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DESIGN STUDY OF A PWR OF 1.300 MWe OF ANGRA-2 TYPE OPERATING IN THE THORIUM CYCLE

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SUMÁRIO

É analisada a utilização de óxidos mistos de tório - urânio altamente enriquecido e de tório-plutônio em um PWR não modificado. O PWR de 1300 MWe da KWU (tipo Angra-2) é tomado como reator de referência para o estudo. Cálculos de projeto do núcleo do reator foram realizados para os dois tipos de combustível, considerando-se ciclos sem reiclagem e com reciclagem. Os cálculos foram realizados com os códigos de projeto da KWU, FASER-3 e MEDIUM-2.2, após a introdução da cadeia do tório e da inclusão de dados de alguns nuclídeos em FASER-3. Um esquema de dois grupos de energia e uma representação bi-dimensional do núcleo (XY) foram utilizados. Não foi encontrado problema técnico algum que excluisse a utilização de qualquer das opções analisadas. A economia nas reservas de urânio introduzida pelo ciclo do tório com reciclagem de combustível varia de 13% a 52%, em comparação com o ciclo usual do urânio sem reciclagem; a economia de trabalho de separação (SWU) vai de 13% a 22%. Entretanto, o ciclo Th/HEU sem reciclagem consome mais urânio e SWU do que o ciclo do urânio sem reciclagem. Investigações adicionais são recomendadas.

ABSTRACT

The utilization of the thorium - highly enriched uranium and of the thorium - plutonium mixed oxide fuels in an unmodified PWR is analysed. The PWR of 1300 MWe from KWU (Angra-2 type) is taken as the reference reactor for the study. Reactor core design calculations were performed for both types of fuels considering once-through and recycle fuels. The calculations were performed with the KWU design codes FASER-3 and MEDIUM 2.2 after introduction of the thorium chain and some addition of nuclide data in FASER-3. A two-energy group scheme and a two-dimensional (XY) representation of the reactor core were utilized. No technical problem that precluded the utilization of any of the options analysed was found. The savings in uranium ore introduced by the thorium cycle with fuel recycling ranges from 13% to 52% as compared with the usual uranium once-through cycle; the SWU savings goes from 13% to 22%. Nevertheless the Th/HEU once-through cycle consumes more uranium and SWU than the uranium once-through cycle. Further investigations are recommended.

ÍNDICE

1. INTRODUCTION

Uranium occurs in nature as a mixture of two main isotopes: ^{238}U and ^{235}U. The content of ^{235}U in the natural mixture is of only 0,7%; ^{235}U is the only fissile isotope occuring in nature. Without its presence it would not be possible to build a fission reactor unless using artificially produced fissile nuclides (^{239}Pu, ^{241}Pu or ^{233}U), which are only produced in appreciable amounts in nuclear reactors initially fueled with ^{235}U, during their operation. Then, the existence and production of artificial fissile nuclides in required amounts for use in power reactors depends on the existence of the ^{235}U nuclides.

The production and utilization of artificial fissile nuclides in nuclear reactors has then great interest, since those nuclides can replace the ^{235}U as the fuel. This enables a better utilization of the natural uranium reserves; one could also say that the energy output per tonne of mined uranium would be higher.

The ^{233}U, one of the artificial fissile nuclides, is produced from ^{232}Th, only nuclide present in the abundant natural thorium, according to the following simplified chain

$$^{232}_{90}Th + ^{1}_{0}n \rightarrow ^{233}_{90}Th \xrightarrow{\beta^-} ^{233}_{91}Pa \xrightarrow{\beta^-} ^{233}_{92}U$$

Under the conditions prevailing in the cores of the present commercial reactors, such as the PWR - Pressurized Water Reactor, the ^{233}U has better nuclear properties than the other fissile nuclides have. This is due to the fact that in the neutron spectrum (thermal) of these reactors more neutrons are produced, on the average, per neutron absorbed in the ^{233}U (higher η value) than in other fissile atoms.

From what has been said one sees the interest of a detailed investigation about the technical feasibility of the use of thorium in nuclear reactors.

The thorium utilization in a type of reactor already commercially proved, without the introduction of modifications in the reactor core and in the nuclear power plant excepting the composition of the fuel itself, has the great advantage of avoiding the necessity of development and proving of a new reactor concept which would require a much greater R&D effort and correspondingly financing costs.

This is the way adopted in the Phase 1 (1979/83) of the joint NUCLEBRÁS-KFA-KWU-NUKEM Research and Development program on Thorium Utilization in PWR [1] *. The reference reactor chosen for the feasibility studies of this program is the PWR of 1.300 MWe standard KWU concept (Angra-2 type), whose main data are presented in Table 1.

Besides the technical feasibility of the thorium utilization in the PWR the following questions are to be answered by this program: Which one of the Th-fuel cycle options is the most advantageous, considered the existing boundary conditions? How do the thorium-fuel cycles compare to usual U-fuel cycles in a system of PWR with respect to resource utilization and economics? At which

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* The development of nuclear fuel containing thorium in the frame of this program is described in another paper presented to this congress.

time is the introduction of Th-fuel cycles advantageous or possible? How to compare Th-fuel cycles for PWR with advanced reactor systems and which long range reactor combination (i.e. strategy scenario) is proposed to exist?

2. CYCLES ANALYSED

As mentioned, thorium can not be utilized alone as a reactor fuel; it requires an accompanying fissile material in the reactor core. Two options for the starting (or complementary) fissile material are being analysed:

- Highly enriched uranium (HEU), with 93 w/o in ^{235}U, mixed with ^{232}Th in a relation of about 4.2 w/o ^{235}U to the total heavy metal (U+Th). (The enriched uranium comes from the enrichment plants).
- Plutonium, normally produced in current commercial power reactors from the predominant ^{238}U, according to the simplified chain

$$^{238}_{92}U + ^{1}_{0}n \rightarrow ^{239}_{92}U \xrightarrow{\beta^-} ^{239}_{93}Np \xrightarrow{\beta^-} ^{239}_{94}Pu$$

A PWR, such as Angra-2 reactor, produces, at equilibrium, about 218 kg of fissile plutonium (^{239}Pu + ^{241}Pu) per GWe.year. This plutonium would then be mixed with thorium in a relation of about 4.4 w/o ^{fiss}Pu to total heavy metal (Pu + Th).

In both cases the fuel considered is in the form of a mixed oxide: $(Th,U)O_2$, i.e. $ThO_2 + UO_2$, or $(Th,Pu)O_2$, i.e. $ThO_2 + PuO_2$.

Two options were considered for both the Th/HEU and the Th/Pu cycles:

- once-through
- recycle.

In the first option the fuel after irradiation in the reactor is discharged and stored; in the second case the irradiated fuel after discharge is reprocessed in a reprocessing plant and its content of fissile material is recovered and used for the fabrication of new fuel elements to be reinserted in the reactor core.

The options analysed up to now are the following:

Th/HEU

- once-through
  - .3-batch cores
  - .4-batch cores
- Recycle of ^{233}U and ^{235}U
  - .3-batch cores: segregated recycling
    mixed recyling
  - .4-batch cores: mixed recycling

Th/Pu

- once-through
  - .3-batch cores
  - .4-batch cores
- Recycle of ^{233}U and Pu
  - .3-batch cores
  - .4-batch cores

3. CALCULATIONS AND RESULTS

The KWU design computer codes FASER-3 [2] and MEDIUM-2.2 [3] were selected to be used in the core nuclear design calculations.

The thorium evolution chain had to be introduced in FASER-3 and the addition of some nuclide data were done in its cross section library, using ENDF/B-4 library, in order to enable the code to treat the thorium fuel cycle. The resulting code - named Th-FASER - was verified through the comparison of its results with experimental results from Brookhaven [4]. No modification in MEDIUM-

2.2 was needed for applying the code.

Th-FASER was employed to generate 2-group cross sections dependent on fuel composition, burnup, boron concentration, fuel temperature, moderator temperature and for the control rods. For the core calculations with MEDIUM 2.2 a two dimensional XY representation was adopted.

For recycling a period of 2 years for cooling, reprocessing and refabricating of spent fuel was adopted. For the 1st recycle batch losses of 5% were assumed (2,5% in reprocessing and 2,5% in refabrication), and for the subsequent batches losses of only 2,5% in reprocessing.

For all the recycle cases the recycling was started at the beginning of cycle number 4 (BOC-4), with fissile material recycled from the 1st cycle of the same reactor. The complementation of fissile material required for the reloads was done in the case of the Th/HEU cycle by the addition of new ^{235}U, and in the case of the Th/Pu cycle by the addition of Pu from a PWR operating in the standard uranium cycle.

For each case analysed it was determined:

- fuel enrichments and initial core configuration,
- reload enrichment and shuffling scheme,
- cycle length,
- isotopic compositions at beginning-and end-of-cycle,
- power form factors,
- reactivity coefficients (moderator temperature, integral Doppler, boron worth),
- shutdown margins.

The enrichment in fissile material for the reloads was determined in such a way that the equilibrium cycle lengths were of approximately 280 EFPD (equivalent full power days).

The requirements in natural uranium and separative work units (SWU) for the equilibrium cycles were also determined. In the case of the Th/Pu cycle there is no direct need of natural uranium or separative work for the Th/Pu fueled reactors. For this reason each time that a PWR(Th/Pu) reactor can be installed in the place of a standard PWR(U) to supply the energy demand in a scenario otherwise constituted of only PWR(U) once-through reactors there is an economy in natural uranium and SWU corresponding to the consumption of these two quantities by the standard PWR(U) that is replaced.

Nevertheless for the installation of PWR(Th/Pu) reactors there must be enough plutonium available. Since the Pu is produced in the PWR(U), and the plutonium production of a 1.300 MWe PWR(U) is at a rate of 218 kg ^{fiss}Pu/GWe.a at equilibrium cycle, the installation of one PWR(Th/Pu) requires the prior installation of a certain number of PWR(U) operating with no recycle.

The savings in natural uranium and SWU obtained for the investigated PWR thorium fuel cycles are shown in Table 2.

4. CONCLUSIONS

The nuclear core desing calculations have indicated that the insertion of full cores $(Th,U)O_2$ and $(Th,Pu)O_2$ fuel into the PWR of present standard design (Angra-2 type) is possible for open and closed fuel cycles. Both 3- and 4-batch loadings fulfill the requirements for safe reactor operation.

The application of the closed thorium fuel cycles to a standard PWR leads to substantial resource savings.

$(Th,Pu)O_2$ fuel offers a potential for further extension of the burnup, to avoid the need for early closing the thorium fuel cycle by reprocessing.

.Further investigations are recommended:

. The switch over from U to Th-cycle operation is to be investigated in detail;
. Safety aspects (e.g. main steam line rupture) should be quantified in support of the results reached so far;
. Burnup limitations for $(Th,Pu)O_2$ cores in a standard PWR, including stainless steel cladding, should be analysed in more detail.
. The contribution of high burnup cores and the advanced fuel cycles (high conversion) with $(Th,Pu)O_2$ fuel should be analysed with respect to core design, resource conservation and fuel cycle economy.
. The build up of higher actinides and the impact on handling and final waste disposal should also be investigated.

Most of these investigations will be performed in the Phase-2A (1984/86) of the Th-Utilization in PWR program.

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Table 1. Design Data of a KWU Standard PWR of 1.300 MWe

Fuel rod diameter [mm]	10,75
Lattice pitch [mm]	14,30
Number of fuel rods per fuel assembly	16x16-20 (=236)
Fuel assembly pitch [mm]	231
V_{H_2O}/V_{fuel} (cell)	1,74
V_{H_2O}/V_{fuel} (core)	2,05
Fuel rod linear power [W/cm]	211
Number of fuel assemblies	193
Fuel rod active length [mm]	3.916,0
Cladding wall thickness [mm]	0,725
Volume of the spacing grids per fuel assembly [cm³]	831,62
Cladding material	Zry-4
Control rod guide tube material	SS-4550 Zry-4
Spacing grids material	Inc-718
Heavy metal mass per fuel assembly [kg]	485,365
Fuel average temperature [°C]	675
Coolant average temperature [°C]	310
Axial "buckling" (cell calculation)	0.
Axial "buckling" (core burnup calculation) A - burnup (MWd/kgHM)	$B_z^2 = 6{,}029\times10^{-5} + 1{,}226\times10^{-4}\times(1-e^{-0{,}3455A})$
For the Present Study	
Mixed oxide density [g/cm³]	9,4 (Th/Pu) 9,5 (Th/HEU)
HEU isotopic vector [w/o]	$^{235}U/^{238}U$ = 93,0/7,0
Pu isotopic vector [w/o] (fresh PWR-Pu)	$^{239}Pu/^{240}Pu/^{241}Pu/^{242}Pu$ 60,1 23,4 12,4 4,1

Table 2. Savings in Natural Uranium and Separative Work of Thorium Cycles Compared to the Current Uranium Once-through Cycle

Th/HEU

	Tail Enrichment	Natural Uranium Requirements [t U_{nat}/GWe.a]	Δ* [%]	SWU Requirements [tSWU/GWe.a]	Δ* [%]
Once-through	0,20	258,6	33	335,1	116
(3-batch)	0,25	286,6	34	306,7	124
Segregated Recycle	0,20	93,1	-52	120,6	-22
	0,25	103,2	-52	110,4	-19
Mixed Recycle	0,20	92,9	-52	120,4	-22
	0,25	102,9	-52	110,2	-22

$$^{*}\ \Delta = \frac{\text{Examined cycle requirement} - \text{U cycle once-through requirement}}{\text{U cycle once-through requirement}}$$

Th/Pu

	Saving in Natural Uranium and Separative Work Units [%]
Once-through	
. 3-batch	13
. 4-batch	15
Recycle	
. 3-batch	29
. 4-batch	29