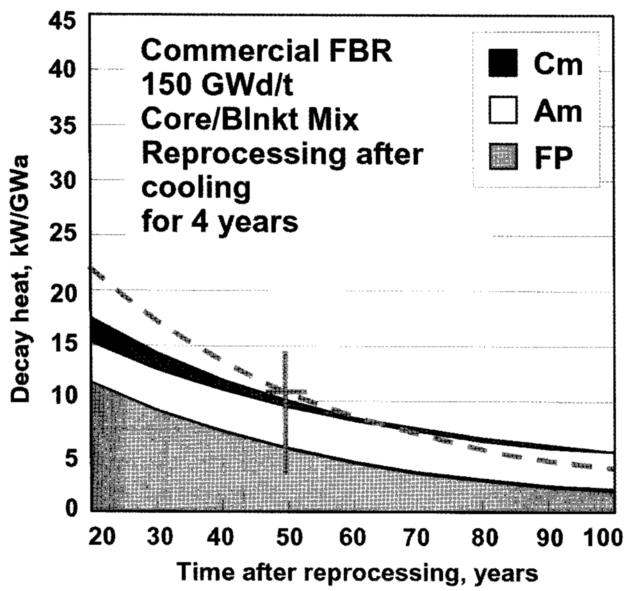

Fig. 12. Decay Heat of HLW from the FBR Cycle Without MA Removal

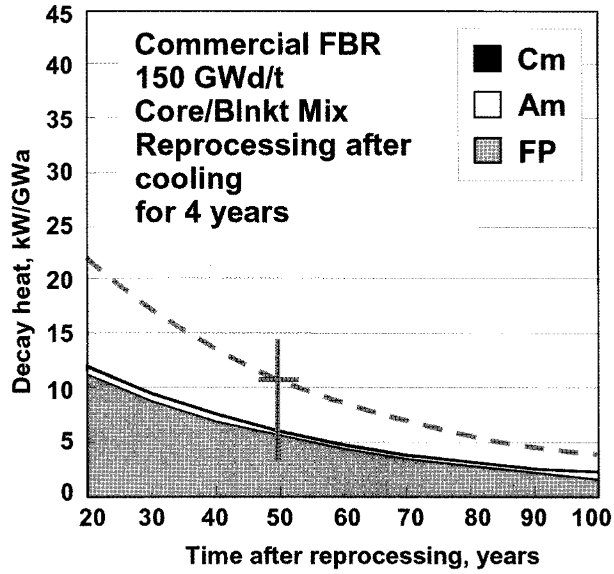

Fig. 13. Decay Heat of HLW from the FBR Cycle with MA Removal

7. TRUE BENEFIT OF MA RECOVERY AND BURNING

Thus far, the benefit of MA recovery and burning in waste management has been discussed in various papers mostly in terms of reducing potential radiotoxicity in

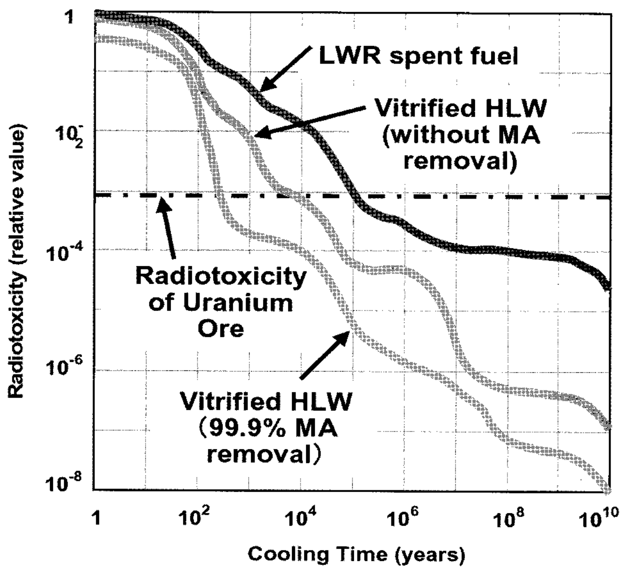

Fig. 14. Reduction of Radiotoxicity by MA Removal

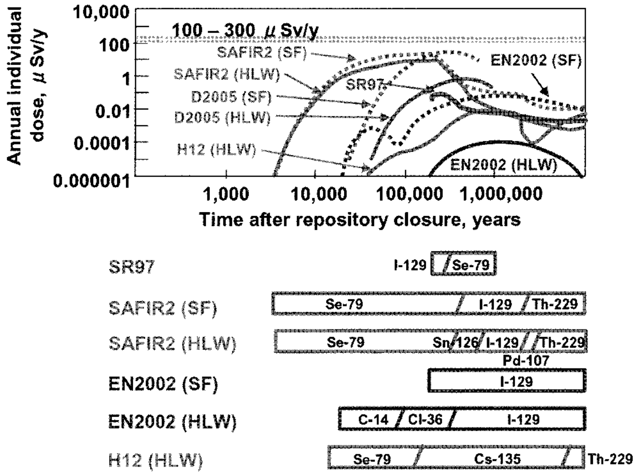

Fig. 15. Results of Safety Assessments in Various Countries and Dominant Nuclides

HLW, as typically illustrated in Fig. 14. However, potential radiotoxicity is by no means a safety indicator in the safety assessments for licensing. Rather, it is considered an aid to easing public acceptance. In nearly all countries with a HLW geological disposal program, the parameter of an individual dose to a critical group is used as a safety indicator. Results of safety assessments conducted in major countries are compared in Fig.5. In the lower part of the figure, nuclides that dominate the calculated dose are indicated along with the time range during which they dominate. As shown in this figure, MAs are not main contributors to dose calculations primarily because they are usually in the form of oxides with very low solubility and mobility. Therefore, from a technical standpoint, MA removal for the purpose of reducing radiotoxicity has little merit, and whether or not to do so is essentially a matter of political judgment. On the other hand, MA recovery and burning for the purpose of reducing the heat load in waste has a true benefit in terms of improving the waste loading density and thus extending the lifetime of a given repository. MA removal as high as 99.9 % is required to achieve a meaningful radiotoxicity reduction; while a 90 % removal factor is sufficient for heat reduction. From an industrialization perspective, attaining 90 % MA removal is far more realistic than 99.9 % MA removal.

The development of the FBR cycle system is indispensable in pursuing sustainable nuclear energy for our future generations. The capability to recover and burn MAs is one of the most important factors that must be realized through R&D because such capability is a key technology to maximize the utilization efficiency of the HLW repository.

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