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D.2. LOS ALAMOS NATIONAL LABORATORY ADS PROJECTS

D.2.1. BASIS AND OBJECTIVES OF THE LOS ALAMOS ACCELERATOR DRIVEN TRANSMUTATION TECHNOLOGY PROJECT

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The Accelerator-Driven Transmutation Technology (ADTT) Project carries three approaches for dealing with waste from the defense and commercial nuclear energy enterprise. First, the problem of excess weapons Plutonium in the U. S. and Russia originating both from stockpile reductions and from defense production site clean-up is one of significant current and long-term concern. The ADTT technology offers the possibility of almost complete destruction of this Plutonium by fission. The technology might be particularly effective for destruction of the low quality Plutonium from defense site clean-up since the system does not require the fabrication of the waste into fuel assemblies and can tolerate a high level of impurities in the feed stream. Second, the ADTT system also can destroy the Plutonium, other higher actinide, and long-lived fission product from commercial nuclear waste which now can only be dealt with by geologic storage. And finally, and probably most importantly the system can be used for the production of virtually unlimited electric power from Thorium with concurrent destruction of its long-lived waste components so that geologic containment for them is not required. In addition Plutonium is not a significant by product of the power generation so that non-proliferation concerns about nuclear power are almost completely eliminated. All of the ADTT systems operate with an accelerator supplementing the neutrons which in reactors are provided only by the fission process, and therefore the system can be designed to eliminate the possibility of a runaway chain reaction. The means for integration of the accelerator into nuclear power technology in order to make these benefits possible is described including estimates of accelerator operating parameters required for the three objectives.

D.2.1.1. INTRODUCTION

Concerns about waste from the defense and commercial nuclear sectors has grown to such an extent in recent years that it now dominates the nuclear enterprise. The emphasis in the nuclear technology field has moved from its earlier reactor-design focus into clean-up of defense production sites and a resolution of the commercial nuclear waste problem. The development of cleaner and safer systems for nuclear energy generation is almost at a standstill because of growing international concerns about the waste issues. The predominant approach to this problem for the past thirty years has been the geologic storage of waste whether it be from the defense or the commercial sector. Geologic storage offers the prospect of confining nuclear waste by the features of a stable geologic structure rather than relying on long-term containment of the waste in man-made containers. In addition the waste is made much less accessible by its placement deep underground. Therefore many countries are providing significant funding for the development and siting of geologic waste storage facilities. While a number of sites might be under study in a given country, the intent is to provide a single site capable of confining the high level waste.

It has become increasingly difficult to convince a community to become host to a nation's single site for storage of waste which many consider to be the nation's most dangerous. The fact that the waste remains dangerous for many tens of thousands of years exacerbates these concerns. The concern that such repositories can become mines for Plutonium has become of even greater concern as the U. S. has made it known that dangerous nuclear weapons can be made from commercial Plutonium [1]. The natural

transformation of commercial Plutonium into weapons Plutonium by radioactive decay[1] means that eventually many thousands of tons of weapons Plutonium will be stored at many sites around the world. Some are becoming concerned about the possibility of natural or induced supercriticality of fissile material stored underground [2]. As a consequence of these and other concerns remaining to be resolved about geologic storage, no nation is expected to begin emplacement of high level waste in a geologic repository before the year 2010 and the ultimate viability of the geologic storage concept remains to be demonstrated.

The world is therefore in need of an acceptably priced and safe alternative to the geologic storage concept. In the U. S. commercial nuclear waste is accumulating at reactor sites and the defense site clean-up effort is struggling to understand what will happen to the Plutonium and other high-level waste which will be gathered together after the clean-up has been completed. The Los Alamos National Laboratory along with a rapidly developing national and international community has therefore been studying Accelerator-Driven Transmutation Technology (ADTT) as a possible means of destruction of this nuclear waste and of generating nuclear power by systems which do not generate the most dangerous components of this waste and which concurrently destroy their own waste. If the full capability of the ADTT systems can be realized at acceptable cost, geologic storage of defense and commercial waste would not be required.

The main elements and function of an ADTT system are illustrated in Fig. 1 for a system which generates nuclear energy from Thorium, avoids the production of Plutonium and concurrently destroys its long-lived high-level fission product waste. This system is referred to as Accelerator-Driven Energy Production (ADEP). The system starts with benign ^{232}Th and converts it by neutron absorption into the excellent fissile fuel ^{233}U from which electric power is produced. The system consists of a reactor-like component referred to in the figure as the target-blanket which contains the fissile material and the waste to be destroyed. For a reactor each fission on average produces enough neutrons after losses to cause another fission so that the chain of fissions is continuous. For all ADTT systems, the losses are made somewhat larger by the expenditure of neutrons on waste destruction so that there are about 5-10 % fewer neutrons than necessary to maintain the chain. Therefore by itself the system is totally passive and inoperative. However, by making up for the 5-10 % loss of neutrons from an external neutron source, the system would function effectively even though the chain reaction would not be self-sustaining.

The essential conceptual difference between the ADTT system and a reactor is the presence of an accelerator to produce neutrons and the presence of a target inside of the reactor-like component to convert a beam of protons from the accelerator into neutrons. All electric-power-producing reactors presently operating have means for removing the heat from the system, converting it to steam, and driving generators for electric power production. These elements are also shown in Fig. 1 with most of the power being sent into the commercial grid except for 10-15 % being used to power the accelerator. Operation of the system stops when the accelerator stops because the system fission chain is not self-sustaining. For this reason the system can be made safe from a runaway chain reaction such as that which occurred at Chernobyl by entirely different means than that incorporated in other reactors, and many of the safety features required in accelerators such as control rods may be omitted.

To understand the value of the accelerator more clearly, consider a system containing ^{233}U fuel which undergoes fission with 92 % probability upon absorption of one thermal neutron and which releases 200 MeV per fission. Assume further that no neutrons are released in fission. The 100,000 MeV released by 500 such fission events would be converted with 42 % efficiency to 42,000 MeV of electric energy. The accelerator would convert this with 45 % efficiency to 18,900 MeV of proton beam power if all of the electric power were fed back to the accelerator. For a proton energy of 800 MeV, the accelerator would produce 18,900/800 = 23.6 protons. At the conversion rate of 25 neutrons per proton which probably can be achieved, a total of 23.6 × 25 = 590 neutrons per 500 fissions is possible. Upon absorption, 92 % of these neutrons would lead to fission of 543 nuclei of ^{233}U. Comparing this number with the original 500 fission events, we see that an accelerator-linked chain reaction is possible even if no neutrons were emitted

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[1]The isotope that makes the principle difference between commercial Plutonium and weapons Plutonium is ^{240}Pu which decays with a half life of 6,600 years. After about 13,000 years commercial Plutonium containing about 24 % ^{240}Pu transforms to weapons Plutonium containing about 6 % ^{240}Pu.

from fission! Of course instead of no neutrons per fission 2.49 neutrons are produced per fission of a ^{233}U nucleus so that altogether one has 590 + 500 × 2.49 = 1835 neutrons per 500 fissions for an increase in the effective number of neutrons from fission from 2.49 to 3.67 if all of the electric power from the target blanket were fed back to the accelerator. This is an increase of more than one neutron per fission and is an enormous increase in the number of neutrons per fission which are available to a nuclear system designer. The latter figure is far more neutrons than are required to maintain the fission process and to breed the ^{233}U from the Thorium, so that only a small portion of the electric power must be consumed by the accelerator. The possibility to dial the neutron production requirement as desired and to operate effectively a system well away from criticality greatly broadens the parameter space available to the nuclear system designer.

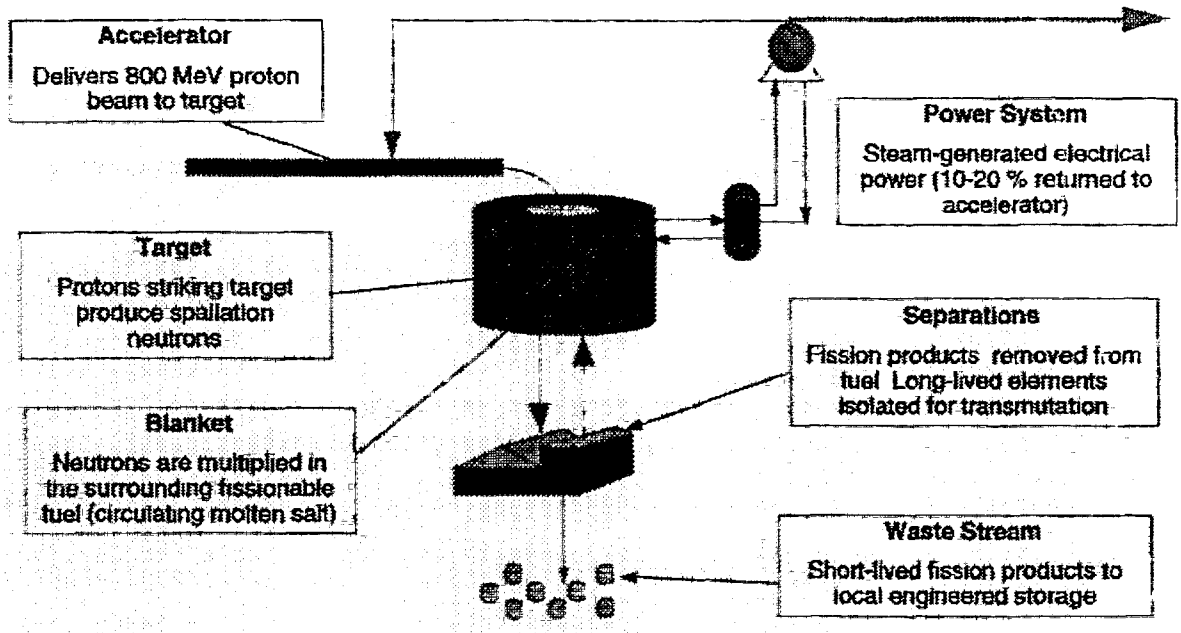

FIG. 1. ADTT System Components. An accelerator produces 800-MeV proton beam which is directed into a reactor-like assembly consisting of a Lead target for the beam and a surrounding blanket containing fissile material. The beam strikes the liquid Lead target and produces about 22 neutrons per proton. The neutrons are moderated in the surrounding blanket which consists mostly of graphite and molten salt which carries the fissile fuel as actinide fluoride. The system operates at $k_{eff} = 0.95$ so that the system multiplies the beam-produced neutrons by about a factor of 20. The blanket contains internal heat exchangers which transfer the heat from the working salt to a secondary external salt stream and then to a steam generator for electric power production. Most of the power is fed into the commercial grid but some of it is used to power the accelerator. The liquid fuel allows the system to be continuously refueled and allows the waste products from fission to be continuously removed.

Owing largely to the enhanced safety of the system, one need no longer remain attached to solid fuel assemblies as in ordinary reactors. Liquid fuel becomes an option with all of the many advantages it provides. In Fig. 1 we show at bottom center a loop carrying the liquid fuel outside of the target-blanket in a continuous flow. An obvious advantage is that the fuel can be continuously added to the system to make up for that which is burned without shutting down for refueling as in the case of the reactor. Of course, the whole process and expense of solid fuel fabrication required for the reactor is avoided as well. But there is even greater benefit from the ability to remove the fission products from the liquid fuel on-line without stopping the system for removal of solid fuel assemblies. By means which will be described later,

the liquid fuel can be continuously cleansed of the fission products which act as neutron poisons. Those long-lived fission products which would ordinarily require geologic storage can be returned to the system to be converted by neutron absorption to stable or short-lived fission product.

Since only fission product is removed from the system, there is no actinide waste except for a very small amount which slips through in the fission product separation process. Because the long-lived waste is destroyed, the only waste from the system is the short-lived and stable fission product. This waste is made up of a number of different species but none of the waste species have half-lives longer than 30 years. Containers can be made to confine this remnant waste until the radioactivity has decayed away by a factor of 1,000 or so. Geologic confinement of the waste is not required because, as is shown later, the remnant waste can be made to satisfy near surface disposal criteria of the NRC and the EPA. If the site of the ADEP system meets the criteria for near-surface disposal, the waste need not leave the site. Therefore only benign Thorium need be brought to the site and no waste need be carried away.

More will be said later about the Thorium-burning system, about weapons Plutonium and commercial waste destruction, and the relationship between the latter two technologies.

D.2 1.2 TARGET-BLANKET DESCRIPTION

More detail on the target-blanket system is shown in Fig. 2. The system consists of a stainless steel tank which contains graphite blocks for neutron moderation and reflection and a molten salt carrier for the fertile and fissile fuel which will be described below. The graphite and molten salt are known to be compatible with one another from extensive experience at Oak Ridge National Laboratory with the Molten Salt Reactor

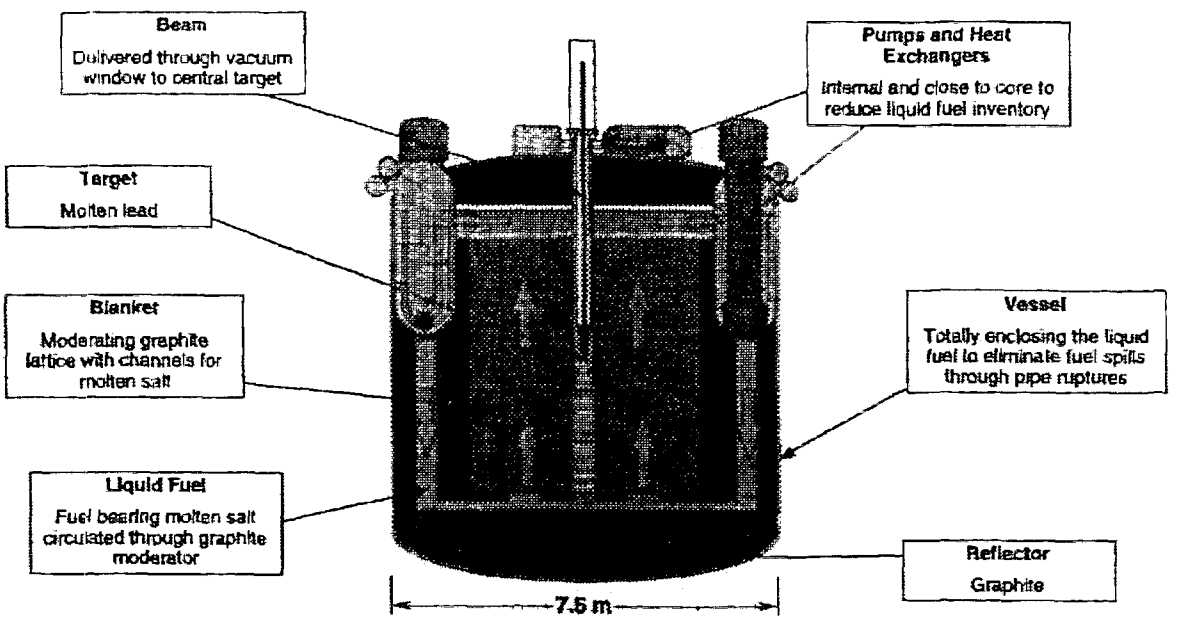

FIG. 2. Target blanket function. The proton beam enters through a window at the top of the system and strikes a liquid Lead target at the center. The Lead is circulated and cooled from above. Five layers of graphite blocks are shown which moderate the neutrons. The molten salt fuel flows upward through holes in the blocks and to the outside through pumps and heat exchangers and back to the bottom. Graphite on all sides serves as a neutron reflector. A cover-gas of helium collects the volatile species and carries them away for appropriate separations.

Experiment (MSRE) [5]. The molten salt flows upward through holes in the graphite blocks across the top of the system to internal magnetic pumps and heat exchangers and back to the bottom of the system. The heat from fission is transferred in the heat exchangers to an external salt loop which carries the heat to steam generators for electric power production. The molten salt is a $LiF\text{-}BeF_2$ eutectic which melts at about 450 degrees centigrade and operates at between 650 and 720 °C. Almost all elements react as fluorides and can be dissolved in small but adequate amounts into the carrier salt for the transmutation and fission requirements. A cover gas of helium is circulated above the molten salt to collect and remove the noble gas and volatile fluoride fission products from the salt. The molten salt never leaves the tank, except for a small slip-stream for on-line refueling and waste removal, and therefore there is no possibility for spillage of the salt through pipe breaks.

Other liquids such as water could be chosen for the carrier. However the salt has the advantage of being an excellent solvent for almost any of the elements present in the system. It also has a low vapor pressure at high temperature which is a major safety advantage allowing operation without a pressure vessel which would be required for a higher vapor pressure medium such as water. The higher operating temperature allows a thermal-to-electric efficiency which might be as high as 44 %. Also the salt is non-reactive with air, nitrogen, or concrete, in contrast with for example, the liquid sodium coolant on which fast reactor technology is now based.

The neutrons are produced at the center in a liquid Lead target. Protons enter from the top through a window and are stopped in the Lead, with the Lead pump and heat exchanger on top of the tank. The Lead is confined by metal resistant to corrosion by the Lead. Since the corrosion properties of the Lead are different from those of the molten salt, the Lead and salt are separated by an inner container compatible with the Lead and an outer container compatible with the salt. The metal for the salt containment probably will be Hastelloy-N developed for the MSRE and for the Lead it probably will be Inconel.

The system operates at a k_{eff} of about 0.95 compared to $k_{eff} = 1$ for a reactor. The neutrons produced by the accelerator therefore are multiplied by a factor of about 20 for and therefore an ADEP system producing power from Thorium with a fission power of 250 MW_t for an electric power output of 100 MW would require an accelerator capable of producing 6 ma at 800 MeV and consuming 11 MW of electric power.

The thermal-to-electric conversion efficiency for such a system would be 44 %. The bussbar-to-beam efficiency would be 45 %.

All systems which produce nuclear power from fission must protect against a potential loss of coolant accident (LOCA) which might occur when the primary coolant system fails and the fission product decay heat builds to dangerous levels. The nuclear reactors now operating have active redundant systems which come into action when the coolant system fails. Newer designs for reactors include passive means to deal with this situation. For example the power density and total power capacity of the reactor might be kept small enough so that the heat can be transferred to the outside of the reactor vessel and from there away from the system into the surroundings by convection or radiant heat loss. The power of such a passive system is usually limited by the rate of heat transfer to the vessel with the components at the center of the reactor being at highest risk.

The ADTT systems are also designed with passive capability for after-heat removal. They have the advantage over water-containing systems that the temperature can be allowed to rise much higher because the ADTT system contains mostly low vapor pressure high melting or vaporization temperature materials such as graphite and molten salt. Therefore much higher temperatures can be tolerated in the ADTT systems without risk of internal damage or dangerously high pressures. In addition to the use of liquid fuel, the incorporation of an internal inside-to-outside flow path and natural convection both contribute to enhanced heat transfer from the inside to the outer wall of the blanket. Therefore ADTT systems can be designed for substantially higher electric power capacity than conventional reactors while still maintaining the passive heat removal capability.

From the beginning of the development of the ADTT program, the discussion has continued as to whether the ADTT system requires a geologic storage facility as back-up for the untransmuted waste. The purpose of this section is to address the question of the requirement for geologic storage of remnant waste after destruction of the actinide and the long lived constituents of the fission products. It will be shown here that near-surface storage of this waste might be made consistent with existing NRC and EPA regulations with an addition to the regulations for storage of Cs, Sr, and Kr for about 200 years until they meet low-level radioactivity levels covered by existing regulations.

D.2.1.3.1. Review of Regulations

To begin the discussion, it is useful to review several aspects of near-surface waste storage. Waste destined for near-surface storage is divided into three classifications as class A, B, and C waste.

Class A waste is the most benign and it can be stored at the surface without stabilization. That is, no special precautions must be made to protect the system from natural dispersion mechanisms such as rainfall, wind, etc. There are of course some restrictions such as exclusion from flood plains and from unstable land. The site must be clearly marked, and monitored for 100 years but no fencing is required. After that time it is assumed that controls are no longer operative and that the site should not be dangerous to an inadvertent intruder. An inadvertent intruder is defined in NRC Regulation 10 CFR 61.2 as;

"a person who might occupy the disposal site after closure and engage in normal activities such as agriculture, dwelling construction, or other pursuits in which the person might be unknowingly exposed to radiation from the waste."

Class B waste must be immobilized or contained by components in the waste site that maintain their "gross physical properties and identity" for 300 years. Surface storage is permitted and institutional control is required for 100 years. Productive use of the land during this 100-year period is possible so long as the "integrity and long-term performance of the site are not disturbed." Therefore the site perhaps might be used as a parking lot. Elsewhere in 10 CFR 61 the use of concrete in such systems is suggested and it is proposed later in this report to use that means for immobilization.

Class C is reserved for waste with even greater radioactivity concentrations. This waste also requires stabilized waste forms or waste containers. This waste must be stored at least five meters below the ground surface such that after 500 years the waste would not be a hazard to an inadvertent intruder or to the public health and safety.

There is no absolute limit on the amount of radioactivity which can be emplaced at one site, whether designated Class A, B, or C. The amount is only limited by the radiation released to the surroundings and risk to an inadvertent intruder. The radiation released from the site "must not result in an annual dose exceeding 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public." This release criterion for near surface disposal is the same as that for a single geologic storage system built to confine all of the radioactive waste of the nation. The siting criteria for surface storage of waste are specifically stated in 10 CFR 60 and are easily met so that such facilities can be sited almost anywhere except in flood plains, areas of unstable land, etc. Therefore there could be many such sites and almost certainly many more than one in every state. If there were 100 such sites in the U. S., the total radioactivity burden in a single site could be 1/100 of that at a national central repository. If in addition these sites were receiving the remnant waste from an ADTT system which reduces the long lived constituents by a factor of about 1000, the long-lived radioactive waste burden for any single site would be smaller by a factor of 100,000 than that of a single geologic repository without transmutation. It therefore seems likely that the surface storage sites for remnant waste following transmutation could meet the whole body and specific organ dose limits for a much smaller radiation source term even though the confinement capability of the surface site would be less than that from a geologic site. This probably would

have to be demonstrated on a site-by-site basis.

It might be argued that if the waste is distributed over 100 sites instead of a single geologic site, that more people would be endangered. The spirit of the release limit however is that the dose received be too low to risk harm to the surrounding population. Therefore the same release limits apply to each of the many low-level waste sites as apply to a geologic storage site for the nation's entire commercial spent fuel. The number of people exposed is considered not to be a factor because no member of the public is to be subjected to a dangerous dose from any waste site...either high or low level.

The other type of restriction for the inadvertent intruder into the site relates to the concentration of the waste and to whether the waste stream from a transmutation system meets the concentration limits for class A, B, or C waste. Classes B and C waste require stabilization before emplacement. The method of stabilization chosen for this report is mixing with concrete, a material already mentioned in 10 CFR 60 as appropriate for use in surface storage systems. The stabilization before emplacement as required by the regulations will obviously result in the dilution of the waste. There are no statements in the regulations about the degree of the dilution allowable....only limits regarding radiation release to the surroundings and dose to the inadvertent intruder , which depend on dose concentration. Weakly contaminated dirt which is being cleaned up from some of the sites at Los Alamos and elsewhere can be disposed of in surface storage if the contaminated material meets the regulatory limits for radiation release or dose to an inadvertent intruder. For the remnant waste after transmutation from a 3000 MW_t commercial reactor, we assume stabilization with 50 m^3 of concrete per year. Assuming the waste to be Class C, these blocks which might be 1m $\times$ 1m $\times$ 2 m = 2m^3 would have to be stored under 5 meters of overburden according to 10 CFR 60. If stacked end-on, one year's remnant waste from a 3000 MW_t system would occupy a surface area of 5m $\times$ 5m. The waste from 35 years of operation of the facility would therefore occupy approximately a 30 m by 30 m area. Since the land can be put to some beneficial use, this area would be much smaller than a 3000 MW_t plant's parking lot and could be used as a small part of the plant's parking lot.

It would be correct to argue that stabilization amounts to dilution, but *stabilization is required* for class C storage and *dilution is not forbidden* by the regulations. In fact stabilization, which is required, demands some level of dilution by the stabilization medium. The operating criteria are (1) dose to an inadvertent intruder and (2) leakage of radiation from the site into the surrounding environment.

D.2.1.3.2. Disposition of remnant waste from ADTT systems

For ADTT the issue then is what should happen to the remnant waste stream. This stream may be considered to have three components for the accelerator-driven transmutation of waste system (ATW). We assume here a system designed to deal with the actinide and fission product waste from a single 3000 MW_t LWR thermal reactor destroying the waste at the rate that the waste is being produced in the reactor. The first components of the waste encountered in transmutation are the Uranium, which is the primary constituent of the spent fuel, and the Zirconium cladding. These components probably could be stored for reuse and are discussed later. The liquid fuel system allows the continuous feed of all of the waste left over after Uranium and Zirconium removal including both higher actinide and fission products. The ATW system destroys the higher actinide waste by fission generating an average fission power of about 750 MW_t per 3000 MW_t reactor. The liquid fuel system also allows the continuous removal of fission products. The only actinides which escape are those which contaminate the fission product removal process. We assume that the atom fraction of actinides in the fission product removal stream can be held to 1 part per 10,000 and next compare the actinide loss rate with the Class C criteria for actinides.

This is illustrated in Table I where the first four columns show the isotopes, the annual production rates, the half-lives, and the decay rates. The fifth column shows the decay rate of the 1/10,000 of the actinide escaping from the transmuter through the separations process into the fission product stream. Stabilization of this annually escaping quantity of actinide waste with 50 m^3 of concrete as described above gives the decay rate per gram of column 6. This may be compared with the class C decay rate limits for these actinides given in 10 CFR 61 and shown in column 7. Except for the shorter half-life nuclides ^{238}Pu, ^{241}Am

and ^{244}Cm, the concentrations are about a factor of 10 lower than the limits. For ^{238}Pu we use the limit given in 10 CFR 61.55 for the parent ^{242}Cm. For ^{241}Am we use the limit given for the parent ^{241}Pu. Applying the sum-of-fractions rule for combining the decay rates for several isotopes as given in paragraph (a) (7) of 10 CFR 61.55, the sum is still well below the decay rate limits. Therefore if the separations can be accomplished at the 1/10,000 level, the remnant could be disposed of as Class C waste.

A similar evaluation of fission products is summarized in Table II where those isotopes with half-lives 10 years or greater are listed along with their production rates, half-lives, and decay rates. The isotopes are divided into groups according to the treatment received and each group is discussed below.

1. ^{137}Cs, ^{135}Cs, and ^{90}Sr

These isotopes cannot be handled as near-surface low-level waste and they cannot be transmuted with significant beneficial effect using accelerators. Therefore they must be removed from the waste stream with a separation factor of 10-100. Column 6 shows the decay rate of the isotope left after the separation. Once isolated, the cesium and strontium must be stored until their radioactivity decays by about a factor of 100 or for about 200 years before they can also be disposed of as Class C waste. Containers can be built for containment for this storage period so that geologic storage is not necessary. The cans must be isolated from the public and protected so that they maintain their integrity. They must be stored with passive means for decay heat removal through this storage period. While geologic containment is not required for these relatively short-lived nuclides, they do not qualify for near-surface storage and new regulations must be developed for handling them on the transmutation site or at a central limited period storage site.

2. ^{107}Pd and ^{93}Zr

These nearly noble metals materials are almost benign with long half-lives and weak decay energies. No Class C limit is given explicitly for these in 10 CFR 61.54. However the limit for the semi-noble metal ^{94}Nb is given as 0.2 curies/m^3. It is more chemically active than either Zirconium or palladium and its decay energy is more than a factor of ten higher than both. Therefore we assume that the Class C limit for ^{107}Pd and ^{93}Zr would be at least a factor of ten higher than for ^{94}Nb and use the ^{94}Nb limit increased by a factor of ten to 2 curies/m^3 to estimate the regulatory limit. With this limit no separation of these isotopes from the rest of the fission product waste would be required before storage as Class C waste and transmutation would not be necessary.

3. ^{79}Se and ^{126}Sn

No regulatory limit has been established for these nuclides. We assume their chemical reactivity is comparable to ^{94}Nb as are their decay energies. Therefore we use the Class C limit for ^{94}Nb of 0.2 Curies/m^3. A separation of a factor of ten must be achieved to reach the assumed Class C limit for each of these. These isotopes must be transmuted.

4. ^{99}Tc, and ^{129}I

These nuclides are perhaps the most chemically mobile of the long-lived fission products and regulatory Class C limits exist for them. To reach these limits, the ^{99}Tc must be separated by a factor of 100 and the ^{129}I by a factor of 10. These nuclides must be transmuted.

5. ^{151}Sm

This nuclide exhibits a very weak decay energy and we therefore assume the limit of 2.0 curies/m^3 derived from the established limit for ^{94}Nb. To reach this limit, the separation factor must be about 300. The separated material must then be transmuted.

TABLE I. CHARACTERIZATION OF THE LONG-LIVED FISSION PRODUCT WASTES

Isotope	Annual Production (Atoms/a) $\times 10^{25}$	Halflife (years)	Decay Rate (Curies)	Separation factor and decay rate (Curies)		Concentration After Stabilization[a] (Curies/m^3)	Class C Decay Rate Limit (Curies/m^3)
^{79}Se	0.13	65,000	11.9	10	1.2	0.024	0.2
^{90}Sr	9.0	29.1	1,860,000	10	190,000	3800	7000
^{93}Zr	15	1,500,000	60	1	60	1.2	2.0
^{99}Tc	15	213,000	426	100	4.26	0.85	3.0
^{107}Pd	4.1	6,500,000	3.8	1	3.8	0.076	2.0
^{126}Sn	0.46	100,000	27	10	2.7	0.054	0.2
^{129}I	2.7	15,700,000	1.0	10	0.1	0.002	0.08
^{135}Cs	4.2	2,300,000	8.4	100	0.084	0.0017	0.08
^{137}Cs	14	30.2	2,800,000	100	28,000	560	4600
^{151}Sm	0.16	90	11,000	300	37	0.74	2.0
^{85}Kr	1.0	10.7	560,000	1	560,000		

[a] Stabilized with 50 m^3 of concrete

TABLE II CHARACTERIZATION OF THE LONG-LIVED ACTINIDE WASTES

Isotope	Annual Production (Atoms/a)	Half life (Years)	Decay Rate (Curies)	Reduction by Separation[c] (Curies)	Concentration After Stabilization[d] (Nanocuries/gram)	Class C Decay Rate Limit[a] (Nanocuries/gram)
^{238}Pu	1.13×10^{25}	88	7.5×10^{4}	7.5	68	20,000[b]
^{239}Pu	41.6×10^{25}	24,100	1.0×10^{4}	1.0	9.1	100
^{240}Pu	19.2×10^{25}	6,560	1.7×10^{4}	1.7	15.	100
^{241}Pu	6.4×10^{25}	14.4	2.7×10^{6}	270.0	2454.5	3500
^{242}Pu	3.9×10^{25}	375,000	6.1×10^{1}	.006	0.05	100
^{237}Np	3.7×10^{25}	2,140,000	1.1×10^{1}	.001	0.009	100
^{241}Am	4.1×10^{25}	433	5.5×10^{4}	5.5	50.	3500[b]
^{243}Am	0.73×10^{25}	7,370	5.9×10^{2}	.059	0.54	100

[a] Used decay limits from 10 CFR 61.55

[b] Used decay limit for parent from 10 CFR 61.55

[c] For a separation factor of 10,000

[d] Stabilization with 50 m^3 of concrete

6. ^{85}Kr

This noble gas is difficult to transmute because its cross section is small and gas-containing systems inside a nearly critical system must be avoided for criticality safety reasons. According to regulation, it must therefore be stored in 100 Curie or smaller amounts in separate containers with a volume of about 100 ml. These containers must then be immobilized in the Class C waste in accordance with 10 CFR 61.54. There are no regulatory limits to the number of containers, but 5600 would be required per year. It would therefore probably be preferable to collect the gas in yearly production volumes of about 10 m^3 and store it along with the cesium and strontium. After 200 years the container confining the remnant could be stored as Class C waste according to regulations.

Of the eleven long-lived fission products, two require no action. The other nine must be separated using eight chemical separations and five of these must be transmuted and stored as Class C waste with the other fission product. The remaining three (Kr, Sr and Cs) must be placed in engineered storage for about 200 years. After the 200-year period, the latter four can be stored permanently as Class C waste also. The five isotopes to be transmuted constitute about 6 % of the fission product and will require about 300 moles of neutrons per 3000 MW_t-year of reactor operation. These neutrons may come either from an accelerator or from the excess neutrons produced by the fission of weapons Plutonium or highly enriched Uranium. Once the five long-lived fission products have been destroyed, the remnant fission product waste can be diluted and stored in concrete at the rate of 50 m^3 per year per 3000 MW_t fission power. For the Los Alamos Thorium burner, which transmutes its own fission product and produces 200 MW_e (500 MW_t) for 35 years, the subsurface storage area required for Class C waste immobilized in concrete if stored two meters thick would be about 12 m × 12 m.

The Uranium and Zirconium cladding are nearly benign materials and could be stored in containers at some central site for probable future use. There is no apparent reason now to place them in geologic storage where they would be almost inaccessible by definition.

In summary, with transmutation and separations factors which need not exceed 10,000 and more nearly 1,000 for actinides and about 100 for fission product, remnant waste would not require geologic storage. For the on-site transmutation of the waste from a commercial nuclear power plant, the fission product immobilized in concrete could stay on the reactor site as Class C near-surface waste. The Cs, Sr, and Kr could stay or be moved in accordance with state and local government decisions. Without the need for a central geologic repository, the federal government need not become involved in the siting of waste storage facilities. Its role would be limited to providing the regulatory framework for near-surface storage.

D.2.1.4. WEAPONS PLUTONIUM DESTRUCTION (ABC SUBPROJECT OF ADTT)

This is the first of the three applications which were mentioned at the beginning of this paper and has been pursued under the acronym ABC for Accelerator-Based Conversion. Excess weapons Plutonium (w-Pu) is being made available by major reductions in the U. S. and Russian stockpile of nuclear weapons and by the clean-up of U. S. and Russian w-Pu production sites. Altogether more than 100 tons of this material exists [1] with perhaps 20 % of it being material reclaimed from the production sites. The ultimate disposition of w-Pu has been the subject of recent intensive study in the U. S. Basically three options are considered: (1) burning of the Plutonium to the point where it has roughly the same isotopic composition as commercial Plutonium (c-Pu), referred to as the "spent fuel standard" followed by geologic storage, (2) geologic storage of the w-Pu without burning after vitrification with defense radioactive waste, or (3) complete burn-up of w-Pu.

Since there is about ten times as much c-Pu as w-Pu in the world today and the c-Pu is increasing rapidly, present U. S. policy appears to favor burning the w-Pu to the spent fuel standard. The advantages of this seem to be that the w-Pu then becomes a small increment on the already larger c-Pu inventory, the

w-Pu is less effective as weapons material, the radioactivity of the burned w-Pu is a deterrent to the handling of this material in nuclear weapons fabrication, geologic storage of the burned Plutonium makes it much less accessible than it now is, and the technology to burn the w-Pu to the spent fuel standard exists now. The arguments against the spent fuel standard are that the resulting material is still quite effective for weapons construction, that it probably could be recovered from geologic storage without great difficulty, and that there is very little near-term political advantage because it will probably take 30-50 years to complete the conversion to the spent fuel standard and the placement of the material in geologic storage.

Perhaps most importantly, disposing of the material this way costs money or yields negative value from the w-Pu whereas there are clearly large positive-value uses for this material for start-up of the ADEP system and for ADTT commercial waste destruction. The destruction of commercial nuclear waste requires supplemental external neutrons all of which could be supplied by an accelerator. However the neutrons could also be supplied by fission of weapons material. The weapons materials are valuable for weapons precisely because they are an excellent source of neutrons. Each fission of ^{239}Pu produces 2.88 neutrons of which one per fission must be used to sustain the chain reaction. An additional 0.35 per fission are lost because not all neutron absorptions in ^{239}Pu lead to fission and a total of about 10 % of the neutrons per fission are lost to parasitic capture and leakage. After subtracting off these losses of neutrons, one is left with about 1.2 excess neutrons per fission available for other uses. The number of neutrons from HEU is slightly smaller. In the burning of commercial waste using the ADTT technology, the accelerator supply of neutrons can be reduced by about a factor of two by the use of w-Pu or HEU. Since the accelerator source can be reduced significantly and we know roughly what the cost of the accelerator-produced neutrons is, the price which could be paid for w-Pu and HEU in this application can be estimated from the savings in cost of the accelerator, which are relatively well known. The value for w-Pu is found to be perhaps as high as $250,000 per kilogram[3]. This value is far more than the value of HEU blended down for commercial reactor fuel. An even higher price could be paid for w-Pu and HEU for the one initial load required for the ADEP system without bootstrapping from the commercial grid using the accelerator.

If one compares the present inventory in the U. S. and Russia of w-Pu and HEU to the amount required for destruction of the world's nuclear waste, there is a surprisingly good match, so that all of these materials could be used for commercial waste destruction. It can be argued that the price quoted above is artificially high because HEU can be separated from natural Uranium at a much smaller price and that therefore a major need for either w-Pu or HEU would be satisfied by lower priced newly produced HEU. However continued production of HEU would not be consistent with international agreements to forego the enrichment of Uranium to HEU when much smaller enrichments are quite sufficient for use as fuel in all of the world's commercial nuclear power plants. International political agreements therefore probably would make it difficult or impossible to produce HEU for commercial waste destruction. Nevertheless, a user of HEU or Plutonium would argue effectively against paying the high accelerator-displacement value when it could be produced anew much more cheaply. A value higher by a factor of two than that for new HEU might be paid for existing HEU or w-Pu in which case the 100 tons of w-Pu might be valued at about $50 billion and the ten times larger amount of HEU at about $500 billion. Such high positive values for these weapons materials would be good news from the perspective of weapons material security since we willingly guard our valuables and grudgingly pay to dispose of our waste. Fortunately Russia still considers its weapons material valuable and we can expect that it will be more carefully guarded if the U. S. policy is directed toward maintaining the high value perspective. Furthermore, since the value for the material is not received until the weapons material is sold for the desired purpose, one can expect the desire for converting the book value to real value to drive the sale of the material as soon as the waste destruction facilities are able to use it. The temptation to hold on to the material for weapons purposes is countered by the high value which could be obtained when it is sold.

Quite obviously these arguments for use of the weapons material for high value purposes are inconsistent with w-Pu destruction which is the purpose of the discussion in this section of the paper. None of the three options for near-term negative value w-Pu disposition identified by the National Academy Study [1] would be favored from the perspective of ADTT. This is especially true since the burning of w-Pu (or HEU) produces many more neutrons than are required to sustain a chain reaction so that the main

purpose of the accelerator, which also is to produce surplus neutrons, is superfluous. The accelerator is however useful if high burn-up of the Plutonium is required so that there is virtually no Plutonium in the waste stream and the isotopic composition is incompatible with use of the remnant as weapons material.

Thus the Los Alamos National Laboratory has proposed an accelerator-driven subcritical system [4] in which fission product poisons are allowed to build up they consume not only the excess fission neutrons from w-Pu fission, but also the supplemental neutrons from the accelerator. The system achieves very high burn up without fuel reprocessing or fuel fabrication and refabrication. Also no chemistry for fission product removal is required. The General Atomic Corporation has proposed a program with a similar objective. Its helium-cooled graphite-moderated reactor with w-Pu fuel particles suspended in the graphite has been proposed as the first stage of w-Pu destruction. After the Pu has been burned sufficiently that it will not sustain criticality, the fuel is transferred to an accelerator-driven assembly which continues to destroy the Plutonium using accelerator-generated neutrons until k_{eff} of the system has dropped to about 0.6. The burn-up of the Los Alamos and the General Atomic systems are similar and are the highest of any of the proposed w-Pu-burning systems; neither require fuel reprocessing or fuel refabrication. Present U. S. DOE policy towards w-Pu burning seems to be to burn the Pu only to the spent fuel standard. The Los Alamos ADTT Project Office position is that preferably the w-Pu either should be burned completely or reserved for enhancing commercial spent fuel waste transmutation as described above with the latter choice much preferred.

D.2.1.5. ACCELERATOR-DRIVEN ENERGY PRODUCTION (ADEP SUBPROJECT OF ADTT)

Perhaps the most important element of the ADTT project in the long term is the Accelerator-Driven Energy Production (ADEP) which uses Thorium as a nuclear fuel. The system is based on the Th-U cycle in which ^{232}Th is converted by neutron capture to thermally fissile ^{233}U. This cycle has been studied extensively [5] for use in commercial nuclear reactor power generation. The primary objective of the molten salt reactor experiment was to show that an effective breeder reactor could be built on this cycle which produced more ^{233}U than it consumed. This reactor technology lost out to the fast breeder based on the U-Pu cycle because its breeding ratio was barely larger than unity even when fission products were promptly removed from the fuel. The U-Pu cycle showed much higher breeding ratios at a time when Plutonium was in demand rather than in excess.

A major advantage in the present climate is that the Th-U cycle produces almost no Plutonium. The Th-U cycle development program also focused on a molten salt liquid fuel program with on-line removal of fission products, and the operation of a liquid fuel reactor was demonstrated with the several-year Molten Salt Reactor Experiment (MSRE) at the Oak Ridge National Laboratory. Not only could fission products be continuously removed from this system but the liquid fuel allowed the reactor to be continuously refueled. For this reason the MSRE still holds the world record for the longest continuous chain reaction. A great deal of successful research was done on the materials to contain the salt and all of the ADTT projects rely on the materials work done for the MSRE. While the MSRE had virtually no actinide waste stream, it had the usual fission product waste and its neutron economy did not allow it to breed as much ^{233}U as it burned and still have excess neutrons left for transmutation of its fission products.

By preserving many of the design features of the MSRE and introducing an accelerator into the system, one achieves the capability to produce as much ^{233}U as is burned so that the nearly unlimited energy available in Thorium can be accessed. In addition the extra accelerator-produced neutrons enable the long-lived fission products to be avoided so that there is no long-term high-level waste stream from this system. Because of the subcriticality of the system a runaway chain reaction can be made much smaller than any reactor and perhaps the probability for such an event can be reduced truly to zero. These three features of "unlimited" energy, criticality safety, and absence of high-level waste are the highly touted features of fusion systems which have been heavily studied for the many years. We believe that we can demonstrate these benefits to society during the coming decade by merging established reactor technology with the existing highly developed accelerator technology. The system produces almost no Plutonium and it is has

excellent non-proliferation features. This system has already been described in some detail at the beginning of this report so we will concentrate mostly here on the non-proliferation features which are of vital importance for any new nuclear power system

All existing commercial reactors for production of nuclear power produce Plutonium as a by-product which is seen by many as an asset because of the additional power which can be derived from it. Others see it as a serious liability since it can be used for nuclear weapons and because of radiological concerns. The established means for separating the Plutonium for reuse in reactors produces a stream of "naked" Plutonium. This Plutonium is pure and unmixed with other material which would inhibit its usefulness in nuclear weapons. This material might be diverted in the separation facility, in storage, in transport to fuel fabrication facilities, etc. There is the fear that in some countries it will simply be stockpiled for planned or possible future use in nuclear weapons. Therefore the U. S. has followed a policy of discouraging the reprocessing of commercial spent fuel and the use of Plutonium for energy generation.

Instead the U. S. and Sweden follow a once-through cycle where the spent fuel would go directly from reactor storage to geologic repository storage. Some are concerned about the consistency of U. S. policy if the once-through policy is proposed as a waste management solution, which will promote the much greater use of nuclear power throughout the world. In that case there eventually would be many repositories spread all over the world which could be mined for Plutonium. Furthermore the reactor-grade Plutonium decays into weapons-grade Plutonium. Therefore neither reprocessing, as it is presently performed, nor once through geologic storage are entirely satisfactory solutions. The ADEP program offers the opportunity to have the benefits of nuclear energy without the weapons potential from Plutonium or other material which could be used for nuclear weapons.

The ADEP system is fed ^{232}Th and transforms it to ^{233}U which is then fissioned to obtain the nuclear electric power. After a stable equilibrium is reached, there will always be a fixed amount of ^{233}U in the system which might be accessed for nuclear weapons.

A number of non-proliferation features of the ADEP system will be described below, which limit the amount of ^{233}U available to a much lower amount than ^{239}Pu in current LWRs. They also limit its accessibility, allow simple detection of any diversion attempt, and allow low impact actions to forcefully terminate any diversion which is underway, if necessary.

Limiting the amount of fissile material present

Fast reactor technology which is being pursued in many countries around the world carries a large inventory of Plutonium. The fundamental reason for this is that the fission cross section for ^{239}Pu in the fast neutron spectrum is smaller by about a factor of 100 than that for thermal spectrum fission of ^{239}Pu. Therefore, other things being equal, the inventory for the thermal spectrum system is smaller by about a factor of 100 than for a fast spectrum system. The neutron flux for the thermal system is about a factor of ten smaller so that as a practical matter the thermal system requires about 10-30 times less material than a fast spectrum system. The same situation is true for ^{233}U when fast and thermal spectrum systems are compared. Generally speaking the ADEP system will carry about the same amount of ^{233}U as an LWR has of ^{235}U and ^{239}Pu together if the flux and power level are the same. The primary point here therefore is that the ADEP system carries a much smaller inventory of potential weapons material than the fast reactors under development in other countries.

Isotopic dilution of ^{233}U in ADEP

If the ^{233}U were diluted with ^{238}U to the 20 % level or lower, the ^{233}U would be classified as non-weapons material according to present regulations. A 500-MW_t thermal ADEP system can be brought immediately into power production by a start-up inventory of 10,000 kg of Th and 700 kg of 20 % low enriched Uranium (LEU) where the 20 % is ^{235}U. The original ^{235}U will be burned out over time and replaced with ^{233}U derived from the Thorium. The distribution of isotopes reached after ten years of operation is given in Fig. 3 where the amount is given in grams. At ten years, which is essentially

equilibrium, the Uranium fissile material inventory will be 100 kg of ^{233}U along with 10 kg of ^{235}U for a total fissile content of 110 kg. The amount of ^{238}U present at this time is about 600 kg so that the required isotopic dilution of about 20 % is maintained. However the inclusion of ^{238}U in the systems will result in the production of a small amount of ^{239}Pu. The isotopic distribution of Plutonium as 239, 240, 241, 241, and 242 is present in the amounts of 1.2, 1.2, 0.3, and 2.5 kilograms. The ratio of fissile to total Plutonium is 0.29 so that the Plutonium would be very poor quality weapons material and there would be only 5.2 kg of Plutonium altogether to be accessed.

"Raiding" the ADEP for ^{233}U through ^{233}Pa

The conversion of ^{232}Th to ^{233}U is a three-step process involving neutron capture by ^{232}Th to produce ^{233}Th which decays almost immediately to ^{233}Pa, which itself subsequently decays with a 26-day half-life to ^{233}U. Fig. 3 shows that the inventory of ^{233}Pa in the system is about 22 kilograms. If operation of the ADEP system were interrupted and the molten salt removed, it would be possible in principle to separate the ^{233}Pa before it decayed to ^{233}U from the 8000 kg of other actinide. If such a separation could be completed in about 26 days, about half of the ^{233}Pa could be recovered. When this half decayed to ^{233}U, the 11 kilograms of ^{233}U resulting would be useful weapons material. The separation in question would be a dangerous activity in view of the very high radioactivity of the salt so soon after shut-down. Ordinarily spent reactor fuel is allowed to decay 300 times longer (about 10 years) before separations begin. A further operational factor would be that the value for k_{eff} for the system would have dropped to about 0.85 from the normal value of about 0.95 by the removal of the 11 kg of ^{233}Pa. This may be compared with about 100 kg of Plutonium which could be recovered from the interruption of operation of an LWR operating at the same thermal power level. A fast spectrum reactor of similar power would carry about 1000 kg of accessible Plutonium.

With the removal of the 11 kg of ^{233}Pa, the thermal power level would have decreased by a factor of three and the net electric power into the commercial grid by a factor of about five while the accelerator power would have remained the same. The power level would recover over a period of several months, but the inconsistency between the accelerator power and the electric power output would be readily observed by infrared mapping from satellites or by other means.

Benefits from "lock-up" of ^{233}Pa

There are two disadvantages if the ^{233}Pa is allowed to circulate freely in the salt. The first is that a "raid" on the ^{233}Pa might be started by draining the salt, although the follow on separations would be exceedingly sophisticated and dangerous. The second is that performance degradation through neutron capture on ^{233}Pa limits the flux to about 2×10^{14} n/(cm^{2}s). The gain from internal isolation of the ^{233}Pa during its decay period would offer many benefits in overall system performance.

Start-up without fissile material

There might be nations which could benefit greatly from nuclear power but which are considered to be substantial proliferation risks. In those cases providing non-radioactive LEU at 20% enrichment to start up the system might be considered a proliferation risk in that much of the enrichment towards highly enriched Uranium has already been done. The start-up load might be diverted for enrichment for weapons use instead of being used for its intended purpose. The ADEP system can be brought into operation with no fissile material at all. For a system initially containing only ^{232}Th, the accelerator can be powered off the commercial grid and the neutrons produced used to produce ^{233}U. As the fission of the ^{233}U increases, the neutron flux also increases generating even more ^{233}U so that over a period of six to twelve months the system bootstraps itself to full power. No reactor existing or under development can operate with absolutely no fissile fuel load.

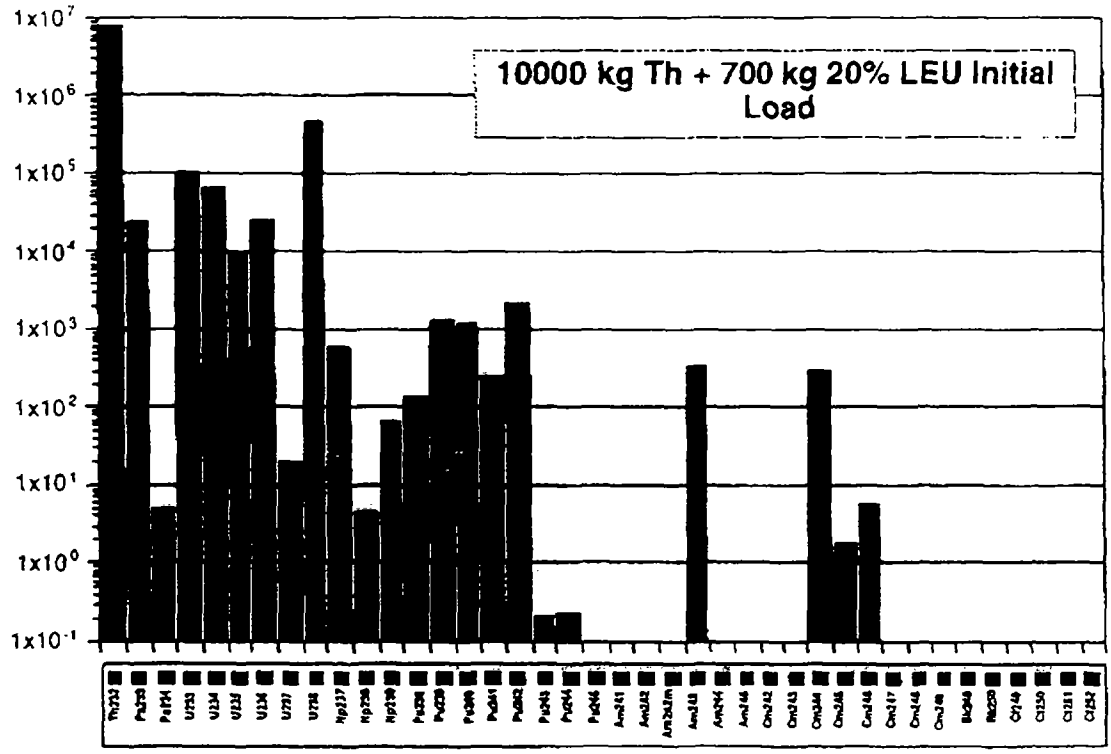

FIG. 3. Actinide isotopic distribution for the ADEP system. The distribution of actinide isotopes for the ADEP system is given ten years after start-up long after the system has reached equilibrium. The fission power level of the system is 500 MW_t. The system was started with 10,000 kg of Thorium and 700 kg of 20 % enriched Uranium. The inventory is given on the ordinate in grams for isotopes from ^{232}Th to ^{246}Cm. The ^{233}U is diluted with ^{238}U such that the ^{233}U is never useful weapons material. The amount of Plutonium in the system is very small and is of a very poor isotopic ratio. The ^{233}Pa decays in 26-days to ^{233}U. As described in the text it might be possible, using heroic measures, to extract a fraction of the 22 kg present. The loss of this material would have a significant effect on k_{eff} and probably could be readily detected remotely.

Remote detection of anomalous operation and possible diversion

All conventional nuclear power systems deployed or under development use solid fuel which must be enriched, fabricated, brought to the site, burned, stored, eventually removed from the site, perhaps reprocessed, returned to the site, and finally placed into a repository. Each of these transfers might require a measurement to confirm the amount of fissile material present in the system. If each measurement could be done to 1 % accuracy and nine were required, the total uncertainty over the fuel cycle for nine independent measurements would be about $9^{1/2} \times 1\% = 3$ %. Since a 3000-MW_t reactor typically burns about 1200 kg of fuel per year, the real uncertainty in the fissile fuel in the system is about 36 kg. About half of this might be Plutonium which could be diverted into nuclear weapons without being missed.

In contrast, no power is generated in ADEP without the operation of the accelerator and all of the fuel is generated internally. No actinide must be removed from the system in the course of normal operation of the system. The accelerator beam power, the fission power, the electric power generated, the electric power consumed by the accelerator and the plant, and the power fed into the external grid must all be internally consistent. If the accelerator power is increased, all of the other power levels must increase in a fixed relationship. If the plant is found to be operating out of balance, for example by power meters at the strategic points or by satellite infrared mapping, it is a signal that material diversion might be

underway. A more detailed study of these anomalous conditions and their dependence on the rate of feed of Thorium and the rate of removal of fission products might provide means to sense remotely when the source of the anomaly is nuclear material diversion.

Limited consequences of extreme measures to control diversion

If an existing operating reactor is suspected or determined to be used for production of nuclear weapon material, the ultimate response by those alarmed could be the destruction of the nuclear reactor. There is a significant possibility that such action could lead to widespread death for the surrounding public and land contamination near the reactor. With such consequences the destruction of a reactor after it has begun operation is probably impractical. The accelerator component of the ADEP system is large and easily damaged into inoperation without significant possibility of damage to the target-blanket itself and the release of radiation. Diversion can therefore be terminated without exposing the surrounding population to significant danger.

D.2.1.6. COMMERCIAL WASTE TRANSMUTATION (ATW SUBPROJECT OF ADTT)

The objective of the Accelerator Transmutation of Waste (ATW) subproject of ADTT is to destroy the actinide and long-lived fission product waste from commercial nuclear reactor spent fuel. If the separations can be done sufficiently well, storage of the remnant waste could be in near-surface sites rather than in geologic storage facilities. The amounts of material requiring transmutation and the selectivity of chemistry separations has already been described in the section of this report entitled, "Geologic Storage and the ADTT System." Separation factors of about 1/100 are shown to be adequate to meet Class C storage criteria for fission product and about 1/10,000 for the Plutonium and other minor actinides.

The ATW system also has means for continuous feed of waste from commercial light water reactors. To many this would appear to require the separation of Plutonium and other components of the waste before feeding them into the system. This is referred to as reprocessing which was forbidden in the U. S. by President Carter by Executive Order. Even though this order expired when he left office, as a practical matter it has continued to govern U. S. internal policy on spent fuel and our foreign policy position has strongly attempted to discourage the reprocessing option for commercial spent fuel. The purpose is to reduce the opportunity for diversion of commercial Plutonium to nuclear weapons purposes and to prevent the accumulation of large inventories of this material which is considered by many to be highly dangerous. Because of the excess neutrons provided by the accelerator, front end reprocessing is not required. The ATW system would require only the removal of the Zirconium cladding and the Uranium. All of the other actinide and all of the fission product can be fed into the blanket, because the capabilities for removal of the fission product already exist in the back-end separations system.

The front-end removal system has not been selected but there are at least two options under consideration. One would involve the crushing of the spent fuel assemblies which contain mostly UO_2 and the oxidation of this to U_3O_8. The volume expansion on the transformation to a higher oxide and the resulting conversion of the spent fuel to fine powder allows the spent fuel to be poured out of the spent fuel assemblies. Separation of the spent fuel from the cladding might approach 99 % for this process, but that might not be adequate and it might be difficult to clean the hulls further. Another means of removing the cladding might be to burn the spent fuel assemblies in a chlorine atmosphere over a plasma torch converting the Zirconium to volatile $ZrCl_4$. The oxide in the cladding however would fall as rubble into the bottom of the chlorination facility and be collected for subsequent fluorination. The bulk of the spent fuel is Uranium and this would be removed as volatile UF_6. All of the other spent fuel material including the fission products, the Plutonium and other higher actinides would be converted to fluorides and fed directly into the ATW system by dissolving them in the molten salt carrier.

In contrast to the aqueous reprocessing system developed long ago and now in common use, the processes described do not produce a pure stream of "naked" Plutonium. The Plutonium is never separated

from the most highly radioactive components of the spent fuel, but only from the relatively benign Zirconium and Uranium. The front-end separation required for the ATW therefore produces a stream which is mostly highly radioactive fission product and separation of the Plutonium from this fission product and from the other actinides would be required before it could be used in weapons. It is also important to mention that the front-end separations for the ATW system would be an integral part of the ATW system so that product stream from the Zr and U removal would be difficult to access.

Commercial nuclear power plants are typically sized at 3000 MW thermal and produce about 300 kg of Plutonium and other higher actinide per year while fissioning 1200 kg of fissile material per year. Therefore an ATW system operating at the same fission power level of the LWRs could burn the waste from four LWRs if its operating life were the same as the LWRs. Destroying the LWR waste arising from the roughly 100 LWRs in the U. S. using ATW systems would require the deployment of about twenty five 3000-MW_t ATW systems if the waste were to be destroyed in about 30 years. Unless the income from electric power sales were sufficient to offset the capital and operating costs of the ATW system, the cost of destroying the waste by this means could be prohibitive. The economic picture for the ATW system will be less favorable than for the ADEP system because the ADEP system need only destroy its own waste and only a modest accelerator is required for the modest neutron supplement. However the ATW system must destroy not only its own waste but also that from the four LWRs. Substantially more accelerator-produced neutrons are required therefore with greater capital cost for the larger accelerator and for the additional power which the accelerator consumes.

There is an attractive way around this. Presently the U. S. and Russia are reducing their weapons stockpiles and freeing up large amounts of highly enriched Uranium and Plutonium. Both are excellent sources of neutrons, which is part of the reason why they are ideal weapons materials. Over the long run it is probably unsafe to store these materials and so they will have to be destroyed almost certainly by fission. If some of these weapons materials are consumed in the ATW system, the excess neutrons can make up for some of the neutrons which otherwise would have to be supplied by the accelerator. Therefore by burning these weapons materials concurrently with the destruction of the commercial nuclear waste, the size of the accelerator probably can be reduced by at least a factor of two. With the resulting benefit to the economic picture for the ATW system, the destruction of the waste using the ATW system might be practical. A comparison between the amount of LWR waste and the amount of excess weapons material available shows that there is a satisfactory match.

There are other practical matters concerned with the practical deployment of the ATW systems. These systems will probably have to be located at government reservations and operated in clusters both because of the sheer size and the use of the weapons material. If four ATWs were located on the same site, the electric power output into the local commercial grid would be about 3-4 GW_{el} from each reservation and there would have to be about three sites if all of the waste was to be burned in 60 years. It is not easy to reliably estimate the U. S. power requirements over the next 30-50 years and how the power will be produced, but having access to a commercial market for the electric power from the ATW systems is an important consideration for this deployment option for the ATW system.

There is a second ATW deployment option for the destruction of the LWR waste which is a hybrid of the ATW and ADEP systems. This would involve the replacement of existing LWRs at the end of their life with an ATW system on the same site feeding the same amount of electric power into the grid. The ATW system would over its life destroy the waste from the LWR and also its own waste stream. About 25 % of its power would be derived from the actinide waste from the LWR and the rest from Thorium. The accelerator requirement would be about the same as that for the other ATW deployment option, but no weapons material would be required. Of course it probably would not be desirable to have these weapons materials being delivered to the approximately 100 ATW systems operating in follow-on to the existing 100 LWRs. An advantage of this deployment scenario is that the waste need not leave the site, some level of radioactivity inventory already exists on the site, and there is probably a clear market for the ATW electric power and an existing distribution system. Under this scenario, the amount of nuclear power would continue to be at least as large as that produced today. The present system would have been replaced with systems which do not produce the waste stream of existing LWRs, which avoid the criticality and after heat

safety concerns of existing reactors, and which nullify the requirement for a nuclear infrastructure of mining, enrichment, fuel fabrication, reactor, fuel storage, reprocessing plants, and fuel refabrication. The requirements for geologic storage of remnant waste would be greatly reduced or perhaps made entirely unnecessary depending on the technical and economic performance of the system.

In summary, the first ATW deployment option carries more of the features which might be associated with a nuclear close-out option. The second option could provide a bridge over the next 30-50 years from the present LWRs with their major infrastructure requirements to the ADEP systems which operate with little infrastructure support .

D.2.1.7. PRESENT STATUS AND SUMMARY

This paper describes a new accelerator-based nuclear technology which offers total destruction of the weapons Plutonium inventory, a solution to the commercial nuclear waste problem which greatly reduces or perhaps eliminates the requirement for geologic waste storage, and a system which generates potentially "unlimited energy from Thorium fuel while destroying its own waste and operating in a new regime of nuclear safety The accelerator technology is already rather mature after 50 years of development and is being driven by other programs. Reactors are also well understood after 50 years of development of many different reactor types. The next essential step in the ADTT program is demonstration of the successful integration of reactor and accelerator technology in an experiment of significant size. Such an experiment has been proposed for the Los Alamos Meson Physics Facility (LAMPF) at Los Alamos and for the Moscow Meson Factory at Troitsk, Russia. For a system operating at a $k_{eff} = 0.96$, LAMPF would drive the system at a power level of 40 MW_t thermal. Of course lower powers are contemplated for the earlier stages of the experiment which might extend over about seven years including both construction and operation. The experiment would be accompanied by research and demonstration, at about the same technical effort as the experiment, on the required separations in the molten salt context. Perhaps seven years hence, an integrated demonstration of the ADTT system could be in operation at the 200 MW_t level, with the deployment of the ADTT system beginning in about fifteen years. This time scale is approximately the same as the earliest planned opening of a geologic repository in the U. S. or elsewhere.

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