ASSESSMENT OF THE EQUILIBRIUM STATE IN REACTOR-BASED PLUTONIUM OR TRANSURANICS MULTI-RECYCLING

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Abstract

An assessment of the radiotoxicity of spent nuclear material and the fuel handling issues associated with plutonium (Pu) or transuranics (TRU) multi-recycling in PWRs using the CORAIL assembly concept has been performed. This necessitated the development of an analytical approach to solve directly for the equilibrium state of repeated recycle, including higher actinide content. The equilibrium states used in the assessment are determined by a one-group transmutation code with cross sections prepared by the WIMS8 code (for transuranics below ^{246}Cm) and ENDF/B-V data (for ^{246}Cm and above). The fuel handling indices suggest that Pu multi-recycling is feasible. However, the high radiotoxicity of the spent assembly content suggests that Pu multi-recycling would provide only minimal benefits to the repository if the goal is to reduce the radiotoxicity of the disposed waste to less than that of the source uranium used in producing the fuel in a thousand years. TRU multi-recycling is the most beneficial to the repository because the amount of TRU in the disposed waste is significantly reduced by keeping the TRU in the PWR fuel cycle. However, TRU multi-recycling in the CORAIL or alternative assembly designs will complicate fuel handling at the separation, fabrication, and core loading stages, due to the high spontaneous fission neutron emission rates of the higher actinides.

Introduction

If nuclear power continues to maintain its share (~20%) of the U.S. electric power generation capacity, the projected repository inventory will continue to grow, making the need for additional repositories a possibility. Given the particular problems in setting up the first repository, the USDOE Office of Nuclear Energy, Science and Technology Advanced Fuel cycle Initiatives Program is assessing approaches for increasing the effective repository capacity. Partitioning and the potential transmutation of plutonium and minor actinides in spent nuclear fuels (P&T) is an approach that is being assessed. If the P&T mission could be completely done in existing or evolutionary PWR fuel cycles, this would provide some cost benefits due to the existing technologies and infrastructure.

For these reasons, PWR fuel cycles that could cap or minimise the growth of TRU are being assessed. One such concept is the CORAIL assembly concept that was originally proposed by the French CEA [1] and has also been investigated by the USDOE. [2-4] The intent of this concept is to stabilise the plutonium (Pu) or transuranics (TRU) in the PWR fuel cycle by mult-recycling. By stabilisation, it is implied that there is no net difference in the amount of the material over a recycle stage. Therefore, the amount of material passed to the repository is limited to that lost in fuel processing between irradiation cycles.

In the CORAIL concept, Pu and TRU would be multi-recycled in an innovative fuel assembly in existing PWR core designs without adversely affecting core safety and operational parameters or fuel cycle infrastructures. The detailed characteristics of Pu or TRU multi-recycling in the assembly are described in Refs. 2-4. The mass balances and radioactive properties of the assemblies were previously evaluated for the seventh recycle stage using a coupled WIMS8-ORIGEN2 procedure. [3,4] The resulting positive mass balance of the concept after seven recycles indicated the need for many recycle stages before an equilibrium state could be reached. Since the evaluation of waste radiotoxicity and fuel handling issues are required for comparative analysis of concepts, an equilibrium state solution was deemed necessary.

A one-group transmutation code, called *TRANSEQM*, has been developed to search for the equilibrium recycle state. The code is designed to solve the transmutation equations of actinides in a heterogeneous assembly, i.e., multiple zones with different fuel compositions are allowed. Although the WIMS8 [5] code terminates the actinide transmutation chain at ^{245}Cm(based on the judgement that the higher actinides do not contribute significantly to the reactivity balance in typical LWR designs), TRANSEQM uses transmutation chains with actinides up to ^{253}Es. It is anticipated that in multi-recycled fuel, the higher actinides could build up and contribute significantly to the neutron source.

In this paper, a brief discussion of the CORAIL and alternative concepts are discussed. The formulation and solution scheme of the TRANSEQM code are discussed. The code has been used to evaluate the mass flow and radioactive properties of the CORAIL multi-recycling cases, and the results from these evaluations are summarised in this paper.

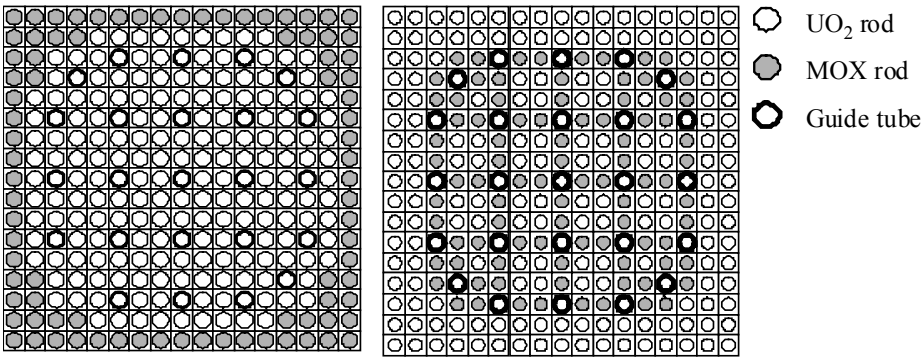
CORAIL and alternative concepts

In the CORAIL concept, a full-core loading of the heterogeneous assembly is assumed. Figure 1 shows the fuel pin configuration in the CORAIL assembly, which employs a standard 17x17 PWR fuel assembly containing 180 UO_2 pins in the interior and 84 MOX pins in the peripheral region. In the CORAIL multi-recycling scheme, the MOX pins are fabricated from Pu-only or TRU extracted from the discharge of the previous recycle stage. A lead-time of two years is assumed from the assembly

fabrication to its loading into the reactor. After the assembly is discharged from the reactor, a five-year post-irradiation cooling time is allowed before separation of the discharged fuel. During the separation, most of the TRU (or Pu only) is recovered, while all fission products and 0.1% TRU (or 0.1% Pu and all minor actinides) are discharged as waste; uranium is cleanly partitioned. It is assumed that the separated uranium will either be used as make-up feed or be passed to low-level storage instead of the repository.

In an effort to lower the TRU content in the assembly at equilibrium recycle, and to reduce the localised power peaking factor, alternative assembly designs have also been considered. Such redesigned assemblies are the *highly-moderated and modified CORAIL-TRU assembly* (simply, HM-TRU; see Figure 1) and the *homogeneous Thorium-TRU assembly* (simply, TMOX). The primary objective of these alternative assembly designs is to reduce the power peaking factor and the conversion of ^{238}U to ^{239}Pu. The HM-TRU assembly design has 88 TRU-containing MOX fuel pins near the guide tubes and a higher moderator to fuel volume ratio (2.47, compared with 2.03 for the CORAIL design). The thorium-containing MOX fuel has also been considered for the transmutation mission. In this case, the uranium fuel in a homogeneous UO_2 assembly is replaced with a blend of uranium, thorium and TRU. The detailed design characteristics and mass balances of these TRU-containing assemblies are reported in Ref. 4.

Figure 1. **CORAIL (left) and highly moderated, modified CORAIL (right) assemblies**



One group transmutation code, TRANSEQM

The one group transmutation code, called TRANSEQM, solves a homogeneous, first-order ordinary, transmutation equation,

$$\frac{d\overline{N}}{dt} = \mathbf{A}\overline{N}\,, \tag{1}$$

where $\overline{N}$ is an M-dimensional vector of nuclide densities and $\mathbf{A}$ is a (M×M) transmutation matrix containing the cross sections, decay constants and yield fractions. The solution of Eq. (1) is known with the initial nuclide densities, $\overline{N}(0)$;

$$\overline{N}(t) = \left(\sum_{m=0}^{\infty} \frac{(\mathbf{A}\cdot\Delta t)^m}{m!} \right)^K \cdot \overline{N}(0)\,, \tag{2}$$

where K denotes the number of time intervals of the total decay time or irradiation time (i.e., $\Delta t = t/K$).

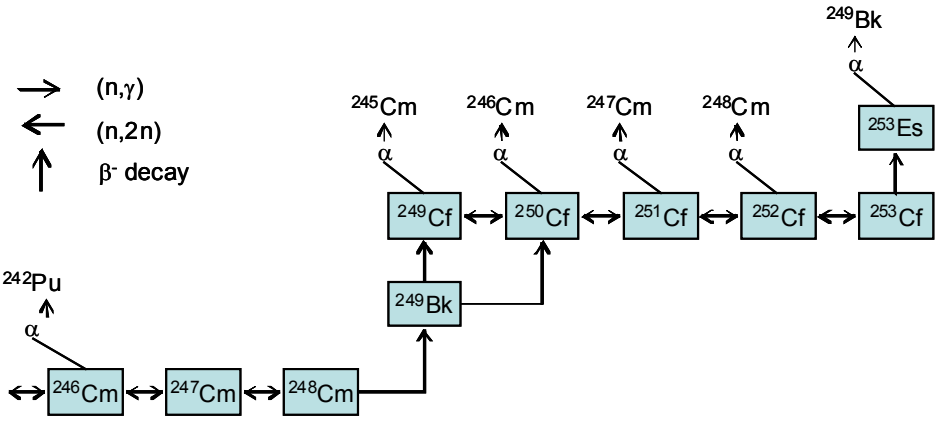
In performing the summation indicated by Eq (2), the accuracy of the solution can be maintained at a desired value by controlling the time step size such that the norm of matrix ($\|\mathbf{A}\|\cdot\Delta t$) is less than a predetermined value. [6] In this work, the norm of matrix **A** is defined by

$$\|\mathbf{A}\| = \min\left\{\max_j \sum_i |a_{ij}|, \max_i \sum_j |a_{ij}|,\right\}, \tag{3}$$

and the value of $\|\mathbf{A}\|\cdot\Delta t$ is restricted to be less than $\|\mathbf{A}\|\cdot\Delta t \le 0.5$. This limits the solution error from summing the first 5 terms of the series indicated by Eq. (2) to < 0.1%. If $\|\mathbf{A}\|\cdot\Delta t$ is greater than the limiting value, it is reduced by repeatedly halving the time-step size. Typical time-step size is about 24 hours.

The transmutation chain for actinides below ^{246}Cm is similar to that of the WIMS8 code, which is used for generating the neutron cross sections for these nuclides. That chain extends from ^{232}Th to ^{245}Cm. Figure 2 shows the transmutation chain of the actinides higher than ^{245}Cm (simply, higher actinides). This chain is introduced to evaluate the build-up of the higher actinides with TRU multi-recycling.

Figure 2. **Transmutation chain for minor actinides beyond ^{245}Cm**



The accuracy of the TRANSEQM code was verified by comparing the code results with those of the ORIGEN-RA code. [7] The comparison was done using the same initial fuel composition and one-group cross sections. The fuel was irradiated to 45 000 MWd/t and additionally cooled for 5 years after discharge. Generally, the results of the TRANSEQM calculation agree well with the results of the ORIGEN-RA calculations; the relative differences are less than 0.1%.

The cross sections of the actinides lower than ^{246}Cm are provided by WIMS8 calculations. In the WIMS8 code, a 172-group neutron cross-section library based on JEF2.2 is available to properly account for the self-shielding of the thermal and epithermal energy resonances of the minor actinides. The method of characteristic option (CACTUS) of the WIMS8 code was used for solving the multi-group neutron transport equations for the CORAIL assembly. The fluxes from this solution are used to collapse the 172-group data to one-group cross sections for the UO_2 and MOX fuel regions.

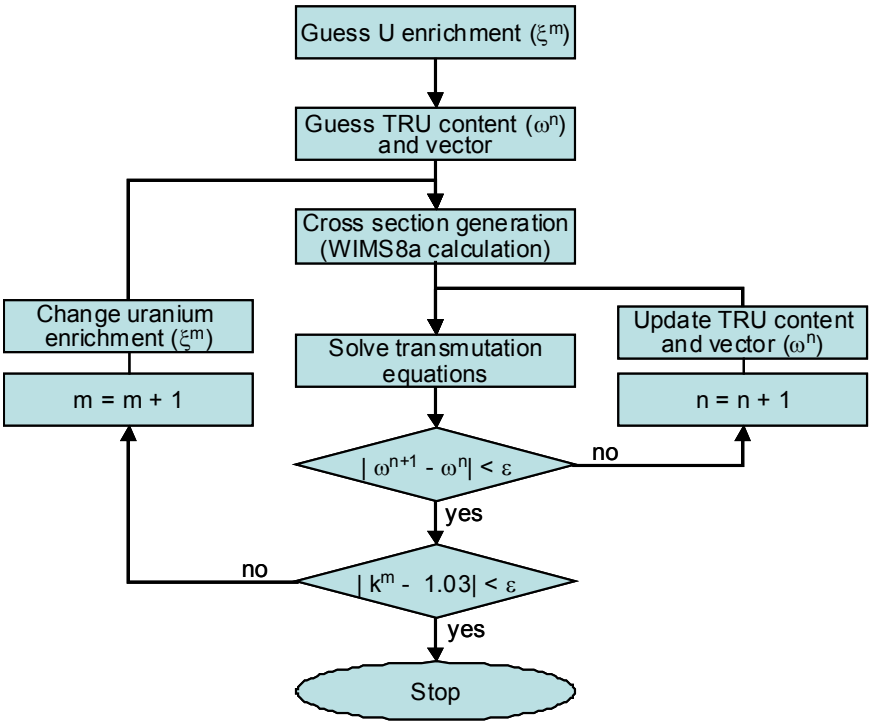
Because the WIMS8 transmutation chain does not include the actinides beyond ^{245}Cm, a tentative approach was developed for providing the pertinent one-group cross sections for these higher nuclides. This approach uses a set of four-group cross sections that were prepared for the higher actinides under a previous EPRI project. [8] These cross sections are based on ENDF/B-V data that were obtained with a typical LWR spectrum. One-group cross sections for these higher actinides are then obtained using four-group neutron spectra from the WIMS8 calculations to collapse the four-group data.

Algorithm to search for equilibrium state

The equilibrium state is a condition in which the charge TRU content and isotopic vector of the $N+1^{th}$ recycle stage are identical to those of the N^{th} stage. To search for the equilibrium state, an iterative algorithm (displayed in Figure 2) is used. There are two levels in this algorithm, designated inner- and outer-iterations. The TRU (or Pu) content and vector of the charge stage are determined with a given uranium enrichment in the inner iteration, while the uranium enrichment is determined in the outer iteration to maintain the desired fuel cycle length.

Initially, a guess of the uranium enrichment of the UO_2 fuel pins and the TRU content and vector of the MOX fuel pins is made. After preparing one group cross section data, the transmutation equations of the UO_2 and MOX regions are solved, independently. Through this calculation, the mass flows at the discharge, separation, fabrication and the charge stages of the next recycle stage are determined. If the TRU content and isotopic vector of the charge stage are not converged (i.e., the values for the next recycle stage are different from those of the current one), the TRU content and vector of the charge stage are replaced with the results determined in the current iteration. After MOX pin loading has been converged, the outer iteration is used to ensure that the specified fuel cycle length is met. This is done by adjusting the uranium enrichment of the UO_2 pins to maintain the specified cycle length. In this study, the specified cycle length is satisfied when the assembly k_∞ is1.03 at the critical burn-up, assuming 3% neutron leakage from the core.

Figure 3. **Equilibrium state searching algorithm**



Equilibrium states of CORAIL and alternative assembly designs

Mass flow

Table 1 provides a summary of the TRU vectors of MOX fuel pins at the charge stage of the equilibrium state for different multi-recycling approaches. For each case, the equilibrium vectors are compared to those obtained using cycle-by-cycle calculations with the WIMS8 code. In this table, CORAIL-Pu and CORAIL-TRU denote Pu and TRU recycling in the CORAIL assembly, respectively. The highly moderated and modified CORAIL assembly and the homogeneous thorium-based TRU assembly concepts are denoted by HM-TRU and TMOX, respectively. Since the results of the cycle-wise TRU recycling were determined by the WIMS8 calculations, Table 1 does not contain the contributions from the higher actinides; the mass fraction of the higher actinides to total TRU is *2.2-5.8% at equilibrium,* in the TRU recycling cases.

Table 1. **Comparison of TRU vector of MOX fuel at charge stage** (%)

		CORAIL-Pu		**CORAIL-TRU**		**HM-TRU**		**TMOX** [a)]	
Cycle		7	Equil.	16	Equil.	7	Equil.	7	Equil.
U enrichment (%)		4.57	4.62	5.06	5.12	5.07	5.11	4.08	4.12
TRU content (%)		8.18	8.45	17.50	20.39	8.16	10.56	3.23	3.83
TRU vector at charge stage (%)	^{237}Np			2.42	1.82	2.79	2.52	2.54	2.34
	^{238}Pu	3.90	3.55	10.61	10.20	6.82	7.42	8.61	8.25
	^{239}Pu	36.10	34.20	26.97	24.40	27.70	24.96	32.31	29.17
	^{240}Pu	27.00	23.33	22.73	21.67	22.39	20.35	18.24	16.96
	^{241}Pu	10.80	10.48	7.51	7.13	8.09	7.80	8.45	7.89
	^{242}Pu	21.10	27.40	14.78	16.93	12.92	11.90	12.95	13.21
	^{241}Am	1.10	1.04	5.93	5.91	4.36	4.37	4.73	4.42
	^{242}mAm			0.07	0.07	0.02	0.02	0.03	0.03
	^{243}Am			4.13	4.59	6.74	6.54	5.01	5.11
	^{243}Cm			0.03	0.03	0.03	0.03	0.03	0.03
	^{244}Cm			3.59	3.90	7.03	6.87	5.81	6.26
	^{245}Cm			1.07	1.14	1.12	1.36	1.29	1.51
	^{246}Cm				1.74		4.62		3.81
	^{247}Cm				0.15		0.35		0.30
	^{248}Cm				0.29		0.82		0.68
	^{249}Bk				3.00E-5		7.00E-5		7.00E-5
	^{249}Cf				1.91E-2		3.51E-2		3.29E-2
	^{250}Cf				3.48E-3		8.36E-3		7.55E-3
	^{251}Cf				4.95E-3		7.39E-3		8.14E-3
	^{252}Cf				9.40E-4		3.02E-3		2.87E-3
	^{253}Cf				0.00E+0		0.00E+0		0.00E+0
	^{253}Es				0.00E+0		0.00E+0		0.00E+0
	Fissile	46.90	44.67	35.65	32.80	36.96	34.21	42.11	38.67

a) Thorium content is 2.43%.

In the case of Pu recycling in the CORAIL assembly, the Pu content of the equilibrium state is similar to that for the 7th recycle stage, which indicates that the CORAIL-Pu assembly would reach the equilibrium state relatively quickly. This is not the case for TRU recycling, however. The TRU content at the 16th recycle stage and the equilibrium state are different by 2.9% (differential). As more of the minor actinides are included in the TRU recycling, the uranium enrichment of the UO_2 fuel pins and TRU content in the MOX fuel pins increase to equilibrium state values of 5.1% and 20.4%, respectively. The TRU fractions are lower in the HM-TRU (10.6%) and TMOX (3.8%) assemblies, by

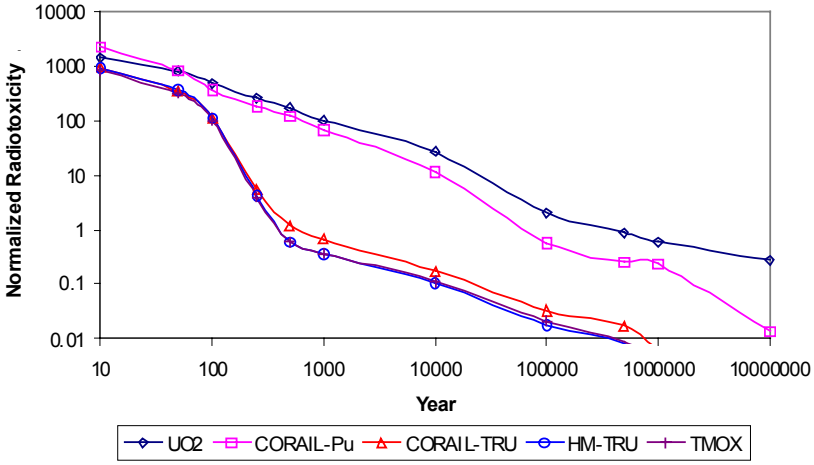
design. The high TRU content in the TRU recycling case may impact the system reactivity coefficients and power peaking factor, but evaluations performed in Refs. 3 and 4 indicate that the higher TRU loading does not adversely affect the core performance, compared to the standard UO_2 assembly case.

Radiotoxicity

The radiotoxicity of the waste discharged from the equilibrium state of the different cases has been estimated by calculating the cancer dose up to 10 million years after disposal. All radiotoxicity values were then normalised to the corresponding values associated with the natural uranium ore needed to produce the charged heavy metal. It was assumed that uranium ore is only needed to produce the enriched UO_2 fuel for the CORAIL or its alternative assemblies; recycled TRU (or Pu) and depleted uranium are used to fuel the MOX pins. Note that in these concepts, the waste passed to the repository is limited to 0.1% of the TRU (or 0.1% Pu and 100% minor actinides in Pu recycling case) and all fission products; most of the TRU (or Pu) are recycled and the discharged uranium will either be used as a make-up feed or stored as low-level waste. On the other hand, all discharged heavy metals from the reference UO_2 assembly are passed to the repository since partitioning of the once-through spent UO_2 fuel would not be practised. In all cases, a five-year cooling time was assumed before disposal.

Figure 4 displays the normalised radiotoxicity of the waste sent to the repository for a number of cases; the meaning of CORAIL-Pu, CORAIL-TRU, HM-TRU, and TMOX are identical to those of Table 1. Initially after disposal, the radiotoxicity of the waste from the TRU-containing assemblies is about a factor 2 smaller than that from the UO_2 assembly. The high ^{244}Cm content in CORAIL-Pu waste is the reason for the initially high radiotoxicity for this concept, but the radiotoxicity decreases below that of the UO_2 assembly within 100 years because ^{244}Cm dies out very quickly ($T_{1/2}$ =18 year). Fission products dominate the radiation hazard after discharge but the hazard associated with most of the fission products dies out within 500 years due to their short half lives. Thus, in those cases which utilise TRU multi-recycling, the normalised radiotoxicity falls below one after 500 years. However, the radiotoxicity of the waste from the CORAIL-Pu assembly is still significant due to the disposal of minor actinides; in particular, the leading contributors (^{240}Pu, ^{239}Pu, and ^{241}Am) in the 1 000 to 10 000 time scale are created from the decay of the Americium and Curium isotopes (e.g., $^{243}Am \rightarrow ^{239}Np \rightarrow ^{239}Pu$, $^{244}Cm \rightarrow ^{240}Pu$). Long-term repository dose release arising from ^{237}Np requires that ^{241}Am be minimised because it is the primary source of the nuclide.

Figure 4. **Comparison of normalised radiotoxicity at repository**



Fuel handling issues

The radiotoxicity of the nuclear waste stored in the repository environment is greatly reduced by keeping the TRU in the reactor fuel cycle. However, the trade-off for this benefit is an increase in the radioactive properties of the TRU-containing assemblies relative to a reference UO_2 assembly. This may complicate fresh fuel handling (e.g., worker dose from neutron and gamma sources) and/or negatively impact fuel separation processes (e.g. temperature increases due to higher decay heat loads may reduce process efficiencies).

To understand the impact of TRU multi-recycling on fuel handling and processing at several stages in the reactor fuel cycle, three key parameters were identified and evaluated: decay heat, gamma energy and neutron emission rates. Comparisons of these parameters for UO_2, full MOX (mono-recycling), and Pu and TRU multi-recycling in the CORAIL and alternative assemblies are provided in Table 2 at several stages in the reactor fuel cycle. In this table, UO_2 and MOX denote a 4% enriched UO_2 assembly and 9.4% Pu content full MOX assembly, respectively. Since the MOX assemblies are currently utilised in the French nuclear programme, it could be considered that the MOX assembly is feasible to fabricate.

For all but the homogenous assemblies (UO_2, MOX and TMOX), the fabrication stage data were evaluated for one metric ton of heavy metal in MOX fuel pin, while the charge stage data were evaluated for one metric ton of heavy metal in the assembly; in the homogeneous assemblies, both stages data were one metric ton of heavy metal in the assembly. Data for discharge and five-year cooling are based on the resulting mass after irradiation from the one metric ton of the charge data (contributions from fission products are included). Note that about 30% of the fuel pins are MOX pins in the CORAIL assembly, except for the MOX and TMOX assemblies (100% MOX pins). This is the main reason for the factor of three differences between the heterogeneous assembly values for the fabrication and charge stages in Table 2.

Table 2. **Comparison of fuel handling indices**

	Stage	**UO_2**	**MOX**	**CORAIL -Pu**	**CORAIL -TRU**	**HM-TRU**	**TMOX**
Decay heat (Watt)	fabrication	0.01045	1 983	1 960	38 390	27 820	9 523
	charge	0.01045	2 054	655	11 930	8 533	8 952
	discharge	2.06E+06	1.98E+06	2.00E+06	2.00E+06	2.00E+06	2.01E+06
	5 year cooling	2 515	5 584	4 404	14 520	11 110	11 540
	disposal	2 515	5 584	3 772	1 963	2 008	2 018
Neutron source (N's/sec)	fabrication	1.23E+04	9.48E+07	1.20E+08	7.65E+12	1.26E+13	4.34E+12
	charge	1.23E+04	9.69E+07	3.96E+07	1.51E+12	2.48E+12	2.60E+12
	discharge	1.23E+09	1.33E+10	9.59E+09	9.08E+12	1.51E+13	1.59E+13
	5 year cooling	5.74E+08	6.47E+08	6.36E+09	2.51E+12	4.14E+12	4.35E+12
	disposal	5.74E+08	6.47E+08	6.32E+09	2.51E+09	4.14E+09	1.54E+09
Gamma energy (MeV/sec)	fabrication	5.10E+08	3.50E+12	3.53E+12	1.45E+14	1.14E+14	3.96E+13
	charge	5.10E+08	5.62E+12	2.00E+12	4.43E+13	3.14E+13	3.35E+13
	discharge	3.54E+18	3.26E+18	3.42E+18	3.33E+18	3.38E+18	3.43E+18
	5 year cooling	6.66E+15	5.60E+15	6.43E+15	6.11E+15	6.23E+15	6.21E+15
	disposal	6.66E+15	5.60E+15	6.43E+15	6.06E+15	6.19E+15	6.17E+15

The decay heat of the fresh UO_2 assembly is essentially zero; only long-lived uranium isotopes are present in the fresh fuel. As can be seen from Table 3, the leading contributors at the fabrication stage of the CORAIL-Pu fuel are ^{238}Pu (88%) and ^{240}Pu (7%), while ^{244}Cm (63-80%) dominates in the TRU-containing assemblies. Compared with the CORAIL-Pu assembly, the decay heat of the TRU containing assemblies is much higher at fabrication and charge stages (factor of 13-18 higher at charge stage), primarily due to the presence of ^{244}Cm in the recycled TRU. All assemblies have essentially the same decay heat loads at discharge, when the fission products dominate.

The protection of workers at the separations and fabrication plants would be complicated by the relatively high neutron emission rates of the TRU-containing assemblies. The leading contributors in the CORAIL-Pu assembly at the fabrication stage are ^{238}Pu, ^{240}Pu and ^{242}Pu. But, in the TRU-containing assemblies, it is predominantly ^{252}Cf (~97%) even though its mass fraction is negligibly small (see Table 1). Note that the specific neutron source (neutrons/g-s) of ^{252}Cf is a factor of 215 000 times that of ^{244}Cm. Thus, at equilibrium, the neutron emission rates from the TRU-containing assemblies are *a factor of 36 170-105 000 higher* at the fabrication stage and a factor of *950-1 660 higher* at the discharge stage, compared with those for the CORAIL-Pu assembly.

The predominant contributor to the gamma source of the CORAIL-Pu assembly at the fabrication stage is ^{238}Pu (~87%). ^{244}Cm, ^{241}Am and ^{252}Cf have higher contributions than ^{238}Pu in the TRU-containing fuels. The fission products (e.g., ^{138}I, ^{134}I, etc) dominate the gamma source at discharge.

Conclusions

The equilibrium states of the CORAIL and alternative assemblies for recycling TRU (or Pu) in PWR fuel cycles were evaluated in terms of fuel handling indices and the radiotoxicity of disposed waste. The equilibrium states are determined using the one-group transmutation code, TRANSEQM. The code can handle higher actinides up to ^{253}Es, therefore making it possible to assess their contributions to the indices of this study. Cross sections and neutron fluxes for TRANSEQM are prepared using the WIMS8 code. However, since the current transmutation chain of the WIMS8 code ends at ^{245}Cm, the cross sections of the actinides beyond ^{245}Cm are provided by collapsing four-group cross sections prepared by EPRI with ENDF/B-V file and four-group neutron spectra calculated by the WIMS8 code.

While the fuel handling indices for the CORAIL-Pu assembly are higher than those for the UO_2 assembly, the Pu recycling in the CORAIL assembly could be considered as a feasible option because its fuel handling indices at the fabrication stage are roughly similar to or lower than those of the full MOX assembly. Multi-recycling of plutonium in the CORAIL assembly has benefits to the repository because plutonium would not be stored in the repository and the long-term radiotoxicity is smaller than that of the standard UO_2 assembly. With this scheme alone, however, the goal of reducing the radiotoxicity to less than that of the source uranium in a thousand years [9] is not met.

Table 3. **Comparison of leading contributors at fabrication stage**

	UO_2		CORAIL-Pu		CORAIL-TRU		HM-TRU		TMOX	
Decay heat (Watt)	TOTAL	1.05E-02	TOTAL	1 960	TOTAL	38 390	TOTAL	27 820	TOTAL	9 523
	^{238}U	8.19E-03	^{238}Pu	1 731	^{244}Cm	24 280	^{244}Cm	22 200	^{244}Cm	7 320
	^{235}U	2.27E-03	^{240}Pu	140	^{238}Pu	12 000	^{238}Pu	4 517	^{238}Pu	1 820
			^{239}Pu	55	^{241}Am	1 213	^{241}Am	435	^{241}Am	159
			^{241}Pu	31	^{240}Pu	309	^{252}Cf	207	^{252}Cf	71
			^{242}Pu	3	^{252}Cf	124	^{240}Pu	149	^{240}Pu	45
					^{243}Cm	117	^{243}Cm	54	^{239}Pu	21
					^{239}Pu	95	^{239}Pu	51	^{243}Cm	21
					^{243}Am	60	^{246}Cm	49	^{246}Cm	15
					^{241}Pu	51	^{243}Am	44	^{250}Cf	13
					^{242}Cm	48	^{250}Cf	40	^{243}Am	13
Neutron source (n's/sec)	TOTAL	1.23E+04	TOTAL	1.20E+08	TOTAL	7.65E+12	TOTAL	1.26E+13	TOTAL	4.34E+12
	^{238}U	1.23E+04	^{238}Pu	5.78E+07	^{252}Cf	7.41E+12	^{252}Cf	1.24E+13	^{252}Cf	4.26E+12
	^{235}U	4.57E+01	^{242}Pu	3.91E+07	^{244}Cm	9.62E+10	^{250}Cf	1.09E+11	^{250}Cf	3.56E+10
			^{240}Pu	2.14E+07	^{250}Cf	8.74E+10	^{244}Cm	8.79E+10	^{244}Cm	2.90E+10
			^{239}Pu	1.31E+06	^{246}Cm	3.16E+10	^{246}Cm	4.34E+10	^{246}Cm	1.30E+10
			^{238}U	1.17E+04	^{248}Cm	2.61E+10	^{248}Cm	3.78E+10	^{248}Cm	1.12E+10
			^{235}U	2.62E+00	^{238}Pu	4.01E+08	^{238}Pu	1.51E+08	^{238}Pu	6.08E+07
					^{242}Pu	5.83E+07	^{240}Pu	2.27E+07	^{242}Pu	8.54E+06
					^{240}Pu	4.72E+07	^{242}Pu	2.12E+07	^{240}Pu	6.83E+06
					^{241}Am	3.50E+07	^{241}Am	1.26E+07	^{241}Am	4.60E+06
					^{242}Cm	1.03E+07	^{242}Cm	4.88E+06	^{242}Cm	1.80E+06
Gamma energy (MeV/sec)	TOTAL	5.10E+08	TOTAL	3.53E+12	TOTAL	1.45E+14	TOTAL	1.14E+14	TOTAL	3.96E+13
	^{235}U	4.96E+08	^{238}Pu	3.08E+12	^{244}Cm	3.92E+13	^{252}Cf	4.23E+13	^{252}Cf	1.46E+13
	^{238}U	1.41E+07	^{240}Pu	2.56E+11	^{241}Am	3.32E+13	^{244}Cm	3.58E+13	^{244}Cm	1.18E+13
			^{241}Pu	1.44E+11	^{252}Cf	2.53E+13	^{241}Am	1.19E+13	^{241}Am	4.35E+12
			^{239}Pu	4.37E+10	^{238}Pu	2.14E+13	^{238}Pu	8.04E+12	^{238}Pu	3.24E+12
			^{242}Pu	6.24E+09	^{243}Cm	1.57E+13	^{243}Cm	7.25E+12	^{243}Cm	2.77E+12
					^{243}Am	3.81E+12	^{249}Cf	3.71E+12	^{249}Cf	1.26E+12
					^{249}Cf	3.80E+12	^{243}Am	2.81E+12	^{243}Am	7.99E+11
					^{245}Cm	1.17E+12	^{245}Cm	7.24E+11	^{245}Cm	2.93E+11
					^{240}Pu	5.62E+11	^{250}Cf	3.43E+11	^{250}Cf	1.12E+11
					^{250}Cf	2.75E+11	^{240}Pu	2.68E+11	^{240}Pu	8.11E+10

Multi-recycling of TRU is the most beneficial to the repository because the amount of TRU to be sent to the repository can be significantly reduced by keeping the TRU in a PWR fuel cycle. The radiotoxicity of the waste discharged from the TRU-containing assemblies decreases below that of the source natural uranium within one thousand years. However, this case may result in the most problems for fuel handling at the fabrication stage due to the extremely high spontaneous fission neutron source generated by ^{252}Cf, ^{250}Cf, and ^{244}Cm, at equilibrium.

The neutron emission rate, and the decay heat and gamma energy emission rate, are generally lower in the first few TRU recycle stages because it takes some time to build up the higher actinides. Thus, a few recyclings of the TRU in the CORAIL or alternative assemblies is attractive. Compared with a conventional UO_2 fuel cycle, it can greatly reduce the nuclear waste to be sent to the repository by keeping the TRU in PWR fuel cycles. This period could be used for perfecting nuclear fuel cycles and reactors that minimise nuclear waste and support a sustainable nuclear enterprise. Because of the attractiveness of TRU multi-recycling in the CORAIL assembly, the practical number of recycles should be determined. Finally, additional improvements could be made to the TRANSEQM code by resolving uncertainties in the one-group cross sections used for the higher actinides.

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