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Thorium Fuel for Light Water Reactors—Reducing Proliferation Potential of Nuclear Power Fuel Cycle

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The proliferation potential of the light water reactor fuel cycle may be significantly reduced by utilization of thorium as a fertile component of the nuclear fuel. The main challenge of thorium utilization is to design a core and a fuel cycle, which would be proliferation-resistant and economically feasible. This challenge is met by the Radkowsky Thorium Reactor (RTR) concept presented in this paper. So far the concept has been applied to a Russian design of a 1,000 MWe pressurized water reactor, known as a VVER-1000, and designated as VVERT. The following are the main results of the preliminary reference design:

- The amount of plutonium contained in the RTR spent fuel stockpile is reduced by 80 percent in comparison with a VVER of a current design.
- The isotopic composition of the RTR-Pu greatly increases the probability of preinitiation and yield degradation of a nuclear explosion.
- An extremely large Pu-238 content causes correspondingly large heat emission, which would complicate the design of an explosive device based on RTR-Pu.
- The economic incentive to reprocess and reuse the fissile component of the RTR spent fuel is decreased. The once-through cycle is economically optimal for the RTR core and cycle.
- To summarize all the items above: the replacement of a standard (uranium-based) fuel for nuclear reactors of current generation by the RTR fuel will provide a strong barrier for nuclear weapon proliferation. This barrier, in combination with existing safeguard measures and procedures is adequate to unambiguously disassociate civilian nuclear power from military nuclear power.

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- The RTR concept is applied to existing power plants to assure its economic feasibility.
- Reductions in waste disposal requirements, as well as in natural uranium and fabrication expenses, as compared to a standard VVER fuel, provide approximately 20 percent reduction in fuel cycle cost.

BACKGROUND

Nuclear power plants (NPPs) produce energy by "burning" nuclear fuel. The nuclear fuel presents a complex engineering product and is a subject of extensive research and development studies. A separate discipline within the nuclear engineering domain, Nuclear Fuel Management, is devoted to analysis of nuclear fuel cycle activities.

The stages of the fuel cycle, typical for a Light Water Reactor (LWR) Plant of current technology, are depicted in figure 1. Naturally-occurring uranium (U) element is mined and processed to produce U_3O_8 concentrate (yellow cake). At the next stage, the yellow cake concentrate is converted into gaseous form and is enriched to a U-235 content level required by the core design and operational requirements. Enriched uranium is then reduced to UO_2 powder, which is used to fabricate fuel elements and finally fuel assemblies.

The fuel assemblies are loaded into a reactor core and are "burned" during the power production period. The loading pattern of fuel assemblies and reshuffling strategies constitute a subdomain of Nuclear Fuel Management, namely in-core Fuel Management. All stages preceding the loading into a reactor core are designated as a front-end of the fuel cycle.

Nuclear fuel is produced from uranium or thorium (Th) which are naturally occurring elements. Natural uranium contains a fissile component, U-235, and a fertile element, U-238, while natural thorium contains only a single fertile isotope Th-232. Therefore, uranium, enriched in U-235, is a standard fuel for the Light Water Reactors (LWR) of current technology. Additional fissile isotopes generated by transmutation of fertile isotopes are Pu-239 and Pu-241 (uranium chain) and U-233 (thorium chain). These isotopes play an important role in the process of generating fission energy during the fuel burnup process; these artificial fissile isotopes are created and "burned" in situ, contributing much of the energy generated by a nuclear fuel.

During the first period, following discharge from a reactor, the spent fuel is stored in water pools at a reactor site. Spent fuel storage racks are placed under water in the fuel storage building adjacent to the reactor building. These racks hold the assemblies and maintain the required spacing between

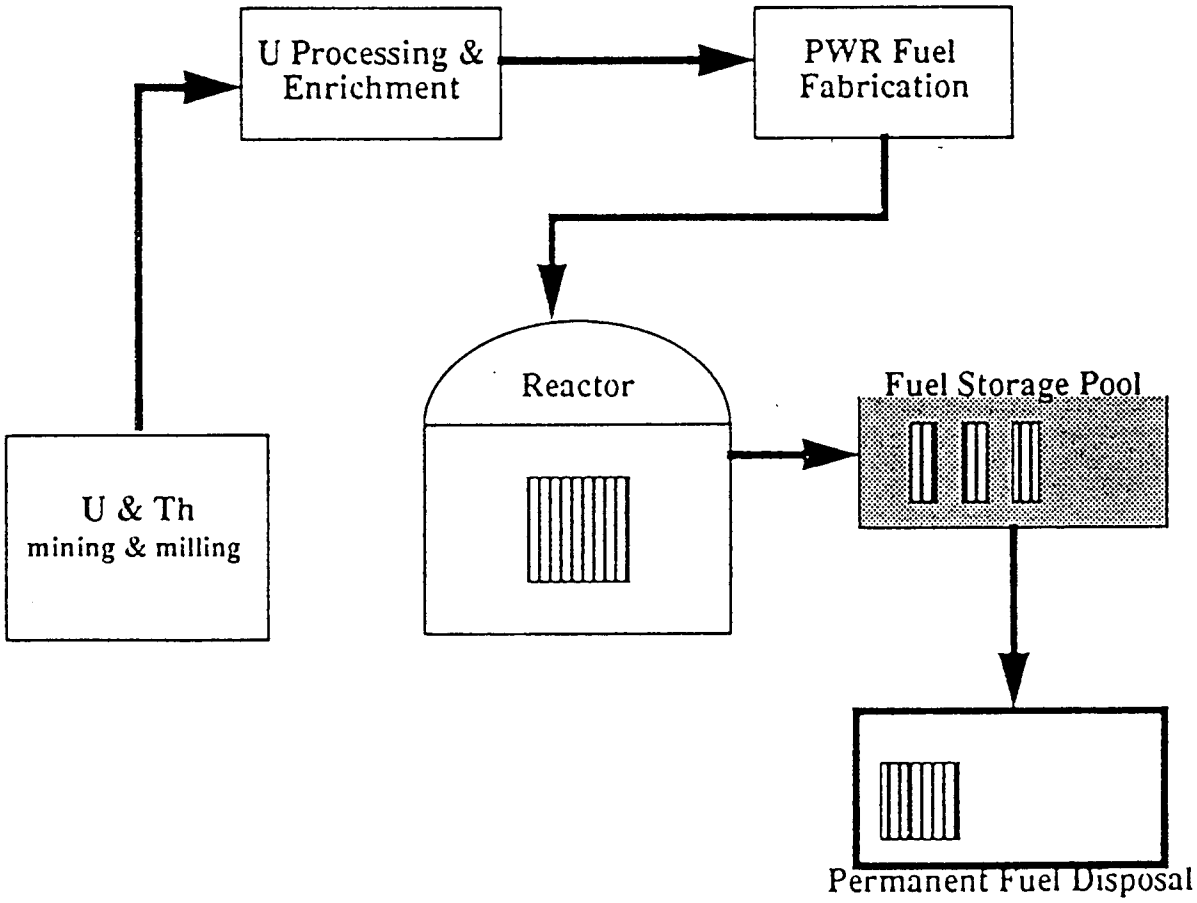

Figure 1: LWR fuel cycle stages without fuel reprocessing.

assemblies to provide criticality control and residual heat removal. Short-lived radioactive isotopes decay during the first several months (up to several years), which leads to a major reduction of the spent fuel radioactivity and heat emission levels. Finally, when the on-site storage pools are filled-up, the spent fuel assemblies would nominally be transported to and deposited into a long-term (permanent) fuel storage facility. All stages following the discharge of the fuel from a reactor core are designated as the "back-end" of the fuel cycle.

It should be noted that fissile isotopes created within the nuclear fuel during the power production period may be separated by chemical methods from the discharged nuclear fuel at one of the back-end stages. The fissile material

separated from the burnt nuclear fuel may be added as a component to fresh fuel and reinserted into the reactor core. Clearly, this path of reprocessing spent fuel assures maximum fuel utilization efficiency.

Nevertheless, the spent fuel reprocessing option is not exercised in most countries using nuclear power. This is due to economics, and to concerns about the proliferation potential of the plutonium (Pu) that is separated from spent fuel. While not an ideal weapons material, this plutonium might potentially be diverted to weapons use. This potential proliferation of nuclear weapon material, produced as a by-product of the nuclear power plant fuel cycle, is responsible for much public concern and may be one of the major obstacles to worldwide expansion of nuclear power.

In order for nuclear power to be accepted as a main source of energy in the next century, it must be based on a fuel cycle which is highly proliferation-resistant. The nonproliferative nature of the nuclear power fuel cycle material flow should be supported not only by a combination of administrative safeguard measures, but mainly by avoiding production of any material of such quantity and quality as to be of potential weapons use.

The extensive Nonproliferation Alternative Systems Assessment Program (NASAP) studies concluded in 1980 that none of the existing or proposed fuel cycle schemes were immune to the possibility of proliferation. Due to the fact that the main proliferation potential is associated with plutonium (Pu), created by transmutation of U-238, thorium presents a natural alternative fertile material.

THORIUM AS A NUCLEAR FUEL

It was noted at an early stage of nuclear technology development that U-233 presents a superior fissile nuclide producing more neutrons per thermal neutron absorbed than all other fissile isotopes. This feature, and the fact that thorium is much more abundant as a natural ore than U prompted numerous attempts to design and implement a nuclear reactor based on thorium fuel. The most notable examples are the Light Water Breeder Reactor (LWBR) and early High Temperature Gas Cooled Reactor (HTGR).

The main challenge encountered in the design of a thorium-based system is the necessity to supplement natural thorium with a pregenerated fissile component. Several design solutions were proposed and investigated, such as the initial start-up of the thorium cycle by enriched uranium, the continuous addition of uranium as a fissile component to supplement self-generated U-233, the reprocessing and recycling of U-233, and the addition of plutonium to supplement self-generated U-233.

The improvement in natural uranium utilization by using thorium could be achieved only if the self-generated U-233 fissile material was separated and recycled into a closed fuel cycle. This approach, adopted by the LWBR, violated the non-proliferation requirement.

Under irradiation, thorium undergoes a rapid increase in U-233 concentration (and k_∞) and retains a value higher than that of plutonium in uranium lattice. Because of the very large fission cross section of plutonium, uranium builds up in plutonium (and k_∞) early in life, but this quantity saturates and declines much sooner than in case of U-233 in thorium. This difference results from the fact that the equilibrium concentration of plutonium is low, by virtue of its high cross section and because of the high capture to fission ratio, α. For thermal neutrons, α(U-233) = 0.102 and α(Pu-239) = 0.339. [1]

Thus, an efficient utilization of thorium in a once-through cycle encounters a "neutron economy" problem: the U-233 build-up process is quite slow (compared with the plutonium build-up of the uranium chain), reaching saturation at a burnup of about 40 GWd/t. During the long build-up the subcritical thorium part of the fuel requires continuous "investment" of neutrons created by fissioning U-235, i.e. a large initial resource investment in uranium. In order to "recover" this investment in terms of fuel utilization gains by taking advantage of superior U-233 properties, the thorium-based fuel must be burned further, to a burnup of at least 70–80 GWD/t, corresponding to 8–9 full-power years.

Thus, the main challenge of efficient utilization of thorium in LWRs is reduced to a problem of achieving very large accumulated burnup of the thorium in a once-through fuel cycle.

It should be noted that, similar to plutonium created by transmutation of U-238 (fertile) isotope, another fissile isotope (U-233) is created by transmutation of Th-232. While pure U-233 is by itself an efficient fissile material and therefore a diversion risk, it may be easily denatured (neutralized) by an addition of a relatively small amount of natural U. Additional non-fissile uranium isotopes created within the thorium transmutation chain, such as U-232, U-234, and U-236 present major natural barriers to diversion of U-233.

THE RADKOWSKY THORIUM REACTOR CORE (RTR) SOLUTION

The RTR concept proposed by Professor A. Radkowsky offers a solution to the thorium utilization problem. The basic idea is to use the heterogeneous, seed/blanket (SBU), fuel assembly. The thorium part of the fuel assembly is sepa-

rated from the uranium part of the assembly. This separation 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 of the blanket is efficient generation and in-situ fissioning of the U-233 isotope, while the design objective of the seed is to supply neutrons to the blanket in the most economic way, i.e. with minimal investment of natural uranium.

The SBU geometry provides the necessary flexibility to satisfy a major design constraint: full compatibility with existing pressurized water reactor (PWR) power plants. In addition, the heterogeneity of the SBU design allows the necessary (and separate) optimization of seed and blanket lattices.

The RTR core may be, in principle, implemented in all light water and heavy water reactors, but the main design effort was concentrated initially on the pressurized-water-reactor type plants. Two pressurized-water-reactor (PWR) design variations are considered: the hexagonal geometry (Russian VVER) design and the square geometry (western PWR) design.

Replacing a standard (U) fuel by the RTR fuel may be justified by the following advantages:

- significant reduction, or if possible, elimination of the fuel cycle proliferation potential;
- reduction of the spent fuel storage/disposal requirements; and
- fuel cycle cost saving.

The design constraints are prescribed mainly by considerations of technical and economic feasibility. These constraints are imposed to support economic justification of the research and development activity required to design, verify, license and implement the RTR fuel within a reasonably short period of time. The design constraints are summarized below:

(i) The RTR concept should be realized as a new fuel design, and thus, be completely compatible with existing power plants. Only minor plant hardware modifications directly related to a different fuel assembly internal arrangement will be acceptable.

(ii) All safety and operational parameters of existing power plants will be preserved.

(iii) The fuel design will be based mainly on existing (although not necessarily commercial) fuel technology. The maximum allowable fresh fuel enrichment will be kept below 20 percent of U-235 content.

The reference design of the RTR core and fuel cycle was carried out based on objectives and constraints discussed above. The VVER option of the RTR (VVERT) is described in detail, and performance parameters related to design objectives and constraints are discussed in following sections.

VVERT DESIGN

The VVERT design is an implementation of a RTR fuel reload for a standard VVER-1000 core, where "T" stands for thorium.[2] The VVERT core is identical to an existing VVER-1000 core with 163 hexagonal fuel assemblies and 3,000 MWth power output. The average power density of 106 kW/liter is somewhat higher than that of a similar PWR core of a western design. A layout of the VVERT core, the hexagonal SBU and a corresponding square SBU (for the PWRT version) are shown in figure 2.

The VVERT fuel assembly, SBU, consists of two spatial regions: an internal region (seed) and external region (blanket). The design objective of such an arrangement is to maximize the power production of the blanket region. The seed region volume and consequently its power share are minimized subject to two constraints: (1) the total amount of uranium loaded in seed each cycle should sustain (drive) the subcritical blanket for a given inter-refuelling interval, and (2) the total surface of seed fuel should be adequate to sustain required temperature and heat flux values. The seed fuel was chosen as U/Zr alloy rods, which is consistent with fuel technology capabilities of the fuel vendor industry of the Russian Federation.[3] The size of the seed rod and unit cell geometry were determined by consideration of neutronic and heat removal aspects as mentioned above.

The blanket fuel considered in present design was thorium oxide with addition of uranium oxide. The uranium was added to blanket fuel for two main reasons: (1) since natural thorium does not include a fissile component, enriched uranium is required to provide a reasonable power density in the blanket during the initial period of U-233 buildup, and (2) addition of U-238 assures that U-233 accumulated and discharged with the blanket fuel is sufficiently diluted to present no diversion potential.

IN-CORE FUEL MANAGEMENT

One of the novel features of the RTR concept is its in-core fuel management scheme. The standard multi-batch fuel management of a PWR is replaced by a more complicated scheme, based on two separate fuel flow routes: seed route and blanket route.

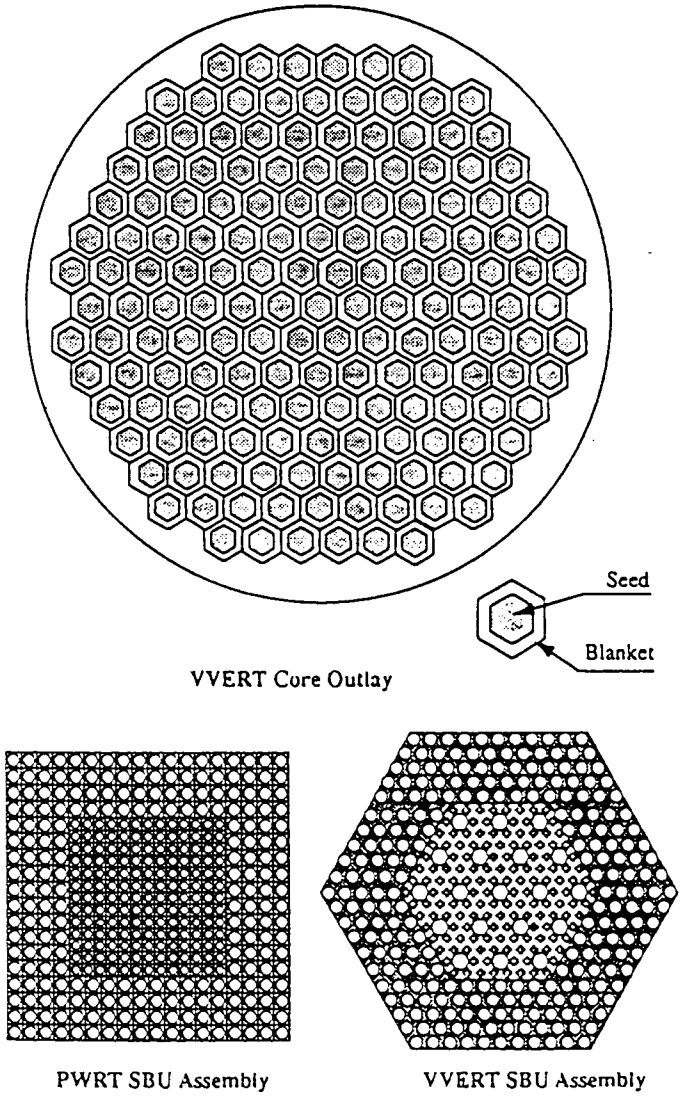

Figure 2: VVERT core outlay and SBU geometry.

Basically, seeds are treated similarly to the standard PWR assemblies, i.e., approximately one-third of seeds are replaced annually by "fresh" seeds, and the remaining two-thirds (partially depleted) seeds are reshuffled. Each seed is loaded into an "empty" blanket, forming a new fuel type. These new fuel type (fresh) assemblies are reshuffled together with partially depleted SBU assemblies to form a reload configuration for the next cycle.

It should be noted that the main difference between the VVERT fuel and a standard VVER (PWR) fuel is the introduction of thorium into the fuel cycle material flow. The thorium is loaded into the blanket part of the SBU. For rea-

sons of fuel economy the Th-blanket in-core residence time is quite long (about 10 years), while the uranium part of the SBU (seed) is replaced on a annual (or 18 month) basis, similar to standard PWR fuel management practice. The long blanket residence time is required to achieve very large accumulated burnup of the thorium part of the fuel, about 100 GWd/*t* (or 10 GWd/*t* on average for each annual cycle).

Optimization of the seed and blanket lattice parameters was aimed at efficient generation and in-situ burning of U-233 (in blanket). This optimization resulted in moderator-to-fuel volume ratios (V_m/ V_f) of 3.2 and 1.9 in seed and blanket regions respectively. The relatively high V_m/V_f in the seed was chosen in order to reduce the epithermal absorption of U-238 and consequently the buildup rate of plutonium isotopes. The blanket V_m/V_f value was chosen as an optimal intersection point of two processes: U-233 buildup (low V_m/V_f) and efficient fission (high V_m/V_f). The resulting value of 1.9 was found almost identical to that of a standard PWR core lattice.

The in-core fuel management adopted for the VVERT cycle was based on a quasi 3-batch reload scheme. One third of all seed subassemblies are replaced annually by fresh seeds, while the remaining two thirds are left within corresponding blanket subassemblies and reshuffled as partially depleted fuel assemblies. Thus, the in-core residence time of a seed subassembly is three years. The blanket subassemblies are burned for ten years, with a fresh seed reinserted every three years. The reload configuration was generated by considering fuel assemblies of three batches: fresh (F), once-burned (O), and twice-burned (T). An "equilibrium" reload pattern, shown in figure 3 demonstrates a typical "low-leakage" configuration repeated for each of the ten seed cycles. For the one sixth of the VVER core shown in figure 3A, no fresh assemblies were loaded in peripheral positions, i.e. at reflector boundary. Most of the peripheral positions are occupied by once-burned fuel. The fresh fuel assemblies are scattered in core inner positions in a near checkerboard pattern. The power density distribution across the core is shown in figure 3B and displays a typical PWR core power map. The variations of power density from beginning of cycle (BOL) and end of cycle (EOL) are also typical for a PWR core. It should be noted that placing fresh fuel into inner core positions requires utilization of burnable poisons to compensate for the local power peaking. The burnable poisons utilization is an integral part of the reactivity control design and is discussed in the next section. The assembly-averaged power peaking factors shown in figure 3 are well within ranges typical for a PWR core.

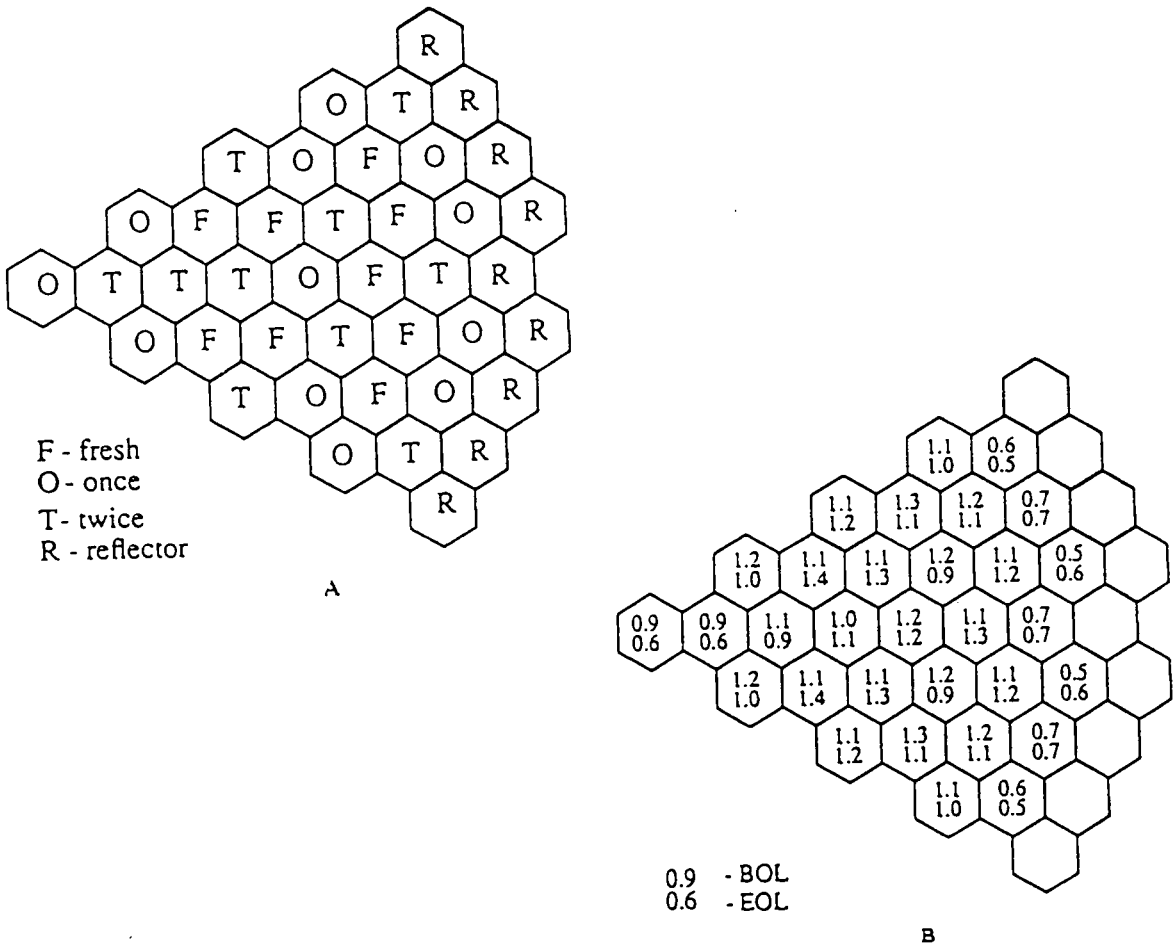

Figure 3: VVERT reload pattern (A) and power distribution (B).

REACTIVITY CONTROL SYSTEM

The reactivity control system of the RTR core is based on a burnable poisons and control rods system, without utilization of the soluble-boron method. The elimination of the soluble-boron system was necessitated by considerations of power sharing between seed and blanket parts of the core. The blanket power share is maximized in the RTR design in order to achieve a maximum power output from the blanket, i.e. thorium component of the fuel. Taking into account that blanket power share is proportional to $1/(1-k_b)$, where k_b is the

blanket multiplication factor, it is clear that a reactivity control mechanism should be designed in such a way that only the seed region of the SBU will be poisoned and thus, the blanket multiplication factor is not reduced.[4]

This consideration excludes utilization of the soluble poison system for the RTR core, leaving only burnable-poison and control-rod methods for reactivity control. This combination is not unusual and is used in control of Boiling Water Reactors. The burnable poison rods are used extensively in the VVERT design in order to compensate for a major part of the burnup related excess reactivity.[5] Two types of burnable poisons rods were used: the WABA type (Westinghouse Advanced Burnable Absorber) and Gd-loaded fuel rods. The WABA, presenting a standard PWR technology, was introduced to compensate the long-term burnup-related excess reactivity (from BOL to about 2/3 of cycle), and the Gd bearing rods were introduced to compensate the excess reactivity at the first 30–50 days of the fuel burnup interval. Resulting curves of typical excess reactivity for cycles 5 and 6 are shown in figures 4a and 4b respectively. The two (almost) identical curves demonstrate that the reactivity control requirements are identical for all cycles and that the total burnup-related reactivity shift is about 5 percent $\Delta\rho$. This value is about half of a typical value for a PWR.

The soluble system components of the plant will be used only for refuelling operations (when reactor cavity is filled with borated water) and possibly for cold-to-hot reactivity compensation as well as an emergency shutdown method. All remaining control functions will be carried out by control rods.

FUEL CYCLE MATERIAL FLOW

The analysis of the core behavior and fuel cycle performance includes a complete 3–D simulation of the reactor core and nuclear fuel during the power production cycle and following the fuel discharge. The fuel burnup and the corresponding change of the fuel material composition are calculated and serve as a basis for analysis of the back-end of the fuel cycle .

The fuel reload weight for each of the ten seed cycles (corresponding to a single blanket lifetime) was adjusted to assure an annual inter-refuelling period, i.e., 300 full power days, assuming a capacity factor of 0.82. The front-end fuel cycle requirements, obtained from the fuel weight and core design data, are summarized in table 1. It may be noted that the cycle 1 reload was divided into three enrichment groups in order to obtain reasonable power distribution for the first core and to serve as a transition to an equilibrium three-batch fuel management scheme.

Table 1: Front-end fuel cycle requirements.

Cycle	U weight kg H.M.	Enrichment U-235 percent	Natural U kg	Separative work[a] (kg swu)	Number of fuel rods per reload
1	2,894	20	112,103	244,454	32,274 (seed 1)
1	2,411	17	79,264	92,142	33,252 (blanket)
1	1,595	12	36,858	41,151	–
Total cycle 1	**6,900**		**228,224**	**377,748**	**65,526**
2	3,299	20	127,844	150,937	10,890
3	3,616	20	140,127	165,438	10,692
4	3,616	20	140,127	165,438	10,692
5	3,616	20	140,114	165,422	10,890
6	3,616	20	140,127	165,438	10,692
7	3,616	20	140,127	165,438	10,692
8	3,781	20	146,529	172,996	10,890
9	3,736	20	144,799	170,954	10,692
10	3,736	20	144,799	170,954	10,692

a. 0.2 enrichment tails are assumed.

The fuel cycle cost savings of approximately 20 percent may be derived from the table 1 data. The amount of natural uranium per cycle is about 140 Mt compared with 170 Mt of a standard PWR, and the number of fuel rods fabricated is 11,000 compared with 15,000 rods per PWR annual reload. The fabrication cost of a metal alloy fuel rod (RTR-seed fuel), produced by an extrusion process, is significantly lower than that of a PWR oxide rod. These three components of the front-end, as well as back-end savings in spent fuel storage expenses, result in a 20 to 25 percent reduction in an overall fuel cycle cost. This estimate is supported by a detailed fuel cycle cost calculation.[6]

The basis for the back-end of cycle analysis is the amount (weight) and composition of the fuel discharged from the core. This fuel is discharged, stored, and finally disposed of in a permanent storage facility. Its fissile con-

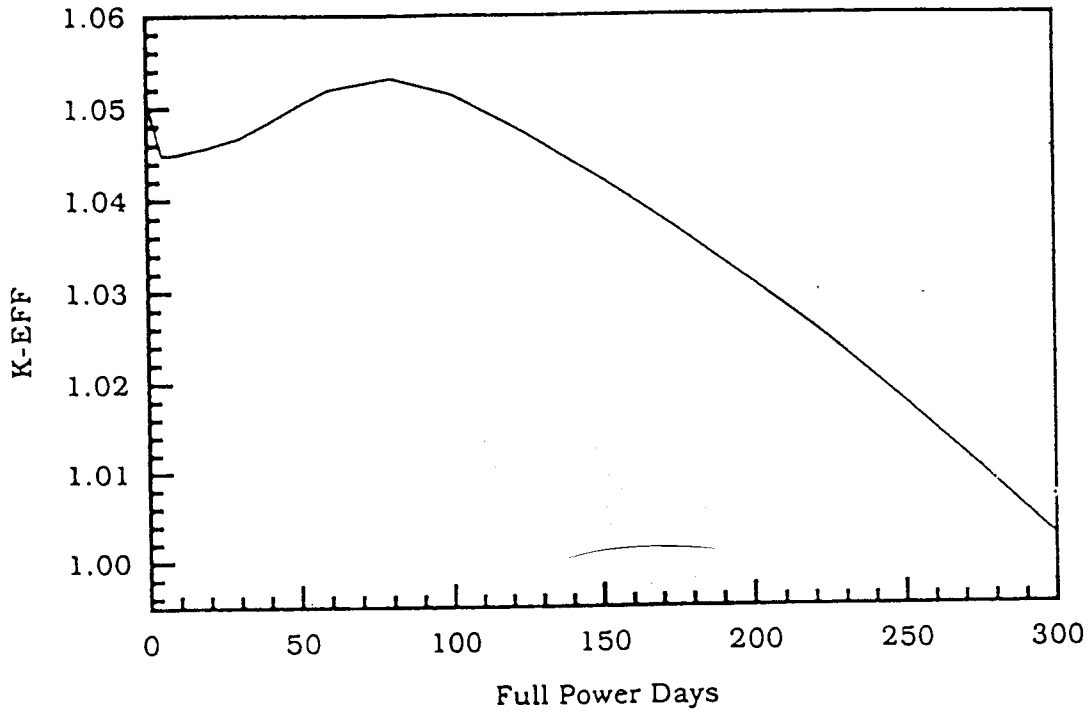

Figure 4a: Core criticality—cycle 5.

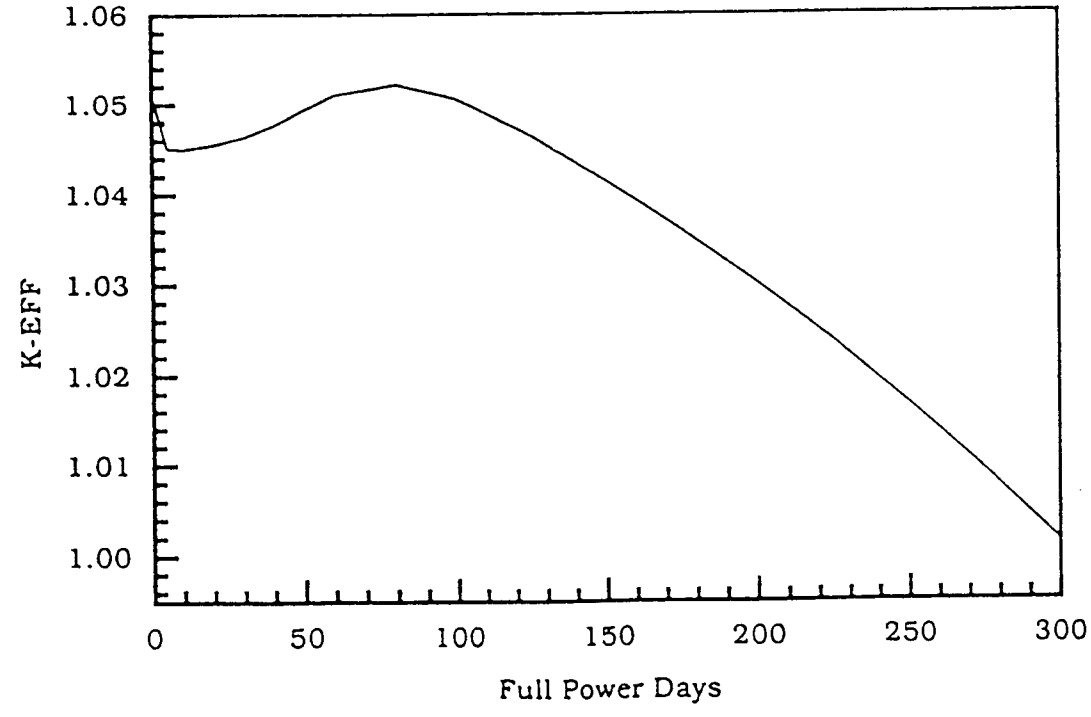

Figure 4b: Core criticality—cycle 6.

Table 2: Discharge fuel inventory and fissile content.

Discharge from cycle #	Seed weight (kg H.M.)	Blanket weight (kg H.M.)	Total U weight (kg)	U-235 weight (kg)	U-233[a] weight (kg)	Total Pu weight (kg)
1	1,595	-	1,445	49.6	-	12.9
2	2,411	-	2,072	87.0	-	24.1
3	2,894	-	2,385	92.4	-	31.8
4	3,299	-	2,732	119.9	-	38.1
5	3,616	-	3,003	140.6	-	42.5
6	3,616	-	3,018	152.8	-	42.5
7	3,616	-	3,059	156.6	-	43.3
8	3,616	-	3,021	155.6	-	43.0
9	3,616	-	3,023	157.9	-	43.0
10	3,781	32,370	3,132	171.1	635.0	36.6[b]
average per year	3,206	323.7	2,689	128.4	63.5	48.4[c]

a. -U-233 weight including Pa-233.
b. -Plutonium from the seed spent fuel.
c. -Plutonium from the seed spent fuel plus blanket spent fuel.

tent, principally the amount and composition of the plutonium, defines the proliferation potential of the fuel cycle. The isotopic composition of the spent fuel stockpile defines its radioactivity and heat emission levels, as well as the overall toxicity of the spent fuel as a function of time. The summary of the spent fuel annual discharge weights and fissile contents are presented in table 2.

The data presented in table 2 show that the plutonium discharged from the seed averaged over the blanket life is 36.6 kg (seed plutonium), and the plutonium contained in the discharged blanket is 118 kg or equivalent to 11.8 kg/year (blanket plutonium). The isotopic composition of this plutonium is presented and analyzed in the following section.

Another important parameter derived from the table 2 data is that the average annual U-235 discharge is 128 kg, compared with approximately 750 kg of U-235 loaded annually into the core. Thus, about 83 percent of initial fissile material loaded into the RTR-seed is depleted. This value is of crucial importance concerning the fuel resources utilization.

The VVERT design analysis demonstrated that the total natural uranium consumption of the RTR is about 20 percent lower than that of the corresponding VVER–1000 (uranium cycle).

SPENT FUEL REPROCESSING OPTION

Chemical reprocessing aims at recovering the fissile value of the spent nuclear fuel while removing the wastes, i.e. neutron absorbing isotopes. The uranium and plutonium isotopes found in the spent fuel of a standard LWR fuel cycle present a considerable economic value in terms of natural uranium and separative work. However, spent fuel reprocessing was abandoned in the United States and several other countries with developed nuclear industry, mainly because of the public concern associated with possible diversion of the separated fissile material for military uses. An additional reason for abandoning fuel reprocessing was a drastic increase in cost of reprocessing due to changing regulatory requirements and other factors.[7]

Nevertheless, a closed fuel cycle is being implemented in several countries (e.g. Japan, France, Switzerland) by utilization of mixed-oxide (MOX) fuel containing separated plutonium. These countries base their approach on the economic value of the fissile material contained in spent fuel and by arguing that permanent disposal of spent fuel leaves a (distant) possibility that fissile material could be eventually recovered from a repository and diverted to weapons uses.

The rationale for not reprocessing the RTR spent fuel is discussed below. The fissile content of the RTR spent fuel stockpile, and its implications for a possible "closed" fuel cycle, i.e. reprocessing, are compared with those of a standard PWR. Tables 3 and 4 present a comparison of the RTR and PWR spent fuel fissile compositions for plutonium and uranium respectively.

The economic value of the fissile material contained in the spent fuel annual discharge may be roughly estimated by considering the total amount of the main elements (uranium or plutonium) and their fissile fraction. The potentially reusable material contained in the standard PWR annual discharge is about 24 Mt of uranium with 1.0 percent of U-235 and approximately 250 kg of plutonium with 73 percent fissile content (Pu-239 + Pu-241). The corresponding values for the RTR–seed annual discharge are 3 Mt of uranium with 4.7 percent of U-235 and 37 kg of plutonium with 62 percent fissile content (Pu-239 + Pu-241).

The RTR–blanket is discharged only once in 10 years and contains about 120 kg of "low-quality" plutonium with 53 percent fissile content and 3.9 Mt of uranium. The RTR–Blanket uranium contains 16 percent of U-233 and 1.4

Table 3: Discharged fuel fissile content of plutonium (fraction of total plutonium).

Nuclide	PWR[a]	RTR-seed[b]	RTR-blanket[c]	Weapon grade[d]	Super grade (Trinity)[e]
Pu-238	0.010	0.065	0.120	0.00012	–
Pu-239	0.590	0.465	0.382	0.938	0.98
Pu-240	0.210	0.225	0.150	0.058	0.02
Pu-241	0.140	0.155	0.147	0.0035	–
Pu-242	0.050	0.090	0.201	0.00022	–
total Pu/year (kg)	~250	36.6	11.8[f]	–	–
total H.M./year (kg)	26,000	3,206	4,450[g]	–	–

a. Plutonium from PWR fuel depleted to 36 GWd/T.
b. Plutonium from RTR fuel, VVERT reference design data.
c. Plutonium from RTR fuel, VVERT reference design data.
d. Weapon grade presented for comparison. Data from endnote 9.
e. Super grade plutonium recovered from very low burnup fuel. Data from endnote 9, presumably close to plutonium composition of the Trinity device.
f. Blanket fuel reloaded once in 10 years in VVERT cycle.
g. Blanket fuel reloaded once in 10 years in VVERT cycle.

percent of U-235. It also contains large amounts of "poison" uranium isotopes, such as U-234, U-236 and mainly U-238. Presence of a sizable amount of U-232 with high-energy gamma emission would necessitate remote separation and refabrication processing, thereby increasing the overall cost of a reprocessed fuel and reducing the economic incentive for closing the RTR-blanket fuel cycle. The same argument applies to the possibility of reenriching the mixture of uranium isotopes separated from the spent blanket fuel. From an economic point of view, this process would be inferior to enriching natural uranium, which does not include highly radioactive nuclides, and, therefore would not require expensive remote handling hardware.

The amount of plutonium contained in the RTR spent fuel stockpile is much smaller and of a lesser "neutronic" quality as shown above. Thus, the economic incentive for recycling this plutonium is also reduced in comparison with a standard PWR fuel cycle. The infinite multiplication factor of the discharged seed is similar to that of a twice-burned PWR fuel. Therefore, the discharged seed fuel may be reused "as is," without reprocessing and refabrication for an additional burnup period. However, separation of the uranium component only from the relatively small total amount of the RTR-seed seems economically unjustifiable.

Table 4: Discharged fuel fissile content of uranium (fraction of total uranium).

Nuclide	PWR[a]	RTR-seed[b]	RTR-blanket[c]
U-232	0	0	0.002
U-233	0	0	0.160
U-234	0	0	0.052
U-235	0.009	0.047	0.014
U-236	0.004	0.033	0.022
U-238	0.987	0.920	0.750
total U/year (kg)	**24,000**	**3,000**	**3,900**

a. Plutonium from PWR fuel depleted to 36 GWd/T.
b. Plutonium from RTR fuel, VVERT reference design data.
c. Blanket fuel reloaded once in 10 years in VVERT cycle.

In summary, the amount and the isotopic composition of the discharged RTR fuel stockpile makes the reprocessing option of the fuel cycle even less attractive from the economic point of view than for a corresponding PWR fuel cycle.

PROLIFERATION RESISTANCE OF THE RTR FUEL CYCLE

The proliferation potential of a fuel cycle, or its proliferation resistance, is determined by the quantity and the quality of the fissile material that could be diverted to military use. An additional factor is the measure of complexity required to separate the fissile component from the normal material flow of the fuel cycle.

The assessment of proliferation potential depends to a great extent on the specific proliferation scenario. The scenarios related to a civilian nuclear power fuel cycle are national diversion scenarios, either clandestine or open.[7] In both cases, international treaties and safeguards may not be effective because they can be abrogated or circumvented. In order to separate the development and expansion of nuclear power from the danger of nuclear weapon proliferation, international safeguards are necessary but not sufficient.

A decisive barrier to proliferation should be based on inherent properties of the fuel cycle itself. The fuel design should provide assurance that the quantity and quality of the fissile component of the fuel cycle material flow reduces the proliferation potential below an acceptable threshold in the context of industrial capabilities and economic realities.

In this section, the proliferation resistance of the RTR fuel cycle is discussed. The quantity and the quality of the fissile material contained in the RTR spent fuel stockpile is compared with that of a standard PWR cycle. The proliferation potential of a material is assessed qualitatively following a simple model given in endnote 9.

The fissile material weapon quality is evaluated by considering three properties:

(i) critical mass. A critical mass is different for different isotopic compositions of plutonium and uranium;

(ii) weapon yield degradation due to preinitiation caused by spontaneous fission neutrons; and

(iii) weapon stability degradation by heat emission.

A comparison of the critical mass for different materials is presented in table 5. The values, obtained by SCALE calculations of metal spheres with water reflectors, have nothing to do with the actual weapon design and are used here only for the comparison of the RTR and PWR grade plutonium mixtures with weapon grade plutonium.[8]

It is clearly demonstrated, that a relatively small critical mass is achieved with any plutonium composition, and that RTR–Pu requires 20 to 50 percent more material compared with the weapon-grade material.

SPONTANEOUS FISSION AND YIELD DEGRADATION

The spontaneous fission source (SFS) defines an important characteristic of the weapon material, namely the yield degradation. Neutrons released by spontaneous fissions cause preinitiation, i.e. the start of the explosion before the device has reached its highest supercriticality value, which in turn causes a reduction of the device yield. A simple qualitative model described in endnote 5 is used to estimate the yield degradation of the explosion device based on RTR–Pu and to compare it with PWR grade plutonium and weapon-grade plutonium.

The smallest value of the explosive yield results from preinitiation, which occurs at the same moment that a device becomes critical. This minimum yield, called the "fizzle" yield, is estimated to be 0.027 of the nominal yield.

Information published in the open literature with regard to the expected performance of the Trinity explosive device allows an estimate of the probabilities of the nominal yield and the fizzle yield.[9] Making a reasonable assumption concerning the plutonium composition used in the Trinity device and

Table 5: Critical mass for different plutonium compositions.

Pu source	Critical mass (kg)
weapon grade	4.3
PWR grade	5.5
RTR–seed	5.9
RTR–blanket	6.5

comparing the spontaneous fission neutron source of alternative plutonium compositions we have estimated the probability of a given explosive yield as: $P = (P_o)^n$ where, P_o is the probability that a Trinity-type device will deliver a given yield (see table 3 or endnote 9), and n is the ratio of Trinity SFS to that of an alternative device. The relevant data for all considered plutonium compositions are summarized in table 6. The first column shows a spontaneous fission source for one gram of a given isotope. The rest of the table shows the SFS in neutrons/sec for a critical mass of all considered plutonium compositions. The last row gives the value of n, i.e., the ratio of a SFS for a given plutonium composition to that of a weapon grade plutonium, assumed for a Trinity device.

The total spontaneous fission source for a critical mass of PWR grade plutonium is 7 times larger than that of a weapon grade plutonium, and for the RTR-seed and RTR-blanket plutonium is 13 and 22 times larger respectively. The nominal and fizzle yields are estimated for each of the considered plutonium compositions and are presented in table 7.

The probability that an explosive device, constructed from RTR–Pu, will deliver a nominal yield is small (seed) to negligible (blanket), and a probability of a fizzle yield is relatively high. Thus, it is shown that the RTR–Pu will produce an unreliable weapon. This result, however, pertains to devices with speed of assembly similar to that of Trinity, which might be the case for crude explosives assembled by a terrorist group or relatively unsophisticated country. A more sophisticated country might be able to design a weapon whose yield would be much less degraded by a spontaneous fission source.[10]

HEAT GENERATION

An additional barrier for a possible diversion of a reactor grade material is the heat emitted by its isotopes. The thermal power produces an increase of the temperature of a device and causes two effects: one is a temperature increase

Table 6: Spontaneous fission rate for plutonium isotopes.

Nuclide	Spontaneous fission rate $(gm\text{–}sec)^{-1}$	Spontaneous fission source $(kg\text{–}sec)^{-1}$				
		Super grade (Trinity)	Weapon grade	PWR grade	RTR–seed grade	RTR–blanket grade
Pu-238	2,600.0	0	312	26×10^3	169×10^3	312×10^3
Pu-239	0.022	0.022	21	13	10	8
Pu-240	910.0	18,200	52,780	191.1×10^3	204,750	136×10^3
Pu-241	0.049	0	0.2	7	8	7
Pu-242	1,700.0	0	374	85×10^3	153×10^3	342×10^3
total/kg of Pu	–	18,200	53,487	302×10^3	526×10^3	790×10^3
total/ critical mass	–	78.3×10^3	230×10^3	$1,661 \times 10^3$	$3,103 \times 10^3$	$5,135 \times 10^3$
ratio to super grade, n	–	1	3	21	40	66

of metallic plutonium which undergoes a metallurgical phase transition at 115°C, and second is an overheating of a high explosive around the plutonium core which may cause the disintegration of this high explosive. The specific heat produced by different plutonium isotopes is summarized in table 8 and is used to estimate the total heat produced by a plutonium metal critical mass in each case.

The total heat produced by the RTR-seed and RTR-blanket plutonium is much higher than that produced by the PWR grade plutonium. Heat loads at the level of seed and blanket plutonium are likely to require special heat removal measures to be incorporated in the design of a weapon. The nature and effectiveness of such measures are beyond the scope of this paper, but it is reasonable to assume that the device stability will be impaired.

Table 7: Probability of an indicated yields.

Yield	Super grade (Trinity)	Weapon grade Pu	PWR grade Pu	RTR-seed grade Pu	RTR-blanket grade Pu
nominal	0.88	0.68	0.07	0.006	0.0002
fizzle	0.02	0.06	0.35	0.55	0.74

U-233 DIVERSION POTENTIAL

The RTR cycle is based on extensive utilization of thorium, which produces through a nuclear reaction the fissile isotope U-233. U-233 has been determined to be superior to U-235 and at least as efficient as Pu-239 as a weapon material. Therefore, a special effort was invested in the RTR design to create effective barriers to diversion of U-233. The total amount of U-233 in the spent blanket fuel, discharged once in 10 years, is about 630 kg (including Pa-233). The annual equivalent of 63 kg may represent major proliferation potential. In order to eliminate this potential, the U-233 created in the blanket was denatured by addition of slightly enriched uranium. The amount of uranium added for a dilution of fissile components was carefully chosen to reduce the overall content of fissile uranium isotopes in a discharged blanket fuel well below 17 percent, which is roughly equivalent to 20 percent of U-235 enrichment. Additional uranium isotopes created during the long in-core residence time of the blanket are U-232, U-234, U-235, and U-236.

In principle, all uranium isotopes may be chemically separated from blanket spent fuel and further enriched by standard industrial methods. However, there are several barriers to the diversion of U-233 through this route:

- The contamination of the recycle material by a hard-gamma emitter (T1-208) originating in the U-232 chain will require that the reprocessing facility be remotely operated.
- The enrichment of the mixture of separated uranium isotopes will be extremely inefficient due to its isotopic composition. An attempt to separate U-233 from the U-238, U-234 and U-236 isotopes will also remove the fissile U-235 from the resulting enriched stream. Residual amounts of U-234 and U-236 will reduce the criticality of the enriched stream.
- Enrichment in U-233 inherently yields a product with increased U-232 content, exacerbating the Tl-208 gamma problem.

Table 8: Decay heat emission for different plutonium compositions.

Nuclide	Specific decay heat (watts/kg)	Weapon grade (watts/kg Pu)	PWR grade (watts/kg Pu)	RTR-seed grade (watts/kg Pu)	RTR-blanket grade (watts/kg Pu)
Pu-238	560	0	12.88	38.64	70.56
Pu-239	1.9	1.77	1.08	0.85	0.65
Pu-240	6.8	0.42	1.54	1.58	0.89
Pu-241	4.2	0.03	0.54	0.60	0.54
Pu-242	0.1	0	0	0.01	0.03
total (watts/kg)	–	2.22	16.04	41.68	72.66
total (watts/ critical mass)	–	10	88	244	475

It should be stressed that all the technical and organizational difficulties on the diversion path of separating and enriching the uranium isotope mixture created in the RTR-blanket may be overcome. A country or a clandestine organization with enrichment technology and installations available may theoretically obtain weapon grade material from the discharged blanket fuel. However, this diversion route is much more difficult and expensive than simple enrichment of readily available natural uranium. Therefore, in cases where enrichment facilities are available, U-233 created in the RTR-blanket does not contribute to the proliferation potential of the fuel cycle.

SPENT FUEL STORAGE AND DISPOSAL

The RTR spent fuel can be readily stored and disposed using technology applicable to conventional LWR fuel. This includes considerations of radiological, thermal and toxicity issues. The dominant consideration for the analysis is simply the amount of material produced and the efforts required to store and dispose of that material. It is assumed that neither PWR nor RTR spent fuel will be reprocessed and will be transported, stored, and disposed as it is loaded into a reactor core.

Table 9: Annual fuel discharge.

Isotope	PWR	RTR		
		Seed	Blanket	Total
total H.M. weight (MT)	26.13	3.50	3.61	7.11
total fuel volume (m^3)	9.33	3.82	1.65	5.47

Three different time periods should be considered. During the first period the spent fuel is stored in water pools at a reactor site. Spent fuel storage racks are placed under water in the fuel storage building adjacent to the reactor building. These racks hold the assemblies and maintain the required spacing between assemblies to provide criticality control and residual heat removal. The space available at a reactor site is limited and additional fuel storage away from the reactor (AFR) is eventually required.

The facilities for the second time period (AFR storage) may be based on wet as well as dry storage technologies. The space in spent fuel pools is virtually always insufficient to store all the spent fuel generated over the expected life of the plant. This is because fuel is typically suitable for shipping within 10 years of discharge, and it was expected that the spent fuel facilities for storing or processing would be available in due time. This is not the current situation in the nuclear power industry. As a spent fuel pool approaches capacity, some of its assemblies must be removed to allow continued plant operation. This fuel would be moved to one of three locations: (1) on-site storage in a special facility owned by the reactor owner, (2) an off-site facility developed and built with the financial contribution of the reactor owner, and (3) directly to a disposal facility, again ultimately at the reactor-owner's expense. The direct economic estimate of the RTR fuel cycle back-end may be assessed and compared with that of a PWR fuel cycle by comparing the annual discharged amounts of spent fuel. The comparison is presented in table 9.

The comparison shown in table 9 is based on a typical PWR plant data, normalized to a total power output of 3,000 MWth with 3–batch in-core fuel management, which is identical to the RTR total power and seed fuel management scheme. The seed volume fraction of the RTR assembly was assumed to be 0.41, according to the VVERT preliminary reference design. All blanket subassemblies are replaced as a single batch every 10 years and the values shown in table 9 represent equivalent annual discharge.

Table 10: Annual MPC requirements.

Cycle	Number of fuel assemblies discharged (annual average)	Number of MCPs required (annual average)
PWR	54.33	2.59
RTR-seed	54.33	0.73
RTR-blanket	16.3	0.78
RTR total	-	1.51

The large reduction in the discharged fuel weight (about 70 percent) and discharged fuel volume (equivalent to about 50 percent) represents potential economic benefits due to reduced waste volume and handling by the utility owner and reduced waste handling in the disposal system, which ultimately come back to the utility owner. In this estimate each discharged blanket subassembly was assumed to nest a discharged seed subassembly.

For the purposes of this analysis the spent fuel from either a RTR or a PWR core is assumed to be introduced into a Multi-Purpose Container (MPC) as defined by the U.S. Department of Energy. This is a container loaded with spent fuel at the reactor site, and then welded and drained. This container is assumed never to be reopened, and it is to be transported, stored and then disposed in overpacks designed specifically for these purposes.

The currently envisioned MPC design will hold 21 PWR assemblies or 44 BWR assemblies. For the RTR spent fuel, conventional MPCs will be used. Some will use PWR design and will contain 21 blanket assemblies with 21 seed assemblies nested inside. The others will contain only seed assemblies. The seed assemblies are somewhat smaller than BWR assemblies; therefore, a seed specific basket could be used, holding 52 seed assemblies. Annual average MPC requirements for PWR and RTR cycles are summarized in table 10.

The costs of spent fuel storage, transportation and disposal are reactor-owner and country-specific. For illustration, the impact of reduced spent fuel volume from an RTR is assessed on a U.S. cost basis. The overall estimate is based on data published in recent estimates of storage, transportation and disposal costs in the U.S. Civilian Radioactive Waste Management Program.[11] The cost of utility support for MPC handling and loading are not covered in the assessment. Another potential benefit would result if the reactor owner ran short of spent fuel storage space before the DOE was scheduled to take the fuel from that reactor. The economic effect of this benefit is difficult to quantify and therefore, no credit is accounted for.

The total handling, transportation, and disposal cost of a single MCP is evaluated as 2.82 million dollars and the corresponding annual benefit of 1.08 MPC saved by the RTR cycle per reactor year is 3.05 million dollars.[12]

A detailed analysis of the spent fuel radioactivity and heat emissions was carried out to assess the long-term effects of replacing standard LWR fuel by RTR fuel. It was found that both radioactivity and heat emission levels of the RTR fuel would be lower than those of the LWR fuel.[13] This effect is due primarily to a much lower content (by about an order of magnitude) of the transuranium isotopes. The short-lived alpha-emitting nuclides introduced by the RTR fuel cycle originated from the long-lived U-233, would not cause a reduction in potential savings of RTR fuel disposal. This is because the nuclides would increase the heat and radioactivity emissions of the RTR fuel relative to PWR fuel for the period starting at 20,000 years after discharge and peaking about 1,000,000 years after discharge. For this period the emission levels would already be reduced by 4 to 5 orders of magnitude, and would therefore be too small to affect the design of the disposal facilities.

SUMMARY

One of the major problems of the nuclear power industry is its nuclear weapon proliferation potential. In particular, there is the risk that reactor grade plutonium could be diverted and used for military purposes. This causes a justified public concern and may be considered as one of the major obstacles to a worldwide expansion of nuclear power.

The proliferation potential of the civilian fuel cycle can be significantly reduced by utilization of thorium as a fertile component of the nuclear fuel. The main challenge of Th utilization is to design a core and a fuel cycle which would be non-proliferative and economically feasible. This challenge is met by the RTR concept presented in this paper. The characteristics of the preliminary reference design of a VVERT option are presented in this paper.

The RTR fuel cycle produces only about 20 percent of plutonium compared with a standard PWR cycle, and economic incentives for reprocessing spent fuel are reduced. Also the isotopic composition of plutonium contained in RTR spent fuel would make the use of the plutonium in weapons much more difficult and complicated. Certainly any crude weapons fabricated from the RTR-originated plutonium would be highly unreliable.

The RTR-generated spent fuel stockpile will still require a safeguarded disposal facility, but taking into account the reduced quantity and quality of the fissile material in the RTR fuel cycle, a nuclear power system based on the RTR would be highly proliferation resistant.

NOTES AND REFERENCES

1. "Large Power Reactor Program," WAPD-lpr-141, (July 1963), Section 5.B.1, "Comparison of Irradiation Effects of ThO_2 and UO_2," p. A–31.

2. "The Nonproliferative Light Water Thorium Nuclear Reactor—VVERT Preliminary Reference Design Report," RTPC/RL-02–96 Report, (March 1996).

3. *Ibid.*

4. A. Radkowsky, M. Segev and A. Galperin, "The Power Sharing Formula for a Seed-Blanket Core—Resolution of a Paradox," *Nuclear Science and Engineering*, (1986), pp. 94, 80–86.

5. "The Nonproliferative Light Water Thorium Nuclear Reactor—VVERT Preliminary Reference Design Report," *op. cit.*

6. R. E. Allen, "RTR Core Fuel Cycle Analysis," Raytheon Nuclear, Inc., Report RNI/RTPC 960223-A, (February 1996).

7. F.J. Rahn, A.G. Adamantiades, J.E. Kenton and C. Braun, "A Guide to Nuclear Technology—A Resource for Decision Making," (J. Wiley and Sons 1984).

8. *SCALE—Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, RSIC Computer Code Collection, CCC-545, (1993 revised).

9. J. C. Mark, "Explosive Properties of Reactor-Grade Plutonium," *Science & Global Security*, Vol. 4, (1993), p. 111.

10. *Management and Disposition of Excess Weapons Plutonium*, National Academy of Sciences, (Washington, D.C. 1994), p. 33.

11. "Analysis of the Total System Life Cycle Cost of the Civilian Radioactive Waste Management Program," U.S. Department of Energy Office of Civilian Radioactive Waste Management, (September 1995).

12. *Ibid.*