

ELECTRICAL ENGINEERING

Cm-244 as Multiplier and Breeder in a ThO₂ Hybrid Blankets of Deuterium-Tritium Driven Source

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Abstract. The safeguard aspects of Cm-244 – a nuclear waste product in LWRs – in a cylindrical hybrid blanket driven by a (D,T) fusion neutron source have been analyzed.

Cm-244 has been investigated for two different applications:

1. As a neutron multiplier between the first wall and the fuel zone in a blanket with ThO₂.
 2. As a component of the mixed fuel, ThO₂-Cm²⁴⁴O₂, used for power flattening in a hybrid blanket.
- The calculations have shown that a relatively small reactor driven with 100 MW_{th} fusion power could produce about 5 kg/year Cm-245, enough to provide nuclear fuel for up to 50 explosives. The study suggests an extension of the safeguarding regulations prior to the commercial introduction of fusion reactors in the energy market.

1. Introduction

Presently, nuclear power plants are producing a substantial amount of actinides in the form of nuclear waste material [1,2,3]. Previous work has demonstrated, that some of the nuclear waste actinides, such as Am-241 and Cm-244 are effective neutron multipliers in a hybrid blanket. They will be converted, partially to new types of nuclear fuel with superior neutronic properties, such as Am-242m and Cm-245 [4-7]. Unfortunately, these studies were conducted for an experimental blanket in plane geometry which can not simulate the spectral conditions in hybrid power plant.

The very hard neutron spectrum of a fast hybrid reactor allows the build up of these new fissile material whereas even an odd americium and curium isotopes are actually burnt up in situ in standard nuclear reactors. Recent measurements show that the fission cross sections of Am-242 m at lower energies are approximately one order of magnitude higher than that of U-235 [8].

Although, the neutron capture reactions in actinides do not permit one to burn them completely, all the same they lead to conversion of the nuclear waste materials

to very precious nuclear fuels of a new type. These nuclear fuels are produced, as follows:

$\text{Am}^{241}(n,\gamma) \text{Am}^{242}$ Half life = 16 hours, probability 65% [9].

$\text{Am}^{241}(n,\gamma) \text{Am}^{242m}$ Half life = 152 years, probability 35% [9].

$\text{Am}^{243}(n,\gamma) \text{Am}^{244}$ Half life = 10 hours.

$\text{Cm}^{244}(n,\gamma) \text{Cm}^{245}$ Half life = 8532 years.

Among these isotopes only Am^{242m} and ^{245}Cm can be considered as a potential precious nuclear fuel due to their relatively long lifetime. They allow one to realize mini-fission systems due to their very superior nuclear properties, compared with standard nuclear fuel, such as U-233, U-235 or Pu-239.

On the other hand, a recent work has shown that only minor quantities (100 gr) of Cm-245 would be sufficient to assemble an explosive device with a yield of 1 kT, by applying the well established implosion technique for plutonium assemblies [10].

By considering the above mentioned facts, it becomes worthwhile to examine, extensively, the production capability of hybrid reactors for advanced fissile materials, such as Cm-245.

These studies can then backup suggestions to International Atomic Energy Agency (IAEA) in Vienna to extend the scope of safe-guarding. Consequently, they can bring also new aspects for the denaturation of nuclear fuel. At present, the IAEA regulations for safeguarding cover only U-233, U-235 and Pu-239.

In this work, the neutron multiplication and breeding performance of Cm-244 is analyzed in connection with a (D,T) driven experimental hybrid blanket in cylindrical geometry within the research program AYMAM [11,12], in order to simulate relatively realistic neutron spectra for future hybrid reactors [13-16].

2. Description of the Blanket

With the framework of the AYMAM, the 14-MeV fusion-neutron will be produced by a moveable target along the axis of a cylindrical hybrid blanket to simulate a line source.

A hybrid blanket, selected for the study, is shown in Figs 1 and 2. It consists mainly of the following components:

- A) The cylindrical first wall is SS-316 with a thickness of 1.3 cm, the same as that of the first wall of the TFTR in Princeton.
- B) The fuel rods are made of ThO_2 (10 mm diameter) with a sintering density of 93% and are covered with an aluminium cladding ($D_0 = 12$ mm, $D_i = 10.4$ mm). In the fuel zone, they are arranged hexagonally, making 10 rows with a volume fraction of 42% of air to simulate a gas cooled blanket. The fuel zone begins with $r = 30$ cm and ends with $r = 43$ cm, making a total thickness of 13 cm.

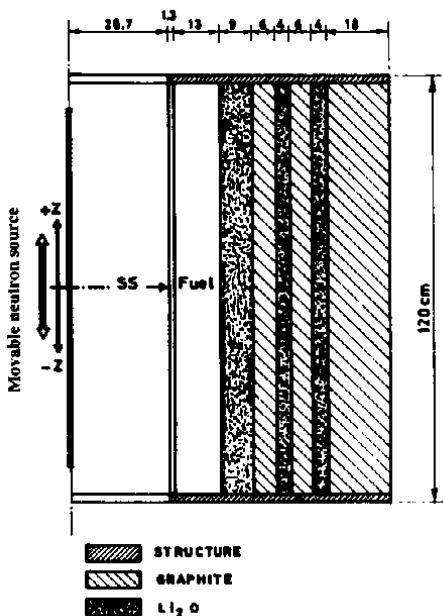


Fig. 1. Basic structure of the investigated experimental blankets

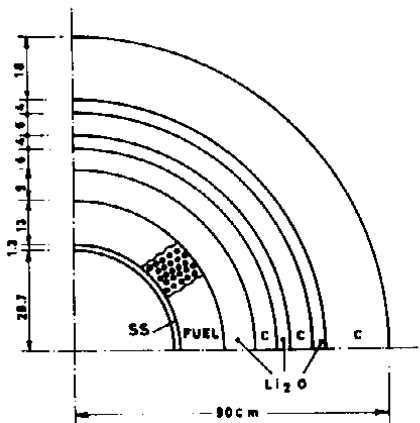


Fig. 2. Cross sectional view of the blanket

Table I. Neutronic performance of the investigated hybrid blankets using different multiplier (per incident 14 MeV neutron)

Reaction Type	*Blanket Type:					
	I	II	III	IV	V	
Multiplier:	None	Cm ²⁴⁴ Multiplier	Cm ²⁴⁴ Breeder	Be (4 cm)	Pb (8 cm)	
	(Reaction rates in hybrid blanket, 1/cm ³ . sec.)					
Li ⁶ (n,α)T	0.6478	0.7215	0.6849	0.6191	0.7694	
Li ⁷ (n,n'α)T	0.1048	0.1051	0.1051	0.0752	0.0449	
Total T Breeding	0.7526	0.8266	0.7900	0.6943	0.8144	
Th ²³² (n,f)	0.0441	0.0424	0.0436	0.0298	0.0203	
Cm ²⁴⁴ (n,f)	-	0.0507	0.0253	-	-	
Th ²³² (n,γ)U ²³³	0.2698	0.2889	0.2719	0.5403	0.3806	
Cm ²⁴⁴ (n,γ)Cm ²⁴⁵	-	0.0148	0.0137	-	-	
Total Breeding (Fusile & Fissile)	1.0224	1.1303	1.0756	1.2346	1.1950	
Be(n,2n)	-	-	-	0.3905	-	
Pb(n,2n)	-	-	-	-	0.4365	
Th ²³² (n,2n)	0.1513	0.1395	0.1470	0.1017	0.0573	
Neutron Leakage	0.0554	0.0590	0.0575	0.0410	0.0421	
Neutron Absorption	1.3362	1.5167	1.4249	1.6281	1.5914	
$\nu \Sigma_f$	0.1613	0.3563	0.2560	0.1042	0.0687	
M	1.8514	2.5765	1.8569	1.6373	1.5529	
k _{eff} ^{**}	0.3007	0.3711	0.3428	0.5037	0.4965	
k _{eff} ^{***}	0.1159	0.2261	0.1727	0.0625	0.0421	

*Structure of blankets, are:

- I: Cavity radius 28.7 cm + 1.3 cm SS + 1.3 cm SS + 13 cm ThO₂ (fuel) + 9 cm Li₂O + 6 cm C + 4 cm Li₂O + 18 cm C.
- II: Cavity radius 28.7 cm + 1.3 cm SS + 13 cm ThO₂ and CmO₂ (Mixed fuel multiplier zone) + 11.7 cm ThO₂ (fuel) + 9 cm Li₂O + 6 cm C + 4 cm Li₂O + 6 cm C + 4 cm Li₂O + 18 cm C.
- III: Cavity radius 24.7 cm + 1.3 cm SS + 5 cm ThO₂ + 4 cm ThO₂ mixed CmO₂ (4.6%) + 4 cm ThO₂ mixed with CmO₂ (8%) + 9 cm Li₂O + 6 cm C + 4 cm Li₂O + 6 cm C + 4 cm Li₂O + 18 cm C.
- IV: Cavity radius 24.7 cm + 1.3 cm SS + 4 cm Be + 13 cm ThO₂ (fuel) + 9 cm Li₂O + 6 cm C + 4 cm Li₂O + 6 cm C + 4 cm Li₂O + 18 cm C.
- V: Cavity radius 20.7 cm + 1.3 cm SS + 8 cm Pb + 13 cm ThO₂ (fuel) + 9 cm Li₂O + 6 cm C + 4 cm Li₂O + 6 cm C + 4 cm Li₂O + 18 cm C.

** defined, as Eq. (2), including (n,2n) reaction

*** defined, as Eq. (3), excluding (n,2n) reaction.

As are demonstrated in Table I, the blanket structures consist of the following:

- i) Blanket type I has only ThO₂ fuel in the 10 rows fuel zone.
 - ii) Blanket type II, has in its first fuel row a mixed fuel made of 50% CmO₂ and 50% ThO₂ (see Section 3.1).
 - iii) Blanket type III, has ThO₂ mixed with 4.6% or 8% CmO₂ in the last 8 cm of the fuel zone as breeder (see Section 3.2).
 - iv) Blankets type IV and V, have 4 cm Beryllium, and 8 cm lead multiplier, respectively, been adopted between the first wall and the fuel zone.
- C) For tritium breeding, Li₂O contained in aluminium cans, has been adopted in the form of a sandwich structure with graphite reflector, as it is schematically shown in Fig. 2 (See ref. [16] for detail).
- D) The neutron reflector is of graphite with a total thickness of 30 cm.

The introduction of Cm-244 into the blanket as neutron multiplier and breeder will be described in the next section.

3. Numerical Calculations

The transport calculations were performed in the S₈-P₃ approximation with the ANISN [17] code in cylindrical geometry with a buckling correction for a height of 120 cm, using 100 neutron energy group data from DLC-2F [18] and DLC-24 [19] libraries. Initially, the neutronic performance of a 4 cm Beryllium multiplier, and then that of 8 cm Lead multiplier has been evaluated.

3.1. Cm-244 as Neutron Multiplier

In the next phase, a Cm-244 multiplier is introduced by replacing the first ThO₂ row with a mixed fuel made of 50% CmO₂ and 50% ThO₂, and by omitting the Beryllium and Lead multiplier.

The main integral data for these blankets are shown in Table I, where the energy multiplication (M) and neutron multiplication (K_{eff}) factors are defined as follows:

$$M = \frac{\text{Heat release in the blanket (MeV)}}{14 \text{ MeV}} + 1 \quad (1)$$

$$K_{\text{eff}} = \frac{\iint (\nu \Sigma_f + 2 \times \Sigma_{2n}) \phi \, dV \, dE}{\iint (\Sigma_a + \Sigma_{2n}) \phi \, dV \, dE + \iint J \, dS \, dE} \quad (2)$$

including (n,2n) reactions

$$K_{\text{eff}} = \frac{\int \int v \Sigma_f \phi \, dV \, dE}{\int \int \Sigma_a \phi \, dV \, dE + \int \int J \, dS \, dE} \quad (3)$$

excluding (n, 2n) reactions.

One can observe in Table I (Blanket Type II) that the Cm-244 multiplier leads to the highest tritium breeding ratio.

U-233 production is reduced compared to Be and Pb multipliers. The Cm-244 in the first row contributes to the fission rate more than all of the Th-232 in the blanket. About 1/5 of Cm-244 is converted into Cm-245, whereas 4/5 of Cm-244 burns up.

Fig. 3 shows the spatial variation of the main neutronic parameters in the fuel zone. While the fissile breeding ratios $\text{Th}^{232}(n, \gamma) \text{U}^{233}$ and $\text{Cm}^{244}(n, \gamma) \text{Cm}^{245}$ are fairly flat, one can observe that the fission reaction rate density in the first row is about one order to magnitude higher than in the rest of the fuel zone. This is an undesirable concentration of the thermal heat production in the blanket and suggests the implementation of Cm-244 in a hybrid blanket in a different way, as discussed in the next section.

Fig. 4 shows the tritium breeding density in the Li_2O zones for this blanket. Due to the sandwich structure of the Li_2O and graphite, the depression of the tritium breeding by density is fairly moderate in the strong absorbing Li^6 zones throughout the blanket.

3.2. Cm-244 as Breeder

The excessive fission rate of Cm-244 in a hybrid blanket can be suppressed to some degree by introducing it in the external rows of the fuel zone. In this case, Cm-244 can also be used to achieve a flat fission power density (FPD) profile in the fuel zone by choosing an appropriate mixture of ThO_2 and $\text{Cm}^{244}\text{O}_2$ with the help of a fission power flattening method, recently developed [20].

In the above method, the procedure of the power flattening is applied on a (D,T) driven cylindrical hybrid blanket concept within the framework of the research project AYMAM [11]. The total height of the cylindrical blanket considered is 250 cm inclusive of 25 cm graphite at the top and bottom. A neutron generator, with a movable target along the axis of the blanket is used to simulate a fusion line source. The flat fission power density is achieved by using a mixed fuel (ThO_2 and natural UO_2) with the thorium/uranium ratio changing from front to back in the ten fuel rows along the radial direction. A straightforward graphic method is used.

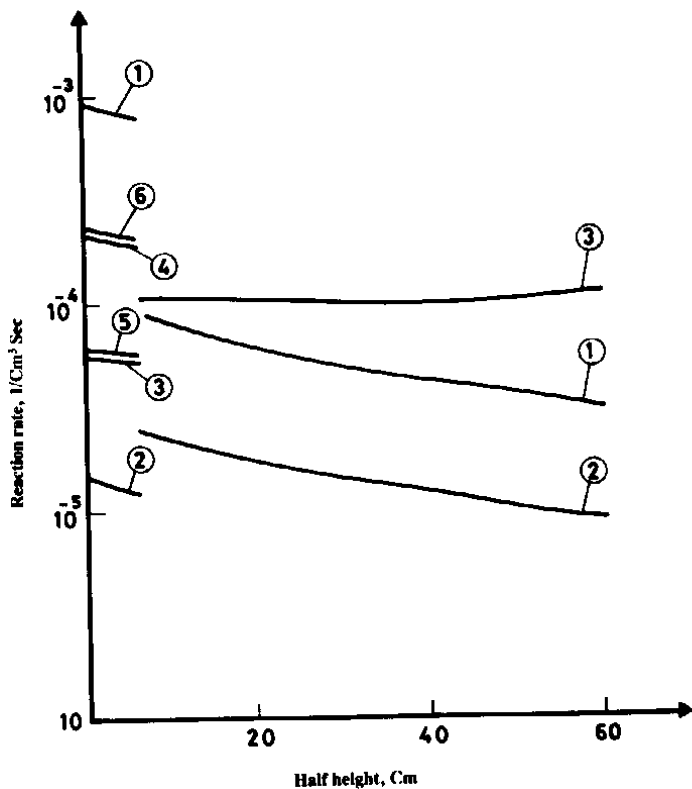


Fig. 3. Main neutronic parameters in the fuel zone, using Cm-244 as neutron multiplier

- | | |
|--------------------------------|---|
| 1) $v\Sigma_p$, total | 4) $\text{Cm}^{244}(n,f)$ |
| 2) $\text{Th}^{232}(n,\gamma)$ | 5) $\text{Cm}^{244}(n,\gamma)$ |
| 3) $\text{Th}^{232}(n,\gamma)$ | 6) $\text{Cm}^{244}(n,f) + \text{Th}^{232}(n,\gamma)$ |

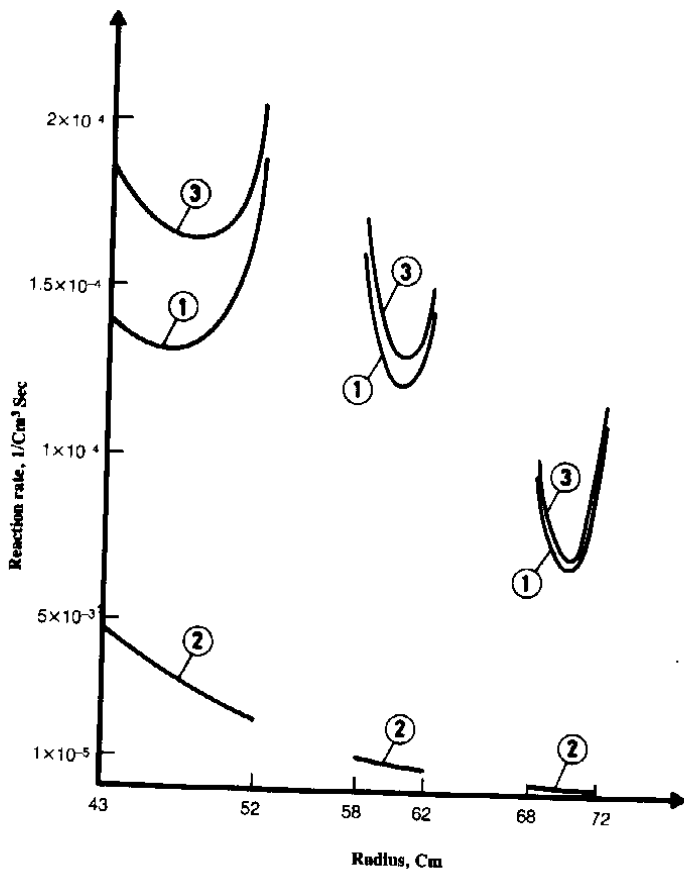


Fig. 4. Tritium breeding density in the blanket with Cm^{244} as multiplier

- 1) $\text{Li}^6(n, \alpha)\text{T}$
- 2) $\text{Li}^7(n, \alpha)n'\text{T}$
- 3) $\text{T}^6 + \text{T}^7$

This method suggests a Cm-244 fraction of 4.6% and 8% in the fuel rows 5 to 7 and 8 to 10, respectively, in order to achieve a quasi-flat FPD in the blanket. The total amount of Cm-244 in the blanket with a flat FPD will then be about 98% of that when Cm-244 is used as a neutron multiplier in the previous case.

Fig. 5 shows the spatial variation of the main neutronic parameters in the fuel zone after the application of the flattening procedure. One can see that the fission power generation in the fuel zone is quite uniform for a source driven blanket. This is very important point for optimum utilization of the blanket with respect to power production, material damage, fuel burn-up, thermal stresses, etc. Hence, one can suggest that this alternative represents a better way toward eventual utilization of Cm-244 in a hybrid blanket.

Table I (Blanket Type III) includes also the integral neutronic data for this alternative of the utilization of Cm-244 in a hybrid blanket. Most of these data are more or less comparable when Cm-244 is used mainly as neutron multiplier in the first row. Only the Cm-244 (n,f) reaction is reduced by about 50%, resulting into a reduction of neutron and energy multiplication factor, k_{eff} and M, respectively.

Fig. 6 shows the tritium breeding density for this blanket with a flat FPD. It has similar shape, when Cm-244 is used as a neutron multiplier (Fig. 4).

4. Discussion

The nuclear waste material Cm-244 can be burnt in a hybrid blanket, as an efficient nuclear fuel. It makes an important contribution to the neutron multiplication and enhances the overall neutronic performance of the blanket. It can also be used for flattening the fission power production in the blanket.

Some of the Cm-244 will be converted to Cm-245 which is a new kind of fissile material with superior neutronic properties which can cause serious concern with respect to proliferation.

By introducing Cm-244 into the external fuel rows (breeder revision), the cumulative fission rate in the Th-232 in the blanket becomes higher than that in Cm-244. About 1/3 of the Cm-244 is converted into Cm-245 and the burn-up fraction in Cm-244 is reduced to 2/3.

The utilization of Cm-244 for power flattening increases the specific production rate of Cm-245 per nuclear waste material.

A rough estimate of this prolific potential can be made as follows:

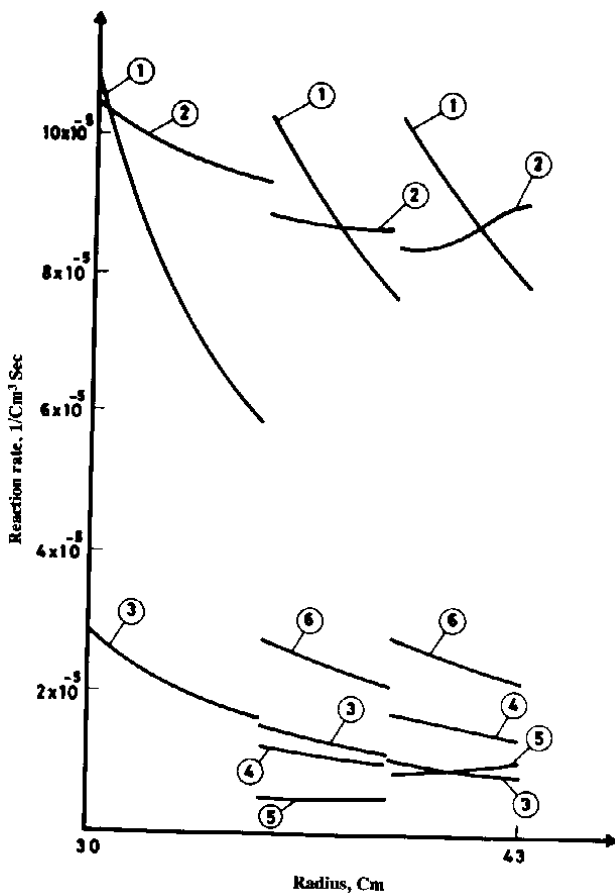


Fig. 5. Main neutronic parameters in the fuel zone, using Cm-244 as breeder

- | | |
|-----------------------------------|--|
| 1) $\nu \Sigma_{\text{total}}$ | 4) $\text{Cm}^{244}_{(n,f)}$ |
| 2) $\text{Th}^{232}_{(n,\gamma)}$ | 5) $\text{Cm}^{244}_{(n,\gamma)}$ |
| 3) $\text{Th}^{232}_{(n,f)}$ | 6) $\text{Cm}^{244}_{(n,f)} + \text{Th}^{232}_{(n,f)}$ |

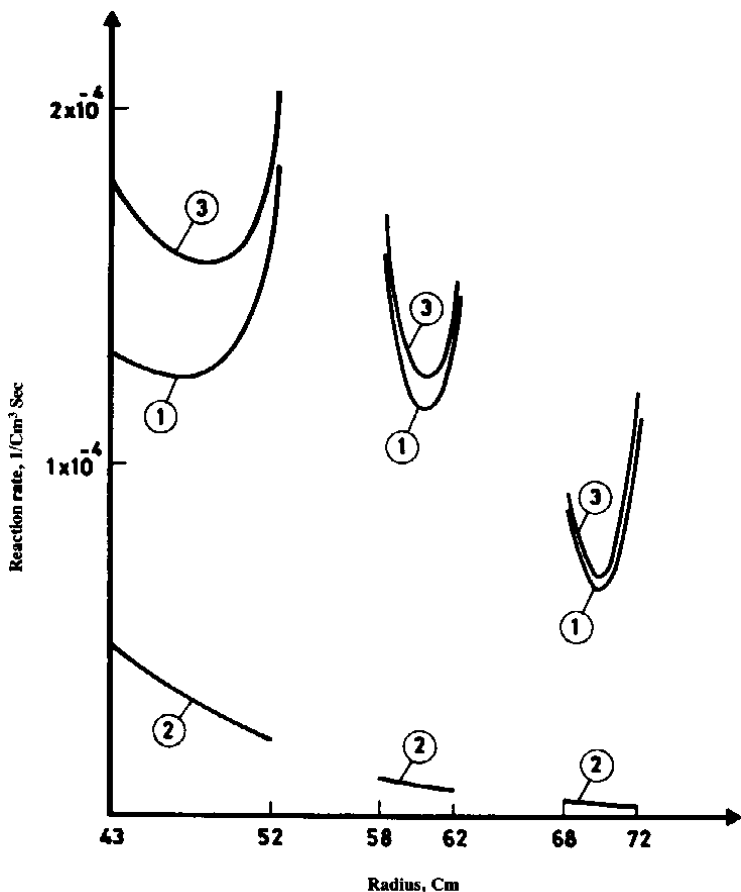


Fig. 6. Tritium breeding density in the blanket with Cm²⁴⁴ as breeder (Legende: See Fig. 4)

In a blanket, using Cm-244 as a neutron multiplier (breeder), the Cm-245 production rate is 0.0148 (0.0137) per (D,T) fusion neutron which corresponds to a production of 50 gr (46 gr) Cm-245 per year-MW_{th} of fusion power with a plant load factor of 75%.

A modest fusion plant with 100 MW_{th} power would make possible the production of about 5 kg (4.6 kg) Cm-245 per year, under similar spectral conditions, enough to provide nuclear fuel for about 50 (46) explosives per year.

The raw material required for this process, would be about 25 kg (14 kg) Cm-244 per year which can be extracted from the inventory of minor actinides from 30 tons (17 tons) heavy metal, after a burn-up of 33 GWd/ton in a 1250 MWe PWR plant, containing 30% conventional oxide fuel (U-Pu) and 70% Uranium with plutonium recycling [3]. These numbers demonstrate clearly the necessity for a revision of the safeguarding regulations prior to the introduction of fusion reactors. It is not too early to consider safeguarding the minor actinides to avoid a new kind of proliferation risk which may arise around the turn of century.

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استعمال مادة الكورיום - ٢٤٤ كمضاعف ومولد في أغلفة أكسيد الثوريوم لمفاعل اندماجي يشتغل بمصدر ديتريوم - تريتيوم

توفيق أحمد القصير

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ملخص البحث. لقد تم تحليل الإجراءات الاحترازية الخاصة بإعادة الكورיום - ٢٤٤ - وهي فضلات نووية ناتجة من مفاعلات الماء الخفيف - في أغلفة المفاعل الأسطواني المزدوج الذي يتم تشغيله بواسطة (ديتريوم - تريتيوم) كمصدر نيوتروني اندماجي.

في هذا البحث تم التحقق من تطبيقات مادة الكورיום - ٢٤٤ :

(١) كمضاعف نيوتروني بين الجدار الأول ومنطقة الوقود في الغلاف الذي يستخدم مادة أكسيد الثوريوم.

(٢) كإداة في وقود مخلوط من أكسيد الثوريوم وأوكسيد الكورיום استعملت لأجل تسوية الطاقة في غلاف المفاعل المزدوج.

لقد أثبتت الحسابات أن مفاعل صغير بقدرة ١٠٠ ميغاوات طاقة اندماجية يمكنه أن ينتج حوالي ٥ كغم / السنة من مادة الكورיום - ٢٤٥ ، وهي كافية لإمداد ٥٠ رأس نووي بالوقود النووي اللازم. تقترح الدراسة إدخال بعض الإضافات للأنظمة الاحترازية المعمول بها في الوكالة الدولية للطاقة النووية قبل دخول المفاعلات الاندماجية تجارياً في أسواق الطاقة.