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NON-PROLIFERATIVE, THORIUM-BASED, CORE AND FUEL CYCLE FOR PRESSURIZED WATER REACTORS

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ABSTRACT

Two of the major barriers to the expansion of worldwide adoption of nuclear power are related to proliferation potential of the nuclear fuel cycle and issues associated with the final disposal of spent fuel. The Radkowsky Thorium Fuel (RTF) concept proposed by Professor A. Radkowsky offers a partial solution to these problems. The main idea of the concept is the utilization of the seed-blanket unit (SBU) fuel assembly geometry which is a direct replacement for a "conventional" assembly in either a Russian pressurized water reactor (VVER-1000) or a Western pressurized water reactor (PWR).

The seed-blanket fuel assembly consists of a fissile (U) zone, known as seed, and a fertile (Th) zone known as blanket. The separation of fissile and fertile allows separate fuel management schemes for the thorium part of the fuel (a subcritical "blanket") and the "driving" part of the core (a supercritical "seed"). The design objective for the blanket is an efficient generation and in-situ fissioning of the U233 isotope, while the design objective for the seed is to supply neutrons to the blanket in a most economic way, i.e. with minimal investment of natural uranium.

The introduction of thorium as a fertile component in the nuclear fuel cycle significantly reduces the quantity of plutonium production and modifies its isotopic composition, reducing the overall proliferation potential of the fuel cycle. Thorium based spent fuel also contains fewer higher actinides, hence reducing the long-term radioactivity of the spent fuel.

The analyses show that the RTF core can satisfy the requirements of fuel cycle length, and the safety margins of conventional pressurized water reactors. The coefficients of reactivity are comparable to currently operating VVER's/PWR's.

The major feature of the RTF cycle is related to the total amount of spent fuel discharged for each cycle from the reactor core. The fuel management scheme adopted for RTF core designs allows a significant decrease in the amount of discharged spent fuel, for a given energy production, compared with standard VVER/PWR. The total Pu production rate of RTF cycles is only 30 % of standard reactor. In addition, the isotopic compositions of the RTF's and standard reactor grade Pu are markedly different due to the very high burnup accumulated by the RTF spent fuel.

INTRODUCTION

Most of the world's nuclear reactors use fuel in which uranium is enriched to less than 5 percent U235, the only naturally occurring, thermally fissile, isotope. However, there has long been interest in using thorium as a fertile material to produce U233, an isotope of uranium with the best neutronic properties, as a thermal reactor fuel. The major reasons proposed for the introduction of the thorium based fuel cycle had been:

a) Increasing the world's fissile resources by breeding U233 from thorium,
b) Improving fissile fuel utilization in thermal reactors,
c) Significantly reducing U235 enrichment requirements,
d) Decreasing production of Pu, and other transuranic elements compared to uranium fuel cycle,

e) Achieving higher fuel burnup than uranium based fuel cycles,
f) Decreasing production of toxic fuel waste or long lived radiotoxic waste.

Thorium based fuel can be used in all proven reactor types and in possible future reactor concepts. The thorium based fuel cycle and thorium fuels are relevant to countries which are having significant thorium deposits and small uranium reserves. The thorium based fuel cycle feasibility was demonstrated for high temperature gas cooled reactors (HTGR), light water reactors (LWR), pressured heavy water reactors (PHWR), liquid metal cooled fast breeder reactors (LMFBR), and molten salt breeder reactors (MSBR). All these activities have been well documented in several extensive publications published by International Atomic Energy Agency (IAEA) [1]. The main idea of the RTF concept is to provide a design which, without reprocessing, would achieve nonproliferation and waste reduction objectives while simultaneously saving natural U and hence extending natural U resources, and remaining cost-competitive. In addition, the once-through fuel cycle avoids complications associated with reprocessing and refabrication of highly radiotoxic U233 based fuels.

There are two ways of introducing thorium in current PWR cores: homogeneously and heterogeneously. The recent investigation of the homogeneously mixed thorium-uranium dioxide fuel cycle was proposed by Herring and MacDonald at INEEL [2]. The fuels produced by the heterogeneous approach can be divided into two sub-categories: macro-heterogeneous thorium-uranium fuel (i.e. so-called seed and blanket configurations) and micro-heterogeneous thorium-uranium fuel.

The concept of micro-heterogeneous Th-U fuel was studied by Bettis as a part of the Light Water Breeder Reactor (LWBR) program in the 1960's [3]. The recent studies of such fuel was accomplished in MIT in 2001 [4].

The representative of the macro-heterogeneous seed and blanket fuel concept are Whole Assembly Seed and Blanket (WASB) and Radkowsky Thorium Fuel (RTF) concepts and the main idea is the separation of the uranium and thorium fuel zones. Such a configuration of the fuel allows applying different management schemes for the uranium and the thorium part of the fuel and different in-core residence time periods.

The WASB concept has been proposed by (M.-H. Kim et al. 1999) [5] [6] for utilization in a PWR core with a high conversion ratio. The RTF concept was proposed by Dr. Alvin Radkowsky and Dr. Alex Galperin [7] [8] and is based in part on the ideas and experiences of the Bettis Atomic Power Laboratory's LWBR program. However, in contrast to the LWBR project, the RTF concept is based on a once-through fuel cycle with no reprocessing. The U233 is burnt in situ, and the fuel rods that contain the U233 are then disposed of. The RTF concept is based on a Seed – Blanket unit (SBU) fuel assembly which is direct replacement for a conventional fuel assembly. The central region of the assembly (seed) contains enriched uranium, while the peripheral region of the assembly (blanket) contains natural thorium spiked by a small amount of enriched uranium.

NUCLEAR DESIGN DESCRIPTION

The main idea of the concept is the utilization of the seed-blanket unit (SBU) fuel assembly geometry. The SBU geometry provides the necessary flexibility to satisfy a major design constraint - full compatibility with existing PWR/VVER power plants. In addition, the heterogeneity of the SBU design allows the needed, separate optimization of the seed and blanket lattices. The design constraints are summarized below:

a) The RTF concept should be realized as a new fuel design, and as such, be completely compatible with existing power plants. Only minor plant hardware modifications, directly related to a different fuel assembly internal arrangement, will be acceptable.
b) All safety and operational parameters of existing power plants will be preserved.
c) The fuel design will be based mainly on existing (not necessarily commercial) fuel technology.
d) No fuel reprocessing is assumed and the maximum allowable fresh fuel enrichment will be kept below 20 w/o of U235 content.

The SBU consists of two spatial regions as shown on Fig. 1 and 2. The internal supercritical region, called seed, contains the higher enriched U and external subcritical region, called blanket, contains a mixture of Th-U. The seed region occupies about 40 % of the assembly volume.

Two design approaches of introducing high enriched seed fuel in SBU were investigated for PWR thorium core. In first one, the seed fuel was chosen as U/Zr alloy rods, while the second is based on regular higher enriched U fuel in dioxide form. The design approaches of seed fuel for VVER thorium core were concentrated on U/Zr metal alloy fuel.

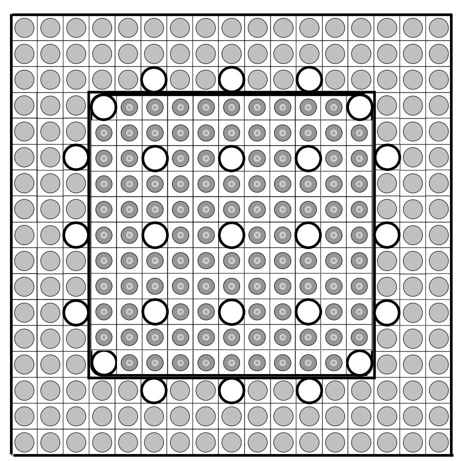

Figure 1. SBU Geometry for PWR

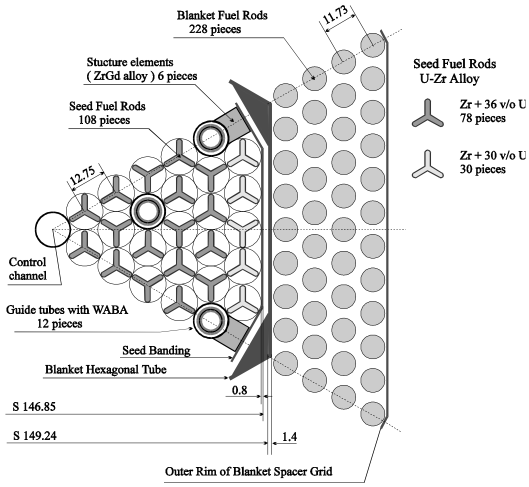

Figure 2. SBU Geometry for VVER

The moderator to fuel ratio (V_m/V_f) of the seed is about 3.5, while the moderator to fuel ratio of blanket is standard (~ 1.65). The seed fuel is 20 % enriched U. The design objective of such a design is to maximize the power production in the blanket region, whereas the main design idea of the seed region is to keep the reactor critical and to achieve the required cycle length, and in addition, to supply thermal neutrons to the blanket area. To supply neutrons to the blanket region in the most efficient way and to minimize Pu production, a high moderator to fuel volume ratio (V_m/V_f) in the seed region is needed. The high enrichment in the seed region is necessary to compensate for the relatively smaller amount of fissile uranium in the core and balance the thorium capture rate.

The size of the seed fuel rod and the unit cell geometry were determined by neutronic and heat removal considerations. The seed region produces about 60 % of the total power, therefore the power density in the seed is relatively high. The main consequence of this fact is that in all designs the seed fuel rod has central un-fueled region as shown on Fig. 3. The central plug allows to reduce fuel central line temperature. In addition, to improve heat removal, the seed rod for VVER has three-petal shape and axially twisted which allow eliminating the grid spacer in the seed region. The fuel rod spacing occurs by fuel rod axial twisting.

The blanket fuel contains a ThO2/UO2 mixture, while the UO2 content is 13 % volume and U enrichment is 12.2 % for PWR design, and UO2 content is 9.5 % volume and U enrichment is 20 % for VVER design. The uranium is added to the blanket fuel for two main reasons: 1) natural thorium does not include a fissile component, thus enriched uranium is required to provide a reasonable power density in the blanket during the initial burnup period of gradual U233 buildup, and 2) the addition of U238 assures that the U233 accumulated and discharged with the blanket fuel is sufficiently diluted to present no diversion potential. Additional uranium isotopes, created during the long in-core residence time of the blanket, are U232, U234, U235, and U236. These present another major natural barrier to the diversion of U233.

One of the novel features of the RTF concept is its in-core fuel management scheme. The fuel management scheme is different for the seed and blanket fuel parts. Seed fuel is managed similar to a standard PWR/VVER fuel, i.e. multi-batch reload with a 12 or 18 month cycle. The number of seed batches is 3. The blanket part of the fuel is treated as a whole single batch and resides in the core for a longer period, usually 6 seed reloads, which is equivalent to 9 years for an 18-month seed cycle.

The major design parameters of the SBU are summarized in Table 1.

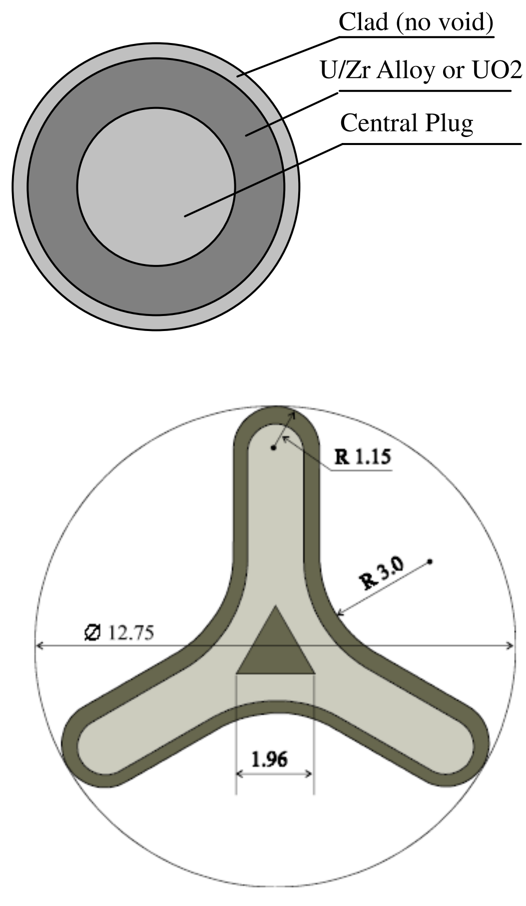

Figure 3. Seed Fuel Rod Geometry for PWR and VVER

Table 1. SBU General Description

Parameter	PWR		VVER	
	Seed	Blanket	Seed	Blanket
Fuel Material Composition	U/Zr metal alloy or UO2 U enrich.=20%	(U+Th)O2 U volume fraction is 0.13 U enrich.=12.2 %	U/Zr metal alloy U enrich.=20%	(U+Th)O2 U volume fraction is 0.095 U enrich.=20%
Number of Fuel Rods	108	156	108	228
Number of Guide Tubes	12	12	12	0
Fuel Rod Outside Radii, cm	0.475	0.475	Eq. radius 0.6375	0.42
Fuel Cell Pitch, cm	1.26	1.26	1.275	1.173
Moderator to Fuel Volume Ratio	~3.5 (~3 for oxide)	1.67	3.9	1.65
Lattice Geometry	Square		Triangular	

CALCULATIONAL TOOLS

All reactor physics calculations were carried out with:

1) PWR thorium core: the ELCOS code system [9] for the static simulation of light water reactor cores. The Eir-Lwr-COdeSystem ELCOS has been developed at the former Swiss Federal Institute for Reactor Research EIR, presently Paul-Scherrer-Institute, PSI. The system can also be used for the generation of most of the nuclear input data needed for the dynamic analysis of nuclear power plants.
2) VVER thorium core: the RECOL code [10] developed by Russian Research Center "Kurchatov Institute" (RRC-KI).

In order to get a comparison of the effect of different methods and data bases, benchmark calculations had to be performed before the start of the actual fuel cycle studies. The verification calculations of the ELCOS computation system were produced in 1994[11]. A PWR core of a current design was used as a test problem. The chosen core was relatively complex, including 11 fuel types, differing by fuel enrichment and burnable poison design. The results of the calculations were compared with utility data, and demonstrated the adequacy of the ELCOS system. It was found, that the relatively simple models are sufficiently accurate in simulating core behavior with accuracy of about 1 mK.

METHODOLOGY

Coupled neutronic and thermal-hydraulic fuel cycle calculations have been performed in few phases, as follows:

1) Seed and blanket fuel rod optimization.
2) Seed-blanket unit physics calculations, based on a two dimensional representation of the actual geometry and depletion characteristics of an individual fuel assembly under operating conditions. These second phase results provided homogenized cross sections, generally on a single assembly basis, to the code which simulates 3D reactor core.
3) The fuel management scheme, including the material weight flow, power density distributions and other neutronic parameters, was evaluated and analyzed. Power density calculations were carried out to identify the hot channel and the hot spot location within a core and to evaluate the average and the maximum power densities at this location. The power densities were then used to assess the basic safety-related thermal-hydraulic parameters, such as fuel temperature, clad

temperature, and departure of nucleate boiling ratio (DNBR) for the hot channel and hot-spot locations.

4) Investigation and analysis of the reactivity control system and reactivity coefficients. The following performance parameters and fuel design characteristics were considered: moderator temperature coefficient (MTC), Doppler coefficient, soluble boron reactivity worth, and control rods system (CR) reactivity worth.

FUEL CYCLE PERFORMANCE PARAMETERS

A complete core simulation in three dimensions including neutronic and thermal-hydraulic coupling was enacted to produce the performance parameters presented in this section.

In order to evaluate the RTF-based cycle it was necessary to simulate a number of seed reloads corresponding to a single blanket reload. The blanket fuel reload, which is carried out as a single batch every nine years, is designed to sustain the blanket K-inf in the vicinity of 0.9. It was found that this value is a compromise between an acceptable radial core power distribution and an optimal power division between the seed and blanket regions.

The U content for each seed reload is adjusted such that the required full power days (FPD's) are sustained, namely criticality is held for the 12 months or 18 months FPD's inter-refueling interval, with a capacity factor of about 0.9.

The Table 2 summarizes the SBU reload sequence for all cases, namely PWR thorium core and VVER thorium core.

The seed part of the fuel is managed similarly to a standard PWR/VVER cycle, i.e. multi-batch reloading. For a 3-batch scheme, adopted in all current RTF designs, all seed sub-assemblies are divided into 3 batches: fresh, once burned, and twice burned. At each reloading the twice-burned batch is replaced by a fresh batch of seeds, all remaining seeds are reshuffled to achieve an acceptable radial core power distribution.

The blanket part of the fuel is treated as a single batch and is reloaded following several seed reloads. Note that the blanket generally does not need to be shuffled; therefore, any shuffling that may occur is as a result of the mechanical characteristics of the SBU. A near-optimal in-core residence time for the Th-based blanket fuel was found to be 9 years. Thus, for an 18 months (1.5 years) seed cycle, the burnt blanket is reloaded as a single batch following 6 seed reloads. This fuel management scheme results in a relatively high blanket discharge burnup (approximately 80,000 MWd/t), which contributes directly to an improved fuel utilization, reduction of discharged fuel mass and volume, and a corresponding savings in fuel cycle costs.

The blanket criticality dependence on burnup creates a different neutron balance for each seed reload cycle, thus resulting in a slightly different seed cycle length. In principle, the length of each cycle may be adjusted to produce the required inter-refueling interval by adjusting the amount of fissile material (U235) loaded. It should be noted, that the U enrichment used in RTF fuel management is constant and equal to 20 %.

Table 2. Summary of SBU Reloads

Cycle #	Total U weight loaded (kg H.M.)		
	PWR 12 months cycle	PWR 18 months cycle	VVER 12 months cycle
1	8134	12356	4798
2	4052	6684	3398
3	3728	6684	3678
4	3728	6684	3678
5	4052	6684	3678
6	4295	7294	3678
1 – 6 Blankets		6915	
7	4295		3764
8	4052		3850
9	4295		3893
1- 9 Blankets	6915		4400

The blanket fuel multiplication factor (K-inf) is a relatively weak function of the accumulated burnup. As noted above, as a rule, the blanket subassemblies are generally not shuffled within the core; exceptions are the first two or three transition cycles in which such shuffling is required to prevent some local power peaking. It should be noted that the blanket reactivity dependence on burnup is quite flat following the buildup of U233. The reactivity values of the individual seeds are quite close, therefore reshuffling of the partially-depleted seeds into different blankets, followed by reshuffling of the blankets is a procedure usually not required to assure an acceptable power distribution. However, such possibility exists in principle and may be pursued to flatten the power density distribution across the core, especially for the first two cycles. The present study assumes an average burnup for all blankets and no blanket reshuffling.

The RTF cycle reactivity control system is identical to that of a standard PWR/VVER and is based on a combination of three independent methods: burnable poisons (BP), soluble poison and control rods.

The reactivity control of the core during the power production cycles was obtained by a combination of BP and varying the concentration of boric acid dissolved in the coolant water. The fissile loading was adjusted to assure inter-refueling intervals of 300 FPD’s for the 12 months cycle and 480 FPD’s for the 18 months cycle. Thus, the core criticality is maintained during burnup by adjusting the critical boron concentration and the EOC state is that time point at which the soluble boron concentration is zero. The critical boron concentration curves for cycle 4 are presented below in Figs. 4. The plotted critical boron concentrations start with equilibrium Xe and are given as functions of full power days.

The power distribution is an essential part of the nuclear design; it determines the thermal-hydraulic feasibility of the concept. For the RTF design the power distribution analysis is complicated by the heterogeneity of the fuel assembly itself. The power distribution calculations are carried out to identify the hot channel and the hot spot location within a core and evaluate the average and the maximum power densities at this location. The power densities are then used to evaluate the basic safety-related thermal-hydraulic parameters: fuel temperature, clad temperature, and departure of nucleate boiling ratio (DNBR) for the hot channel and hot-spot locations.

The analysis of the RTF core performance shows that the radial power peaks are similar to typical values for a standard PWR/VVER core and do not exceed 1.4. On the other hand, the axial power peaks are slightly higher than 1.5, the typical value. This problem can be eliminated by improvement in the radial configuration of the BP rods, as well as by axial zoning of the BP rods. The latter measure will contribute also to reducing the maximum linear power in the fuel.

The analysis of the reactivity worth coefficients shows that RTF moderator temperature coefficient (MTC) values are similar to those of a standard PWR/VVER core. Taking into account possible variations of the soluble boron the MTC value range for all operating conditions is -20 to -35 pcm/^{o}C. The range of the boron worth coefficient values for the RTF designs are in the range of -6 to -10 pcm/1 ppm, and Doppler coefficient (DC) values of the SBU at nominal operating

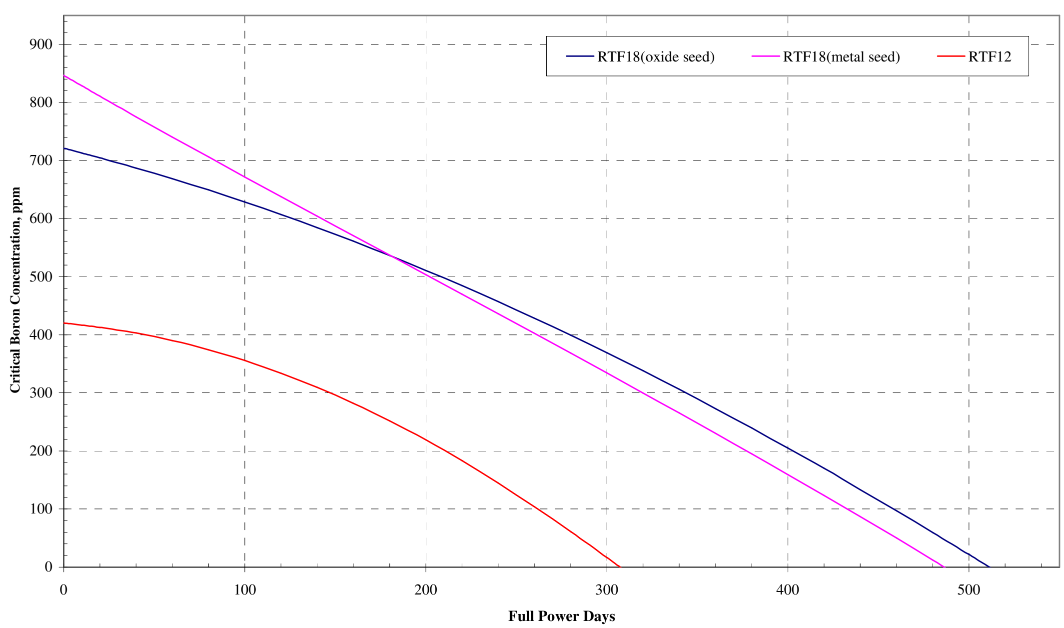

Figure 4. Critical Boron Concentration, cycle 4

conditions are within a range similar to that of a standard PWR/VVER core, i.e. -2 to -5 pcm/^{o}C.

The control rod (CR) reactivity worth values do not provide a sufficient solution for the reactivity control of the RTF cycles. The control rod worth could be further improved if higher enrichment B-10 is used for the poison material.

The total burnup of the discharged seed fuel is estimated to be in the range of 140 to 150 GWd/t. The average burnup of the discharged blanket fuel accumulates approximately to 80 GWd/t. The heterogeneous approach of rods divided between seed and blanket necessitates in-core residence times and burnups significantly higher than those encountered in present operating experience. Although, the relatively low power in the blanket, and the robust characteristics of thorium oxide, are expected to mitigate some of these concerns, the situation with the seed is considerably more challenging.

The major feature of the RTF cycle is related to the total amount of spent fuel discharged each cycle from the reactor core. The fuel management scheme adopted for RTF core designs allows a significant decrease in the amount of discharged spent fuel , for a given energy production, compared with a standard PWR/VVER. There are two separate fuel flows: the three-batch scheme is used for the seed, while the single-batch scheme is used for the blanket. At each refueling period, one third of all seed sub-assemblies are discharged with the total weight of 3.1, 3.2, and 4.7 tons respectively for the VVER (12 months cycle) and PWR with 12 and 18 months cycles. At the end of the ninth year, all blanket sub-assemblies are discharged with the total weight of about 43 tons. Table 3 shows the average amount of U (U+Pa for the blanket) and the Pu discharged each cycle.

The Pu inventory and isotopic mix in discharged fuel is the major concern related to proliferation potential. The RTF fuel cycle produces two separate material flows: seed and blanket, each with its characteristic material composition. The fissile Pu (Pu239 + Pu241) annual production rates in the PWR with 12 months and 18 months cycles are, respectively, 23 % and 30 % of the corresponding rate for a standard PWR. The total Pu production rate of RTF cycles is only 30 % of standard PWR and VVER. In addition, the isotopic compositions of the RTF's and PWR/VVER Pu are markedly different due to the very high burnup accumulated by the RTF spent fuel. The Pu238 content of the PWR/VVER spent fuel is about 2 %, not presenting a significant barrier to diversion, while the Pu238 contents of the RTF spent fuel is about 6 % for VVER and PWR seed (except for the first two transient cycles). The blankets Pu238 concentration is about 9.5 % for PWR blanket and is about 14.5 % for the VVER blanket.

Table 3. Average amount of discharged U + Pu

Cycle #	Total weight discharged (kg H.M.)		
	PWR 12 months cycle	PWR 18 months cycle	VVER 12 months cycle
1	1806.4	2297.5	679.4
2	2299.1	3760.2	1356.7
3	3034.8	4986.6	1961.4
4	3467.3	5740.8	2863.3
5	3185.4	5773.2	3121.1
6	3186.7	5864.7	3125.2
1 – 6 Blankets		6488.0	
7	3493.2		3147.2
8	3723.8		3150.6
9	3800.4		3245.5
1- 9 Blankets	6439.0		4013.9

EXPERIMENTS

Efforts to develop and demonstrate the RTF concept have been focused at the RRC-KI in Russia. Efforts are aimed at eventually testing full-scale SBU in an operating VVER-1000. Thermal-hydraulic experiments and irradiation of fuel samples at the IR-8 reactor at RRC-KI have also been performed.

One of the main challenges of the SBU approach is to ensure adequate heat removal from the seed region. Several experiments were performed at the KS thermal-hydraulic test facility at RRC-KI to validate thermal-hydraulic calculational tools, and confirm the feasibility of the proposed design. The facility reproduces the thermal-hydraulic conditions in a VVER reactor under normal, transient and accident conditions. Tested configurations included part-length and full-length rods, and clusters ranging up to 19-rods. The main parameters measured were related to hydraulic resistance and critical power. The results show that the thermal-hydraulic performance of the seed part is bounded by that of a conventional VVER-1000 fuel assembly.

In addition, mechanical and thermal hydraulic test were performed on full cross-section of reduced length SBU at OKBM Gidropress. Thorium-oxide/uranium-oxide pellets for the blanket were fabricated and have been irradiated in the IR-8 reactor at RRC-KI.

SUMMARY

Results of the study presented in this paper demonstrate the feasibility of a thorium-based fuel cycle for pressurized water reactors of current technology. The problem of a relatively high power density within a seed region is addressed by utilizing chosen fuel rod geometry for the seed region for PWR/VVER core. The RTF concept shows better economic potential than the homogeneous mixture approach based on preliminary estimates of fuel cycle costs, and is comparable to those of a conventional PWR. While the fuel has to be designed to withstand very high burnup, above 100 GWD/T, no serious problems have been identified in the technical, safety and licensing performance of this fuel.

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