THORIUM-FUELED UNDERGROUND POWER PLANT BASED ON MOLTEN SALT TECHNOLOGY

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This paper addresses the problems posed by running out of oil and gas supplies and the environmental problems that are due to greenhouse gases by suggesting the use of the energy available in the resource thorium, which is much more plentiful than the conventional nuclear fuel uranium. We propose the burning of this thorium dissolved as a fluoride in molten salt in the minimum viscosity mixture of LiF and BeF$_2$ together with a small amount of ^{235}U or plutonium fluoride to initiate the process to be located at least 10 m underground. The fission products could be stored at the same underground location. With graphite replacement or new cores and with the liquid fuel transferred to the new cores periodically, the power plant could operate for up to 200 yr with no transport of fissile material to the reactor or of wastes from the reactor during this period. Advantages that include utilization of an abundant fuel, inaccessibility of that fuel to terrorists or for diversion to weapons use, together with good economics and safety features such as an underground location will diminish public concerns. We call for the construction of a small prototype thorium-burning reactor.

I. POWER PLANT DESIGN

This paper brings together many known ideas for nuclear power plants. We propose a new combination including non-proliferation features, undergrounding, limited separations, and long-term, but temporary, storage of reactor products also underground. All these ideas are intended to make the plant economical, resistant to terrorist activities, and conserve resources in order to be available to greatly expand nuclear power if needed as envisioned by Generation IV reactor requirements.

We propose the adoption of the molten salt thorium reactor that uses flowing molten salt both as the fuel carrier and as a coolant. The inventors of the molten salt reactor were E. S. Bettis and R. C. Briant, and the development was carried out by many people under the direction of A. Weinberg at Oak Ridge National Laboratory.[1] The present version of this reactor is based on the Molten Salt Reactor Experiment[2–4] that operated between 1965 and 1969 at Oak Ridge National Laboratory at 7-MW(thermal) power level and is shown in Fig. 1. The solvent molten salt is lithium fluoride (LiF, ~70 mol%) mixed with beryllium fluoride (BeF$_2$, 20%), in which thorium fluoride (ThF$_4$, 8%) and uranium fluorides are dissolved (1% as ^{238}U and 0.2% as ^{235}U in the form of UF$_4$ and UF$_3$, UF$_3$/UF$_4 \geq$ 0.025).[a] This mixture is pumped into the reactor at a temperature of ~560°C and is heated up by fission reactions to 700°C by the time it leaves the reactor core, always near or at atmospheric pressure. The materials for the vessel, piping, pumps, and heat exchangers are made of a nickel alloy.[5,6][b] The vapor pressure of the molten salt at the temperatures of interest is very low ($<10^{-4}$ atm), and the projected boiling point at atmospheric pressure is very high (~1400°C). This heat is transferred by a heat exchanger to a nonradioactive molten fluoride salt coolant[c] with an inlet temperature of 450°C and the outlet liquid temperature of 620°C that is pumped to the conventional electricity-producing part of the power plant located aboveground. This heat is converted to electricity in a modern steam power plant at an efficiency of ~43%.

The fluid circulates at a moderate speed of 0.5 m/s in 5-cm-diam channels amounting to between 10 and 20% of the volume within graphite blocks of a total height of a few meters.

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†We are sorry to inform our readers that Edward Teller is deceased
 September 9, 2003.

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[a]Instead of the Be and Li combination, we might consider sodium and zirconium fluorides in some applications to reduce hazards of Be and tritium production from lithium.

[b]It seems likely all these components could be made of composite carbon-based materials instead of nickel alloy that would allow raising the operating temperature so that a direct cycle helium turbine could be used rather than a steam cycle (~900°C) and hydrogen could be made in a thermochemical cycle (~1050°C). A modest size research and development program should be able to establish the feasibility of these high-temperature applications.

[c]A secondary coolant option is the molten salt, sodium fluoroborate, which is a mixture of NaBF$_4$ and NaF. Other coolants are possible depending on design requirements such as low melting temperature to avoid freeze-up.
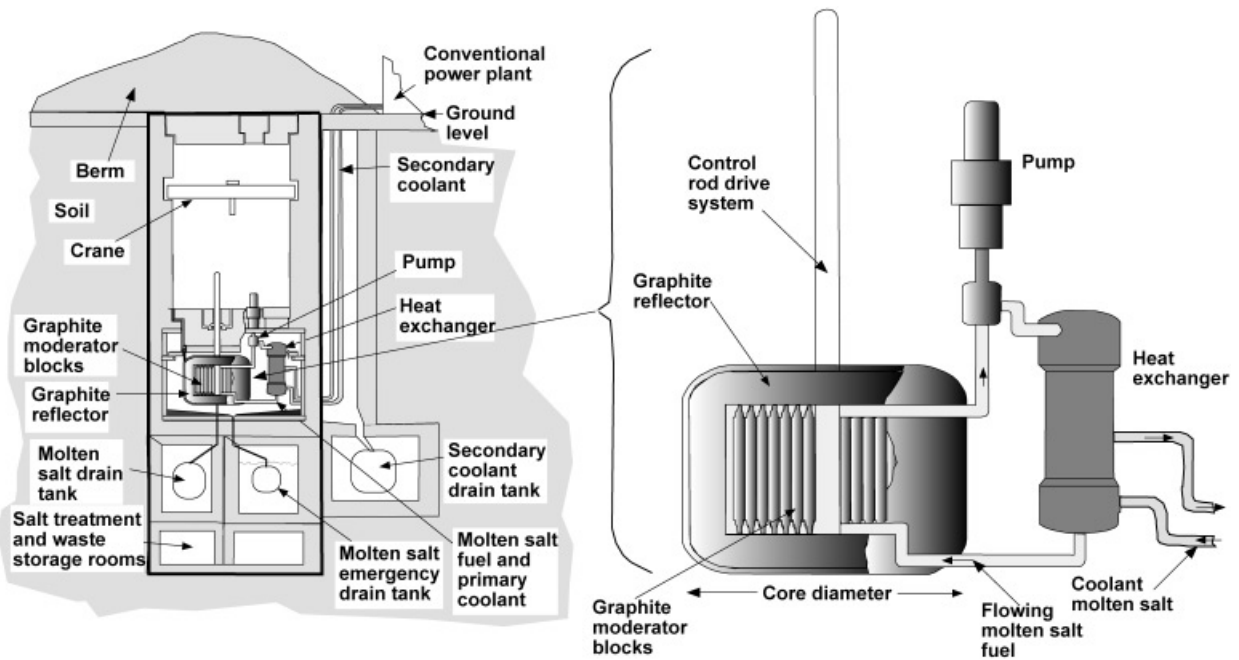
Fig. 1. The nuclear part of the molten salt power plant[7] is illustrated belowground with the nonradioactive conventional part aboveground; many rooms and components are not shown. New cores would be installed after each continuous operating period of possibly 30 yr or the graphite in the cores can be replaced.

Of all these components, only graphite can burn and then slowly. Leakage of air or water into the molten salt is to be minimized to limit corrosion, as oxidation rates are low. In case of an accident, the fuel would be isolated from the graphite by passively draining the molten salt to the drain tank thus removing the decay heat source making the graphite hot.

The graphite slows down the fast neutrons produced by the fission reaction. The slowed neutrons produce fission and another generation of neutrons to sustain the chain reaction.

One of the slowed neutrons is absorbed in ^{232}Th producing ^{233}Th, which undergoes a 22-min beta decay to ^{233}Pa. The ^{233}Pa undergoes a month-long beta decay into ^{233}U, which with a further neutron produces fission and repeats the cycle. The reactions are illustrated in Fig. 2. Note that the cycle does not include ^{235}U, which is used only to initiate the process. The result is a drastic reduction of the need for mined uranium.

The initial fuel to start up the reactor can be mined and enriched ^{235}U [~3500 kg for 1000 MW(electric)]. An alternative
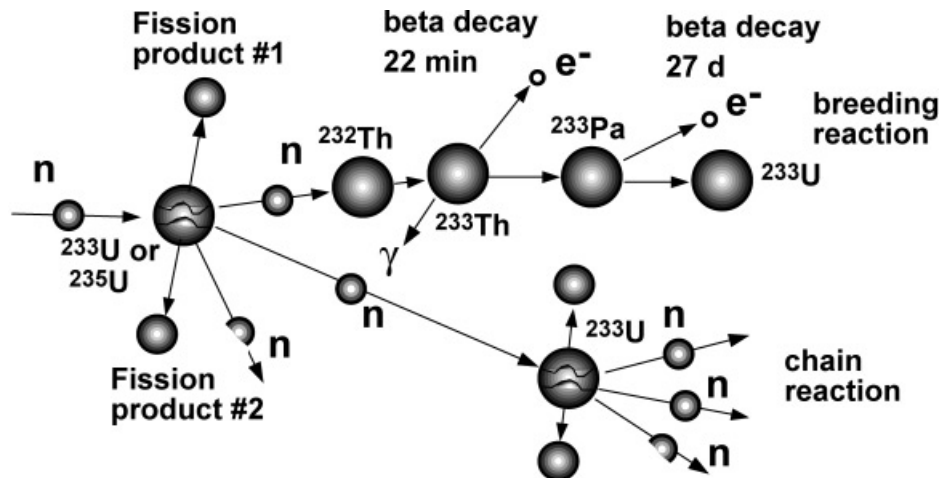

Fig. 2. Illustration of the process of breeding or producing new fuel, ^{233}U, from neutron capture in ^{232}Th as a part of the chain reaction. Each fission reaction produces two or three neutrons (about 2.5 on average as illustrated by the "half neutron" above).

might be to start up on discharged light water reactor (LWR) spent fuel, particularly ^{239}Pu. Actually, ^{239}Pu is contained in a waste of transuranium elements that people might actually pay to give this fuel away. As the plant operates, the plutonium and higher actinides and ^{235}U would be fissioned and replaced with ^{233}U produced from thorium, which is even a better fuel than ^{235}U, because nonfission thermal neutron captures are about half as likely.

An important feature of our proposal is to locate everything that is radioactive at least 10 m underground—where all fissions occur—while the electric generators are located in the open, being fed by hot, nonradioactive liquids. The reactor's heat-producing core is constructed to operate with a minimum of human interaction and limited fuel additions for decades. Of the three underground options,[8] excavation into mountains with tunnel or vertical access or surface excavation with a berm covering, we prefer the berm as illustrated in Fig. 1. Undergrounding will preclude the possibility of radioactive contamination in case of airplane disasters. A combination of 10 m of concrete and soil is enough mass to stop most objects. It would eliminate tornado hazards and, most particularly, contribute to defense against terrorist activities. In case of accidents, undergrounding, in addition to the usual containment structures, enhances containment of radioactive material. The 10-m figure is a compromise between safety and plant construction expense. We anticipate the cost to construct underground with only 10 m of overburden using the berm technique will add <10% to the cost.

The molten salt reactor that operated in the 1960s had a big advantage in the removal of many fission products without much effort. Gases (Kr and Xe) simply bubble off aided by helium gas bubbling, where these gases are separated from the helium and stored in sealed tanks to decay. Noble and seminoble metals[d] precipitated. In the planned reactor, the old method of removing the gases may be repeated. The precipitation process might conceivably be enhanced by using a centrifuge and filtering rather than the old uncontrolled method of precipitation. In this way, the need to remove the remaining fission products, e.g., the rare-earth elements (Sm, Pm, Nd, Pr, Eu, and Ce) and alkali-earth elements with valence two and three fluoride formers, is reduced and may be postponed to intervals, perhaps as long as once every 30 yr. The accumulation of these elements has a small effect on neutron economy and on chemistry such as corrosion. Experience is needed on these long-term effects.

Most fission products have half-lives of ~30 yr or less. These "short-lived" fission products can be stored and monitored at the plant site for hundreds of years, while their hazard decreases by three orders of magnitude or more by the natural process of radioactive decay. Three elements are notable because they need to be separated for special treatment because of their extra long lives: ^{99}Tc, ^{129}I, and ^{135}Cs [with half-lives of 210 000 yr, 1.6 million yr, and 2.3 million yr; capture cross sections of 20, 30, and 9 b (10^{-24} cm^2); and production rates of 23, 3.8, and 34 kg/GW(electric)·yr, respectively]. New ways should be found for separating these long-lived products (≫30-yr half-life) from short-lived products (≤30-yr half-life).

After a period of operation, perhaps as long as 30 yr, the reactor is shut down, owing to the swelling of the graphite blocks as shown in Fig. 3. The criterion[4,9] used here is 30 yr for a 10-m-diam core at 1000 MW(electric), for a neutron dose of $<3 \times 10^{26}$ n/m^2 for $E > 50$ keV and a swelling of 3 vol% at 750°C for a capacity factor of 85%. Robotic technology is developing so rapidly that graphite replacement might be a quick and a low-cost operation. Another process that might be life limiting is corrosion. At the time that a new or refurbished power-generating graphite core is put into operation and the corroded parts are replaced, the fuel dissolved in the molten salt is transferred to the new core in a liquid state. This fuel transfer and core refurbishment allows the power station to continue operating for several more decades. At this time, the remaining fission products in solution can be removed by the chemical process known as *reductive extraction* to limit the neutron loss to absorption. The bulk of materials (lithium, beryllium, and thorium fluorides) may last for several hundred years before they are transmuted to other elements by nuclear reactions.

This process might conceivably be continued as long as we operate the power station, perhaps even hundreds of years, making operations and ownership similar to a dam but with less impact. The fission products will be separated and stored at the power plant site in a suitable form under careful supervision or they will be transported to a permanent disposal site. We propose the twofold argument for the safe interim storage of radioactive material: first, that the location will be underground, and second, that the storage will be at the site of operating reactors, which require carefully planned defense anyway.

When the site with its collection of reactors is to be shut down, careful considerations will have to be used in the choice between whether the accumulated radioactivity should be transported to a permanent storage site or whether continuation of established supervision is safer and less expensive. The idea is to transport only mildly radioactive fuel to the power plant but have a minimum of transport of highly radioactive fission products and fuel away from the plant, thus minimizing the chance of accidents or terrorist activities. One conclusion is obvious: It
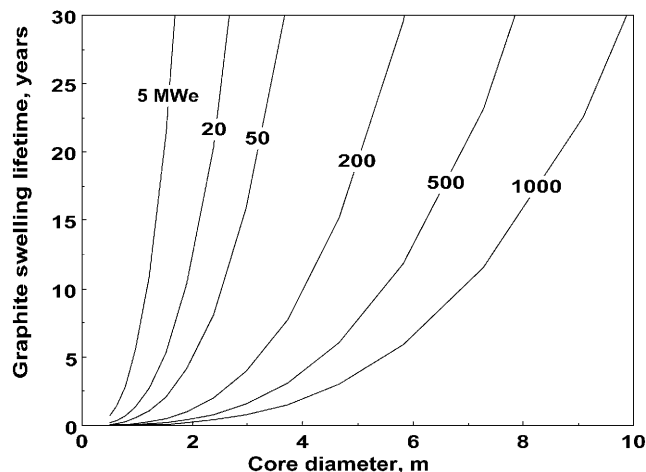

Fig. 3. The core lifetime versus diameter (see Fig. 1) limited by graphite swelling is shown for a wide range of output power.

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[d]Noble and seminoble metals are Zn, Ga, Ge, As, Nb, Mo, Tc, Ru, Rh, Pd, Ag, Cd, In, Sn, and Sb. Seminoble here means they do not form fluorides but rather precipitate in elemental form.

will become important to find useful applications of radioactivity such as radioactive tracers, thereby converting a serious worry into a potential asset.

II. WASTE FORM: SUBSTITUTED FLUORAPATITE

A possible waste form for the molten salt reactor might be based on the naturally occurring mineral that has been found to contain ancient actinides in the natural reactor in Africa in mineral deposits called fluorapatite[10,11] $Ca_5(PO_4)_3F$. This low solubility mineral is much like fluoridated tooth enamel. If we substitute the fission product ions, for example, Sm, for the Ca ions, we call this substituted fluorapatite

$$SmF_3 + 4.5Ca_3(PO_4)_2 \rightarrow 3(Sm_{0.33}Ca_{4.5})(PO_4)_3F \ . \quad (1)$$
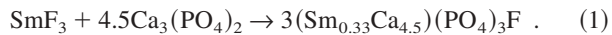
The result of this reaction is a ceramic powder that can be melted into bricks for long-term storage either at the power plant site or at a repository.

It might be preferable to transport it in a more compact fluoride form and produce the more stable but larger mass and volume form of material at the permanent repository site. The stored fluoride wastes could be melted and transferred in liquid form to a shipping container much like that used for sulfur shipping except more massive and shipped to a permanent storage site where again they are transferred in liquid form to be made into substituted fluorapatite bricks. If permanent storage is decided upon, we estimate the space needed in a Yucca Mountain–like repository for molten salt wastes to be ten and maybe closer to 100 times less than for once-through LWR spent fuel based on the heat generation rate of the wastes.

III. SAFETY

The molten salt reactor is designed to have a negative temperature coefficient of reactivity. This means the reactor's power quickly drops if its temperature rises above the operating point, which is an important and necessary safety feature. The molten salt reactor is especially good in this respect—it has little excess reactivity because it is refueled frequently online and has a high conversion rate that automatically replaces fuel consumed. Failure to provide makeup fuel is fail-safe as the reactivity is self-limiting by the burnup of available fuel. A small amount of excess reactivity would be compensated by a temporary interruption of adding makeup fuel online. Present reactors have ∼20% excess reactivity. Control rods and burnable poisons are used not only in accident control but also to barely maintain criticality. In the molten salt reactor, control rods are used to control excess reactivity of perhaps only 2%, which is necessary to warm the salt from the cooler start-up temperature to the operating temperature (i.e., overcome the negative temperature coefficient). That is, only enough fissile fuel is in the core to maintain a chain reaction and little more.

Gaseous fission products are continually removed and stored separately from the reactor in pressurized storage tanks. By contrast, in conventional reactors the gaseous fission products build up in the Zr-clad fuel tubes to a high pressure that presents a hazard and can cause trouble. If an unforeseen ac-

cident were to occur, the constant fission-product removal means the molten salt reactor has much less radioactivity to potentially spread.

The usual requirement of containing fission products within three barriers is enhanced by adding a fourth barrier. The primary vessel and piping boundary, including drain tanks, constitute one barrier. These components are located in a room that is lined with a second barrier, including an emergency drain or storage tank for spills. The third barrier is achieved by surrounding the entire reactor building in a confinement vessel. A fourth safety measure is locating the reactor underground, which itself is one extra "gravity barrier" aiding confinement. A leakage of material would have to move against gravity for 10 m before reaching the atmosphere.

In case of accidents or spills of radioactive material, the rooms underground would remain isolated. However, the residual decay heat that continues to be generated at a low rate would be transferred through heat exchangers that passively carry the heat to the environment aboveground, while retaining the radioactive material belowground. This passive heat removal concept perhaps using heat pipes will be used to cool the stored fission products as well.

The initial fuel needed including the amount circulating outside the core is considerably less than half that of other breeding reactors such as the liquid metal–cooled fast reactor. This is a consequence of fast reactors having much larger critical mass than thermal reactors and for the molten salt case, avoiding the need for extra fuel at beginning of life to account for burnup of fuel.

IV. FUEL CYCLE WITHOUT FUEL PROCESSING AND WITHOUT WEAPONS-USABLE MATERIAL

When the 1000-MW(electric) reactor is started up, the initial fissile fuel is 20% enriched uranium (20% ^{235}U and 80% ^{238}U) along with thorium, actually 3.5 tons of ^{235}U, 14 tons of ^{238}U, and 110 tons of thorium. This low enrichment makes the uranium undesirable as weapons material without isotope separation, and therefore it does not have to be guarded so vigorously. An important side product is a small amount of ^{232}U produced by $(n, 2n)$ and (γ, n) reactions on ^{233}U producing ^{232}U. Uranium-232 is highly radioactive and has unusually strong and penetrating gamma radiation (2.6 MeV), making diversion of this fuel for misuse extra difficult and easier to detect if stolen; the resulting weapons would be highly radioactive and therefore dangerous to those nearby as well as making detection easier.

The uranium in the core starts at 20% fissile and drops so it is never weapons usable.[e] The plutonium produced from neutron capture in ^{238}U rather quickly develops higher isotopes of plutonium, making it a poor material for weapons.[f] Safeguarding is still necessary but less important. The advantage of this fuel cycle is that 80% of its fuel is made in the

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[e]For example after 15 yr of operation, the isotopes of the uranium in the molten salt are ∼0.02% ^{232}U; 8% ^{233}U; 2% ^{234}U; 4% ^{235}U; 3% ^{236}U; and 83% ^{238}U.

[f]After operation for 15 yr, the plutonium in the molten salt has the following isotopes: 7% ^{238}Pu; 36% ^{239}Pu; 21% ^{240}Pu; 15% ^{241}Pu; and 20% ^{242}Pu.

reactor, and the fuel shipments to the plant during its operation are nonweapons usable.

$$\text{Conversion ratio} = \frac{^{233}\text{U and fissile Pu production rate}}{\text{all fissile consumption rate}} \quad . \quad (2)$$

The conversion ratio starts out at 0.8, and after 30 yr of operation drops to 0.77 (Ref. 4). Today's LWRs[g] each require 5700 tons of mined uranium in 30 yr. Our molten salt reactor example also at 1000-MW(electric) size, in 30 yr of operation at 75% capacity factor would consume by fissioning, 17 tons of thorium, 3.8 tons of ^{238}U, and 6.7 tons of ^{235}U. This requires 1500 tons of mined uranium.[h] Our worries about the consumption of uranium are reduced by a factor of 4 relative to today's reactors while the depletion of thorium remains entirely negligible.

In our example, 14% of the heavy atoms that have been transported to the reactor are burned up or fissioned in 30 yr of operation.[i] If we include the 1500 tons of mined uranium that went into the depletion process and was not used in the reactor, then the percentage of burnup is 1.3%. This compares to our present-day reactor example with once-through fueling of 0.5% burnup of mined uranium with the assumptions in footnote g.

V. ALTERNATIVE FUEL CYCLE

If we decide in future versions of the molten salt reactor to move toward the pure thorium-^{233}U cycle with fuel processing, then the conversion ratio approaches unity and the use of mined uranium will drop by over an order of magnitude or be eliminated once started up. This cycle would start up the reactor with only ^{235}U and thorium dissolved in the molten salt.[j] Neutrons absorbed in thorium would produce ^{233}U. Although this fuel is highly radioactive, after chemical separation it is directly usable in nuclear weapons and therefore poses a danger that would have to be guarded against with extra measures. We should avoid designs that permit separation of protactinium because it decays into ^{233}U without the highly radioactive ^{232}U "spike" previously mentioned.

The strong advantage of this fuel cycle is that it breeds essentially all of its own fuel, thus removing the need for transportation of weapons-usable material to the reactor site once it is started up. Also it makes no further demands for mined uranium for several hundred years although the graphite had to be changed a number of times.[4] For example, a present-day reactor would use 38 000 tons of mined uranium over 200 yr, while the molten salt reactor once started up on ^{235}U and thorium would need only 600 tons of mined uranium and could operate for 200 yr (see footnote g again). One hundred thirty-seven tons of thorium would be fissioned.[k] The burnup of the 600 tons of uranium and 137 tons of thorium would be ~18%.

Even a small amount of fissile material removed from the reactor would cause it to cease operation, and this mitigates the danger of diversion from the plant site. Diversion of the material for weapons use would be an interruption of normal procedures, which could be carried out only by insiders. It is clear

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[g]The assumption on LWR fuel usage can be seen:

$$\frac{1000 \text{ MW(electric)} \cdot 0.75 \cdot 365 \text{ day/yr} \cdot 30 \text{ yr} \cdot 5\%}{0.32 \dfrac{\text{MW(electric)}}{\text{MW}} \cdot 50\,000 \text{ MWd}/T \cdot 0.45\%} = 5700 \text{ tons}$$

of mined uranium in 30 yr with tails of 0.25%. (5700 tons $\cdot$ 200 yr)/30 yr = 38 000 tons in 200 yr.

$$\text{Burnup of heavy atoms} = \frac{1000 \text{ MW(electric)} \cdot 235 \text{ amu} \cdot 1.67 \cdot 10^{-27} \text{ kg/amu} \cdot 365 \text{ days/yr} \cdot 24 \text{ h/day} \cdot 3600 \text{ s/h}}{0.32 \text{ MW(electric)/MW} \cdot 195 \text{ MeV} \cdot 1.6 \cdot 10^{-19} \text{ J/eV}}$$

$$= 1240 \text{ kg/full power year} \quad .$$

$$\text{Burnup fraction} = 1.24 \text{ tons} \times 30 \text{ yr} \times 0.75/5700 \text{ tons} = 0.49\% \quad .$$

Mined uranium for the molten salt reactor to start up is 3.5 tons ^{235}U/0.0045 = 780 tons of mined uranium. For the alternative fuel cycle, the start-up is 2.8 tons ^{235}U/0.0045 = 620 tons of mined uranium.

[h]6.7 tons of ^{235}U/0.0045 = 1500 tons of mined uranium where we assume the ^{235}U content of 0.7% of uranium can be used with tails of 0.25%.

[i]Burnup of heavy atoms $= \dfrac{1000 \text{ MW(electric)} \cdot 233 \text{ amu} \cdot 1.67 \cdot 10^{-27} \text{ kg/amu } 365 \text{ days/yr} \cdot 24 \text{ h/day} \cdot 3600 \text{ s/h}}{0.43 \text{ MW(electric)/MW} \cdot 195 \text{ MeV} \cdot 1.6 \cdot 10^{-19} \text{ J/eV}}$

$$= 915 \text{ kg per full power year} \quad .$$

$$\text{Burnup fraction} = \text{atoms burned (fissioned) in 30 yr/all heavy atoms}$$

$$= \frac{915 \text{ kg} \cdot 0.75 \cdot 30 \text{ yr}}{110\,000 \text{ kg Th} + 32\,400 \text{ kg } ^{238}\text{U} + 7900 \text{ kg } ^{235}\text{U}} = \frac{20\,600 \text{ kg}}{150\,300 \text{ kg}} = 13.7\% \quad .$$

We use 30-yr period and 75% capacity factor consistently for all cases, so that relative comparisons are unaffected by this assumption. The fissile consumption is then $0.75 \times 915 = 690$ kg/yr.

[j]Uranium-233 for start-up fuel could be produced externally from accelerator or thermonuclear fusion produced neutrons absorbed in thorium if these technologies become developed successfully. This fissile source or use of discharge fuel from current fission reactor designs would virtually eliminate the need for further uranium mining but would introduce proliferation issues that could and would have to be dealt with.

[k]Burnup of heavy atoms in 200 yr = 0.915 tons/yr per full power year $\times$ 0.75 capacity factor $\times$ 200 yr = 137 tons in 200 yr. Burnup in 200 yr = (137 tons Th)/(620 tons mined U + 137 tons Th) = 18%.

that continuous operation would be needed. Thus, it should be easily noticed unless carried out by separating small amounts for a long period.

We advocate full compliance and even strengthened international safeguard agreements including inspection regimes and technical means for monitoring the reactor and all its operations. Monitoring devices including cameras and transceivers possibly in miniature or even subgram sizes might aid monitoring systems to find out whether all components in the system are in place and operating normally. It is difficult to exclude the possibility that considerable quantities of components of nuclear explosives might be produced in reactors, and therefore information on the production of these materials should be readily available. This requirement should be considered a crucial part of a policy of openness (to be introduced gradually), which, in a general sense, will be necessary to insure the stability of the world. Openness is not an easy condition to fulfill but perhaps better than any obvious alternative.

VI. ECONOMIC COMPETITIVENESS

Our economic goal is to achieve a cost of electrical energy averaged over the life of the power station to be no more than that from burning fossil fuels at the same location. Past studies have shown a potential for the molten salt reactor to be somewhat lower in cost of electricity than both coal and LWRs (Refs. 4 and 12). There are several reasons for substantial cost savings: low pressure operation, low operations and maintenance costs, lack of fuel fabrication, easy fuel handling, low fissile inventory, use of multiple plants at one site allowing sharing of facilities, and building large plant sizes. The cost of undergrounding the nuclear part of the plant obviously needs to be determined and will likely not offset the cost advantages of a liquid-fueled low-pressure reactor.

VII. WHY HAS THE MOLTEN SALT REACTOR NOT ALREADY BEEN DEVELOPED?

If the molten salt reactor appears to meet our criteria so well, why has it not already been developed since the molten salt reactor experiment operated over 30 yr ago?

Several decades ago an intense development was undertaken to address the problem of rapid expansion of reactors to meet a high growth rate of electricity while the known uranium resources were low. The competition came down to a liquid-metal fast breeder reactor (LMFBR) on the uranium-plutonium cycle and a thermal reactor on the thorium-^{233}U cycle, the molten salt breeder reactor. The LMFBR had a larger breeding rate, a property of fast reactors having more neutrons per fission and less loss of neutrons by parasitic capture, and won the competition. This fact and the plan to reduce the number of candidate reactors being developed were used as arguments to stop the development of the molten salt reactor rather than keep an effort going as a backup option. In our opinion, this was an excusable mistake.

As a result there has been little work done on the molten salt reactor during the last 30 yr. As it turned out, a far larger amount of uranium was found than was thought to exist, and the electricity growth rate has turned out to be much smaller than predicted. High excess breeding rates have turned out not to be essential. A reactor is advantageous that once started up needs no other fuel except thorium because it makes most or all its own fuel.

Studies of possible next-generation reactors, called Generation IV, have included the molten salt reactor among six reactor types recommended for further development. In addition the program called Advanced Fuel Cycle Initiative has the goal of separating fission products and recycling for further fissioning.

VIII. DEVELOPMENT REQUIREMENTS AND CONCLUSIONS

In conclusion, we believe a small prototype plant should be built to provide experience in all aspects of a commercial plant. The liquid nature of the molten salt reactor permits an unusually small plant that could serve the role just so that the temperatures, power densities, and flow speeds are similar to that in larger plants. A test reactor, e.g., 10 MW(electric) or maybe even as small as 1 MW(electric) would suffice and still have full commercial plant power density and therefore the same graphite damage or corrosion limited lifetime. Supporting research and development would be needed on corrosion of materials, process development, and waste forms, all of which, however, are not needed for the first prototype.

We give some examples of development needs. We need to show adequate long corrosion lifetime for nickel alloy resistant to the tellurium cracking observed after the past reactor ran for only 4 yr. If carbon composites are successful, corrosion will likely become less important. We want to prove feasible extraction of valence two and three fluorides, especially rare-earth elements, which will then allow the fuel to burn far longer than 30 yr (200 yr). We need to study and demonstrate an interim waste form suggested to be solid and liquid fluorides and substitute fluorapatite for the permanent waste form of fission products with minimal carryover of actinides during the separation process. This solution holds the promise to diminish the need for repository space by up to two orders of magnitude based on waste heat generation rate. We need a study to show the feasibility of passive heat removal from the reactor afterheat and stored fission products to the atmosphere without material leakage and at reasonable cost. Another study needs to show that all aspects of the molten salt reactor can be done competitively with fossil fuel. The cost for such a program would likely be well under $1 billion with operation costs likely on the order of $100 million per year. In this way a very large-scale nuclear power plan could be established, including even the developing nations, in a decade.

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