Fuel Cycle for a Uranium-Plutonium Thermal Breeder Reactor

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INTRODUCTION

Background

The generally accepted idea that there aren't enough neutrons for a uranium/plutonium thermal reactor to be a breeder is based on the assumption that few of the emitted fast neutrons will cause fission of U-238 even though most of them have enough energy to do so. The discussion below in the second subsection (Theoretical Basis) shows that with some physical arrangements of the fuel there may be enough neutrons because of an increased rate of fast fission of U-238.

Because the proposed reactor design is novel and unfamiliar, the second section (The Proposed Reactor Design) is devoted to it. In the proposed reactor three design features are used to enable the reactor to be a uranium/plutonium breeder:
1. The fuel is in large tubes so that enough of the fast neutrons cause fission of U-238 before leaving the fuel and passing into the moderator,
2. Neutron loss is minimized by using heavy water as the moderator and by controlling the reactor by moderator displacement as described in the third subsection (Reactor control),
3. The fuel is molten uranium metal, uranium alloy, or uranium salt so the reactor is cooled by bulk transport of the fuel.

The resulting reactor has many advantages:
1. Its fuel can be used LWR fuel with minimal processing,
2. Very long fuel life (30 to 100 years),
3. Roughly 90% reduction in waste.
4. Effective load following is possible.

Theoretical Basis

It is generally understood [1] that a uranium fueled (U-238 + U-235) thermal nuclear reactor cannot be a breeder because there aren't enough neutrons. Each thermal fission of U-235 releases around 2.435 neutrons but requires around 1.17 thermal neutrons be absorbed in U-235 because only about 84% of thermal neutrons absorbed in U-235 cause fission, the remainder convert the U-235 to U-236. To have a conversion ratio of at least 1.000 requires that an additional 1.17 neutrons be absorbed in U-238 to produce new Pu-239 to replace the U-235 lost. This leaves only about .09 extra neutrons left of the 2.435 neutrons released per fission; but the neutrons absorbed by xenon, samarium, and other fission products and all the rest of the reactor hardware are more than this.

This analysis ignores the probability that some of the fast neutrons generated will cause fast fission of U-238 because in existing thermal reactor designs the probability of this is low. However, as the smallest dimension of the fuel bodies increases towards the mean free path of fast neutrons in U-238, the probability of fast fission of U-238 increases. Each fast fission of U-238 yields about 2.82 nuetrons but only costs 1 neutron becuse the U-238 does not need to be replaced to have a breeder. This provides about 1.82 extra neutrons for each fast fission of U-238 so that if 15 or 20 percent of all fissions which occur are fast fissions of U-238 there may be enough extra neutrons to have a breeder. This theoretical analysis provides a conceptual framework, MCNP [2] analysis can provide data on the actual effect with different fuel body sizes.

The simplest theoretical thermal reactor consists of fuel in moderator. To explore the theoretical limits, MCNP simulations have been done with a hypothetical reactor consisting of a hexagonal prism of heavy water moderator with a hexagonal array of uranium fuel cylinders embedded in it. The moderator prism modeled is not infinite but large at 20 meters high and about 20 meters wide. The fuel cylinders are slightly under 20 meters long. The parameters which are varied in the model are the radius of the fuel cylinders, the pitch (center to center spacing of the fuel cylinders in the moderator), and the percentage of the uranium which is U-235, the remainder is U-238. The primary variable is the fuel cylinder radius. For each radius tried the pitch is varied to find the pitch which provides the maximum multiplication factor. Then the fuel enrichment is varied to find the minimum enrichment which gives a multiplication factor of 1.0000.

Radius cm	Pitch cm	U-235 %	% fast fission	neutrons gained
.25	6	.44	1.2	21
2	26	.42	6.9	119
5	38	.46	13	224
8	47	.55	16	275
14	64	.75	20	344
20	75	.95	22	378

TABLE I. MCNP fast fission % vs fuel radius

Table I suggests that a physical uranium/plutonium thermal breeder reactor could be built. The last column shows the number of neutrons gained per 1000 fissions compared to 0% fast fissions of U-238. Because the theoretical model doesn't take into account the neutron losses in the reactor hardware and the fission products of an actual reactor, the fuel enrichment would have to be higher in an actual reactor than the theoretical minimums shown in the table.

This paper describes a design for a thermal reactor, the moderator to fuel volume ratio is over 4 to one and most neutrons which escape from the fuel are well moderated, most fission of fissile isotopes occurs at thermal or epi-thermal energies. The neutron gain needed to have a breeder comes from some fast fissions of U-238 (with emitted fast neutrons before they ever leave the fuel body) which works because the U-238 which fissions does not need to be replaced to have a breeder. Compared to a thermal fission of U-235 which produces around 0.09 extra neutrons, a fast fission of U-238 produces around 1.82 extra neutrons or over 18 times as many. Only around 20% fast fissions of U-238 are required to have enough extra neutrons to have a breeder.

This approach is completely different from that used in the Hitachi-RBWR [3] where neutrons which leave the fuel are very poorly moderated and most fission of fissile isotopes occurs at higher neutron energies which release more neutrons per fission. The moderator to fuel volume ratio is around 0.5, and few fissions are thermal. The neutron gain needed to have a breeder comes from most fissions of thermal isotopes being at higher energies which release somewhat more neutrons per fission, perhaps 4 or 5 times as many extra neutrons as with thermal fissions. This means that almost all fissions of fissile isotopes in the RBWR must be at high energies to have enough extra neutrons for a breeder.

Control Method

To achieve a conversion ratio greater than 1.0 in a thermal nuclear reactor requires a reactor design with very low neutron losses. The proposed design [4] is heavy water moderated and uses molten uranium metal for both the fuel and the coolant (To reduce the high temperature of the molten fuel, the fuel might be a uranium salt or a uranium alloy). Most of the energy is removed by the flow of the molten uranium fuel, the same as in a molten salt reactor. Because the heat energy is removed by circulating the molten fuel to a heat exchanger, the fuel temperature does not need to increase with increasing reactor power, the fuel flow rate can be increased as the power is increased to keep the maximum fuel temperature constant. Some of the energy produced heats the moderator, both by the slowing of fast neutrons in the moderator and by thermal conduction of heat energy from the hot fuel to the cooler moderator. The proposed reactor is controlled by displacement of moderator from the core to the reflector [5] & [6]. The displacement of the moderator is by self generated moderator steam with negative feedback. The moderator cooling system and the reactor control system are integrated into a single system.

Most of the heavy water moderator in the core is contained in inverted control cavities which are closed at the top and open at the bottom. Each cavity is cooled by a flow of cooler heavy water into the cavity and each cavity has a bubble of heavy water steam trapped at the top of the cavity. If the energy deposited in the control cavity by the fast neutrons is greater than the heat energy removed by the flow of cooler moderator the excess energy boils some of the moderator and increases the size of the steam bubble. This displaces liquid moderator out of the cavity into the reflector and decreases the multiplication factor, reducing the reactor power. The reverse happens when less energy is deposited in the cavity by the fast neutrons than is removed by the cool moderator flow, steam condenses and liquid moderator moves in from the reflector to replace it. There is a size of the steam bubble which makes the multiplication factor equal to one, this size is independent of the reactor power but varies slowly over time as the reactivity of the fuel changes with burnout.

Because a small amount of heavy water boiled generates a large volume of steam and displaces a large volume of heavy water out of the reactor core the control method is very sensitive. It is also very fast because the Control cavity steam bubble size (equivalent to void fraction in a conventional reactor) starts changing as soon as the fast neutrons emitted by the fuel reach the moderator in less than a microsecond, much faster than in a conventional reactor where an increase in the fission rate causes the fuel temperature to rise after some delay which then increases the rate of heat transfer to the cladding which will eventually increase the temperature of the cladding and increase the void fraction after a delay of a few milliseconds to several seconds. The strong negative feedback results in the total reactor power following the rate of the moderator coolant flow, the reactor power is increased or decreased by increasing or decreasing the moderator coolant flow. This allows for practical load following. A general power failure or moderator coolant pump failure will stop the moderator coolant flow and cause the reactor to quickly shut down.

This control method is simple and safer than traditional control methods. It has the additional advantages of a very wide control range of around 300 mk, effective load following, and low neutron loss, all excess neutrons are less well moderated and are absorbed by U-238 in resonance capture, increasing the conversion ratio.

THE PROPOSED REACTOR DESIGN

Full Sized Reactor Diagrams

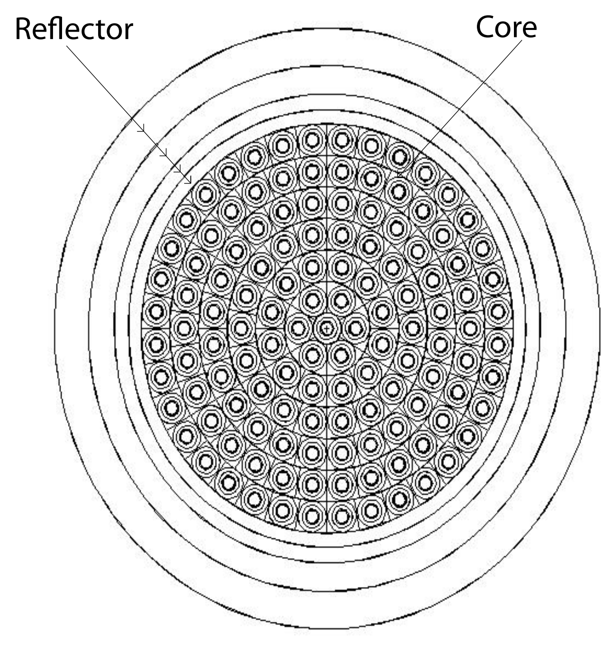

Fig. 1. MCNP model top view cross section.

Fig. 1 and Fig. 2 show the proposed reactor core and reflector regions in a top view and a side view. They are generated by the MCNPX visual editor and are accurate but

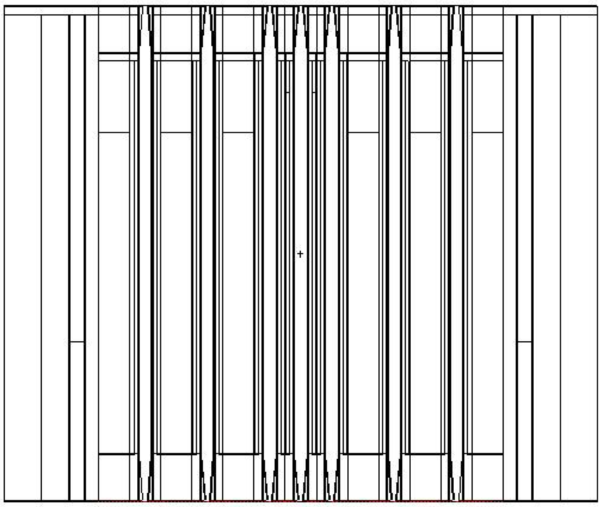

Fig. 2. MCNP model side view cross section.

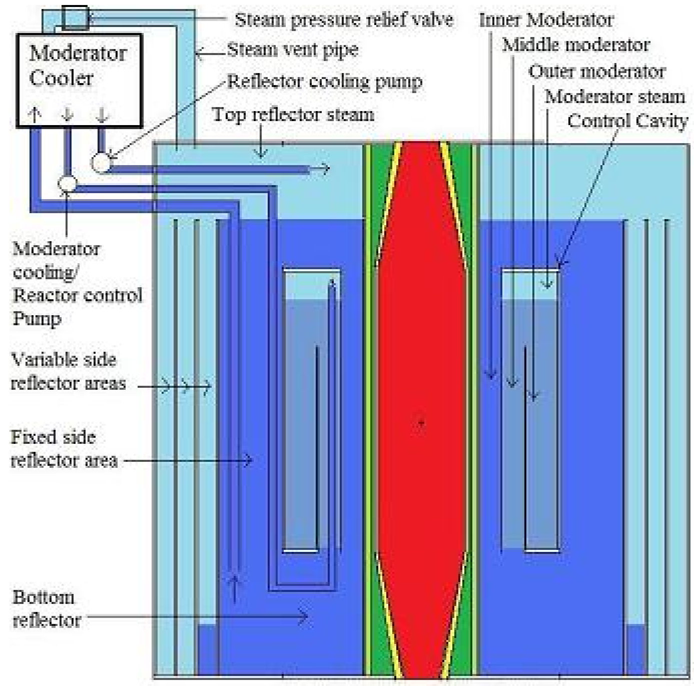

Fig. 3. One tube reactor with moderator cooling system

the details are so small that they are hard to understand. The reactor has 127 vertical fuel tubes with a central tube surrounded by six concentric rings of fuel tubes. Each fuel tube is surrounded by a moderator control cavity which is filled with liquid heavy water moderator and moderator steam. The moderator steam is generated by the energy deposited in the moderator by the fast neutrons released by fission in the fuel. Each control cavity is closed except at the bottom so liquid moderator is free to move between the cavity and the reflector area. As can be seen in the top view (1) the x axis passes between the fuel tubes in rings 2, 4, and six of the fuel tubes so the side view (2) does not show the fuel tubes and control cavities in the even numbered rings of fuel tubes.

The reflector area at the sides of the reactor is composed of four concentric rings. The inner ring is fixed and is always full of liquid moderator. The three outer rings are variable, as liquid moderator is displaced from the core it overflows the fixed reflector ring into the inner variable reflector ring, then the next, and finally the last. These details are easier to see in Fig. 3

Cartoon Reactor Diagram

Fig. 3 shows a simplified reactor with only one fuel tube. In this figure the vertical scale appears much too small and the width of the reflector appears much too large. In Fig. 2 many MCNP surfaces are shown which are not physical, those have been deleted in Fig. 3 for clarity and the control system plumbing and the colors added. Starting in the middle of Fig. 3 and going to the side the diagram shows the fuel/coolant (red, uranium). Next is the fuel tube (yellow, zirconium carbide), next is a layered structure to reduce heat flow (green), then the moderator wall (black, zircaloy), then the inner moderator region (Blue, heavy water), a partition wall (black, zircaloy), the middle moderator and partition wall, the outer moderator, then the ring divider wall (black, zircaloy). The liquid moderator in the control cavities is shown in a lighter shade of blue. All of the preceding are repeated for each ring of fuel tubes in the actual reactor, then there is the fixed side reflector (blue, heavy water), a partition wall (black, zircalloy), the inner variable side reflector and another partition wall, the middle variable side reflector and partition, and the outer variable side reflector followed by the outside containment wall. The light blue areas are moderator steam. The materials listed above are suggested, not final.

The moderator/reflector cooling and control system consists of a moderator cooler and cold moderator tank, a cooler input tube which runs up from low in the fixed reflector area to the steam area above the reflector then horizontally to the cooler located outside the reactor confinement wall. The return flow is by two pipes each with a controlled pump. One pump sprays cool moderator into the steam area above the top reflector. It is controlled by a pressure sensor and keeps the pressure in the reflector area constant so that small changes of pressure in the control cavities will cause liquid moderator to move between the control cavities and the reflector. The other pump sprays cool moderator into the steam bubbles at the top of the control cavities. It is controlled so its flow rate is proportional to the desired reactor power. As a backup system there is also a steam pipe from the steam area above the top reflector to a pressure relief valve and then to the moderator cooler. This keeps the reflector pressure constant even without electrical power for the pumps.

It is to be noted that there is no physical barrier between the inner moderator, the middle moderator, the bottom reflector, the fixed side reflector, and the top reflector. The heavy water is free to move between those areas. The amount of moderator steam in the control cavities (the middle, outer, and top moderator areas, shown in a lighter shade of blue) is maintained by negative feedback to keep keff equal to 1.000. Over the initial years of the fuel life the reactivity of the fuel slowly increases because the conversion ratio is greater than one and the trapped steam bubble in the control cavity slowly expands displacing liquid moderator out into the reflector where it spills over from the fixed side reflector into the inner variable side reflector, then over into the middle variable side reflector, and finally into the outer variable side reflector. Later in the fuel life as the concentration of neutron absorbing fission products builds up the fuel reactivity will start to slowly decrease and the liquid moderator will flow back from the reflector area into the control cavities middle moderator area, then spill over into the control cavities outer moderator area, and finally fill up the control cavities top moderator area.

The side reflector variable inner, middle, and outer areas and the control cavities outer moderator areas have no passage for liquid moderator to leave them. Any liquid moderator in those areas is heated by the fast neutron flux and by thermal conduction and is cooled by evaporation, so the liquid will all eventually boil away unless it is replenished by liquid spilling over the side wall.

The MCNP Model

The proposed design is modeled with MCNP to evaluate various design options (this MCNP model is different from the MCNP model mentioned above in the introduction for a simple hypothetical reactor, this model includes all the reactor hardware and materials in the reactor core and reflector). The MCNP input files are generated by a shell program based on a design pattern and driven by parameters. The most important parameters are the radius of the fuel tubes, the pitch (center to center spacing between the fuel tubes), the number of rings of fuel tubes, the isotopic composition of the fuel, and the height of the fuel tubes relative to the diameter of the reactor core. Other parameters describe the composition and temperature of all materials in the reactor, and the thickness of various materials. The shell program generates the MCNP input file, executes MCNP, and reads the MCNP output file and writes the most important results into a log file. The shell program can be run iteratively over a range of values for one or two parameters to explore systematic variations in the design.

The MCNP model is used to validate the basic reactor design, evaluate the effects of changes to various parameter values, and to generate accurate diagrams of the reactor using the MCNPX visual editor. Since this reactor design has a number of engineering and materials challenges it is important to be able to evaluate the effects of any engineering or material choices. This will suggest tradeoffs which can be made in the design to achieve the desired features.

FUEL CYCLE

Initial Fuel

The initial fuel for the proposed reactor requires a minimum fissile content of around 1.4%. This fuel can come from any one of or some combination of three sources:

1. Low enriched uranium,
2. Used LWR fuel,
3. Used fuel from a prior reactor of this type, possibly diluted with depleted or natural uranium.

The fuel mass in this reactor is much higher than in LWR or CANDU reactors. Because it is liquid fueled and cooled, around half of the fuel will be outside the core in the primary heat exchanger or in transit. In addition while the pitch increases as the fuel tube radius increases it increases proportionately less so that as the fuel tubes become large the percentage of the core occupied by fuel increases. The total required fuel may be around 1000 tonnes for a 1000 MWe reactor. In the case where the fuel is used LWR fuel it must be separated from the cladding and reduced from uranium oxide to metal. However no other separation is required, the fissile content is high enough to compensate for the reactivity reduction caused by the existing fission products.

In Core Fuel Evolution

During operation of the reactor the composition of the fuel changes slowly. During the initial period of operation U-235 is depleted and replaced with Pu-239. The Pu-239 has a higher fission cross section than U-235 so this replacement will increase the reactivity of the fuel. If the conversion ratio is greater than 1.000 this also increases fissile content percentage and the reactivity of the fuel. The accumulation of neutron absorbing fission products will decrease the reactivity of the fuel. These three trends occur at different rates at different points in the fuel life, so the fuel reactivity may be quite different at various times in the reactor life.

The control method compensates for increased fuel reactivity by under-moderating excess neutrons, causing them to undergo resonance capture in the U-238 and increasing the conversion ratio. This is in contrast to the usual reactor control by absorbing the excess neutrons which decreases the conversion ratio. Since the difference in reactivity between a reactor core which is full of moderator and one which has no moderator is very large (around 300 milli-k), the control method can properly control the reactor through wide reactivity changes over the course of the fuel life.

The total fission cross section of the fuel fissile content may be increasing because of the first two trends mentioned above. This means that to have the same thermal fission rate the thermal neutron flux is decreasing. With reduced thermal neutron flux the number of neutrons lost to absorption in the fission products and the reactor hardware may also decrease so the conversion ratio may increase with fuel burnout at some points in the fuel life.

Because the fuel is continuously circulating it is always well mixed, there are no hot spots or areas of uneven burnout. Fission products which are vapor at high temperature and low pressure (notably xenon and its precursor iodine) will naturally separate from the fuel above the heat exchanger and can be captured and removed from the fuel flow. Some other isotopes may be insoluble in the fuel and float as dross above the heat exchanger where they may also be captured and removed from the circulating fuel. The removal of some fission products from the circulating fuel may result in a significant decrease in the eventual neutron losses due to the buildup of neutron absorbing fission products in the circulating fuel. The geometry which causes a significant rate of fast fission of U-238 will also increase the rate of fast fission of transuranic isotopes in the circulating fuel.

Because of the high temperature of the molten fuel, high efficiency Air-Brayton cycle turbine generators may be used. Assuming an efficiency of 40% a 1000 MWe plant would fission about 2.5 kg of fuel per day or around 1 tonne per year. Since the fuel fissile content is not decreasing, a fuel burnout of only 35,000 grams per tonne would represent 35 years of fuel life. The actual achievable fuel life might be much greater than that depending on what percentage of the circulating fission products can be removed.

Post Core Handling

When the fuel is removed from the reactor it is mostly U-238. The fissile content will be around 1.5 to 2.5 % depending on the achieved burnout, conversion ratio, and starting fuel. Most of that will be Pu-239 with some Pu-241 and even less U-235. Remaining fission products will be 2 to 5 % depending on the effectiveness of in-line fission product removal and the achieved burnout. There may be 1 to 2 % of other fissionable isotopes (besides U-238), mostly Pu-240, Pu-242, U-236, and other transuranic isotopes.

The fuel can be reused with only the removal of part of the fission products. The waste requiring disposal is the fission products removed from the fuel during post processing and the fission products already removed during the in-line processing while the reactor was operating. The remainder of the fuel can all be re-used as fuel in the same type of reactor with the missing mass replaced by depleted or natural uranium.

The nuclear proliferation risk is reduced because there is no need to separate the plutonium from the uranium and the plutonium has a high percentage of Pu-240. Also, there is no need for uranium enrichment if the starting fuel is used LWR fuel.

CONCLUSIONS

The proposed reactor with its uranium/plutonium breeder fuel cycle has many advantages. It can be fueled with used LWR fuel with minimal processing, it can dispose of large quantities of LWR waste, and it requires no enrichment capabilities (although it can optionally be fueled with very low enriched uraniun). The waste it produces is limited to the fission products; its fuel can be reused indefinitely with the mass of the fission products replaced with depleted or natural uranium. The time until refueling is required is very long, 30 to 100 years. The control system has a wide range of control, so the fissile content of the initial fuel and the amount of neutron absorbing fission products can vary over a wide range. It also has the advantages of inherent safety and simple and effective load following.

The challenges with this design are mostly engineering and materials problems related to the high temperature of the molten uranium fuel. The MCNP model allows quick evaluation of different options for reactor materials and dimensions and different fuel choices such as uranium alloy or uranium salts.

ACKNOWLEDGMENTS

The author thanks Dr. Mihai (Mike) G. Pop - AREVA Senior International Expert (Retired) for his assistance, advice, and encouragement.

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