FISSION FUSION THORIUM HYBRID

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INTRODUCTION

The thorium fission fusion hybrid is discussed as a sustainable longer term and larger resource base to the fast breeder fission reactor concept and as an early introduction of fusion energy. In addition, it offers a manageable waste disposal process, burning of the produced actinides and serious nonproliferation characteristics. A first generation thorium hybrid would use the DT fusion reaction as a neutron source instead of the fissile isotopes U^{235} and/or Pu^{239} as external seed material feeds, breeding tritium from lithium. A second generation would use the catalyzed DD fusion reaction eliminating the need for tritium breeding and providing a practically unlimited supply of the deuterium at 150 ppm in water from the world oceans. Either system could use the U^{233} bred from Th^{232} for power generation on-site or for export to satellite smaller-size fission reactors.

The nuclear performance of a fusion-fission hybrid reactor having a molten salt composed of Na-Th-F-Be as the blanket fertile material and operating with a catalyzed Deuterium-Deuterium (DD) plasma is compared to a system with a Li-Th-F-Be salt operating with a Deuterium-Tritium (DT) plasma. In a reactor with a 42-cm thick salt blanket followed by a 40-cm thick graphite reflector, the catalyzed DD system exhibits a fissile nuclide production rate of 0.88 Th(n, γ) reactions per fusion source neutron. The DT system, in addition to breeding tritium from lithium for the DT reaction yields 0.74 Th(n, γ) breeding reactions per fusion source neutron. Both approaches provide substantial energy amplification through the fusion-fission coupling process.

The catalyzed DD fusion approach to U^{233} breeding from Th^{232} offers distinct advantages. It eliminates the need for breeding tritium from lithium in the blanket. The requirement of tritium storage and its environmental leakage hazard are eliminated. Any tritium produced in the plasma of the catalyzed DD cycle can be directly injected into the plasma, so the active tritium inventory in the plasma loop is reduced by a factor of 3 compared with the DT fusion cycle. In addition, the competition for the available neutrons to compete for fusile and fissile breeding as occurs in the DT system is eliminated, leading to higher fissile breeding in the catalyzed DD cycle. Continuous extraction of the bred U^{233} and its Pa^{233} precursor from a molten salt would lead to a blanket relatively clean from fission products contamination, neutron poisoning and power swings caused by fissioning of the produced U^{233} as would occur in a solid blanket.
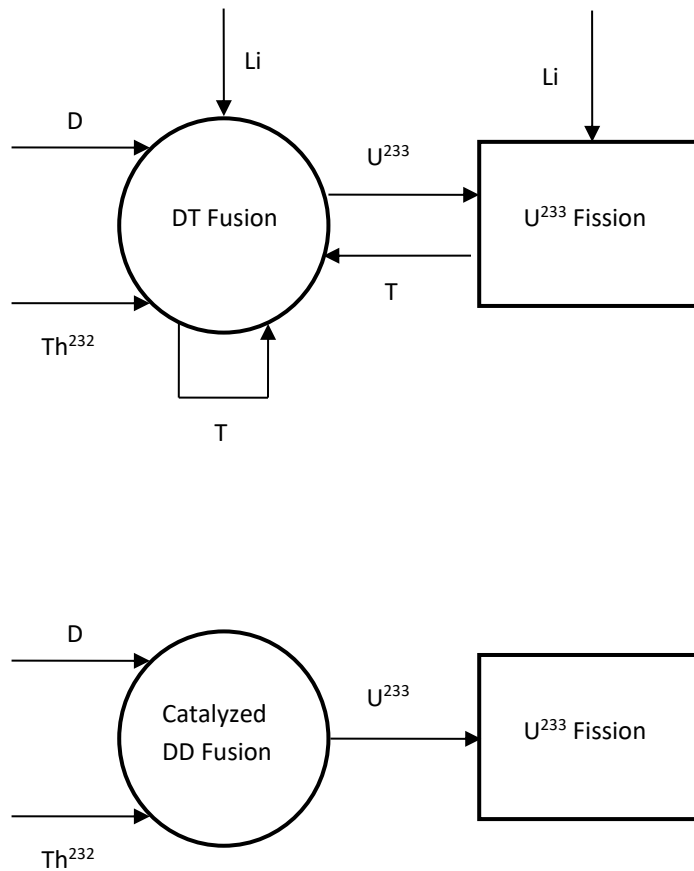
Figure 1. Material flows in the DT (top) and Catalyzed DD fusion-fission hybrid (bottom) alternatives with U^{233} breeding from Th^{232}.

Figure 2. Thorium dioxide ore is more common than uranium ores in the Earth's crust (left). ThO_2 incandescent gas lantern mantle (right).

HISTORICAL JUSTIFICATION

The thorium fission fuel cycle was investigated over the period 1950-1976 in the Molten Salt Breeder Reactor (MSBR) at the Oak Ridge National Laboratory (ORNL) as well as in the pilot Shippingport fission reactor plant, which was a replica of a naval reactor design, and operated for 5 years on a single load of fuel and ended the run with more bred fissile fuel U^{233} fuel than the fissile U^{235} feed that it was started with.

The existing U^{238}-U^{235} fuel cycle has been favored to the Th^{232}-U^{233} for historical reasons. The early reactors needed to be powered using the U^{235} isotope that occurs in natural uranium to produce Pu^{239} from U^{238}. Thorium could not have been used since it is fissionable but not fissile to build a self sustained chain reaction.

In a natural-uranium fuelled reactor using heavy water D_2O as a coolant/moderator, about 1 gram per day of Pu^{239} is produced by neutron capture in U^{238} per MW of thermal power at low U^{235} burnup. Production of the fissile U^{233} isotope requires the irradiation of the fertile Th^{232} isotope in a neutron spectrum. The thermal neutron absorption cross section of Th^{232} is 3 times that of U^{238}, and at the maximum 7 percent ThO_2 needed to maintain criticality in a Heavy Water Reactor (HWR), only 0.2 gm/day per MW of U^{233} can be produced at low fuel burnup rates below 1,000 thermal megawatts per ton of heavy U metal (MWd / tU). Thus most of the fissile breeding in a HWR would be as Pu^{239} and not as U^{233}.

With the availability of uranium enrichment capabilities, the fissile production can be totally shifted towards U^{233} production with highly enriched U^{235} fuel. In fact, the USA produced most of its fissile Pu^{239} as well as fusile tritium in its heavy water reactors at Savannah River using highly enriched naval reactors fuel as a driver neutron source and depleted uranium metal and a LiAl alloy as target elements.

With the present day availability of fissile U^{235} and Pu^{239}, and available fusion and accelerator neutron sources, a new look at the thorium cycle is warranted. Since no more than 7 percent of ThO_2 fuel can be added to a HWR before criticality would not be achievable, this suggests that fusion and accelerator sources are the appropriate alternative for the implementation of the Th fuel cycle.

India, because of its large thorium resources but limited uranium resources has adopted a three stages plan to implement the thorium fuel cycle. The first stage involves the use of HWRs fueled by natural uranium and Light Water Reactors, LWRs fueled by Low Enriched Uranium (LEU). In the second stage, the plutonium extracted from the spent fuel of these reactors would be used as a startup fuel for liquid sodium cooled fast reactors. In the third stage, the U^{233} produced by neutron capture in Th^{232} blankets in the fast breeder reactors would be mixed with thorium and used to start up HWRs and High temperature Gas Cooled Reactors (HTGRs) using a closed U^{233}-Th^{232} fuel cycle for the long term.

India is developing the Advanced Heavy water Reactor (AHWR) using low enrichment uranium (LEU) fuel at 19.75 percent U^{235} for export at the 300 MWe power level; a third of the

capacity of a typical 1,000 MWe unit. It is designed to operate for 100 years, is manageable with modest industrial infrastructure in developing countries and has a level of safety providing the operators a three days period for recovery from a potential accident. The spent fuel would contain a large proportion of Pu^{238} making it unusable for weapons manufacture and half the amount of the long lived actinides per unit of energy produced as typical LWRs. The first AHWR is meant to start construction in 2012. A prototype 500 MWe fast breeder reactor is projected for completion in 2011 at Kalapakkam.

Whereas the U^{233}-Th^{232} fuel cycle is undergoing a grassroots scientists and engineers as well as members of the public revival as a replacement of the existing LWRs system, there exists a highly promising approach in its use in fusion-fission hybrid reactors as an eventual bridge and technology development for future pure fusion reactors, bypassing the intermediate stage of the fast fission breeder reactors.

It may be a suitable time to leap to the proposed approach to take advantage of the Th cycle benefits in the form of a well optimized fission-fusion thorium hybrid. Such an alternative sustainable paradigm or architecture would provide the possibility of long term fuel availability with the added advantages of higher temperatures thermal efficiency for process heat production, the possibility of dry cooling in arid areas of the world, proliferation resistance and minimal waste disposal characteristics.

BOILING WATER REACTOR EXPERIMENT IV, BORAX-4

The 1958 BORAX-IV reactor, with a power level of 20 MWth, tested fuel elements made from mixed oxides (ceramics) of uranium and thorium. These materials had a high capacity to operate in the extreme heat of a reactor before they failed.

The ceramic core demonstrated that a reactor loaded with this fuel could operate safely and feasibly. The fuel could operate in higher temperatures, was less reactive with the water coolant in case the cladding ruptured, was cheaper to manufacture, and burned a larger percentage of the fuel before losing its reactivity.

The reactor produced measurable quantities of the artificial thorium-derived fuel, U^{233}. One series of BORAX-IV tests involved operating the reactor with experimentally defective fuel elements in the core.

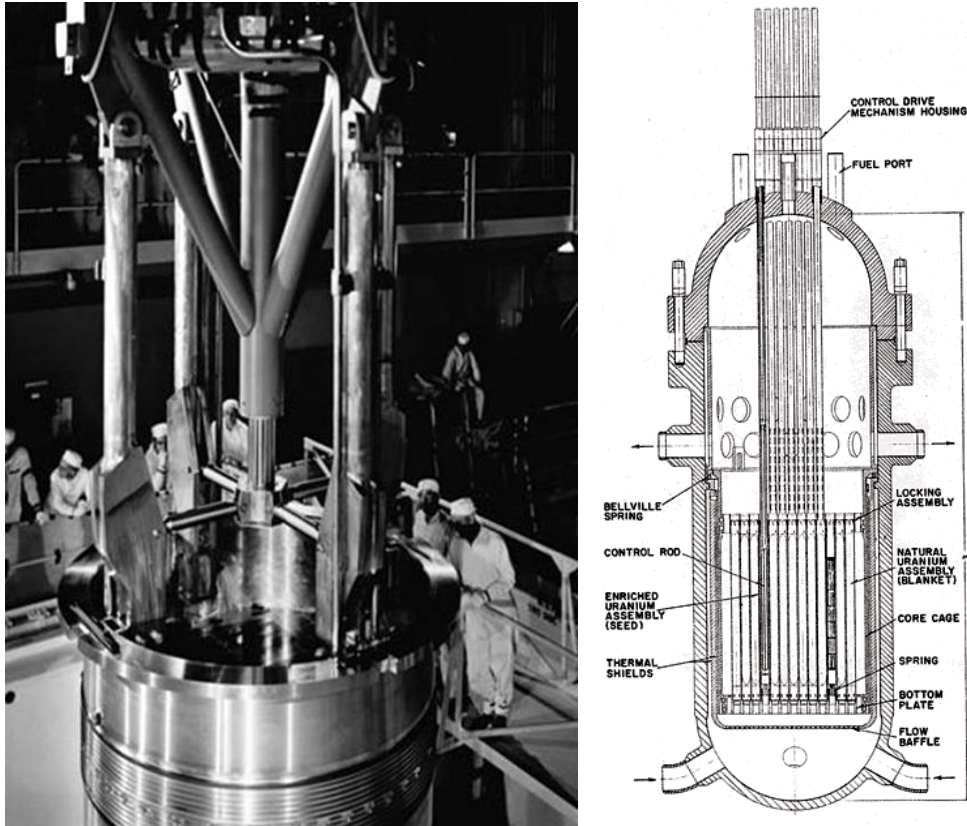## SHIPPINGPORT PRESSURIZED WATER REACTOR AND LIGHT WATER BREEDER REACTOR

Figure 3. The Shippingport reactor was the first commercial and experimental nuclear power plant in the USA and second in the world after Calder Hall in the UK.

The Shippingport power station first operated in December 1957 and was the first USA's commercial nuclear power reactor operated by the Duquesne Light Company. The UK's Calder Hall dual purpose pressurized CO_2 cooled pile producing electrical power and plutonium plant at Seascale, Cumbria, went online on August 27, 1956, more than a year before Shippingport, earning the honor of being the world's first large-scale commercial nuclear power plant.

The Shippingport plant was a Pressurized Water Reactor (PWR) with the first two reactor cores as "seed and blanket" cores. The seed assemblies had highly enriched uranium plate fuel clad in zirconium, similar to submarine cores, and the blanket assemblies had natural uranium.

Using low-enriched uranium LEU oxide for fuel instead of the highly enriched uranium HEU employed in naval reactors was adopted in order to prevent diversion of the nuclear fuel elements for use in nuclear weapons, severely affecting the size of the core and hence increasing the capital cost.

The first core, PWR-1, had 32 seed assemblies with each seed assembly including four subassemblies for a total of 128. Each subassembly contained 15 fuel elements for a total of 1920. The U^{235} loading for the first seed core 75 kgs and the subsequent seeds had 90 kgs loadings.

In a subsequent breeding core, small ceramic fuel pellets inside the fuel rods were ¼ inch in diameter and 1/2 inch long made of U^{233} and Th^{232}. This seed provided the neutrons for the chain reaction while the outer part of the fuel module was a blanket of fuel rods containing

pellets consisting of a slightly different composition of the isotope U^{233} that provided the excess neutrons required to breed new fuel and Th^{232}.

The seed and the blanket were surrounded by a reflector to reduce neutron losses by reflecting neutrons back into the core to enhance breeding of new fuel. Instead of using neutron-absorbing hafnium control rods used in the earlier reactor core, the chain reaction in the breeder reactor had a better neutron economy and was controlled by moving the seed to a position level with the blanket to increase the power and withdrawing the seed from the blanket to slow or stop the chain reaction.
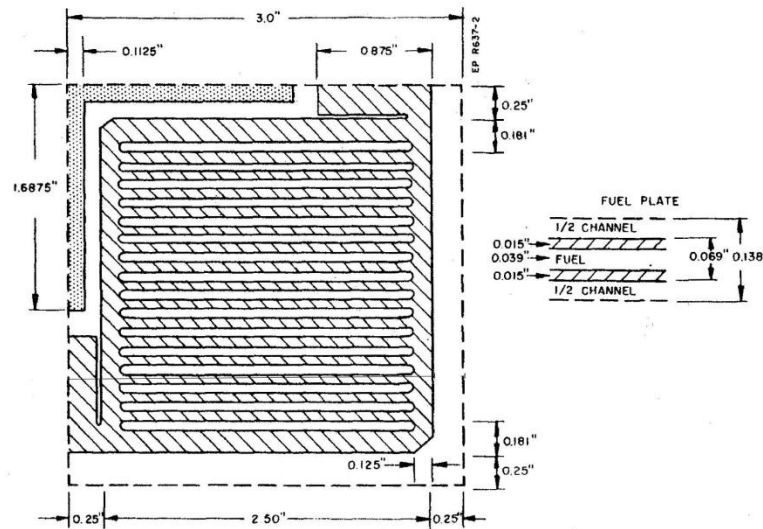

Figure 4. Shippingport Reactor PWR-1 seed subassembly showing the highly enriched zirconium clad plate fuel and coolant channels. Dimensions in inches.
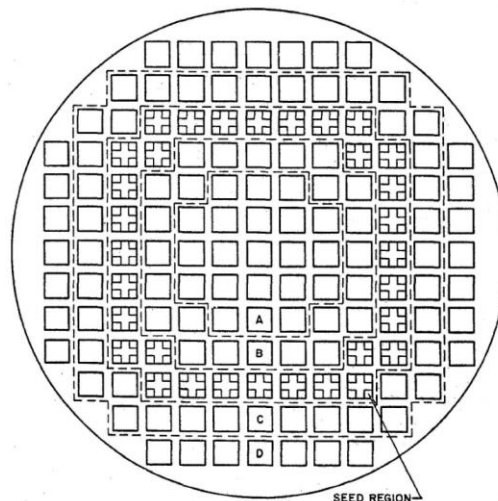

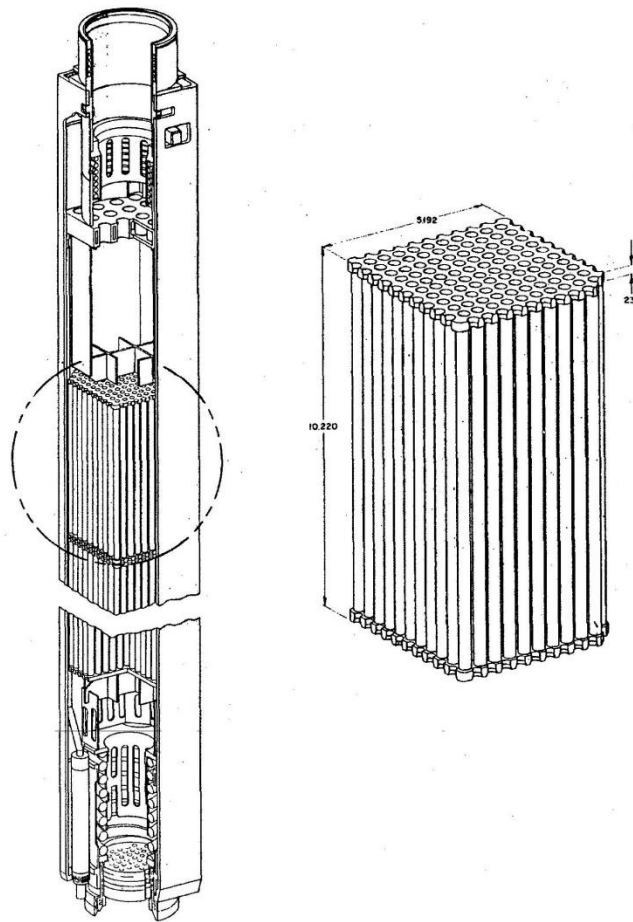Figure 5. Cross section of Shippinport PWR-1 core showing the seed region and the blanket regions A, B, C and D.

Figure 6. Shippingport reactor blanket fuel assembly.

The PWR-1 blanket fuel was made of natural uranium in the form of natural UO_2 pellets clad with Zircaloy tubes. Each blanket assembly was made from seven stacked fuel bundles. Each fuel bundle was an array of short Zircaloy tubes with natural uranium oxide pellets in the tubes. PWR-1 had 113 blanket assemblies each containing seven fuel bundles for a total of 791, and each bundle contained 120 short fuel rods for a total of 94,920. The natural uranium loading for the blanket fuel was 12,850 kgs of natural uranium.

Subsequently, the Shippingport blanket was replaced by a thorium control assembly to introduce the light water breeder concept where U^{233} is bred from Th^{232} in a thermal neutron spectrum. The core for the breeder reactor consisted of twelve hexagonal fuel modules that together were nine feet high, eight feet in diameter and weighed ninety tons. An inner movable seed attached to a control rod was operated by an external drive. This seed consisted of fuel rods made of Zircaloy-4 that were each 8.7 feet long and .3 inch in diameter.

Having satisfactorily proven the concept of thermal breeding of U^{233} from Th^{232}, in 1989, after more than 25 years in operation, the Shippingport plant became the USA's first successfully decommissioned commercial nuclear power plant, with the site released for public use.

HIGH TEMPERATURE GAS COOLED REACTOR HTGR, THERMAL THORIUM BREEDERS

High Temperature graphite moderated reactor designs have historically used thermal breeding within the Th^{232}-U^{233} fuel cycle. The smaller modular designs of HTGRs have been under development beginning with a design originating in Germany in 1979. The supporting HTGR technology has been under development with major programs in the UK, the USA and Germany from the 1950's through the early 1990's. Important milestones have been achieved in the design and successful operation of three steel vessel HTGRs during the 1960's and 1970's, and in the production and demonstration of robust, high quality fuel and other key elements of the technology.

The technology has developed along two distinct paths: pebble bed fuel consisting of ceramic spheres 6 cm in diameter with continuous refueling, and prismatic fuel consisting of hexagonal blocks approximately 35 cm across the flats and 75 cm in height with periodic batch refueling. Both fuel systems utilize ceramic coated microparticles of less than 1 millimeter in diameter.
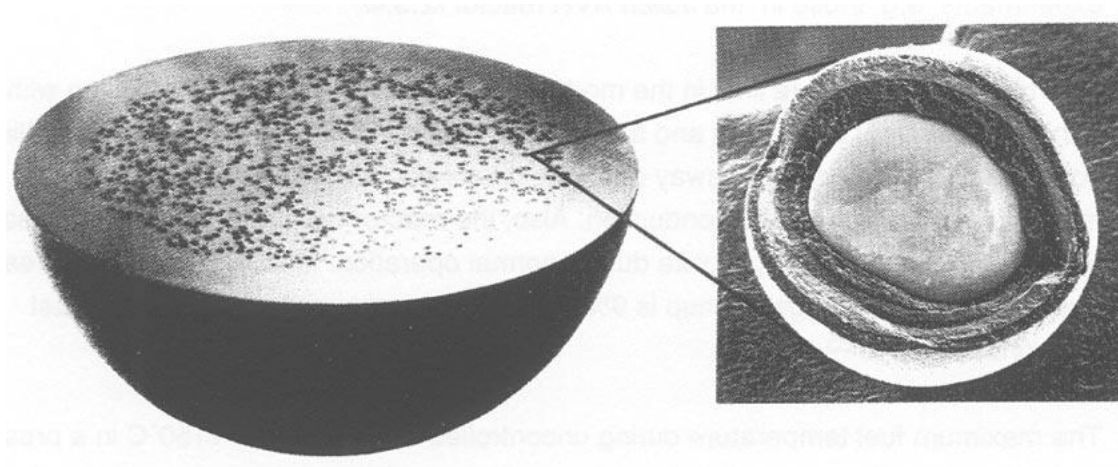

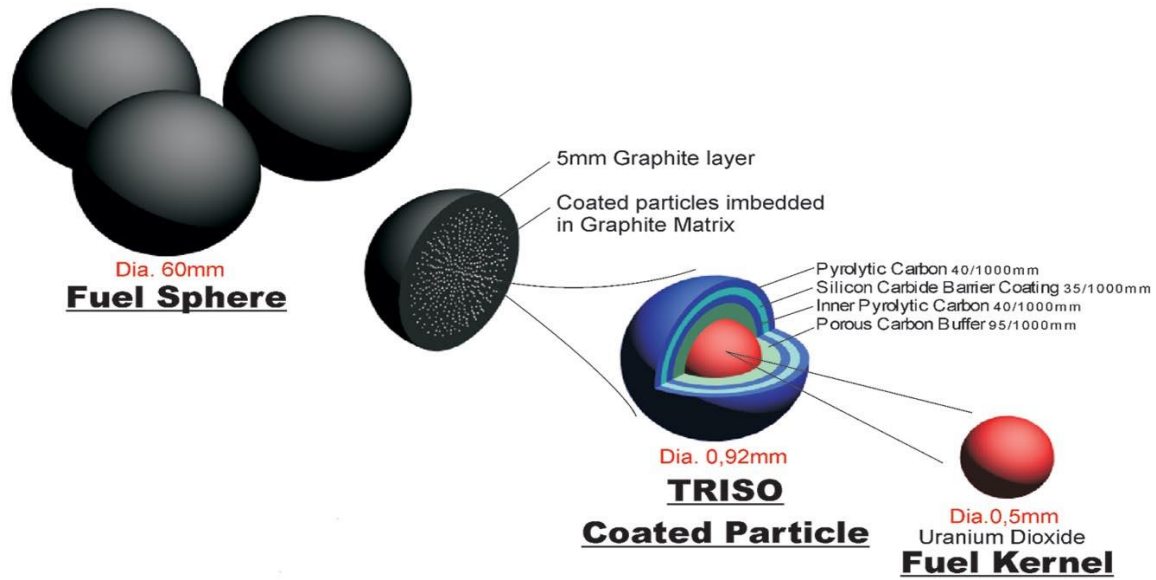Figure 7. Fuel particles used in the pebble bed and the prismatic fuel designs.

Figure 8. Fuel kernel, coated triso fuel particle and pebble fuel sphere design.
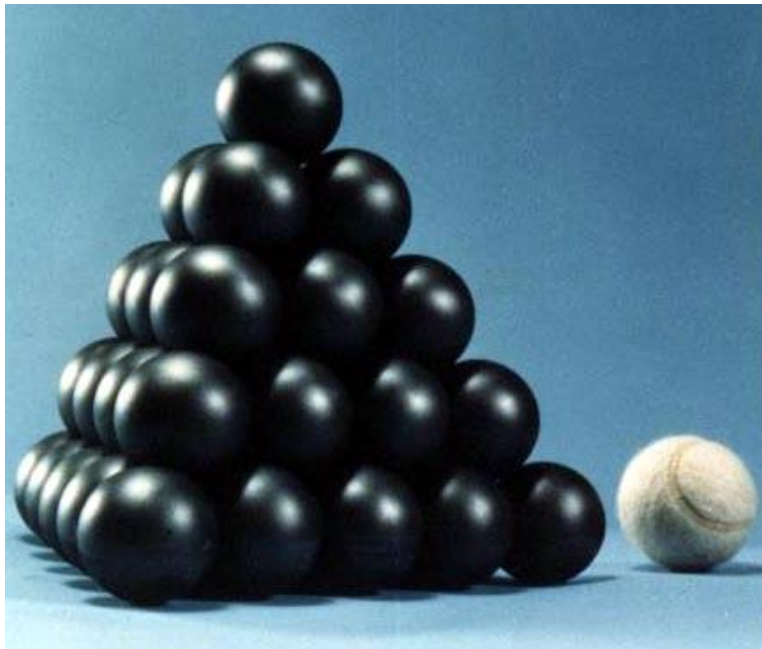

Figure 9. Graphite fuel pebbles used in the PBMR.

PEACH BOTTOM AND FORT SAINT VRAIN GAS COOLED THORIUM REACTORS

Figure 10. Fort Saint-Vrain He cooled graphite moderated reactor, Colorado.

Thorium as a breeder fuel has been used in the High Temperature Gas Cooled Reactor (HTGR) in a pebble bed and a prismatic moderator and fuel configurations.

The General Atomics (GA) Company built two thorium reactors over the 1960-1970 period. The first was a 40 MWe prototype at Peach Bottom, Pennsylvania operated by Philadelphia Electric. The second was the 330 MWe Fort St. Vrain reactor for Public service of Colorado which operated between 1971 and 1975.

The Fort St. Vrain reactor was a helium cooled reactor using water driven gas circulators with water bearings that leaked. Whenever they leaked, the plant had to be shut down and the coolant had to be dried. Circulators with magnetic bearings were developed as a remedy but were never used. Around 1974-1975, the oil embargo placed the utilities in financial distress and they cancelled the capital projects of 8 full size HTGRs that were on order with GA, which bought its way out of two more for fear that they would not pay for the engineering costs.

A problem with the HTGR design was its operation at low power density, implying a large reactor size and a high capital cost. Advantages were its high level of safety compared with conventional Light Water Reactors (LWRs) resulting from the high thermal inertia of its graphite core and operation at higher temperature, hence improved thermal efficiency.

THE GAS TURBINE MODULAR HELIUM REACTOR: GT-MHR

The Gas Turbine Modular Helium Reactor GT-MHR concept was based on an agreement between General Atomics in the USA and Russia. This concept pioneered the development of a next-generation modular nuclear reactor using the direct-cycle gas turbine, rather than the steam cycle. This approach operates at higher temperatures than light water reactors, increasing the

overall thermal efficiency of the plant by 50 percent. This in turn reduces the cost of producing electricity to the range of 3 cents per kilowatt-hour, compared with the cost of 5 cents per kilowatt-hour in existing designs.

The ability of handling higher temperatures is possible because of the use of ceramics such as silicon carbide. Helium as a coolant replaces water, which even though a good coolant, is corrosive. This limits the use of water to a temperature of 700 degrees C. Helium, in contrast, is inert and noncorrosive. It has no thermal limit, is in the gaseous form whether at room temperature or 3,000 degrees C, which is a tremendous advantage.
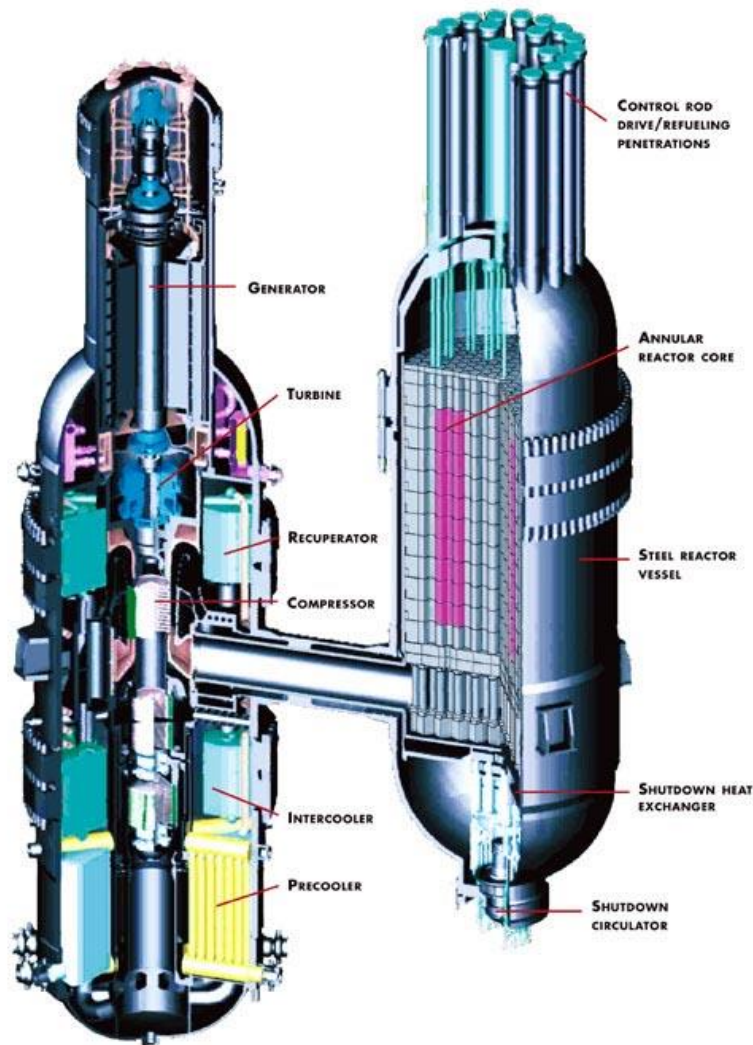
Figure 11. The entire Gas Turbine Modular Helium Reactor power plant is contained in two pressure vessels, enclosed into a concrete containment structure below ground.

Every time a coolant is passed through a heat exchanger before it can do its work, significant losses in efficiency occur. The turbine is directly driven by the helium gas as it comes out of the reactor core. Since no heat exchange is needed, this improves the overall efficiency of the plant.

A 300 MWe unit will be able to burn either uranium fuel, or fuel made from weapons grade plutonium, thus converting weapons grade material into electricity. If plutonium is used, up to 95 percent of it is used during power production. A mixture of uranium and thorium can also be used.

This kind of development project is also advocated as a way for western nations to turn around the economic depression and cultural pessimism that are eroding scientific and technological capabilities threatening their futures. In addition it would help the former nations of the Soviet Union and Third world countries out of their current economic devastation. It has been even advocated as a contribution for preventing atomic scientists in these countries from getting involved in nuclear weapons projects in proliferation prone countries. Along this line, the USA pledged $ 1.6 billion in terms of cleaning up problems in Russia's nuclear and weapons programs.

The new technology of magnetic bearings virtually eliminates friction. It has also great properties in the dynamic dampening of rotating shafts; a technology that did not exist a few years ago. The gas turbines builds on new turbine technology developed for jet engines such as the Boeing 747. High-strength high-temperature steel vessels are used.

New technology has also been become available in the recuperators from the fossil fuel power production field. Today's recuperators are five times smaller than the designs of a few years ago. Highly efficient plate-fin heat exchange equipment is used.

The design consists of two pressure vessels, both located underground in a concrete containment structure. The first vessel houses the reactor system. The second vessel houses the power conversion system including the gas turbine, two compressors, and a generator.

The helium gas in the core is heated to 1,562 degrees F. The heated helium flows to the power generator unit generating electricity at an overall thermal efficiency of 48 percent, compared with the value of 28-30 percent in water cooled reactors.

The reactor possesses passive inherently safe features in that it can shut itself down and cool itself down in the case of an emergency. Its cooling towers are one-sixth the size of those of conventional power plants, which reduces the overall cost. The cooling towers can be air-cooled rather than water cooled, which suggests that the plants can be located where water resources are scarce.

Compared with light water reactors, a reduction of 50 percent in the amounts of nuclear waste and thermal discharge is claimed.

The control and monitoring system are based using digital programmable equipment. Panels based on traditional equipment are used for controlling the safety systems.

Operation at high temperatures open the possibility for process heat applications such as the extraction and production of chemical products and mineral fertilizers, coal gas saturation, production of synthetic natural gas from coal, and ferrous and nonferrous metallurgy, as well as district heating.

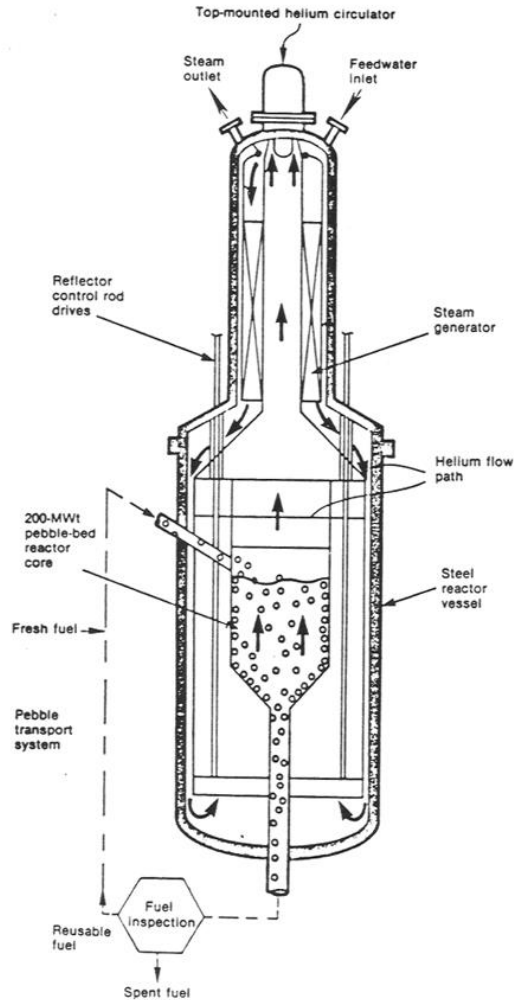## THE PEBBLE BED MODULAR REACTOR: PBMR

Figure 12. Conceptual Design of Pebble Bed Gas Cooled Reactor with a steam cycle rather than a gas cooled cycle.

The pebble Bed Modular Reactor, PBMR was pursued jointly by the Exelon corporation in the USA and South Africa's state owned utility Eskom. This concept is a high temperature helium cooled reactor with unit sizes of 110 MWe. These small sizes can be factory built before assembly at a site.

This design depends on fuel in the form of pebbles 6 cm in diameter. About 400,000 of these pebbles lie within a graphite lined vessel that is 20 m high and 20 m in diameter. Each pebble contains about 15,000 fuel particles where the fuel is enclosed in layers of pyrolitic graphite and silicon carbide.

The use of graphite as fuel element cladding, moderator, core structural material, and reflector, provides the reactor with a high degree of thermal inertia. A core melt situation would be practically unlikely, since a large difference exists between the normal average operating temperature of 1,095 degrees C, and the maximum tolerable fuel temperature of 2,800 degrees C.

Helium at a temperature of about 500 degrees C is pumped in at the top of the reactor, and withdrawn after sufficient burnup from the bottom of the reactor. The coolant gas extracts heat from the fuel pebbles at a temperature of 900 degrees C. The gas is diverted to three turbines. The first two turbines drive compressors, and the third drives an electrical generator from which electrical power is produced.

Upon exit from the compressors or generator, the gas is at 530 degrees C. It passes through recuperators where it loses excess energy and leaves at 140 degrees C. A water cooler takes it further to about 30 degrees C. The gas is then repressurized in a turbo-compressor. It then moves back to the regenerator heat-exchanger, where it picks up the residual energy before being fed to the reactor.
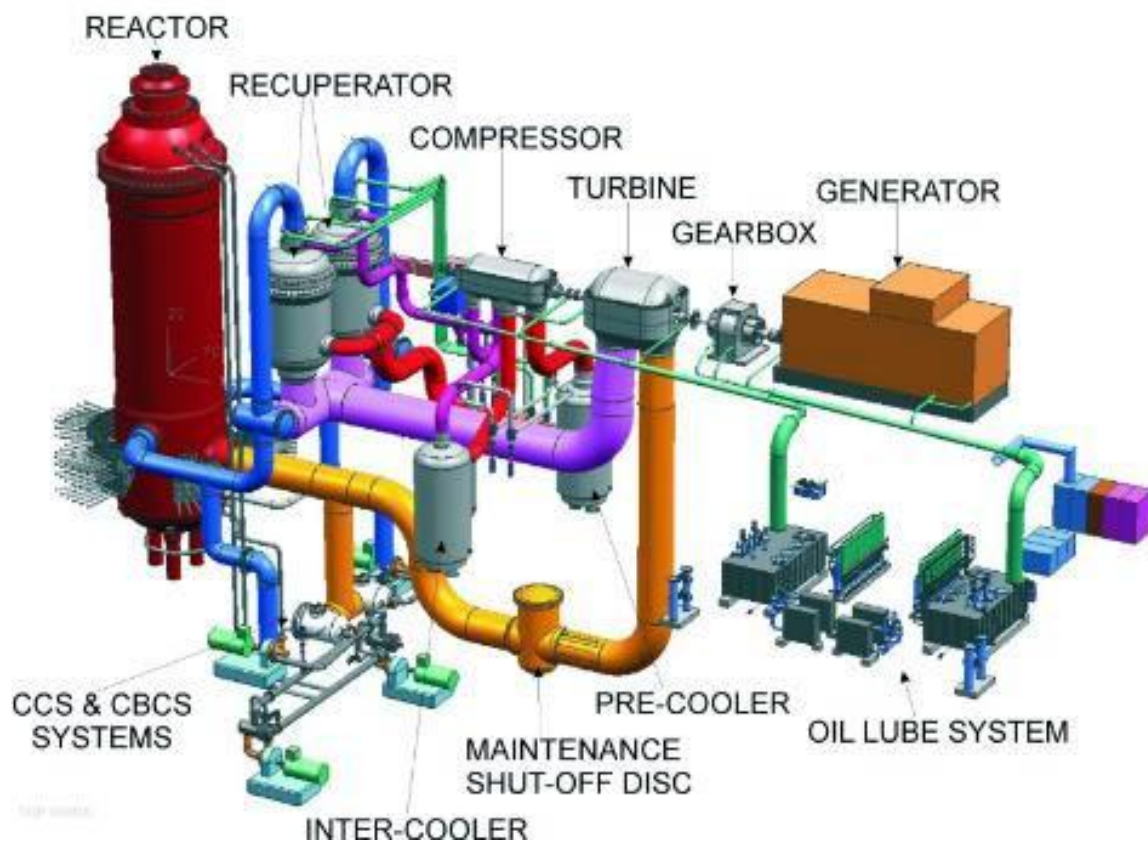
Figure 13. Single reactor unit arrangement of the Pebble Bed Modular Design PBMR design using the gas turbine or Brayton thermodynamic cycle.

Refueling is done online, eliminating refueling outages. The PBMR would shut down every few years for maintenance of other mechanical parts of the plant. The staff would be constantly taking pebbles out of the bottom, checking their burnup, eliminating any leakers, and then reloading them back from the top, or adding fresh pebbles to replace the discarded ones..

The spent fuel pebbles are passed pneumatically to large storage tanks at the base of the plant. This storage space can hold all spent fuel throughout the plant's life. These tanks can hold the fuel for 40 to 50 years after shutdown. About 2.5 million walls are normally used over the 40

years design life of a typical reactor. The silicon carbide coating on the fuel particles can isolate the fission products, at least in theory for a million years. For permanent storage, these pebbles are easier to store than fuel rods from PWRs.

The PBMR is expected to have an overall thermal efficiency of bout 40-42 percent. It operates at a low power density of less than 4.5 MW/m^3, compared with a value of 100 for a PWR.

The economic advantage of the PBMR is that it can allow a utility to make a decision on investing about 120 million dollars rather than the 2-3 billion dollars for other power plants. In addition, the construction time can be reduced to the 18-36 months level as opposed to the 5 or more years for light water reactors. The operating costs of the concept because of the staffing characteristics and the lower fuel costs. It can satisfy the information technology needs which are increasing the load demand in the USA at 4-4 1/2 percent rate per year, for instance around Chicago.

It possesses a high degree of inherent safety. The worst case scenario would produce temperatures below fuel damage temperatures.

The design provides a containment structure provide for regulatory needs only the reason is that the type of accidents envisioned would take hours to days to develop, as compared to minutes in PWRs. In the USA a Nuclear Regulatory Commission (NRC) policy statement designated as SECY 93-092, provides guidance in this regard. It concludes that conventional containment is not needed for such a design. Nevertheless, the PBMR design allows for the release of the helium coolant in the case of a loss of coolant accident. The containment structure is there to intercept any fission products release within days of the initiation of any accident.

The design would require a smaller Emergency Planning Zone (EPZ). The control rods are used only for compensating for the initial heat up and for achieving full cold shutdown. For temperature control, the helium pressure is lowered or raised. To decrease the power level, the temperature is decreased by increasing the helium pressure and vice versa.

In a bridging to a helium economy in the future, high temperature systems can be used to dissociate water into oxygen and hydrogen on a global scale. This would satisfy in a nonpolluting manner the needs of both industrialized and developed nations. Expectations are for the doubling of electricity demand worldwide by 2020.

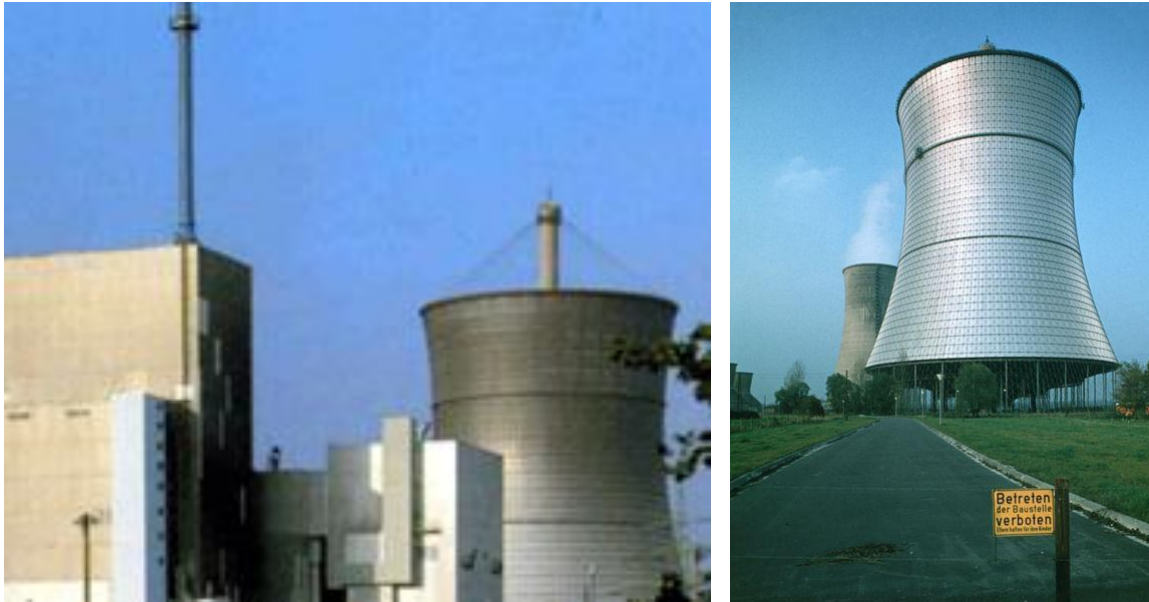## THE THORIUM HIGH TEMPERATURE REACTOR THTR-300 PEBBLE BED REACTOR

Figure 14. The TFTR-300 U^{235}-Th^{232} high temperature pebble bed reactor used a 180 m high dry cooling tower.

The Thorium High Temperature Reactor THTR-300 was operated at a 300 MWe level over the period 1983-1989 in Germany. It used U^{235} as a seed fissile fuel and Th^{232} as a breeder material.

It was preceded by the AVR experimental reactor where the pebble graphite BISO fuel was developed.

One of its unique features was that it used a dry cooling tower using air rather than water cooling. This was possible because of its high temperature operation resulting in a thermal efficiency of 50 percent compared with about 33 percent for water cooled reactors. This feature makes the concept for siting in arid regions of the world without a source of cooling water.

The pebble bed core consisted of 670,000 spherical fuel pebbles 6 cm in diameter each. Graphite was the moderator material and helium as the coolant. The pressure vessel was pre-stressed concrete. The power conversion system was similar to the Fort St. Vrain reactor in the USA with the use of a steam cycle rather than a Brayton cycle.

On May 4, 1985, a fuel pellet became lodged in a fuel feed pipe to the core and the plant was later decommissioned. A follow up design of 500 MWe power was not pursued.

HIGH TEMPERATURE ENGINEERING TEST REACTOR, HTTR

The High Temperature Engineering Test Reactor (HTTR) is a 30 MW(th) prismatic core HTGR designed, constructed and operated by the Japan Atomic Energy Research Institute (JAERI). The reactor achieved initial criticality in 1998 with full power operation started in 2000.

It supports the development of high temperature process heat and closed cycle gas turbine technology. It uses prismatic fuel in the form of hexagonal blocks 35 cm in width and 75 cm in height.

JAERI has been constructing the High Temperature Engineering Test Reactor with a nominal outlet coolant temperature of 850 ºC, and a maximum of 950 ºC, as the facility for the development of a hydrogen production system using high temperature gas from the HTTR. It is also meant as a source for process heat for chemical manufacture.
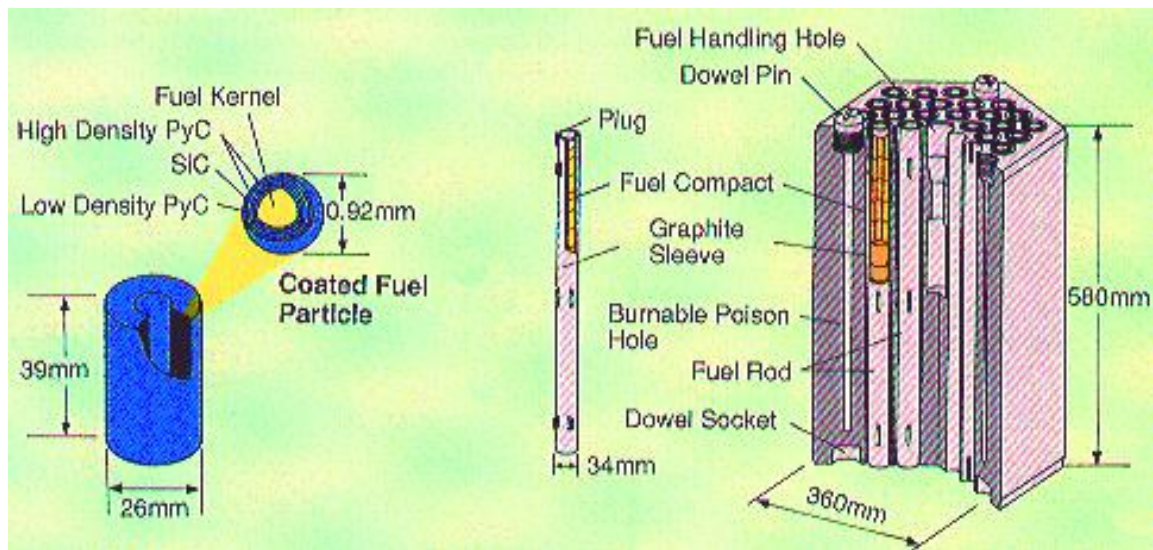

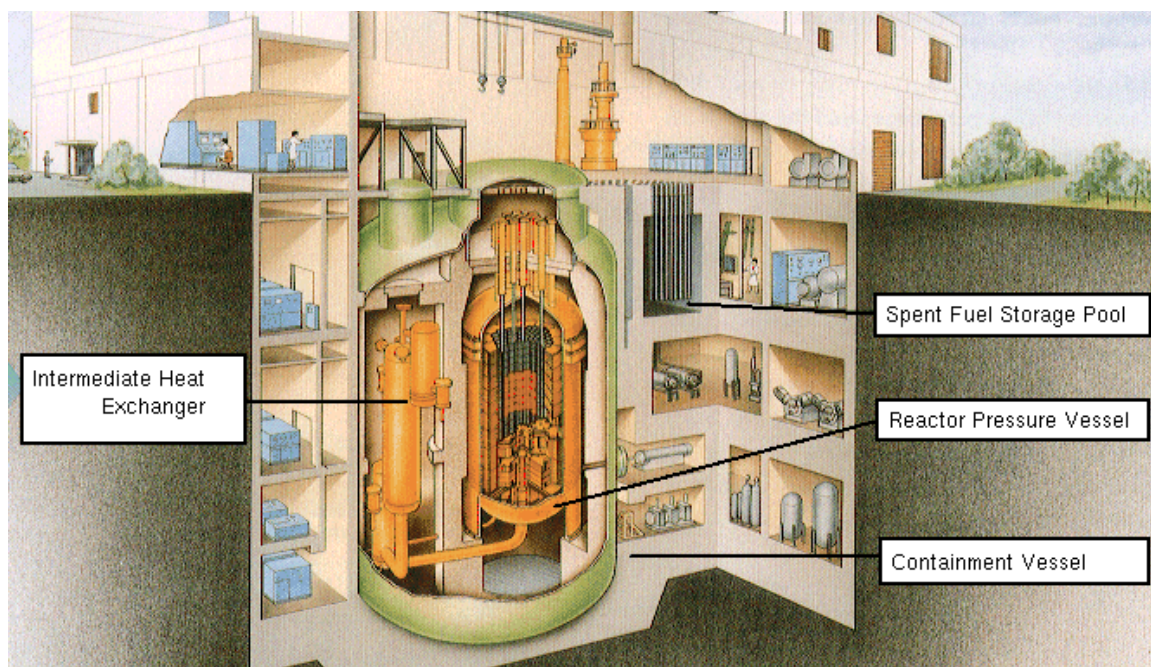Figure 15. Prismatic hexagonal graphite fuel block.

Figure 16. High Temperature Engineering Test Reactor, HTTR is sited underground.

HIGH TEMPERATURE REACTOR HTR-10

The HTR-10 is a 10 MW(th) pebble bed core HTGR designed, constructed and operated by the Institute for Nuclear Energy Technology (INET) at Tsinghua University in China. The reactor achieved initial criticality in 2000. The design employs pebble fuel composed of 6 cm diameter spheres. It operates with a core outlet temperature of 950 ℃ to support the development of high temperature process heat and electricity generation.

THE MOLTEN SALT THORIUM CYCLE
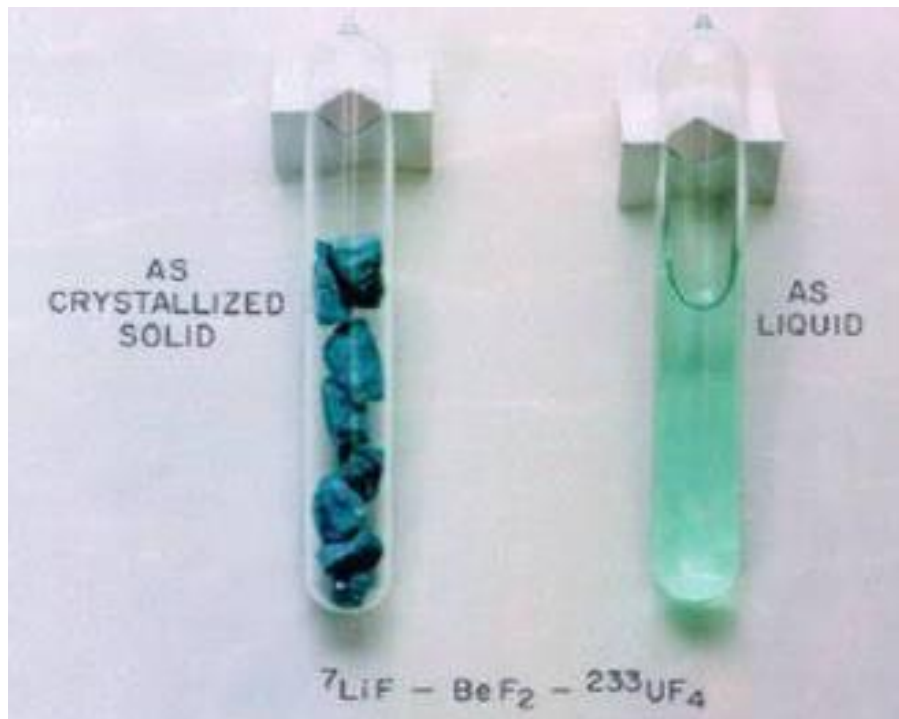
Figure 17. $_3Li^7F$-BeF_2-$_{92}U^{233}F_4$ salt in the cold solid and hot liquid states.

In solid fuel reactors the fission product Xenon^{135} acts as a neutron poison and remains confined in the fuel. In the fluid fuelled thorium cycle it can be readily removed.

The use of a molten salt offers the possibility of developing a sophisticated reprocessing approach eliminating the Np, Am and Cm actinides from the cycle without criticality constraints. One can think about feeding them back into the reactor for eventual fissioning. Their fissioning would be more prominent in a fast neutron spectrum in the case of the hybrid than it would be in a thermal neutron spectrum in the case of a fission system. The current molten salt reprocessing involves:

a) The chemical separation of the bred U^{233} by fluorination to the uranium hexafluoride gas $U^{233}F_6$, and later its reduction into the $U^{233}F_4$ powder for reincorporation into the liquid fuel stream.

b) The strong neutron absorber bred Pa^{233} is stripped using metallic bismuth, and allowing it enough time to decay into U^{233} to be returned to the fuel stream again.

c) The remaining salt is distilled to remove the lighter fission products, with the transuranics remaining in the stream to be burned in the reactor.

FOURTH GENERATION MOLTEN SALT REACTOR: MSR

As one of six Fourth Generation reactors concepts under consideration, the Molten Salt Reactor (MSR) system produces fission power in a circulating molten salt fuel mixture with an epithermal neutron spectrum reactor with graphite core channels, and a full actinide recycle fuel cycle. The MSR can be designed to be a thermal breeder using the Th^{232} to U^{233} fuel cycle.
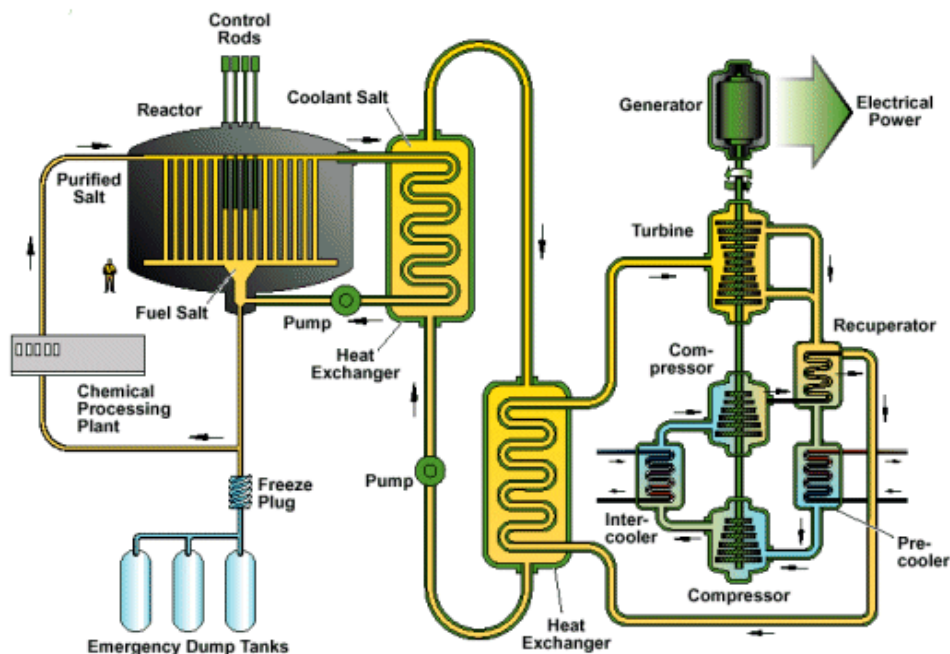
Figure 18. Molten Salt Reactor, MSR using a Brayton, Joule gas turbine cycle. It shows a freeze plug leading to emergency coolant dump tanks as an Engineered Safety Feature, ESF. The MSR is proposed as an actinides burner. Source: INEL.

It is advanced as an actinide burner to solve the problem of actinides production in the U^{238}-Pu^{239} fuel cycle.

In the MSR system, the fuel is a circulating liquid mixture of sodium, zirconium, and uranium fluorides. The molten salt fuel flows through graphite core channels, producing an epithermal spectrum. The heat generated in the molten salt is transferred to a secondary coolant system through an intermediate heat exchanger, and then through a tertiary heat exchanger to the power conversion system. The reference plant has a power level of 1,000 MWe. The system has

a coolant outlet temperature of 700 ºC, possibly ranging up to 800 ºC, affording improved thermal efficiency.

The closed fuel cycle can be tailored to the efficient burnup of plutonium and the minor actinides. The MSR's liquid fuel allows addition of actinides such as plutonium and avoids the need for fuel fabrication. The actinides and most fission products form fluorinides in the liquid coolant. Molten fluoride salts have excellent heat transfer characteristics and a very low vapor pressure, which reduce stresses on the vessel and piping.

An Engineered Safety Feature, ESF involves a Freeze Plug where the coolant is cooled into a frozen state. Upon an unforeseen increase in temperature, this plug would melt leading to the dumping of the coolant into emergency dump tanks. In the absence of moderation by the graphite, the coolant would reside there in a subcritical safe state until pumped back into the core.

Research and development on that concept addresses the selection of a fuel salt with small neutron cross section of the fuel solvent, radiation stability, and a negative temperature coefficient of reactivity. It needs a low melting point good thermal stability, low vapor pressure and adequate heat transfer and viscosity coolant. The secondary salt must be corrosion resistant to the primary salt. Graphite as a moderator would have to be replaced every four years due to radiation damage to its matrix.

COMPARISON TO THE MIXED OXIDE, MOX CYCLE

Plutonium is not a very good light water reactor fuel since in a moderated thermal reactor spectrum, only 2/3 of the Pu^{239} undergoes fission. The other 1/3 absorbs a neutron turning into fissionable but not fissile Pu^{240}, which again absorbs neutrons turning into Pu^{241}. This makes the neutron economy of the Mixed Oxide fuel (MOX) poor, and the use of MOX in LWRs does not solve the problem of actinide nuclear waste.

In addition, the use of the uranium and plutonium oxides complicates fuel fabrication and reprocessing, and it is difficult to recycle plutonium oxides with a 100 percent plutonium recovery. The problems related to the loss of plutonium to the processing waste stream are not manageable.

Plutonium burning becomes more effective if thorium is included in the core fuel or in a surrounding blanket. This favors the use of Molten Salt Reactors as a primary tool for burning plutonium. Plutonium from dismantled nuclear devices or nuclear waste can be used in LFTR start up charges. The residual Pu and minor actinides would not leave the reactor and are burned within. Plutonium can be extracted from the fluoride salts with a relatively simple and low cost chemical process, and it is possible to avoid a plutonium contamination problem in processing the waste stream.

FUSION ISLAND CONSIDERATIONS

For an immediate application of the fusion hybrid using the Th cycle, the DT fusion fuel cycle can be used:

$$_1D^2 + {}_1T^3 \rightarrow {}_2He^4(3.52 \text{ MeV}) + {}_0n^1(14.06 \text{ MeV}) + 17.58 \text{ MeV} \tag{1}$$
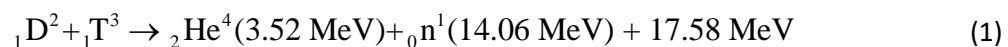
The tritium would have to be bred from the abundant supplies of lithium using the reactions:

$$_3Li^6 + _0n^1(\text{thermal}) \rightarrow {}_2He^4(2.05 \text{ MeV}) + {}_1T^3(2.73 \text{ MeV}) + 4.78 \text{ MeV}$$
$$_3Li^7 + _0n^1(\text{fast}) \rightarrow {}_2He^4 + _0n^1 + {}_1T^3 - 2.47 \text{ MeV}$$

(2)
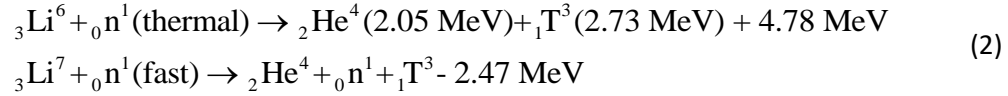
In this case a molten salt containing Li for tritium breeding as well as Th for U^{233} breeding can be envisioned:

$$LiF.BeF_2.ThF_4$$

For a practically unlimited supply of deuterium from water at a deuterium to hydrogen ratio of D/H = 150 ppm in the world oceans, one can envision the use of the catalyzed DD reaction in the fusion island:

$$_1D^2 + _1D^2 \rightarrow {}_1T^3(1.01) + {}_1H^1(3.03) + 4.04 \text{ } MeV$$
$$_1D^2 + _1D^2 \rightarrow {}_2He^3(0.82) + _0n^1(2.45) + 3.27 \text{ } MeV$$
$$_1D^2 + _1T^3 \rightarrow {}_2He^4(3.52) + _0n^1(14.06) + 17.58 \text{ } MeV$$
$$_1D^2 + _2He^3 \rightarrow {}_2He^4(3.67) + {}_1H^1(14.67) + 18.34 \text{ } MeV$$

(3)

_____

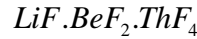$$6{}_1D^2 \rightarrow 2{}_1H^1 + 2{}_2He^4 + 2{}_0n^1 + 43.23 MeV$$
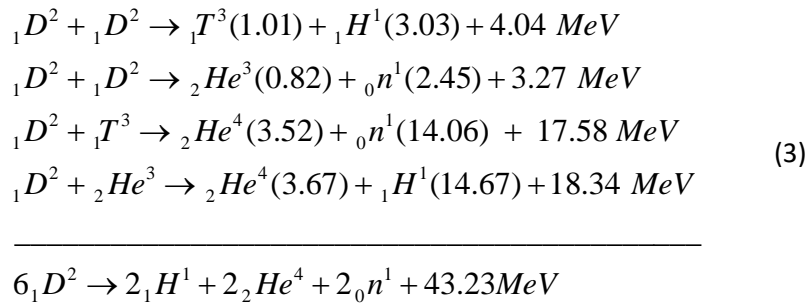
with each of the five deuterons contributing an energy release of 43.23/6 = 7.205 MeV.

For plasma kinetic reactions temperatures below 50 keV, the DHe^3 reaction is not significant and the energy release would be 43.23 - 18.34 = 24.89 with each of the five deuterons contributing an energy release of 24.89/5 = 4.978 MeV.

$$_1D^2 + _1D^2 \rightarrow {}_1T^3(1.01) + {}_1H^1(3.03) + 4.04 \text{ } MeV$$
$$_1D^2 + _1D^2 \rightarrow {}_2He^3(0.82) + _0n^1(2.45) + 3.27 \text{ } MeV$$
$$_1D^2 + _1T^3 \rightarrow {}_2He^4(3.52) + _0n^1(14.06) + 17.58 \text{ } MeV$$

(4)

_____

$$5{}_1D^2 \rightarrow {}_1H^1 + {}_2He^3 + {}_2He^4 + 2{}_0n^1 + 24.89 MeV$$
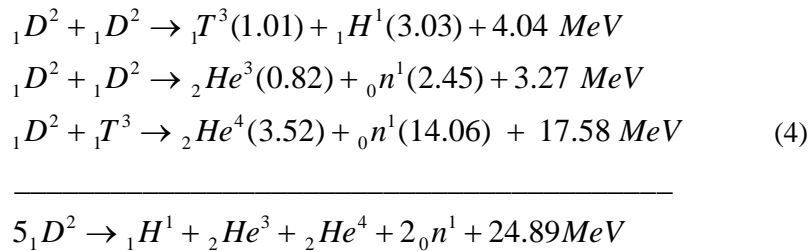
In this case, there would be no need to breed tritium, and the lithium can be replaced by Na in a molten salt with the following composition:

$$NaF.BeF_2.ThF_4$$

With a density and percentage molecular composition of:

$$\rho = 4.52 \frac{gm}{cm^3}, (71 - 2 - 27\, mol\%)$$

FUSION FISSION HYBRID LIQUID METAL THORIUM BREEDER

REACTOR CONFIGURATION AND CALCULATIONAL MODEL

A one dimensional calculational model considers a plasma cavity with a 150 cm radius. The plasma neutron source is uniformly distributed in the central 100 cm radial zone and is isolated from the first structural wall by a 50 cm vacuum zone.

The blanket module consists of a 1 cm thick Type 316 stanless steel first structural wall that is cooled by a 0.5 cm thick water channel, a 42 cm thick molten salt filled energy absorbing and breeding compartment, and a 40 cm thick graphite neutron reflector.

The molten salt and graphite are contained within 1 cm thick Type 316 stainless steel structural shells.

Table 1. Fusion-fission reactor geometrical model.

Material	Zone	Outer Radius (cm)	Thickness (cm	Remarks
Plasma	1	100.0	100.0	DT(14.06 MeV) or, Catalyzed DD (50 % 2.45 MeV + 50 % 14.06 MeV)
Void	2	150.0	50.0	Vacuum zone
First wall	3	151.0	1.0	Type 316 stainless steel
Water coolant	4	151.5	0.5	H_2O cooling channel
Structure	5	152.5	1.0	
Molten salt	6	194.5	42.0	$NaF.BeF_2.ThF_4$ or: $LiF.BeF_2.ThF_4$ ρ =4.52 gm/cm^3 (71-2-27 mol %)
Structure	7	195.5	1.0	Type 316 stainless steel
Neutron reflector	8	235.5	40.0	Graphite as C^{12}
Structure	9	236.5	1.0	Type 316 stainless steel
Albedo	10	-	-	20 percent albedo surface to simulate neutron and gamma ray reflection

Table 2. Fusion-fission material compositions.

Material	Composition	Nuclide Density [nuclei/(b.cm)]
1. LiF.BeF$_2$.ThF$_4$ salt ρ = 4.52 gm/cm^3 71-2-27 mol %	$_3$Li^6 $_3$Li^7 $_4$Be^9 $_{90}$Th^{230} $_9$F^{19}	1.414x10^{-3} 1.744x10^{-2} 5.310x10^{-4} 7.169x10^{-3} 4.859x10^{-2}
2. NaF.BeF$_2$.ThF$_4$ salt ρ =4.52 gm/cm^3 71-2-27 mol %	$_{11}$Na^{23} $_4$Be^9 $_{90}$Th^{230} $_9$F^{19}	1.697x10^{-2} 4.799x10^{-4} 6.452x10^{-3} 4.373x10^{-2}
3. Type 316 stainless steel 63.6 wt% Fe, 18 wt% Cr, 13 wt% Ni, 2.6 wt% Mo, 1.9 wt% Mn, 0.9 wt% (Si+Ti+C) ρ = 7.98 gm/cm^3	C Si Ti Cr Mn Fe Ni Mo	1.990x10^{-4} 1.360x10^{-3} 4.980x10^{-5} 1.150x10^{-2} 1.650x10^{-3} 5.430x10^{-2} 1.060x10^{-2} 1.290x10^{-3}
4. Graphite ρ =2.25 gm/cm^3	C	1.128x10^{-1}
5. H$_2$O ρ = 1.0 gm/cm^3	H O	6.687x10^{-2} 3.343x10^{-2}

BREEDING ESTIMATES

Computations were conducted using the one dimensional discrete ordinates transport ANISN code with a P$_3$ Legendre expansion and an S$_{12}$ angular quadrature.

The catalyzed DD system exhibits a fissile nuclide production rate of 0.880 Th(n, γ) reactions per fusion source neutron. The DT system, in addition to breeding tritium from lithium for the DT reaction yields 0.737 Th(n, γ) breeding reactions per fusion source neutron. Both approaches provide substantial energy amplification through the fusion-fission coupling process.

Table 3. Fissile and fusile breeding for sodium and lithium salts in DT and DD symbiotic fusion-fission fuel factories. Blanket thickness = 42 cm, reflector thickness =40 cm; no structure in the salt region.

Source	Li-Be-Th-F Salt						Na-Be-Th-F Salt			
	Li^6(n,α)T	Li^7(n,n'α)T	Be^9(n,T)	F(n,T)	Total T	Th(n,γ)	Be^9(n,T)	F(n,T)	Total T	Th(n,γ)
	(Nuclei / fusion source neutron)									
DD 100% 2.45 MeV	0.311	0.001	4.03x10^{-10}	1.01x10^{-7}	0.312	0.579	4.18x10^{-10}	1.04x10^{-7}	1.04x10^{-7}	0.794
DT	0.391	0.073	1.08x10^{-4}	3.33x10^{-3}	0.467	0.737	1.04x10^{-4}	3.08x10^{-3}	3.18x10^{-3}	0.966

| 100% 14.06 MeV | | | | | | | | | |
| Catalyzed DD 50% 2.45 MeV 50% 14.06 MeV | 0.351 | 0.037 | 5.40×10^{-5} | 1.67×10^{-3} | 0.390 | 0.658 | 5.20×10^{-5} | 1.54×10^{-3} | 1.59×10^{-3} | 0.880 |

The largest Th(n,γ) reaction rate (0.966) occurs when the sodium salt is used in conjunction with the DT reaction. For this case, however, the tritium required to fuel the plasma must be supplied to the system, since that produced in the blanket would be negligible (3.18×10^{-3}). A system of such kind has been proposed and studied by Blinken and Novikov.

For the catalyzed DD plasma, the Th(n,γ) reaction rate is 0.880 in the sodium salt compared with 0.737 obtained in the DT system using the lithium salt. For the DT system the tritium production rate is 0.467 tritium nuclei per source neutron. The tritium production rate is too low to sustain the DT plasma, and an extra supply of the isotope is required to supplement the plasma. This could be internal by allowing some fissions to occur in the blanket supplying extra neutrons, or this could be external in the external satellite fission reactors producing the needed difference as a byproduct and feeding it back to the DT plasma.

The fissile fuel production rate is higher in the DD system because of the absence of lithium in the salt that would introduce a competition between the Th(n,γ) and the Li^6(n,T) reactions.

SYMBIOTIC THORIUM FUSION GENERATORS AND FISSION SATELLITES

INTRODUCTION

An approach to the fusion-fission hybrid is the fuel factory, where fissile fuel for fission reactors is bred in fusion reactors.

In this approach, there is a need to maximize the fissile breeding with the constraint of maintaining self sufficiency in tritium production and realistically accounting for the modeling for structural and coolant compositions and configurations imposed by the thermal hydraulic and mechanical designs.

For the DT fusion reaction cycle there appears a need for neutron multiplication using threshold neutron multiplication reactions such as from Pb and Be.
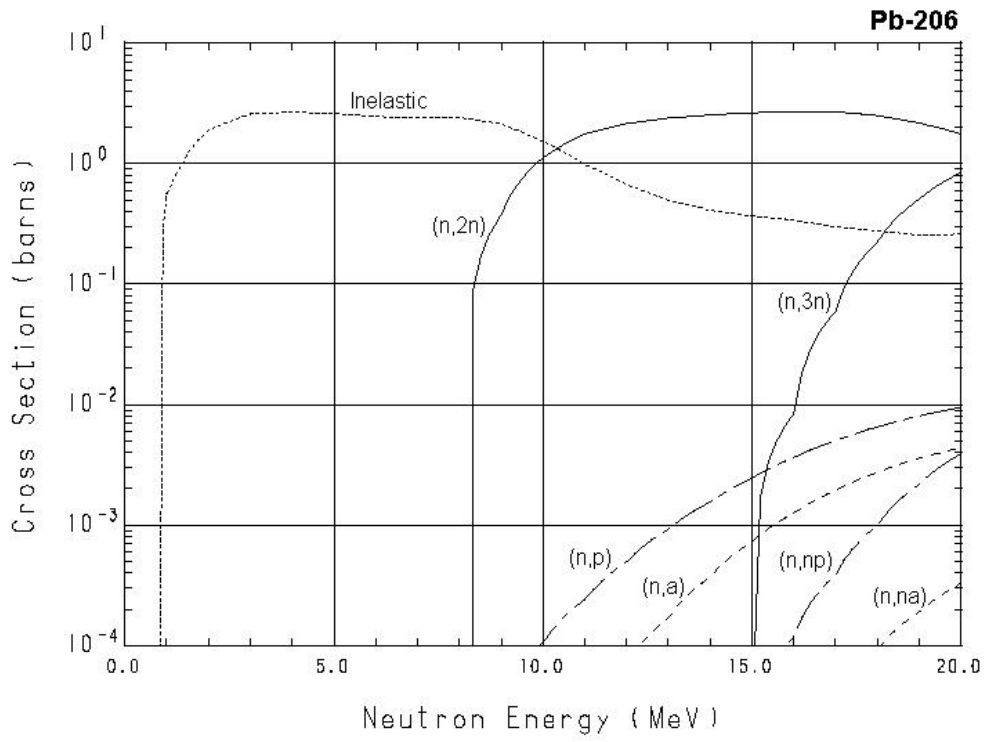
Figure 19. Neutron multiplication threshold reactions in Pb^{206}. Source: JENDL.
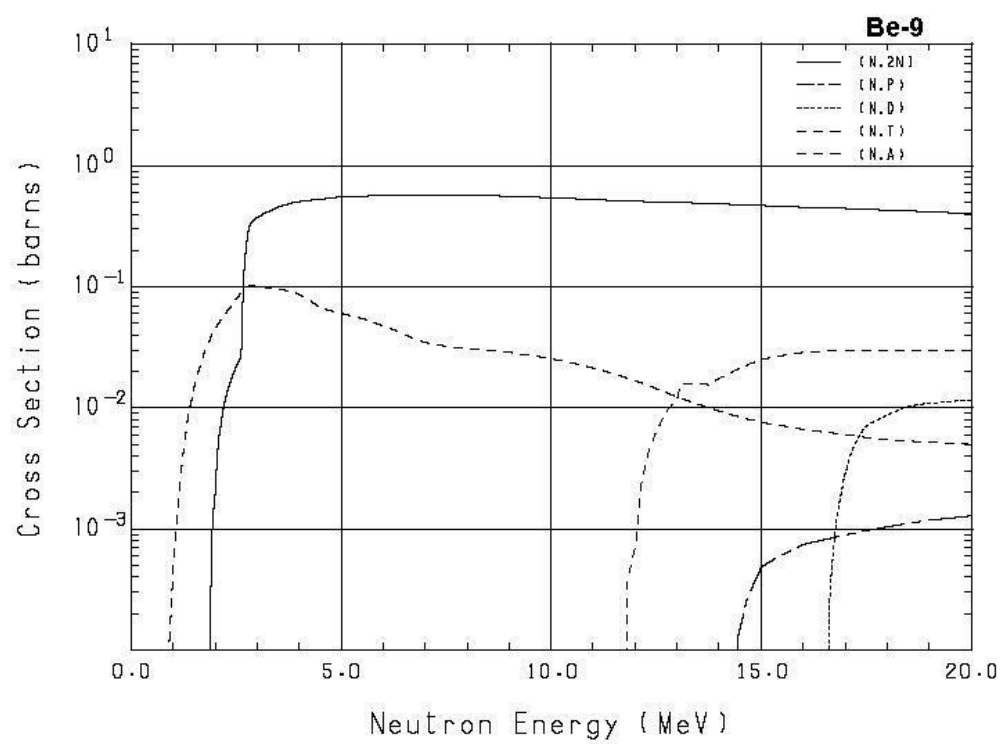
Figure 20. Neutron multiplication threshold reactions in Be[9]. Source: JENDL.
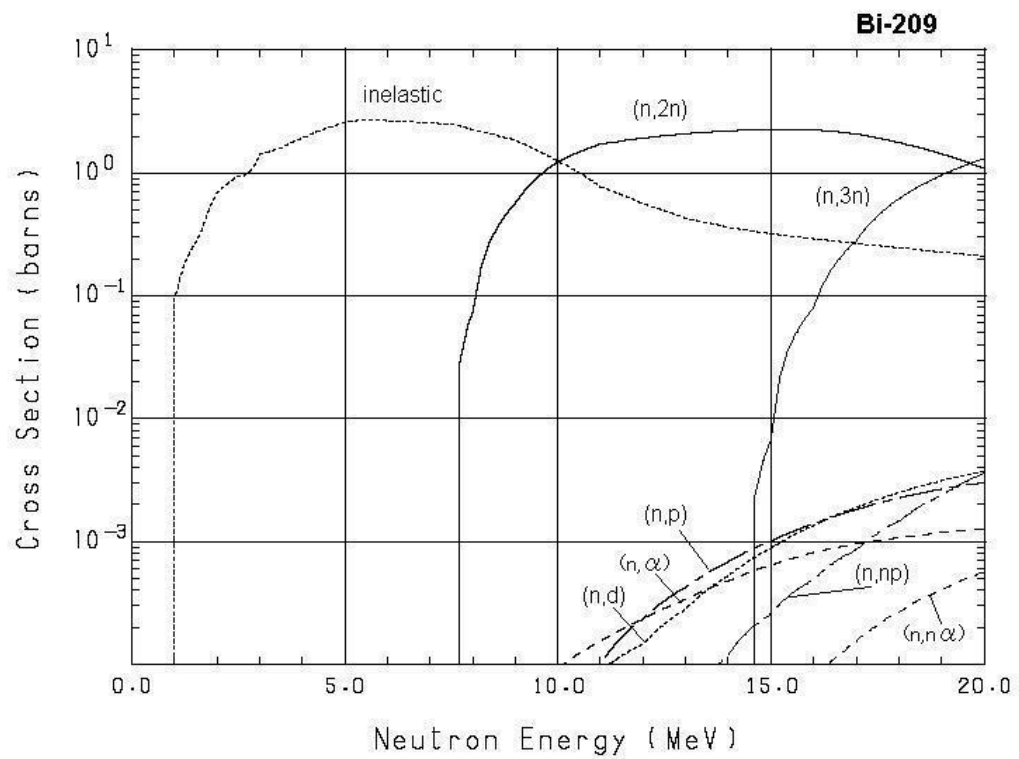
Figure 21. Neutron multiplication threshold reactions in Bi^{209}. Source: JENDL.
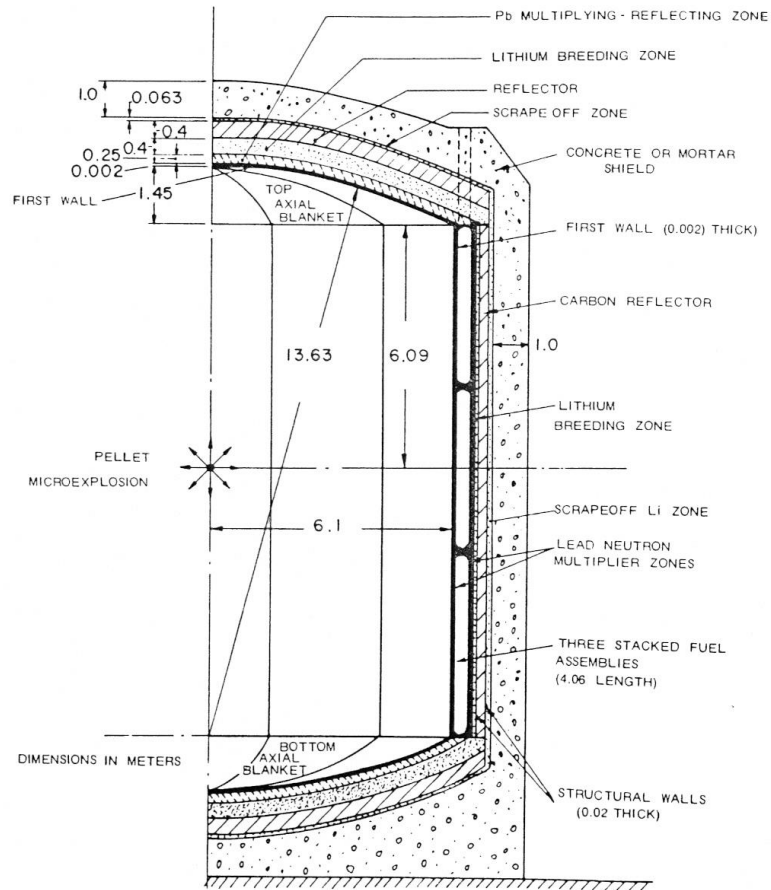
Figure 22. Laser fusion fissile generator plant with U^{233} breeding.
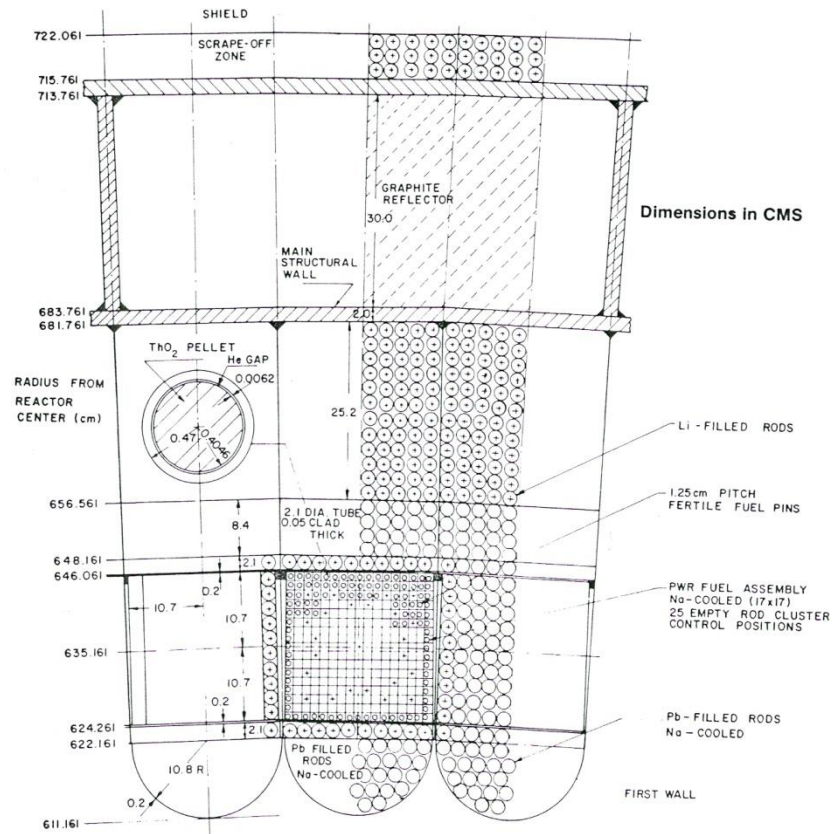
Figure 23. ThO$_2$ Pressurized Water Reactor fuel elements within the neutron multiplication zone, followed by a tritium breeding zone and a graphite reflector.
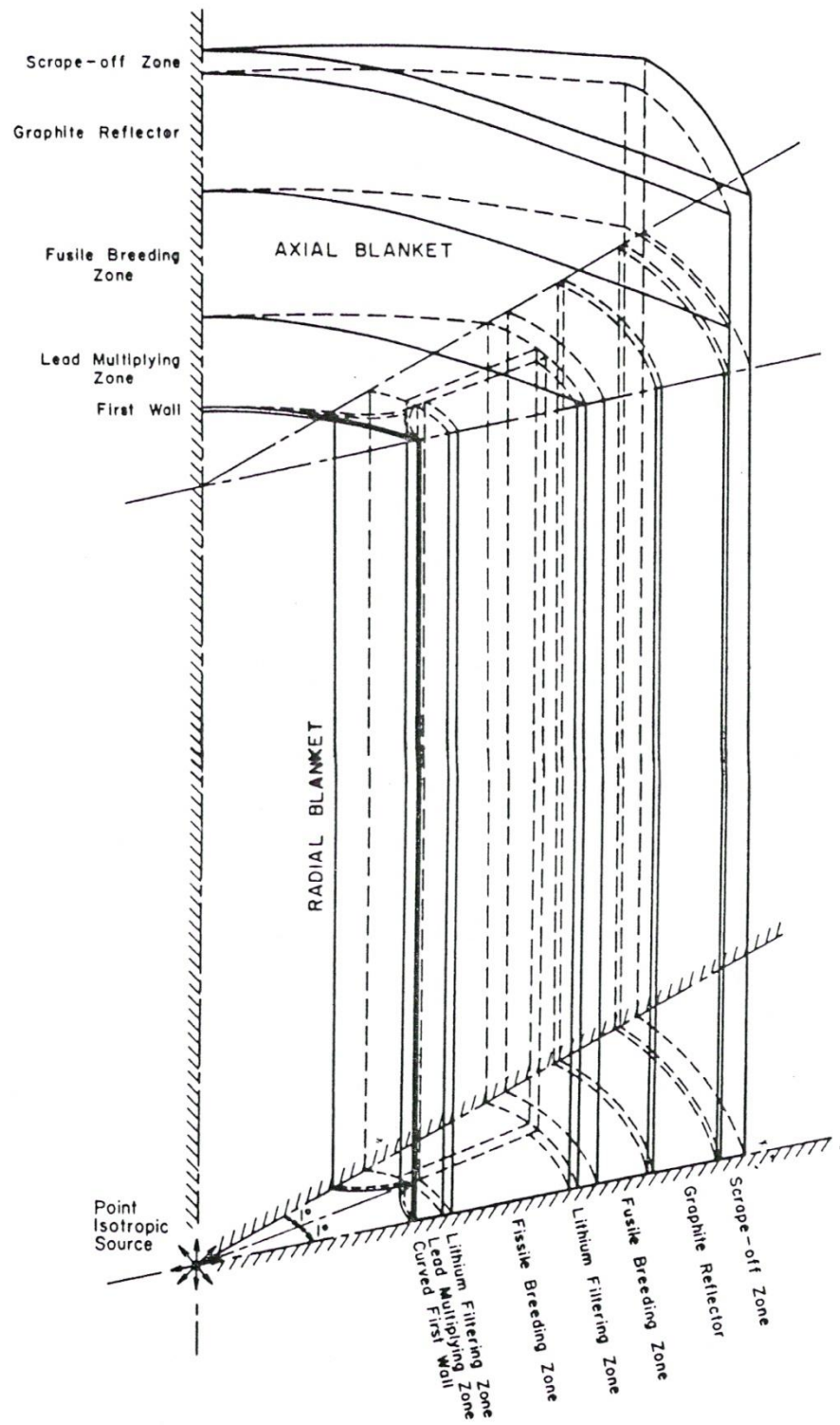
Figure 24. Three-dimensional unit cell of computational model of laser fusion hybrid plant.
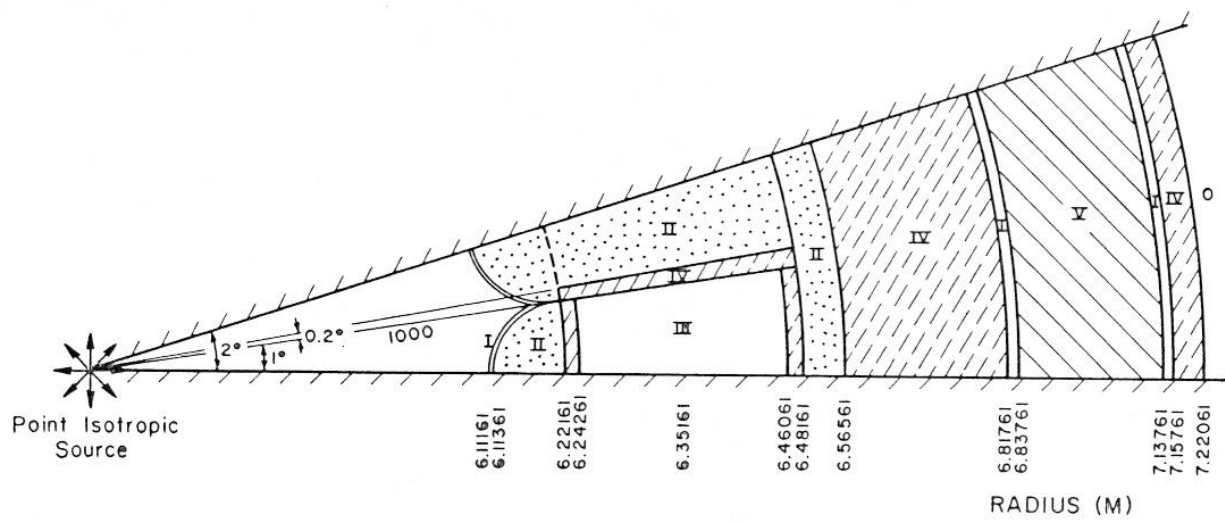
Figure 25. Horizontal cut through unit cell of three dimensional computational model.

Table 4. Material compositions and number densities.

Material composition	Element	Atomic densities [atoms/(barn·m)]
1. First wall and structural wall		
100 v/o Zircaloy-4	Zr	4.374+0
98.24 w/o Zr+1.5 w/o Sn	Sn	4.962−2
+0.21 w/o Fe	Cr	7.812−3
+0.10 w/o Cr	Fe	1.527−2
ρ(Zircaloy-4)=6.745×10³ kg/m³		
2. Reflector	^{12}C	8.373+0
100% Reactor-grade graphite		
ρ(graphite)=1.67×10³ kg/m³		
3. Neutron multiplication zones	Pb	2.145+0
65.03 v/o Pb+844 v/o	Zr	3.692−1
Zircaloy-4+26.53 v/o	Sn	4.188−3
Na coolant.	Cr	6.593−4
ρ(Pb)=11.35×10³ kg/m³	Fe	1.289−3
ρ(Na)=9.71×10² kg/m³	Na	6.748−1
4. Fusile breeding zones	^{6}Li	2.364−1
68.78 v/o natural lithium	^{7}Li	2.950+0
+7.97 v/o Zircaloy-4	Zr	3.486−1
+23.25 v/o Na Coolant	Sn	3.955−3
ρ(Li)=0.534×10³ kg/m³,	Cr	6.226−4
7.22 a/o ^{6}Li+92.58 a/o	Fe	1.217−3
^{7}Li	Na	5.914−1
5. Fissile breeding zone	Th	6.415−1
28.10 v/o ThO₂ + 10.47 v/o	^{16}0	1.283+0
Zircaloy-4	Zr	4.580−1
+60.98 v/o Na Coolant	Sn	5.195−3
+1.15 v/o He Fill Gas	Cr	8.179−4
ρ(ThO₂)=10.01×10³ kg/m³	Fe	1.599−3
	Na	1.533+0

BREEDING ESTIMATES

A tritium yield per source neutron of 1.08 and a Th (n, γ) reaction yield of 0.43 can be obtained in the concept where ThO_2 Zircaloy-clad fuel assemblies for Light Water Reactors (LWRs) are enriched in the U^{233} isotope by irradiating them in a Pb flux trap.

This corresponds to 0.77kg/[MW(th).year] of fissile fuel production, and 1.94 years of irradiation in the fusion reactor to attain an average 3 w/o fissile enrichment in the fuel assemblies.
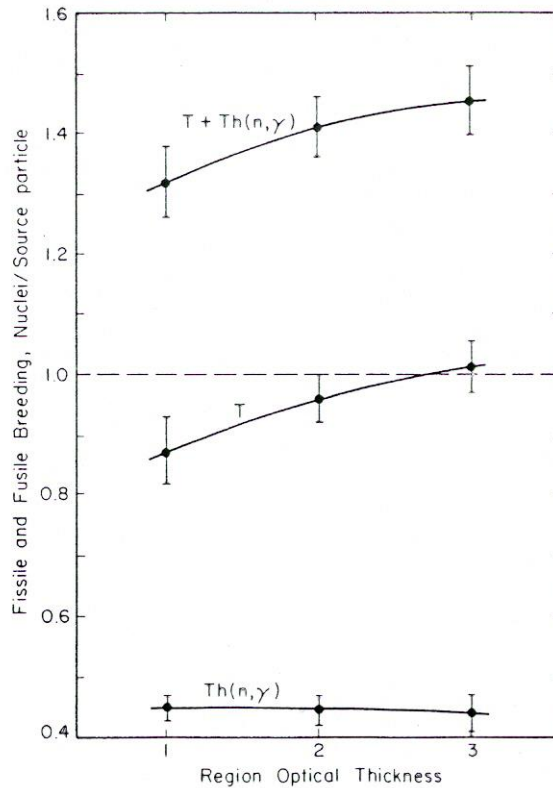
Figure 26. Optimization of fissile U^{233} and fusile tritium (T) breeding.

For a once through LWR cycle, a support ratio of 2-3 is estimated. However, with fuel recycling, more attractive support ratios of 4-6 may be attainable for a conversion ratio of 0.55, and 5-8 for a conversion ratio of 0.70.

ACTINIDES PRODUCTION

There has been a new interest in the Th cycle in Europe and the USA since it can be used to increase the achievable fuel burnup in LWRs in a once through fuel cycle while significantly reducing the transuranic elements in the spent fuel. A nonproliferation as well as transuranics waste disposal consideration is that just a single neutron capture reaction in U^{238} is needed to produce Pu^{239} from U^{238}:

$$
\begin{aligned}
&_{0}n^{1} + {}_{92}U^{238} \rightarrow {}_{92}U^{239} + \gamma \\
&\quad {}_{92}U^{239} \xrightarrow{23.5m} {}_{93}Np^{239} + {}_{-1}e^{0} \\
&\quad {}_{93}Np^{239} \xrightarrow{2.35d} {}_{94}Pu^{239} + {}_{-1}e^{0}
\end{aligned}
\tag{5}
$$
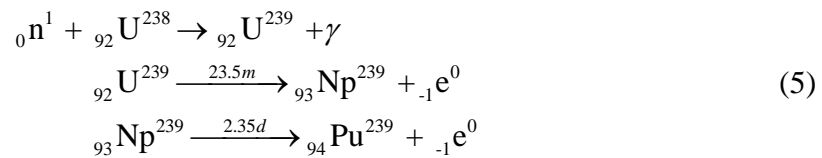
whereas a more difficult process of fully 5 successive neutron captures are needed to produce the transuranic Np^{237} from Th^{232}:

$$
\begin{aligned}
&_0n^1 + {}_{90}Th^{232} \rightarrow {}_{90}Th^{233} + \gamma \\
&_0n^1 + {}_{90}Th^{233} \rightarrow {}_{90}Th^{234} + \gamma \\
&_{90}Th^{234} \xrightarrow{24.1d} {}_{91}Pa^{234} + {}_{-1}e^0 \\
&_{91}Pa^{234} \xrightarrow{6.70h} {}_{92}U^{234} + {}_{-1}e^0 \\
&_0n^1 + {}_{92}U^{234} \rightarrow {}_{92}U^{235} + \gamma \\
&_0n^1 + {}_{92}U^{235} \rightarrow {}_{92}U^{236} + \gamma \\
&_0n^1 + {}_{92}U^{236} \rightarrow {}_{92}U^{237} + \gamma \\
&_{92}U^{237} \xrightarrow{6.75d} {}_{93}Np^{237} + {}_{-1}e^0
\end{aligned}
\tag{6}
$$
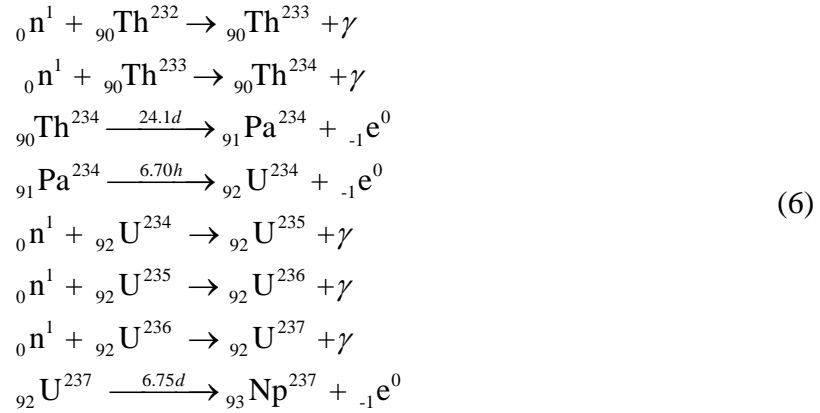
This implies a low yield of Np^{237} however, as an odd numbered mass number isotope posing a possible proliferation concern; whatever small quantities of it are produced, provisions must be provided in the design to have it promptly recycled back for burning in the fast neutron spectrum of the fusion part of the hybrid.

In fact, it is more prominently produced in thermal fission light water reactors using the uranium cycle and would be produced; and burned, in fast fission reactors through the (n, 2n) reaction channel with U^{238} according to the much simpler path:

$$
\begin{aligned}
&_0n^1 + {}_{92}U^{238} \rightarrow 2{}_0n^1 + {}_{92}U^{237} \\
&_{92}U^{237} \xrightarrow{6.75d} {}_{93}Np^{237} + {}_{-1}e^0
\end{aligned}
\tag{6$'$}
$$
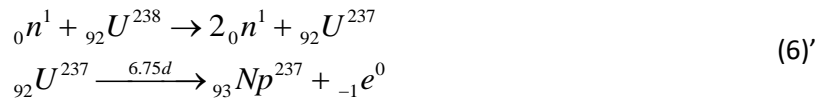
The Np^{237} gets transmuted in the Th^{232} fuel cycle into Pu^{238}:

$$
\begin{aligned}
&_0n^1 + {}_{93}Np^{237} \rightarrow {}_{93}Np^{238} + \gamma \\
&_{93}Np^{238} \xrightarrow{2.12d} {}_{94}Pu^{238} + {}_{-1}e^0
\end{aligned}
$$
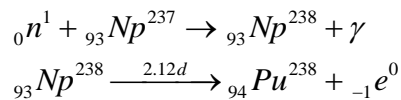
A typical 1,000 MWe Light Water Reactor (LWR) operating at an 80 percent capacity factor produces about 13 kgs of Np^{237} per year.

This has led to suggested designs where Th^{232} replaces U^{238} in LWRs fuel and accelerator driven fast neutron subcritical reactors that would breed U^{233} from Th^{232}.

Incidentally, whereas the Pu238 isotope is producrd in the Th fuel cycle, it is the Pu240 isotope that is produced in the U fuel cycle:

$$_0n^1 + {_{92}}U^{238} \rightarrow {_{92}}U^{239} + \gamma$$

$$_{92}U^{239} \rightarrow {_{93}}Np^{239} + {_{-1}}e^0 + v^* + \gamma$$

$$_{93}Np^{239} \rightarrow {_{94}}Pu^{239} + {_{-1}}e^0 + v^* + \gamma$$

$$_0n^1 + {_{94}}Pu^{239} \rightarrow {_{94}}Pu^{240} + \gamma$$
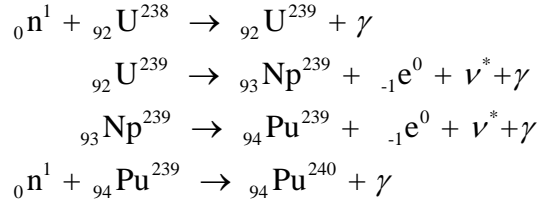
PHYSICAL AND CHEMICAL PROPERTIES

Thorium (Th) is named after Thor, the Scandinavian god of war. It occurs in nature in the form of a single isotope: Th^{232}. Twelve radioactive isotopes are known for Th. It occurs in Thorite ($ThSiO_4$) and Thorianite ($ThO_2 + UO_2$). It is 3 times as abundant as uranium in the earth's crust and as abundant as lead and molybdenum. It is commercially obtained from the monazite mineral containing 3-9 percent ThO_2 with other rare-earths minerals. Its large abundance will make it in the future valuable for electrical energy generation, sea water desalination and hydrogen production with supplies exceeding both coal and uranium combined. This would depend on breeding of the fissile isotope U^{233} from the fertile isotope Th^{232} according to the breeding reactions:

$$_0n^1 + {_{90}}Th^{232} \rightarrow {_{90}}Th^{233} + \gamma$$

$$_{90}Th^{233} \xrightarrow{\quad 22.2m \quad} {_{91}}Pa^{233} + {_{-1}}e^0 \qquad (7)$$

$$_{91}Pa^{233} \xrightarrow{\quad 27.0d \quad} {_{92}}U^{233} + {_{-1}}e^0$$

Together with uranium, the thorium's radioactive decay chain leads to stable lead isotopes with a half-life of 1.4×10^{10} years for Th^{232}. It contributes to the internal heat generation in the Earth.

As Th^{232} decays into the stable Pb^{208} isotope, $Radon^{220}$, initially designated as Thoron, forms in the chain. Rn^{220} has a low boiling point and 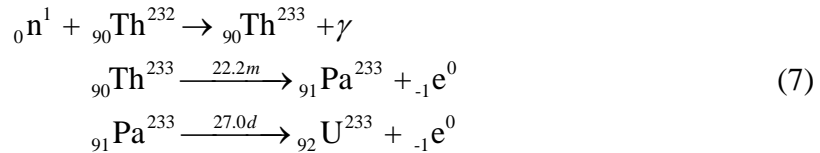exists in gaseous form at room temperature. It poses a radiation hazard through its own daughter nuclei. Radon from uranium and thoron from thorium tests are needed to check for their presence in new homes that are possibly built on rocks like granite or sediments like shale or phosphate rock containing significant amounts of uranium and thorium and their decay daughter nuclei. Adequate ventilation of homes that are over-insulated becomes a design consideration in this case.

Thorium, in the metallic form, can be produced by reduction of ThO_2 using calcium or magnesium in crucibles. Also by electrolysis of anhydrous thorium chloride in a fused mixture of Na and K chlorides, by calcium reduction of Th tetrachloride mixed with anhydrous zinc chloride, and by reduction with an alkali metal of Th tetrachloride.

Thorium is the second member of the actinides series in the periodic table of the elements. When pure, it is soft and ductile, can be cold rolled and drawn and it is a silvery white

metal retaining its luster in air for several months. If contaminated by the oxide, it tarnishes in air into a gray then black color.

Thorium oxide has the highest melting temperature of all the oxides at 3,300 degrees C. Just a few other elements and compounds have a higher melting point such as tungsten and tantalum carbide. Water attacks it slowly, and acids do not attack it except for hydrochloric acid.

Thorium in the powder form is pyrophyric and can burn in air with a bright white light. In portable gas lights, the Welsbach mantle is prepared with ThO_2 with an added 1 percent cerium oxide and other ingredients. As an alloying element in magnesium, it gives high strength and creep resistance at high temperatures. Tungsten wire and electrodes used in electrical and electronic equipment such as electron guns in x-ray tubes or video screens are coated with Th due to its low work function and associated high electron emission. Its oxide is used to control the grain size of tungsten used in light bulbs and in high temperature laboratory crucibles. Glasses for lenses in cameras and scientific instruments are doped with Th to give them a high refractive index and low light dispersion. In the petroleum industry, it is used as a catalyst in the conversion of ammonia to nitric acid, in oil cracking, and in the production of sulfuric acid.

ABUNDANCE AND OCCURENCE

Thorium is more abundant than uranium and provides a fertile isotope for breeding of the fissile uranium isotope U^{233} in a thermal or fast neutron spectrum. In the Shippingport reactor it was used in the oxide form. In the HTGR it was used in metallic form embedded in graphite. The MSBR used graphite as a moderator and hence was a thermal breeder and a chemically stable fluoride salt, eliminating the need to process or to dispose of fabricated solid fuel elements. The fluid fuel allows the separation of the stable and radioactive fission products for disposal. It also offers the possibility of burning existing actinides elements and does need an enrichment process like the U^{235}-Pu^{239} fuel cycle.

Table 5. Relative abundances of some elements in the Earth's crust. Thorium is 4 times as abundant as uranium.

Element	Symbol	Abundance ppm	Abundance [gms/ton]
Lead	Pb		16
Gallium	Ga		15
Thorium	Th	10	10
Samarium	Sm		7
Gadolinium	Gd		6
Praseodymium	Pr		6
Boron	B		3
Bromine	Br		3
Uranium	U		2.5

Beryllium	Be		2
Tin	Sn		1.5
Tungsten	W	1	1
Molybdenum	Mo		1
Mercury	Hg	0.1	0.2
Silver	Ag		0.1
Uranium^{235}	U^{235}		0.018
Platinum	Pt		0.005
Gold	Au		0.02

Thorium is abundant in the Earth's crust, estimated at 120 trillion tons. The monazite black sand deposits are composed of 12 percent of thorium. It can be extracted from granite rocks and from phosphate rocks, rare earths, tin ores, coal and uranium mines tailings. It could also be extracted from the ash of coal power plants.

A 1,000 MWe coal power plant generates about 13 tons of thorium per year in its ash. Each ton of thorium can in turn generate 1,000 MWe of power in a well optimized thorium reactor. Thus a coal power plant can conceptually fuel 13 thorium plants of its own power. From a different perspective, 1 pound of Th has the energy equivalent of 5,000 tons of coal. There are 31 pounds of Th in 5,000 tons of coal. If the Th were extracted from the coal, it would thus yield 31 times the energy equivalent of the coal.

The calcium sulfate or phospho-gypsum resulting as a waste from phosphate rocks processing into phosphate fertilizer contains substantial amounts of unextracted thorium and uranium.

Uranium mines with brannerite ores generated millions of tons of surface tailings containing thoria and rare earths.

The United States Geological Survey (USGS), as of 2008, estimated that the USA has reserves of 915,000 tons of thorium ore existing on properties held by Thorium Energy Inc. in Montana and Idaho. This compares to a previously estimated 160,000 tons for the entire USA.

The next highest global thorium ores estimates are for Australia at 300,000 tons and India with 290,000 tons.

CHARACTERISTICS OF THE THORIUM FUEL CYCLE

The need for U^{235} enrichment for nuclear naval propulsion and for nuclear devices using both U^{235} and Pu^{239} were factors in favoring the adoption of the U^{235} fuel cycle. Historically, among the nuclear energy pioneer scientists, Enrico Fermi advocated the use of the U^{235}-Pu^{239} fuel cycle, whereas Eugene Wigner favored the longer term Th^{232}-U^{233} fuel cycle. Pu^{239} is considered a better fuel for fast reactors using liquid metals such as sodium or lead as coolants. The thorium cycle technology was tested in the Shippingport pilot plant during the 1970-1980 period.

The MSBR operates at low pressures implying a higher level of safety than existing Light Water Reactors (LWRs) where depressurization can lead to the water coolant flashing into steam leading to a Loss of Coolant Accident (LOCA). It can operate at high temperatures, implying a higher thermal efficiency reaching 50 percent compared with 33 percent in the LWR system. It offers the possibility of high temperature process heat production for hydrogen production as a fission energy carrier for use as a transportation fuel in a fuel cells based infrastructure.

The MSBR was 100-300 times more fuel efficient than the LWR, reducing the volume of the disposed fission products waste and using for centuries the existing stockpiles of thorium and uranium. Over the last 40 years, $25 million have been collected into a trust fund from the electrical utilities to fund waste disposal facilities. A part of these could be used for the development of the thorium fuel cycle preventing the creation of new long-lived nuclear waste.

ADVANTAGES OF THE THORIUM FUEL CYCLE

The following advantages of the thorium fuel cycle over the U^{235}-Pu^{239} fuel cycle can be enumerated:

1. Expanded nuclear fuel resources due to the higher abundance of the fertile Th^{232} than U^{238}. The USA proven resources in the state of Idaho amount to 600,000 tons of 30 percent of Th oxides. The probable reserves amount to 1.5 million tons. There exists about 3,000 tons of already milled thorium in a USA strategic stockpile stored in Nevada.
2. Lower nuclear proliferation concerns due to the reduced limited needs for enrichment of the U^{235} isotope that is needed for starting up the fission cycle and can then be later replaced by the bred U^{233}. The fusion fission fusion hybrid totally eliminates that need.
3. A superior system of handling fission products wastes than other nuclear technologies and a much lower production of the long lived transuranic elements as waste. One ton of natural Th^{232}, not requiring enrichment, is needed to power a 1,000 MWe reactor per year compared with about 33 tons of uranium solid fuel to produce the same amount of power. The thorium just needs to be purified then converted into a fluoride. The same initial fuel loading of one ton per year is discharged primarily as fission products to be disposed of for the fission thorium cycle.
4. Ease of separation of the lower volume and short lived fission products for eventual disposal.
5. Higher fuel burnup and fuel utilization than the U^{235}-Pu^{239} cycle.
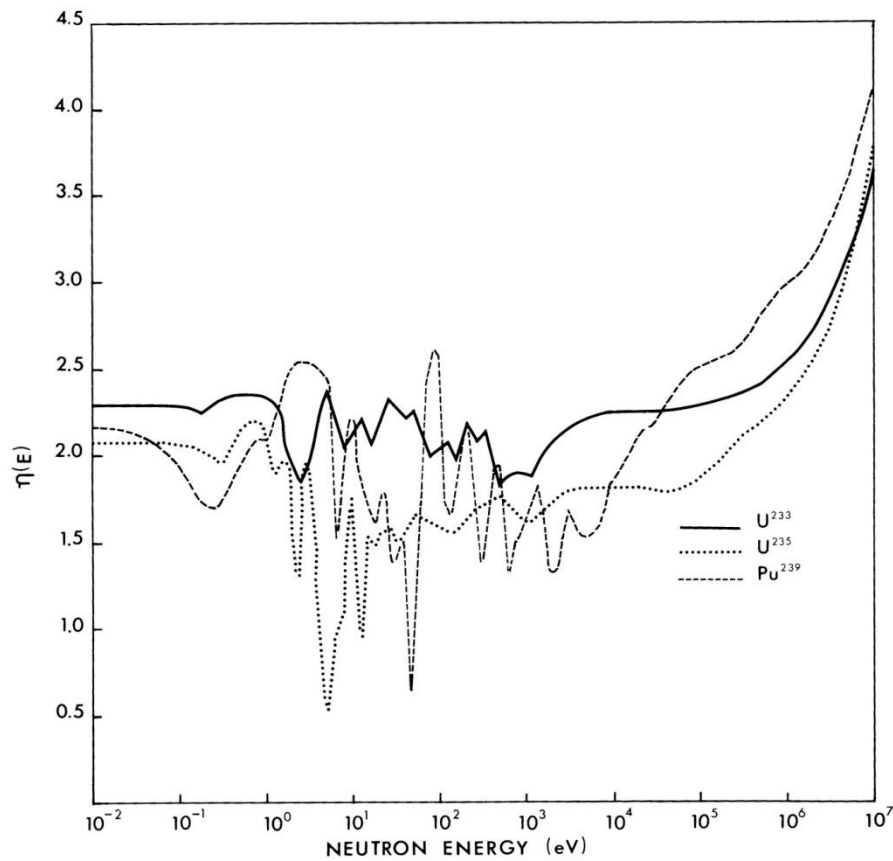
Figure 27. Regeneration factor as a function of neutron energy for the different fissile isotopes. U^{233} breeding is possible in both the thermal and fast neutrons energy ranges.

6. Enhanced nuclear safety associated with better temperature and void reactivity coefficients and lower excess reactivity in the core.

7. With a tailored breeding ratio of unity, a fission thorium fuelled reactor can generate its own fuel, after a small amount of fissile fuel is used as an initial loading.

8. The operation at high temperature implies higher thermal efficiency with a Brayton gas turbine cycle instead of a Joule or Rankine steam cycle, and lower waste heat that can be used for desalination or space heating. If thermal efficiency can be sacrified, an open air cooled cycle can be contemplated eliminating the need for cooling water and the associated heat exchange equipment.

9. A thorium cycle for base-load electrical operation would a perfect match to peak-load cycle wind turbines generation. The produced wind energy can be stored as compressed air which can be used to cool a thorium open cycle reactor, substantially increasing its thermal efficiency, yet not requiring a water supply for cooling.

10. The unit powers are scalable over a wide range for different applications such as process heat or electrical production. Units of 100 MWe each can be designed, built and combined for larger power needs.

11. Operation at atmospheric pressure without pressurization implies the use of standard equipment with a lower cost than the equipment operated at high pressure in the LWRs cycle.

NONPROLIFERATION CHARACTERISTICS

The hard gamma rays associated with the decay chain of the formed isotope U^{232} with a half life of 72 years and its spontaneous fission makes the U^{233} in the thorium cycle with high fuel burnup a higher radiation hazard from the perspective of proliferation than Pu^{239}.

The U^{232} is formed from the fertile Th^{232} from two paths involving an (n, 2n) reaction, which incidentally makes Th^{232} a good neutron multiplier in a fast neutron spectrum:

$$
\begin{aligned}
&_0n^1 + {}_{90}Th^{232} \rightarrow 2\,_0n^1 + {}_{90}Th^{231} \\
&_{90}Th^{231} \xrightarrow{\ 25.52h\ } {}_{-1}e^0 + {}_{91}Pa^{231} \\
&_0n^1 + {}_{91}Pa^{231} \rightarrow \gamma + {}_{91}Pa^{232} \\
&_{91}Pa^{232} \xrightarrow{\ 1.31d\ } {}_{-1}e^0 + {}_{92}U^{232}
\end{aligned}
\tag{8}
$$
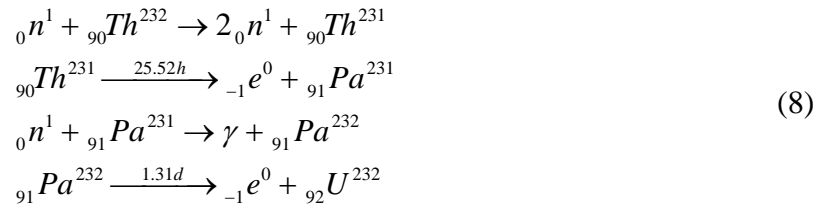
and another involving an (n, γ) capture reaction:

$$
\begin{aligned}
&_0n^1 + {}_{90}Th^{232} \rightarrow \gamma + {}_{90}Th^{233} \\
&_{90}Th^{233} \xrightarrow{\ 22.2m\ } {}_{-1}e^0 + {}_{91}Pa^{233} \\
&_{91}Pa^{233} \xrightarrow{\ 27d\ } {}_{-1}e^0 + {}_{92}U^{233} \\
&_{92}U^{233} + {}_0n^1 \rightarrow 2\,_0n^1 + {}_{92}U^{232}
\end{aligned}
\tag{9}
$$
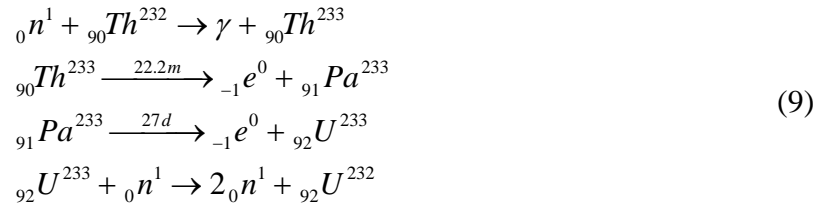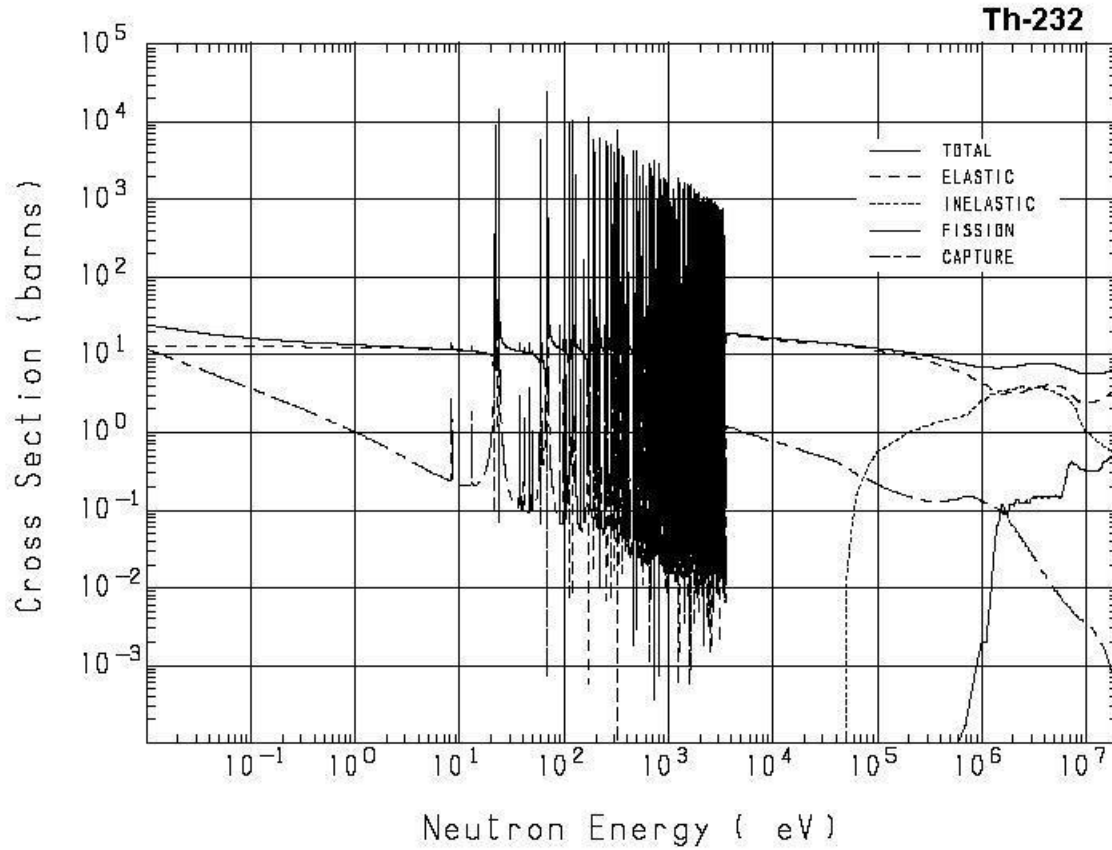
Figure 28. The cross sections of Th^{232} including the (n, γ) neutron radiative capture reaction.
Source: JENDL.

The isotope U^{232} is also formed from a reversible (n, 2n) and (n, γ) path acting on the bred U^{233}:

$$_{0}n^{1} + {}_{92}U^{233} \rightarrow 2\,_{0}n^{1} + {}_{92}U^{232}$$
$$_{0}n^{1} + {}_{92}U^{232} \rightarrow \gamma + {}_{92}U^{233}$$

(10)
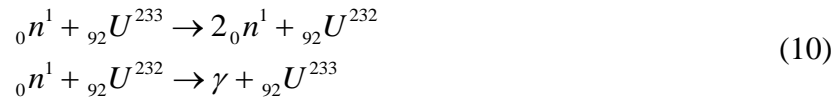
The isotope Th^{230} occurs in trace quantities in thorium ores that are mixtures of uranium and thorium. U^{234} is a decay product of U^{238} and it decays into Th^{230} that becomes mixed with the naturally abundant Th^{232}. It occurs in secular equilibrium in the decay chain of natural uranium at a concentration of 17 ppm. The isotope U^{232} can thus also be produced from two successive neutron captures in Th^{230}:

$$_{0}n^{1} + {}_{90}Th^{230} \rightarrow \gamma + {}_{90}Th^{231}$$

$$_{90}Th^{231} \xrightarrow{\;25.52h\;} {}_{-1}e^{0} + {}_{91}Pa^{231}$$

$$_{0}n^{1} + {}_{91}Pa^{231} \rightarrow \gamma + {}_{91}Pa^{232} \tag{11}$$

$$_{91}Pa^{232} \xrightarrow{\;1.31d\;} {}_{-1}e^{0} + {}_{92}U^{232}$$
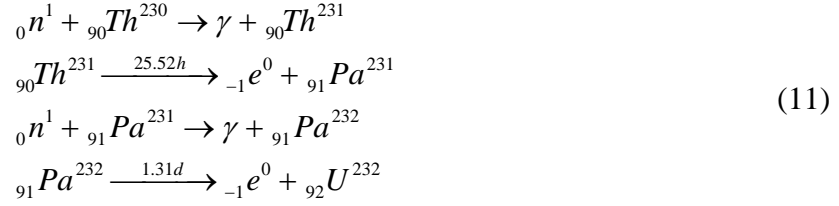
The hard 2.6 MeV gamma rays originate from Tl^{208} isotope in the decay chain of aged U^{232} which eventually decays into the stable Pb^{208} isotope:

$$_{92}U^{232} \xrightarrow{\;72a\;} {}_{90}Th^{228} + {}_{2}He^{4}$$

$$_{90}Th^{228} \xrightarrow{\;1.913a\;} {}_{88}Ra^{224} + {}_{2}He^{4}$$

$$_{88}Ra^{224} \xrightarrow{\;3.66d\;} {}_{86}Rn^{220} + {}_{2}He^{4}$$

$$_{86}Rn^{220} \xrightarrow{\;55.6s\;} {}_{82}Po^{216} + {}_{2}He^{4}$$

$$_{84}Po^{216} \xrightarrow{\;0.15s\;} {}_{82}Pb^{212} + {}_{2}He^{4}$$

$$_{82}Pb^{212} \xrightarrow{\;10.64h\;} {}_{83}Bi^{212} + {}_{-1}e^{0} \tag{12}$$

$$_{83}Bi^{212} \xrightarrow[64\%]{\;60.6m\;} {}_{84}Po^{212} + {}_{-1}e^{0}$$

$$_{83}Bi^{212} \xrightarrow[36\%]{\;60.6m\;} {}_{81}Tl^{208} + {}_{2}He^{4}$$

$$_{84}Po^{212} \xrightarrow{\;0.298\mu s\;} {}_{82}Pb^{208}(stable) + {}_{2}He^{4}$$

$$_{81}Tl^{208} \xrightarrow{\;3.053m\;} {}_{82}Pb^{208}(stable) + {}_{-1}e^{0} + \gamma(2.6146MeV)$$
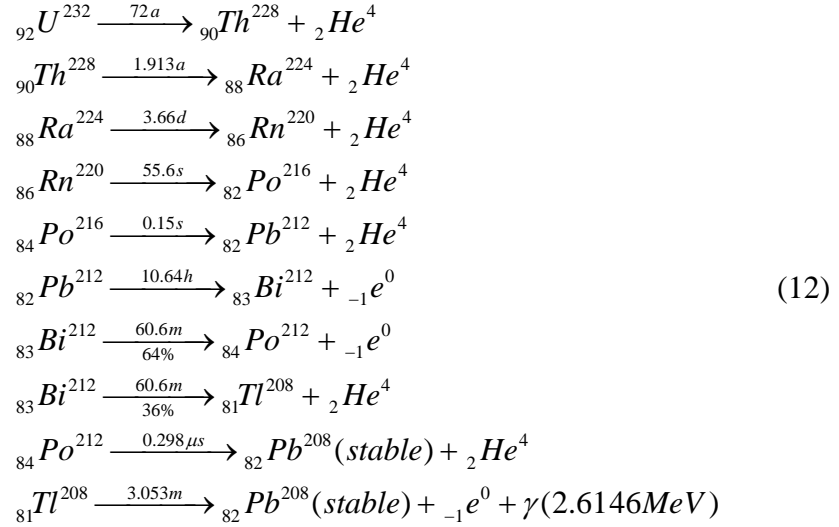
For comparison, the U^{233} decay chain eventually decays into the stable Bi^{209} isotope:

$$_{92}U^{233} \xrightarrow{\;1.592 x10^{5}a\;} {}_{90}Th^{229} + {}_{2}He^{4}$$

$$_{90}Th^{229} \xrightarrow{\;7340a\;} {}_{88}Ra^{225} + {}_{2}He^{4}$$

$$_{88}Ra^{225} \xrightarrow{\;14.8d\;} {}_{89}Ac^{225} + {}_{-1}e^{0}$$

$$_{89}Ac^{225} \xrightarrow{\;10.0d\;} {}_{87}Fr^{221} + {}_{2}He^{4}$$

$$_{87}Fr^{221} \xrightarrow{\;4.8m\;} {}_{85}At^{217} + {}_{2}He^{4} \tag{13}$$

$$_{85}At^{217} \xrightarrow{\;32.3ms\;} {}_{83}Bi^{213} + {}_{2}He^{4}$$

$$_{83}Bi^{213} \xrightarrow{\;45.6m\;} {}_{84}Po^{213} + {}_{-1}e^{0}$$

$$_{84}Po^{213} \xrightarrow{\;4.2\mu s\;} {}_{82}Pb^{209} + {}_{2}He^{4}$$

$$_{82}Pb^{209} \xrightarrow{\;3.28h\;} {}_{83}Bi^{209}(stable) + {}_{-1}e^{0}$$
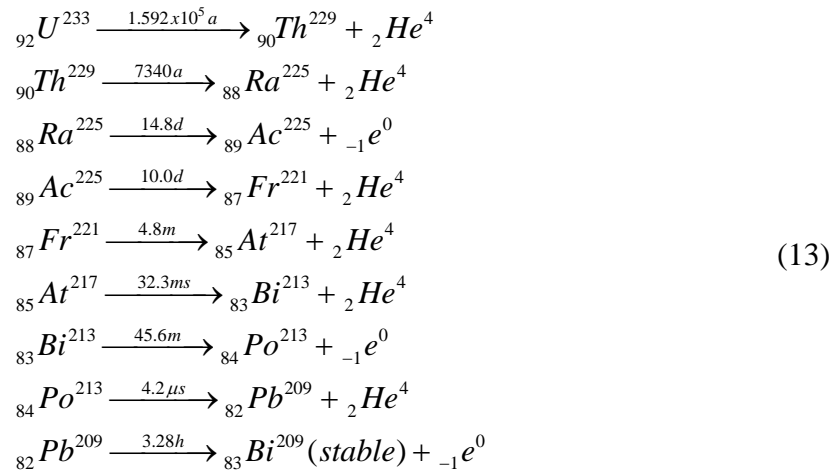
A 5-10 proportion of U^{232} in the U^{232}-U^{233} mixture has a radiation equivalent dose rate of about 1,000 cSv (rem)/hr at a 1 meter distance for decades making it a highly proliferation resistant cycle if the Pa^{233} is not separately extracted and allowed to decay into pure U^{233}.

The Pa^{233} cannot be chemically separated from the U^{232} if the design forces the fuel to be exposed to the neutron flux without a separate blanket region, making the design failsafe with respect to proliferation and if a breeding ratio of unity is incorporated in the design.

Such high radiation exposures would lead to incapacitation within 1-2 hours and death within 1-2 days of any potential proliferators.

The International Atomic Energy Agency (IAEA) criterion for fuel self protection is a lower dose equivalent rate of 100 cSv(rem)/hr at a 1 meter distance. Its denaturing requirement for U^{235} is 20 percent, for U^{233} with U^{238} it is 12 percent, and for U^{233} denaturing with U^{232} it is 1 percent.

The Indian Department of Atomic Energy (DAE) had plans on cleaning U^{233} down to a few ppm of U^{232} using Laser Isotopic Separation (LIS) to reduce the dose to occupational workers.

The contamination of U^{233} by the U^{232} isotope is mirrored by another introduced problem from the generation of U^{232} in the recycling of Th^{232} due to the presence of the highly radioactive Th^{228} from the decay chain of U^{232}.

CONVERSION, BREEDING AND NEUTRON MULTIPLICATION

The Th^{232}-U^{233} fuel cycle in fission reactors can have a conversion ratio of larger than unity and become a net breeder of fissile material in thermal spectra if the use of burnable neutron poisons to control long term reactivity is minimized by the use of continuous fueling in a liquid salt or a slurry or the use a geometrical control of the reactivity by the control rods.

The threshold for the (n, 2n) reaction at 6 MeV and the (n, 3n) reaction would be more significant in the high energy neutron spectrum of the fusion-fission hybrid than in the case of a thermal neutron spectrum and needs to be quantified.

In addition, in the hard neutron spectrum, fissions will also occur in Th^{232} adding more neutrons to be available for fusile as well as fissile breeding.

Table 6. Averaged neutron cross sections for Th^{232}. Source: JENDL-3.3.

Reaction	Thermal 0.0253 eV average	Maxwellian. average at 0.025 eV	g-factor	Resonance Integral	14-MeV spectrum average	Fission Spectrum average
total	20.12	19.26	1.080	-	5.740	7.633
elastic	12.72	12.71	1.128	-	2.681	4.579
inelastic	$E_{threshold} = 4.921 \times 10^4$ eV				726.0×10^{-3}	2.875
(n,2n)	$E_{threshold} = 6.465 \times 10^6$ eV				1.181	14.46×10^{-3}
(n,3n)	$E_{threshold} = 1.161 \times 10^7$ eV				800.0×10^{-3}	113.6×10^{-6}
fission	53.68×10^{-6}	47.65×10^{-6}	1.002	635.9×10^{-3}	350.0×10^{-3}	78.32×10^{-3}

capture	7.400	6.545	0.998	84.94	1.920×10^{-3}	85.93×10^{-3}

The fission spectrum average cross section is 14.46 mb, and is a much larger value of 1.181 b for a fusion 14 Mev spectrum for the (n, 2n) reaction:

$$_{0}n^{1} + {}_{92}U^{233} \rightarrow 2\,_{0}n^{1} + {}_{92}U^{232} \tag{14}$$

and it is 4.08 mb for the (n, γ) reaction:

$$_{0}n^{1} + {}_{90}Th^{232} \rightarrow \gamma + {}_{90}Th^{233} \tag{15}$$

The U^{232}/U^{233} ratio will depend on the fraction of neutrons above the 6.465 MeV neutron energy threshold. In a fusion spectrum this fraction would be large enough to produce fuel that is unusable for weapons manufacture.
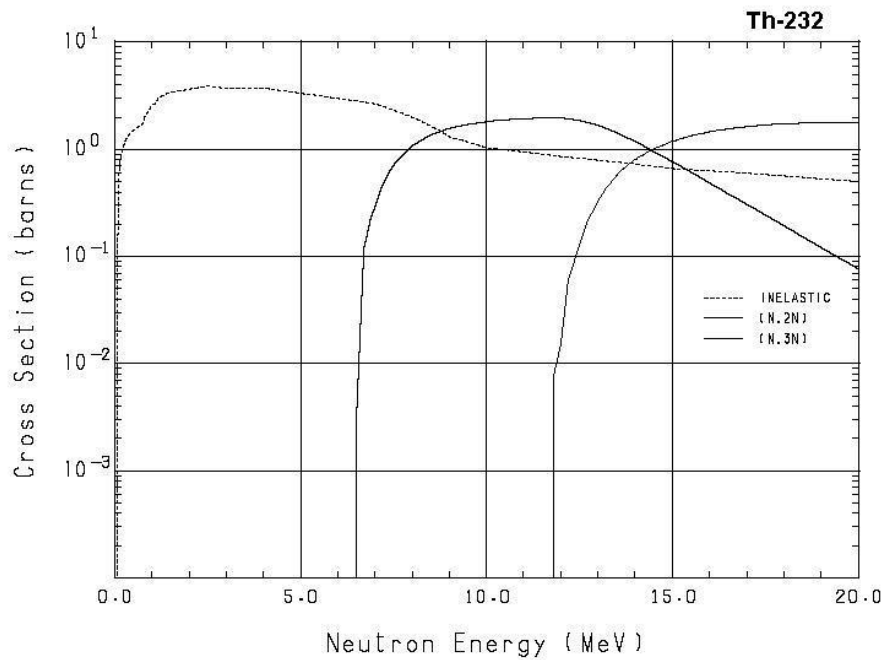

Figure 29. The cross section distribution for the (n, 2n) and (n, 3n) neutron multiplication reactions in Th^{232} showing energy thresholds at 6.465 and 11.61 MeV. Source: JENDL.
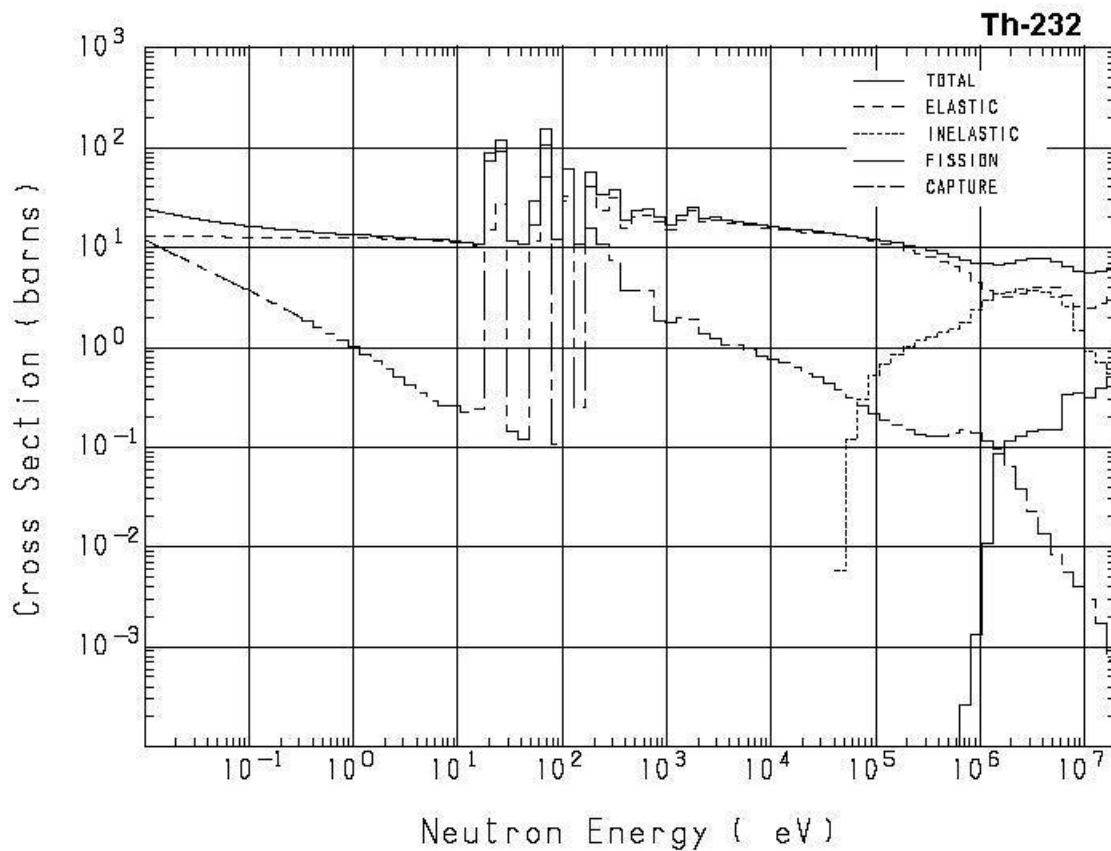
Figure 30. The cross section for the (n, f) fission cross reaction for the fissionable isotope Th^{232}.
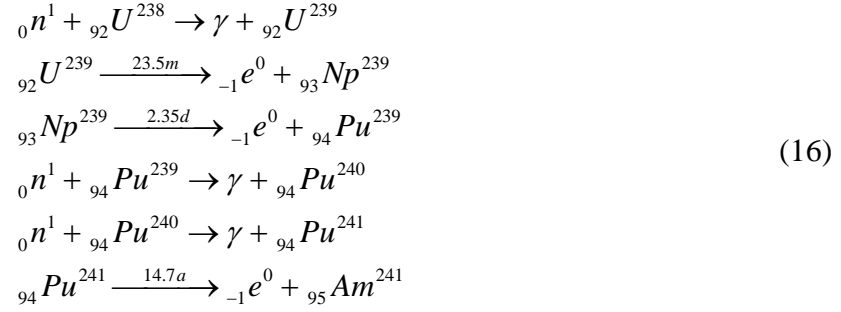Source: JENDL.

DOSIMETRY

The International Atomic Energy Agency (IAEA) criterion for occupational protection is an effective dose of 100 cSv (rem)/hr at a 1 meter distance from the radiation source.

It is the decay product Tl^{208} in the decay chain of U^{232} and not U^{232} itself that generates hard gamma rays. The Tl^{208} would appear in aged U^{233} over time after separation emitting a hard 2.6416 MeV gamma ray photon. It accounts for 85 percent of the total effective dose 2 years after separation. This implies that manufacturing of U^{233} should be undertaken in freshly purified U^{233}. Aged U^{233} would require heavy shielding against gamma radiation.

In comparison, Pu^{239} containing Pu^{241} with a half life of 14.4 years, the most important source of gamma ray radiation is from the Am^{241} isotope with a 433 years half life that emits low energy gamma rays of less than 0.1 MeV in energy. For weapons grade Pu^{239} with about 0.36 percent Pu^{241} this does not present a major hazard but the radiological hazard becomes significant for reactor grade Pu^{239} containing about 9-10 percent Pu^{241}.
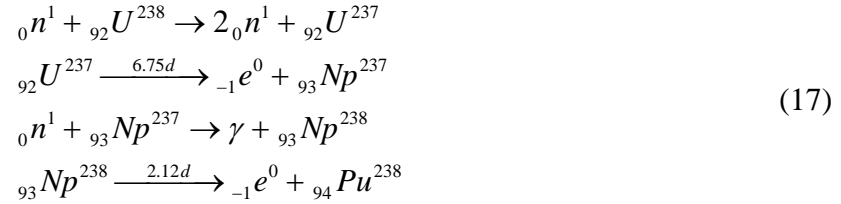
The generation of Pu^{241} as well as Pu^{240} and Am^{241} from U^{238} follows the following path:

$$
\begin{aligned}
&{}_{0}n^{1} + {}_{92}U^{238} \rightarrow \gamma + {}_{92}U^{239} \\
&{}_{92}U^{239} \xrightarrow{\ 23.5m\ } {}_{-1}e^{0} + {}_{93}Np^{239} \\
&{}_{93}Np^{239} \xrightarrow{\ 2.35d\ } {}_{-1}e^{0} + {}_{94}Pu^{239} \\
&{}_{0}n^{1} + {}_{94}Pu^{239} \rightarrow \gamma + {}_{94}Pu^{240} \\
&{}_{0}n^{1} + {}_{94}Pu^{240} \rightarrow \gamma + {}_{94}Pu^{241} \\
&{}_{94}Pu^{241} \xrightarrow{\ 14.7a\ } {}_{-1}e^{0} + {}_{95}Am^{241}
\end{aligned}
\tag{16}
$$

Plutonium containing less than 6 percent Pu^{240} is considered as weapons-grade.

The gamma rays from Am^{241} are easily shielded against with Pb shielding. Shielding against the neutrons from the spontaneous fissions in the even numbered Pu^{238} and Pu^{240} isotopes accumulated in reactor grade plutonium requires the additional use of a thick layer of a neutron moderator containing hydrogen such as paraffin or plastic, followed by a layer of neutron absorbing material and then additional shielding against the gamma rays generated from the neutron captures.

The generation of Pu^{238} and Np^{237} by way of (n, 2n) rather than (n, γ) reactions follows the path:

$$
\begin{aligned}
&{}_{0}n^{1} + {}_{92}U^{238} \rightarrow 2\,{}_{0}n^{1} + {}_{92}U^{237} \\
&{}_{92}U^{237} \xrightarrow{\ 6.75d\ } {}_{-1}e^{0} + {}_{93}Np^{237} \\
&{}_{0}n^{1} + {}_{93}Np^{237} \rightarrow \gamma + {}_{93}Np^{238} \\
&{}_{93}Np^{238} \xrightarrow{\ 2.12d\ } {}_{-1}e^{0} + {}_{94}Pu^{238}
\end{aligned}
\tag{17}
$$

The production of Pu^{238} for radioisotopic sources for space applications follows the path of chemically separating Np^{237} from spent LWR fuel and then neutron irradiating it to produce Pu^{238}.

Table 7. Typical compositions of fuels in the uranium and thorium fuel cycles.

Isotopic composition [percent]	Pu^{239} weapons grade	Pu^{239} reactors grade	U^{233}	U^{233} + 1 ppm U^{232}
U^{232}			0.0000	0.0001
U^{233}			100.0000	99.9999
Pu^{238}	0.0100	1.3000		
Pu^{239}	93.8000	60.3000		
Pu^{240}	5.8000	24.3000		
Pu^{241}	0.3500	9.1000		

Pu^{242}	0.0200	5.0000		
Density [gm/cm^3]	19.86	19.86	19.05	19.05
Radius [cm]	3.92	3.92	3.96	3.96
Weight [kg]	5	5	5	5

Table 8. Glove box operation dose rate required to accumulate a limiting occupational 5 cSv (rem) dose equivalent from a 5 kg metal sphere, one year after separation at a 1/2 meter distance [11].

Fuel, U^{232}/U^{233}	Time to 5 cSv effective dose [hr]	Effective dose rate cSv/hr
0.01	0.039	127.0000
100 ppm	3.937	1.2700
5 ppm	84.746	0.0590
1 ppm	384.615	0.0130
Reactor grade Pu^{239}	609.756	0.0082
Weapons grade Pu^{239}	3846.154	0.0013

Both reactor-grade plutonium and U^{233} with U^{232} would pose a significant radiation dose equivalent hazard for manufacturing personnel as well as military personnel, which precludes their use in weapons manufacture in favor of U^{235} and weapons-grade Pu^{239}.

Table 9. Dose equivalent rates in cSv (rem)/hr from 5 kg metal spheres at a 1/2 meter distance for different times after separation [11].

Material	Type of radiation	Dose equivalent rate at time after separation [cSv(rem)/hr]				
		0 yr	1 yr	5 yr	10 yr	15 yr
Pure U^{233}	γ total	0.32	0.42	0.84	1.35	1.89
U^{233} +1 ppm U^{232}	γ total	0.32	13.08	35.10	39.57	39.17
	γ from Tl^{208}	*0.00*	*11.12*	*29.96*	*33.48*	*32.64*
Pu^{239}, weapons grade	γ	0.49	0.71	1.16	1.57	1.84
	neutrons	0.56	0.56	0.56	0.56	0.56
	γ + neutron	1.05	1.27	1.72	2.13	2.40
Pu^{239}, Reactor grade	γ total	0.49	5.54	16.72	28.64	37.54
	γ from	*0.00*	*3.24*	*14.60*	*26.00*	*34.80*

	Am^{241}					
	neutrons	2.66	2.66	2.65	2.64	2.63
	γ + neutrons	3.15	8.20	19.37	31.28	40.17

SAFETY FEATURES

The molten salt has a substantial negative temperature coefficient of reactivity. As the temperature increases, the coolant's density decreases, and the power level subsequently decreases, leading to a decrease in temperature.

A freeze plug in a drain at the reactor vessel would be continuously cooled. If the cooling were stopped; caused for instance by a power failure, the plug would melt draining the entire molten salt through gravity to storage tanks shutting down the chain reaction.

DISCUSSION

The use of the thorium cycle in a fusion fission hybrid could bypass the stage of fourth generation breeder reactors in that the energy multiplication in the fission part allows the satisfaction of energy breakeven and the Lawson condition in magnetic and inertial fusion reactor designs. This allows for the incremental development of the technology for the eventual introduction of a pure fusion technology.

There is a need to start the research on vessels, piping and pumps made of ceramics or carbon aiming at operation at 900 ºC or higher. A 5-10 MWth test reactor would be built to test the materials to be later scaled to commercial size. Within the current state of knowledge, plants can be readily built for operation at around 650 ºC.

Such an alternative sustainable paradigm or architecture would provide the possibility of a well optimized fusion-fission thorium hybrid for sustainable long term fuel availability with the added advantages of higher temperatures thermal efficiency for process heat production, proliferation resistance and minimized waste disposal characteristics.

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