Reactor Physics Assessment of Potential Feasibility of Using Advanced, Nonconventional Fuels in a Pressure Tube Heavy Water Reactor to Destroy Americium and Curium

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REACTOR PHYSICS ASSESSMENT OF POTENTIAL FEASIBILITY OF USING ADVANCED, NON-CONVENTIONAL FUELS IN A PRESSURE-TUBE HEAVY WATER REACTOR TO DESTROY AMERICIUM AND CURIUM

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KEY WORDS: Pressure Tube Heavy Water Reactor, Transmutation, Americium, Curium, Thorium

ABSTRACT

Lattice physics and core physics studies have been carried out to investigate the reactor physics feasibility of destroying americium (Am) and curium (Cm) using special target fuel bundles in blanket fuel channels in a heterogeneous seed-blanket pressure tube heavy water reactor (PT-HWR) core fueled primarily with natural uranium. Results indicate that it should be feasible to achieve net zero production of Am in a single PT-HWR core using 10 to 16 dedicated blanket channels containing Am-based target bundles, while only one dedicated blanket channel would be required for achieving net zero production of Cm. While using target blanket fuel bundles with fuel elements made of Am or Cm mixed with thorium in oxide form ($(Am,Th)O_2$, $(Cm,Th)O_2$) are expected to be suitable for transmutation purposes, the use of fuel elements made of pure americium oxide, especially those in the form of $AmO_{1.55}$, may not be suitable for transmutation purposes due to potential issues with fuel melting under high power operations or postulated accident scenarios. The potential to achieve net zero production of Am and Cm in a single thermal-spectrum reactor, such as a PT-HWR, could help eliminate the need to build and qualify a Deep Geological Repository (DGR) capable of storing minor actinides for a long time (> 1 million years). At the very least, the size and/or number of DGRs required for storing radioactive waste could be reduced significantly. Thus, destroying Am and Cm in PT-HWRs could be regarded as a viable solution to perceived problem of nuclear waste, and may help improve public acceptance of the use of nuclear energy. In addition, it may be possible to apply a similar approach for destroying MAs in other Gen-III+/Gen-IV/SMR technologies.

I. INTRODUCTION

The operation of nuclear reactors (including Gen-III+, Gen-IV, and Small Modular Reactor (SMR) technologies) produces high level radioactive waste (HLW) in the spent fuel, which contains numerous radioactive fission products, plutonium isotopes, and minor actinides (MAs, such as isotopes of Np, Am, Cm, and other heavier elements). Aside from the fission products and certain neutron activation products, a key long-term problem for radioactive waste storage are the isotopes of Am and Cm, particularly those that are non-fissile, and those that have a long half-life, ranging from 432 years (for Am-241) to 15,600,000 (for Cm-247), as shown in Table I. Long-term isolation and storage of MAs in a Deep Geological Repository (DGR) to ensure the protection of the environment for millions of years is considered scientifically and technologically feasible [1], [2]. However, it may be preferable to separate and destroy Am and Cm through direct fission, or transmutation into fissile isotopes by neutron capture, followed by fission.

Table I. Expected Time Scales and Thermal Neutron Fluxes Required to Destroy MAs and non-Fissile Isotopes of Pu

Actinide	**Half Life (years)**	**Dominant Decay Mode**	**Years to Decay to 1.0×10^{-4} ***	**Thermal σ_{absorb} (barns)**	**Thermal $\sigma_{fission}$ (barns)**	**Fission Spectrum $\sigma_{fission}$ (barns)**	**Years to Reduce to 1.0×10^{-4} with $1.0\times10^{+14}$ n/cm^2/s Neutron Flux****	**Flux to Reduce to 1.0×10^{-4} within 100 years (n/cm^2/s) ****
Np-236	154,000	EC to U-236***	2,046,742	3073.9	2,453	1.92	1.0	9.50E+11
Np-237	2,144,000	Alpha to Pa-233	28,494,906	144	0.019	1.34	20.3	2.03E+13
Pu-238	87.7	Alpha to U-234	1,166	473.2	15.18	1.99	6.1	5.64E+12
Pu-240	6564	Alpha to U-236	87,239	263.7	0.053	1.36	11.1	1.11E+13
Pu-242	373,300	Alpha to U-238	4,961,357	16.8	0.0023	1.13	173.6	1.74E+14
Am-241	432	Alpha to Np-237	5,744	534.7	2.711	1.38	5.5	5.37E+12
Am-242m	141	IT to Am-242***	1,874	7,460.0	6,235	1.84	0.4	3.71E+11
Am-243	7,370	Alpha to Np-239	97,951	70.6	0.103	1.08	41.4	4.13E+13
Cm-243	29	Alpha to Pu-239	387	662.8	549.6	1.94	4.4	3.27E+12
Cm-244	18	Alpha to Pu-240	241	14.3	0.906	1.57	110.6	1.20E+14
Cm-245	8,500	Alpha to Pu-241	112,970	1,961.4	1,674	1.74	1.5	1.49E+12
Cm-246	4,760	Alpha to Pu-242	63,263	1.3	0.124	1.23	2,212	2.29E+15
Cm-247	15,600,000	Alpha to Pu-243	207,332,337	121.0	70.8	1.91	24.1	2.41E+13
Cm-248	348,000	Alpha to Pu-244	4,625,106	2.6	0.328	1.25	1,118	1.12E+15

* Calculated using equation $N(t)/N(0) = \exp(-1.0 * 0.693/T_{half} * t)$

** Calculated using equation $N(t)/N(0) = \exp(-1.0 * (0.693/T_{half} * t) + \sigma * \phi * t))$.

*** EC = Electron Capture; IT = Internal Transition; Am-242 has a very short half-life (~16 hours) to decay to Cm-242 (by beta decay) or Pu-242 (by electron capture). Am-242 also has a large capture cross section ($\sigma_{capture}$=4872 barns) to create Am-243.

I.A Option of Thermal Spectrum to Destroy MAs

Although fast-spectrum reactors are usually proposed for the destruction of MAs [3], the fission cross sections for MAs and non-fissile Pu isotopes in a fast spectrum (which range from 1.1 barns to 1.9 barns, see Table I) are much lower than the absorption cross sections (including both capture and fission) in a thermal neutron energy spectrum. With the exception of Cm-246 and Cm-248, the thermal-spectrum absorption cross sections range from 14 barns (for Cm-244) to 7,460 barns (for Am-242m). Many MAs are both fissionable and fertile, and can be transmuted into fissile isotopes that have a high thermal fission cross section. A good example is Am-241 which has cross sections of $\sigma_{capture}$= 532 barns, $\sigma_{fission}$= 2.7 barns in the thermal spectrum. With thermal neutron capture, Am-241 will be transmuted to Am-242 or Am-242m, both which have very high thermal fission cross section ($\sigma_{fission}$=1861 barns and $\sigma_{fission}$=6235 barns respectively). It is noted that Am-242 is short-lived ($t_{half\text{-}life}$=16.02 hours) and will decay to Cm-242 (82.7%) and Pu-242 (17.3%). The Cm-242 then decays (with a 162.8-day half-life) to Pu-238. As shown in Table I, both Pu-238 and Pu-242 have much lower thermal fission cross sections than Am-242m.

Previous studies [4], [5] have established the feasibility of using current-generation thermal-spectrum nuclear reactors to carry out the destruction of recycled Pu in mixed oxide fuels, such as $(Pu,Th)O_2$ and $(Pu,U)O_2$. It may also be feasible to use thermal-spectrum reactors to transmute and destroy Am and Cm. For example, as shown in Table I, the use of a thermal neutron flux of 1.0e+14 n/cm^2/s (typical for current reactors) could reduce the inventory of most MAs and non-fissile Pu isotopes by a factor of 10,000 within 1 to 42 years. Much higher thermal neutron fluxes (1.2e+14 n/cm^2/s to 2.3e+15 n/cm^2/s) would be needed to reduce the inventories of the other MAs (such as Pu-242, Cm-244, Cm-246, and Cm-248) to less than 1.0×10^{-4} within 100 years. Such high thermal neutron fluxes may be achieved in specially-designed reactors, perhaps similar to the HFIR at ORNL [6].

However, even the lower neutron fluxes of conventional reactors (along with those in various Gen-IV and SMR technologies) may still be sufficient to achieve a net zero production of Am/Cm, if the consumption rate of Am/Cm in specially-designed targets matches the production rate of Am/Cm in the fuel in the reactor core. The reduction or elimination of Am/Cm (and other MAs) would help reduce the volume of waste sent to a DGR, and it

would also eliminate the need to qualify a DGR for a long time period (> 1 million years). The added potential benefit of actively destroying MAs is that it could address a fundamental underlying public concern about nuclear waste, and thus would help increase public acceptance of the use of nuclear energy.

I.B Previous Studies

Various research groups within the international community have investigated the feasibility of using thermal-spectrum reactors as an alternative to fast reactors for destroying plutonium and minor actinides [7], and many studies have been published on the topic, as discussed in Reference [8]. Conventional thermal-spectrum reactors, such as LWRS (including PWRs and BWRs) and pressure tube heavy water reactors (PT-HWRs), such as CANDU (CANada Deuterium Uranium) reactors [9], [10], [11], have been considered for use in consuming/destroying the plutonium, minor actinides and some long-lived fission products found in the accumulated inventories of spent fuel [12] to [19].

In the late 1990s, Meneley and Dastur [20], and also Chan and co-workers [21] suggested that advanced burner CANDU reactors (CANDU-AB) with alternative / advanced fuels could potentially be adapted for consuming/destroying plutonium and various minor actinides. The practical destruction capability that could be achieved with a CANDU-AB would be comparable or better than alternative approaches, such as using large fast-spectrum advanced liquid metal-cooled reactors (ALMR), or even sub-critical fast reactors driven by a proton accelerator (ADS). In the studies by Chan et al. [21], the emphasis was on mixing plutonium with an inert matrix material such as SiC, or using plutonium mixed with depleted uranium or thorium in a mixed-oxide fuel to use subsequently in a CANDU reactor. Consideration was also given for mixing Np, Am and Pu altogether with SiC for burning in fuel. Preliminary calculations in that study indicated that 60% of the original actinide inventory (which was mainly fissile Pu) could be consumed / destroyed, although there would be a net increase in the amount of radio-active heavier actinides, such as Pu-242, Am-243, Cm-242, and Cm-244.

In the more recent period of 2007-2013, researchers at Atomic Energy of Canada Limited (AECL) – Chalk River Laboratories (CRL) carried out various lattice physics studies [22]-[24] looking at the feasibility of

destroying transuranic elements (TRU = Np + Pu + Am + Cm), Am + Cm, or just Am in various fuel bundle concepts that could be used in an entire core in a CANDU-type PT-HWR. In those studies, it was assumed that the TRU would be obtained from spent LWR fuel. In most of these studies, TRU was mixed with enriched uranium fuel, thorium, or an inert matrix material (such as ZrO_2) and placed in one fuel element, one ring of fuel elements, or in all fuel elements in a fuel bundle. In some studies, very small amounts of Am and Am/Cm were mixed homogeneously with LEU, and acted as a burnable neutron absorbing poison. Various bundle concepts for destroying TRU, Am/Cm or just Am were investigated, including 43-element and 55-element bundles. Different fuel matrix materials were considered, including mixed oxide (MOX) fuels, and inert matrix fuels (IMF). In a number of other studies, Am/Cm or just Am was mixed and placed in a single central fuel element. It was found in these various studies by Hyland and co-workers that in a single pass through a CANDU core, at exit burnup, anywhere from 40% to 75% of the total Pu, and 40% to 80% of the total Am could be destroyed/transmuted. However, the content of some of the heavier radioactive isotopes would experience a net increase due to neutron captures on the lighter isotopes of Pu and Am. For example, the content of Pu-242 would increase, ranging between by 40% and 300%, while the content of Cm would increase as well, ranging between 260% and 3,200%. The earlier work by researchers at CRL suggested that one type of fuel bundle containing TRU or Am/Cm would be used to fuel the entire core of a CANDU PT-HWR. At the time, it appears that no consideration was given for using different parts of a PT-HWR core, such as the outer ring of fuel channels, or dedicated regions of the core for irradiating targets made of MAs.

In a later publication by Hyland and Gihm [25], it was suggested that a heterogeneous PT-HWR core could be set up, using dedicated fuel channels to irradiate fuel bundle targets containing higher concentrations of minor actinides. In that particular case, Am and Cm was mixed with an inert matrix material (SiC), from 14 wt% to 35 wt% of Am+Cm in the inert matrix fuel (IMF), and placed in fuel bundles (using various bundle concepts, ranging from 21-element to 43-element bundles) in 30 exterior fuel channels in a 380-channel PT-HWR core. It was found that 30% to 90% of the Am could be transmuted, depending on the fuel bundle design, initial Am+Cm content, and the residence time in the PT-HWR. Up to 10% per year of the Am+Cm could be transmuted, although 10 to 18 years of residence time in the PT-HWR core would be required to achieve nearly 90%

transmutation. One of the trade-offs was that the content of various Pu isotopes (such as Pu-242, Pu-244) and Cm isotopes (such as Cm-242 and Cm-244) would increase due to neutron capture and decay on the Am isotopes.

In addition to researchers at AECL/CRL, recent work on evaluating the use of CANDU-type / PT-HWRs and PWRs for consuming / destroying Pu and/or MAs has been carried by other research staff at other institutions. Examples include the work by A. Nuttin and co-workers [26], [27], [28], Z. Gholamzadeh and co-workers [29], A. Morreale and co-workers [30], [31], and D. Hartanto and co-workers [32]. The reader is advised to review those other references for more information and details. In most of these studies, recycled Pu, Am, or TRU (Np + Pu + Am + Cm) was mixed with uranium (depleted, natural, or low enrichment) or thorium in either homogeneous or heterogeneous fuel bundles or fuel assemblies, and then burned in a homogeneous reactor core with one type of fuel bundle or one type of fuel assembly. For example, Morreale and co-workers [30] found that TRU mixed with natural uranium in a heterogeneous 43-element CANFLEX-type fuel bundle, and then used in a 900-MWe-class PT-HWR (such as a CANDU-9) could achieve reductions in the Np, Pu, and Am content of -45%, -33% and -63% respectively, although the Cm content increased substantially (11.6 grams per fuel bundle, or +2,491%). In the studies by Hartanto and co-workers [32], calculations were performed to evaluate the consumption of TRU in a PT-HWR using fuel pellets made of TRISO-type (TRI-structural ISOtropic) fuel particles containing TRU in a silicon carbide (SiC) matrix. With discharge burnup values close to 700 MWd/kg-TRU, it was found that up to 70% of the TRU and up to 78% of the americium would be consumed, while there was a net increase of curium (from ~0.600 grams/kg-TRU at zero burnup to ~30-33 grams/kg-TRU at exit burnup).

I.C A Modified Approach for Destroying MAs Using PT-HWRs

Another option for destroying Am or Cm is to use special target bundles in the peripheral fuel channels of a PT-HWR, such as the CANDU reactor, a Gen-III/Gen-III+ reactor technology. PT-HWRs would be well-suited for this application, since they have excellent neutron economy due to the use of heavy water moderator/coolant, and can also carry out online re-fueling to replace fuel bundles and Am or Cm target bundles. Thus, the objective

of this study was to carry out preliminary lattice physics and core physics assessments of the feasibility of destroying Am or Cm in PT-HWRs using special targets in non-conventional fuel bundles in seed/blanket cores, where the blanket region contains special fuel bundles with Am or Cm. This paper is considered a companion to an earlier one on the use of PT-HWRs for destroying long-lived fission products (LLFPs) [33].

II. DESCRIPTION OF PROBLEM

The proposed approach for destroying Am or Cm in a conventional 700-MWe-class PT-HWR is illustrated in Fig. 1 (core concept) and Fig. 2 (lattice concepts). In the 380-fuel-channel core, the inner 320 "seed" channels are fueled with 37-element (B37) natural uranium (NUO_2) fuel bundles. The outer 60 "blanket" fuel channels adjacent to the radial heavy water reflector are also fueled with 37-element bundles with either homogeneous or heterogeneous designs. In the homogeneous blanket design, all 37 elements are made of a mixture of 10 vol% AmO_2 (or Cm_2O_3) and 90 vol% ThO_2. The ThO_2 serves as fertile matrix material. In the heterogeneous blanket bundle design, the outer 18 elements are made of NUO_2, while the inner 19 elements are made of pure AmO_2. It is recognized that using fuel elements made of pure AmO_2 will require special fuel manufacturing and handling facilities [8], given the decay heat and radiation generated by Am-241 and trace amounts of Am-242m and Am-243.

For the purpose of these studies, it was assumed that the americium oxide would be stoichiometric (AmO_2), although the practical manufacture of americium oxide may be oxygen-deficient. In previous computational physics studies where the transmutation of americium in thermal-spectrum reactors has been investigated [34], it has usually been assumed that the composition of americium would be stoichiometric (AmO_2) or only slightly oxygen-deficient ($AmO_{1.9}$). For example, fuel pellets made of $(U,Am)O_{1.94}$ were irradiated in the High Flux Reactor (HFR) (a thermal spectrum reactor) at Petten, Netherlands, over the period of 2011 March to 2012 May [35], and these pellets operated at fuel temperatures ranging from 1,000°C to 1,200°C. Depending on the manufacturing methods used, and the operating temperature, the oxygen content in the americium oxide could be even lower, ranging from $AmO_{1.5}$ (or Am_2O_3) to $AmO_{1.9}$. Previous studies investigating the manufacturing of americium oxide and mixed-oxide compounds containing americium, along with measurements and predictions of thermo-physical properties have shown that the oxygen-to-americium atom ratio will vary from 1.5 to 2.0 [35], [36], [37]. A potential concern is that at higher operating fuel temperatures, particularly above 700°C, the americium oxide may become more oxygen deficient, shifting from AmO_2 to AmO_x, with $1.5 \leq x \leq 2.0$ [36], [37],

[38], [39], [40]. This shift will cause its thermal conductivity to drop [41], [42], and could lead to higher fuel temperatures unless the power density in the fuel is reduced. This reduction in thermal conductivity is expected to be more of a concern for target fuel elements made of pure AmO_2, rather than $(Am,Th)O_2$, where the thermal-conductivity in the latter will be dominated by that of ThO_2. However, as will be shown later, the power levels of Am-based fuel bundles in blanket channels are expected to be less than one tenth (1/10) of those of UO_2 fuel in seed channels. With lower power levels, but with comparable coolant flow rates, it is expected that there should be sufficient thermal margin to prevent melting of the fuel, although thermal-hydraulic and heat transfer calculations will be required eventually to confirm this supposition, especially considering the reduction in thermal conductivity for $AmO_{1.5}$ relative to AmO_2, and relative to UO_2. For comparison, $AmO_{1.5}$ (Am_2O_3) melts at ~2,205°C, and AmO_2 melts at ~2,500° C [42] which are 21% to 12% lower than UO_2 (T_{melt} ~ 2,865°C) [39]. Another americium compound, $AmO_{1.55}$, appears to have a lower melting point (T_{melt} ~1,750°C) [43], although more recent work suggests that it may be closer to ~2,000°C [44]. With regards to thermal conductivity, that for $AmO_{1.5}$ ranges from ~2.3 W/m/K at 300° C to ~1.0 W/m/K at 1200 °C [36], while that for AmO_2 ranges from ~4.7 W/m/K at 300° C to ~2.8 W/m/K at 1,200 °C [42], which is comparable to that for UO_2, ranging from ~4.4 W/m/K at 300 °C to ~2.4 W/m/K at 1,200 °C [42]. Thus, the thermal-conductivity for $AmO_{1.5}$ is nearly half that for AmO_2 and UO_2.

In PT-HWRs with NUO_2 fuel, the peak fuel centerline temperature in high-power channels (~6,500 kW) is usually less than 1,700°C , and during transient postulated accident scenarios, such as a loss-of-coolant-accident (LOCA), is less than 1,850°C [45]. If the power level remained the same, but the thermal conductivity of the UO_2 was reduced to half its value, comparable to $AmO_{1.5}$, then the peak centerline temperature would jump to approximately 3,100°C during steady state operations, and to approximately 3,400°C during a transient LOCA event, which are clearly above the melting points of UO_2, AmO_2, and $AmO_{1.5}$. However, as mentioned above, and as will be shown later, the blanket channels operating with Am-based fuels operate at power levels that are less than one-tenth (1/10) that of the power levels of the seed channels containing UO_2 fuel. Thus, it would appear unlikely, even with degradation of the thermal-conductivity, that the blanket channels containing Am-based fuels

would experience peak fuel centerline temperatures above 1,700°C in steady-state operations, and above 1,850°C during a transient LOCA event, the latter which is at least 350°C below the melting point for $AmO_{1.5}$. However, in the unlikely event that the power levels in the blanket channels approached nearly 50% of the peak power levels in the seed channels, and if the pure americium oxide fuel elements was in the form of $AmO_{1.55}$ (with an estimated melting point of ~1,750°C), then there would be almost zero margin to melting at the fuel centerline for the fuel elements in the blanket fuel bundle with the peak centerline temperatures during steady state operations. Under the same assumptions, then during a postulated transient LOCA event, it appears that blanket fuel bundles containing fuel elements made of pure $AmO_{1.55}$ could briefly experience fuel centerline melting. Such a scenario may discourage the use of fuel elements made of pure americium oxide, in the form of $AmO_{1.55}$. To ensure that Am-based fuels have sufficient margin to prevent melting, it may be necessary to increase the coolant flow rate in the blanket fuel channels and/or de-rate the reactor to a lower power level to reduce the power density in the Am-based blanket fuel.

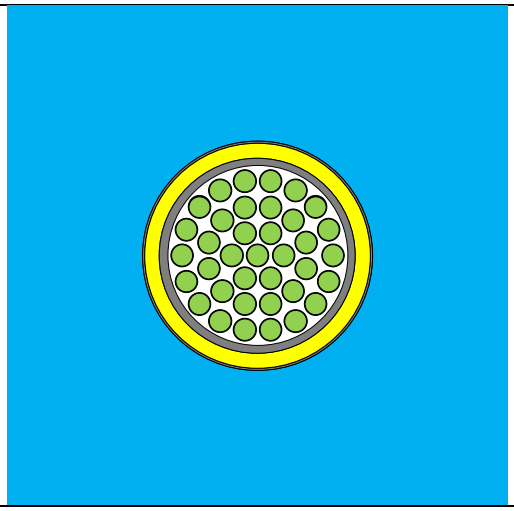
In the heterogeneous blanket bundle design, the outer 18 elements are made of NUO_2, while the inner 19 elements are made of pure AmO_2. It is recognized that using fuel elements made of pure AmO_2 will require special fuel manufacturing and handling facilities [8], given the decay heat and radiation generated by Am-241 and trace amounts of Am-242m and Am-243.

The PT-HWR specifications are shown in Table II, while the lattice concept specifications and fuel compositions are shown in Table III and Table IV. The three heterogeneous seed/blanket core concepts are shown in Table V. Two of the cores have Am-containing blanket lattices (MA-03 and MA-04), while one core has Cm-containing blanket lattices (MA-07). The MA-containing blanket bundles are pushed to a higher burnup (~20 MWd/kg) to achieve a modest consumption rate of MAs, and to limit the accumulation of fission products. The NUO_2 seed bundles will have core-average burnups ranging from 6.8 to 7.4 MWd/kg, providing sufficient reactivity to maintain core criticality (k_{eff}=1.000) during normal operations.

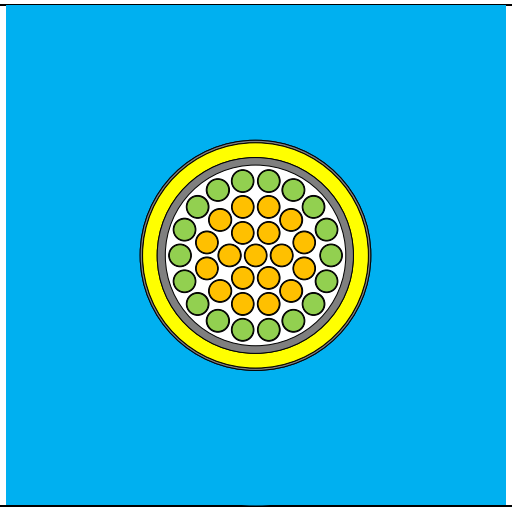
Row\Col	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	Row\Col
A									B	B	B	B	B	B									A
B						B	B	B	S	S	S	S	S	S	B	B	B						B
C					B	S	S	S	S	S	S	S	S	S	S	S	S	B					C
D				B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B				D
E			B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B			E
F			B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B			F
G		B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B		G
H		B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B		H
J	B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B	J
K	B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B	K
L	B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B	L
M	B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B	M
N	B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B	N
O	B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B	O
P		B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B		P
Q		B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B		Q
R			B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B			R
S			B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B			S
T				B	S	S	S	S	S	S	S	S	S	S	S	S	S	S	B				T
U					B	S	S	S	S	S	S	S	S	S	S	S	S	B					U
V						B	B	B	S	S	S	S	S	S	B	B	B						V
W									B	B	B	B	B	B									W
Row\Col	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	Row\Col

* S = Seed Fuel, B = Blanket Fuel containing MAs.

Fig. 1. PT-HWR Core Configuration for Burning MAs in Blanket Bundles



a) 37-element (B37) Bundle with NUO_2 or MA Matrix Fuel



b) Outer 18 Elements Contain NUO_2. Inner 19 Elements Contain MA Matrix

Fig. 2. Seed Fuel Bundles and MA-Containing Blanket Fuel Bundles

Table II PT-HWR Specifications

Quantity	Value, Units
Nominal Reactor Power	2,061 MW_{th}
# of fuel channels	380
# bundles per channel	12
Bundle Power	600 kW
Lattice pitch (square)	28.575 cm
Length of fuel channel	594 cm
Reflector thickness	66 cm
Moderator	99.8 wt% D_2O, ~69°C
Pressure Tube (PT)	Zr-2.5%Nb
Calandria Tube (CT)	Zircaloy-2 (Zr-2)
Coolant	99.1 wt% D_2O, ~288°C
Coolant mass flow rate	21 to 24 kg/s per channel

Table III Lattice Specifications

Bundle Concept	**B37**
# Fuel Elements	37
Bun. / Fuel Stack Length (cm)	49.5 / 48.0
UO_2 Fuel Density (g/cm^3)	10
ThO_2 Fuel Density (g/cm^3)	9.7
AmO_2 Fuel Density (g/cm^3)	9.7 to 11.68
Cm_2O_3 Fuel Density (g/cm^3)	9.7 to 12.00
Nominal Fuel Temp. (K)	941
Fuel Element OR (cm)	0.65
Fuel Pellet OR (cm)	0.61
Cladding Material	Zircaloy-4
B37 NUO_2 Bundle HM Mass (kg)	18.9
B37 MA Bundle HM Mass (kg)	18.3 to 18.6
Target burnup (MWd/kg) for NUO_2	6.5-7.5
Target burnup (MWd/kg) for MA Bundles	20

Table IV Fuel Composition

Lattice	**Fuel Composition**
LC-01	NUO_2 in B37
MA-03	10 vol% AmO_2, 90 vol% ThO_2
MA-04	Outer 18 Pins: NUO_2 Inner 19 Pins: 100% AmO_2
MA-07	10 vol% Cm_2O_3, 90 vol% ThO_2

Table V. Heterogeneous Seed/Blanket Core Concepts

Core Concept	Seed Fuel*	Blanket Fuel*	MA Target Material
CC-MA-03	LC-01	MA-03	10 vol% AmO_2, 90 vol% ThO_2
CC-MA-04	LC-01	MA-04	Outer 18: NUO2 Inner 19:: AmO_2
CC-MA-07	LC-01	MA-07	10 vol% Cm_2O_3, 90 vol% ThO_2

* The inner 320 fuel channels are filled with Seed fuel bundles, while the outer 60 fuel channels are filled with MA-containing Blanket fuel bundles

The isotopic content of the Am and Cm is based on that found in spent PT-HWR NUO_2 fuel, burned to ~7.25 MWd/kg, and then allowed to undergo decay for ~10 years. These compositions, and the mass of each MA element in a spent and decayed fuel bundle are shown in Table VI. Data are also shown for Np and Pu. Approximately 97 wt% of the Am (1.54 grams per bundle) is Am-241, while nearly 97% of the Cm (2.2 milligrams per bundle) is Cm-244. Most the Am-241 has been produced from the decay of Pu-241, while most of the Cm-244 has been produced from neutron capture on Cm-243, and also the decay of Am-244.

Table VI MAs in Spent NU Fuel (After 10 Years Decay)

Isotope	Mass of Isotope in Bundle (g/bundle)	Mass Fraction in Element	Total for Element in Bundle (g/bundle)
Am-241	**1.4985**	**0.970975**	1.543294
Am-242	1.0962×10^{-8}	0.000000	
Am-242M	2.1665×10^{-4}	0.000140	
Am-243	4.4577×10^{-2}	0.028884	
Am-244	2.1525×10^{-12}	0.000000	
Am-244M	1.6836×10^{-12}	0.000000	
Cm-242	2.1290×10^{-6}	0.000980	0.002172
Cm-243	4.0217×10^{-5}	0.018513	
Cm-244	**2.1027×10^{-3}**	**0.967940**	
Cm-245	2.5176×10^{-5}	0.011590	
Cm-246	2.1189×10^{-6}	0.000975	
Cm-247	3.8784×10^{-9}	0.000002	
Cm-248	7.2558×10^{-11}	0.000000	
Cm-249	5.6808×10^{-22}	0.000000	
Cm-250	1.7553×10^{-13}	0.000000	
Np-236	2.2971 ×10 -7	0.000000	0.514505
Np-237	**0.51450**	**0.999996**	
Np-238	2.6253×10^{-9}	0.000000	
Np-239	1.9907×10^{-6}	0.000004	
Pu-238	6.0287×10^{-2}	0.000866	69.62711
Pu-239	48.111	0.690983	
Pu-240	18.048	0.259211	
Pu-241	2.3738	0.034093	
Pu-242	1.0337	0.014847	
Pu-243	2.3852×10^{-10}	0.000000	
Pu-244	1.2472×10^{-5}	0.000000	

III. METHODS AND ANALYSES

The methods used for analyzing seed/blanket PT-HWR cores with Am or Cm target blanket bundles were similar to those described in previous studies [4], [33], [46]. Lattice physics modeling is performed using WIMS-AECL Version 3.1 [47], in combination with an 89-group nuclear data library based on ENDF/B-VII.0 [48]. The nuclear data library used with WIMS-AECL has data for heavy elements and isotopes up to Cm-250. Heavier elements and isotopes, such as berkelium (Bk), californium (Cf), einsteinium (Es), Fermium (Fm) and others are not included, although it is expected that the production of such heavy isotopes would be relatively low.

Lattice burnup (BU) calculations were performed using an assumed, "nominal" bundle power of ~600 kW for the NUO_2 B37 seed fuel (LC-01) and ~150 kW for the MA blanket fuel bundles (MA-03, MA-04, and MA-07). The 600-kW value is the approximate flux-mean-squared bundle power in a conventional 700-MWe-class PT-HWR operating with 37-element fuel bundles with NUO_2 fuel, and such a value is typically used in lattice physics calculations [11]. The 150-kW nominal value is simply the 600-kW value reduced by a factor of 4, guided by the fact that blanket fuel, with a lower content of fissile fuel, and being adjacent to the outer radial reflector where the neutron flux is lower, is expected to have lower bundle power levels than seed fuel.

In previous studies of seed/blanket cores with thorium-based blanket fuel bundles, a nominal blanket bundle power of 150 kW was used in lattice physics calculations [46]. In principle, data from core physics calculations could be used to determine the flux-mean-squared bundle power in the blanket region, and such values could then be used in updated lattice physics calculations, in an iterative approach between core physics and lattice physics calculations [49]. However, for preliminary reactor physics assessments, it was considered sufficient to use a bundle power of 150 kW for the blanket fuel. As will be seen later, the channel power level in the blanket channels ranges from ~300 kW to 2,000 kW, giving average blanket bundle power levels ranging from 24 kW to 169 kW; thus, the 150-kW value initially chosen within the same order-of-magnitude.

WIMS Utilities Version 2.0 [50] was used for processing the data from WIMS-AECL to produce homogenized two-group diffusion data of the D_2O reflector and the fuel as a function of burnup/irradiation, for use in core physics models with RFSP [51]. More details on similar analyses are found in previous studies [4], and [46].

Time-average equilibrium core physics and refueling modeling were carried out using RFSP 3.5.1 [51], a two-group, 3D (x,y,z) diffusion code. RFSP is used to determine the power distributions, core k_{eff}, and channel refueling rates for user-specified burnup distributions in the core. Modeling approximations are similar to those described in previous studies [4], [46]. The refueling rate of the MA blanket fuel was adjusted such that the blanket burnup would ~20 MWd/kg, a nominal value similar to that used in previous studies [33]. The refueling rates of the NUO_2 B37 fuel in various regions within the 320-channel seed region were adjusted to make the radial power distribution in the seed region as flat as possible, and also to ensure that k_{eff} ~1.015. A 15-mk (1 mk = 100 pcm = 0.001 $\Delta k/k$) excess reactivity allowance accounts for various reactivity devices (such as adjusters for flux-shaping) that are were not modeled explicitly. The power level of the reactor was also adjusted from the nominal value of 2,061 MW_{th} to ensure that the peak channel power would be less than 6,500 kW, and that the peak bundle power would remain below 800 kW. These limits ensure that the coolant does not experience significant bulk boiling upon channel exit, and that the maximum linear element rating (LER) of the seed fuel will stay below 50 kW/m for fuel integrity [52].

IV. LATTICE PHYSICS RESULTS

Sample results of lattice physics calculations with WIMS-AECL 3.1 for the seed fuel (LC-01) and the different MA blanket bundles (MA-03, MA-04 and MA-07) are shown in Fig. 3, Fig. 4, and Fig. 5, and also in Table VII, Table VIII, and Table IX.

IV.A. Neutron Multiplication

The plots of the infinite neutron multiplication factor (k-infinity) vs. burnup in Fig. 3 demonstrate that the blanket lattices have a highly reduced k-infinity relative to the NUO_2 seed bundle, due the reduction in the initial content of fissile fuel (particularly for MA-03), but also due to the large neutron absorption occurring in the non-fissile isotopes of Am-241 and Cm-244. However, as burnup proceeds, the k-infinity values of the MA lattices increase as new fissile fuel is bred by neutron capture to produce Am-242/Am-242m, and Cm-245, and also U-233 (bred from Th-232). The k-infinity for MA-03 ranges from 0.05 to 0.70 (at 20 MWdk/kg), while MA-4, which as 18-element ring of NUO_2, has k-infinity ranging from 0.45 to 0.60 (at 20 MWd/kg). The Cm-containing lattice MA-07 is much more reactive, due to the presence of small amounts of fissile Cm-243 (1.85 wt% in Cm) and fissile Cm-245 (1.16 wt% in Cm), as shown in Table VI. As shown earlier Table I, Cm-243 has a thermal fission cross section of 549.6 barns, while Cm-245 has a very high thermal fission cross section of 1,674 barns. For the Cm-containing lattice (MA-07), the k-infinity ranges from ~0.8 to 1.0 (at 20 MWd/kg).

Plots of the change in the value of k-infinity (Δk_{void} = k-infinity$_{void}$ – k-infinity$_{cool}$) when the coolant density drops to nearly zero (ρ_{cool} = 0.81 g/cm^3 $\rightarrow$ ρ_{void} ~ 0.001 g/cm^3) are shown in Fig. 4. The change in multiplication factor with voiding (Δk_{void}) is similar to the coolant void reactivity (CVR = (k-infinity$_{void}$ – k-infinity$_{cool}$) / (k-infinity$_{void}$ × k-infinity$_{cool}$)), and it is nearly identical in value (Δk_{void} ~ CVR) when the values of k-infinity are close to unity (1.000). The Δk_{void} is an important reactor physics safety parameter for PT-HWRs (along with the fuel temperature coefficient (FTC) and others [9], [10]), and it is desirable to minimize the value of Δk_{void} (and

CVR). For a PT-HWR core filled with NUO_2 in 37-element bundles, the Δk_{void} varies from ~+20 mk (for fresh fuel) to ~+13 mk (at exit burnup), with a core-average value of ~+18 mk, which corresponds to a CVR value of ~+15 mk [53], [54]. For the blanket fuel containing Cm (MA-07), the Δk_{void} varies from +22 mk to +19 mk, which is somewhat higher than that for NUO_2. However, the blanket bundles containing Am have much lower values of Δk_{void}, ranging from +3.5 mk to +8 mk.

Since the seed-blanket cores in this study use conventional 37-element NUO_2 fuel in 320 out of 380 channels, and since the neutron importance of the outer 60 blanket channels is reduced due to their lower power levels (as will be shown later), the reactor physics behavior of the seed-blanket cores will dominated by the seed region. Hence, the reactor physics behavior of these seed-blanket cores should be very similar to that of a conventional PT-HWR core fueled entirely with NUO_2 fuel. Although full-core reactor physics studies of seed/blanket cores to evaluate reactivity coefficients (such as CVR, FTC, and others) are yet to be performed, it is expected that such seed-blanket cores using Am-based blanket fuels would have a slightly lower CVR (~13.5 mk), while seed-blanket cores using Cm-based blanket fuels would have a slightly larger (CVR ~ 15.5 mk).

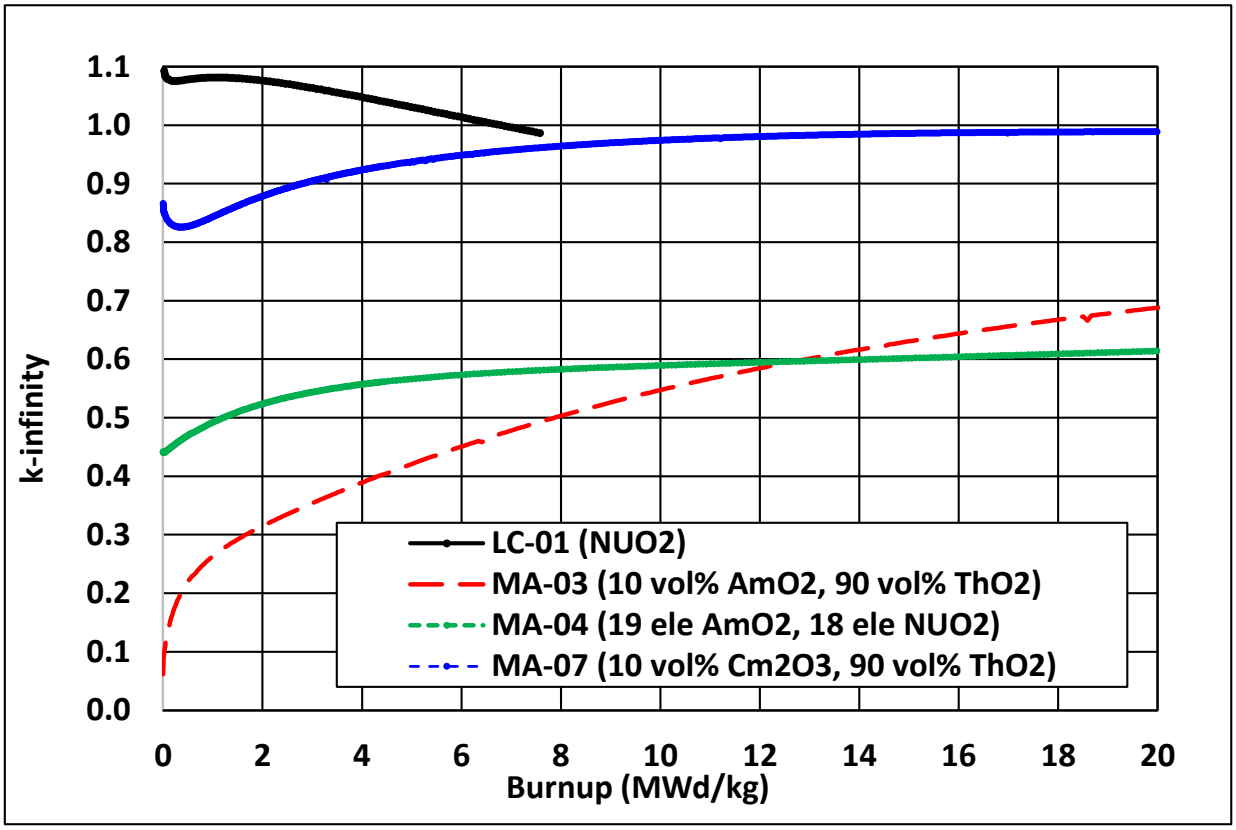


Fig. 3. Neutron Multiplication Factor for the B37 NUO_2 Seed and MA-Containing Blanket Lattices

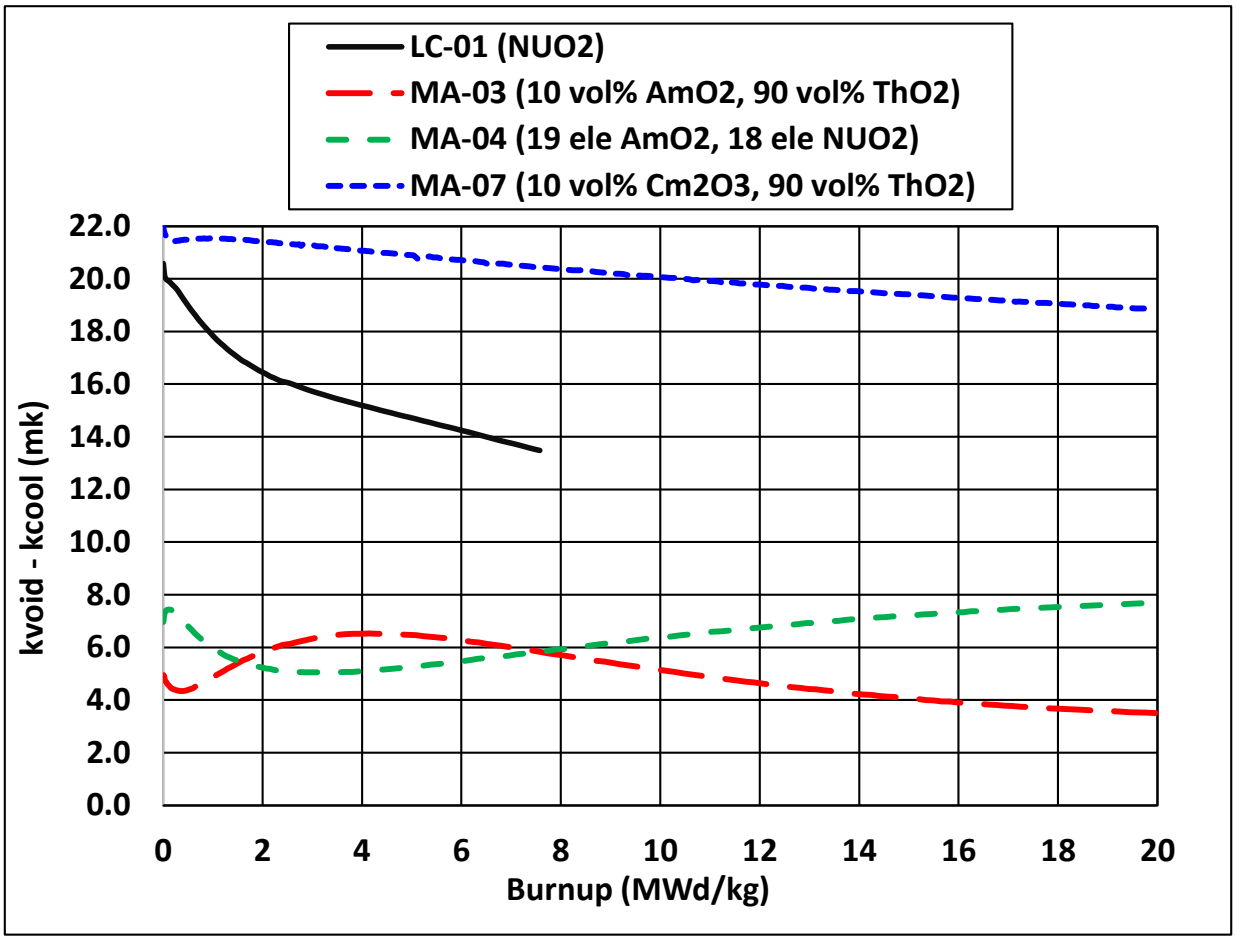


Fig. 4. Change in Neutron Multiplication Factor with Coolant Voiding for the B37 NUO_2 Seed and MA-Containing Blanket Lattices

IV.B. Mass Inventories of Actinides

The mass inventory of key MAs (along with uranium, and the plutonium isotopes) in the seed and blanket bundles at near-zero and near-exit burnup are shown in Table VII. The net change in the mass of U, Np, Pu, Am, Cm, and the total for all MAs plus Pu are also shown in Table VII and Fig. 5. In addition, the net change in the mass of individual Am and Cm isotopes in the different blanket lattices are shown in Table VIII and Table IX respectively.

For the seed fuel (LC-01), data are shown for the content of U, Np, Pu, Am, and Cm at nearly zero burnup (~0.003 MWd/kg), and also at a nominal exit burnup of 6.95 MWd/kg. This exit burnup is typical of what will be expected in core physics calculations to be shown later. For LC-01 at nearly zero burnup, there is essentially zero amounts of Np, Pu, Am, and Cm. Data is also shown for LC-01 with a nominal exit burnup of 7.25 MWd/kg, followed by 10 years of decay. The 7.25-MWd/kg burnup is slightly different (~4.3% higher burnup) than the data from the 6.95-MWd/kg burnup. However, the Np, U, Pu, Am, and Cm found in the LC-01 fuel at 7.25 MWd/kg of burnup is allowed to undergo 10 years of radioactive decay, which will change the amount of certain types of isotopes, since several of the radioactive actinides have relatively short half-lives, ranging from hours to a few decades of years. With shorter half-lives, significant changes in inventories of certain isotopes can occur within a 10-year storage period after removal from a reactor core. Examples of such isotopes include U-237 (6.75 days), Np-238 (2.117 days), Np-239 (2.357 days), Pu-238 (87.7 years), Pu-241 (14.35 years), Pu-243 (4.956 hours), Am-242 (16.02 hours), Am-244 (10.1 hours), Am-244m (26 minutes), Cm-242 (162.8 days), Cm-243 (29.1 years), and Cm-244 (18.1 years). In particular, the amount of Am-241 found in spent fuel after 10 years is largely determined by the decay of Pu-241. Although Am-242 is found in spent fuel immediately after it is removed from the reactor, in addition to Am-242m, it decays relatively quickly to Cm-242 (beta decay) and Pu-242 (electron capture), while Cm-242 decays to Pu-238.

For the blanket bundles containing Am (such as MA-03 and MA-04), the inventory of Am is reduced by 1,100 grams/bundle (~60% of initial inventory) to 1,300 grams/bundle (~14% of initial inventory). However, the reduction in Am is offset by the production of Pu-238 from the alpha decay of Cm-242 (which comes from beta

decay of Am-242), Pu-242 from electron capture on Am-242, and other Pu isotopes. The Am-241 also undergoes alpha decay to produce Np-237. The net impact is that the total for Np+Am+Cm+Pu is reduced by 271 grams/bundle for MA-03 and 142 grams/bundle for MA-04. What is also notable is that the plutonium composition in MA-03 and MA-04 is 62 wt% to 71 wt% Pu-238, and 18 wt% to 19 wt% Pu-242. It is anticipated that the Pu found the spent MA bundles could be recycled in $(Pu,Th)O_2$ bundles for subsequent destruction, although the Pu-242 would be a more persistent problem, requiring neutron irradiation until it is converted into Pu-243, which will decay to Am-243, which then can be irradiated to produce fissile Am-244/Am-244m (with a fission cross section of 2,096 barns/1,561 barns). The Am-244/Am-244m have relatively short half-lives (10.1 hours, 26 minutes), and will decay to Cm-244.

For the blanket bundle containing Cm (MA-07), the inventory of Cm is reduced by ~561 grams/bundle (~30% of initial inventory). However, the reduction in Cm is offset by the production of Pu-240 (from the alpha decay of Cm-244), Pu-241, and Pu-242. The net impact is that the total for Np+Am+Cm+Pu is reduced by 215 grams/bundle for MA-07. At exit burnup, the plutonium composition in MA-07 is ~80 wt% Pu-240, 13 wt% Pu-241, and 5.7 wt% Pu-242. Similarly, it is anticipated the Pu from MA-07 could be recycled in $(Pu,Th)O_2$ bundles and subsequently destroyed by thermal neutron capture and fission. Although the destruction of Am and Cm is offset by the production of Pu, the net consumption of MAs in the blanket bundles at 20 MWd/kg of burnup may be sufficient, as long as the total consumption rate in the blanket bundles (as to be determined by core physics calculations) is comparable or higher than the total production rate in seed bundles in a PT-HWR core.

As mentioned previously, tabulated data for the inventory of the different Cm isotopes in the blanket fuel are shown in Table IX. Upon irradiation to ~20 MWd/kg, it is mainly the lighter-weight isotopes of Cm (Cm-242, Cm-243, Cm-244 and Cm-245) that are produced in the Am-based blanket fuels (MA-03 and MA-04). For the Cm-based blanket fuel (MA-07), there is a significant reduction in the inventories of Cm-244 (-549.6 grams/bundle), follow by Cm-243 (-31.04 grams/bundle) and Cm-242 (-1.42 grams/bundle). There is modest net production of Cm-245 (4.28 grams/bundle), Cm-246 (16.17 grams/bundle) and Cm-247 (0.30 grams/bundle).

For the Am-based blanket fuels, a significant portion of the curium produced will be Cm-242 (50% for MA-03, and 78% for MA-04). The isotope Cm-242 is a strong alpha-emitter, with a half-life of 163 days (turning into

Pu-238). Thus, there will be a build-up of helium gas in Am-based blanket fuels, which could build up pressure within the fuel element. Although optimization of the fuel element design is not the focus of this study, it is anticipated that future studies will need to evaluate the impact of the accumulation and pressurization of helium gas, and several volatile fission products, making use of fuel performance analysis codes, such as ELESTRES [55], [56], modified and adapted for Am and Cm-based fuels. In addition, further design modifications will be required to accommodate the accumulation of helium, while maintaining fuel element structural integrity. Examples of such design modification would be to use a thicker Zircaloy-4 clad to withstand higher pressures, the use of additives (such as graphite powder) to improve the thermal conductivity of the fuel matrix and to reduce fuel and helium temperature and pressure, and also the use of an annular fuel pellet with a central void space to allow the accumulation of helium and fission product gases. Recent irradiation experience with Am-based fuels (15 wt% Am, 85 wt% NU) in (Am, U)$O_{1.94}$ in the High Flux Reactor (HFR) in Petten, Netherlands [35] suggest that it should be possible to irradiate such fuel at high temperatures (1,000°C to 1,200°C) for neutron fluences up to 9.0×10^{21} n/cm^2 (total), and 1.5×10^{21} n/cm^2/s, without significant degradation in the fuel matrix due to helium production.

The net production of the heavier Cm isotopes (such as Cm-248, Cm-249, and Cm-250), which are precursors to creating even heavier elements and isotopes (such as Berkelium (Bk) and Californium (Cf)), are very small, totaling less than 0.03 grams/bundle (or approximately 0.002 % of all the Cm in the blanket bundle). While the production of heavier, highly radioactive isotopes such as Cf-252 ($_{Thalf\text{-}life}$=2.645 years, alpha-emitter) and Cf-250 ($_{Thalf\text{-}life}$=13.08 years, alpha emitter) in the Cm-based fuels are of potential concern due to alpha-heating in the spent fuel, it is expected that the concentration of such isotopes should be relatively low. However, it is recognized that temporary storage of the spent blanket fuel for ~30 years may be necessary and advantageous to allow all the short-lived MAs (such as Cf-250 and Cf-252) to decay to relatively small levels before reprocessing/recycling the MAs for further cycles of irradiation and consumption/destruction.

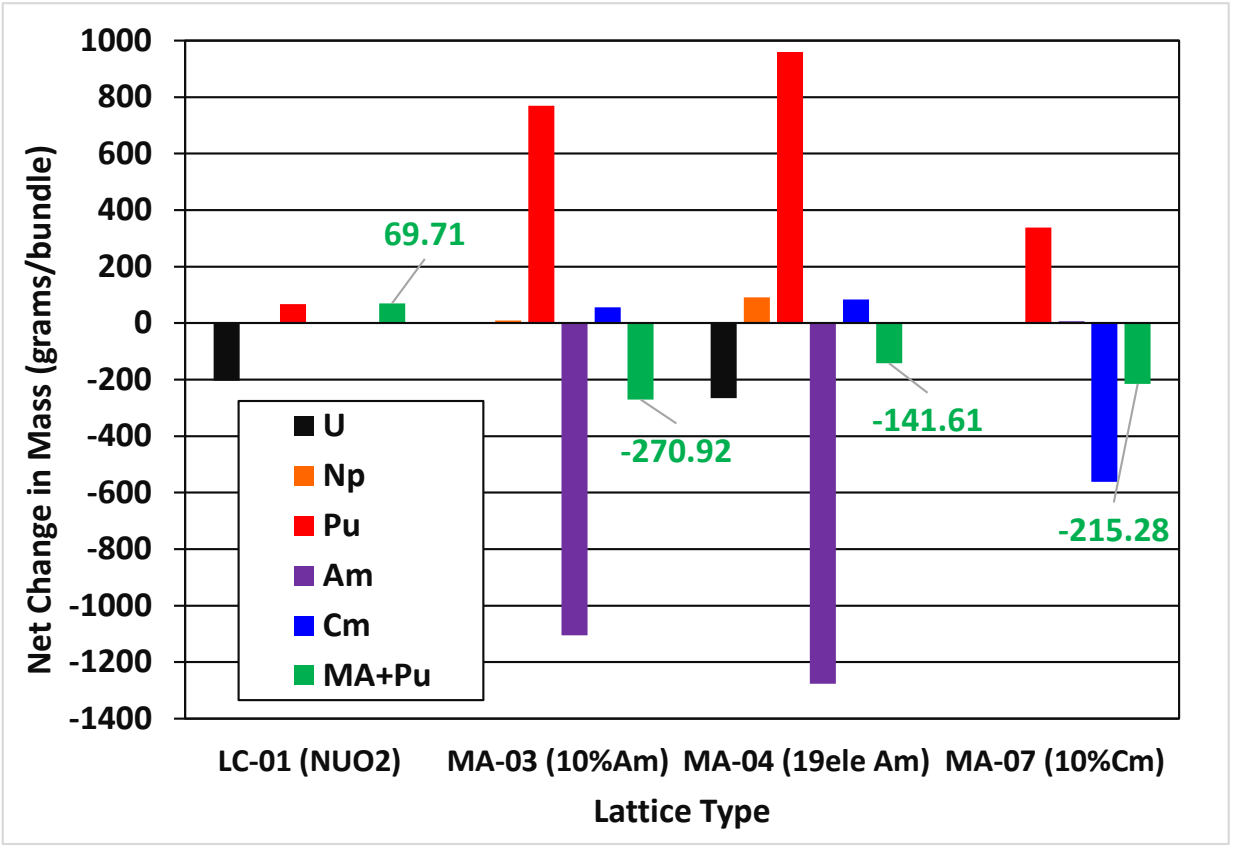


Fig. 5. Net Change in Mass of U, Pu, and MAs in Seed and Blanket Bundle Concepts at Exit Burnup

Table VII Mass (grams/bundle) of MAs, U and Pu Isotopes at Exit Burnup

Lattice Concept	Burnup (MWd/kg)	U	Np	Pu	Am	Cm	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Total (Np+Pu+ Am+Cm)
LC-01*	0.003	$1.89\times10^{+4}$	4.30×10^{-2}	5.29×10^{-4}	8.57×10^{-16}	9.09×10^{-18}	8.39×10^{-13}	5.27×10^{-4}	1.52×10^{-6}	2.93×10^{-10}	2.30×10^{-14}	0.043
	6.947	$1.87\times10^{+4}$	$2.38\times10^{+0}$	$6.73\times10^{+1}$	6.59×10^{-2}	8.58×10^{-3}	5.10×10^{-2}	$4.56\times10^{+1}$	$1.71\times10^{+1}$	$3.61\times10^{+00}$	9.17×10^{-1}	69.758
Delta		-203.985	2.336	67.304	0.066	0.009	0.051	45.596	17.128	3.612	0.917	69.714
Delta (%)		-1.06	N/A**	N/A**	N/A**	N/A**	N/A**	N/A**	N/A**	N/A**	N/A**	N/A**
LC-01+ 10 yr Decay*	3.15×10^{-5}	$1.89\times10^{+4}$	1.19×10^{-5}	1.21×10^{-9}	5.15×10^{-18}	8.77×10^{-18}	8.63×10^{-19}	1.20×10^{-9}	3.33×10^{-12}	6.90×10^{-18}	8.62×10^{-19}	1.19×10^{-5}
	7.247	$1.87\times10^{+4}$	5.15×10^{-1}	$6.96\times10^{+1}$	$1.55\times10^{+00}$	2.17×10^{-3}	6.03×10^{-2}	$4.81\times10^{+1}$	$1.80\times10^{+1}$	$2.37\times10^{+00}$	$1.03\times10^{+00}$	71.687
Delta		-212.016	0.515	69.621	1.549	0.002	0.060	48.111	18.048	2.367	1.034	71.687
Delta (%)		-1.12	N/A**	N/A**	N/A**	N/A**	N/A**	N/A**	N/A**	N/A**	N/A**	N/A**
MA-03	0.001	2.08×10^{-2}	7.71×10^{-4}	4.69×10^{-3}	$1.84\times10^{+3}$	2.76×10^{-2}	3.15×10^{-6}	2.12×10^{-6}	2.21×10^{-8}	2.66×10^{-11}	4.68×10^{-3}	1,836.794
	20.113	$2.94\times10^{+0}$	$9.11\times10^{+0}$	$7.70\times10^{+2}$	$7.31\times10^{+2}$	$5.59\times10^{+1}$	$4.76\times10^{+2}$	$1.07\times10^{+2}$	$3.37\times10^{+1}$	$8.09\times10^{+0}$	$1.45\times10^{+2}$	1,565.875
Delta		2.92	9.104	769.496	-1105.428	55.909	476.106	106.940	33.665	8.093	144.692	-270.919
Delta (%)		N/A**	N/A**	N/A**	-60.18	N/A**	N/A**	N/A**	N/A**	N/A**	N/A**	-14.75
MA-04	0.001	$9.20\times10^{+3}$	1.16×10^{-2}	6.08×10^{-4}	$9.43\times10^{+3}$	3.14×10^{-3}	3.46×10^{-7}	9.43×10^{-5}	8.98×10^{-8}	$5.61E^{-12}$	5.14×10^{-4}	9,432.239
	20.093	$8.93\times10^{+3}$	$9.13\times10^{+1}$	$9.60\times10^{+2}$	$8.16\times10^{+3}$	$8.40\times10^{+1}$	$6.76\times10^{+2}$	$7.59\times10^{+1}$	$2.72\times10^{+1}$	$6.55\times10^{+0}$	$1.74\times10^{+2}$	9,290.631
Delta		-265.295	91.328	959.815	-1276.776	84.024	676.213	75.884	27.238	6.546	173.933	-141.608
Delta (%)		-2.89	N/A**	N/A**	-13.54	N/A**	N/A**	N/A**	N/A**	N/A**	N/A**	-1.50
MA-07	0.001	2.08×10^{-4}	8.49×10^{-15}	1.99×10^{-2}	6.63×10^{-7}	$1.89\times10^{+3}$	7.77×10^{-4}	2.24×10^{-4}	1.89×10^{-2}	2.42×10^{-6}	7.25×10^{-8}	1,894.180
	20.027	$1.82\times10^{+0}$	2.33×10^{-2}	$3.39\times10^{+2}$	$7.21\times10^{+0}$	$1.33\times10^{+3}$	$2.05\times10^{+0}$	8.53×10^{-1}	$2.72\times10^{+2}$	$4.47\times10^{+1}$	$1.92\times10^{+1}$	1,678.904
Delta		1.824	0.023	338.782	7.210	-561.291	2.050	0.853	272.013	44.665	19.201	-215.275
Delta (%)		N/A**	N/A**	N/A**	N/A**	-29.63	N/A**	N/A**	N/A**	N/A**	N/A**	-11.37

* Values for Np, Pu, Am, and Cm for LC-01 differ from that shown in Table VI because of differences in burnup (6.95 MWd/kg vs. 7.25 MWd/kg), and also because of the decay of various actinides (particularly Np-239, Pu-241, Cm-242)

** N/A = Not Applicable. Percentage difference changes are not meaningful when the initial values are close to zero, making the percentage change exceeding 100%.

Table VIII Mass (grams/bundle) of Am Isotopes in Blanket Bundles

Lattice Concept	Burnup (MWd/kg)	Am-241	Am-242	Am-242m	Am-243	Am-244	Am-244m	Total
MA-03	0.001	1.78E+03	5.12E-01	3.17E-01	5.30E+01	3.52E-04	1.77E-03	1836.76
	20.113	6.54E+02	1.25E-01	6.91E+00	7.06E+01	3.03E-04	2.38E-04	731.33
Fraction of Am		8.94E-01	1.71E-04	9.45E-03	9.65E-02	4.15E-07	3.25E-07	1.00E+00
Delta (g/bundle)		-1129.185	-0.387	6.597	17.549	0.000	-0.002	-1105.428
Delta (%)		-63.33	-75.57	2083.11	33.08	-13.71	-86.60	-60.18
MA-04	0.001	9.16E+03	5.61E-02	1.33E+00	2.72E+02	4.62E-05	2.33E-04	9432.22
	20.093	7.84E+03	3.16E-01	4.66E+01	2.68E+02	2.45E-04	1.91E-04	8155.45
Fraction of Am		9.61E-01	3.88E-05	5.72E-03	3.29E-02	3.00E-08	2.35E-08	1.00E+00
Delta (g/bundle)		-1317.944	0.260	45.295	-4.388	0.000	0.000	-1276.776
Delta (%)		-14.39	463.70	3404.20	-1.61	429.51	-17.78	-13.54
MA-07	0.001	1.19E-11	6.65E-16	5.97E-17	6.63E-07	7.01E-13	5.24E-12	6.63E-07
	20.027	4.96E+00	2.42E-03	4.23E-02	2.20E+00	1.23E-05	9.61E-06	7.21
Fraction of Am		6.88E-01	3.36E-04	5.86E-03	3.06E-01	1.70E-06	1.33E-06	1.00E+00
Delta (g/bundle)		4.961	0.002	0.042	2.204	0.000	0.000	7.210
Delta (%)		N/A***	N/A***	N/A***	N/A***	N/A***	N/A***	N/A***

* Note: Am-242 is fissionable, while Am-242m, Am-244 and Am-244m are fissile. Am-244 and Am-244m are short-lived, decaying eventually to Cm-244.

** Note: the scientific notation format 1.0E+4 is equivalent to 1.0×10^4

*** N/A = Not Applicable. Percentage difference changes are not meaningful when the initial values are close to zero, making the percentage change exceeding 100%.

Table IX Mass (grams/bundle) of Cm Isotopes in Blanket Bundles

Lattice Concept	Burnup (MWd/kg)	Cm-241	Cm-242	Cm-243	Cm-244	Cm-245	Cm-246	Cm-247	Cm-248*	Cm-249*	Cm-250*	Total
MA-03	0.001	2.97E-12	2.24E-02	2.09E-07	5.21E-03	6.76E-07	4.43E-11	2.26E-16	8.87E-19	1.87E-19	8.90E-19	2.76E-02
	20.113	4.05E-07	2.78E+01	2.27E+00	2.40E+01	1.50E+00	3.28E-01	5.18E-03	3.08E-04	6.32E-10	1.54E-08	55.94
Fraction of Cm		7.23E-09	4.97E-01	4.05E-02	4.29E-01	2.68E-02	5.87E-03	9.27E-05	5.51E-06	1.13E-11	2.76E-10	1.00E+00
Delta (g/bundle)		4.05E-07	27.803	2.267	24.006	1.499	0.328	5.18E-03	3.08E-04	6.32E-10	1.54E-08	55.909
Delta (%)		N/A***	N/A***	N/A***	N/A***	N/A***	N/A***	N/A***	N/A***	N/A***	N/A***	N/A***
MA-04	0.001	4.55E-14	2.45E-03	9.38E-10	6.84E-04	2.53E-09	5.64E-15	8.81E-19	8.83E-19	1.87E-19	8.90E-19	3.14E-03
	20.093	1.48E-06	6.54E+01	1.22E+00	1.69E+01	4.83E-01	1.59E-02	8.16E-05	1.12E-06	1.38E-09	6.50E-11	84.03
Fraction of Cm		1.76E-08	7.78E-01	1.45E-02	2.02E-01	5.75E-03	1.89E-04	9.71E-07	1.34E-08	1.64E-11	7.73E-13	1.00E+00
Delta (g/bundle)		1.48E-06	65.370	1.221	16.935	0.483	0.016	8.16E-05	1.12E-06	1.38E-09	6.50E-11	84.024
Delta (%)		N/A***	N/A***	N/A***	N/A***	N/A***	N/A***	N/A***	N/A***	N/A***	N/A***	N/A***
MA-07	0.001	5.97E-11	1.86E+00	3.51E+01	1.83E+03	2.20E+01	1.85E+00	3.39E-03	6.33E-05	7.64E-11	1.53E-07	1894.16
	20.027	4.27E-09	4.32E-01	4.02E+00	1.28E+03	2.62E+01	1.80E+01	3.01E-01	2.69E-02	2.80E-08	1.23E-07	1332.87
Fraction of Cm		3.20E-12	3.24E-04	3.02E-03	9.63E-01	1.97E-02	1.35E-02	2.26E-04	2.02E-05	2.10E-11	9.25E-11	1.00E+00
Delta (g/bundle)		4.21E-09	-1.424	-31.040	-549.600	4.279	16.169	0.298	0.027	2.79E-08	-2.98E-08	-561.291
Delta (%)		7,045.08	-76.73	-88.53	-29.98	19.49	874.56	8791.57	42,434.86	36,510.77	-19.49	-29.63

* Note: the heavier isotopes of Cm (Cm-248, Cm-249, and Cm-250), which are pre-cursors to the production of heavier elements and isotopes (such as Bk and Cf), represent less than 0.002% of all the Cm in the fuel.

** Note: the scientific notation format 1.0E+4 is equivalent to 1.0×10^4

*** N/A = Not Applicable. Percentage difference changes are not meaningful when the initial values are close to zero, making the percentage change exceeding 100%.

IV.C. Supplementary Calculations – Cf-252 Production in Cm-Based Fuels

Previous studies of the transmutation of Cm in sodium-cooled fast reactors (SFRs) [57] have suggested that targets made $U_{0.892}$ $Cm_{0.108}O_2$ should be suitable to avoid potential issues with initial heat decay heat power, and neutron source rate (from the spontaneous fission of Cm-244 and Cm-250). Thus, it would be expected that the 10 vol% Cm_2O_3 + 90 vol% ThO_2 blanket fuel bundles proposed in this study should be manageable.

It is recognized, however, that rather than allowing the Cm found in spent PT-HWR NU fuel (as shown previously in Table VI) to decay for 10 years before attempting recycling and transmuting in Cm-based target blanket bundles, it may be easier and more practical to allow the curium to decay for an even longer period of time (perhaps 60 years or more), so that 88% of the initial Cm-244 can decay to Pu-240, while 75% of the initial Cm-243 decays to Pu-239. In addition to alpha-decay, Cm-244 has a significant spontaneous fission rate (1.3×10^{-4} % of all decays). Thus, reducing the amount of Cm-244 by natural decay will reduce both decay heat power, and the neutron source rate from curium. While the absolute amount of Cm left to attempt to recycle and transmute would be reduced by allowing >60 years of decay first, it would also increase the fissile fraction of Cm from ~ 3 wt% $Cm_{fissile}$/Cm to ~ 14 wt% $Cm_{fissile}$/Cm, since the amount of Cm-144 is reduced from nearly 96.7 wt% (see Table VI) to ~84 wt%. Attempting to use such Cm mixed with ThO_2 may require an even smaller fraction of Cm, perhaps ~ 10% × 3/14 ~ 2.1 vol% of Cm_2O_3 mixed with 97.0 vol% ThO_2, if it is desired to maintain approximately the same reactivity level in the blanket fuel.

However, an issue of potential concern when irradiating targets made of (10 vol% Cm_2O_3 + 90 vol% ThO_2), which have a relatively high initial content of curium, in a thermal-spectrum reactor such as a PT-HWR is that the resultant production of Cf-252 in the spent fuel may be orders-of-magnitude higher than what may be produced from the irradiation of Cm-based targets in a fast spectrum reactor. Since Cf-252 can undergo spontaneous fission (3.09% of all decay events), the handling of spent Cm-based fuel from a thermal-spectrum reactor may become more difficult [57], [58]. Thus, key questions are what will be the expected inventory of Cf-252 in the spent Cm-based blanket fuels, and what will be the neutron source rate in the spent fuel due to the spontaneous fission of Cf-252?

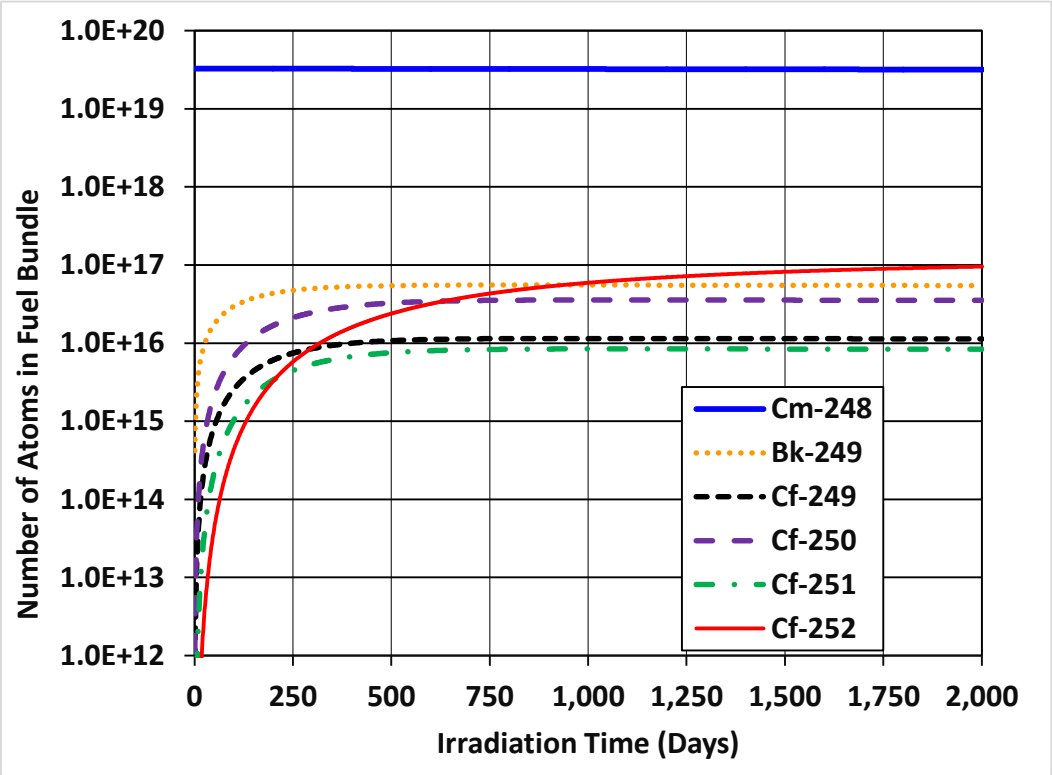
As mentioned previously, it is recognized that the current nuclear data library used with WIMS-AECL in the present study does not include heavier isotopes such as Berkelium (Bk) and Californium (Cf), and thus, output data is not provided on the inventory of the different Bk and Cf isotopes, including Cf-252. However, by using some simplified approximations in the solution of the equations for the production and decay of isotopes (the Bateman equations), it is possible to obtain order-of-magnitude estimates of the amount of Cf-252 that may be found in the spent Cm-based fuel.

Sample calculations for the estimates of the inventories of different Berkelium and Californium isotopes in MA-07 are shown in Fig. 6. To carry out these calculations, it was assumed that the initial inventory of Cm-248 was 0.0135 grams/bundle, which is the mid-point value between fresh fuel and that at an exit burnup of ~20 MWd/kg, as shown previously in Table IX. However, it is recognized that the amount of Cm-248 is continually changing during irradiation due to both its production and consumption, and will not be in a state of equilibrium over the 2000-day period of irradiation of the blanket fuel. For the purpose of making order-of-magnitude estimates of the production of Bk and Cf, the initial amount of Cm-248 was set at the mid-point value.

Based on lattice physics and core physics calculations, it was estimated that the average thermal neutron flux in the fuel in the blanket fuel bundles was ~2.5e+13 n/cm^2/s. Based on refueling rates from core physics calculations to be shown later, it was estimated that the Cm-based blanket fuel bundles (MA-07) reside in the reactor core for ~2000 days. It was assumed that neutron capture on Cm-248 would lead to direct production of Bk-249, since Cm-249 has a relatively short half-life (64.15 minutes). Hence, the inventories of Cm-249 and Cm-250 were not tracked in these simplified calculations. Similarly, it was also assumed that neutron capture on Bk-249 would lead to direct production of Cf-250, since Bk-250 also has a relatively short half-life (3.217 hours), and the inventory of Bk-250 was not tracked. Thus, the inventories were computed for Cm-248, Bk-249, Cf-249, Cf-250, Cf-251, and Cf-252, as shown in Fig. 6. The associated neutron source rate from the spontaneous fission of Cf-252 is shown in Fig. 7. After 2,000 days of irradiation, there are approximately 2.72e+16 atoms (or 1.14e-5 grams, 11.4 micrograms) of Cf-252 in the Cm-based MA-07 fuel bundle, with a neutron source rate of approximately 1.75e+7 n/s. For comparison, the initial neutron source rate in the MA-07 fuel bundle, due to the

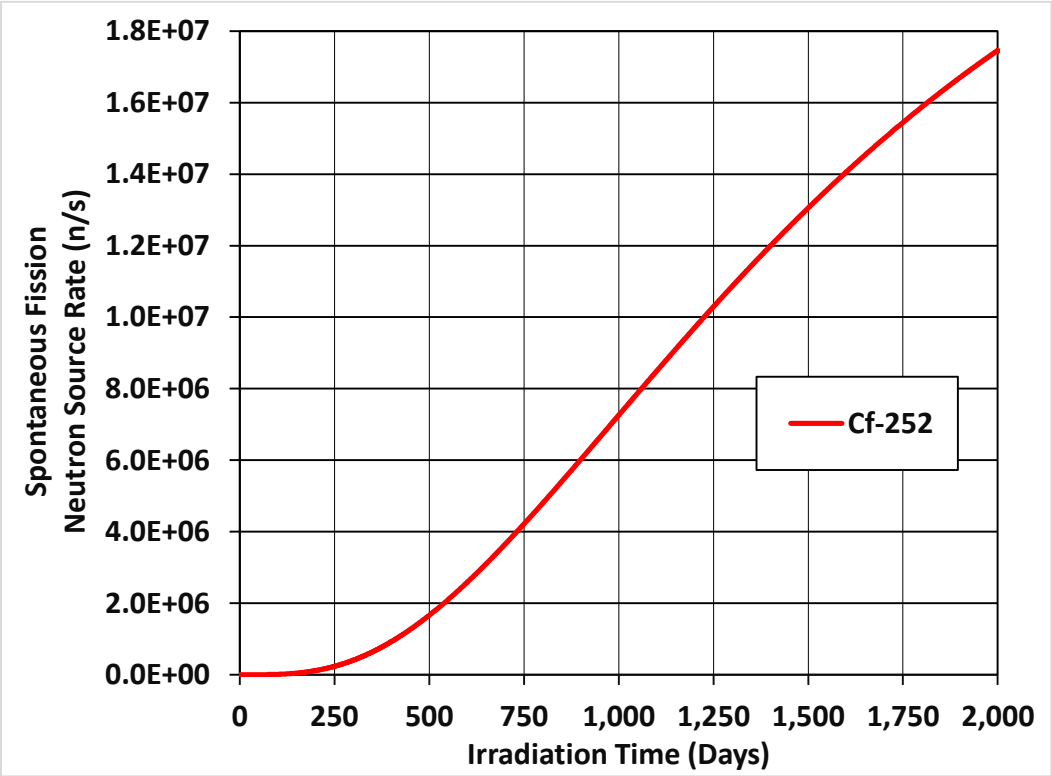
spontaneous fission of Cm-244 ($T_{half\text{-}life}$ = 18.1 years, with 1.3e-6 of all decays resulting in spontaneous fission), is approximately 1.78e+10 n/s per bundle. Thus, the neutron source rate from the spontaneous fission of Cf-252 in the spent fuel is nearly one tenth that of neutron source rate from the spontaneous fission of Cm-244 in the fresh fuel. For a blanket fuel channel with twelve (12) Cm-based fuel bundles, the total initial neutron source rate would be ~ 2.1e+11 n/s, while the total neutron source rate from the spontaneous fission of Cf-252 in spent fuel would be ~2.1e+8 n/s. For comparison, in previous studies of transmutation of Cm-based fuel blanket assemblies in fast reactors [57], the neutron source rate in fresh fuel assemblies made of $U_{0.80}Cm_{0.20}O_2$ was ~2.5e+11 n/s, although it was suggested that reducing the curium content by approximately half to $U_{0.892}Cm_{0.108}O_2$, giving a neutron source rate of ~1.3e+11 n/s would satisfy fuel handling limits due to decay heat. Thus, based on approximate estimates, it would appear that the spontaneous neutron source rate due to Cf-252 in the spent Cm-based fuel will be much smaller than that from the Cm-244. However, future calculations could be carried out in the future to obtain a more rigorous and accurate estimate.

Of course, these results are for a single irradiation pass of Cm-based blanket fuel, and do not consider the impacts of multiple stages of recycling of irradiated Cm-based fuel in a PT-HWR, which has a thermal neutron energy spectrum. From previous studies of recycling of TRU [59], [60], it was noted that there could be issues with significant production of Cf-252 from the multiple recycling of TRU (Np + Pu + Am and Cm) in PWRs in homogeneous cores where all the TRU is mixed with low-enriched uranium (~ 10 wt% TRU, ~90 wt% LEU). In this scenario, after multiple recycles, the inventory and neutron source rate from the spontaneous fission of Cf-252 could be become nearly 300 times that of the spontaneous fission neutron source rate from Cm-244, after 10 years of post-irradiation cooling. Thus, if multiple stages of recycling of Am or Cm-based fuels in blanket fuels in a PT-HWR were to be carried out, it is expected that the trace amounts of Bk and Cf found in the spent fuel would need to be separated after each stage of recycling before new fuel fabrication is carried out to recycle the leftover Am and Cm.



* Note: Irradiation time is ~2000 days for MA-07 Blanket Fuel Bundle, to achieve ~20 MWd/kg burnup in the blanket region. Average thermal neutron flux in the fuel ~2.5e+13 n/cm^2/s. Mass of Cm-248 in bundle ranges from 6.33e-5 grams (at near-zero burnup) to ~0.0269 grams (at near-exit burnup). For calculations to estimate production of Berkelium and Californium, an initial value of ~0.0135 grams (mid-point value) for Cm-248 was assumed.

Fig. 6. Estimate of Inventory of Berkelium and Californium Isotopes from Irradiation of Curium in Cm-Based Fuel Bundle (MA-07).



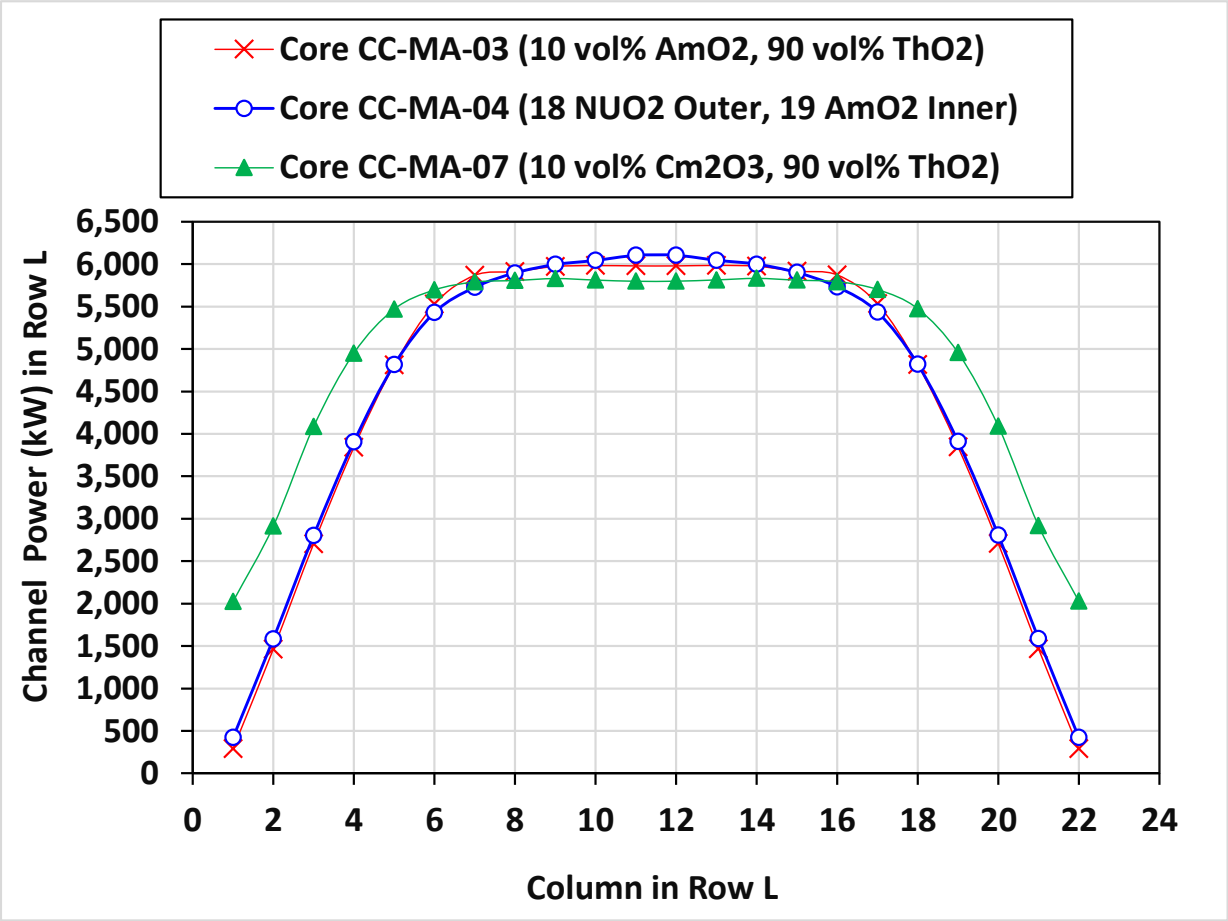
* Note: Half-life of Cf-252 is 2.645 years, and 3.09% of decays result in spontaneous fission.

Fig. 7. Estimate of Neutron Source Rate from the Spontaneous Fission of Cf-252 as a Function of Irradiation Time for Cm-based Blanket Fuel Bundle (MA-07)

V. CORE PHYSICS RESULTS

Three different seed/blanket PT-HWR cores were investigated, using LC-01 (NUO_2 in B37) seed fuel in the inner 320 fuel channels, and one of three different MA blanket bundles in the surrounding 60 blanket channels. The blanket bundles tested in the core analyses included MA-03 (10 vol% AmO_2, 90 vol% ThO_2), MA-04 (18 outer elements with NUO_2, 19 inner elements made of AmO_2), and MA-07 (10 vol% Cm_2O_3, 90 vol% ThO_2).

The results for the reactor power level, burnup, and refueling rates in the different seed-blanket cores are shown in Table X. For the cores with AmO_2-based blanket bundles, the total reactor power was reduced to ~63% of the nominal (2,061 MWth) power level, while the Cm_2O_3-based blanket bundles required a reduction to ~80%. This reduction occurred due to the low reactivity and power level of the blanket bundles, particularly for CC-MA-03 (290 kW channel power) and CC-MA-04 (422 kW), as shown in Fig. 8 The more reactive MA-07 fuel had a much higher power level (2000 kW). The seed burnup in these cores ranged from 6.8 MWd/kg (with AmO_2 blankets) to 7.4 MWd/kg (for $Cm2O_3$ blankets). To maintain core criticality (with k_{eff} ~1.015), the seed burnup had to be reduced to compensate for the low reactivity of the MA-03 and MA-04 blanket bundles. Typically, 10 to 11 seed bundles are refueled per day, while 0.05 to 0.4 blanket bundles must be replaced per day (or 1 bundle every 2.5 to 20 days).



* Row L in Heterogeneous Seed-Blanket PT-HWR Core is illustrated previously in Fig. 1.

** Here, it is seen that the blanket channels are at power levels less than half of those of the high-power seed channels

Fig. 8. Radial Power Distribution in PT-HWR Seed-Blanket Cores with MA-containing Fuel Bundles in the Outer 60 Blanket Channels

Table X Reactor Power, Burnup, and Refueling Rate for Seed and Blanket Bundles in PT-HWR

Core Case	**MA Bundle Matrix**	**Reactor Power (% FP)**	**Seed Burnup (MWd/kg)**	**Seed Refueling Rate (Bundles/day)**	**Blanket Burnup (MWd/kg)**	**Blanket Refueling Rate (Bundles/day)**
CC-MA-03	10 vol% AmO_2, 90 vol% ThO_2	63.0%	6.765	10.477	20.122	0.0526
CC-MA-04	Outer 18 Elements: NUO_2 Inner 19 Elements: AmO_2	63.9%	6.764	10.571	19.574	0.0767
CC-MA-07	10 vol% Cm_2O_3, 90 vol% ThO_2	80.0%	7.414	11.390	18.966	0.375

Using the mass inventory data based on the lattice physics data, and the refueling rates based on core physics calculations, estimates were obtained for the net consumption rate of MAs in the 60 blanket channels and the net production rate of MAs in the 320 seed channels. The results are shown in Table XI. The estimate of production of MAs in the seed is more conservative in that it uses the data found in Table VI, which involves a burnup of ~7.25 MWd/kg, and also accounts for 10 years of decay. This approach is considered a good approximation, given that spent fuel is likely to be stored for at least 10 years before attempting to extract and recycle MAs. The 10-year decay of spent fuel will lead to larger amounts of Am (since the Pu-241 will decay to Am-241), but also lesser amounts of Cm (since the Cm-242 will decay to Pu-238). The consumption rate of americium in the blankets well exceeds the production rate in the seed region by a factor of 3.6 (for MA-03) to 6.0 (for MA-4). Hence, it may only be necessary to use 10 to 17 dedicated blanket channels to achieve net zero production of Am. For curium, the consumption rate far exceeds the production rate, by a factor of 8,500, and only one dedicated channel (or perhaps even just one bundle) would be needed to achieve net zero production. If the production of other actinides in the blanket channels are taken into account, such as Np, and also the plutonium, which has higher fractions of Pu-238 (up to 70 wt% for CC-MA-04), Pu-240 (up to 80 wt% for CC-MA-07) and Pu-242 (up to 19 wt% for CC-MA-03), then 90 to 121 channels of AmO_2-based blanket fuel, and 18 channels of CmO_2-based blanket fuel would be needed to achieve net zero production of MAs (if the Pu in the spent blanket bundles were treated as MAs as well). These estimates neglect the production of Pu in the seed fuel, which has a relatively low content of Pu-238 (~0.1 wt%) and Pu-242 (~1.5 wt%), and which could be recycled and consumed in $(Pu,Th)O_2$ fuel bundles, as demonstrated in previous studies [4].

The approximate estimate of the number blanket channels required to achieve net-zero production of Am, Cm, or minor actinides (Np+Am+Cm) plus blanket-based plutonium in the seed-blanket core is simply the number of blanket channels (60) multiplied by the ratio of seed production to blanket consumption. For example, for core CC-MA-03, which is intended for destroying Am, the number of channels that may be required ~60 ×5.925/21.223 = 16.75 blanket channels. To balance the net consumption of Np+Am+Cm+Pu in the blanket with production of Np+Am+Cm in the seed, the number of blanket channels that may be required ~ 60 × 7.901 / 5.201

~ 91.15 blanket channels. However, this estimate is a highly simplified approximation. Since the seed-blanket core is fixed with 380 channels total, the number of seed channels would need to be adjusted from 320 channels to accommodate the change in the number of blanket channels. Thus, updated core physics calculations would be required to analyze the different core configuration, since there would also be changes in the core power distributions and refueling rates in both the seed and blanket regions. The actual number of blanket channels and seed channels required to achieve net zero production may be different from the initial estimate provided, and thus a number of iterations on the seed/blanket core configuration and associated refueling rates for the seed region may be required.

Ultimately, an important goal is to maximize the destruction of Am and/or Cm in the blanket channels through fission, but without producing Np, or Pu in blanket channels through neutron capture and decay processes. Any Np or Pu found in the spent blanket fuel becomes a new liability to deal with. While the Np (many of its isotopes which have high capture and/or fission cross sections) could be recycled into irradiation targets, the spent Pu found in blanket fuel will be somewhat more problematic, given that it has a higher fraction of non-fissile Pu isotopes, with Pu-242 being the most problematic, given its long half-life and much smaller cross sections for neutron capture and fission (see Table I). In contrast, the Pu found in spent seed fuel has a relatively high fraction of fissile isotopes, and could be recycled into new fuel for subsequent consumption/destruction in another reactor.

The lower net consumption rate of MAs (Np + Am + Cm) plus Pu in the blanket channels for MA-03 (-5.2 kg/year) and MA-04 (-4.0 kg/year) are due to the low channel power levels (290 kW to 422 kW). It may be possible to significantly increase these channel power levels (and neutron flux levels), either by increasing the reactivity of the AmO_2-based blanket lattices, or by increasing the power levels of the NUO_2 seed channels adjacent to blanket. The former could be achieved by using slightly enriched uranium (for example, 1.5 wt% U-235/U) in the outer 18 elements of MA-04. The reactivity and power level of MA-03 could be boosted by mixing in fissile fuel into the $(Am,Th)O_2$, perhaps using recycled U-233 (up to 1.4 wt% U-233/(U+Th+Am), or by mixing the Cm with the Am (since the Cm-based fuel is highly reactive), or by mixing in high-assay low

enriched uranium (HALEU, 19.75 wt% U-235/U), with ~10 vol% UO_2, 10 vol% AmO_2 and 80 vol% ThO_2. The potential tradeoff is that there would be more production of Pu and other MAs from neutron capture on the U-238 found in the HALEU. Alternatively, the MA-03 fuel could be pushed to higher burnup levels, which will permit higher reactivity levels as more fissile Am-242/Am-242m and U-233 are bred into the fuel. The power level in the seed channels adjacent to the blanket could be increased simply by reducing their respective exit burnup; the exit burnup of the other seed channels would also need to be adjusted to maintain a flat radial power distribution, and also to ensure that the core reactivity was k_{eff}=1.015.

Table XI Net Production of MAs as Determined from Core Calculations

Core Case	Nuclides of Interest	Blanket Production (kg/yr))	Seed Production* (kg/yr)	Core Net (kg/yr)	Ratio Net Core to Seed Production	Ratio of Blanket to Seed Production	Expected Number of Blanket Channels Needed***
CC-MA-03	Np-Total	0.175	1.968	2.143	1.09	0.089	
	Pu-Total	14.774	266.232	281.006	1.06	0.055	
	Pu-NonFiss*	12.565	73.200	85.765	1.17	0.172	
	Am-Total	-21.223	5.925	-15.298	-2.58	-3.582	16.75
	Cm-Total	1.073	0.008	1.082	130.48	129.480	
	A-Total**	-5.201	7.901	2.700	0.34	-0.658	91.15
CC-MA-04	Np-Total	2.555	1.986	4.541	2.29	1.287	
	Pu-Total	26.853	268.625	295.477	1.10	0.100	
	Pu-NonFiss*	24.547	73.858	98.404	1.33	0.332	
	Am-Total	-35.721	5.978	-29.742	-4.97	-5.975	10.04
	Cm-Total	2.351	0.008	2.359	282.04	281.041	
	A-Total**	-3.962	7.972	4.011	0.50	-0.497	120.74
CC-MA-07	Np-Total	0.003	2.139	2.143	1.00	0.001	
	Pu-Total	46.362	289.442	335.803	1.16	0.160	
	Pu-NonFiss*	40.133	79.581	119.714	1.50	0.504	
	Am-Total	0.987	6.442	7.428	1.15	0.153	
	Cm-Total	-76.811	0.009	-76.802	-8521.54	-8522.541	0.01
	A-Total**	-29.460	8.590	-20.870	-2.43	-3.429	17.50

* Non-fissile plutonium includes sum of Pu-238, Pu-240 and Pu-242.

** The "A-Total" for the blanket includes Np + Pu + Am + Cm (minor actinides + plutonium), since the Pu found in the blanket bundles contains higher amounts of non-fissile isotopes (which are non-desirable), and because the ultimate goal is to destroy Am and Cm, but without producing Np and Pu. The "A-Total" for the seed fuel (made of NUO_2) includes Np + Am + Cm, but not Pu, because the Pu in the spent seed fuel has higher relative amounts of fissile isotopes (which is more desirable), and can be separated and recycled in a different reactor.

*** The number of blanket channels required to achieve net zero production, where the consumption rate in the blanket is balanced by the consumption rate in the seed, is a highly simplified approximate estimate. New core physics calculations with a modified seed-blanket core would be required to refine this estimate.

VI. CONCLUSIONS AND FUTURE WORK

Lattice physics and core physics studies have been carried out to investigate the feasibility of destroying Am and Cm using special target fuel bundles in blanket fuel channels in a seed-blanket PT-HWR core fueled primarily with natural uranium. Results indicate that it should be feasible to achieve net zero production of Am using 10 to 16 dedicated blanket channels containing Am-based target bundles, while only one dedicated blanket fuel bundle would be required for achieving net zero production of Cm. To achieve net zero production of MAs, more blanket channels (from 90 to 120) would be needed, but it is expected that the plutonium (and perhaps also the neptunium) found in spent MA-based fuel could be recycled in $(Pu,Th)O_2$ fuel bundles. Modifications to the blanket and seed exit burnup and refueling rates, or modifications to the blanket bundles could potentially be made to boost the power and neutron flux levels in the blanket bundles to increase the consumption rates of Am and the net consumption rate of MAs.

Although calculations have demonstrated that only a single fuel bundle made of (10 vol% Cm_2O_3 + 90 vol% ThO_2) may be needed to achieve net-zero production of Cm in a PT-HWR, the potential issues with higher concentrations of Cf-252 (which can undergo spontaneous fission) in spent Cm-based fuel suggest that it may be more practical to use blanket bundles made of Am mixed with trace amounts of Cm for the purpose of destroying Am and Cm simultaneously, rather than using Cm-only based fuel bundles.

The potential to achieve net zero production of Am and Cm in a single thermal-spectrum reactor, such as a PT-HWR, could help eliminate the need to build and qualify a DGR capable of storing minor actinides for a long time (> 1 million years). At the very least, the size and/or number of DGRs required for storing radioactive waste could be reduced significantly. Thus, destroying Am and Cm in PT-HWRs could be regarded as a viable solution to perceived problem of nuclear waste, and may help improve public acceptance of the use of nuclear energy. In addition, it may be possible to apply a similar approach for destroying MAs in other Gen-III+/Gen-IV/SMR technologies.

As mentioned previously, it is expected that the impact of the outer 60 blanket channels containing MA-based blanket fuels on the core-averaged reactivity coefficients and kinetics data should be relatively small. The reactor

physics behavior will be dominated by the inner seed region of 320 channels containing 37-element NUO_2 fuel bundles. However, for future studies, it would be useful to carry out full-core reactor physics evaluations of reactivity coefficients and kinetics data for the various heterogeneous seed-blanket cores.

In the next stage of concept development, it will be useful to evaluate the production rate of helium and fission product gases in the Am-based and Cm-based blanket fuels, along with fuel performance using updated analysis codes such as ELESTRES [55], [56]. It may also be necessary to evaluate alternative fuel element design concepts and fuel matrix materials that would be able to accommodate the accumulation of helium produced form the decay of short-lived MAs, such as Cm-242. Future studies could also involve thermal-hydraulic and heat transfer calculations to evaluate the temperatures of MA-based fuels in heterogeneous seed-blanket cores.

Although it is expected that the production of heavier elements and isotopes (such as isotopes of Bk and Cf) from blanket lattices containing Am or Cm will be relatively low (since up to eight steps of neutron absorption are required to transition from Cm-244 to Cf-252), it would be useful for future lattice physics studies with WIMS-AECL to use updated nuclear data libraries that include several of the heavier elements and isotopes (Bk, Cf, Es and Fm), and/or to use an alternative lattice physics code and associated nuclear data set, such as SERPENT [61].

The current studies have considered a single stage of recycling. However, future studies should also consider analyzing multiple stages of recycling of blanket fuel, to evaluate the potential impact on the buildup of Bk and Cf, as observed in other studies [59], [60].

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