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Studies of Advanced Fuel Cycles in Indian Pressurized Heavy Water Reactors and Advanced Heavy Water Reactor

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In order to conserve the natural uranium resources, a study has been performed to use the depleted uranium and plutonium discharged from the PHWRs. A new fuel cluster design, MOX-888, has been proposed which contains 0.8 wt % PuO_2 mixed with depleted uranium having 0.25 wt % U^{235}. It has been found that it is possible to use MOX-888 fuel clusters in the outer 190 channels in conjunction with natural UO_2 bundles in the central 116 channels of the reactor without making any changes in the other hardware of the reactor. The average discharge burnup of fuel (Natural UO_2 and MOX-888) can be improved to more than 10000 MWD/T from the present value of 6700 MWD/T resulting in the substantial saving of natural uranium per year. The paper discusses the physics studies pertaining to this fuel cluster. It also compares the worth of various control devices including primary and secondary shut down systems for the proposed core with present core having all natural uranium fuel clusters.

An Advanced Heavy Water Reactor is being designed making use of thorium fuel. The design envisages the recycling of ^{233}U produced due to capture of neutrons in the thorium. ^{234}U is also produced in this process. The paper discusses the results of multicycle studies performed to estimate the increase in the concentration of ^{234}U and associated burnup penalty in the reprocessed uranium in each recycle.

The paper also compares the production of minor actinides produced in the discharged fuel, their associated radioactivity and toxicity for PHWR, AHWR and PWR fuels. Comparison shows that the production of minor actinides per unit energy is less in AHWR fuel than those of PWR fuel, but it is comparable to the case of PHWR fuel. This is mainly due to presence of (Th-Pu) MOX pins in the AHWR fuel cluster.

KEYWORDS: PWR, PHWR, AHWR, MOX fuel, void reactivity, BOC, EOC, Burnup, Burnable Poison, Minor Actinides, Radioactive Decay, Toxicity

1. Introduction

Presently, Pressurized Heavy Water Reactors (PHWRs) form the most important part of Indian nuclear power programme. There are fourteen operating units of 220 MWe. Two 540 MWe PHWRs and four 220 MWe PHWRs are under construction and few more units have been planned. These reactors make use of natural uranium as fuel. In order to conserve the natural uranium resources, a study was made to use mixed oxide fuel in which 0.4 wt % PuO_2 was mixed with natural UO_2 in the central 7 fuel pins of the 19-rod fuel cluster [1-2]. There is a modest increase in the burnup of these fuel clusters from the present value of 6700 MWD/T to approximately 10500 MWD/T. It is proposed to irradiate 50 such fuel clusters in one of the 220 MWe PHWR to acquire first hand experience. In order to further conserve the

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natural uranium resources, we have now studied the use of depleted uranium discharged from the PHWRs. A new fuel cluster design, MOX-888, has been proposed which contains 0.8 wt % PuO_2 mixed with depleted uranium having 0.25 wt % U^{235}. The paper discusses the physics studies pertaining to this fuel cluster. It also compares the worth of various control devices including primary and secondary shut down systems for the proposed core with present core having all natural uranium fuel clusters.

There are large reserves of thorium in India. A high priority is accorded to the development of fuel cycle based on thorium fuel. An Advanced Heavy Water Reactor (AHWR) of 300 MWe power is being designed making use of thorium fuel. The design envisages the recycling of ^{233}U produced due to capture of neutrons in the thorium. ^{234}U is also produced in this process. Recently, multicycle studies were performed to estimate the increase in the concentration of ^{234}U and associated burnup penalty in the reprocessed uranium in each recycle. The paper discusses these aspects of the thorium fuel cycle.

The studies have also been performed towards the backend of the fuel cycles using the computer code ORIGEN-2 [3]. The paper also compares the production of minor actinides produced in the discharged fuel, their associated radioactivity and toxicity for PHWR, AHWR and PWR fuels.

2. Use of Depleted Uranium-Plutonium Oxide Fuel in 220 MWe PHWRs

Presently 14 PHWRs of 220 MWe are under operation in India and out of them three reactors have completed about 10 Full Power Years of operation and another 4 reactors are approaching the landmark very fast. Every full power year of operation of each reactor discharges about 40tonnes of irradiated fuel in the spent fuel bay with average discharge burnup of about 6700 MWD/T. The piling inventory of discharged fuel is a matter of concern because of limited storage capacity for discharged fuel. To maintain sufficient storage capacity of spent fuel storage bay for discharged fuel, one of the possible alternate is to reprocess the discharged fuel and recycle it back to PHWR as natural/depleted Uranium-Plutonium MOX or Thorium-Plutonium MOX. This recycling will not only help to conserve the limited Uranium reserves but also reduce the volume of the radioactive waste. For this purpose, following types of MOX fuel were considered in such a way that the use of MOX fuel in Indian PHWRs does not require any changes in other hardware of the reactor.

1. MOX-7: The central 7 pencils contain 0.4% PuO_2 mixed with Natural UO_2 and remaining 12 pencils (outer ring) contains only Natural UO_2.
2. MOX-97: The central 7 pencils contain 0.9% PuO_2 mixed with Depleted UO_2 having 0.25% of U-235 and remaining 12 pencils (outer ring) contain 0.7% PuO_2 mixed with Depleted UO_2 having only 0.25% of U-235.
3. MOX-888: All 19 pencils contain 0.8% PuO_2 mixed with Depleted UO_2 having 0.25% of U-235.
4. MOX-TH24: All 19 pencils contain 2.4% PuO_2 mixed with Thorium Oxide (ThO_2).
5. MOX-TH20: All 19 pencils contain 2.0% PuO_2 mixed with Thorium Oxide (ThO_2).

The MOX-7 has already been studied in detail and its trial irradiation (50 bundles) in one reactor is under progress presently. The concentration of fissile material added to Thorium Oxide to give MOX-TH24 and MOX-TH20 is such that it gives average discharge burnup equal to about 15000 MWD/T and 8000 MWD/T, respectively.

2.1 Core Calculations

Natural UO_2 fuel bundle of PHWR is 19 rods cluster and discharged at about 6700 MWD/T which contains about 0.25 % U-235. The reprocessed fuel (UO_2) of PHWR with 0.25% U-235 is termed as depleted UO_2. This depleted UO_2 can be mixed with different concentration of PuO_2. The concentration of fissile material is chosen in such a way that the bundle power limits of the MOX fuel is comparable to the Natural UO_2 fuel. The linear heat rating of all the five MOX fuel clusters and natural UO_2 fuel cluster has been assumed to be same. The isotopic composition of Plutonium (70.0%, 24.0%, 5.0%, and 1.0% of

Pu-239, Pu-240, Pu-241 and Pu-242 respectively) has been also taken corresponding to burnup of 6700 MWD/T of Natural UO_2 fuel.

The transport theory code CLUB [4-5] was used to perform the lattice calculations. Two-group cross sections generated were used to perform core calculation. The normalized ring power distribution of fuel cluster was used to calculate the bundle power limits. The comparison of their bundle power limits with Natural UO_2 fuel is given Table 1. From the bundle power limit point of view, MOX-97 is the best among all others. MOX-97 fuel cluster requires differential concentration of PuO_2 with Depleted Uranium in different rings. Hence MOX-97 requires high precision during manufacture of the fuel on commercial scale. Initially the bundle power limit for MOX-7 is lower than Natural UO_2. Therefore, it demands the loading of MOX-7 in relatively low flux region initially. The bundle power limits for MOX-TH24 & MOX-TH20 are further lower as compared to Natural UO_2 and these limits are extended up to a burnup of 6000 MWD/T. The bundle power limits for MOX-888 are comparable to Natural UO_2 fuel, hence it can be directly used like a natural UO_2 fuel bundle.

Table-1 Bundle power limits for 19-rod PHWR cluster (kw)

Burnup MWD/T	Natural	MOX-7	MOX-97	MOX-888	MOX-TH24	MOX-TH20
0	462	431	482	459	419	426
2000	462	474	487	464	427	435
4000	453	488	481	460	436	444
6000	432	465	458	440	444	452
8000	409	434	432	417	451	456
10000	380	400	400	387	456	459
12000	359	376	377	367	459	459
15000	337	350	353	344	460	457

Equilibrium core burnup optimisation study for all the above mentioned five fuel clusters with two burnup zones and multi-bundle shift schemes was carried out by using computer code TAQUIL [6]. The two, four, six and eight bundle shift schemes were studied for all types of MOX fuel clusters. The results of the optimisation study are given in Table 2. For all types of MOX fuel the smaller is the bundle shift lesser is the average discharge burnup, which is contrary to the trend observed with natural UO_2 fuel. This is because the total fissile contents of MOX are higher than the Natural UO_2 fuel. As a result smaller bundle shift of MOX fuel leads to more axial flux peaking at the ends of a channel with respect to natural UO_2 fuel. The peaking at end leads to more neutron leakage, thus there is more loss of burnup with smaller bundle shift scheme.

Table-2 Average discharge burnup for an equilibrium core with multi bundle shift (MWD/T)

Fuel Type	2 Bundle Shift	4 Bundle Shift	6 Bundle Shift	8 Bundle Shift
MOX-7	10891	11004	11037	11084
MOX-97	11409	11696	12140	12298
MOX-888	11887	12225	12725	12947
MOX-TH24	14578	14956	15632	15735
MOX-TH20	7700	7840	8091	8037
Natural UO_2	6553	6464	6472	6363

The fuel requirement for each type of cluster is given in Table 3. The daily feed rates for MOX-TH20 are comparable to Natural UO_2. Whereas the feed rates for all other MOX fuels is almost half as compared to Natural UO_2. Thus the use of MOX fuel may result in fuel saving of about 50%.

Table-3 Feed Rates and Annual Fuel Requirements for 4 Bundle ShiftSscheme (Annual CF=85%)

Fuel Type	Fuel Bundles per FPD				Components of Oxide Fuel in tonnes per year				
	2BSS	4BSS	6BSS	8BSS	N-UO_2	ThO_2	D-UO_2	PuO_2	Total
MOX-7	5.5	5.4	5.4	5.4	25.76	Nil	Nil	0.04	25.8
MOX-97	5.2	5.1	4.9	4.8	Nil	Nil	24.21	0.19	24.4
MOX-888	5.0	4.9	4.7	4.6	Nil	Nil	23.21	0.19	23.4
MOX-TH24	5.2	5.1	4.9	4.8	Nil	23.80	Nil	0.60	24.4
MOX-TH20	9.9	9.7	9.5	9.5	Nil	45.37	Nil	0.93	46.3
Natural UO_2	9.1	9.2	9.2	9.4	44.0	Nil	Nil	Nil	44.0

BSS = Bundles Shift Scheme ; N-UO_2 = Natural UO_2 ; D-UO_2 = Depleted UO_2

The time averaged burnup distribution generated in the equilibrium core optimisation study was used to generate the instantaneous power distribution for each type of cluster to estimate the worth of shut down systems and regulating devices. The reactivity worths of protective devices were compared with Natural UO_2 fuel cluster and given in Table 4. The requirement of PuO_2 is relatively higher for MOX-TH24 and MOX-TH20 and the reactivity worth of shut down system in these two cases are relatively low. Hence it was not considered for further studies in the present context. Therefore, these two fuels require to be studied separately under slightly different scheme by using reshuffling option to make the best use of Thorium characteristics at higher burnup.

The poison tubes of Secondary Shut down System (SSS) are located in the center of the core. Therefore the worth of 3 banks of SSS (3-SSS) is maximum for 6 bundles shift scheme (6-BSS). Whereas the 4 & 2 bundles shift scheme results in more axial flux peaking towards the ends of the channel, therefore the worth of 3-SSS is relatively small. Also as the net fissile contents are more for all the MOX fuels as compared to Natural UO_2, the worth of shut down devices are lower than Natural UO_2.

Table-4 Reactivity worth (mk) of shut down systems in equilibrium state of core

Fuel Type	13-PSS				3-SSS			
	2BSS	4BSS	6BSS	8BSS	2BSS	4BSS	6BSS	8BSS
MOX-7	26.9	27.4	26.4	27.7	25.3	25.6	30.8	23.8
MOX-97	25.4	26.3	25.0	26.7	17.6	20.8	29.1	22.5
MOX-888	24.9	25.9	24.6	26.5	16.9	20.2	28.8	22.3
MOX-TH24	20.7	21.6	20.9	22.2	14.2	17.2	24.7	19.1
MOX-TH20	21.4	21.9	21.3	22.2	17.7	19.5	25.0	17.6
Natural UO_2	28.3	28.3	28.0	29.0	29.9	28.5	30.9	25.5

13-PSS = 13 rods of Primary Shut down System 3-SSS = 3 banks of Secondary Shut down System

2.2 Selection Of MOX Loading Pattern in the Core

The equilibrium core burnup optimisation study and worth estimation of shut down devices reveal that full core with only MOX fuel shall not be acceptable due to high bundle power and channel power and reduced shut down margin in the shut down state of the reactor. Hence a suitable combination was evolved to meet these requirements so that MOX may be accepted as fuel of the future for Indian PHWRs without any hardware changes. The pattern thus evolved uses 116 central channels loaded with Natural UO_2 adopting 8 bundles shift scheme and remaining 190 channels loaded with MOX-888 fuel adopting only 4 bundles shift scheme. The average discharge burnup of Natural UO_2 achieved is 9400 MWD/T and for the MOX-888 the average discharge burnup is 10680 MWD/T. Thus the core average discharge burnup is estimated to be about 10105 MWD/T. In the study, it has been assumed that the corner and central adjusters are in their nominal configuration (Adjuster Rods fully IN and Regulating Rods 80% IN). Presently, Indian PHWRs operate with partial adjusters in the core, the above pattern was re-optimized with core average discharge burnup of 10573 MWD/T (Natural UO_2 burnup was estimated to be 9900 MWD/T and MOX-888 burnup was estimated to be 11130 MWD/T).

The salient features of this pattern are given below.

a) The total requirement of fuel in a PHWR can be reduced to about 30tonnes annually, which is about 2/3 of the present requirement.
b) The requirement of Natural UO_2 can be reduced to about 15tonnes annually, which is about 1/3 of the present requirement of Natural UO_2. Almost same amount (15tonnes) of MOX-888 has been used to make up the total fuel (Natural UO_2 & MOX-888) requirement of about 30tonnes annually.
c) The average refueling rate has been estimated to about 1.1 channels per FPD. Hence there is no extra load on fuelling machine even by adopting the smaller bundle shift scheme.
d) The worth of 13 rods of Primary Shut down System (13 PSS) is lower by about 3 mk, where as the worth of other important devices like 3 banks of Secondary Shut down System (3-SSS), Shim Rods (SR), Adjuster Rods (AR), Regulating Rods (RR) are comparable to all Natural UO_2 core.
e) However with the present operational policy of only 4 mk with adjusters, with Adjuster Rods (AR) fully withdrawn, the worth of PSS and SSS will become comparable to Natural UO_2 case.

3. Burnup Penalty of ^{234}U in Successive Recycling of AHWR fuel

There are large reserves of thorium in India. A high priority is accorded to the development of fuel cycle based on thorium fuel. An Advanced Heavy Water Reactor (AHWR) is being designed making use of thorium fuel. AHWR is a 300 MWe, vertical, pressure tube type reactor cooled by boiling light water and moderated by heavy water [7]. The reactor is fuelled with 54-rod fuel cluster (D5 cluster). The fuel uses 3.0% ^{233}U in the inner 12 $(Th,U)O_2$ pins, 3.75% ^{233}U in the middle 18 $(Th,U)O_2$ pins and 3.0% plutonium in the outer 24 $(Th,Pu)O_2$ pins [8]. A central displacer rod with about 3% dysprosium and a void tube in the lattice corner have been provided, to make the spectrum harder and make the void reactivity negative. The reactor consists of 500 lattice positions with a pitch of 27 cm, with 452 occupied by the fuel and the rest by control and safety devices. The core is optimised to give an average exit burnup of about 24000 MWD/T for the reference fuel cluster design mentioned above. The design envisages the recycling of ^{233}U produced due to capture of neutrons in the thorium. ^{234}U is also produced in this process. The results of multicycle studies performed to estimate the increase in the concentration of ^{234}U and associated burnup penalty in the reprocessed uranium in each recycle are discussed below.

3.1 Multicycle Analysis

In this study, uranium from both the 30 $(Th, U)O_2$ and 24 $(Th, Pu)O_2$ pins were mixed and recycled and the burnup penalty was estimated. However, the initial uranium feed for each cycle was kept constant. The dysprosium content in each recycle was optimized to get similar void reactivity in each cycle. The calculations have been done using the WIMSD lattice code system [9]. The core average discharge burnup would actually be double of the core average burnup. Hence, in order to estimate the burnup penalty, we have taken the infinite multiplication factor at a burnup of 12000 MWD/T in the base reference case to be the cut off for all the following cases.

The results are tabulated in table 5. The initial fuel charge is given as percentage of heavy metal (%

H.M.) in the inner 30 pins and outer 24 pins respectively. The uranium vector is given as % H.M at both beginning-of-cycle (BOC) and end-of-cycle (EOC). The cycle length is calculated as the maximum discharge burnup attainable without change in the initial fuel content. The lattice averaged void reactivity is also given over this cycle length. At EOC or discharge, the uranium vector in both the inner (Th, U) pins and the outer (Th, Pu) pins are given. The total uranium in the cluster is given in kilograms for BOC and for both kinds of fuel pins at EOC. The ^{233}U content is calculated in kilograms again for both BOC and EOC. The net ^{233}U indicated below is the difference in the total ^{233}U content in both the inner and outer pins at EOC and BOC. The burnup penalties are indicated in parentheses and calculated as difference from the core average discharge burnup calculated in the reference case (Case 1).

Table 5 Lattice Characteristics of AHWR D5 Cluster with Recycle* of Uranium

Case No. / Burnup	**Initial Fuel U / Pu I / O % H.M**	**Uranium compositions at discharge ^{232}U/^{233}U / ^{234}U / ^{235}U / ^{236}U (%)**	**Lattice avge. void reactivity mk**	**Total U in the cluster kg**	**Net ^{233}U in the cluster kg**	**Average Burnup MWd/t**
0 0 GWd/t **24 GWd/t** **Base Ref. Case**	3.45 / 3.0 **I** **O**	**0.0/ 100.0/ 0.0/ 0.0** 90.28 / 8.65 / 0.98 / 0.075 91.34 / 7.78 / 0.8 / 0.08	-5.033	**2.305** 1.758 0.665	**2.305** 1.587 0.608 **-0.110**	**24000**
1 0 GWd/t **22 GWd/t** **ReferenceCase** **Dy = 2.6 %**	3.45 / 3.0 **I** **O**	**0.0 / 93.70 / 5.90 / 0.4** 0.036 / 84.45/ 13.26 / 2.066 / 0.181 0.084 / 91.93 / 7.23 / 0.672 / 0.084	-5.0299	**2.305** 1.843 0.636	**2.160** 1.557 0.585 **-0.018**	**21892**
2 0 GWd/t **19.52 GWd/t** Dy = 2.3 % **I Recycle**	3.45 / 3.0 **I** **O**	**0.048 / 86.36 / 11.71 / 1.708 / 0.156** 0.069 / 78.45/ 17.54 / 3.46 / 0.484 0.09 / 92.86 / 6.51 / 0.542 / 0.0	-5.399	**2.305** 1.933 0.591	**1.990** 1.516 0.549 **+0.075**	**19526** **(2366)**
3 0 GWd/t **18.57 GWd/t** Dy = 2.0 % **II Recycle**	3.45 / 3.0 **I** **O**	**0.074 / 81.82 / 14.96 / 2.78 / 0.371** 0.101 / 74.65/ 19.99 / 4.43 / 0.829 0.09 / 93.19 / 6.20 / 0.513 / 0.0	-5.28	**2.305** 1.975 0.573	**1.886** 1.475 0.534 **+0.123**	**18569** **(3323)**
4 0 GWd/t **17.7 GWd/t** Dy = 1.9 % **III Recycle**	3.45 / 3.0 **I** **O**	**0.098 / 78.82 / 16.88 / 3.55 / 0.642** 0.099 / 72.40/ 21.3 / 5.04 / 1.15 0.096 / 93.57 / 5.9 / 0.435 / 0.0	-4.96	**2.305** 2.007 0.552	**1.816** 1.453 0.517 **+0.154**	**17691** **(4201)**
5 0 GWd/t **17.32 GWd/t** Dy = 1.8 % **IV Recycle**	3.45 / 3.0 **I** **O**	**0.098 / 76.97 / 17.98 / 4.05 / 0.902** 0.099 / 70.93/ 22.07 / 5.44 / 1.451 0.096 / 93.79 / 5.71 / 0.394 / 0.0	-5.04	**2.305** 2.025 0.542	**1.774** 1.436 0.509 **+0.171**	**17318** **(4574)**
6 0 GWd/t **17.0 GWd/t** Dy = 1.75 % **V Recycle**	3.45 / 3.0 **I** **O**	**0.098 / 75.76 / 18.61 / 4.37 / 1.144** 0.099 / 69.89/ 22.6 / 5.69 / 1.74 0.096 / 93.71 / 5.8 / 0.393 / 0.0	-4.78	**2.305** 2.030 0.544	**1.746** 1.419 0.509 **+0.182**	**17018** **(4874)**
Discharge Composition		**0.099 / 74.92 / 19.05 /4.57 / 1.372**				

The first case (Case 0) tabulated in table 5 is called as the ***base reference case*** which is the AHWR D5 cluster calculations as reported in the Physics DBR of AHWR [10]. The core average discharge burnup is estimated as 24000 MWD/T and the lattice averaged void reactivity over this burnup is –5.033 mk.

Case 1 is called as the ***Reference Case***. In this case we have assumed the uranium reprocessed from the thorium discharged in PHWRs having an average composition of ^{233}U/ ^{234}U/^{235}U in the ratio 93.7 / 5.9 / 0.4 respectively. As mentioned earlier the initial charge of uranium and plutonium has not been altered implying that fuel enrichments would be the same, i.e., 3.45% average uranium in the inner pins and 3.0% plutonium in the outer pins. The cluster has 2.305 kg of uranium as the initial charge. The optimized Dy content was found to be 2.6 % to get similar void reactivity as the *base reference case*. The core average discharge burnup is calculated as 21892 MWD/T. The burnup penalty estimated for this case is 2108 MWD/T and the lattice averaged void reactivity over this burnup is –5.03 mk. There is a deficit of 18 g in the ^{233}U of this cluster.

The burnup penalty for successive recycles has been estimated as a departure from this reference case and the results for 5 recycles are given in Table 5. The burnup penalty increases with recycling from 2366 MWD/T in the first recycle to 4874 MWD/T in the fifth recycle. This is partly due to decrease in ^{233}U content and partly due to increase in ^{234}U content with increased recycling. The Dy content has been reduced from 2.6 % to about 1.75 %. Also, the net ^{233}U in a recycle improves from 75g to 182 g. It may be mentioned that the burnup penalty can be further reduced if uranium enrichments in each recycle are altered so as to keep U-233 content same [11].

4. Radio-Toxicities of the Minor Actinides in the Discharged Fuel

In this section, we present a comparison of the radioactivity and long-term toxicity of high–level radioactive wastes from PHWR, PWR and AHWR D5 fuel cluster using the point depletion and burnup computer code ORIGEN-2 [3]. The code needs one group cross sections that can be provided from the inbuilt library for different reactor systems or can be provided externally by generating them by an external program for the particular system that is simulated. For the present analysis the inbuilt library cross sections were used. For PHWR and PWR analysis, the respective spectra were used. For AHWR analysis, standard PHWR spectrum was assumed to be valid as it uses heavy water as moderator.

The 19-rod cluster of natural uranium with specific power of 21 kw/kg and burn-up 6500 MWD/T has been considered for PHWR. The PWR fuel uses enriched U (3.25% U-235), specific power 33.0 kw/kg and burn-up of 33,000 MWD/T. The AHWR fuel assembly employs two types of fuel pins, namely, $(Th\text{-}Pu)O_2$ and $(Th\text{-}U233)O_2$ MOX. These fuel pins are arranged in three rings in the fuel cluster. The outer most ring consists of 30 pins of Th-Pu (3.0 %) MOX. The middle ring consists of Th-U233 (3.75%) MOX fuel and the inner most ring consists of Th-U-233 (3.25%) MOX fuel. The average burnup for the AHWR cluster is taken as 24,000 MWD/T and its specific power has been taken as 14.625 kw/kg.

Tables 6-8 give the concentration of minor actinides produced in the discharged fuel (g/Te) after 10 years of cooling, their radioactivity, toxicity index and toxicity potential in air as well as water for PHWR, PWR and AHWR fuel. The Toxicity index gives the amount of water/air required to dilute the toxicity of nuclide to the level of MPC (Maximum Permissible Concentration) and the toxicity potential is nothing but the radioactivity in curies multiplied by the DCI (Dose Coefficient of Intake) values [12]. In order to compare the results for different systems, they have also been computed for unit energy (per GWye) produced.

It is seen from Tables 6-7 that the main contributor to the production of minor actinides in PHWRs and PWRs is Np-237. This nuclide is generated because of decay of U-237, which is generated by the following nuclear reactions: $^{238}U(n,2n)^{237}U$, $^{235}U(n,\gamma)^{236}U(n,\gamma)^{237}U$. The later process seems to be dominating in this case. There is a significant difference between the Np-237 content in the two reactors at discharge, because of the difference in the discharge burnups. After ten years of cooling the difference

Table 6 Production of Minor Actinides, Their Radioactivity, Toxicity Index and Toxicity Potential for PHWR Fuel With 10 Yrs Cooling

Nuclide	Concentration (g / Te)	Radioactivity (C/Te)	Toxicity Index Air (m^3)	Toxicity Index Water (m^3)	Toxicity Potential Air (Sv/Te)	Toxicity Potential Water (Sv/Te)
Pa-231	--	8.993E-07	2.248E+07	1.03000E+0	3.3274	2.3292E-02
Np-237	3.214E+01	2.266E-02	2.266E+11	7.63000E+3	1.6768E+04	8.3842E+01
Am-241	9.827E+01	3.374E+02	1.687E+15	8.49700E+7	4.9935E+08	2.4968E+06
Am-243	2.810E+00	5.604E-01	2.802E+12	1.41800E+5	8.2939E+05	4.147E+03
Cm-242	6.134E-05	2.029E-01	5.073E+10	1.23100E+4	3.7537E+04	7.5073E+01
Cm-243	6.254E-03	3.230E-01	1.615E+12	7.64800E+4	3.5853E+05	2.3902E+03
Cm-244	-	1.633E+01	5.445E+13	2.36900E+6	1.8126E+07	6.0421E+04
Total	1.3323E+02	3.5484E+02	1.7461E+15	8.75772E+07	5.1872E+08	2.5639E+06
/GWye	2.4133E+04	6.4276E+04	3.1630E+17	1.58638E+10	9.3962E+10	4.6442E+08

Table 7 Production of Minor Actinides, Their Radioactivity, Toxicity Index and Toxicity Potential for PWR Fuel With 10 Yrs Cooling

Nuclide	Concentration (g / Te)	Radioactivity (C/Te)	Toxicity Index Air (m^3)	Toxicity Index Water (m^3)	Toxicity Potential Air (Sv/Te)	Toxicity Potential Water (Sv/Te)
Pa-231	-	3.176E-06	7.940E+07	3.529E+00	1.1751E+01	8.2258E-02
Np-237	5.807E+02	4.095E-01	4.095E+12	1.365E+05	3.0303E+05	1.5152E+03
Am-241	6.169E+02	2.118E+03	1.059E+16	5.295E+08	3.1346E+09	1.5673E+07
Am-243	1.465E+02	2.921E+01	1.461E+14	7.303E+06	4.3231E+07	2.1615E+05
Cm-242	2.159E-03	7.142E+00	1.786E+12	3.571E+05	1.3213E+06	2.6425E+03
Cm-243	4.339E-01	2.241E+01	1.120E+14	4.482E+06	2.4875E+07	1.6583E+05
Cm-244	3.328E+01	2.694E+03	8.979E+15	3.848E+08	2.9903E+09	9.9678E+06
Total	1.3778E+03	4.8712E+03	1.9833E+16	9.2658E+08	6.1947E+09	2.6027E+07
/GWye	4.9160E+04	1.7380E+05	7.0763E+17	3.3060E+10	2.2102E+11	9.2863E+08

Table 8 Production of Minor Actinides, Their Radioactivity, Toxicity Index and Toxicity Potential for AHWR Fuel With 10 Yrs Cooling (Average of all the pins)

	Nuclide	Concentration (g / Te)	Radioactivity (C/Te)	Toxicity Index Air (m^3)	Toxicity Index Water (m^3)	Toxicity Potential Air (Sv/Te)	Toxicity Potential Water (Sv/Te)
	Pa-231	2.2199E+00	1.0487E-01	2.6200E+12	1.1648E+05	3.8803E+05	2.7162E+03
	Np-237	5.7377E+00	4.0450E-03	4.0500E+10	1.3486E+03	2.9933E+03	1.4966E+01
	Am-241	5.1200E+02	1.7573E+03	8.7900E+15	4.3900E+08	2.6009E+09	1.3004E+07
	Am-243	2.5729E+01	5.1289E+00	2.5700E+13	1.2829E+06	7.5908E+06	3.7954E+04
	Cm-242	4.1960E-03	1.3880E+01	3.4700E+12	6.9404E+05	2.5678E+06	5.1356E+03
	Cm-243	2.0813E-01	1.0747E+01	5.3700E+13	2.1496E+06	1.1929E+07	7.9525E+04
	Cm-244	1.8747E+00	1.5173E+02	5.0600E+14	2.1668E+07	1.6842E+08	5.6141E+05
Average of all pins	Total	5.4777E+02	1.9389E+03	9.3815E+15	4.6491E+08	2.7918E+09	1.3691E+07
	/GWye	2.6873E+04	9.5122E+04	4.6025E+17	2.2808E+10	1.3696E+11	6.7167E+08
Th- Pu pins	Total	1.2302E+03	4.3625E+03	2.1104E+16	1.0470E+09	6.2811E+9	3.0869E+07
	/GWye	5.3165E+04	1.88540E+5	9.1204E+17	4.5249E+10	2.7146E+11	1.3312E+09
Th – U pins	Total	1.7989E+00	8.4616E-02	2.1131E+12	9.3922E+04	3.1306E+05	2.1914E+03
	/GWye	9.8942E+01	4.6542E+00	1.1623E+14	5.166E+06	1.722E+07	1.2054E+05

between the total minor actinides in the case of these two reactors reduces because Pu-241 ($T_{1/2}$ ~13 yrs) decays to Am-241.

Table 8 gives the results for AHWR fuel. Last four rows of the Table 8 give the results for two types of pins ((Th-U233)O_2 and (Th-Pu)O_2) used in AHWR at their respective burnups of 21406 MWD/T and 27243 MWD/T for an average discharge burnup of 24000 MWD/T. Average of all pins combines these results to give the values for the composite cluster. The minor actinides in the Th-U pin per unit energy (per GWye) are significantly less compared to Uranium cycle (Tables 6&7). This is mainly because except for Pa-231 most of the minor actinides are having mass numbers much higher than Th-232, so at least 5 to 6 capture and decay reactions are required to generate those minor actinides. The minor actinides per unit energy for the Th-Pu pins are relatively higher because of Am and Cm isotopes. Due to these pins, the minor actinides in AHWR fuel are comparable with PHWR fuel but it is small compared to PWR fuel and is approximately 29 % of PWR fuel.

The variation in the toxicity levels in the three different types of reactors PWR, PHWR and AHWR are also shown in Tables 6-8. The following observations are made based on the above analysis:

a) It is seen that the toxicity of the PWR fuel is higher than those of PHWR and AHWR fuel.
b) The average toxicity of the AHWR fuel (Table-8) is comparable with the PHWR fuel.
c) The toxicity of (Th-U233) pins (Table-8) is about 10^4 order of magnitude less than the toxicity of any other fuel shown in the Tables 6-8.

5. Conclusions

- The use of Natural UO_2 in the central 116 channels with 8 bundle shift and rest with MOX-888 with 4 bundle shift gives a very good combination which can be used in PHWRs in future. The overall fuel requirement reduces to about 65% of the present level, whereas the requirement of Natural UO_2 reduces to about 33 % of the present level. The average discharge burnup of Natural UO_2 has improved by 40% to 9400 MWD/T.
- The multicycle study for AHWR fuel showed that successive recycles of uranium in AHWR increases the content of ^{234}U from 5.9% H. M. to 18.6% H. M. As pointed out earlier, ^{234}U is a load only to the extent of reduction in the burnup of about 2108 MWD/T using the reprocessed uranium from PHWRs and this further increases by 4874 MWD/T in five recycles. In all these cases, the initial charge of both uranium and plutonium has not been changed.
- The present analysis of production of minor actinides for AHWR was performed by using the inbuilt cross sections in the computer program ORIGIN-2. The preliminary results indicate that the production of minor actinides and their toxicity for the AHWR fuel is comparable to PHWR fuel and it is slightly less than PWR fuel. The AHWR fuel provides a good opportunity of incinerating plutonium and producing U-233.

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