Waste Transmutation by High Flux DT Fusion Neutrons from Inertial-Electrostatic Fusion (IEF) Systems

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Abstract

Studies have been made of fission product (FP) transmutation by irradiation with DT fusion neutrons at very high fluxes and fluences, from unique, compact, replaceable IEF fusion systems. These use quasi-spherical electrostatic potential wells to focus fusion ions to high densities and reaction rates. Such a device operating at a fusion power P_f can "burn up" FP waste from fission reactors with total capacity about $20P_c$. The add-on cost of such burn-up is about 5% of the base cost of power, ignoring sale of power from the burner plant, itself. Both short- and long-lived FP isotopes can be destroyed by such systems. The actinides are gone within a few weeks, while the Sr^{90} and Cs^{137} isotopes require neutron fluences of ≈200 MW/m² to reach residual hazard storage lives markedly less than 100 years.

Introduction and Background

Fission product transmutation resulting, from irradiation by neutrons has been studied since 1973.[1,2,3] Fusion neutrons from reactions between deuterium (D) and tritium (T) are of interest for this application to transmutation waste burning (TWB) if they can be supplied at very high fluxes and fluences. This fusion reaction yields 17.6 MeV. of which 14.1 MeV is carried by a fast neutron produced in the fusion process. The remaining 3.5 MeV is in the alpha particle (^4He) product of the reaction. These high energy neutrons can be used to generate even larger numbers of neutrons at lower energies in an appropriate FP waste-bearing blanket around the neutron (fusion) source, by multiplication in blanket nuclei that allow (n, 2n / 3n / etc) reactions (e.g. Be).

To be effective in the TWB process, these neutrons must be thermalized to reach the large (1/v)-dependent thermal capture cross-sections at which they can best be utilized. Thus an optimum blanket must contain a good slowing-down material as well as a neutron multiplier, in addition to the fission waste that it is desired to transmute. Graphite as blanket thermalizing material is less effective than Be, as it docs not give neutron multiplication, and will require somewhat higher blanket first wall fluxes. The use of fission multipliers (e.g. ^{238}U) is unfavorable because of their added in-situ FP production and associated increased hazard and complexity in blanket design.

It is important to note that the location of the blanket with respect to the fusion source plasma and its confinement structures can be a dominating factor affecting the overall performance of the TWB plant. With the exception of very compact high-power density tokamaks, most magnetic fusion confinement concepts are forced to interpose the blanket between the plasma vessel and the large superconducting confinement magnets in order to obtain adequate blanket performance (for required tritium breeding) and to minimize radiation damage to the magnets. This forces the blanket to operate inside the complex irradiated structure of the fusion system. and makes its maintenance and replacement hazardous, complex and time-consuming.

However, these older, large-scale, hypothetical fusion systems were conceived for electric power production rather than for efficient disposal of fission wastes. A variety of past studies[1,2,4] have shown that the simplest operation of fusion systems for TWB is possible only when their blankets are located completely *external* to the fusion neutron source device. This is not possible with big machines, but is a geometry ideally suited to small-scale high-power-density electric fusion concepts.

II. IEF Concepts and Devices

Analysis shows that fusion-neutron-driven transmutation (or "incineration" can be a powerful and effective means of FP "incineration" for all fission products (i.e. *not* just the actinides) **only** if the neutron flux in the blanket is sufficiently large. Thermal fluxes required are in the range of 1.5 x 10^{17} neutrons / cm^2 sec corresponding to fast fusion neutron fluxes incident on the blanket inner face of approximately 0.2-1 x 10^{16} neutrons / cm^2 sec. This gives blanket wall irradiation at 45-225 MW/m^2 of 14.1 MeV fusion neutrons. for blankets using Be multipliers, that can give significant (n, 2n) multiplication and good neutron thermalization with minimal capture. These correspond to thermal (charged particle) fluxes of about 9-45 MWth/m^2 on IEF and blanket system structures. Cooling at this level can not be handled by conventional stainless steels without excessive thermal stresses, but other materials of higher thermal conductivity (e.g. TZM, copper alloys, et al) can be used successfully.

To reach the neutron flux levels cited above, the neutron source, within the blanket shell, must be very compact and able to operate at very high output fluxes, itself. At these fluxes the first wall material will suffer neutron damage (due to atomic displacement) at a rate that requires its replacement at intervals of about 1-6 months. This will be practical (i.e. economical) only if the neutron source can be made sufficiently small in size that it can be handled as a single module for ease of replacement. and if its replacement cost is small compared to the value of the transmutation accomplished during its life. Similarly, the blanket structures must be able to be replaced at frequent intervals, with minimum down time and at low cost. This forces simplicity and modularity on the blanket/FP-handling system design.

Conventional concepts for DT fusion sources can not meet these requirements. They are all very large in size and power, and very complex in geometry and mechanization (e.g. the complicated structures of large toroidal superconducting magnet tokamaks). Their projected costs are measured in billions of dollars, and their first wall fluxes are all below a few MW/m2, two orders of magnitude too low for the high flux TWB incinerator of interest here.[5] What is needed is a very small, compact, relatively structureless, intense source of DT fusion neutrons, that can be manufactured easily and run to end-of-life with no maintenance much in the fashion of an intense light bulb.

Work done since 1987 has defined such systems,[6,7] based on the principle of inertial-electrostatic-fusion (IEF). In this system, fusion fuel ions are energized and spherically focussed to a high-density central convergence by being accelerated through a spherical negative electric potential. This can be produced either by an electron-injection-driven negative potential well,[8] or by direct acceleration through biased grids.[9] These two approaches to potential well formation and ion-acceleration are called IXL and EXL electric fusion systems, respectively.

Figures 1 and 2 show the general principles of these IEF devices. IXL systems are limited by heating of the mechanical grids required to provide the spherical ion-accelerating potential and are not suited for high-flux TWB application.

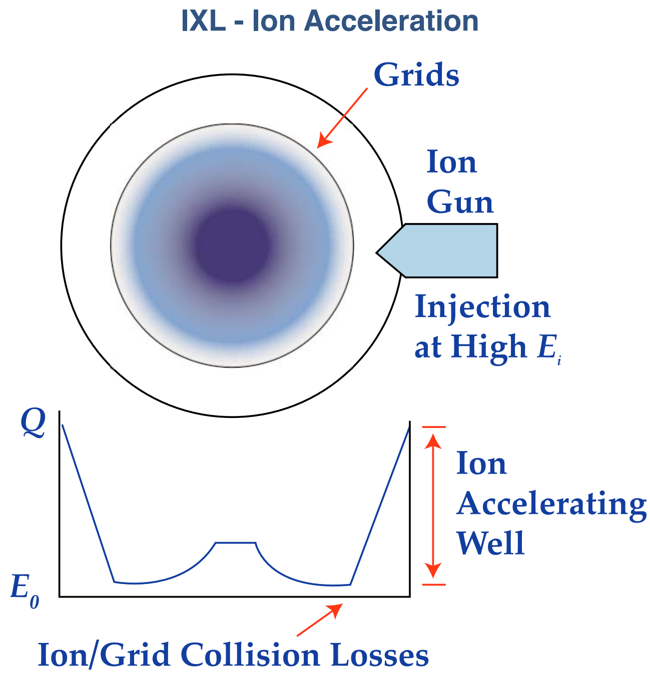

Figure 1 — Ion Acceleration in Spherical Geometry by electrically-biased grids; the IXL System

In contrast, EXL systems have no internal structure, can operate with water-cooled cooper alloy magnets located at the outer boundary of the ion-confining region, and thus are suitable for use in TWB incineration systems.

EXL - Electron Acceleration

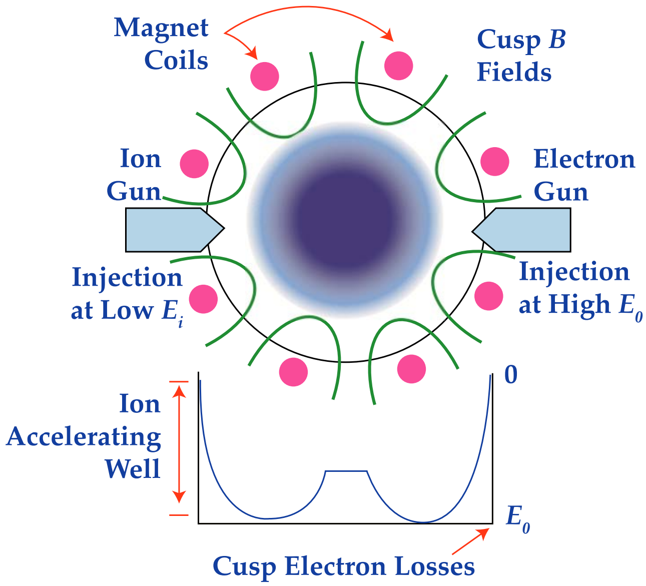

Figure 2 — Ion Acceleration in Electron-Driven Negative Potential Well; the EXL System.

All of the fusion power and neutrons originate in a small (a few cm diameter) central "core" of reacting ions in the center of these devices. Neutrons from DT fusion will readily escape from the IEF source region and exit radially towards any blanket structure placed around the unit. The detailed features and characteristics of this concept for electric fusion are discussed in several reports[10,11] of prior research and development work on the concept. Figure 3 shows an example of an EXL fusion device mounted on a removable base plug, inside a cylindrical blanket. All drive power, fueling, cooling water, control and vacuum lines go through the base plug to a radiation-safe tunnel below the power unit. This allows easy quick-disconnect operations for rapid IEF unit replacement at end-of-life.

IEF Unit Inside TWB Irradiation Blanket

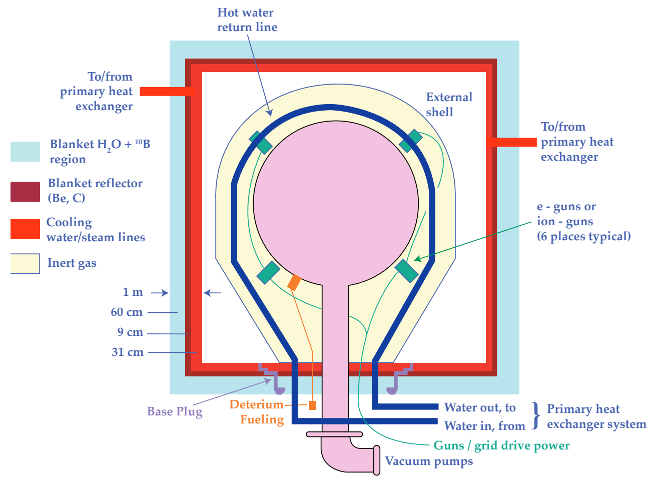

Figure 3— IEF Unit Inside TWB Irradiation Blanket

Figure 4 shows a TWB reactor building using three replaceable baseline IEF power cell units in a modular array for power generation and waste burn-up. and Figure 5 shows an artist's conception of a 3300 MWth TWB plant based on use of 1100 MWth modules.

IEF Reactor Building

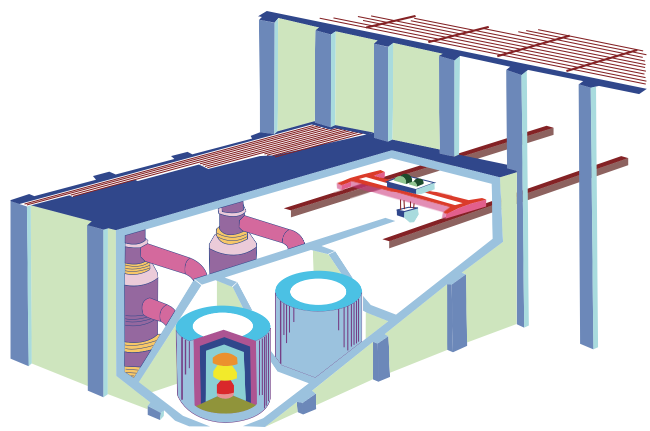

Figure 4 — IEF Power Cells in TWB Fusion Reactor Building.

Analysis of the performance of such systems has been made with two principal computer programs developed by EMC2 in studies of IEF systems. These are the EIXL code, a 1/1.5-D Vlaasov-Maxwell Poisson solver, to determine ion and electron density and potential distributions and associated fusion reaction rates and a system power-balance phenomenological code (the PBAL code). which includes all loss mechanisms in the complete fusion system. These show that DT fusion may be achieved readily at very high power (and neutron flux) in devices of less than two meters radius.

TWB Plant

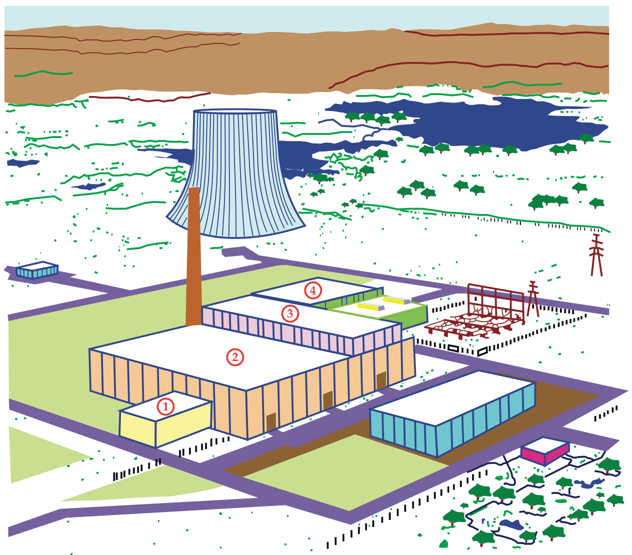

Figure 5 — TWB Plant, Showing: (1) IEF Reactor Building; (2) FP Handling; (3) Turbine-Generator Hall; (4) Electrical Equipment Building.

Figures 6. 7 and 8 show gross electric power gain G_{gr}, net electric power $P_{net.}$, and total fusion power P_{fus}. as a function of ion core energy or central potential well depth (generally equivalent to electron injection energy E_o), over a range of confinement system radii. The figures are for EXI/IEF DT fusion power systems operated at a thermal conversion efficiency of 33%, B field of 6 kG with copper coil magnets, and include 2.4 MeV per reaction in blanket reactions.

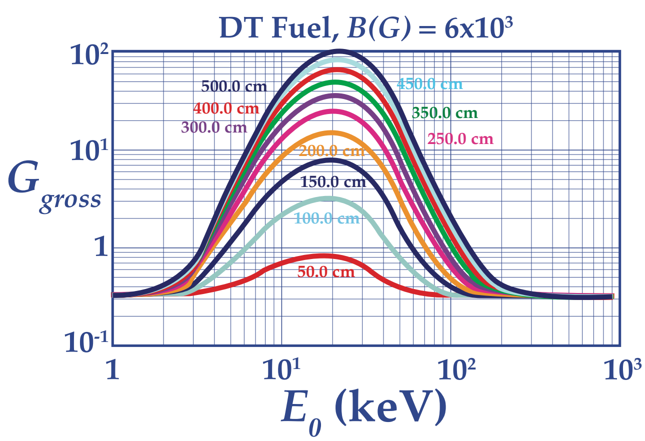

Figure 6 — Gross Gain vs. Well Depth for Various Radii.

Consider a baseline system of radius 150 cm to the magnet coil center, operating at an electron injection energy of E_o = 24 keV. This machine will generate a total fusion power (neutrons and charged particles) of about 1600 MWth. and produce net electric power of 530 MWe with a gross electric gain of 8.0, as indicated on the figures. The fusion neutrons come from a reacting "core" region 1.5-2.0 cm in radius, and deliver a neutron power flux of 25 MW(neutrons)/m^2 to the blanket first wall at 200 em.

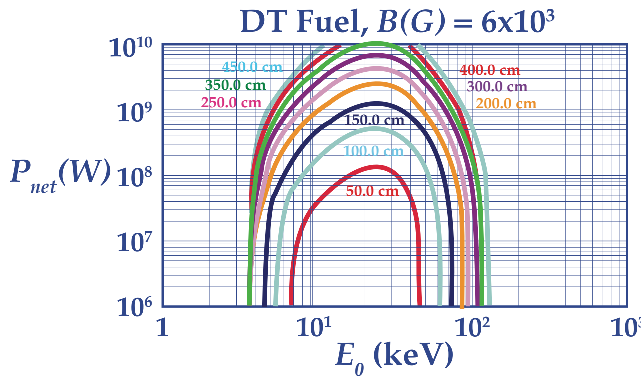

Figure 7 — Net Power vs. Well Depth for Various Radii

Increasing the B field to 9 kG would raise the fusion power to 8000 MWth and give a blanket fast neutron power flux of 125 MW(neutrons)/m^2. This can be doubled by raising the B field another 2 kG, to 11 kG. IEF system net electric power gain here is found to be well above 20:1, thus the recirculating power fraction is negligible, in contrast to that of hypothetical laser-fusion DT sources[10] or large-scale accelerator/target systems.[11]

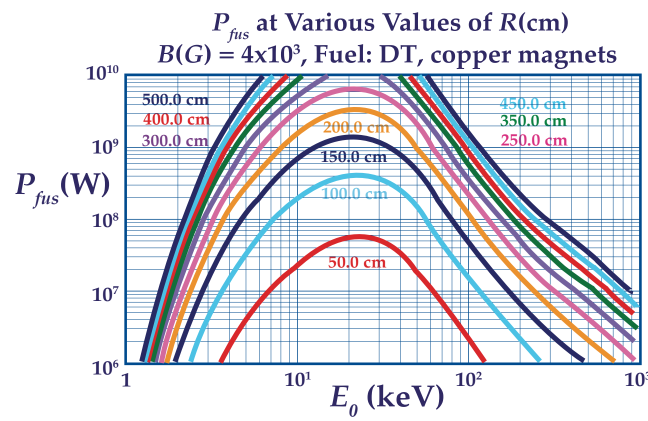

Figure 8 — Fusion Power vs. Well Depth for Various Radii.

The lifetime of the IEF neutron source is limited to about 20 MW(neutrons)year/m^2 neutron fluence. If operated at 8000 MWth, as above, coil flux is 225 MW(neutrons)/m^2 and machine life will be roughly 4.5 weeks. Because of (n, 2n) reactions in the coil conductor metal, neutron number flux on the blanket first wall will be appreciably higher than that computed from simple geometry; if the coils intercept 15-20% of the source neutrons, the blanket flux is approximately 0.88 of coil number flux. Blanket materials damage still depends on power flux, thus blanket first wall can live about 80% longer than IEF coil structures.

Since the IEF fast neutron sources are inherently very small, light weight, and compact, they can be constructed as removable plug-in/out units (e.g. as in Figure

3), able to be replaced relatively easily and quickly. This means of rapid replacement/removal was employed for most of the Rover nuclear rocket reactor tests conducted at the Nevada Test Site during the 1960's. The waste disposal problem this represents is only that of neutron-activated structural materials; no radioactive fusion products are involved.

III. Blanket Concepts and Performance

To use such IEF sources for waste incineration. they must be surrounded by a burn-up blanket region as described above. Correct determination of transmutation rates requires specification of system and blanket geometry, blanket composition, and the use of multi-group and multidimensional neutronics codes. Such computations have been made for mockup IEF systems by use of TRIDENT -CTR, ANISN, and a variety of special codes devised to follow the evolution of the transmutation chains for the isotopes of concern.[2] Using these, blanket design studies[1,4,12] have been made to establish fission product disposal rates with plausible blanket geometries, compositions and locations. However, the general transmutation effect can be illustrated by appeal to a simpler model. The time rate of change of concentration (N) of radioactive species under neutron irradiation at a flux (Φ), neglecting the natural half-life, is given by

$$\frac{dN}{dt} = N\sigma\phi f_n \qquad (1)$$

where σ is the cross-section for neutron transmutation, and f_n is a "neutron utilization efficiency" for the process. If both Φ and f_n are time-independent, then

$$\frac{dN}{dt} = N_0 EXP(-\sigma\phi f_n t) \qquad (2)$$

This equation is analogous to that for natural decay, with the decay constant replaced by $\sigma\Phi f_n$. Hence, one can write an "effective" half-life (τ_{eff}) as

$$\tau_{eff} = \frac{LN2}{\sigma\phi f_n} = \frac{0.693}{\sigma\phi f_n} \qquad (3)$$

The important feature of this approximation is that τ_{eff} is inversely proportional to the neutron current or flux. From this formula it is possible to estimate the neutron flux required to reach a desired "transmutation-half-life" for any given isotope (fixed σ). FP isotopes with large cross-sections (e.g. ^{129}I, actinides) can be most easily transmuted, while small cross-sections (e.g. ^{90}Sr, ^{137}Cs) are much more difficult. For example, ^{90}Sr has a thermal neutron "burnout" cross-section of about 1.2 b. If a transmutation half-life of 0.2 years is desired (so that 10 half-lives equivalent decay could be forced in only 2 years of irradiation), the required thermal neutron flux is 10^{17} neutrons/cm^{2}sec. This, in turn, demands a 14.1 MeV DT neutron flux of about 10^{16} neutrons/cm^{2}sec for f_n = 0.5. This factor depends on details of blanket design.

Among the blanket concepts considered, two generic designs were found that gave useful fission product burn-up rates. The first of these consisted of a beryllium blanket loaded with infinitely dilute short-lived products, in this case ^{90}Sr and ^{137}Cs. The second class used other moderators (e.g. D_2O, C) with little neutron-multiplying ability, and performed more poorly. The Be blanket could use the n, 2n reactions of the very fast (14.1 MeV) DT neutrons to increase neutron blanket flux, while the D_2O and C/SiC blankets could not. Use of Be thus makes DT neutrons more valuable for transmutation than neutrons from DD fusion. for the same neutron power, for example.

Table 1 — 14.1 MeV Neutron Flux (Φ_i) on Blanket First Wail Required for Two Year Transmutation Half-life of Selected FP Waste Isotopes;1 Units of Φ_i are MW(neutrons)/m^2.

	Blanket Type		
Fission Product Nuclide	Be-Be-Be	D2O-Al-Al	C-SiC-C
^{90}Sr	7	20	20
^{137}Cs	35	77	75
^{154}Eu	0.003	0.01	0.01
^{151}Sm	0.0005	0.001	0.001
^{129}I	0.3	0.8	0.8
^{99}Tc	0.3	1	0.9
^{93}Zr	0.3	7	7
^{135}Cs	0.8	2	2

Table 1 summarizes results of computer code calculations[1] for these two classes using the basic geometry shown in Figure 9. This took the baseline IEF system discussed above, as a DT neutron source mounted centrally on the axis of a 100 cm thick cylindrical blanket with an inner radius of 200 cm. The blanket consisted of 20-30 cm of active FP target mixed with blanket moderator material, all inside an outer reflector/moderator region.

The table lists the incident DT neutron flux required to achieve a two-year transmutation half-life for various FP isotopes. Ten-fold higher fluxes would be required to reach the 0.2 year equivalent "half-life" used illustratively above. Note the Be effect described previously, and that only the ^{90}Sr and ^{137}Cs, pose scant problems for transmutation incineration. In particular, note the low flux loadings required to dispose of the actinides. These very long-lived radioactive isotopes will be readily reduced in any system that can transmute the Sr/Cs components.

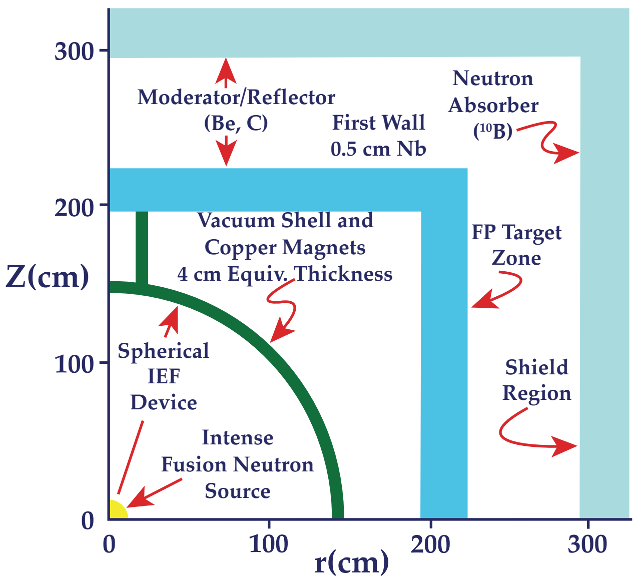

Figure 9 — Geometry for Blanket Irradiation Calculations.

More complex computer calculations have been made[12] that show the time-variation of ^{90}Sr residual radioactivity under DT neutron irradiation in a dilute Be blanket. Figure 10 presents the resulting reduction in hazard index (H) for a range of total neutron fluences, scaled from a nominal base case of 50 MW year/m² over a period of 5 years. The figure shows the residual hazard index as a function of time after transmutation irradiation. Note that a blanket fluence of DT neutrons of 250 MWyear/m² reduces the ^{90}Sr burden essentially to the residual low level background from bred-in ^{93}Zr and ^{93}Nb. and well below the H-value for natural uranium, reached at only about half of this fluence.

Volume of water to dilute the transmuted waste from one ton of 33,000 MWD-irradiated LWR fuel to RCG (10CFR-20) after reprocessing and 10 year storage.

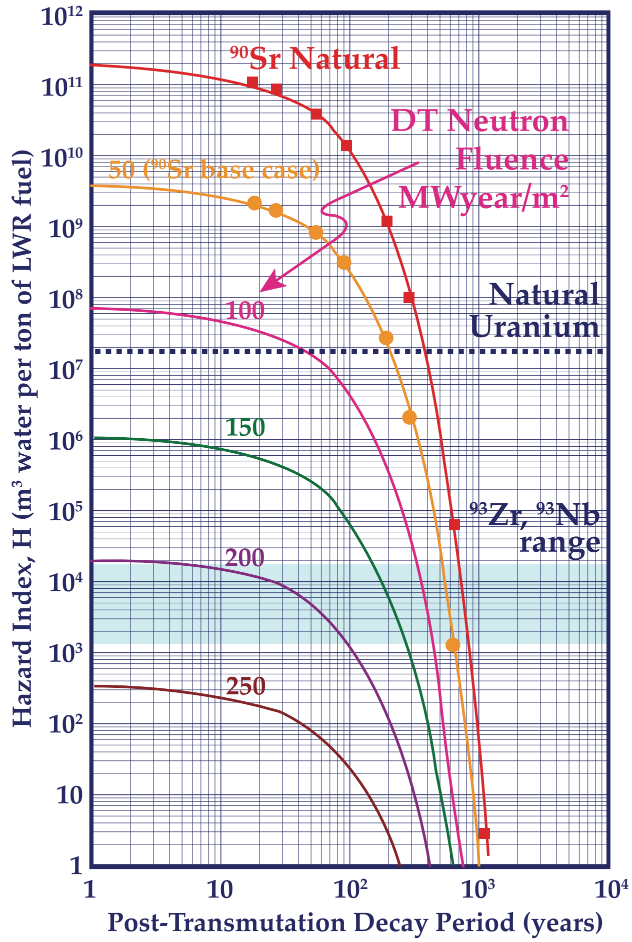

Figure 10 — Residual Hazard Index from ^{90}Sr Decay following Transmutation in a Beryllium Blanket, Driven by 14.1 MeV DT Neutrons from the Source Geometry of Figure 9.[11]

Taking an extreme example, to provide the very high tluence cited above (250 MWyear/m²) over a period of 2 years, requires a neutron flux of 10^{14} neutrons/cm²sec at the blanket first wall. Using the baseline EXL/IEF system with f_{n_c} = 0.85 (coil/structure intercept fraction), the neutron flux at the magnet radius (150 em) must be 1.1 x 10^{16} neutrons/cm²sec. Allowing 20 Mw(neutrons)year/m² for structure life[5] gives a machine lifetime of about one month. Less stringent numbers apply to longer irradiation times and lower fluences. An optimum system might be one that reached its transmutation goals in 5-6 years, with IEF device life of 2.5-3 months, and corresponding blanket first-wall structure replacement only every 0.5 years. However, independent of flux, the number of IEF unit and blanket replacements for this extreme irradiation level must be about 22 and 12, respectively. Lesser total fluence requires fewer replacement units, in direct proportion to the reduced neutron dose.

Since fluence is just a product of the neutron current per unit area and the irradiation time, fusion source concepts capable of high neutron flux can achieve much taster hazard reduction rates than those with low fluxes, at the expense of more complicated blanket fuel management processes (albeit without significant added difficulties in structure replacement requirements). In general, ^{90}Sr and ^{137}Cs can be burned up in times short compared to their natural decay half-lives only by compact IEF fusion systems. These can transmute the actinides in only a few weeks, in contrast to fission reactors or accelerator/target systems, which require several years of irradiation. Fission reactors and conventional fusion concepts can not provide fluxes high enough to drive ^{90}Sr or ^{137}Cs reduction at rates significantly shorter than their natural decay half-lives.

IV. Operating System Considerations

A complete TWB/fission-reactor/processing/storage system might be as shown in Figure 11.[1] Early and recent studies[4,10] have been made of the processing capacity of such generic TWB systems. Results showed that an IEF DT incinerator operating at fusion power P_f can provide a transmutation rate equal to the rate of generation of FP wastes from fission reactors with total power output capacity of nearly $40P_f$.[1] The add-on cost of such burn-up will be only about 5% of the base cost of power, neglecting income from sale of power from the TWB plant, itself. Both short- and long-lived FP isotopes can be destroyed by such systems. If operated at 200 MW/m² blanket flux of fusion neutrons, the actinides will be gone within a few weeks. Using continuous bypass processing and chemically separated mixed Sr and Cs wastes containing the ^{90}Sr and ^{137}Cs isotopes, the residual mixed wastes will be reduced to hazard storage lives of less than 100 years, with about 2-2.5 years of IEF system DT neutron irradiation.

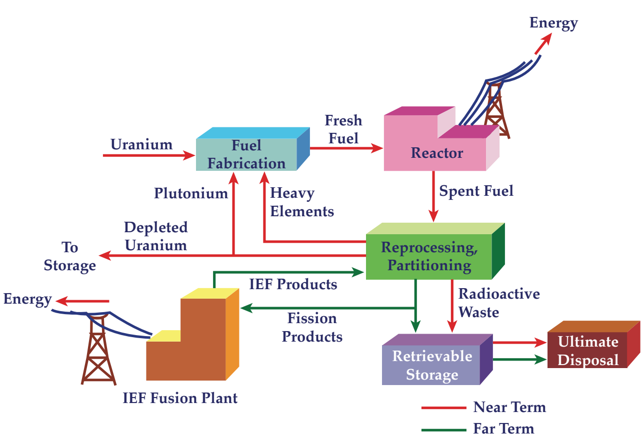

Figure 11 — Schematic Layout of Generic TWB Cycle and Plant System,[1] Driven by an IEF DT Fusion Neutron Source.

To analyze the dynamics of TWB incineration, take N_b as the number of TWB plants, each capable of handling the waste stream from R_b fission power plants. Then, assume N_f extant fission power plants, all of equal power, which have been operating for T_o years, accumulating FP wastes. Writing Y for years from time of startup of all of the burner plants, the waste inventory will then be given by

$$W(Y) = W_0\left\{1-\left(\frac{Y}{T_f}\right)\left[\left(\frac{R_bN_b}{N_f}\right)-1\right]\right\} \quad (4)$$

where the initial FP inventory (in plant-years equivalent) is $W_o = N_f T_o$. From this it is obvious that the number of burner plants must satisfy $N_b \geq N_f/R_b$ if the FP waste inventory is to be reduced. The initial rate of change of inventory is

$$\frac{d(W/W_0)}{dY} = 1-\frac{R_bN_b}{N_f} \quad (5)$$

As an example, assume that $N_f = 100$ plants exist and have been operating for $T_f = 20$ years, and $N_b = 10$ burner plants are built and operated, each capable of handling the waste from $R_b = 20$ fission plants. Then, equation 5 gives the initial rate of decrease of fission wastes requiring long-lived storage (e.g. 100's to 1000's of years) as 0.05/year (5%/year). Within 20 years the burner plant system would be able to handle all on-going waste production on a real-time steady-state basis. with only half of the assumed installed TWB capacity.

As is evident from Figure 10, fission product waste disposal by fusion sources is interesting primarily at large neutron fluences. The order of magnitude of these doses mandates careful analysis of the need for chemical separation, reprocessing and refabrication, as the fluence level on the blanket itself reaches the limits for selected materials. The use of fast neutron fluence as high as 200 MW years/m² implies a fairly sophisticated blanket management plan to achieve the number of irradiation cycles that are required.

Isotopic enrichment of selected target nuclides could help to avoid useless neutron capture in nuclei of benign isotopes, and thus reduce fluence requirements. However, for ^{137}Cs it has been found[2] that such enrichment yields only a modest (ca. 30%) reduction in time required to reach background radiation levels, thus is of little real interest The value of isotopic enrichment for ^{90}Sr may be greater, but may still not be worthwhile at the very high flux levels offered by the IEF TWB approach. More study of this issue is needed. In addition to consideration of isotopic enrichment, another factor affects blanket management; lower isotopic loadings per unit of blanket material result in higher FP burn-up rates. These conditions and those above will determine the fuel cycle processing requirements for best use of these unique high-power-density fusion neutron sources.

In any event, the practical application of fusion neutrons to TWB disposal of FP wastes requires the development and deployment of monitored retrievable storage (MRS) sites and of radiochemical fuel processing (RFP) plants, able to separate and compact Sr, Cs, the actinides and any other desired FP wastes. These must be packaged into forms suitable for use in the TWB blanket system and subsequent processing, and for eventual disposal by other means.

V. Gross Economics

These two developments (TWB and MRS) go hand-in-hand, as shown in Figure 11. Taken together they offer promise for complete and final solution of the FP waste problem, in a short and timely fashion and at modest cost. Though not yet proven by demonstration experiments, development of the new IEF concept for high-power-density electric fusion devices of small size can be undertaken at such small cost that it would seem prudent to do so as soon as possible.

Approximately $16M has been invested since 1985 in R&D on this concept. The work has involved EMC2 (which holds the basic patent rights), Los Alamos National Laboratory, Illinois and Columbia Universities, and several other private companies; supported by the DoD, NASA, DoE and EPRI.

Because of the inherent small size and low electron current drive features of the IEF system, it is possible to undertake its full development at modest cost. Proof-of-principle to net power break-even in a DT machine is estimated to cost about $150-180M over a period of 5-6 years.. A first pilot TWB plant could be built in an ensuing 5-6 year period at an additional cost of about $500-700M at the 1000 MWth level; large enough to handle FP wastes from fission reactors of 8,000-16,000 MWth power output. The radchem processing plant required to separate, compact and package this amount of FP waste material for irradiation is estimated to cost about $800-1200 M. The total cost of a first full pilot TWB system would then be in the range of $1600-1800M.

This first generation plant would be flexible, and highly instrumented and controlled, to allow operational determination of optimal conditions for subsequent plant design and construction; second generation plants would be less expensive per unit capacity. Since approximately 8-10 such plants could handle all the US nuclear power industry's FP wastes, the *total* cost of all TWB systems required to support the national nuclear power system could be less than $10-11B.

These costs are well below (less than half) those estimated for the current "permanent" geologic storage (PGS) plans for FP waste vitrification and deep underground storage in Yucca Mountain (YM). Even more significant is the fact that the cost of *proving* the scientific and technical basis of the TWB approach is less than 1% of the projected future cost of the YM/PGS program, and its success (or failure) would be known within 5-6 years. If successful, an additional 3-4% cost could complete development of the first pilot fusion neutron TWB plant, and the first complete system including radchem processing would cost another 4-5%. Thus technical and economic feasibility of the TWB approach could be fully proven at very small cost relative to (and thus of little impact on) the YM/PGS program.

Finally, IEF development for TWB plant application has clear, measurable milestones attainable at very low cost, before proceeding with development and construction of a first full pilot plant system. If successful in this approach, most FP wastes would be forever safely destroyed, leaving residual storage times all within a single human lifetime. Great savings could accrue to the nuclear power industry, since future operational costs would be reduced and further development of the YM/PGS approach would not be needed. Estimated savings range up to ca. $12-20B over the next 20-25 years. And, such a development could help to relieve the current environmental public and political pressure impeding nuclear power deployment, since a safe and effective means of short-term FP waste disposal and storage would, at long last. be available.

Publishing History

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