

**Discussion Paper**

**ARE CHEMICALLY SEPARABLE, WEAPONS-USABLE FISSILE MATERIALS A  
CHARACTERISTIC OF NUCLEAR POWER SYSTEMS?**

—An Example: A Proliferation-Resistant Molten Salt Reactor—

Charles W. Forsberg  
Oak Ridge National Laboratory\*  
P.O. Box 2008  
Oak Ridge, Tennessee 37831-6180  
Tel: (865) 574-6783  
Fax: (865) 574-9512  
Email: forsbergcw@ornl.gov

Preparation Date: May 5, 2000

For  
U.S. Department of Energy-Russian Federation Workshop  
Proliferation Resistance of Nuclear Technologies  
Moscow, Russia  
June 27-29, 2000

The submitted manuscript has been authored by a contractor of the U.S. Government under contract DE-AC05-00OR22725. Accordingly, the U.S. Government retains a nonexclusive, royalty-free license to publish or reproduce the published form of this contribution, or allow others to do so, for U.S. Government purposes.

---

\*Oak Ridge National Laboratory, managed by UT-Battelle, LLC, for the U.S. Dept. of Energy under contract DE-AC05-00OR22725.

## ABSTRACT

A major concern with nuclear power is the existence in reactors and spent nuclear fuel of chemically separable, weapons-usable materials, which could be diverted to construct nuclear weapons. Over the lifetime of a large light-water reactor, sufficient plutonium is produced such as to build several hundred weapons. The historical “belief” has been that any large nuclear power reactor will contain large quantities of chemically separable, weapons-usable materials (i.e., production of weapons-usable material is an intrinsic characteristic of the production of nuclear power). However, that assumption may not be true. It may be possible to build a proliferation-resistant nuclear power system with no significant quantities of chemically separable, weapons-usable fissile materials and thus break the potential connection between nuclear power and nuclear proliferation.

An evaluation indicates that the smallest inventories of weapons-usable fissile materials would be in reactors with the following characteristics: (1)  $^{233}\text{U}$ –thorium fuel cycle with sufficient  $^{238}\text{U}$  to make the  $^{233}\text{U}$  non-weapons-usable, (2) low reactor fissile inventory, (3) thermal neutron spectrum to suppress plutonium inventories relative to  $^{233}\text{U}$  inventories, and (4) long fuel irradiation times to maximize the ratio of higher plutonium isotopes to  $^{239}\text{Pu}$ . One candidate reactor is a modified molten salt reactor.

## 1. INTRODUCTION

The historical “belief” has been that any large nuclear power reactor will contain large quantities of chemically separable, weapons-usable materials (i.e., that production of weapons-usable material is an intrinsic characteristic of nuclear power). When nuclear power was first developed, the emphasis was on breeding fissile materials because it was thought that uranium resources were limited. Proliferation resistance was not a consideration. Consequently, no effort was made to minimize the inventory of chemically separable, weapons-usable materials in the reactor or spent nuclear fuel (SNF).

Could a nuclear power system be designed with no significant inventories of chemically separable, weapons-usable materials? Could such a system effectively break the potential link between nuclear power and nuclear weapons proliferation? Would such a system be practical? To examine these issues, a set of criteria is first defined that can be used to judge proliferation resistance. Second, using the criteria and nuclear reactor physics constraints, a logic is developed that defines some of the technical characteristics of such a reactor. Based on these considerations, a selection was made of a promising reactor concept that may meet the criteria. A description of the candidate system is provided in order to explore and understand the issues with such concepts.

## 2. DEFINITION OF PROLIFERATION RESISTANCE

One can not identify a proliferation-resistant (PR) nuclear power system unless one defines the term *proliferation-resistance*. The goal is a system where both terrorists and nation states conclude that other approaches to obtain weapons-usable materials are clearly preferable to obtaining these materials from the civilian nuclear power industry or using facilities associated with the industry. Three criteria are used to define the idealized characteristics of a reactor and associated fuel cycle. The criteria are similar to those defined by Williams and Feiveson (July–August 1990).

**2.1 CRITERION I:** *The waste shall not contain significant quantities of chemically separable, weapons-usable fissile materials.*

The primary waste from existing power reactors is SNF, which contains large quantities of chemically separable, weapons-usable plutonium. SNF, like many other wastes, may be processed to recover potentially valuable materials (plutonium). Whether recycle is viable depends upon economics and other considerations. It has been assumed that chemically separating plutonium from SNF is technically difficult and, therefore, that once-through fuel cycles that dispose of SNF as a waste are PR. The primary technical difficulty in recovering plutonium from SNF is the high radiation levels associated with the SNF soon after its discharge from the reactor. Unfortunately, for a number of reasons, the barrier is not as robust as is generally assumed.

- *Old SNF.* The radiation levels of SNF decrease with time and thus reducing the difficulty in recovery of plutonium. Many nonproliferation studies assumed that this problem would be overcome if the United States accepted foreign SNF for disposal before the radiation levels decayed to low levels and thereby removed the SNF from countries where plutonium recovery might be attempted. However, domestic political considerations have made it impossible for the United States or any other weapons state to accept foreign power-reactor SNF. One must assume that SNF will remain where it is generated.
- *New Technologies.* Recovery of plutonium from SNF requires remotely operated equipment to conduct the chemical separation processes within highly radioactive environments. Unfortunately, in terms of weapons proliferation, the required technology to do this job is improving rapidly and becoming more widely available—partly because of a revolution in offshore oil-field technology and partly because of other technical advances. The offshore oil industry is beginning to use seabed production platforms with equipment on the seabed rather than traditional oil platforms with the equipment above the surface of the ocean. Seabed production platforms require remote operation, construction, and maintenance techniques. That industrial demand is rapidly developing and spreading advanced robotic technologies.

The most viable long-term proliferation-resistant solution is to avoid waste streams that contain significant quantities of chemically separable, weapons-usable fissile materials.

**2.2 CRITERION II:** *The reactor shall not contain significant quantities of chemically separable, weapons-usable fissile materials.*

The goal for a 1,000-MW(e) plant is to reduce the quantity of chemically separable, weapons-usable material to less than that defined by the International Atomic Energy Agency (IAEA) as sufficient such as to construct one nuclear weapon (equivalent to 8 kg <sup>239</sup>Pu, 8 kg <sup>233</sup>U, or 25 kg <sup>235</sup>U) (Williams and Feiveson July–August 1990). If the reactor contains large quantities of chemically separable, weapons-usable materials, it may become a potential source of weapons-usable materials by a nation-state.

**2.3 CRITERION III:** *The reactor and fuel cycle shall strongly inhibit conversion of the facilities and organizations for the production of weapons-usable fissile materials.*

This criterion assures that if a nation-state chooses to develop nuclear weapons, it will choose to build special-purpose production facilities rather than to convert or otherwise use the power reactor or fuel cycle facilities as sources of weapons-usable fissile materials.

A corollary is the need to avoid uranium enrichment facilities for long-term production of fuel. Enrichment facilities can be used to produce (1) fuel or (2) weapons-usable, high-enriched uranium (HEU). Many nations have the legitimate desire to use nuclear power to assure energy independence. For such nations, it would be acceptable to import low-enriched uranium (LEU) one time to start a reactor. LEU contains <20%  $^{235}\text{U}$  in  $^{238}\text{U}$  and is defined as non-weapons-usable uranium. Enriched uranium could be supplied from existing facilities. This would be no different than importing a pressure vessel or other special reactor component. However, many nations would be concerned if continued importation of enriched uranium was required to maintain reactor operations. The only way to eliminate the legitimate energy-independence concern and the corresponding incentive to build uranium enrichment plants is to eliminate the need for importation of enriched uranium for fuel after initial reactor startup.

### 3. SELECTION OF A PR REACTOR AND FUEL CYCLE

Using these idealized criteria, reactor constraints, and engineering constraints, an analysis (Fig. 1) was conducted to determine what kinds of reactors are most likely to meet all criteria. The logic also defines an approach to minimize the inventory of chemically separable, weapons-usable fissile materials in other reactors.

#### 3.1 BASIS FOR THE SELECTION OF A $^{233}\text{U}$ -THORIUM REACTOR FUEL CYCLE

There are three practical nuclear fuels:  $^{239}\text{Pu}$ ,  $^{235}\text{U}$ , and  $^{233}\text{U}$ . Plutonium-239 is made by the neutron irradiation of  $^{238}\text{U}$ . Uranium-235 is found in nature—isotopically mixed with the fertile material  $^{238}\text{U}$ . Uranium-233 is made by the neutron irradiation of  $^{232}\text{Th}$ . The thorium is found in ore bodies. An examination of each fuel, as described below, leads to the conclusion that only a  $^{233}\text{U}$ – $^{232}\text{Th}$  fuel cycle could possibly meet stated goals.

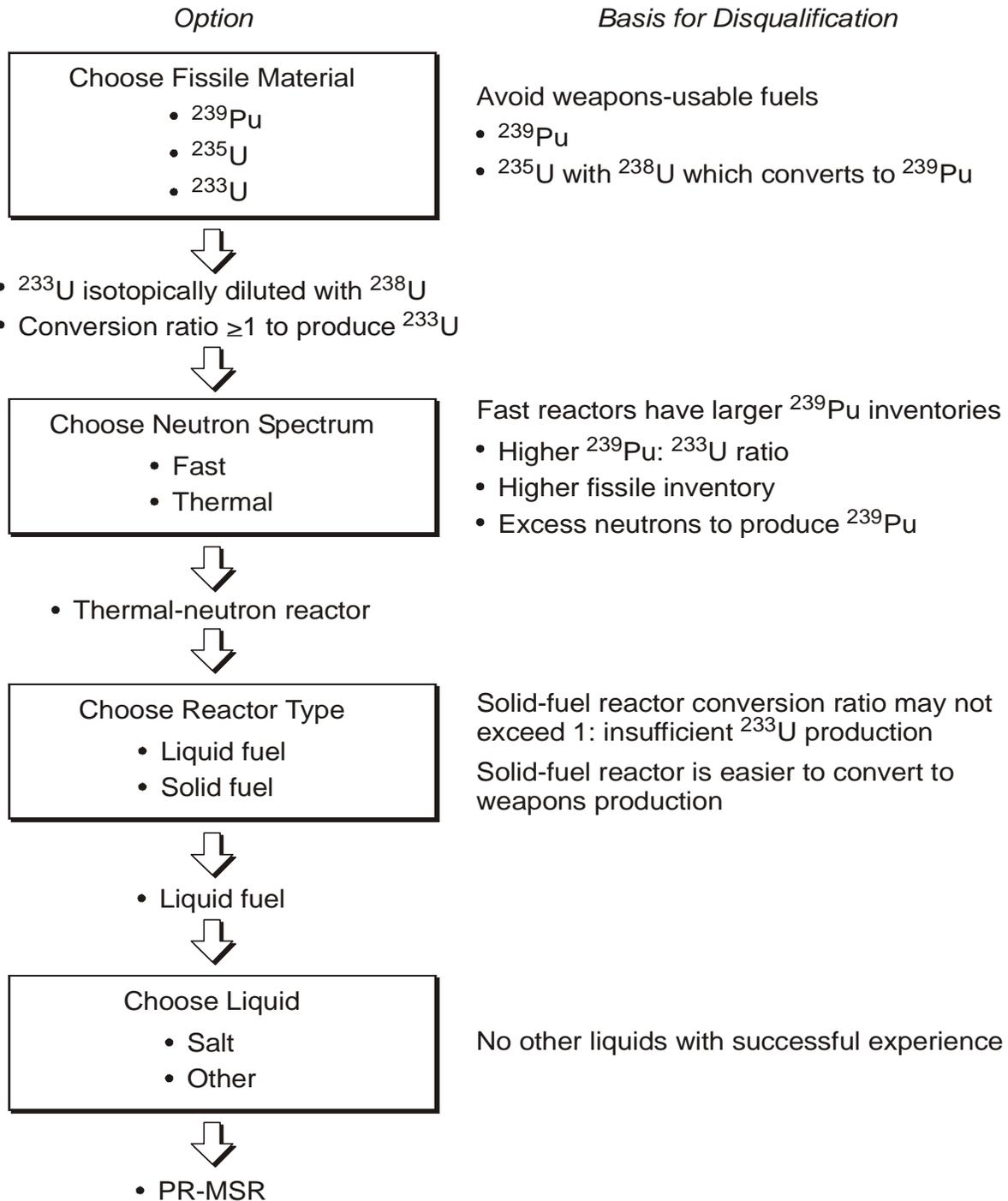
##### 3.1.1 Plutonium

The key component of most nuclear weapons is  $^{239}\text{Pu}$ . If a reactor and fuel cycle are to be PR, plutonium inventories must be minimized (*Criteria I and II*). Since  $^{239}\text{Pu}$  is produced by the neutron irradiation of  $^{238}\text{U}$  in a reactor, the use of  $^{238}\text{U}$  in a reactor should be minimized to minimize plutonium production.

##### 3.1.2 Uranium-235

Uranium-235 can not be manufactured, and there is no significant quantity of natural fertile material that can be irradiated to produce it since  $^{234}\text{U}$  exists in nature only in trace quantities. There is no  $^{235}\text{U}$  breeder reactor fuel cycle. Because  $^{235}\text{U}$  is mined, not made, the only way to fuel a reactor with non-weapons-usable  $^{235}\text{U}$  is to add LEU at intervals to the reactor and remove the SNF with (a) its lower  $^{235}\text{U}$  content, (b)  $^{239}\text{Pu}$  made from the  $^{238}\text{U}$ , and (c) the remaining  $^{238}\text{U}$ . Non-weapons-usable  $^{235}\text{U}$  fuel cycles generate significant quantities of plutonium in the SNF because the continued addition of  $^{235}\text{U}$  fuel includes the implicit cycling of  $^{238}\text{U}$  through the reactor with resultant plutonium production and discharge as SNF. A  $^{235}\text{U}$  fuel cycle does not meet *Criterion I*.

Power reactor fuel cycles that use enriched uranium also require uranium enrichment facilities, in which  $^{235}\text{U}$  is separated from  $^{238}\text{U}$ , and these facilities can be converted to produce weapons-usable HEU. The existence of such facilities creates potential proliferation risks (*Criterion III*).



**Logic for Selection of Proliferation-Resistant Molten-Salt Reactor**

### 3.1.3 Uranium-233

Uranium-233 can be used in nuclear weapons in pure form, but it can be converted to non-weapons-usable  $^{233}\text{U}$  by two different approaches:

- *Isotopic dilution with  $^{238}\text{U}$ .* Uranium-233 can be isotopically diluted to <12 wt %  $^{233}\text{U}$  in  $^{238}\text{U}$  to convert the  $^{233}\text{U}$  to non-weapons-usable uranium (Forsberg March 1998). This is equivalent to <20 wt%  $^{235}\text{U}$  in  $^{238}\text{U}$ .
- *High-radiation levels with  $^{232}\text{U}$ .* The production of  $^{233}\text{U}$  results in secondary production of  $^{232}\text{U}$ , which decays through several decay products to thallium-208 ( $^{208}\text{Tl}$ ), which then emits a 2.6-MeV gamma ray. Consequently, the radiation field associated with chemically purified  $^{233}\text{U}$  increases with time, as shown in Fig. 2. The radiation dose is calculated for an 8 kg  $^{233}\text{U}$  metal sphere—the International Atomic Energy Agency (IAEA) defined quantity of  $^{233}\text{U}$  that is needed to produce a weapon. The  $^{232}\text{U}$  level is that found in typical power reactor fueled with  $^{233}\text{U}$ . As a point of comparison, the IAEA defines *SNF* as fuel with a radiation level that exceeds 100 R/h at 1 m.

Isotopic dilution (as a proliferation-resistance barrier) is preferred because isotopic dilution exists forever as a barrier for use of the  $^{233}\text{U}$  in weapons. Isotopic dilution does not decay with time. Because there is no natural source of  $^{233}\text{U}$ , a viable  $^{233}\text{U}$  reactor and fuel cycle must produce  $^{233}\text{U}$  as fast as it consumes  $^{233}\text{U}$ ; that is, the conversion ratio (CR) must equal or exceed 1. The CR is the rate of production of  $^{233}\text{U}$  divided by the rate of consumption of  $^{233}\text{U}$ . A CR >1 implies that the reactor produces one or more kilograms of new  $^{233}\text{U}$  from  $^{232}\text{Th}$  for every kilogram of  $^{233}\text{U}$  consumed. If the CR is less than 1, the reactor can not operate only on  $^{233}\text{U}$ ; consequently, an alternative source of fissile material would have to be supplied.

### 3.2 BASIS FOR SELECTION OF A THERMAL-NEUTRON REACTOR

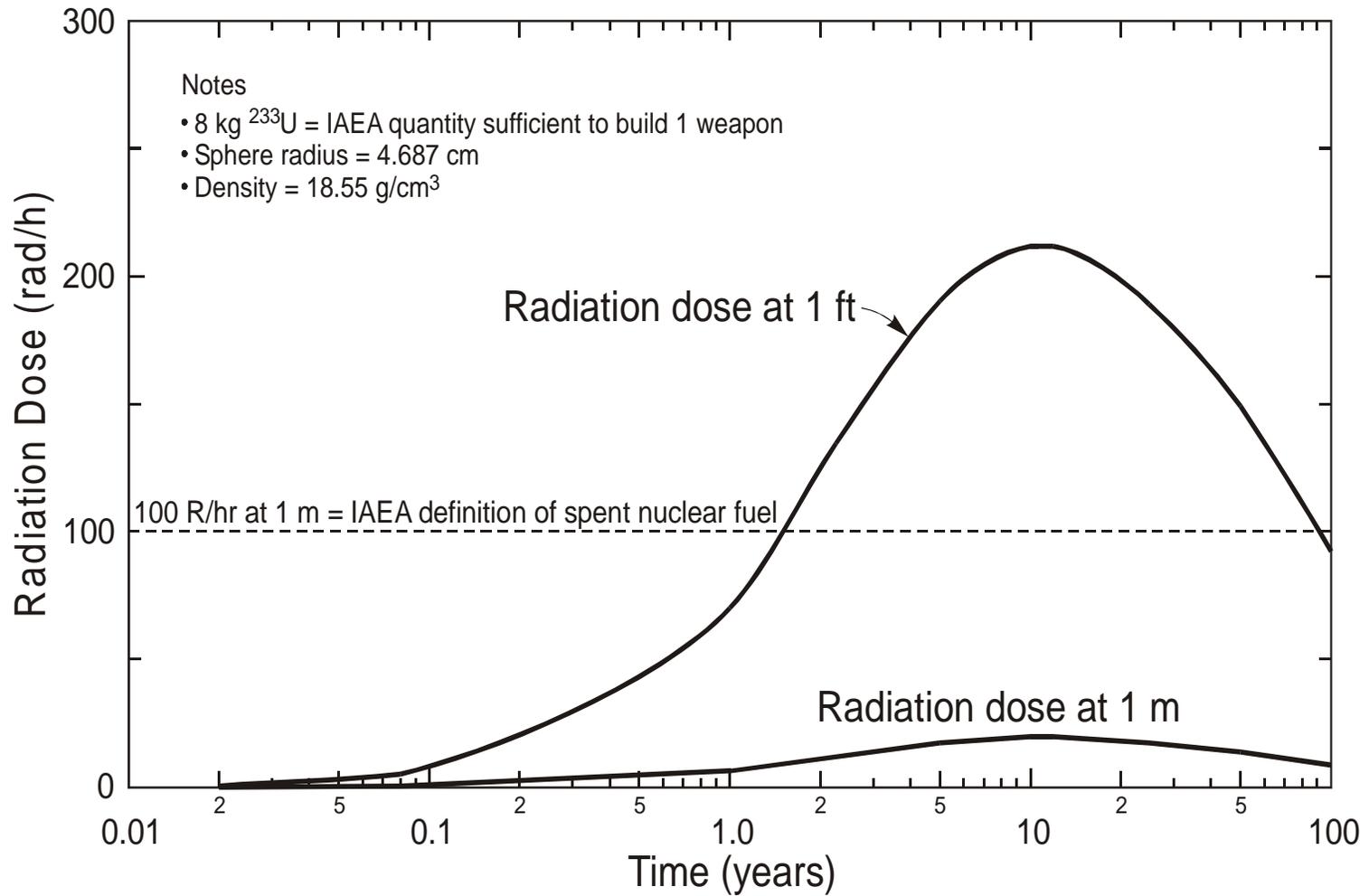
Nuclear reactors can be built with fast, intermediate, and thermal-neutron spectrums. In a fast-neutron spectrum, the neutrons have high energies and travel at high speeds. In a thermal-neutron spectrum, the neutrons have low energies and travel at low speeds. A moderator, such as carbon, is used to slow fast neutrons down to create thermal neutrons. Atoms can capture neutrons or be fissioned by neutrons. The nuclear cross section is a measure of the probability of an atom capturing a neutron or being fissioned by a neutron. It is measured in units called barns ( $10^{-24}\text{ cm}^2$ ). Neutron capture or fission strongly depends upon the neutron energy as shown in Table 1. Based on several considerations that derive from the data in this table, a thermal-neutron reactor with the CR >1 is the only option that may meet the criteria.

**Table 1. Nuclear cross sections (F) for fissile and fertile materials (in barns)**

Isotope	Thermal-neutron Cross Sections <sup>1</sup>			Fast-neutron Cross Sections <sup>2</sup>				
	Capture	Fission	Total	Capture	Fission	Total		
$^{232}\text{Th}$		7.40	0.00	7.40		0.15	0.02	0.17
$^{233}\text{U}$		47.70	531.10	578.80		0.14	2.05	2.19
$^{238}\text{U}$		2.70	0.00	2.70		0.12	0.08	0.20
$^{239}\text{Pu}$		268.80	742.50	1011.30		0.13	1.64	1.77

<sup>1</sup>Cross section measured at 2,200 m/s.

<sup>2</sup>Cross section in a fast-neutron spectrum of 50 keV and up



## Radiation Dose from a Metal Sphere with 1000 ppm $^{232}\text{U}$ and 8 kg $^{233}\text{U}$

### 3.2.1 Thermal-Neutron Reactor: Minimize Total Fissile Material Inventory

The goal is to minimize plutonium in the reactor. Since plutonium is generated from the  $^{238}\text{U}$ , the quantity of  $^{238}\text{U}$  should be minimized. The quantity of  $^{238}\text{U}$  in the reactor is determined by how much is needed to make the  $^{233}\text{U}$  non-weapons usable. To minimize the required  $^{238}\text{U}$ , the  $^{233}\text{U}$  inventory should be minimized. Fast reactors typically require ten times the fissile inventory of that of thermal reactors because of the low fast-neutron fission cross sections compared to thermal-neutron fission cross sections (Table 1). If the chance of a neutron causing fission is low, more fissile material is needed to increase the fission rate. For example, a 1,000-MW(e) light-water reactor (LWR) with a thermal-neutron spectra has about 3 tons of  $^{235}\text{U}$  and  $^{239}\text{Pu}$  in its reactor core. An equivalent fast reactor has 25 to 35 tons of  $^{235}\text{U}$  and  $^{239}\text{Pu}$ . A thermal-neutron breeder reactor is preferred because it requires a smaller inventory of  $^{233}\text{U}$  to operate and hence a smaller inventory of  $^{238}\text{U}$  to isotopically dilute the  $^{233}\text{U}$  to non-weapons-usable  $^{233}\text{U}$ .

### 3.2.2 Thermal-Neutron Reactor: Minimize Ratio of $^{239}\text{Pu}$ to $^{233}\text{U}$

For a PR  $^{233}\text{U}$  fuel cycle, the  $^{233}\text{U}$  should be produced from  $^{232}\text{Th}$  and fissioned in the presence of  $^{238}\text{U}$ , while minimizing the production of  $^{239}\text{Pu}$ . Ideally, the  $^{238}\text{U}$  would be "invisible" to the reactor while assuring the  $^{233}\text{U}$  is non-weapons-usable.

In a fast-neutron reactor, nuclear cross sections and thus nuclear reaction rates vary only by a factor of 10. Consequently, in a fast reactor, there is no way to selectively breed  $^{233}\text{U}$  fuel from  $^{232}\text{Th}$  in the presence of  $^{238}\text{U}$  without creating large quantities of  $^{239}\text{Pu}$  from  $^{238}\text{U}$ .

In a thermal-neutron spectrum, there are orders-of-magnitude differences in nuclear cross sections between isotopes and, thus, nuclear reaction rates. In particular, the large differences in nuclear cross sections can be used to minimize plutonium production in the reactor while maximizing breeding of  $^{233}\text{U}$  in the presence of  $^{238}\text{U}$ . There are several important synergistic nuclear effects:

- *Fissile production.* The production of  $^{239}\text{Pu}$  relative to  $^{233}\text{U}$  ( $^{239}\text{Pu}/^{233}\text{U}$ ), assuming equal concentrations of  $^{238}\text{U}$  and  $^{232}\text{Th}$ , is smaller in a thermal-neutron than a fast-neutron reactor. This factor is determined by the ratio of neutron-capture cross sections of (1)  $^{238}\text{U}$ , which converts to  $^{239}\text{Pu}$  upon absorbing a neutron, to (2)  $^{232}\text{Th}$ , which converts to  $^{233}\text{U}$  upon absorbing a neutron. In a fast-neutron reactor, this ratio is 0.8 ( $F_{238\text{U}} / F_{232\text{Th}} = 0.12/0.15$ ). In a thermal-neutron reactor, this ratio is 0.36 ( $F_{238\text{U}} / F_{232\text{Th}} = 2.7/7.4$ ). Thermal-neutron fluxes suppress production of  $^{239}\text{Pu}$  relative to  $^{233}\text{U}$ .
- *Fissile destruction.* The destruction of  $^{239}\text{Pu}$  relative to  $^{233}\text{U}$ , assuming equal concentrations of  $^{239}\text{Pu}$  and  $^{233}\text{U}$ , is larger in a thermal-neutron reactor than in a fast-neutron reactor. This factor is determined by the ratio of the nuclear cross sections of  $^{239}\text{Pu}$  to  $^{233}\text{U}$ . In a fast-neutron reactor, this ratio is 0.81 ( $F_{239\text{Pu}} / F_{233\text{U}} = 1.77 / 2.19$ ). In a thermal-neutron reactor, this ratio is 1.75 ( $F_{239\text{Pu}} / F_{233\text{U}} = 1011.3/578.8$ ). The relative destruction rate of  $^{239}\text{Pu}$  to  $^{233}\text{U}$  is more than twice as high in a thermal-neutron reactor as it is in a fast-neutron reactor. The inventory of  $^{239}\text{Pu}$  is preferentially destroyed in a thermal-neutron reactor, but not in a fast-neutron flux reactor.
- *Relative quantities of materials.* In a thermal-neutron reactor, fertile materials, such as  $^{232}\text{Th}$  and  $^{238}\text{U}$ , have low neutron-capture cross sections, whereas the fissile materials, such as  $^{233}\text{U}$  and  $^{239}\text{Pu}$ , have high fission cross sections. In a  $^{233}\text{U}$ -thorium reactor with a CR >1, there are large quantities of thorium—usually 40 to 60 times as much  $^{232}\text{Th}$  as  $^{233}\text{U}$ . What this implies is that

most of the neutrons are absorbed in  $^{233}\text{U}$  and  $^{232}\text{Th}$ . The  $^{233}\text{U}$  absorbs many neutrons because of its large cross section. The  $^{232}\text{Th}$  absorbs many neutrons because there is so much of it. In contrast, the  $^{238}\text{U}$  has a low-neutron cross section, and there is not much of it in the reactor—only enough for isotopic dilution of the  $^{233}\text{U}$ . Plutonium production versus  $^{233}\text{U}$  production is suppressed by the small quantities of  $^{238}\text{U}$  compared to thorium.

### 3.3 BASIS FOR SELECTION OF A LIQUID FUEL REACTOR

The previous criteria define the general characteristics of a PR reactor— but not the specific design. There are several choices of specific reactor concepts that involve either solid fuels or liquid fuels. Either may be possible. The existing data indicate that a liquid-fueled reactor is more likely to meet the stated goals, but, *unlike the earlier basis for selection of a PR reactor, this conclusion is not as definitive.*

In a liquid-fueled reactor, the uranium and other fuel materials are dissolved in some liquid. The liquid fuel flows through a wide section of piping (the reactor vessel), where nuclear criticality occurs and heat is generated. The hot liquid (with dissolved uranium and fission products) flows through a primary heat exchanger, where the heat is transferred to another fluid. The liquid fuel then flows back to the reactor. In the heat-transfer loop, the heat is transferred by the heat-transfer fluid to a steam generator, where steam is produced and sent to a turbine generator to produce electricity.

#### 3.3.1 Minimum Fissile Inventory

In a molten fuel reactor, there is no SNF. Fission products are removed from the liquid, solidified, and sent to disposal. All fissile materials remain in the reactor until they fission. The lack of SNF minimizes the total system weapons-usable fissile inventory. In a solid fuel reactor, the inventory of chemically separable weapons-usable fissile materials in the SNF often exceeds that in the reactor.

#### 3.3.2 Viability

A  $^{233}\text{U}$ – $^{232}\text{Th}$  reactor with a CR that equals or exceeds 1 is desired to avoid the need to add  $^{235}\text{U}$  to the reactor with its excess  $^{238}\text{U}$  and resultant effect—the production of  $^{239}\text{Pu}$ . However, even the best thermal-neutron reactors, unlike fast reactors, do not have high CRs. A thermal-neutron reactor with a CR >1 can be made using a  $^{233}\text{U}$ – $^{232}\text{Th}$  fuel cycle. Thermal-neutron reactors with CRs equal to or exceeding 1 can not be made using a  $^{239}\text{Pu}$ – $^{238}\text{U}$  fuel cycle—the nuclear physics does not allow this. Studies (Engel 1978) indicate that after accounting for losses and the higher actinides, 1.06 neutrons are ultimately produced for every neutron absorbed in  $^{232}\text{Th}$ . In contrast, only 0.9 neutrons are produced for every neutron absorbed in  $^{238}\text{U}$ .

Denaturing  $^{233}\text{U}$  with  $^{238}\text{U}$  reduces the reactor CR and brings it close to 1. It is unclear whether a practical, solid-fuel, thermal-neutron reactor with a CR equal to or exceeding 1 is possible with this degradation in neutronic performance. The CR in a liquid-fuel thermal breeder reactor is better than in a solid fueled reactor for several reasons.

- *Continuous fission product removal.* The fission process creates fission products with high neutron cross-sections and resultant parasitic neutron capture, which lowers the value of the CR. In liquid-fuel reactors, some of the most important parasitic fission products (xenon isotopes) are rapidly removed (stripped out of the liquid fuel as gases). This stripping significantly improves neutron utilization and thus the CR.

- *Continuous fueling.* In most solid-fuel reactors, sufficient fuel is placed in the reactor at startup so as to allow operation for extended time periods. This implies excess fissile material over what is needed at reactor startup, the subsequent need for burnable absorbers and control rods, and the resultant parasitic loss of neutrons. A few solid-fuel reactors, such as the Canadian Deuterium Uranium Reactor, allow continuous on-line fueling (Banerjee June 1977).

### 3.4 BASIS FOR SELECTION OF A MOLTEN SALT REACTOR SYSTEM

Many types of  $^{233}\text{U}$ -thorium, thermal-neutron, liquid-fueled, reactors with a CR >1 are possible. Studies (e.g., *Nuclear Technology* February 1970, Bettis February 1970, and MacPherson 1985) were conducted in the 1950s and 1960s on alternative reactor concepts with these characteristics. The conclusion was that the best power-reactor candidate is the molten salt breeder reactor (MSBR). This is the only liquid-fuel-concept reactor which has been demonstrated on a significant scale. The Molten Salt Reactor Experiment, an 8-MW(th) reactor, demonstrated the technology using first a fuel with  $^{235}\text{U}$  fluorides dissolved in molten salts and later  $^{233}\text{U}$  fluorides dissolved in molten salts. It operated for 13,000 equivalent full-power hours. The MSBR is fueled with weapons-usable  $^{233}\text{U}$ . A PR molten salt reactor (MSR) (PR-MSR) would be a variant of this concept.

## 4. PR-MSR

A potential example of a PR reactor is provided herein to better understand some of the characteristics, limitations, and issues associated with this approach to a PR nuclear-power system. The example is the PR-MSR. Only limited studies have been done; thus, there are *significant uncertainties*. The concept of minimizing proliferation risks by minimizing the inventories of chemically separable, weapons-usable materials has not received significant study.

### 4.1 SYSTEM DESCRIPTION

The PR-MSR is a variant of the MSBR. Similar characteristics of these two reactor concepts are described; this description is followed by a discussion of the differences in their designs.

#### 4.1.1 Common Characteristics

In the late 1960s, a detailed conceptual design of a 1,000-MW(e) MSBR was developed (Robertson 1971). A schematic of the reactor is shown as Fig. 3. Some of the design characteristics of the reactor are presented in Table 2.

The reactor fuel consists of thorium fluorides and uranium fluorides dissolved in a molten salt containing lithium-7 fluoride and beryllium fluoride. During operation, various fission products and actinides also form fluorides in the liquid. Nuclear criticality occurs in the reactor vessel containing unclad graphite. The molten fuel salt flows upward through vertical channels in the graphite. The graphite slows down fast-fission neutrons and creates a thermal-neutron flux. The heat is primarily generated in the molten fuel salt.

The molten fuel salt has a high boiling point; thus, the pressure at the top of the reactor core is at atmospheric pressure. The molten fuel salt enters the reactor vessel at 565EC (1,050EF) and exits at 705EC (1,300EF). An inert cover gas is used to prevent unwanted chemical reactions.

**Table 2. Characteristics of a molten-salt reactor**

Net electric generation	1,000 MW		Core flow velocity	2.6 m/s (maximum)
Thermal efficiency	44.4 %		Total fuel salt	48.7 m <sup>3</sup>
Core height	3.96 m		<sup>233</sup> U	1,500 kg
Vessel design pressure	5.2 10 <sup>5</sup> N/m <sup>2</sup> (75 psi)		Thorium	68,100 kg
Power density	22.2 kW/L (average)		Salt components	<sup>7</sup> LiF-BeF <sub>2</sub> -ThF <sub>4</sub> -UF <sub>4</sub>
Graphite mass	304,000 kg		Salt composition (See line above)	71.7-16-12-0.3 mol %

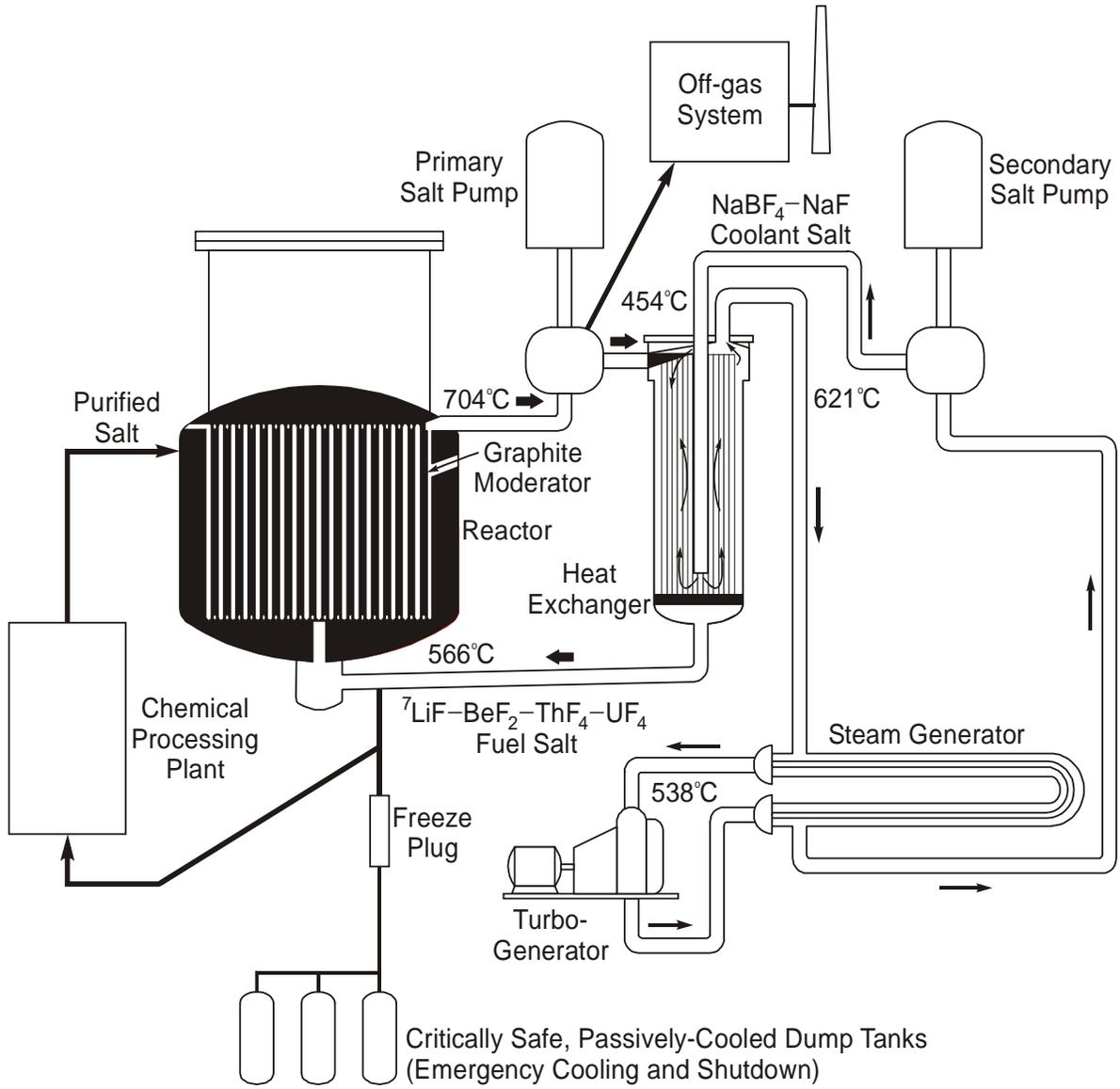
In the molten fuel salt, xenon and other fission product gases are stripped from the salt in the primary-system circulation pumps. The reactor has control rods for rapid shutdown; however, during normal operation, the control rods are fully withdrawn. The reactor and primary system are constructed of Hastelloy for corrosion resistance to the molten salt.

The hot, molten fuel salt from the reactor flows to a primary heat exchanger, where the heat is transferred to a heat-transfer fluid. The molten fuel salt flows back to the reactor core. The heat-transfer coolant (NaBF<sub>4</sub>-NaF) provides isolation between the molten fuel salt and the steam cycle. The heat-transfer fluid flows to a steam generator, where it produces steam, and then back to the primary heat exchanger. A conventional, steam cycle converts the heat to electricity. The plant's electrical efficiency is ~44%. This high thermal efficiency is a consequence of the high operating temperatures of the reactor. The temperatures are determined by the need to ensure low-salt viscosity and a significant margin between the salt melting point and the system operating temperature. The operating temperature is a consequence of the selection of the salt composition.

#### **4.1.2 History**

There is a strong technical foundation to the MSBR. The original MSR program was (1) a large program in the 1960s and (2) the primary backup to the liquid-metal, fast-breeder reactor program. In addition to the construction and operation of the Molten Salt Reactor Experiment as described above, the program investigated many design options, operated large test loops, developed special alloys for the reactor system, and included industrial partners with an interest to commercialize the technology.

When the MSBR was being developed, the emphasis was on a breeder reactor because it was thought that uranium resources were extremely limited and thus the cost of fuel would control nuclear power economics. Proliferation resistance was not a design consideration. Safety was not a public concern and was to be addressed by traditional, active safety systems. Waste management was not an issue. Consequently, it is a reactor for which a large base of information exists but the reactor concept has not been seriously examined in the context of current requirements.



## Proliferation-Resistant Molten-Salt Reactor

### 4.1.3 Differences

A PR-MSR would be similar to the MSBR *except for changes in reactor core design and the associated fuel-salt processing system to change the proliferation-resistance characteristics*. The major changes in design are:

- *Uranium-233*. The MSBR fuel was weapons-usable  $^{233}\text{U}$ . The PR-MSR fuel would be a mixture of non-weapons usable  $^{233}\text{U}$ ,  $^{238}\text{U}$ , and other uranium isotopes.
- *Plutonium*. The PR-MSR core design and salt-processing systems are modified to suppress plutonium inventories.
- *Other sources of weapons-usable materials*. The PR-MSR is modified to minimize the potential for the reactor to be used for production of other weapons-usable materials.

The fuel cycles of the MSBR and PR-MSR are similar. The reactors are started on LEU. The molten salt contains thorium, which is converted to  $^{233}\text{U}$ . As the  $^{235}\text{U}$  burns out, the reactor fuel becomes  $^{233}\text{U}$ , which was created in the reactor. Fresh thorium is added to replace that converted to  $^{233}\text{U}$ . A chemical process system removes the fission products from the salt. Fissile and fertile materials are never removed from the fuel salt. The fissile materials only leave the reactor as fission products. Fertile materials are converted to fissile materials and also leave the reactor as fission products.

The major difference in fuel cycles between the MSBR and the PR-MSR is that natural or depleted uranium is added during the lifetime of the PR-MSR to replace  $^{238}\text{U}$ , which is converted to plutonium and then fissioned. Sufficient  $^{238}\text{U}$  is added, when needed, such as to ensure that the  $^{233}\text{U}$  is always non-weapons usable by isotopic dilution.

In the 1970s, several limited studies were undertaken to identify methods to improve the proliferation resistance of the MSBR. One study (Engel 1978 and 1979) examined the possibility of an MSBR that operates with isotopically diluted  $^{233}\text{U}$ —as described above. The study indicated that isotopic dilution of  $^{233}\text{U}$  is feasible, but it did not examine how to reduce the resultant plutonium inventory. However, this study and other studies showed that the reactor had (1) low inventories of plutonium, (2) unusual plutonium and uranium isotopics, and (3) strong dependencies of the plutonium inventory on particular core design features. The limited study did not examine the implications of these observations.

## 4.2 HOW LOW CAN THE WEAPONS-USABLE FISSILE INVENTORY BE IN A PR-MSR?

The design objective for a PR reactor is to minimize the weapons-usable fissile material inventory. Scoping calculations can provide a perspective on the possibility of obtaining the objectives.

### 4.2.1 Allowable Inventory of Fissile Materials in the Reactor

The IAEA defines 8 kg of plutonium as sufficient to construct one nuclear weapon. If this is used as a limit for weapons-usable materials in the reactor, the total allowable fissile inventory (weapons-usable and non-weapons-usable) can be determined.

- *Ratio of plutonium to isotopically diluted  $^{233}\text{U}$* . For any particular reactor design using a  $^{233}\text{U}$ -Th fuel cycle, there will be a ratio of the plutonium inventory to the inventory of isotopically diluted,

non-weapons-usable  $^{233}\text{U}$ . This ratio should be minimized to minimize the quantities of weapons-usable fissile materials. Using the logic developed in Sect. 3 for an MSR and imposing the requirement that the  $^{233}\text{U}$  be isotopically diluted to 12 wt %  $^{233}\text{U}$  in  $^{238}\text{U}$  (non-weapons-usable  $^{233}\text{U}$ ), it was determined that equilibrium concentration ratio of  $^{239}\text{Pu}$  to  $^{233}\text{U}$  is about 0.0195 (see Appendix A). In LWR SNF, the corresponding  $^{239}\text{Pu}$  to  $^{233}\text{U}$  ratio typically exceeds 0.70.

- *Fissile inventories.* If the quantity of  $^{239}\text{Pu}$  allowed in the reactor is that needed for one weapon (8 kg), the allowable  $^{233}\text{U}$  in the reactor is 410 kg (8kg/0.0195).

#### 4.2.2 Impacts of Fuel Isotopics

The previous  $^{233}\text{U}$  inventory calculation does not account for fuel isotopics—this has a significant impact on the allowable  $^{233}\text{U}$  inventory in the reactor. An MSR does not remove uranium or plutonium from the reactor—they remain in infinite, continuous recycle until fissioned or transmuted (Engel 1978). This results in very unusual uranium and plutonium isotopics.

Under these conditions, there is a significant buildup of  $^{234}\text{U}$  and  $^{236}\text{U}$  in the fuel. Not all  $^{233}\text{U}$  fissions. Some is converted to  $^{234}\text{U}$ —which does not fission. The  $^{234}\text{U}$  absorbs neutrons and becomes  $^{235}\text{U}$ —a fissionable isotope. Most of the  $^{235}\text{U}$  fissions, but some does not and is converted into  $^{236}\text{U}$ —a non-fissionable isotope. These isotopes, like  $^{238}\text{U}$ , isotopically dilute the  $^{233}\text{U}$  and reduce its usability in a nuclear weapon. This isotopic dilution reduces the quantity of  $^{238}\text{U}$  required to convert  $^{233}\text{U}$  to non-weapons-usable  $^{233}\text{U}$  and, consequently, reduces the quantities of plutonium that are produced. Assessments have not yet been made on how much the  $^{238}\text{U}$  inventory of the reactor may be reduced in the presence of other uranium isotopes while maintaining the  $^{233}\text{U}$  as nonweapons usable. Any reduction in  $^{238}\text{U}$  directly reduces the plutonium inventory of the reactor and increases the allowable  $^{233}\text{U}$  in the reactor.

Unlike conventional fuel cycles, in an MSR (Engel 1978, Benedict 1957) the  $^{242}\text{Pu}$  becomes the dominant plutonium isotope, and  $^{239}\text{Pu}$  becomes a minor plutonium isotope. This is a characteristic of thermal-neutron reactors with very high fuel burnups. In the specific case of the PR-MSR, the plutonium is never removed from the reactor and thus the plutonium burnups are higher than those seen in any other reactor. Not all  $^{239}\text{Pu}$  fissions. After neutron absorption, a small fraction of the  $^{239}\text{Pu}$  is converted to  $^{240}\text{Pu}$ . Further neutron irradiation converts the  $^{240}\text{Pu}$  to  $^{241}\text{Pu}$ . With neutron irradiation, most of the  $^{241}\text{Pu}$  is fissioned, but a small fraction is converted to  $^{242}\text{Pu}$ . Plutonium-242 has a much lower neutron absorption cross section than any other plutonium isotope and thus builds up in the reactor.

When the IAEA definition of 8 kg as the quantity of plutonium necessary to manufacture one weapon was developed, power reactors produced plutonium, which was 80+%  $^{239}\text{Pu}$ . No one was considering a reactor in which  $^{239}\text{Pu}$  is a minor plutonium isotope. The critical mass of  $^{242}\text{Pu}$  is about one order of magnitude greater than that for  $^{239}\text{Pu}$ . Furthermore, the high concentrations of  $^{240}\text{Pu}$  and other isotopes that produce spontaneous fission neutrons complicate weapons design (Sahin 1980). The implication is that for a PR-MSR (or related concepts), the IAEA definition should be modified to account for plutonium isotopics. If 8 kg of  $^{239}\text{Pu}$  is required to build a weapon, a larger—but currently undefined—quantity of plutonium would be required if the primary plutonium isotope is  $^{242}\text{Pu}$ .

The implication of these considerations is that the allowable quantity of  $^{233}\text{U}$  for a 1,000-MW(e) PR-MSR is some multiple of 410 kg because more than (a) 8 kg of plutonium with  $^{242}\text{Pu}$  and (b) >12%  $^{233}\text{U}$  in  $^{238}\text{U}$  in some mixture of other uranium isotopes are allowable (if the goals are to limit the plutonium inventory to less than that in one nuclear weapon while still meeting nonproliferation goals).

### 4.2.3 Fissile Material Requirements for an MSR

MSRs have the desirable characteristic of low total fissile inventories, as compared to most other reactors. The conceptual design of the 1,000-MW(e) MSBR system (reactor core, heat exchangers, processing systems, etc.), had a total fissile inventory ( $^{233}\text{U}$ ) of only 1,500 kg (Robertson 1971, Bettis February 1970)—about 3 times the inventory defined above before accounting for the unusual uranium and plutonium isotopes. This is half the fissile inventory of an LWR and is partly a result of eliminating most of the heat-transfer limitations in the reactor core. The power density in the reactor core of the MSBR was about half that of a modern PWR. However, the power density in the fuel salt with the dissolved uranium is higher. Because most of the heat is generated in the molten salt with the uranium, heat transfer in the reactor core does not limit fuel power density. In contrast, in solid-fueled reactors, many fuel elements are required to provide a sufficiently large heat-transfer surface. This, in turn, increases the inventory of fissile material in the reactor.

It is unclear what the practical lower fissile-inventory limits of a MSRs are. In the 1960s, the swelling of graphite in radiation fields limited reactor power densities. Higher power densities (neutron fluxes) implies more frequent replacement of the graphite. In the MSBR, the power densities were chosen to limit graphite replacement to once in every four years. In four decades, there have been major improvements in graphite.

These and other considerations indicate the potential for very-low inventories of weapons-usable fissile materials in some types of power reactors. There are significant uncertainties. Some of these are associated with reactor design. Others are associated with the lack of information on (1) the relative value in weapons of different isotopes of plutonium—such as  $^{242}\text{Pu}$ — and (2) the impact of uranium isotopes other than  $^{238}\text{U}$  on denaturing  $^{233}\text{U}$  and  $^{235}\text{U}$ . If PR nuclear power systems are to be developed, internationally agreed-to measures of the relative weapons-usable potential of different fissile materials are needed.

### 4.2.4 Minor Fissile Isotopes

The quantities of minor fissile isotopes are very small. Neptunium production is low because the fuel is  $^{233}\text{U}$  and four neutron absorptions are required to reach  $^{237}\text{Np}$ . Americium and curium inventories are low because the reactor is designed to minimize plutonium inventories, which, in turn, minimizes the production of the higher actinides.

## 4.3 OTHER FACTORS IMPACTING PROLIFERATION RESISTANCE

There are several other systems and characteristics of the PR-MSR that impact proliferation resistance. These are primarily associated with the unique characteristics of liquid-fueled reactors that have advantages and disadvantages compared to solid fueled reactors.

### 4.3.1 Chemical Separations

Fission products accumulate in the fuel salt and must be separated to (1) avoid parasitic neutron capture (that is, ensure that the CR is greater than 1) and (2) avoid exceeding fission product solubility limits in salt and precipitating fission products as solids. The removal rate of a particular fission product may be determined by any one of these limits. The uranium and actinides remain with the molten salt until fissioned.

MSRs separate fission products from the fuel. This is opposite of conventional reprocessing fuel cycles where the SNF is processed for recovery of plutonium and subsequent recycle of the plutonium back to the reactor. This approach is highly desirable in terms of proliferation resistance since it avoids the existence of purified fissile materials. However, the historical rationale for this approach is based on economics, not proliferation resistance.

- *Desired products.* The desired product from reprocessing LWR SNF is plutonium. A ton of SNF typically contains >950 kg of uranium, 10 kg of plutonium, 300+ kg of zircalloy, and tens of kgs of fission products. The processes are designed to extract the small quantities of plutonium from large quantities of other materials. In an MSR, the desired products are: uranium ( $^{233}\text{U}$  in  $^{238}\text{U}$ ), expensive isotopically-separated  $^7\text{Li}$ , and plutonium (very small quantities). The beryllium, thorium, and fluorides are expensive to treat as wastes; consequently, it is desirable to recycle these chemicals. The waste to be extracted—the fission products—is less than 0.1% of the salt. Under such circumstances, there are strong incentives to extract the fission products from the salt rather than separate and recycle 99.9% of the salt components from the fission products.
- *Product purity.* Fission product removal is dictated by the two factors described above. In most cases, reducing fission product concentrations by a factor of ten in the salt more than meets the requirements. There are no incentives for clean salts. In conventional fuel cycles with reprocessing, the fission product concentrations in the plutonium and uranium are reduced by a factor of a million or more. This reduction is necessary to reduce the radiation levels such as to allow economic fabrication of complex, precision fuel elements. An MSR with a liquid fuel has no comparable fuel fabrication.

The most important fission-product-removal operation is the removal of noble gases. The noble gases, particularly certain xenon isotopes, are strong neutron absorbers. Without the quick removal of the gases, the neutrons absorbed by these gases would prevent the reactor from having a CR equal to or exceeding 1. The most important fission-product-separation processes—including removal of noble gases—were demonstrated in the MSRE.

For proliferation resistance, it is important that the chemical separation facilities not have the capability and can not be easily altered to allow recovery of any weapons-usable materials (Appendix B).

#### **4.3.2 Waste Management and Waste Thresholds**

The IAEA has defined waste thresholds (Forsberg July 1998) for some wastes. The *waste threshold* is the concentration of fissile material in the waste below which international safeguards may be terminated. The IAEA recognizes that at some concentration of fissile material in waste, the fissile material is not practically recoverable. It would be easier for the nation-state to make new fissile materials. The waste threshold concentration is not a single number. The concentration varies depending upon (1) the fissile material and (2) the waste form.

In a PR-MSR or other PR reactor concept, the waste system must be designed to assure that the concentrations of weapons-usable fissile materials in the final waste form are below these IAEA-defined waste thresholds. If this is not done, the waste becomes a potential source of weapons-usable materials. However, at the same time, small quantities of fissile materials can be sent to the waste system to reduce the inventories of specific weapons-usable fissile materials in the reactor. For proposed MSR waste systems, this is easily accomplished.

#### 4.4 PROLIFERATION RESISTANCE

Theft of nuclear materials by subnational groups from a reactor such as PR-MSR is not credible. The initial fuel is non-weapons-usable LEU or non-weapons-usable  $^{233}\text{U}$ . There is no SNF. The total inventory of potentially usable weapons materials in the reactor is small and is mixed with >100,000 kg of highly radioactive fuel salt in the reactor. Building, testing, and operating (1) a large transfer system to move the salt from the reactor to shipping casks for offsite processing or (2) a separations plant on site to recover weapons-usable fissile materials are not credible for a subnational group.

Secret diversion of nuclear materials by nation-states may be possible—but may not be attractive for realistic scenarios. With a small total inventory of weapons-usable fissile materials, loss of relatively small amounts of material result in large percentage changes of particular isotopes in the fuel salt. This makes secret diversion very difficult. For a nation-state to divert a significant quantity of weapons-usable material successfully, the nation would have to divert weapons-usable fissile materials for many years—perhaps decades—without detection.

A nation-state may abrogate its treaty commitments and divert weapons-usable material from a power reactor. A nation that builds 1,000-MW(e) reactors has significant resources. There are many options to produce weapons-usable fissile materials.

- *SNF.* If SNF is available, it is unquestionably a preferable source of chemically separable, weapons-usable fissile materials, as compared to obtaining such materials from a PR-MSR. The concentrations of weapons-usable materials in SNF are much higher than those in the PR-MSR fuel salt. SNF plutonium is primarily  $^{239}\text{Pu}$  (the preferred isotope for nuclear weapons).
- *Alternative technologies.* Some of the other technologies, such as centrifuge isotopic enrichment, are becoming widely available because the underlying technologies are widely used in other industries (e.g., for gas turbines and high-strength composites for aircraft and sporting goods). Old technologies, such as calutrons (Benedict 1957), have been available for decades.
- *PR-MSR.* The quantities of weapons-usable fissile materials are limited. The unusual isotopics would create major challenges to the weapons designer. If significant quantities of weapons-usable materials are removed, the reactor will be shut down because of a lack of fuel.

The goal for a PR nuclear power system is a system about which both terrorists and nation-states conclude that other approaches to obtain weapons-usable materials are clearly preferable to obtaining weapons-usable materials from the civilian nuclear power industry. The previous considerations indicate the potential to achieve this goal. Whether the goal can be achieved depends upon both the detailed characteristics of the reactor and the viability of alternative methods to weapons-usable materials.

#### 5. SOLID-FUELED REACTORS

As noted in Sect. 3, there may be solid fueled reactors with low inventories of weapons-usable materials. However, solid-fueled reactors discharge SNF. Minimizing weapons-usable fissile materials in the SNF to very low levels is a major challenge. There are several possible candidate reactors—such as a modified high-temperature, gas-cooled reactor (HTGR) and a modified Canadian Deuterium Uranium reactor (Banerjee June 1977)—using modified thorium fuel cycles with non-weapons-usable  $^{233}\text{U}$  or  $^{235}\text{U}$ . In some respects, the HTGR is the solid-fueled version of the PR-MSR ( $^{233}\text{U}$ -Th fuel cycle, thermal neutrons,

and graphite moderator). HTGR studies (Teuchert 1978; Abbott 1978) indicate this reactor can have high conversion ratios (but less than 1), low plutonium production, and  $^{242}\text{Pu}$  as the primary plutonium isotope in the SNF.

If nuclear power is used on a global scale, it may be used for different applications. The characteristics of the PR-MSR and other liquid-fueled reactors are probably unsuitable for some applications. The HTGR is used as example herein of a solid-fueled reactor because this particular reactor is currently under development to address the two nuclear applications that a liquid-fueled reactor might not fill:

(1) producing very-high temperatures for chemical processes such as future production of hydrogen for fuel and (2) small modular reactors. Because of these types of uses, the approaches described herein to reduce the inventory of weapons-usable materials should be considered for solid-fuel reactors.

## 6. GLOBAL NUCLEAR SYSTEMS AND ALTERNATIVE FUEL CYCLES

A PR-MSR may be used as a component of a larger nuclear system to destroy inventories of chemically separable, weapons-usable materials from other reactors. *If it is used in these alternative fuel-cycle modes, the inventory of weapons-usable materials in the reactor will be significant.* For a PR-MSR, the decision of whether it is desirable to burn plutonium and SNF is a complex economic, political, and environmental decision based on non-proliferation, waste management, and other considerations.

### 6.1 BURNING OF SNF

The PR-MSR could be used to burn SNF. Most of the world's SNF (including LWR SNF) is in the chemical form of uranium dioxide. This oxide (a mixture of uranium, plutonium, fission products, and other actinides) can be converted to a fluoride salt. The existing technologies provide viable methods to remove most of the uranium from the salt, but these methods do not remove the plutonium—the plutonium remains with the fission products. This chemical conversion does not separate weapons-usable fissile materials from the fission products; thus, the proliferation risks are minimized. This capability creates two options.

- *Initial fueling.* The PR-MSR could be initially fueled with a plutonium-fission-product salt rather than LEU. The plutonium will burn out over time with the conversion of thorium to  $^{233}\text{U}$ . Some  $^{238}\text{U}$  from the SNF would also be included to denature the  $^{233}\text{U}$  as it is created. Because of the small fissile inventory of this type of reactor, the current and future SNF inventory from existing reactors would be sufficient such as to start up several thousand PR-MSRs.
- *Conversion of weapons-usable to non-weapons-usable fissile material.* The PR-MSR can be modified to become an intrinsically efficient nuclear system to convert weapons-usable plutonium (separated or mixed with fission products from SNF) into a non-weapons-usable mixture of  $^{233}\text{U}$  and  $^{238}\text{U}$ . A single large reactor could convert ~700 kg of plutonium to non-weapons-usable  $^{233}\text{U}$  each year. This characteristic is a consequence of the PR-MSR being a liquid-fuel reactor with a  $\text{CR} = 1$ . When operated in this mode, the reactor fuel contains plutonium, thorium, and only small concentrations of  $^{233}\text{U}$  and  $^{238}\text{U}$ . The excess  $^{233}\text{U}$ - $^{238}\text{U}$  mixture is removed continuously. When the concentration of uranium is kept very low, little of the  $^{233}\text{U}$  is fissioned, and very little  $^{239}\text{Pu}$  is produced from  $^{238}\text{U}$ ; consequently, almost all fissions involve the fissioning of plutonium that was added to the reactor. This maximizes the plutonium destroyed per unit of energy produced. The non-weapons-usable  $^{233}\text{U}$ - $^{238}\text{U}$  mixtures could be used to fuel other reactors or stored if the consumption of the  $^{233}\text{U}$ - $^{238}\text{U}$  did not match production rates.

In solid fueled reactors with plutonium and thorium, the  $^{233}\text{U}$  is produced and partly consumed because it stays in the reactor until the reactor is de-fueled. Much of the reactor energy results from the fissioning of  $^{233}\text{U}$ — not the fissioning of plutonium. To ensure that the  $^{233}\text{U}$  is non-weapons-usable,  $^{238}\text{U}$  must be in the initial fuel to isotopically dilute the  $^{233}\text{U}$  to non-weapons-usable  $^{233}\text{U}$  as the  $^{233}\text{U}$  is made. Some of this  $^{238}\text{U}$  will be converted to plutonium, which must also be burned. These factors imply lower plutonium destruction rates per unit of energy produced, as compared to what happens in a PR-MSR.

## 6.2 SECURE NUCLEAR PARKS

The potential capability of a PR-MSR to convert weapons-usable plutonium into a non-weapons-usable mixture of  $^{233}\text{U}$  and  $^{238}\text{U}$  has two implications. First, it is an efficient method for destroying excess stocks of plutonium. Second, it becomes the preferred reactor for certain types of PR fuel cycles. There have been many proposals for nuclear systems (Pigford 1978) that contain two types of reactors.

- *Distributed reactors.* The distributed reactors would be fueled with non-weapons-usable LEU or non-weapons-usable  $^{233}\text{U}$ . Any SNF discharged from these reactors would be immediately sent to a secure nuclear park.
- *Secure nuclear parks.* In a few secure nuclear parks, the SNF would be processed, and the plutonium would be destroyed by conversion to non-weapons-usable  $^{233}\text{U}$ . The non-weapons-usable  $^{233}\text{U}$  could fuel distributed reactors. All proliferation-sensitive operations would be conducted inside the secure nuclear parks.

Several characteristics of the PR-MSR can improve the proliferation resistance of this type of fuel cycle.

- *Minimize the security envelope.* This minimization of security needs is accomplished by minimizing the number of reactors and quantities of chemically-separable fissile materials located in the secure nuclear park versus those located elsewhere. This can be effected by: (1) choosing reactors for distributed deployment that minimize production of weapons-usable materials and (2) choosing reactors in the secure nuclear park with the highest rates of conversion of weapons-usable materials to non-weapons-usable materials per unit energy produced. Inside the park, the PR-MSR, as a liquid fueled reactor, is a preferred machine for this task.
- *Minimizing proliferation risks inside the secure nuclear park.* A PR-MSR, as a liquid fueled reactor, can accept a plutonium-fission product mixture as a fuel. The use of this highly radioactive mixture from SNF as a feed avoids the presence of chemically separated, weapons-usable material inside the secure nuclear park. Plutonium-fission product mixtures are handled only once before being placed in a PR-MSR and converted to non-weapons-usable  $^{233}\text{U}$ . This avoids large flows of plutonium in any form in the secure nuclear park.

This type of operation provides a basis for destruction of SNF and plutonium from existing LWRs or future reactors such as HTGRs. The non-weapons-usable  $^{233}\text{U}$  could be recycled back to these reactors. This system allows burning of the plutonium without creating chemically separated, weapons-usable materials.

### 6.3 GLOBAL ENERGY SUPPLY

The long-term strategic incentive for nuclear power is the potential, if necessary, to meet the world energy demand for thousands of years. There have been many studies of how to meet global demands including studies using secure nuclear parks (Abbott December 1978, DOE 1979). Most of these studies were done in the late 1970s under a set of ground rules that assumed (1) very rapid growth in electric power demand, (2) limited uranium resources, and (3) small existing inventories of SNF. Under these conditions, fast-breeder reactors were required to produce fuel rapidly to meet energy demands.

Conditions have changed. There are large inventories of SNF, uranium resources are larger than originally thought, and ultimate world energy demands are lower than originally projected. Under these changed conditions, a very-long-term, nuclear power enterprise that could meet the world's energy demands, if required, may be viable with a reactor such as the PR-MSR. There are two resource issues: (1) fissile material to start the reactor and (2) fissile material to continue operation of the reactor. Because the PR-MSR has a  $CR > 1$ , only very small quantities of thorium and uranium are required for continued operation. Under these circumstances, the price of thorium and uranium has almost no impact on the cost of power from such a reactor; hence, long-term operation is not an issue.

The major question is the availability of fissile resources to start up large number of reactors if nuclear power becomes the primary energy source for the planet. In this context, a thermal “breeder” reactor with a CR equal to or greater than one has an important advantage. The fissile material inventory for reactor startup is an order-of-magnitude less than for a fast reactor. This characteristic—combined with known SNF and fissile uranium resources—suggests that fissile material will not limit reactor use. For the next several decades, uranium prices are likely to be sufficiently low such that there is no limitation on available fissile resources. In the longer term, the options exist to convert the SNF into a  $^{233}\text{U}$ - $^{238}\text{U}$  mixture to provide fissile material for reactor startup at dispersed locations.

The total fissile inventory of this type of nuclear system is one to two orders of magnitude less than that based on fast reactors because of (1) the low fissile inventory of as MSR as compared to a fast reactor and (2) the lack of SNF with its associated fissile inventories. The inventories of chemically separable, weapons-usable fissile materials in such a thermal “breeder” reactor system are several orders of magnitude less than the fast-breeder reactor systems that have been historically studied. There are many uncertainties with such an approach because only limited studies have been conducted.

## 7. CONCLUSIONS

The historical “belief” is that nuclear power and nuclear weapons proliferation are tightly connected. Radioactivity has been considered the primary barrier to recovering chemically separable, weapons-usable fissile materials from power reactor SNF. However, there is another approach to improve the proliferation resistance of nuclear power—reduce the inventory of chemically separable, weapons-usable fissile materials.

Can we build practical nuclear power systems that disconnect nuclear power from nuclear proliferation using this alternative approach? Are there multiple reactor concepts with this property—solid and liquid fueled reactors? Can practical concepts for burning SNF be developed? The definitive answers do not yet exist. What can be said is that the old belief that nuclear power implies significant quantities of chemically separable, weapons-usable materials is an assumption, not necessarily a reality. Clearly, the inventories of weapons-usable materials in nuclear systems can be significantly reduced.

## APPENDIX A: RATIO OF WEAPONS-USABLE TO NON-WEAPONS-USABLE MATERIALS

Power reactors require large inventories of fissile materials. The key to minimizing the quantities of weapons-usable fissile materials in nuclear power systems is to minimize the ratio of weapons-usable to non-weapons-usable fissile material—that is, plutonium to non-weapons-usable  $^{233}\text{U}$ . The analysis herein estimates this ratio for an idealized PR-MSR.

Four nuclear reactions are important: production of  $^{233}\text{U}$  from  $^{232}\text{Th}$ , fissioning of  $^{233}\text{U}$ , production of  $^{239}\text{Pu}$  from  $^{238}\text{U}$ , and the fissioning of  $^{239}\text{Pu}$ . To explain how the ratio of plutonium to isotopically diluted  $^{233}\text{U}$  is minimized, the following simplified assumptions are used: (1) all neutrons are at thermal energies, (2) all fissile materials fission with one neutron, (3) all fertile materials are converted to fissile materials with one neutron, and (4) the absorption cross sections of structural materials, moderators, and fission products are small and can be ignored. Using these assumptions, the ratio of plutonium to isotopically diluted  $^{233}\text{U}$  in the reactor can be determined.

- *Production ratio of  $^{239}\text{Pu}$  to  $^{233}\text{U}$ .* The production ratio of  $^{239}\text{Pu}$  to  $^{233}\text{U}$  in the reactor is.

$$\text{Production ratio } ^{239}\text{Pu} : ^{233}\text{U} = (R_{238\text{U}} \times F_{238\text{U}}) / (R_{232\text{Th}} \times F_{232\text{Th}})$$

where

$R_{238\text{U}}$  is the ratio of the number density of  $^{238}\text{U}$  atoms in the salt to the  $^{233}\text{U}$  in the salt. To ensure that the  $^{233}\text{U}$  is non-weapons-usable, the  $^{233}\text{U}$  is isotopically diluted with  $^{238}\text{U}$  where  $R_{238\text{U}}$  equals 7.3. This value of  $R_{238\text{U}}$  equals 12%  $^{233}\text{U}$  in  $^{238}\text{U}$ —the dividing line between weapons-usable and non-weapons-usable  $^{233}\text{U}$ .

$F_{238\text{U}}$  is the thermal-neutron absorption cross section for  $^{238}\text{U}$  and is equal to 2.7 barns

$R_{232\text{Th}}$  is the ratio of the number density of  $^{232}\text{Th}$  to  $^{233}\text{U}$ .

$F_{232\text{Th}}$  is the thermal cross section of  $^{232}\text{Th}$ .

For every fission of  $^{233}\text{U}$ , another  $^{233}\text{U}$  will be generated in the thermal-breeder reactor by absorption of neutrons by  $^{232}\text{Th}$ . In this system, the CR equals  $\sim 1$  ( $^{233}\text{U}$  production =  $^{233}\text{U}$  destruction). As a consequence,  $R_{232\text{Th}} \times F_{232\text{Th}} = F_{233\text{U}}$ . The thermal-neutron-absorption cross section of  $^{233}\text{U}$  is 578.8 barns. Substituting these values into the above equation yields:

$$\text{Production ratio } ^{239}\text{Pu} : ^{233}\text{U} = (R_{238\text{U}} \times F_{238\text{U}}) / (F_{233\text{U}}) = (7.3 \times 2.7) / (578.8) = 0.03$$

- *Inventory ratio of  $^{239}\text{Pu}$  to  $^{233}\text{U}$ .* The thermal cross section of  $^{239}\text{Pu}$  ( $F_{239\text{Pu}} = 1011.3$ ) is larger than the thermal cross section of  $^{233}\text{U}$  ( $F_{233\text{U}} = 578.8$  barns), so the  $^{239}\text{Pu}$  is preferentially destroyed, and the equilibrium concentration ratio of  $^{239}\text{Pu}$  to  $^{233}\text{U}$  is about 0.0195— $(0.03 \times 578.8 / 1011.3)$ .

The IAEA definition of the quantity of plutonium required for building one nuclear weapon is 8 kg. If the quantity of  $^{239}\text{Pu}$  allowed in the reactor is that needed for one weapon, the allowable  $^{233}\text{U}$  in the reactor is 410 kg (8kg/0.016). This calculation does not account for other plutonium and uranium isotopic effects (Sect. 4.2.2).

## APPENDIX B: CHEMICAL SEPARATIONS AND PROTACTINIUM MANAGEMENT

Most of the chemical separation systems associated with an MSR have little impact on proliferation resistance. There is one potential exception: protactinium (Pa) separations. Protactinium management is a unique characteristic of liquid-fueled reactors using a  $^{233}\text{U}$ -Th fuel cycle. When  $^{232}\text{Th}$  absorbs a neutron, it is converted to  $^{233}\text{Pa}$ , which then decays to  $^{233}\text{U}$ . Protactinium-233 has a 27-d half-life. The total  $^{233}\text{Pa}$  inventory is a several tens of kilograms dissolved in >100,000 kg of highly-radioactive molten fuel salt. If the  $^{233}\text{Pa}$  absorbs a neutron, it will no longer decay to fissile  $^{233}\text{U}$ . If  $^{233}\text{Pa}$  losses are too high, the reactor will no longer have a CR equal to or greater than 1 and will not meet nonproliferation goals. This must be avoided. There are two options to minimize  $^{233}\text{Pa}$  losses:

- *Limit reactor-core power density with no  $^{233}\text{Pa}$  separation capabilities.* As the reactor core power density is lowered, neutron absorption by  $^{233}\text{Pa}$  is reduced, and a larger fraction of the  $^{233}\text{Pa}$  decays to  $^{233}\text{U}$ . However, this reduction in power density implies a larger reactor core, a larger  $^{233}\text{U}$ - $^{238}\text{U}$  inventory in the reactor, and consequently a larger plutonium inventory in the reactor. This requires careful trade-offs in core design to ensure a CR of 1, which is needed to meet nonproliferation goals.
- *Separate  $^{233}\text{Pa}$ .* The  $^{233}\text{Pa}$  can be separated from the fuel salt and allowed to decay to  $^{233}\text{U}$  outside the reactor. The resultant  $^{233}\text{U}$  can then be added back to the reactor. This process reduces losses of  $^{233}\text{Pa}$  by neutron absorption in the reactor core and maximizes  $^{233}\text{U}$  production. It improves fuel economy and increases the CR. However, if the  $^{233}\text{Pa}$  were completely separated from the isotopically diluted  $^{233}\text{U}$  and then allowed to decay, it would decay into  $^{238}\text{U}$ , which was not isotopically diluted with  $^{238}\text{U}$ —a potential proliferation risk. In theory, excess fuel salt with  $^{233}\text{Pa}$ ,  $^{233}\text{U}$ , and  $^{238}\text{U}$  could be stored outside the reactor until the  $^{233}\text{Pa}$  decayed; however, the inventory of salt and  $^{233}\text{U}$  would be cost prohibitive. If  $^{233}\text{Pa}$  is to be stored outside the reactor until it decays, it must be separated from most (not all) of the fuel salt.

These considerations require that any on-line chemical processing of the fuel salt not allow for rapid, efficient separation of  $^{233}\text{Pa}$  from the  $^{233}\text{U}$ - $^{238}\text{U}$  in the fuel salt *if isotopic dilution is to be used as the only acceptable barrier to prevent access to weapons-usable materials*. There are many methods to continuously remove most fission products from the fuel salt that are not capable of separating protactinium from uranium. As long as this condition exists, there is no concern about separated  $^{233}\text{Pa}$  and subsequent decay of the  $^{233}\text{Pa}$  to  $^{233}\text{U}$ .

There are incentives to separate the  $^{233}\text{Pa}$  from the fuel salt to improve the CR of the reactor. The improvement in neutron economy can be used to modify the reactor core to minimize plutonium production and improve economics (small reactor core, etc.). There are several enabling characteristics of this system that may allow  $^{233}\text{Pa}$  separation from the fuel salt to be feasible while maintaining high proliferation resistance.

- *Limited possible  $^{233}\text{U}$  production.* For the reactor to operate, the  $^{233}\text{U}$  from the decaying of  $^{233}\text{Pa}$  must be recycled back to the reactor. It is the fuel. Only limited amounts of  $^{233}\text{U}$  (from  $^{233}\text{Pa}$ ) can be removed before the reactor shuts down because of a lack of fuel and  $^{233}\text{Pa}$  production is stopped. For the nation-state, the choice is electric power or a small inventory of  $^{233}\text{U}$ .
- *Hot  $^{233}\text{U}$ .* Any chemical separation process for protactinium separates all protactinium isotopes equally from the fuel salt—including  $^{232}\text{Pa}$ , the other protactinium isotope which decays to  $^{232}\text{U}$ . Unlike  $^{233}\text{U}$ ,  $^{232}\text{U}$  has a decay product that emits a 2.6-MeV gamma-ray. Uranium-thorium feeds, salt cleanup systems, and the reactor design can be chosen to boost the  $^{232}\text{U}$  content and radiation

levels. The variable radiation fields with time and other factors make quantifying proliferation resistance complex with this approach.

- *Incomplete  $^{233}\text{Pa}$  separation from isotopically diluted  $^{233}\text{U}$ .* The quantity of  $^{233}\text{Pa}$  in the reactor is very small, as compared to the quantity of  $^{238}\text{U}$  in the reactor. If a small fraction of a percent of the  $^{233}\text{U}$ - $^{238}\text{U}$  inventory can not be separated from the  $^{233}\text{Pa}$  because of some physical effect such as azeotrope formation, that  $^{233}\text{U}$ - $^{238}\text{U}$  mixture will isotopically dilute the new  $^{233}\text{U}$  from decay of  $^{233}\text{Pa}$  to non-weapons-usable  $^{233}\text{U}$ . There are potential  $^{233}\text{Pa}$  separation technologies with this characteristic.

Significant additional technical work is required before a full understanding of the  $^{233}\text{Pa}$  options will exist.

As a proliferation concern,  $^{233}\text{Pa}$  is only an issue if a nation abrogates its treaty commitments. Building, testing, and operating (1) a large transfer system to move the salt from the reactor to shipping casks for off-site processing or (2) a separations plant on-site to recover  $^{233}\text{Pa}$  are not credible for a subnational group and can not be hidden. For a nation, it is a high-risk method to acquire weapons-usable materials. The potential inventories of  $^{233}\text{Pa}$  are small. Any delay in  $^{233}\text{Pa}$  removal causes its decay in the presence of  $^{238}\text{U}$  and thus becomes a non-weapons-usable material. If very much  $^{233}\text{Pa}$  is removed, the reactor shuts down, and there is no power generation. High gamma radiation levels with  $^{233}\text{U}$  containing  $^{232}\text{U}$  complicate weapons fabrication, storage, and transport.

Any nation with access to SNF would clearly choose SNF over  $^{233}\text{Pa}$  as a source of weapons-usable materials: concentrations of plutonium in SNF are much higher, radiation levels are lower, and the plutonium does not rapidly decay away. Any nation that planned in advance the acquisition of weapons-usable material would likely use a different route because other routes can be pursued in secret whereas this approach implies large modifications at the power plant—an easily monitored site.

## APPENDIX C: MSR SAFETY, WASTE MANAGEMENT, AND ECONOMICS

A PR nuclear power system is of no value unless it is safe, wastes can be disposed of, and it is economic. A brief description of safety, waste management, and economic issues associated with the PR-MSR is provided herein.

### C.1 SAFETY

The characteristics of MSRs imply major differences in their safety systems, as compared to those of solid-fueled reactors. Some of these changes have impacts on proliferation resistance in nuclear power systems.

The greatest danger during a nuclear reactor accident is the catastrophic release of radionuclides to the environment. In solid-fueled reactors, the accident *source term* (quantity of fission products and actinides in the reactor) is a given. In an MSR, radionuclides are continuously removed from the molten fuel salt, solidified, and stored. This approach is a significant safety advantage because the inventory of fission products in an MSR is significantly less than that in a conventional reactor. Severe accident consequences are less because the inventory of fission products is less.

Improving safety by reducing the source term implies that the PR-MSR breaks the historical trade-off between proliferation resistance and safety. Most proposed proliferation-resistance fuel cycles use higher-burnup fuels with higher inventories of fission products in the reactor core to increase proliferation resistance by increasing radiation levels. Unfortunately this process increases the accident source term and implies greater consequences if an accident occurs. The fission product inventory of the reactor should be lowered for safety. The PR-MSR uses isotopic dilution, not radiation, as the primary proliferation barrier. In the PR-MSR, the need for the CR to be at least 1 dictates what fission products are removed and the rate of removal. The safety advantages of removing fission products from the reactor and solidifying them into a stable waste form has long been recognized but has not been quantified.

The one safety disadvantage of an MSR is that the liquid fuel requires an off-gas system to prevent releases of gaseous fission products. In solid-fuel reactors, these gases are contained in sealed fuel pins.

### C.2 WASTE MANAGEMENT

The wastes from the  $^{233}\text{U}$ -Th fuel cycle have very low actinide concentrations, as compared to those of (1) SNF or (2) conventional reprocessing HLW streams. The primary fuel is  $^{233}\text{U}$ . Most  $^{233}\text{U}$  fissions with only small quantities of  $^{234}\text{U}$  produced. The  $^{234}\text{U}$  eventually absorbs neutrons and is converted to  $^{235}\text{U}$ . Most of the  $^{235}\text{U}$  is fissioned. It takes many more neutron absorptions to create an actinide in this fuel cycle than it does during a  $^{235}\text{U}$  or  $^{239}\text{Pu}$  fuel cycle. The addition of  $^{238}\text{U}$  for isotopic dilution of  $^{233}\text{U}$  will generate some actinides. However, suppressing plutonium production to meet nonproliferation goals aids waste management by minimizing actinides in the waste.

The different waste characteristics aid waste management. The waste has (1) no significant fissile content (thus no repository nuclear criticality or safeguards issues) and (2) a low actinide content (and thus limited concerns about long-lived actinides in the repository—a particular concern to part of the public). The long-term (but small) health risks from the proposed Yucca Mountain Repository (YM) (DOE November 13, 1999) and many other proposed repositories (Nuclear Energy Agency 1999) are partly

controlled by neptunium, which is generated from  $^{235}\text{U}$  by multiple neutron capture. A  $^{233}\text{U}$  fuel cycle minimizes neptunium production and much of the neptunium that is produced is destroyed. This fuel cycle reduces neptunium quantities to the repository by one or more orders of magnitude, as compared to LWRs, and results in equivalent reductions in repository risk.

### **C.3 ECONOMICS**

The early studies indicated that the MSBR economics were slightly better than those of LWRs. These studies were performed a decade before the Three Mile Island accident and the ensuing changes in safety requirements. No recent studies have been done. There are several factors that indicate the potential for significantly lower costs (low fuel-cycle costs with no fuel fabrication, reduced safety-system costs, and reduced balance-of-plant costs from the higher-temperature steam cycle) than those of advanced LWRs and other reactors; but significant work is required before any definitive economic conclusions can be drawn.

The studies suggest that such a reactor would be more economic in larger sizes (1,000 MWe). This is a consequence of some specific characteristics of these reactors. The chemical processing, waste management, and off-gas systems are more complex than those with solid-fuel reactors. There are very large economics of scale in these auxiliary systems. The fluid flows are small. Much of the equipment is sized to avoid plugging or similar problems. In effect, there is a significant fraction of the plant for which the cost is almost independent of the size.

## REFERENCES

- Abbott, L. S., et al., December 1978. *Interim Assessment of the Denatured <sup>233</sup>U Fuel Cycle: Feasibility and Nonproliferation Characteristics*, ORNL-5388, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
- Banerjee, S., R. Hatcher, A. D. Lane, H. Tamm, and J. I. Veeder, June 1977. "Some Aspects of the Thorium Fuel Cycle in Heavy-Water-Moderated Pressure Tube Reactors," *Nuclear Technol* **34**(1),58–68.
- Benedict, M., and T. H. Pigford, 1957. *Nuclear Chemical Engineering*, McGraw-Hill Book Company, Inc., New York
- Bettis, E. S., and R. C. Roberson, February 1979. "The Design and Performance Features of A Single-Fluid Molten-Salt Breeder Reactor," *Nuclear Applications Technol. (Today: Nuclear Technology)*. **8**, 190. (The entire issue is devoted to liquid-fueled reactors.)
- Engel, J. R., W. R. Grimes, W. A. Rhoades, and J. F. Dearing, August, 1978. *Molten-Salt Reactors For Efficient Nuclear Fuel Utilization Without Plutonium Separation*, ORNL/TM-6413, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
- Engel, J. R., W. A. Rhoades, W. R. Grimes, and J. F. Dearing, November, 1979. "Molten-Salt Reactors For Efficient Nuclear Fuel Utilization Without Plutonium Separation," *Nuclear Technol.* **46** (1), 30–43.
- Forsberg, C. W., S. N. Storch, and L. C. Lewis, July 1998. *Uranium-233 Waste Definition: Disposal Options, Safeguards, Criticality Control, and Arms Control*, ORNL/TM-13591, Oak Ridge National Laboratory, Oak Ridge, Tennessee
- Forsberg, C. W., C. M. Hopper, J. L. Richter and H. C. Vantine, March 1998. *Definition of Weapons-Usable Uranium-233*, ORNL/TM-13517, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
- Fox, M. and E. Mulder, August 1996. "Pebble Bed Modular Reactor–South Africa," *Design and Development of Gas-Cooled Reactors With Closed Cycle Gas Turbines, Proc. of a Technical Committee Meeting, Beijing, China, October 30–November 2, 1995, International Atomic Energy Agency, IAEA-TECDOC-899*, Vienna, Austria.
- MacPherson, H. G., 1985. "The Molten Salt Reactor Adventure," *Nuclear Science. Engineering*, **90**, 374–380
- Nicholls, D. R., December 1998. "Eskom Sees A Nuclear Future in the Pebble Bed," *Nuclear Engineering Int.*, 12–16.
- Nuclear Technology*, February 1970, *Nuclear Applications Technol. (Today: Nuclear Technology)*. **8**. (The entire issue is devoted to liquid-fueled reactors.)
- Pigford, T. H., C. S. Yang, and M. Maeda, November 1978. "Denatured Fuel Cycles for International Safeguards," *Nuclear Technol.* **41**(1), 46–59.

Robertson, R. C., et. al., June 1971. *Conceptual Design Study of a Single-Fluid Molten-Salt Breeder Reactor*, ORNL-4541, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Sahin, S., and J. Ligou, Mid-August 1980. "The Effect of the Spontaneous Fission of Plutonium-240 on the Energy Release in a Nuclear Explosion," *Nuclear Technol.* **50**(1), 88–94.

Teuchert, E, H. J. Rutten, and H. Werner, May 1978. "Performance of Thorium Fuel Cycles in the Pebble-Bed Reactor," *Nuclear Technol.* **38**, 374–383.

U.S. Department of Energy, December 1979. *Nuclear Proliferation and Civilian Nuclear Power: Report of the Nonproliferation Alternative Systems Assessment Program*, DOE/NE-0001. Washington D.C.

Williams, R. H., and H. A. Feiveson, July–August 1990. "Diversion-Resistance Criteria For Future Nuclear Power," *Energy Policy*, pp. 543–549.