Discussion Paper

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

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

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ABSTRACT

A major concern with nuclear power is the existence in reactors and spent nuclear fuel (SNF) of chemically separable, weapons-usable materials, which could be diverted to construct nuclear weapons. Over the lifetime of a large light-water reactor (LWR), 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. The approach is described herein.

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., production of weapons-usable material is an intrinsic characteristic of the production of nuclear power). When nuclear power was first developed, the emphasis was on breeding fissile materials because it was thought that uranium resources were extremely 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 SNF. However, could a nuclear power system be designed with no significant inventories of chemically separable weapons-usable materials? Would it be practical?

To examine this issue, a set of criteria are 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 description of one reactor concept that may meet the criteria is provided. This description is helpful for understanding some of the issues associated with such a reactor.

2. DEFINITION OF PROLIFERATION RESISTANCE

One can not identify a proliferation-resistant nuclear power system unless one defines the term *proliferation-resistance*. Three criteria are used to define the idealized characteristics of a reactor and associated fuel cycle that are designed to minimize the potential connection between nuclear weapons and nuclear power.

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 spent nuclear fuel (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 proliferation resistant. 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. Within decades, plutonium recovery is greatly simplified with the ensuing reduction in radiation levels. SNF repositories become plutonium mines. 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, 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.

For example, the historical method for offshore oil production has been the construction of offshore oil platforms from which to drill oil wells and to recover the oil. In deep water, the construction costs for a single production platform are several billion dollars. In the last decade, alternative seabed production platforms have been developed using remote operation, construction, and maintenance techniques. Because the total oil-industry cost savings are measured in tens of billions of dollars, major research and development programs have been conducted in robotics with dramatic improvements in capabilities. A recent demonstration of this capability was the recovery of EgyptAir Flight 990 debris on the seabed off the eastern seaboard of the United States. Private oil industry robots designed to work on seabeds were used to allow rapid recovery of electronic black boxes and other debris in seas so rough that debris recovery by even sophisticated U.S. Navy equipment was not possible. The technology to aid in the rapid recovery of plutonium from SNF is becoming an item of international commerce.

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 ^{239}Pu, 8 kg ^{233}U, or 25 kg ^{235}U). 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 enriched uranium one time to start a reactor. 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.

A second corollary is the need to minimize other infrastructure requirements that could be diverted to support a nuclear weapons program.

3. SELECTION OF A PROLIFERATION-RESISTANT REACTOR AND FUEL CYCLE

Using these 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}U-Thorium Reactor Fuel Cycle

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

3.1.1 Plutonium

The key component of most nuclear weapons is ^{239}Pu. If a reactor and fuel cycle are to be proliferation resistant, plutonium inventories must be minimized (*Criteria I and II*). Since ^{239}Pu is produced by the neutron irradiation of ^{238}U in a reactor, the use of ^{238}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}U exists in nature only in trace quantities. There is no ^{235}U breeder reactor fuel cycle. Because ^{235}U is mined, not made, the only way to fuel a reactor with non-weapons-usable ^{235}U is to add ^{235}U (<20% ^{235}U in ^{238}U so that the ^{235}U remains nonweapons usable) at intervals to the reactor and remove the SNF with (a) its lower ^{235}U content, (b) ^{239}Pu made from the ^{238}U, and (c) the remaining ^{238}U. All non-weapons-usable ^{235}U fuel cycles generate significant quantities of plutonium in the SNF because the continued addition of ^{235}U fuel includes the implicit cycling of ^{238}U through the reactor with resultant plutonium production and discharge as SNF. A ^{235}U fuel cycle does not meet *Criterion I*.

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

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}U by two different approaches:

- *Isotopic dilution with ^{238}U*. Uranium-233 can be isotopically diluted to <12 wt % ^{233}U in ^{238}U to convert the ^{233}U to non-weapons-usable uranium (Forsberg March 1998). This is equivalent to <20% ^{235}U in ^{238}U.

- *High-radiation levels with ^{232}U*. Unless very special production techniques are used, the production of ^{233}U results in secondary production of ^{232}U, which decays through several decay products to thallium-208 (^{208}Tl), which then emits a 2.6-MeV gamma ray. Consequently, the

Option   *Basis for Disqualification*

Choose Fissile Material

- ^{239}Pu
- ^{235}U
- ^{233}U

Avoid weapons-usable fuels

- ^{239}Pu
- ^{235}U with ^{238}U which converts to ^{239}Pu

- ^{233}U isotopically diluted with ^{238}U
- Conversion ratio ≥ 1 to produce ^{233}U

Choose Neutron Spectrum

- Fast
- Thermal

Fast reactors have larger ^{239}Pu inventories

- Higher ^{239}Pu: ^{233}U ratio
- Higher fissile inventory
- Excess neutrons to produce ^{239}Pu

- Thermal-neutron reactor

Choose Reactor Type

- Liquid fuel
- Solid fuel

Solid-fuel reactor conversion ratio may not exceed 1: insufficient ^{233}U production

Solid-fuel reactor is easier to convert to weapons production

- Liquid fuel

Choose Liquid

- Salt
- Other

No other liquids with successful experience
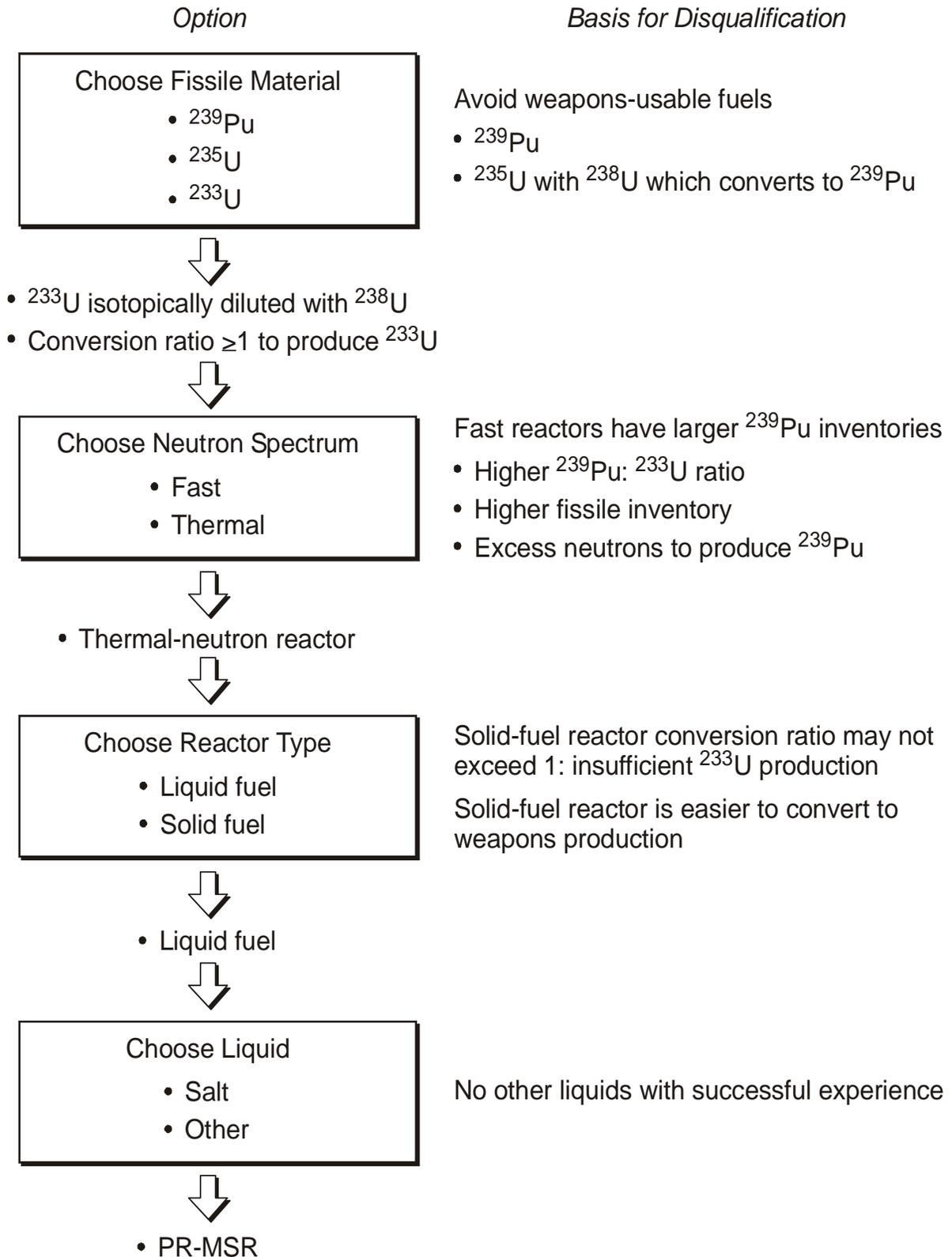
- PR-MSR

Fig 1. Logic for selection of a proliferation-resistant nuclear power system.

radiation field associated with ^{233}U increases with time. Within a decade, the radiation levels from an 8-kg sphere of ^{233}U with 1,000 ppm ^{232}U will be many tens of R/h at 1 m and hundreds of R/h at 1 ft. *Eight kilograms* is the IAEA equivalent definition of the quantity of ^{233}U needed to produce a weapon. In a typical power reactor with a ^{233}U fuel cycle, the ^{232}U content is 1,000–2,000 ppm ^{232}U. With these ^{232}U concentrations, the radiation levels become significant barriers against the use of ^{233}U in nuclear weapons or purification of ^{233}U by isotopic separation from other uranium isotopes. Only isotopic separations can separate the ^{232}U from the ^{233}U and permanently eliminate the radiation hazard. However, the decay product impurities can be removed chemically from the ^{233}U and thereby reduce the radiation hazard for short periods of time until the radiation levels build up again.

Uranium-233 can be used as the basis for a proliferation-resistant fuel cycle if one or both of the previous mechanisms are used. Isotopic dilution is preferred because it converts ^{233}U into non-weapons-usable ^{233}U. Because there is no natural source of ^{233}U, a viable ^{233}U reactor and fuel cycle must produce ^{233}U as fast as it consumes ^{233}U; that is, the conversion ratio (CR) must equal or exceed 1. The CR is the rate of production of ^{233}U divided by the rate of consumption of ^{233}U. A CR >1 implies that the reactor produces one or more kilograms of new ^{233}U from ^{232}Th for every kilogram of ^{233}U consumed. If the CR is less than 1, the reactor can not operate only on ^{233}U; consequently, an alternative source of fissile material must be supplied. The only alternative source that is practicable is ^{235}U, but the use of this material and the associated ^{238}U increases plutonium production in the reactor (*Criterion I*).

3.2 Basis for Selection of a Thermal-Neutron Reactor

Nuclear reactors can be built with fast, intermediate, and thermal neutron spectrums. Based on three considerations, a thermal-neutron reactor with the CR >1 is the only option that may meet the criteria.

3.2.1 Thermal-Neutron Reactor: Minimize Ratio of ^{239}Pu to ^{233}U Production

For a proliferation-resistant ^{233}U fuel cycle, the ^{233}U should be produced from ^{232}Th and fissioned in the presence of ^{238}U, while minimizing the production of ^{239}Pu. Sufficient ^{238}U is required such as to make the ^{233}U nonweapons usable by isotopic dilution. Ideally, the ^{238}U would be "invisible" to the reactor.

In a fast-neutron reactor, all isotopes have similar nuclear cross sections and thus similar nuclear reaction rates. Consequently, in a fast reactor, there is no way to selectively breed ^{233}U fuel from ^{232}Th in the presence of ^{238}U without creating large quantities of ^{239}Pu from ^{238}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, as discussed later in Sect. 3.2, the different nuclear cross sections can be used to minimize plutonium production in the reactor while maximizing breeding of ^{233}U in the presence of ^{238}U. There are several important synergistic nuclear effects:

- In a thermal neutron flux, fertile materials, such as ^{238}U and ^{232}Th, have low neutron-capture cross sections, whereas the fissile materials, such as ^{233}U and ^{239}Pu, have high fission cross sections. In a ^{233}U-thorium reactor with a CR >1, there are large quantities of thorium—usually 40 to 60 times as much ^{232}Th as ^{233}U. What this implies is that most of the neutrons are absorbed in ^{233}U and ^{232}Th. The ^{233}U absorbs many neutrons because of its large cross section. The ^{232}Th absorbs many neutrons because there is so much of it. In contrast, the ^{238}U has a low nuclear cross section, and there is not much of it in the reactor—only enough for isotopic dilution of the ^{233}U. The ^{238}U does not absorb many neutrons; therefore little plutonium is made.

- The production efficiency ratio of ^{239}Pu:^{233}U is smaller in a thermal flux than a fast flux. The production efficiency ratio is the ratio of nuclear absorption cross sections of ^{238}U (that converts to ^{239}Pu) to ^{232}Th (that converts to ^{233}U). In a fast flux (50 kev and up), this ratio is 0.8 (σ_{238U} = 0.12; σ_{232Th} = 0.15). In a thermal flux , this ratio is 0.36 (σ_{238U} = 2.7; σ_{233Th} = 7.4). Thermal neutron fluxes suppress production of ^{239}Pu relative to ^{233}U.

- The destruction efficiency ratio of ^{239}Pu:^{233}U is larger in a thermal neutron flux than in a fast flux. The destruction efficiency ratio is the ratio of nuclear cross sections of ^{239}Pu to ^{233}U. In a fast flux this ratio is 0.81 (σ_{239Pu} =1.77; σ_{233U} = 2.19). In a thermal flux, this ratio is 1.92 (σ_{239Pu} = 1111.3; σ_{233U} = 578.8). The relative destruction rate of ^{239}Pu to ^{233}U is more than twice as high in a thermal neutron flux as a fast flux. The inventory of ^{239}Pu is preferentially destroyed in a thermal neutron flux compared to a fast neutron flux.

3.2.2 Thermal-Neutron Reactor: Minimize Total Fissile Material Inventory

The goal is to minimize plutonium in the reactor. Since plutonium is generated from the ^{238}U, the quantity of ^{238}U should be minimized. The quantity of ^{238}U in the reactor is determined by how much is needed to make the ^{233}U non-weapons usable. To minimize the required ^{238}U, the fissile reactor fuel (^{233}U) should be minimized. Fast reactors typically require ten times the fissile inventory of that of thermal reactors because of the low neutron absorption cross sections at high neutron energies. This implies ten times as much ^{233}U and ten times as much plutonium. It would be much more difficult to achieve a proliferation-resistant fast reactor. For example, a 1,000-MW(e) LWR with a thermal neutron spectra has about 3 tons of ^{235}U and ^{239}Pu in its reactor core. An equivalent fast reactor has 25 to 35 tons of ^{235}U and ^{239}Pu. A thermal-neutron breeder reactor is preferred.

3.2.3 Thermal-Neutron Reactor: Minimize Potential Conversion to Weapons Production

Excess neutrons can be used for production of weapons-usable materials. A reactor with a CR significantly above 1 is less proliferation resistant because the excess neutrons can be efficiently used to make plutonium. Fast breeder reactors have typical breeding ratios of 1.1 to 1.3. The best thermal-neutron reactors intrinsically have CRs near 1. A thermal-neutron, ^{233}U–thorium reactor with a CR near 1 is not very useful for weapons production (*Criterion III*) because most of the neutrons are required to make new fuel for continued reactor operation. Few excess neutrons are available for other uses.

3.3 Basis for Selection of a Liquid-Fuel Reactor

The previous criteria define the general characteristics of a proliferation-resistant 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 proliferation-resistant 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 a heat transfer fluid. The liquid fuel flows back to the reactor. In the heat-transfer loop, the heat is transferred by a heat-transfer fluid to a steam generator, where steam is produced and sent to a turbine generator to produce electricity. There are several reasons for selection of a liquid-fuel reactor.

3.3.1 Liquid-Fuel Reactor: Assure Conversion Ratio Exceeds 1

A ^{233}U–^{232}Th reactor with a CR that equals or exceeds 1 is desired to avoid the need to add ^{235}U to the reactor with its excess ^{238}U and resultant production of ^{239}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}U–^{232}Th fuel cycle. *Thermal-neutron reactors with CRs equal to or exceeding 1 can not be made using a ^{239}Pu-^{238}U fuel cycle—the nuclear physics does not allow this.* Earlier studies (Engel 1978) indicate that after accounting for loses and the higher actinides, 1.06 neutrons are ultimately produced for every neutron absorbed in ^{232}Th. In contrast, only 0.9 neutrons are produced for every neutron absorbed in ^{238}U.

Denaturing ^{233}U with ^{238}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 fuel reactor for several reasons.

- *Continuous fission product removal.* The fission process creates xenon and other fission products with high neutron cross-sections. 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 significantly improves neutron utilization and thus the CR.

- *Continuous fueling.* In most solid-fuel reactors, sufficient fuel is placed in the reactor 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. Some solid fuel reactors, such as the Canadian Deuterium Uranium Reactor with on-line fueling, allow continuous fueling.

3.3.2 Liquid-Fuel Reactor: Assure Difficulty in Conversion to Weapons Production

A liquid-fueled reactor is more difficult to convert to weapons use than is a solid-fuel machine (*Criteria III*). With solid-fuel reactors, the fuel designs can be changed. This versatility has allowed U.S. production reactors to produce ^{239}Pu, ^{233}U, and tritium. The only way to ensure that this is not being done is to inspect and track all SNF—a serious challenge. It is much more difficult to change the characteristics of a liquid-fueled reactor. Furthermore, a single monitor that tracks the uranium isotopic concentration in the liquid fuel would detect any changes in reactor operation.

3.4 Basis for Selection of a Molten Salt Reactor System

Many types of ^{233}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 that 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}U fluorides dissolved in molten salts and later ^{233}U fluorides dissolved in molten salts. It operated for 13,000 equivalent full-power hours.

The MSBR is fueled with weapons-usable ^{233}U. A proliferation-resistant molten salt reactor (PR-MSR) would be a variant of this concept.

4. PROLIFERATION-RESISTANT MOLTEN SALT REACTOR

A potential example of a proliferation-resistant reactor is provided herein to better understand some of the characteristics, limitations, and issues associated with this approach to a proliferation-resistant 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 first described followed by the differences in design.

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. 1. Some of the design characteristics of the reactor are presented in Table 1.

Table 1. 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^3
Core height	3.96 m		^{233}U	1,500 kg
Vessel design pressure	5.2 10^5 N/m^2 (75 psi)		Thorium	68,100 kg
Power density	22.2 kW/L (average)		Salt components	^{7}LiF-BeF$_2$-ThF$_4$-UF$_4$
Graphite mass	304,000 kg		Salt composition (See line above)	71.7-16-12-0.3 mol %

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 product and actinide fluorides 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 565°C (1,050°F) and exits at 705°C (1,300°F). An inert cover gas is used to prevent unwanted chemical reactions.
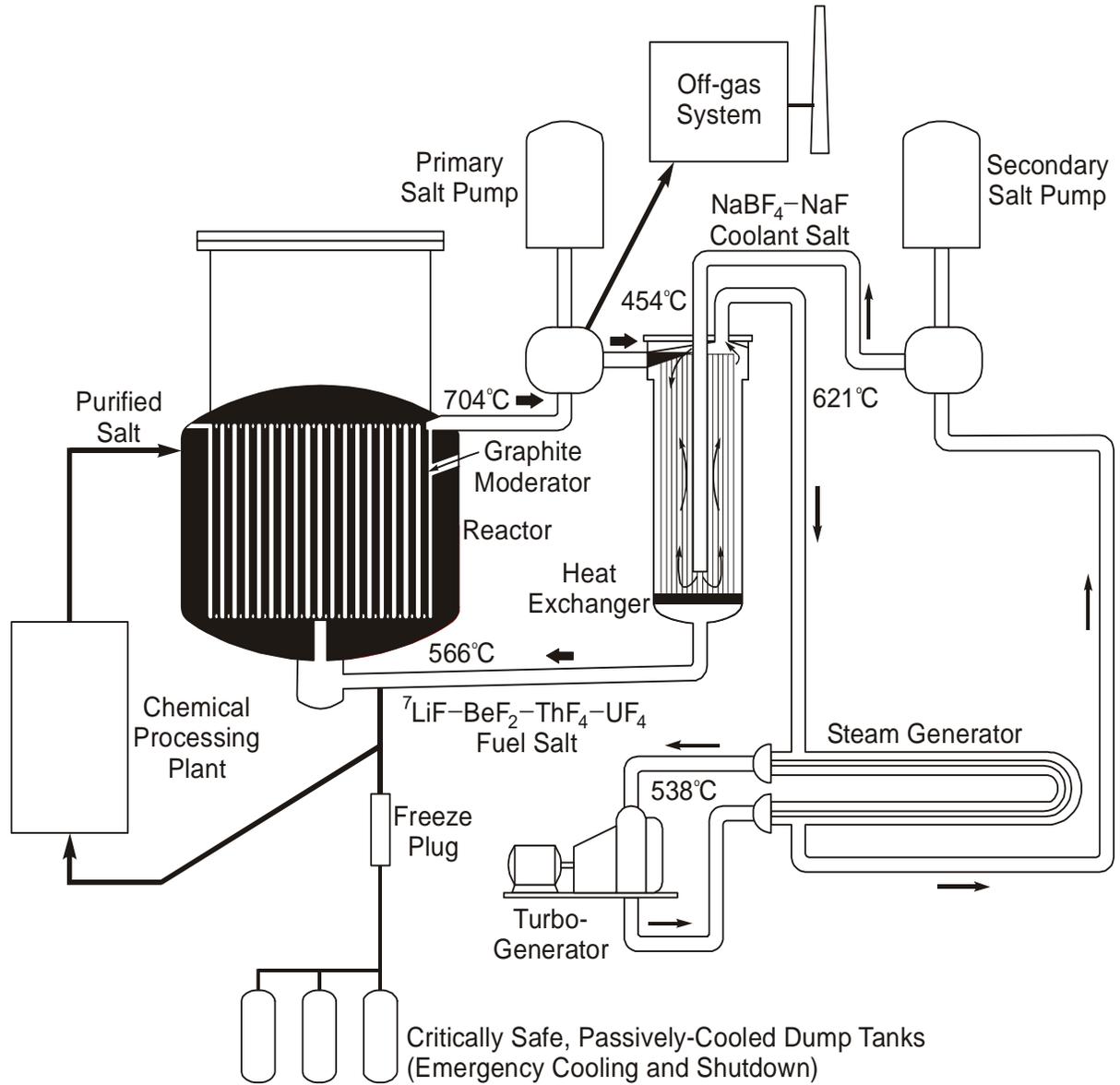
Fig 2. Proliferation-resistant MSR.

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 ($NaBF_4$-NaF) provides isolation between the molten fuel salt and the steam cycle. The heat-transfer fluid flows to a steam generator to produce steam and 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 assure low salt viscosity and a significant margin between the salt melting point and the system operating temperature. It is a consequence of the selection of the salt composition.

4.1.2 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}U. The PR-MSR fuel would be a mixture of non-weapons usable ^{233}U, ^{238}U, and other uranium isotopes.

- *Plutonium*. The PR-MSR core design and salt processing systems are modified to suppress plutonium production and 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—except for one difference. The reactor is started on low-enriched uranium. The molten salt contains thorium that is converted to ^{233}U. As the ^{235}U burns out, the reactor fuel becomes ^{233}U that was created in the reactor. Fresh thorium is added to replace that converted to ^{233}U. A chemical process system removes the fission products from the salt and converts them to an acceptable high-level waste form. 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}U that is converted to plutonium. Sufficient ^{238}U is added, when needed, to assure the ^{233}U is always non-weapons usable by isotopic dilution. The plutonium from the ^{238}U is fissioned.

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

4.2 DESCRIPTION OF THE MECHANISMS THAT MINIMIZE THE WEAPONS-USABLE FISSILE INVENTORY

A description is provided of (1) why the quantities of weapons-usable fissile materials are so small in a PR-MSR and (2) the uncertainties in terms of minimizing the weapons-usable inventory of the reactor.

4.2.1 Inventory Of Fissile Fuel And Weapons-Usable Materials

Four nuclear reactions are important in the PR-MSR: production of ^{233}U from ^{232}Th, fissioning of ^{233}U, production of ^{239}Pu from ^{238}U, and the fissioning of ^{239}Pu. To explain the proliferation-resistance characteristics of the reactor, 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. The following logic is the basis for the reactor's nonproliferation characteristics.

- For every fission of ^{233}U, another ^{233}U will be generated in the thermal breeder reactor by absorption of neutrons by ^{232}Th. In this system, the breeding ratio equals ~1 (^{233}U production = ^{233}U destruction). If σ is the nuclear neutron absorption cross section and R_{232Th} is the ratio of thorium atoms to ^{233}U atoms (number density of ^{232}Th atoms to ^{233}U atoms), then:

$$R_{232Th} \times \sigma_{232Th} = \sigma_{233U} \quad \text{(With constant thermal neutron flux)}$$

- In the same neutron flux, for every ^{233}U atom that is created, 0.03 ^{239}Pu atoms are created from ^{238}U. To ensure that the material is nonweapons usable, there are R_{238U} times as many ^{238}U atoms as ^{233}U atoms, where R_{238U} equals 7.3 (this equals 12% ^{233}U in ^{238}U—the dividing line between weapons-usable and non-weapons-usable ^{233}U). Consequently, the relative production ratio of ^{239}Pu to ^{233}U is as follows:

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

The thermal cross section of ^{238}U is 2.7 barns. Furthermore, from above we know that $R_{232Th} \times \sigma_{232Th}$ is equal to σ_{233U}. The thermal cross section of ^{233}U is 578.8 barns. Consequently, the production ratio is as follows:

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

- The thermal cross section of ^{239}Pu ($\sigma_{238U} = 1111.3$) is larger than the thermal cross section of ^{233}U ($\sigma_{233U} = 578.8$ barns), so the ^{239}Pu is preferentially destroyed, and the equilibrium concentration ratio of ^{239}Pu to ^{233}U is about 0.016—($578.8 \times 0.03/1111.3$).

The IAEA definition of the quantity of plutonium required for building one nuclear weapon is 8 kg. If the quantity of ^{239}Pu allowed in the reactor is that needed for one weapon, the allowable ^{233}U in the reactor is 500 kg (8kg/0.016).

4.2.2 Fuel Inventory

Whatever the reactor physics, there will be a ratio of weapons-usable fissile materials to isotopically diluted fissile materials in the reactor (ratio of plutonium to isotopically diluted ^{233}U). The weapons-usable inventory can be reduced by reducing the total fissile inventory (the denominator). The total fissile inventory (Robertson 1971, Bettis February 1970) in a 1,000-MW(e) MSBR system (reactor core, heat exchangers, processing systems, etc.) is only 1500 kg—significantly smaller than that of other power reactors. There are several reasons for this.

- *Thermal neutron reactor*. The fissile inventory is more than an order of magnitude lower than in a fast-breeder reactor. The lower fissile content (as compared to that of a fast reactor) is a result of the larger nuclear cross sections of fissile materials in a thermal neutron flux.

- *Reduced heat-transfer limits*. The fissile content is about half that of an LWR. This 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 pressurized water reactor. However, the power density in the salt with the dissolved uranium is higher. Because most of the heat is generated in the molten salt with the uranium, heat transfer does not limit fuel power density. A large inventory of fissile material in many fuel elements to provide a large heat-transfer surface is not a requirement. It is unclear what the practical lower limits of the fissile inventory of a molten salt reactor are.

4.2.3 Fuel Isotopics

The previous ^{233}U inventory calculation does not account for fuel isotopics—this has a major impact on the allowable ^{233}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). With the low fissile inventory, the fuel sees a higher effective neutron flux than in other thermal reactors. After one year in a MSR, the relative concentrations of different uranium and plutonium isotopes will begin to resemble LWR fuel that has been irradiated for several years. After five years of operation, the relative concentrations of MSR isotopes will be different from anything previously seen in any other type of power reactor.

Under these conditions, there is a significant buildup of ^{234}U and ^{236}U in the fuel. These isotopes, like ^{238}U, isotopically dilute the ^{233}U and reduce its usability in a nuclear weapon. This isotopic dilution reduces the quantity of ^{238}U required to convert ^{233}U to non-weapons-usable ^{233}U and consequently reduces the quantities of plutonium that are produced. Assessments have not yet been made on how much the ^{238}U inventory of the reactor may be reduced in the presence of other uranium isotopes while maintaining the ^{233}U as non-weapons usable. Any reduction in ^{238}U directly reduces the plutonium inventory of the reactor.

Molten salt reactor studies have determined plutonium isotopics when the plutonium is in an infinite recycle loop (Engel 1978, Benedict 1957). Under such conditions, ^{242}Pu becomes the dominant plutonium isotope, and ^{239}Pu becomes a minor plutonium isotope. Not all ^{239}Pu fissions. After neutron absorption, a small fraction of the ^{239}Pu is converted to ^{240}Pu. Further neutron irradiation converts the ^{240}Pu to ^{241}Pu. With neutron irradiation, most of the ^{241}Pu is fissioned, but a small fraction is converted to ^{242}Pu. Plutonium-242 has a low neutron absorption cross section and therefore 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 that was 80+% ^{239}Pu. No one was considering a reactor in which ^{239}Pu is a minor plutonium isotope. The critical mass of ^{242}Pu is about an order of magnitude greater than that for ^{239}Pu. *Different nuclear isotopes have different nuclear properties.* 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}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}Pu.

The implication of these considerations is that the allowable quantity of ^{233}U for a 1,000-MW(e) PR-MSR may be some multiple of 500 kg because more than (a) 8 kg of plutonium with ^{242}Pu and (b) >12% ^{233}U in ^{238}U in some mixture of other uranium isotopes is allowable (if the goal is to limit the plutonium inventory to less than that in one nuclear weapon) while still meeting nonproliferation goals. This is for a reactor system that typically operates on 1,500 kg of fissile material.

4.2.4 Minor Fissile Isotopes

The quantities of minor fissile isotopes are very small. Neptunium production is low because the fuel is ^{233}U and four neutron absorptions are required to reach ^{237}Np. Americium and curium inventories are low because the reactor is designed to minimize plutonium isotopes, which, in turn, minimizes the production of the higher actinides. Furthermore, americium and curium have chemical behaviors similar to the rare earths and thus will tend to migrate to the rare earth waste stream (See Sect. 4.4).

4.2.5 Chemical Separations

Fission products accumulate in the fuel salt and must be separated to (1) avoid excessive parasitic neutron capture—assure the CR is greater than 1, (2) avoid increasing the volume of salt beyond the capacity of the reactor, and (3) 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.

Molten salt reactors separate fission products from the fuel. This is opposite of conventional reprocessing fuel cycles where the SNF is processed for recovery of plutonium and recycle of the plutonium back to the reactor. The MSR approach is highly desirable in terms of proliferation resistance since it avoids the existence of purified fissile materials. However, the historical rational for this different approach is based on economics, not proliferation resistance. There are two reasons for this approach.

- *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 a MSR, the desired products are: expensive isotopically-separated ^{7}Li, the uranium, and the plutonium. The beryllium, the thorium, and the fluoride are expensive to treat as wastes; consequently, it is highly 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 99.9 % of the salt components from the fission products.

- *Fuel fabrication.* 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 is necessary to reduce the radiation levels to allow economic fabrication of complex, highly-engineered, precision fuel elements. A MSR with a liquid fuel has no fuel fabrication. Fission product removal is dictated by the 3 factors described above. In most cases, reducing fission product concentrations by a factor of 10 in the salt more than meets the requirements. This reduces costs but implies that the "clean" salt is highly radioactive.

The most important fission product removal operation is 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. Many alternative flowsheets were developed and tested.

4.2.6 Protactinium Management

Protactinium management is a unique characteristic of molten salt reactors using a ^{233}U-Th fuel cycle. When ^{232}Th absorbs a neutron, it is converted to ^{233}Pa, which then decays to ^{233}U. Protactinium-233 has a 27-d half-life. If the ^{233}Pa absorbs a neutron, it will no longer decay to fissile ^{233}U. If ^{233}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}Pa loses:

- *Limit reactor-core power density.* As the reactor core power density is lowered, neutron absorption by ^{233}Pa is reduced with more production of ^{233}U. However, this reduction in power density implies a larger reactor core, a larger ^{233}U inventory in the reactor, and consequently a larger plutonium inventory in the reactor. This requires careful trade-offs in core design to assure a breeder reactor that meets nonproliferation goals.

- *Separate ^{233}Pa.* The ^{233}Pa can be separated from the fuel salt and allowed to decay to ^{233}U outside the reactor. The resultant ^{233}U can then be added back to the reactor. This process reduces losses of ^{233}Pa by neutron absorption in the reactor core and maximizes ^{233}U production. It improves fuel economy and increases the CR. However, if the ^{233}Pa were completely separated from the isotopically diluted ^{233}U, its decay would produce ^{233}U, which was not isotopically diluted with ^{238}U—a potential proliferation risk. In theory, excess fuel salt with ^{233}Pa, ^{233}U, and ^{238}U could be stored outside the reactor until the ^{233}Pa decayed; however, the inventory of salt and ^{233}U would be cost prohibitive. If ^{233}Pa is to be stored outside the reactor until it is decayed, it must be separated from most 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}Pa from the ^{233}U/^{238}U in the fuel salt *if isotopic dilution is to be used as the only acceptable barrier to prevent access to weapons-usable materials*. This is not an issue with offsite processing of the salt because the ^{233}Pa rapidly decays to ^{233}U in the presence of ^{238}U. By the time off-site processing could be done, there would be no significant quantity of ^{233}Pa and would be diluted by the ^{238}U that is already present. 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}Pa and subsequent decay of the ^{233}Pa to ^{233}U.

There are incentives to separate the ^{233}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.). However, if a ^{233}Pa separation system is installed, it must be assured that it is not a significant proliferation risk. *The choice of ^{233}Pa management strategy (no ^{233}Pa separation or ^{233}Pa separation) is a major design decision for a PR-MSR and a significant uncertainty.* There are several enabling characteristics of this system that may make ^{233}Pa separation from the fuel salt feasible while maintaining high proliferation resistance.

- *Limited possible ^{233}U production.* For the reactor to operate, the ^{233}U from decay of ^{233}Pa must be recycled. It is the fuel. Only limited amounts of ^{233}U (from ^{233}Pa) can be removed before the reactor shuts down because of a lack of fuel and ^{233}Pa production is stopped. For the nation state, the choice is electric power or a small inventory of ^{233}U.

- *Hot ^{233}U.* Any chemical separation process for protactinium separates all protactinium isotopes equally—including ^{232}Pa, which decays to ^{232}U. Unlike ^{233}U, ^{232}U has a decay product that emits a 2.6-MeV gamma-ray. The choice of uranium-thorium feeds, salt cleanup systems, and the reactor design determines the ^{232}U content. It may be possible to modify the reactor so that the secular equilibrium radiation levels from the ^{232}U and decay products exceed the IAEA definition of SNF—fuel with radiation levels exceeding 100 R/h at 1 m. Because of variable radiation fields with time and other factors, quantifying proliferation resistance with this approach is complex.

- *Inefficient ^{233}Pa separation from isotopically diluted ^{233}U.* The quantity of ^{233}Pa in the reactor is very small compared to the quantity of ^{238}U in the reactor. If weapons-usable ^{233}U is to be produced by separation of the ^{233}Pa from the uranium in the fuel salt, the separation process to separate the ^{233}Pa from the uranium in the fuel salt must be efficient. If a small fraction of a percent of the ^{233}U/^{238}U inventory is not separated from the ^{233}Pa, that ^{233}U-^{238}U mixture will isotopically dilute the new ^{233}U from decay of ^{233}Pa to non-weapons-usable ^{233}U. The separation must also be done quickly or the ^{233}Pa decays to ^{233}U in the presence of ^{238}U in the fuel salt.

  A proliferation-resistant system that allows separation of ^{233}Pa from the fuel salt is any technology that, because of thermodynamic equilibrium limits (azeotrope or eutectic formation, etc.) or other mechanisms, can not efficiently separate all of the ^{233}U and ^{238}U from the ^{233}Pa. If some ^{233}U or ^{238}U remains with the ^{233}Pa, the ^{233}U from its decay will be isotopically diluted to non-weapons-usable ^{233}U outside the reactor. There are several possible technologies with these characteristics (See Sect. 4.3).

Protactinium management is secondary to reducing plutonium inventories. The total ^{233}Pa inventory is a several tens of kg dissolved in >100,000 kg of highly-radioactive molten salt. Because the ^{233}Pa decays rapidly to ^{233}U, any nation that wants to separate the ^{233}Pa has to build a chemical separations plant next to the reactor. Uranium-233 output will be limited before reactor shutdown because there is insufficient fuel and because the separation facility can not be hidden since it must be close to the reactor. Any international on-line monitoring of the fuel salt composition would indicate removal of ^{233}Pa from the reactor salt. In contrast, plutonium remains weapons usable for tens of thousands of years; thus, the just-in-time separations requirements for ^{233}Pa do not exist for plutonium. Off-site processing of feeds for plutonium recovery is viable with the option to build a plutonium separations plant and test it before diverting feed materials and making the facility's existence known.

An appropriate perspective on protactinium management is required. It is, by default, the area of proliferation concern that receives attention with molten salt reactors. This is because all of the much larger and potentially more accessible sources of weapons-usable materials (SNF, large inventories of plutonium in the reactor core, and enrichment plants) have been eliminated. As a point of comparison, it is noted that (1) a typical LWR has a plutonium inventory of ~750 kg, (2) a typical LWR discharges several hundred kilograms of plutonium in 30,000 kg of SNF per year, (3) the plutonium does not rapidly decay away—off-site processing is clearly viable, and (4) chemically separated, clean plutonium radiation levels are relatively low and acceptable for weapons purposes.

4.3 ALTERNATIVE FUEL CYCLES

The basic fuel cycle is simple. After startup on low enriched uranium, the reactor would be fed (1) either natural uranium or depleted uranium and (2) thorium. Only fission products are removed from the molten salt and leave the reactor as waste. Inside the reactor, fissile materials are fissioned and fertile materials are converted to fissile materials that fission. Previous studies have examined this system as it approaches equilibrium in several decades (Engel 1978).

The characteristics of this reactor create several alternative fuel cycle options. A reactor that minimizes the inventory of weapons-usable materials has the capability for rapid destruction of plutonium by (1) rapid burnout of plutonium, (2) a very low rate of plutonium generation compared to other reactors, and (3) rapid conversion of the remaining plutonium to isotopic compositions that are very undesirable for constructing a nuclear weapon. If it is used in this alternative fuel-cycle mode, the inventory of weapons-usable materials in the reactor will be significant. There are two sets of plutonium fuel cycle options.

- *Burn plutonium*. The PR-MSR could be used to rapidly convert weapons-usable plutonium into a mixture of nonweapons usable ^{233}U and ^{238}U. In this mode, the reactor fuel contains plutonium, thorium, and only small concentrations of ^{233}U with ^{238}U. When the concentration of uranium is kept very low, little of the ^{233}U is fissioned, and very little ^{239}Pu is produced from ^{238}U; consequently, almost all fissions involve the fissioning of plutonium. Excess neutrons are absorbed in ^{232}Th to produce added ^{233}U. The excess $^{233}U{:}^{238}U$ mixture is removed continuously. The non-weapons-usable ^{233}U-^{238}U mixtures could be used to startup other PR-MSRs or stored if the growth rate for new reactors did not match ^{233}U-^{238}U production rates. The reactor becomes an efficient method to convert weapons-usable fissile plutonium to non-weapons-usable fissile ^{233}U mixed in ^{238}U.

- *Burn 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 mixture (uranium, plutonium, fission products, and other actinides) can be directly converted to a fluoride salt. The existing technologies provide viable methods to remove most of the uranium from the salt but do not remove the plutonium—the plutonium remains with the fission products. This form of processing reduces proliferation risks, as compared to conventional reprocessing technologies. This capability creates two options.

  - *Initial fueling*. The PR-MSR could be initially fueled with a plutonium-fission product salt rather than low-enriched uranium. Some of the uranium in the initial SNF has to be removed to obtain an acceptable starting fuel. The plutonium will burn out over time. Because of the small fissile inventory of this type of reactor, the expected SNF inventory from existing reactors would be sufficient to start up several thousand PR-MSRs.

► *Continuous fueling.* The PR-MSR requires addition of ^{238}U for proliferation resistance. The fluoride salt from the direct conversion of SNF can be used to makeup the ^{238}U requirements. Light-water reactor SNF is primarily ^{238}U. The plutonium and ^{235}U burn out over time.

The decision of whether it is desirable to burn plutonium and SNF is fundamentally a political decision based on non-proliferation considerations. Because the PR-MSR has a CR >1 and thus requires only small quantities of relatively inexpensive natural uranium and thorium, there is not a significant economic incentive to use plutonium or SNF as a fuel source. The decision involves trade-offs between risks of storing plutonium essentially forever in repositories or monitored storage areas vs placing it into reactors and destroying it.

4.4 WASTE MANAGEMENT AND WASTE THRESHOLDS

In any power reactor, the waste system is where the fissile materials become available for diversion to weapons. In a conventional LWR, the waste is SNF. In a PR-MSR, the waste is a complex mixture of fission products separated from the molten salt. As with any chemical separation process, separations are not perfect and there will be traces of fissile materials in the waste streams. In particular, two fissile materials, americium and curium, have chemical properties similar to the rare earths and thus tend to follow the rare earths to the final waste forms. The loss rate for fissile materials and the waste form determines the long-term proliferation risk.

The IAEA has defined waste thresholds (Forsberg July 1998). 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, the fissile material is practically unrecoverable and is not weapons-usable. It would be easier for the nation state to make new fissile material. 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 proliferation resistant reactor concept, the waste system must be designed to reduce the concentrations of fissile materials in the final waste form below these IAEA defined waste thresholds. 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.

4.5. SAFETY

The historical approach to improve proliferation resistance of nuclear power systems has been to use radiation barriers to prevent diversion of weapons-usable fissile materials from power reactors to weapons programs. This is the primary basis for the U.S. policy against reprocessing of SNF. Reprocessing removes the radiation barrier from weapons-usable fissile materials. Almost all proposals for more proliferation-resistant reactor fuel cycles involve increasing the fuel burnup and thus the radiation field associated with the SNF. Unfortunately, this strategy increases the inventory of fission products in the reactor core and thus increases the consequences if a severe accident occurs. The fission product inventory of the reactor should be lowered for safety. There is an implied tradeoff between this approach to proliferation resistance and safety.

The use of isotopic dilution, not radiation, as the primary proliferation barrier breaks the tradeoff between safety and proliferation resistance. The PR-MSR is a liquid-fueled reactor where fission products can be removed on-line to minimize the accident source term. In the specific example of the PR-MSR, fission product removal is a requirement to assure a CR >1 and thus minimization of plutonium production. The accident source term may be a small fraction of that of a conventional power reactor.

5. OTHER REACTOR CONCEPTS

The example above was of a PR-MSR because our limited knowledge indicates that this concept may offer the highest degree of proliferation resistance of any nuclear power system. This assume that the best approach to proliferation resistance is to minimize inventories of weapons-usable materials. The basic approach (^{233}U-^{238}U-thorium fuel cycle, thermal neutrons, modification of a reactor to minimize plutonium production, etc.) could be applied to other reactors. It is unknown if there are other reactor concepts that could match or exceed the PR-MSR in terms of proliferation resistance. This approach to proliferation resistance has not been studied in detail and is not well understood. Earlier thorium fuel cycle studies did not consider the impact of unusual isotopics on proliferation resistance or several other factors. In part this reflects the earlier emphasis on uranium-plutonium fuel cycles and in part this reflects our better understanding today of proliferation issues.

6. BREAKING THE POWER REACTOR—WEAPONS CONNECTION

Can we build nuclear power systems that disconnect nuclear power from nuclear proliferation? The definitive answer does not yet exist.

Theft of nuclear materials by subnational groups from a reactor such as PR-MSR is not credible. The initial fuel for the PR-MSR is non-weapons-usable LEU or non-weapons-usable ^{233}U mixed with ^{238}U. There is no SNF. The total inventory of potentially usable weapons materials in the reactor (plutonium isotopes and ^{233}Pa that decays over time to weapons-usable ^{233}U—if not in the presence of ^{238}U) is measured in a few tens of kg of material dissolved in >100,000 kg of highly radioactive salt. Most of the potentially usable weapons-usable inventory is in the form of ^{233}Pa that rapidly decays to ^{233}U that, in turn, is isotopically diluted by the ^{238}U in the molten salt to non-weapons-usable ^{233}U. To recover any weapons-usable ^{233}U, the chemical recovery process for ^{233}Pa from the salt must be fast and efficient. Building, testing, and operating a separations plant for extended periods of time at an operating reactor is not credible for a subnational group. The ^{233}U from the decay of ^{233}Pa will be highly radioactive because it contains ^{232}U and the ^{232}U decay products that emit 2.6-MeV gamma rays, which are easy to detect. Only very-small quantities of plutonium are in the reactor core and that plutonium has isotopics (^{242}Pu) that present serious challenges to any weapons designer.

Diversion of nuclear materials by nation states is possible—but may not attractive. Diversion of ^{233}Pa or plutonium shuts down the reactor because of a lack of fuel. The quantities of weapons-usable fissile materials are small, and any mistake that causes delay causes a rapid reduction of the available inventory (^{233}Pa decays away quickly). The isotopic quality of weapons-usable fissile materials from a PR-MSR are very poor—a major challenge to an experienced weapons designer and maybe beyond a first-time weapons designer. Last, any diversion becomes obvious. It is unclear whether there are credible circumstances under which such diversion would be undertaken.

For a nation, there are many options for acquisition of weapons-usable materials. A nation that buys 1,000-MW(e) reactors has significant resources. Almost all of the other options to produce weapons-usable fissile materials are easier to hide and more likely to succeed. Some of the technologies, such as centrifuge isotopic enrichment are becoming widely available because the underlying technologies are widely used in other industries (gas turbines and high-strength composites for aircraft and sporting goods). Other technologies, such as calutrons (Benedict 1957), have been available for decades.

7. CONCLUSIONS

The historical "belief" is that nuclear power and nuclear weapons proliferation are connected. Radiation in SNF has been considered the primary barrier against recovery of chemically separable, weapons-usable fissile materials from power reactors. However, there are other approaches—one is the reduction in the inventory of weapons-usable fissile materials. It may be possible to reduce this inventory and change the characteristics of nuclear power so that the connection with nuclear weapons is not significant when addressing the potential for a nation state to build nuclear weapons. Such an approach makes acquisition of weapons-usable materials by subnational groups not credible.

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