A FUSION TRANSMUTATION OF WASTE REACTOR

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ABSTRACT

A design concept and the performance characteristics for a fusion transmutation of waste reactor (FTWR)—a sub-critical fast reactor driven by a tokamak fusion neutron source--are presented. The present design concept is based on nuclear, processing and fusion technologies that either exist or are at an advanced stage of development and on the existing tokamak plasma physics database. A FTWR, operating with $k_{eff} \leq 0.95$ at a thermal power output of about 3 GW and with a fusion neutron source operating at $Q_p = 1.5\text{-}2$, could fission the transuranic content of about a hundred metric tons of spent nuclear fuel per full-power-year and would be self-sufficient in both electricity and tritium production. In equilibrium, a nuclear fleet consisting of LWRs and FTWRs in the electrical power ratio of 3/1 would reduce the actinides discharged from the LWRs in a once-through fuel cycle by 99.4% in the waste stream that must be stored in high-level waste repositories.

I. INTRODUCTION

There is a substantial worldwide R&D activity devoted to the transmutation of spent nuclear fuel (e.g. Refs. 1-3). The objective of this activity is to technically evaluate the possibility of reducing the requirements for long term geological repositories for the storage of high-level radioactive waste from spent nuclear fuel (SNF), by neutron fission of the plutonium and higher actinides remaining in the spent fuel discharged from fission power reactors. Repeated recycling of this spent fuel in commercial thermal spectrum fission power reactors would not significantly reduce the repository requirements, because the destruction of actinides by fission would be offset by the production of actinides by neutron capture in ^{238}U [1,2]. Repeated recycling of the spent fuel in special purpose fast spectrum reactors could reduce the radiotoxicity of the spent nuclear fuel by a factor of about 100, limited by safety and criticality constraints [1]. These constraints could be relaxed if the reactors (fast or thermal spectrum) could be operated sub-critical, which would require a neutron source. There is a general consensus that significantly higher levels of actinide destruction can be achieved by repeated recycling of spent fuel in sub-critical reactors with a neutron source. An accelerator-spallation neutron source has been extensively studied for this application (e.g. Refs. 1-6).

D-T fusion neutron sources could also be used to drive sub-critical reactors for the destruction of actinides, and a few scoping studies [7-13] have been carried out. In particular, Ref. 13 reviewed the requirements for a neutron source vis-à-vis the present tokamak database and found that the physics parameters routinely achieved in operating tokamaks ($H \approx 1$, $\beta_N = 2\text{-}3$) and operation at Q_p as low as 1.5-2.0 would be sufficient for a tokamak neutron source with major radius R = 3-5 m to produce transmutation rates of hundreds to thousands of kg/FPY (full-power-year) of SNF in a sub-critical transmutation reactor.

Our purposes in this paper are to identify the physical and performance characteristics of a subcritical transmutation reactor driven by a tokamak fusion neutron source at the lower end of this range of sizes and performance capabilities. The general design objectives for this Fusion Transmutation of Waste Reactor (FTWR) are that it: 1) destroy the transuranic content of hundreds of metric tonnes/FPY of spent nuclear fuel; 2) utilize nuclear and processing technologies that either exist or are under development; 3) operate at a neutron multiplication factor $k_{eff} \leq 0.95$ to enhance safety; 4) be based on the existing tokamak plasma and fusion technology databases to the maximum extent possible; and 5) be self-sufficient in tritium and electricity production. In this initial effort, we concentrate on those aspects of the design which most influence the configuration and performance characteristics.

II. DESIGN SUMMARY

II.A Geometric Configuration and Materials

The geometric configuration of the FTWR is shown in Figs. 1 and 2. The transmutation reactor consists of a ≈ 40 cm thick ring of vertical hexagonal fuel assemblies located outboard of the plasma chamber of the tokamak fusion neutron source. The reactor metallic fuel consists of a zirconium alloy containing transuranics from SNF dispersed in a zirconium matrix and clad with a steel similar to HT-9. The coolant for the reactor, reflector and shield, first-wall and divertor is Li17-Pb83 eutectic enriched to 20 % ^{6}Li to meet the tritium self-sufficiency requirement. Reflector and shield are located inboard of, above and below the plasma chamber and above, below and outboard of the reactor to protect the magnets from radiation damage and to reflect neutrons towards the reactor. The toroidal and poloidal magnets employ Oxygen-Free High Conductivity (OFHC) copper conductor and liquid nitrogen (LN2) coolant. The materials composition of the FTWR is summarized in Table 1.

Table 1 Materials Composition of FTWR

Component	Material
Reactor	
Fuel	Zr-transuranic alloy in Zr matrix
Clad & Structure	HT-9-like steel
Coolant	Li17Pb83 (^{6}Li enrich 20%)
Reflector	HT-9, Li17Pb83
Shield	HT-9, Li17Pb83, B_4C
Magnets	
Conductor	OFHC
Coolant	LN2
Structure	Steel
First-Wall & Divertor	
Structure	HT-9-like steel
Coolant	Li17Pb83

II.B Major Design Parameters

The neutron source is a D-T tokamak with the parameters shown in Table 2, most of which are in the range routinely achieved on operating tokamaks [14]. The only two parameters which fall outside this range are the plasma energy amplification factor Q_p and the steady-state pulse length. The required value of Q_p is only a factor of about 2 greater than what has been achieved on the Joint European Torus (JET) device, and there is a proposal for $Q_p \approx 2$ operation in JET. Perhaps the greatest advance beyond the present state of the art in tokamak operation is the steady-state pulse length. Using a conservative estimate of a current drive efficiency $\eta_{CD} = 0.03$ A/W, we estimate that steady state could be achieved with $Q_p = 1.55$, at 150 MW fusion power. If advances in tokamak R&D [15] enable achievement of $\eta_{CD} = 0.05\text{-}0.06$ A/W or a higher bootstrap current fraction, it should be possible to achieve steady-state pulse length at the reference value of $Q_p = 2$.

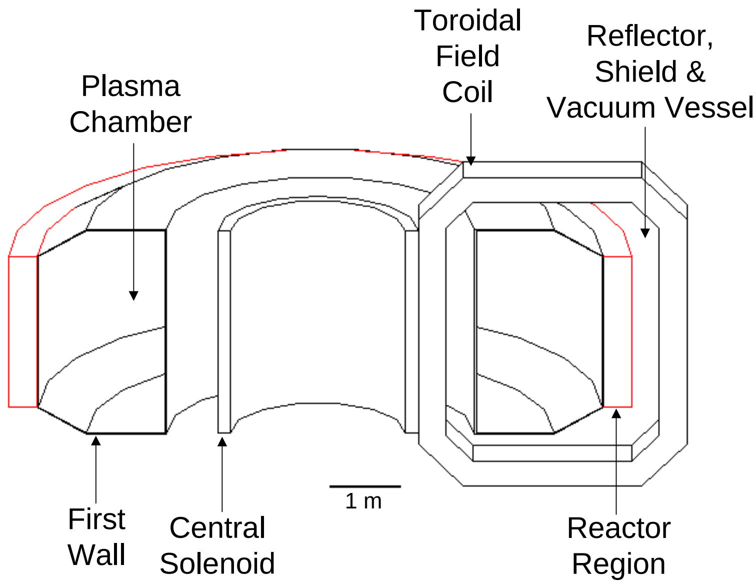
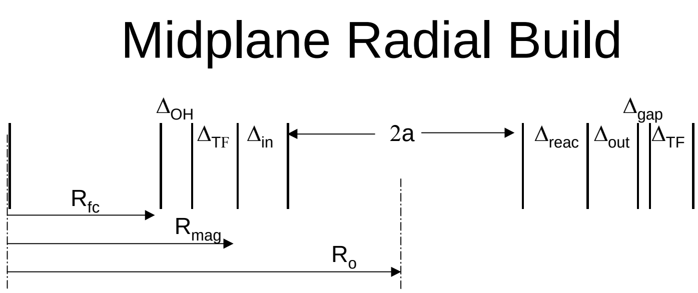


Fig. 1 Schematic of Geometric Configuration of FTWR



R_{fc}	=	flux core radius	=	1.24 m
Δ_{OH}	=	OH solenoid Δ	=	0.18
Δ_{TF}	=	TF coil Δ	=	0.39
Δ_{in}	=	inner refl/shld	=	0.40
a	=	minor radius	=	0.89
Δ_{reac}	=	reactor	=	0.40
Δ_{out}	=	outer refl/shld	=	0.30
Δ_{gap}	=	outer gap	=	tbd
R_o	=	major radius	=	3.10
R_{mag}	=	magnet radius	=	1.81

Note: refl/shld includes first wall, reflector, shield, and vacuum vessel

Fig. 2 Radial Build of FTWR

Table 2 Neutron Source Parameters

Parameter	Value
Plasma	
Major radius, R_0 (m)	3.1
Minor radius, a (m)	0.89
Elongation, κ	1.7
Magnetic field, B_0 (T)	6.1
Plasma current, I_p (MA)	7.0
Bootstrap current fraction	0.38
Normalized beta, β_N (%)	2.5
Confinement factor, H ITER IPB98(y,2)	1.1
Fusion power (MWth)	150
Plasma energy amplification, Q_p	2.0
Pulse length	steady-state
Magnets	
Toroidal field @ coil (T)	10.45
Central solenoid field @ coil (T)	8.0
Inductive flux (V-s)	90
Temperature (K)	80-100
Power dissipation & refrigeration (MWe)	972
Lifetime radiation dose (rads)	1.5×10^{12}
Lifetime fast neutron dose (n/cm^2)	1.8×10^{22}
First-Wall	
14 MeV neutron wall load (MW/m^2)	0.79
Surface heat load (MW/m^2)	0.34
Radiation damage (dpa/623 d cycle)	21
Tritium Inventory	
Beginning of cycle (g)	120
Maximum (g)	1000

The FTWR magnetic system is based on existing technology. The magnetic field levels are well within the range of existing tokamaks. The Joule heating and, even more, the LN2 refrigeration for the resistive magnets constitute the major electrical power requirement for the FTWR. The lifetime radiation and neutron doses to the toroidal field coils are intended to be at the limit for ceramic insulators, and may be beyond the limit for organic insulators, although these limits are not well defined. The poloidal coil system (central solenoid plus ring coils) is designed to provide adequate Volt-seconds for inductive startup and a minute or so of burn.

The FTWR first-wall design is an adaptation of the ITER design [16], albeit with HT-9-like steel structure. Although the qualification of HT-9-like steel for operation in a neutron irradiation environment is in progress, the radiation damage limit is not yet known. However, we believe that this limit will probably allow about 5-10 (623 day) cycles (> 100-200 dpa) before it is necessary to replace the first-wall of the neutron source.

The main parameters of the transmutation reactor are given in Table 3. The design is an adaptation of the ANL design of a transmutation reactor for an accelerator (ATW) neutron source [17], which has a fast neutron spectrum to maximize the fission probability per neutron absorbed in transuranics.

Table 3 Transmutation Reactor Parameters

Parameter	Value
Maximum multiplication constant, k_{eff}	0.95
Actinide loading (MT)	27
Maximum actinide enrichment (V/O)	45
# Hexagonal fuel assemblies	470
Fuel assembly pitch (cm)	16.1
Fuel assembly length (cm)	228
Fuel pin diameter (cm)	0.635
Average power density (kW/liter)	124
Fuel cycle	4 batch
Clad irradiation @ discharge (dpa)	150
Coolant Tin/Tout (K)	548/848
Coolant flow velocity (m/s)	0.76
Coolant mass flow rate (kg/s)	51630
Coolant pumping power (MWe)	131

II.C Performance Summary

The performance of the FTWR is summarized in Table 4. A FTWR operating at 3000 MWth can destroy the transuranic content of about 100 metric tons of SNF per full-power-year (FPY). By repeatedly recycling the unburned FTWR fuel and using transuranics from LWR SNF as the makeup material, the equilibrium FTWR fuel cycle would ultimately result in an effective reduction in the waste streams of 99.4% of the transuranics discharged from a LWR in the OTC. While mass alone does not characterize the high-level waste repository requirements, this reduction in mass provides some indication of the corresponding reduction in high-level waste repository requirements. A single FTWR (3 GWth) with 60% availability can transmute the transuranic content of the SNF produced in three typical Light Water Reactors (LWRs ,1 GWe).

Table 4 Major Performance Parameters of FTWR

Parameter	Value
Total Power (MWth)	3000
Thermal-to-electrical conversion (%)	40
Fusion Neutron Source Strength (#/s)	5.32×10^{19}
SNF Transmutation Rate (MTU/FPY)	102
Transuranic Mass Reduction in SNF (%)	99.4
Support Ratio (GWe LWR/FTWR)	3
Electrical Power Amplification, Q_e	> 1
Lifetime (FPY)	40
Availability (%)	60

The toxicity (defined as the volume of water required to dilute the SNF to the maximum permissible concentration for human consumption) of the original SNF from a once-through LWR cycle and the toxicity from the same SNF after transmutation in a FTWR (without the uranium, which is assumed to be recovered and disposed of as low level waste in both cases) are compared with the toxicity of the original as-mined uranium ore from which the fuel was fabricated in Fig. 3. The toxicity of the LWR SNF including the uranium is also shown to illustrate the effect of just removing the uranium from the SNF. The SNF from the LWR becomes less toxic than the natural as-mined uranium ore from which it was fabricated in about 7,500 years. If this same SNF were irradiated in the FTWR, it would become less toxic than the natural as-mined uranium ore from which it was fabricated in about 500 years. While toxicity is only one of many measures of the hazard potential of radioactive waste, this comparison does indicate the magnitude of the benefit of the transmutation of SNF.

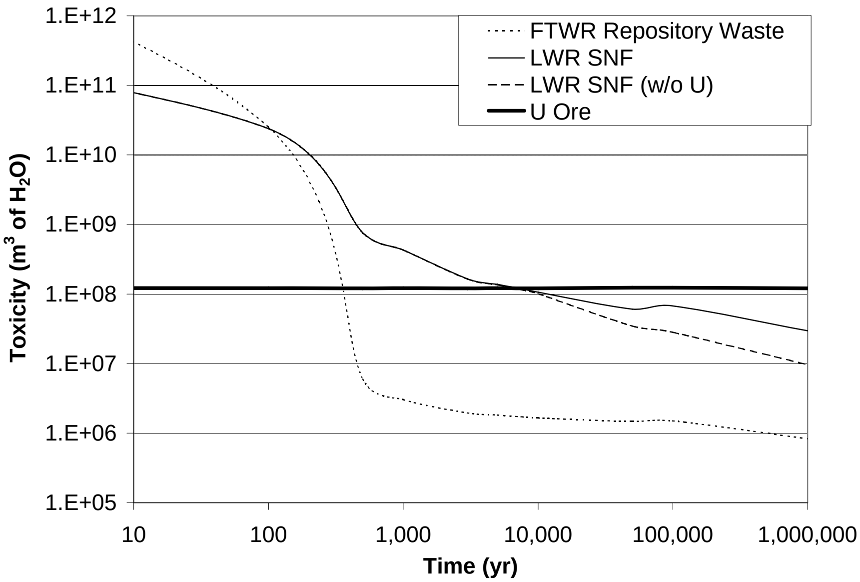


Fig. 3 Toxicity of SNF (uranium recovered) with and without transmutation in FTWR compared to toxicity of natural uranium ore.

At 3 GWth, a FTWR is just self-sufficient in electrical power production (i.e. $Q_e \approx 1$). The principal electrical power requirement is associated with refrigeration of the LN2 that is required to remove the Joule heating from the magnets. If the FTWR design was extended to produce 6 GWth by increasing the number of fuel assemblies, the power requirements would increase slightly and the electrical power amplification factor would become $Q_e \approx 1.8$, which would allow $\approx$ 1 GWe surplus electricity to be produced, as well as doubling the transmutation rate and number of LWRs supported by a single FTWR.

III. DESIGN TRADE-OFF STUDIES

The size and geometry of a sub-critical reactor driven by a tokamak fusion neutron source are determined by the size and geometry of the tokamak neutron source and by the transmutation rate (power level) and power density of the surrounding sub-critical transmutation reactor. The design objective of identifying the 'minimal' tokamak neutron source that would produce a relevant transmutation rate in a sub-critical ($k_{eff} \leq 0.95$) reactor and produce electrical power self-sufficiency led to the selection of copper (rather than superconducting) magnets at the outset. The choice of materials was strongly influenced by the design objective of using fusion, nuclear and processing technologies which either existed or are well along in their development.

III.A Tokamak Neutron Source Physics Constraints

The standard design methodology used in the ITER design studies, where the major parameters of the machine (R_0, a, I_p, B_0, etc.) are determined by a relatively small number of equations and assumptions [14,18], was employed. The starting point of this approach is a simple equation for the radial build of the reactor,

$$R_0 = R_{mag} + \Delta_{in} + a \tag{1}$$

where R_{mag} is the major radius at the inner leg of the toroidal field (TF) coil, Δ_{in} is the thickness of the inner shield and reflector region between the plasma and the TF coil and a is the minor radius (see Fig. 2). Using equation (1) along with expressions for the edge safety factor q_{95}, the beta limit and the Greenwald density limit, taking into account the $1/R$ dependence of the toroidal magnetic field, and assuming that the plasma energy confinement time, τ_E, is described by one of the usual confinement scalings such as the ITER IPB98(y,2) scaling [19], an equation can be derived coupling the performance characteristics of the reactor to its major geometric and operational parameters:

$$\frac{\left(nT\tau_E\right)_\infty}{1+\dfrac{5}{Q_p}} = F\left(\beta_N, G_n, \kappa, \delta, q_{95}, \Delta_{RS}, H, A, I_p, B_{TF}, \ldots\right) \tag{2}$$

where $(nT\tau_E)_\infty$ is the value of the triple product $nT\tau_E$ required for ignition (usually taken to be equal to 5×10^{21} m^{-3} keV s for D-T reactors), $Q_p = P_{fus} / P_{aux}$, and F is a nonlinear function of various operating and constraint parameters (see Appendix B). If we select reasonable values for the shape parameters and constraint limits δ, κ, q_{95}, β_N and G_n, and aspect ratio A, we can use Eq. 2 to perform trade-off studies between the size and the major operational parameters (plasma current and maximum toroidal field), for given performance requirements (Q_p and H). An example of such a trade-off study is shown in Fig. 4, where the major parameters of the reactor are plotted vs. the maximum toroidal field at the inside leg of the TF coil for a reasonable set of assumptions and performance requirements ($\delta = 0.4$, $\kappa = 1.7$, $q_{95} = 3$, $\beta_N = 2\%$, $G_n = 0.75$, $A = 3.47$, $\Delta_{in} = 0.4$ m, $Q_p = 5$).

It should be emphasized here that while most of the physics constraints are inequalities ($\beta_t \leq \beta_{max}$, etc.) they are treated as equalities in our analysis. This means that the performance and power output of the reactor designs obtained via this procedure are the maximum attainable under the assumed constraints. Once the major reactor size parameters (a, R_0, etc.) are fixed, a wide operating space with more modest performance (Q_p) and fusion powers can be identified by selecting appropriate operating densities and temperatures, or even reducing the plasma current and the toroidal magnetic field.

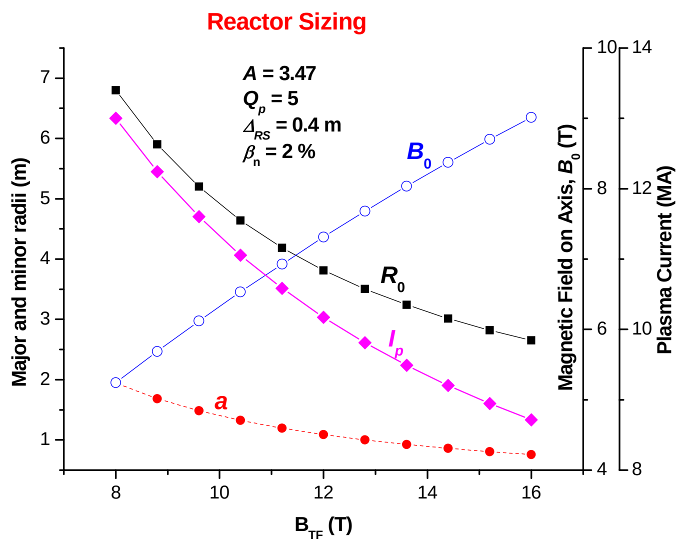


Fig. 4 Various reactor parameters (R_0, a, B_0, I_p) vs. the maximum toroidal field at the coil, B_{TF}, for a set of fixed shape and performance parameters.

Based on the results shown in Fig. 4 and on other similar analyses, a major radius of about 3.1 m, corresponding to a maximum field of 14 T at the TF coil and a plasma current of 9.4 MA, was selected for our initial design point. More detailed considerations of the performance characteristics vis-à-vis the neutron source requirements led us to a reference design point with the same size ($R_0 = 3.1$ m, $A = 3.47$) but lower field at the TF coil and lower plasma current (10.4 T and 7 MA, respectively). This choice represents a reasonable trade-off between low cost (small size and low current) and reasonable Joule heat removal requirements for the TF system.

III.B Magnet Conductor and Coolant

The selection of the magnet technology, materials and magnet cooling options were among the most significant design choices we had to make. While most recent steady-state power producing tokamak reactor designs rely on superconducting magnets in order to minimize the high Joule losses associated with resistive magnets, we opted to trade-off recirculating power for simpler technology and smaller size (superconducting magnets require a thicker shield, which leads to a larger reactor size, since they are more sensitive to nuclear heating and irradiation). Such a choice of magnet technology would not have been viable for a stand-alone power producing fusion reactor (the total power required to operate and cool the resistive magnets is several times the reference fusion power output of the neutron source), but since in the FTWR most of the useful power originates in the fission (transmutation) part of the system, it is an acceptable tradeoff.

Several copper alloys were considered for the magnet conductor, including Oxygen-Free High Conductivity (OFHC) Copper and Beryllium Copper (BeCu). OFHC Copper, strengthen by steel support, was selected, because of its lower resistivity, even though BeCu has better structural properties.

The choice of a coolant for the magnets was also a critical part of the design. Water was considered but rejected, since the resulting Joule heating power was very large (~1.3 GW) due to the high resistivity of copper at the high operational temperatures (~450 K) of the magnets with water coolant. It was decided to use liquid nitrogen (LN2) to cool the magnets to cryogenic temperatures (~80–100 K), because the copper resistivity drops substantially at these temperatures, resulting in substantially reduced Joule losses (~ 110 MW). Although the power required to refrigerate the magnets for steady-state operation is considerable, overall the cryogenic option was more attractive than the water-cooled one.

III. C Transmutation Reactor Technologies

One of the design objectives was to use nuclear and processing technologies that exist or are being developed, to the maximum extent possible. Metal fuel with HT-9-like steel clad and pyrolytic processing technology have been under development at Argonne National Laboratory for the fast reactor program for a number of years and have been adopted for further development in the U. S. ATW program [6]. Moreover, a fast neutron spectrum maximizes the transmutation rate per neutron absorbed in actinides. Thus, we use HT-9-like steel clad and metal fuel, following the ATW design [17].

We considered two coolants, a lead-bismuth eutectic and a lead-lithium eutectic. The lead bismuth eutectic (LBE [$Pb_{45.5}Bi_{55.5}$]) has been used in the Soviet nuclear submarine program and is currently under development for the ATW program [6]. We decided to incorporate lithium within the circulating coolant, rather than as a solid component, in order to achieve continuous tritium recovery, which would necessitate the addition of lithium to LBE. There is also a significant development program for the lead lithium eutectic (Li17Pb83) for fusion applications, primarily in Europe [20].

The physical properties of lithium lead eutectic and lead bismuth eutectic are quite different (see Appendix A). LBE has a much lower melting point, 397 K, than does LiPb, 508 K. However, in general, the other properties of LiPb are far more favorable than those of LBE. The specific heat of LiPb is nearly 50% greater than for LBE. This results in a requirement for much higher flow velocities with the LBE coolant which, only partially compensated by the lower electrical conductivity of LBE, would require a significantly larger MHD pumping power for LBE than for LiPb. Furthermore, since the LBE would have to be doped with lithium in order to produce the required tritium, the properties of this new alloy may vary significantly from those of LBE, requiring substantial further development. It was found that even at 100% ^{6}Li, as much as 2% lithium would have to be added to the LBE. At more reasonable ^{6}Li enrichments, this value could easily exceed 5%. Therefore, Li17Pb83 was chosen as the primary coolant.

III.D Reflector and Shield

The magnets must be shielded to protect against radiation damage effects of the fusion neutrons, fission neutrons, and secondary gammas. The blanket region surrounding the plasma will necessarily consist of a first wall and vacuum vessel that are designed based primarily on structural, not shielding, considerations. An additional region must be added to reduce the damage rates to an acceptable level. Furthermore, to enhance the transmutation rate, a reflector is needed to redirect neutrons heading away from the transmutation reactor. The reflector and shield compositions from the ANL ATW design study [17] were adopted. We found that we might be able to design a pure shield as small as 25 cm, but then would need a relatively large heavy metal loading and ^{6}Li enrichment. On the other hand, using only a reflector, with no shield, would require a reflector thickness of 40 cm. We chose a combined reflector-shield with a thickness of 30 cm, which provided adequate shielding and sufficient tritium production at a reasonable ^{6}Li enrichment, and beyond which no significant further reduction in heavy metal loading could be obtained. We allowed an extra 10 cm for gaps or additional shielding on the inboard. Since the plasma is shifted outward, we did not otherwise allow for a gap between the circular plasma in our model and the wall on the inboard side.

IV. NEUTRON SOURCE PLASMA PHYSICS ANALYSIS

IV.A Reference Plasma Parameters & Neutron Source Performance

Based on the methodology outlined in the previous section and taking into account the neutron source requirements of the subcritical fission reactor, a R_0 = 3.1 m design with 7 MA current and 6.1 T central magnetic field was selected as the FTWR reference design point. While our trade-off studies had assumed a 14 T field at the toroidal coil and Q_p = 5 (Fig. 4), subsequent simulations and concerns about the impact of the resistive losses in the TF coil system on the recirculating power of the plant led us to adopt a less demanding set of magnet and performance parameters, namely B_{TF} = 10.45 T and Q_p = 2, for the reference design point. The major plasma-related parameters of the reference design point are listed in Table 5.

Table 5 Reference Plasma Parameters of the Fusion Neutron Source

Parameter	Value
Major Radius, R_0 (m)	3.1
Minor Radius, a (m)	0.89
Aspect Ratio, A	3.47
Plasma Elongation, κ	1.70
Plasma Triangularity, δ	0.40
Safety Factor at 95% flux, q_{95}	3.0
Toroidal Field @ R_0, B_0 (T)	6.1
Plasma Current, I_p (MA)	7.0
Normalized Beta, β_N (%)	2.5
Confinement multiplier, H, ITER IPB98(y,2)	1.1
P_{fus} (MW)	150
$Q_p = P_{fus} / P_{aux}$	2
$<n_e>$ (m^{-3})	2.0×10^{20}
$<n_e> / n_{GW}$ (Greenwald density ratio)	0.75
$<T>_n$ (keV)	7.6
Density profile exponent, α_n	0.1
Temperature profile exponent, α_T	1.0
Neutron Wall Load (MW/m^2)	0.79
First Wall Power Density (MW/m^2)	0.34
Total DT Fusion Neutron Rate (#/s)	5.32×10^{19}
H-Mode Power Flux Margin, P_{sep} / P_{LH}^{thr}	4.5
Bootstrap Current Fraction	0.38

A Plasma Operating Contour (POPCON) was constructed for the reference design to help us select an appropriate operating point and to scope out the operating range of the machine. It can be seen from Fig. 5 that an operating point with Q_p = 2 and $P_{fus} \approx$ 150 MW, which satisfies the neutron source performance requirements, is within the allowable operating range.

The 7 MA/6.1 T design is also capable of higher performance operation with Q_p = 5, if higher levels of confinement or beta limits can be attained. In Fig. 6, a POPCON plot for an enhanced confinement factor H = 1.3 relative to the ITER IPB98(y,2) scaling is shown. It can be seen that operating points with higher Q_p's and fusion powers, and with densities below the Greenwald limit, are possible. It also should be emphasized that such confinement enhancements are rather modest and are routinely observed in today's experiments [21].

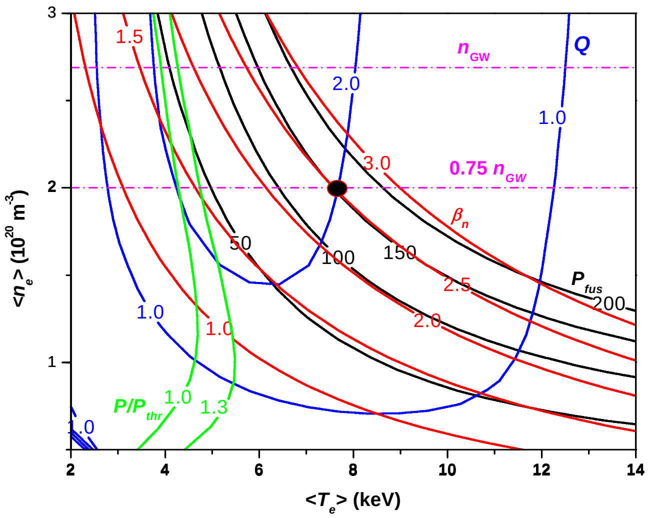


Fig. 5 POPCON Plot for the reference design of the fusion neutron source. Contours of constant fusion power, Q_p, normalized beta and P_{sep} / P_{LH}^{thr} ratio are shown. In addition, lines of constant $<n_e>$ / n_{GW} ratio are also shown. The reference operating point is marked by a solid circle.

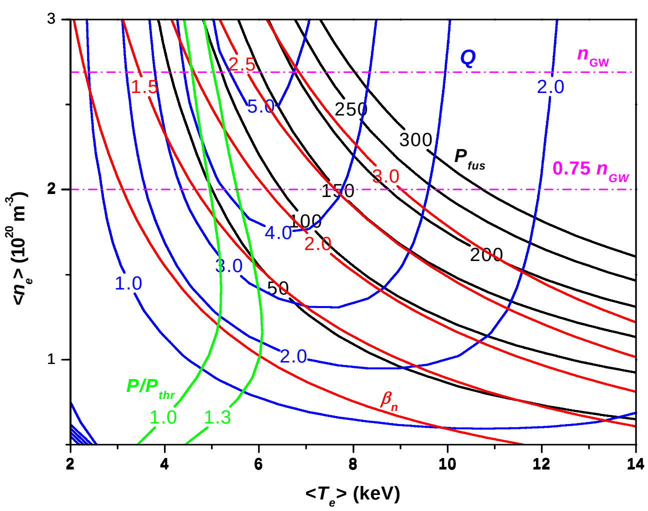


Fig. 6 POPCON plot assuming a confinement enhancement factor H = 1.3

IV.B Current Drive Considerations

Steady-state operation is one of the goals of the FTWR design. This means that external current drive will be required to supply part of the plasma current in the fusion reactor core. Since most current drive methods for reactor-grade plasmas are rather inefficient and expensive, every effort should be made to minimize the external current drive requirements by maximizing the bootstrap current fraction. For the reference design point, this fraction is estimated to be about 38 % using a simple scaling formula (Appendix B). However, it is believed that higher bootstrap currents can be attained by optimizing various plasma profiles.

To get an idea of the influence of the bootstrap current fraction on the demands on the current drive system, the current drive efficiency $\eta_{CD} \equiv I_{CD}/P_{CD}$ (Amps/Watt) required for steady state operation is calculated for our reference design for two values of the plasma Q_p. This calculation assumes that all of the auxiliary power injected into the plasma is also available to drive current, therefore $I_p\left(1-f_{bs}\right)=\eta_{CD}\,P_{fus}/Q_p$. The reference values for fusion power and plasma current (150 MW and 7 MA respectively) have been assumed.

Table 6 Current Drive Efficiencies required for steady-state operation for various bootstrap fractions and Q_p values.

Bootstrap Current Fraction	η_{CD} (A/W), $Q_p = 2$	η_{CD} (A/W), $Q_p = 5$
0.2	0.075	0.187
0.4	0.056	0.140
0.6	0.037	0.093
0.8	0.019	0.047

It can be seen from Table 6 that for the reference design point, a current-drive efficiency in the range of 0.05 – 0.06 A/W would be necessary to achieve steady-state operation. Although a detailed analysis of the current drive and heating system of this design has not been performed, a system based on fast waves (FW) in the ICRF regime for central current drive and lower hybrid (LH) waves for off-axis drive would be a reasonable choice [22]. An estimate of the FW current drive efficiency of such a system can be obtained by using a simple scaling formula developed for the ARIES RS design study [22,23]. For our reference design point, this simple scaling predicts a current drive efficiency of 0.03 A/W, resulting in a driven current of 2.34 MA, less than the 4.34 MA that are needed. However, this is a very conservative estimate. A fraction of the current would be driven by LH, which has a higher current drive efficiency than ICRF FW.

Furthermore, even if all the current had to be driven by FW current drive, we could operate at higher temperatures and lower densities to increase the current drive efficiency. As can be seen from the POPCON plot in Fig. 5, by moving along the 150 MW fusion power line (which almost coincides with the constant $\beta_N = 2.5$ contour) we can produce the same amount of fusion power at higher temperatures and lower densities. We would have to accept slightly lower Q_p operation, but this also works to our advantage in this case since the extra auxiliary power would be available to drive more current. A simple calculation shows that we could drive all of the 4.34 MA current needed for steady-state operation with FW current drive alone by operating at $Q_p \approx 1.55$ with $< T_e >_n \approx 9$ keV and $< n_e > \approx 1.6 \times 10^{20} m^{-3}$ to achieve a higher CD efficiency (η_{CD} = 0.044 A/W).

It should also be mentioned that intensive research is being carried out in the area of tokamak current drive, and the relevant experimental database is rapidly growing [15]. More efficient methods, such as Electron Cyclotron (EC) current drive, may soon be available.

IV.C Extrapolations Beyond Present Experimental Database

Since the objective of this design is a relatively near-term neutron source to transmute spent nuclear fuel, one of our design requirements was to remain as close as possible to the present tokamak experimental database. However, even small extrapolations from this database can greatly enhance the performance and hence attractiveness of a fusion neutron source. Such extrapolations allow operation at a higher beta and enhanced confinement level (simultaneous attainment of higher beta and enhanced confinement is usually required) and result in higher fusion power densities and higher bootstrap current fractions. Tokamaks operating under these improved conditions are usually called *Advanced Tokamaks*, and are being vigorously studied by the fusion community [21]. Several tokamak experiments around the world have achieved advanced tokamak operation for short pulses, and this database is rapidly growing.

V. NEUTRON SOURCE TECHNOLOGY DESIGN

V.A Magnets

A tokamak fusion neutron source requires several sets of magnets. A toroidal magnet system produces the toroidal magnetic field (TF) needed to stabilize the plasma, while a central solenoid (CS) and a set of poloidal ring coils (PF) provide the changing magnetic flux (Volt-seconds) to drive the inductive plasma current and provide the equilibrium field for plasma position control and shaping.

In this initial analysis, we have focused our attention on the TF and CS systems, since they are the ones that affect the size of the FTWR and can have a major impact on the recirculating power fraction of the plant.

Our reference design is based on resistive copper magnets (with ceramic or organic insulators) cooled at cryogenic temperatures (80-100 K) by liquid nitrogen (LN2). This choice follows recent designs of pulsed tokamak plasma burning experiments, which have also adopted resistive magnets cooled at cryogenic temperatures, for simplicity and size reduction [24-26].

A wedged design with 18 TF coils was adopted. One unique characteristic of the FTWR design is that the TF coils are larger than would be expected for a tokamak of this size, since there must be enough space between the plasma and outer TF coil leg to accommodate the transmutation reactor (see Fig. 1). Another unique feature is the requirement for steady state operation, which imposes rather demanding requirements on the LN2 refrigeration system.

To minimize Joule losses, Oxygen-Free High Conductivity (OFHC) copper was selected as the conductor material, with 3% steel added for structural support. Cooling channels occupy 5% of the TF coil cross section. To minimize the resistance of the coils while maintaining structural integrity, the cross sections of the top, bottom and outer legs of the TF coils were 50% larger than the cross section of the inner leg.

To ensure that our TF coil design meets ASME structural design criteria, the various stresses and forces (centering and tensile forces, bending stresses etc.) were evaluated using standard analytic expressions (Appendix C). The yield and ultimate strengths for the magnet materials are listed in Appendix A.

The central solenoid (CS) coil was designed to produce about 45 Volt-seconds, which, along with the contribution from the PF coil system (assumed to be equal to the CS contribution), is sufficient to start up the plasma and provide enough Volt-seconds for a few minutes of burn (flattop) time. The same materials (OFHC and steel) were used for the CS coil, but the steel fraction was higher (20%) compared to the TF coil design. Twenty-five coolant channels (5% of the cross sectional area of the CS coil) were used.

The major design and operational parameters of the TF and CS coil systems are summarized in Table 7.

Table 7 Major TF and CS magnetic coil parameters

Parameter	TF Coils	CS Coil
Conductor	OFHC Cu	OFHC Cu
Coolant	LN2	LN2
Field @ Conductor (T)	10.45	8.0
Cross section area (m^2)	0.22	0.547
coolant fraction (%)	5	5
steel fraction (%)	3	20
Maximum tensile stress (MPa)	132	246
ASME allowable Sm (MPa)	132	251
Ohmic Heating (MW)	82 (all magnets)	27
Magnet resistance (Ω)	1.645×10^{-7} (per magnet)	7.133×10^{-8}

Radiation effects on the magnets, particularly the insulators, are a concern for the FTWR design. Transport calculations indicate that the lifetime fast neutron fluence at the TF coil of $1.8\text{x}10^{22}$ n/cm^2 is about a factor of 2 less than the limiting value for ceramic insulators, but that the lifetime dose of $1.5\text{x}10^{12}$ rads exceeds the limit for organic insulators. Although the radiation damage limits for insulators are uncertain, the values used here are comparable to those used in other design studies [27-29].

V.B Heat Removal from Magnets

Joule heating and heat removal calculations have been made using standard analytical expressions. The use of LN2 cooled resistive magnets has helped to drop the Joule heating losses from the TF and CS coils to 109 MW. This is more than an order of magnitude less than the amount of heat that would have to be removed (~ 1.3 GW) had we selected water-cooled magnets. The power required to operate the LN2 refrigeration cycle is, however, considerable. We estimate that 7 W of electricity is needed for each W of heat that must be removed [30]. Therefore, the total amount of electric power required to operate and cool the TF and CS coils is 872 MW. Although we have not designed the PF coil system, there is no comparable constraint on the cross section area that can be used to reduce resistance, and we allow 100 MWe for dissipation and LN2 refrigeration in the PF coils. The total, 972 MWe, is a significant fraction of the total recirculating power of the plant, but it is still smaller than the amount that would be needed with water as the coolant.

V.C First Wall

The first wall is the material surface closest to the plasma and, along with the divertor plates, absorbs the radiation and charged particle energy escaping from the plasma. It is necessary to insure that the first-wall material can withstand the various thermal and coolant stresses (i.e. satisfy the ASME structural criteria) and that the coolant can remove the heat that is deposited in the first wall. Some constraints on the FTWR design are the need for compatibility between the first wall and the transmutation reactor coolants and the need to minimize neutron absorption in the first-wall.

A steel similar to HT-9 was selected as the first wall material, since there is considerable experience with this material in the nuclear field [31], it is being developed in the fusion program, and it has been investigated as a first wall material in several fusion reactor studies [32-35].

The same lead-lithium coolant used in the transmutation reactor was chosen as the first wall coolant. Although liquid metals have a significantly lower heat capacity than water, their selection as a first wall coolant avoids any potentially adverse reactions between water and the transmutation reactor's lead-lithium coolant. In addition, since pressurization is not required, a thinner first wall is possible, which is important from the neutron economy point of view.

The basic first-wall design configuration was adapted from the ITER design [16]. A two-loop design was adopted (i.e. there are two independent coolant loops, one on the inboard and one on the outboard sides). The plasma facing surface is coated with a 0.5 cm layer of Beryllium, and the structural part of the first-wall consists of a 2 cm thickness of HT-9-like steel, with 9 mm diameter coolant channels spaced at a distance of two centimeters.

The first wall was designed for a heat flux of 0.5 MW/m^2, higher than the reference value of 0.34 MW/m^2, to provide some margin for peaking, unexpected transients and possible lower Q_p or higher fusion power operation. For the design heat flux, 38.25 MW of power will have to be removed from each of the two coolant loops. There are 700 coolant channels in the inner part and 1260 in the outer. A flow velocity of 1.13 m/s is required for inlet and outlet coolant temperatures of 548 K and 848 K, respectively.

Although the radiation damage limit for HT-9 type ferritic steels is not yet known, estimates in the range 100-200 dpa have been used (32-35) for fusion neutron spectra. The 623 day reference fuel cycle for the FTWR produces 21 dpa in the first wall. Thus, using the lifetime range 100-200 dpa, we estimate the first wall will have a lifetime of 5-10 fuel cycles. The plant design lifetime of 40 FPY is slightly more than 23 fuel cycles. This means that the first wall will have to be replaced (during a refueling shutdown) about 2-4 times over the plant lifetime.

VI. TRANSMUTATION REACTOR DESIGN

VI.A Materials & Geometry

The transmutation reactor consists of the following materials. The fuel is a transuranic zirconium alloy (TRU-10Zr) dispersed in a zirconium matrix and clad with a steel similar to HT-9. The relative amounts of actinides and zirconium in the fuel region are adjusted to achieve the desired neutron multiplication (k_{eff} = 0.95) at the beginning of each cycle. At equilibrium, the actinides will constitute approximately 45% of the fuel volume. The coolant and tritium breeding material is the eutectic Li17Pb83. Properties of these materials are given in Appendix A.

The geometric configuration of the FTWR is shown in Fig. 1. The blanket is the region inside of the toroidal field coils and outside of the plasma chamber. The blanket consists of the transmutation reactor, reflector, shield, first wall, and vacuum vessel. The transmutation reactor region, where the actinide-containing fuel assemblies are located, is outboard of the plasma and inside the toroidal field coils. The design of the FTWR transmutation reactor is based on the ANL ATW blanket design studies [17,36]. The same pin and assembly geometry was used, with the exception that the length of the assembly was increased to 228 cm. Table 8 gives the basic data for the fuel assembly design.

Table 8 Fuel Assembly Design

Pin Diameter (cm)		0.635
Clad thickness (cm)		0.05588
Pitch		Triangular
Pitch to Diameter		1.727
Pins per assembly		217
Structure Pins		7
Fuel Smear density		85%
Hexagonal Assembly Pitch		16.1
Assembly Length (cm)		228
Assemblies		470
Average Power Density (kW/l)		124
Volume %	Fuel	17.01
	Structure	10.44
	Coolant	69.55
Materials	Fuel	TRU-10Zr/Zr
	Structure	HT-9
	Coolant	Li17Pb83

The assemblies will be placed on the outboard side of the plasma chamber as shown Figure 7. The reactor region is approximately 40 cm thick and will consist of 470 assemblies, of which approximately 1/5 will be configured as 'half assemblies' placed in the gaps along the interior and exterior surfaces of the reactor region to produce a more uniform annular distribution.

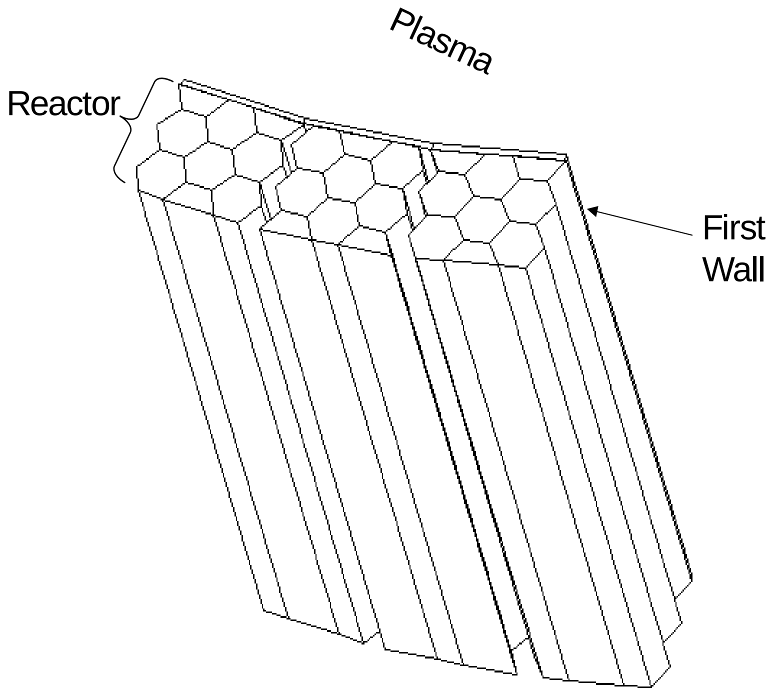


Fig. 7 Transmutation Reactor Configuration Outboard of Plasma Chamber

VI.B Nuclear Design

The nuclear analysis was performed with the same codes and similar methodology used in the ANL ATW design studies [17,36]. The fuel cycle analysis was performed with the REBUS fuel cycle code [37]. Within this code, the neutronics calculations were performed using the DANT [38] code to perform 2-D discrete ordinates transport calculations with material-dependent multi-group cross section libraries based on the ENDF/B-V.2 nuclear data library processed using the MC^2-2 [39] and SDX [40] codes for a 34 group energy structure. The REBUS input specifies the compositions, geometries, and all other necessary fuel cycle parameters. The neutronics calculations were performed using an R-Z geometry model, with R in the direction of the major radius.

The blanket power consists of the fission power, the energy deposited by the 14.1 MeV fusion neutrons, and all of the exoergic reactions such as ^{6}Li (n,T). The total power includes the 3000 MW of blanket power, the alpha fraction (1/5) of the fusion power, and the auxiliary heating power of the plasma.

Under this operating scheme, the fission power, fusion power, and total power will vary over the cycle. In the reference cycle, the total system power will rise from 3029 MW at beginning of cycle (BOC) to 3093 MW at the end of cycle (EOC).

The nuclear design requirements are shown in Table 9. The neutron multiplication at the beginning of each cycle was limited to $k_{eff} = 0.95$, to provide a large margin against accidental criticality. To achieve this value of k_{eff}, the fuel enrichment is adjusted. Compensation for reactivity decrease with burnup could be accomplished in a number of ways. We have chosen to proceed with a simple operational scheme in which the fusion neutron source strength will increase over the cycle to maintain a constant power of 3000 MW in the transmutation reactor. The fusion power will rise from 41 MW at the beginning of cycle to 133 MW at the end of cycle. Reactivity loss from fuel burnup is not compensated by any type of reactivity adjustment.

Table 9 Nuclear Design Requirements

Requirement	Limit
Criticality Safety	$k_{eff} \leq 0.95$
Tritium self-sufficient	Tritium inventory $\geq$ required startup inventory
Fuel Integrity	Fuel Cladding Irradiation < 200 dpa [41]
First Wall Integrity	First Wall Irradiation < 200 dpa [41]
Neutron source strength	$P_{fus} \leq 150$ MW
Fuel Material Composition	≤ 45% actinides by volume [17]
Heat Removal	$P_{blanket} \approx 3000$ MW
Plant life	40 FPY
Plant Availability	60%

The fusion power limit of 150 MW determines the maximum reactivity swing over a cycle. Radiation damage of the fuel cladding limits the achievable burnup in a fuel element. To maximize the cumulative burnup while limiting the reactivity swing, a batch fueling scheme was adopted. The assemblies will be loaded with 4 batches of fuel in a roughly out-to-in pattern, with the 'fresh' SNF fuel being loaded furthest from the plasma and the highest burned fuel closest to the plasma neutron source.

The nuclear performance is summarized in Table 10. The reference fuel cycle length is 623 full power days (FPD). A discharge burnup of 25% of the actinides will be achieved during the fuel in-reactor residence of four 623 FPD cycles. The actinide fission rate is 1.13 metric tonnes (MT) per full power year (FPY). Over the 40 FPY lifetime, the transuranic inventory recycled from approximately 4,500 MTU of spent light water reactor fuel will be fissioned in a single FTWR.

Table 10 Fuel Cycle Performance Summary

Cycle Length (full power days)	623
Enrichment (volume fraction TRU)	47%
Beginning of Cycle Neutron Multiplication	0.950
End of Cycle Neutron Multiplication	0.852
Fuel Batches	4
Actinide Loading (MT)	27
Discharge Burnup (4 cycles)	25%
First Wall Irradiation Rate (dpa/cycle)	21
Cladding Irradiation (dpa/4 cycles)	150
SNF Waste Transmutation (MTU/FPY)	102

VI.C Heat Removal and Pumping Power

We have divided the blanket into four large regions in order to estimate the pumping power required to remove the heat. These regions are the reactor, reflector, shield, and first wall. The requirements of the first wall were discussed in section V. The majority of the energy will be from fission in the reactor region, with much smaller amounts deposited in the other regions. The inlet temperature in each region is the same and the flow velocity through each region is adjusted so that the outlet temperatures are also the same. Following the ARIES-I design report [42], the inlet and outlet temperatures are set at 548 K and 848 K, respectively. The resulting flow velocity is 0.76 m/s in the reactor region. The total coolant mass flow rate for the entire blanket is 53 MT/s.

Comment: Flow velocity

Comment: Coolant Mass Flow Rate

The pumping power requirement has three separate components--the conventional friction losses, the potential energy gains, and the magnetohydrodynamic (MHD) losses. The density of the heavy Li17Pb83 coolant causes the power required to lift the fluid to be significant. We assume that the resulting potential energy increase will be lost in the heat exchanger. The pumping power calculation is described in Appendix D. Each region has three flow paths that need to be included. The first is the path for moving the coolant horizontally from outside the toroidal field coils into the region of interest, secondly, the flow path for moving it vertically through the region of interest, and finally, the flow path for moving it horizontally back outside the toroidal field coils.

Table 11 summarizes the pumping power calculations. The total pumping power, based on a 90% pumping efficiency [34], is 130 MW, of which 123 MW is from MHD losses. If an electrical insulator were to be developed to coat the piping and fuel cladding, the MHD pumping loss could be reduced to effectively zero.

Table 11 Blanket Pumping Power

Parameter	Reactor	Reflector	Shield
Radius from centerline (m)	4.25	3.75	3.61
Flow Length Through Region (m)	2.28	2.38	2.48
Magnetic Field in Region(T)	4.25	4.82	5.00
Flow Length To/From Region (m)	0.90	1.47	1.60
Magnetic Field To/From Region(T)	3.84	4.03	4.09
Region Power (MW)	2922	28	63
Peaking Factor	1.50	2.05	7.33
Mass Flow Rate (kg/s)	51630	491	1115
Flow Velocity (m/s)	0.76	0.04	0.08
MHD Pumping Through Region (MW)	74	0.0	0.2
MHD Pumping Power to/From Region (MW)	48	0.04	0.18
Friction Pumping Power (MW)	6.7	0.00	0.00
Gravity Pumping Power (MW)	1.1	0.01	0.02
Total Pumping Power (MW)	130	0.09	0.42

VI.D Tritium Breeding

The requirement to produce tritium from neutron capture in lithium has a significant impact on the overall design. We have chosen to include lithium in a liquid form as part of the Li17Pb83 to allow continuous recovery of the tritium. The FTWR is designed to be tritium self-sufficient; i.e., although the initial tritium inventory required to startup the reactor will be acquired externally, no additional tritium will be required from external sources for the life of the plant.

The BOC inventory is a function of the fusion rate and the operating parameters of the tritium system. We used a simple estimate of the beginning of cycle tritium inventory--a tritium inventory equivalent to the total number of fusions occurring in the first 30 full power days of operation must be available at the beginning of each cycle. The BOC tritium inventory for the reference fuel cycle is 120 g. The cycle length is not very sensitive to this parameter. A larger inventory requires slightly higher tritium production to offset the higher radioactive decay rate, which is very small relative to the fusion rate.

The fusion power, hence the tritium consumption rate, will increase by nearly a factor of 4 over a cycle. The tritium production rate will also increase somewhat because of changes in the spectrum due to a higher fraction of 14.1 MeV neutrons. Over a cycle, the tritium production will initially be larger than the fusion rate, and the inventory will grow until the tritium consumption rate equals the tritium production rate and then fall rapidly as the tritium is burned at an increasing rate. The peak tritium inventory for the reference cycle in the FTWR is ≈ 1000 g. The cycle length is limited to the time at which there would be just enough tritium to satisfy the startup requirements for the next cycle, allowing for a conservative 90 days of decay between cycles, which requires an EOC tritium inventory of 121g. To achieve this, the lithium must be enriched to 20% ^{6}Li.

The threshold for tritium production in ^{7}Li is 2.82 MeV, well below the energy of most fission neutrons. In this design the neutron spectrum is predominately in the region of the minima of the natural lithium cross section, below the threshold of ^{7}Li and above the epithermal resonance of ^{6}Li. We have not attempted to exploit the larger cross sections of ^{6}Li at lower energies, but if tritium production needs to be increased, the addition of graphite or other moderators in the reflector and/or shield should allow for large increases in the tritium production by shifting the spectrum down into the ^{6}Li resonance.

VII. REFLECTOR & SHIELD DESIGN

The purpose of the shield is to protect the magnets from radiation damage, and the purpose of the reflector is to redirect escaping neutrons back into the transmutation reactor. The shield-reflector is located just in front of the toroidal field magnets between the magnets and the sources of neutrons from the plasma and the transmutation reactor, and on the top and bottom of the plasma and the transmutation reactor (see Figs. 1 and 2). We used the compositions of the reflector and shield from the ANL ATW design studies [17] shown in Table 12.

The magnets are designed as lifetime components. Radiation damage limits to magnet insulators of 10^{11} rads for organic insulators and 4×10^{22} fast neutrons per cm^2 for the inorganic insulators [27] were used as design criteria. Transport calculations determined that the maximum radiation doses in the TF magnets would be 1.5×10^{12} rads and 1.8×10^{22} n/cm^2, which implies that the present shield design would allow the use of ceramic insulators, but not organic insulators. However, the insulator radiation damage limits are rather uncertain, and it is possible to choose more effective shield materials.

The minimum thickness of the inboard reflector plus shield plus vacuum vessel plus first- wall is approximately 30 cm. The reflector and shield are 8.5 cm and 17 cm thick, respectively. Varying the composition of the reflector/shield ratio showed that this is about the minimum. The same thicknesses are used above and below the plasma and the transmutation reactor and outboard the reactor.

The total thickness inboard of the plasma is 40 cm. This includes 30 cm for the reflector, shield, first wall, and vacuum vessel, plus a 10 cm gap to accommodate the assembly of the components.

Table 12 Reflector and Shield Composition

Region	Structure (HT-9)	Coolant (Li17Pb83)	Boron Carbide (B_4C)
Reflector	70%	30%	
Shield	25%	18%	45%

VIII. TRANSMUTATION FUEL CYCLE ANALYSIS

VIII.A LWR Waste / Transmutation Reactor Feed Composition

The spent nuclear fuel that will be transmuted by the FTWR will ultimately come from a very large number of light water reactors that have been and will be operated under a wide range of operating conditions with varying fuel design, discharge burnups, and storage times. This will result in significant variance in the feed composition to the FTWR. For the reference fuel cycle analysis, we use a single feed composition that is representative of the material we would expect to receive. The composition is based on removal of 99.995% of the uranium [43] from the remaining actinides. Since many of the minor components are important to some of the parameters we are evaluating, a complete isotopic composition is needed. This was not available, so the depletion of a PWR pin cell was performed using SCALE 4.4 [44]. A design and burnup calculation that gives reasonably good agreement with Ref. 42 for the major isotopes was performed. This should be representative of the composition of the minor isotopes and fission products that will be present. Table 13 shows the reference composition compared with that used by ANL for their design studies [43] and with the average composition from the Yucca Mountain Environmental Impact Statement [45]. The differences all tend to be fairly small and should not have a significant impact on the reference fuel cycle calculations.

Table 13 Transmuter Feed Actinide Composition

		Absolute Difference with	
Isotope	Design Composition	ANL [43]	YMEIS [45]
U235	0.0039%	0.00%	0.00%
U236	0.0018%	0.00%	0.00%
U238	0.4234%	-0.05%	0.00%
Np237	4.3128%	-0.71%	-1.29%
Pu239	53.9014%	0.71%	1.73%
Pu240	21.2309%	-0.30%	0.15%
Pu241	3.8702%	0.09%	0.33%
Pu242	4.6769%	-0.01%	0.05%
Am241	9.1838%	0.22%	-0.25%
Am242m	0.0067%	-0.01%	-0.01%
Am243	1.0205%	0.09%	-0.18%
Cm243	0.0018%	0.00%	0.00%
Cm244	0.1158%	0.01%	-0.04%
Cm245	0.0125%	0.00%	-0.01%
Cm246	0.0010%	0.00%	0.00%

VIII.B Waste Processing

The waste processing system for the FTWR will be identical to the waste processing system being developed for the ATW system [46]. The general concept of this system is shown in Figure 8. The waste processing system consists of three basic components. The first is a uranium extraction system (UREX) that will separate the bulk uranium and fission products in the SNF from the transuranic elements. The transuranic elements and the rare earth fission products will then be transferred to a pyro-metallurgical system (Pyro-A) that will separate the rare earths from the transuranic elements and convert the latter to a metallic form for fuel manufacturing. The discharged FTWR fuel will be sent to a separate pyrometallurgical system (Pyro-B) where the residual actinides will be recovered. The recovered materials from Pyro-A and Pyro-B will be blended together and manufactured into new fuel elements for the FTWR.

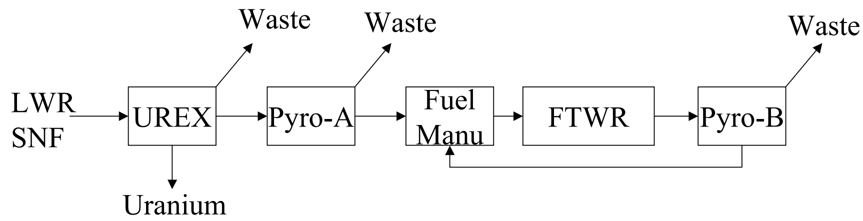


Fig. 8 Waste Processing Flow Diagram

Many of the performance parameters are very sensitive to the performance of the waste processing systems. The UREX system is assumed to remove 99.995% of the uranium [43] and all of the fission products that are not rare earth elements. The Pyro A system is assumed to remove 95% of the rare earth fission products [17] and recover 99.9% of the actinide elements. The Pyro B system is assumed to remove 95% of the rare earth fission products, remove 100% of all other fission products, and recover 99.9% of the actinide elements. In addition to the recovery fractions, the total fraction of transuranics that end up in the waste stream is a strong function of fractional burnup achieved during each residence in the FTWR. The following equation shows this relationship.

$$f_{waste} = f_{UREX}^{TRU} + (1 - f_{UREX}^{TRU} + M_{SNF}^{U}(1 - f_{UREX}^{U}))(f_{Pyro-A} + \frac{(1 - f_{Pyro-A})(1 - f_{BU})f_{Pyro-B}}{1-(1-f_{BU})(1-f_{Pyro-B})}) \quad (3)$$

f_{UREX}^{TRU} = Fraction of transuranics lossed in UREX process = 0

f_{UREX}^{U} = Fraction of uranium recovered in UREX process = 99.995%

M_{SNF}^{U} = Relative mass of uranium in LWR SNF = 95

f_{Pyro-A} = Fraction of actinides lossed in Pyro-A process = 0.1%

f_{Pyro-B} = Fraction of actinides lossed in Pyro-B process = 0.1%

f_{BU} = Fractional burnup of actinides in a single residence in the FTWR

This relationship includes the uranium that is not recovered in the UREX process as transuranics, since most of this will be converted to transuranics in the FTWR. For the FTWR, each MTU of SNF will result in 70 g of transuranics in the waste stream. This is a 99.4% destruction of the transuranics originally present in the SNF.

VIII.D Equilibrium Cycle Calculations

Initially, the first generation of FTWRs will only have feed from LWR SNF. The pyrometallurgical technology allows for a very short decay period of the SNF before recycling. The reference fuel cycle assumes that the FTWR fuel will remain in the reactor for 4 cycles and then be reprocessed, blended with 'fresh' SNF and fabricated into new fuel elements for re-insertion into an FTWR. Over the 40 FPY plant life of the first generation of FTWRs, the original charge of LWR feed will be reprocessed 5 times.

This implies that fuel composition in the first generation FTWRs will be very near equilibrium well before the end of life. The earlier cycles can be loaded to perform similarly to the equilibrium cycle. The initial charge of the reactor and the first reload batch will require approximately 3500 MTU of LWR SNF to manufacture these fuel elements. Following this, approximately 190 MTU of LWR SNF will be processed in each subsequent 623-day cycle. A first generation FTWR will process approximately 74 MT of transuranics from LWR SNF of which approximately 56% will be fissioned, 0.2% will be lost to the waste streams, and 44% will be used in a second generation FTWR.

The second and subsequent generations of FTWRs will use the fuel from the previous generation FTWRs and therefore operate in the equilibrium mode over their entire life. Repeated recycling of the discharged transuranics from FTWRs in successive generations of FTWRs will ultimately result in the destruction of 99.4% of the transuranics discharged from LWRs operating on the OTC.

The change in composition is summarized in Table 14. All values are the mass fraction of the total actinide inventory. This table shows that, even in this very hard neutron spectrum, there is a significant shift to the higher elements. The curium concentration increases by nearly a factor of 10. The increase in the uranium concentration results from the build up of ^{234}U. A high concentration of ^{238}Pu builds up in the reactor from absorption in ^{237}Np and alpha decay of ^{242}Cm. The ^{238}Pu decays to ^{234}U with a half-life of 87.7 years. Plutonium will still constitute the majority of the mass of actinides in the reference fuel cycle, but the isotopic composition will change dramatically. The ^{239}Pu fraction drops from 54% of the mass of all actinides and 63% of the plutonium mass in the LWR SNF feed to 23% of the actinide mass and 30% of the plutonium mass in the fuel discharged from the reference FTWR cycle.

VIII.E Transmutation Performance Characteristics

The FTWR is essentially a hazardous waste incinerator. Its primary goal is to take a hazardous material and convert it to less hazardous materials that are easier to dispose of in a manner that is cheaper than the alternative of directly disposing of the original spent nuclear fuel. The FTWR would not eliminate the need for a high-level waste repository, but it would greatly reduce the performance that must be achieved by the repository to protect the public.

Table 14 Change in FTWR Actinide Composition over the Four 623 Day Cycles between Reprocessing

Element	SNF Feed	FTWR Charged	FTWR Discharge
Mass (MT)	1.93	7.87	5.88
U	0.4%	4.0%	5.1%
Np	4.3%	3.5%	3.2%
Pu238	1.2%	4.8%	5.9%
Pu239	53.9%	31.0%	23.1%
Pu240	21.2%	30.6%	33.8%
Pu241	3.9%	3.1%	3.3%
Pu242	4.7%	9.1%	10.7%
Am	10.2%	12.9	13.5%
Cm	0.1%	1.0%	1.5%

The hazardous waste incinerator will charge a fee for taking the waste, perhaps generate revenue by selling its net electrical production, and be assessed a fee for disposing of its own, hopefully, less hazardous waste.

This initial analysis is not sophisticated enough to evaluate the cost or to do a thorough assessment of the relative hazard of the OTC versus the FTWR cycle. Therefore, we evaluate surrogate figures of merit to provide indications of performance of the FTWR as a hazardous waste incinerator. We examine two parameters, mass flow and toxicity flow.

The mass flows of the various elements and of a few specific isotopes are given in Table 14. For the Once Through Cycle, roughly 11,000 g of transuranics will be placed directly in a repository for each metric tonne of initial uranium content (MTU) of light water reactor fuel discharged. When the same SNF discharged from a LWR is reprocessed and cycled through the FTWR transmutation cycle, approximately 70 g of transuranics per MTU will end up in the repository. This is a 99.4% reduction in the mass of the transuranics that ultimately end up in the repository.

The toxicity flow is often evaluated for transmutation systems. Figure 9 shows the flow diagram for the FTWR cycle. The toxicity is defined as the cubic meters of water required to dilute the given material to the radioactive concentration guides for continuous ingestion from water. The toxicity was calculated using the water dilution factors included in SCALE 4.4 [44].

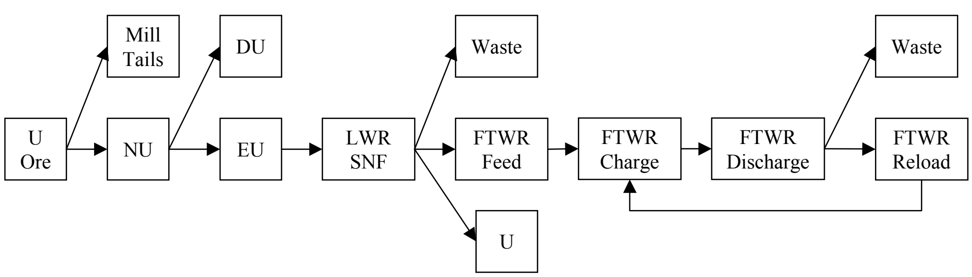


Fig. 9 FTWR Toxicity Flow Diagram

The toxicity is strongly time dependent. Initially, the toxicity is dominated by the highly radioactive, but short-lived, fission product isotopes. As the short-lived isotopes decay away, the medium-lived actinides and fission products become important. At very long decay times, the daughter products of the very long-lived isotopes, such as ^{238}U, will dominate the toxicity. To put these numbers in context, the toxicity flow for the entire nuclear fuel cycle (as depicted in Fig. 9) is given in Table 15. This table includes the toxicity as a function of time for each stage of the fuel cycle and for the separate components produced at each stage. For example, line 1 gives the toxicity of uranium ore as it exists in nature, while lines 2 and 3 give the toxicity of the two components of uranium ore - natural uranium and the mill tails.

Table 15 Fuel Cycle Toxicity Flow For 1 Metric Tonne of Enriched Uranium in Original LWR Fuel

Source	Component	Mass (MT)	Toxicity (m^3 of H_2O) at Time (yr) 0	100	500	1,000	10,000	100,000	1,000,000
U Ore	U Ore	7.700	1.2E+08	1.2E+08	1.2E+08	1.2E+08	1.2E+08	1.2E+08	1.2E+08
U Ore	Tails	0.000	1.2E+08	1.2E+08	1.2E+08	1.2E+08	1.1E+08	4.8E+07	1.2E+04
	NU	7.700	1.5E+05	2.9E+05	3.9E+05	5.8E+05	9.1E+06	7.5E+07	1.2E+08
NU	DU	6.700	7.4E+04	1.8E+05	2.1E+05	2.5E+05	2.3E+06	2.4E+07	9.2E+07
	EU	1.000	8.0E+04	1.1E+05	1.8E+05	3.3E+05	6.8E+06	5.2E+07	2.9E+07
EU	LWR SNF	1.000	4.6E+11	2.4E+10	7.6E+08	4.3E+08	1.1E+08	6.8E+07	3.0E+07
LWR SNF	Recovered U	0.955	7.1E+04	9.4E+04	1.3E+05	2.3E+05	5.1E+06	4.0E+07	2.0E+07
	Waste	0.033	4.5E+11	2.2E+10	2.9E+06	1.3E+06	9.8E+05	7.8E+05	5.6E+05
	FTWR Feed	0.011	8.0E+09	1.5E+09	7.6E+08	4.3E+08	1.0E+08	2.7E+07	9.1E+06
FTWR	Charge	0.046	3.3E+12	9.1E+09	3.4E+09	1.8E+09	4.6E+08	4.8E+08	9.8E+07
	Discharge	0.046	2.8E+13	1.0E+10	2.6E+09	1.4E+09	3.6E+08	4.6E+08	8.8E+07
FTWR Discharge	Reload	0.034	1.4E+12	7.7E+09	2.6E+09	1.4E+09	3.6E+08	4.6E+08	8.8E+07
	Waste	0.012	2.7E+13	2.8E+09	3.1E+06	1.7E+06	6.7E+05	7.2E+05	2.7E+05
All Waste Streams	OTC Waste	7.700	4.6E+11	2.4E+10	8.8E+08	5.5E+08	2.2E+08	1.4E+08	1.2E+08
	FTWR Waste	7.700	2.7E+13	2.5E+10	1.3E+08	1.2E+08	1.2E+08	1.1E+08	1.1E+08
High Level Waste	OTC Waste	1.000	4.6E+11	2.4E+10	7.6E+08	4.3E+08	1.1E+08	6.8E+07	3.0E+07
	FTWR Waste	0.045	2.7E+13	2.5E+10	6.0E+06	3.0E+06	1.7E+06	1.5E+06	8.3E+05

NU - Natural Uranium; EU - Enriched Uranium; DU - Depleted Uranium; LWR SNF - Light Water Reactor Spent Nuclear Fuel

There are two waste system summaries for the OTC and FTWR scenarios. The first (lines 14 and 15) treats the uranium streams as part of the total waste stream, and the second (lines 16 and 17) includes only the wastes that must end up in a high-level waste repository. The repository waste for the OTC is the LWR SNF shown in the sixth row of Table 15. For the FTWR, all fission products and actinides that are not recovered and recycled back into the FTWR are assumed to go to a repository, as indicated in lines 8 and 13.

Initially, the waste from the FTWR fuel cycle has a greater toxicity than that of the OTC, because of the creation of additional fission products and actinides in the SNF that is recycled in the FTWR. In the 100 to 500 year time period, the toxicity of the FTWR waste falls below that of the OTC. At very long times, the radiotoxicity of the ^{238}U daughters in the depleted uranium dominates the toxicity and only a small reduction in radiotoxicity is produced by the FTWR fuel cycle compared to the OTC, when the uranium is considered as part of the waste stream.

However, the uranium can be separated from the high-level waste stream and stored in a low-level waste facility. If only the repository requirements for high-level wastes are considered (the case depicted in the last two lines of Table 15 and in Fig. 3), the toxicity of the FTWR fuel cycle waste will fall well below that of as-mined U ore in about 500 years. The toxicity of the high-level waste in the SNF from the OTC, on the other hand, requires about 7,500 years to be reduced to this level of toxicity. There are other hazard metrics which would dictate longer periods of storage, but this comparison is indicative of the reduction in hazard potential that can be achieved by recycling SNF in FTWRs.

VIII.F System Deployment

According to the DoE Integrated Data Base Report [47], the US inventory of discharged LWR SNF was 34,252 MTU in 1996. The inventory is expected to grow at a rate of slightly over 2,000 MTU/yr for more than 10 years. There are a number of scenarios for the total inventory of SNF that will be discharged after that. A realistic estimate is that the present nuclear capacity will be maintained into the foreseeable future, in which case the feed into the inventory will be slightly greater than 2000 MTU/yr. If the nuclear capacity is increased or decreased, then the feed into the SNF inventory will increase or decrease accordingly.

We assessed a simple scenario for the deployment of a fleet of FTWRs in order to give a sense of the magnitude and time frame that would be needed to destroy the backlog of LWR SNF and to support a fleet of LWRs at equilibrium conditions in the future. This assessment is based on the assumption of a constant electrical power generation of 100 GWe from LWRs. This scenario assumes that the first FTWR demonstration facility is deployed in 2020 and operates for 10 years before we enter the slow growth phase. During the slow growth phase a single FTWR is added each year for the next 10 years. This phase is followed by the fast growth phase, during which two FTWRs are added every year. The fast growth phase would last for more than 20 years. This phase is followed by the equilibrium phase, during which FTWRs are added at a rate just sufficient to maintain the equilibrium FTWR fleet. The key parameters are given in Table 16. At equilibrium, each 3.0 GWth FTWR would support 3.0 GWe of LWRs.

This scenario would essentially replace one of every four LWRs that would otherwise be required for a given systems power requirement. The cost of a FTWR would be greater than the cost of the displaced LWR, and the multiple processing costs involved in the fuel cycle of an FTWR would probably be greater than the cost of producing fresh fuel for an LWR. On the other hand, the costs of building additional repositories for long-term storage of LWR SNF would be decreased substantially by using the FTWR. A quantitative cost analysis is necessary, but beyond the scope of this paper.

Table 16 FTWR Fleet Deployment Parameters

Installed LWR Capacity (GWe)	100
LWR Capacity Factor	80%
LWR Thermal Efficiency	35%
Average LWR Burnup (GWd/MTU)	40
LWR TRU concentration in future discharges	1.1%
LWR TRU Inventory Feed Rate (MTU/yr)	2,087
LWR SNF Inventory (MTU) in 2000	42,600
TRU Concentration in SNF in 2000	1.0%
FTWR Fission Capacity (MT/FPY)	1.14
FTWR Availability	60%
Support Ratio (GWe LWR / FTWR)	3.0
Steady State # of 3000 MWth FTWRs	34

Figure 10 shows the inventory of transuranic waste as a function of time for the above scenario, for different assumptions about the availability of the FTWR. The higher the availability in the ‘slow growth’ and ‘fast growth’ phases defined above, the earlier in time the maximum inventory occurs and the lower are both the maximum and equilibrium inventories. With the reference availability (60%), the transuranic inventory would begin to decline after 2050 and would approach equilibrium by approximately 2120.

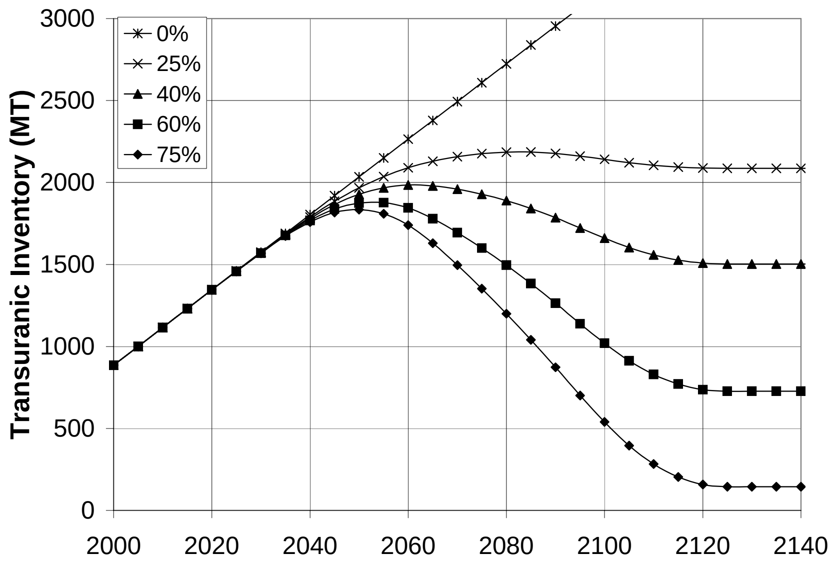


Fig. 10 Estimated Future Transuranic Inventory as a Function of FTWR Availability.
(0% corresponds to the Once Through Cycle without FTWRs)

IX. ELECTRIC POWER PERFORMANCE ANALYSIS

A design objective of the FTWR is electric power self-sufficiency. The level of self-sufficiency of the design is characterized by the electric power amplification factor, also known as the engineering "*Q*" of the reactor, which is just the inverse of the recirculating power fraction, being at least unity:

$$Q_e = \frac{\text{Gross Electric Power Produced}}{\text{Gross Electric Power Consumed}} \geq 1 \tag{4}$$

The gross electric power produced, P_{EG}, is given by

$$P_{EG} = \eta_{th}\left[P_{fus}\left(\frac{1}{5}+\frac{1}{Q_p}\right)+P_{reac}\right] \tag{5}$$

where the first term represents the power deposited on the plasma facing components (mainly charged particles and radiation) and P_{reac} represents the total power (including the fusion neutron contribution) deposited in the transmutation reactor region (predominantly fission power).

The total electric power consumed by the power plant in order to operate its various components, P_{plant}, is then given by

$$P_{plant} = \frac{P_{fus}}{\eta_{CD}^{e} Q_p} + P_{tot}^{TF} + P_{tot}^{CS} + P_{tot}^{PF} + P_{p-FW} + P_{p-reac} + P_{repro} + P_{BOP} + P_{other} \tag{6}$$

where η_{CD}^{e} is the wall-to-plasma electric efficiency of the current drive and heating system, P_{tot}^{TF}, P_{tot}^{CS} and P_{tot}^{PF} are the total (I^2R Joule heating term plus refrigeration) electric powers required to operate the TF, CS and PF magnetic coil systems, P_{repro} is the power required to reprocess fuel on site, P_{BOP} is the balance-of-plant power, $P_{p\text{-}reac}$ is the total pumping power for the transmutation reactor, $P_{p\text{-}FW}$ is the pumping power for the first wall and P_{other} accounts for miscellaneous powers that are not accounted for explicitly.

Values for these powers and efficiency factors for the FTWR reference design are shown in Table 17. Most of these values are calculated. However, some numbers (η_{th}, P_{tot}^{PF}, P_{repro}, P_{BOP}, P_{other}) have been estimated by direct scaling from comparable design studies. P_{repro} and P_{BOP} were estimated from a cost estimate of these same facilities for an ATW design [48].

Table 17 Reference Design Powers & Efficiencies

P_{fus} (MW)	150
P_{reac} (MW)	3000
P_{tot}^{TF} (MW)	656
P_{tot}^{CS} (MW)	216
P_{tot}^{PF} (MW)	100
$P_{p\text{-}FW}$ (MW)	2.5
$P_{p\text{-}reac}$ (MW)	131
P_{repro} (MW)	23
P_{BOP} (MW)	6
P_{other} (MW)	5
η_{th} (%)	40
η_{CD}^{e} (%)	70

Using these values, the calculated electric power amplification factor Q_e for the reference design is about 1.0, i.e. the FTWR produces all the electricity that it needs to perform its mission, transmuting spent nuclear fuel.

In case one or more of the numbers in Table 17 turn out to be less favorable than anticipated, we can maintain electrical self-sufficiency by adding fuel assemblies to the fission reactor in order to increase the power output. This is demonstrated in Table 18, where the electric power amplification factor Q_e is calculated for the reference design with one-half and one additional rows of fuel assemblies. The power requirements increase slightly (only the P_{BOP}, P_{repro}, and $P_{p\text{-}reac}$ terms would increase), so the excess electricity raises the value of Q_e. It can be seen that there is enough margin to accommodate reasonable uncertainties in the powers and efficiencies listed in Table 17.

Table 18 Effects of Added Fuel Assemblies

Added Rows	0	1/2	1
P_{fus} (MW)	150	150	150
P_{reac} (MW)	3000	3600	4200
Q_e (MW)	1.0	1.17	1.33

This calculation also suggests that the FTWR can produce surplus electricity by increasing the number of fuel assemblies. For example, if the FTWR operated at 6 GWth, the resulting Q_e would be equal to 1.77, resulting in about 1 GWe of surplus electricity.

X. CONCLUSIONS & DISCUSSION

The first major conclusion of this study is that a Fusion Transmutation of Waste Reactor (FTWR) based on Liquid Metal—Metal Fuel Fast Reactor technology and a D-T tokamak fusion neutron source is a feasible option for substantially reducing the quantity and hazard potential of high-level radioactive waste from spent nuclear fuel (SNF) that must be stored in geological repositories. A FTWR which produces 3000 MWth would transmute the transuranic content of about 100 metric tones of SNF per full-power-year and would be self-sufficient in producing all the tritium and electricity required for its operation. By repeated recycle of transuranics from SNF in a series of FTWRs, more than 99 % of the transuranics would be destroyed by fission. One FTWR operating with 60% availability would 'support' three commercial Light Water Reactors (LWRs, 1000 MWe each), so that an equilibrium fleet of 34 FTWRs (3000 MWth each) would support the present US commercial nuclear capacity of 100 GWe. This same support level applies also to a mix of FTW and ATW reactors.

The second major conclusion is that a fusion neutron source that met all the requirements, except high availability, for a FTWR could be designed and built today, based on the existing tokamak physics and fusion technology databases. The plasma confinement and stability parameters needed in a FTWR (H ≥ 1, $\beta_N \approx 2.5$) are routinely achieved in operating tokamaks. The required plasma current, plasma energy amplification factor and auxiliary heating power ($I_p = 7$ MA, $Q_p = 1.5$-2, $P_{aux} \approx 80$ MW) are only modest extrapolations from existing tokamaks. Empirical scaling laws predict that steady-state current drive can be achieved with these parameters, based on experience in existing tokamaks. The resistive magnet and liquid nitrogen cooling technology to achieve 8-10.5 T fields is well established. The tritium processing system technology that has been developed for JET and TFTR and in the ITER R&D program should provide an adequate design base for the FTWR. The remote handling technology that has been developed in the ITER R&D program should provide an adequate design base for the fusion neutron source for the FTWR.

The third major conclusion is that availability is the major issue for the FTWR. The equilibrium transuranic inventory (hence the repository requirement) and the size of the FTWR fleet needed to achieve this equilibrium inventory are sensitive to the availability of the FTWR. Achieving an availability of > 50% in the second generation of FTWRs is important. Since we have based the FTWR design on the nuclear and processing technology that is being developed in the US fast reactor program for the ATW, we assume the same high availability for the transmutation reactor in the FTWR as is anticipated in the ATW design. Thus, the availability of the FTWR will be determined by the availability of the fusion neutron source.

There are two elements to the issue of availability of the fusion neutron source: 1) reliable, high availability, routine operation of the neutron source and 2) downtime for the replacement of failed components. Both of these issues suggest the need to build a prototype tokamak fusion neutron source as soon as possible to learn how to achieve routine high availability operation and to learn about any short-term failure modes of the components.

This leaves the issue of long term component failure due to radiation damage, which is common to all transmutation reactors and other devices with a high neutron fluence mission. The most inaccessible components in the FTWR, the toroidal and central solenoid magnets, are shielded sufficiently to be lifetime components. However, some structural components (e.g. the first-wall of the neutron source and the clad and structure in the reactor fuel assemblies) will accumulate high levels of radiation damage. The radiation damage limit for the HT-9-like steel components is not known, but estimated lifetimes in a fusion neutron spectrum are in the range 100-200 dpa. These damage limits would require that first-wall of the fusion neutron source be replaced 2-4 times during the 40 FPY lifetime of a FTWR. Since the wall replacement could be scheduled to coincide with a plan outage for refueling it should not have a substantial impact on average lifetime availability, even if first-wall lifetimes < 100 are encountered.

We tried to make this initial assessment of a FTWR realistic by basing the design concept for the neutron source on the existing tokamak physics and fusion technology databases and by basing the design concept for the transmutation reactor on the nuclear and processing technology that is being developed for the ATW reactor. The major uncertainties in this existing database vis-a-vis the FTWR requirements are in the areas of high availability, steady-state tokamak operation and structural materials lifetime, as discussed above, and only the former would substantially impact availability. However, there will inevitably be design-specific R&D requirements identified by a more detailed assessment of the FTWR at the conceptual design level that includes the mechanical and thermal designs of the magnet systems, the transmutation reactor, the fuel changeout and reprocessing systems, etc. and the safety and environmental analysis.

APPENDIX A - MATERIALS PROPERTIES

COOLANTS

Properties	Li17PB83 [49]	LBE [50]
Density (kg/m^3)	9270	10190
Resistivity (Ω-m)	9.71×10^{-8}	4.29×10^{-7}
Specific Heat, C_p (J/kg -°K)	187	129
Viscosity (mPa-s) @ 698 °K	1.39×10^{6}	1.46×10^{6}

HT-9 [51]

Property	Value
Yield strength (MPa)	307
Ultimate strength (MPa)	396
Thermal conductivity (W/m-°K)	30
Poisson's ratio	3
Density (kg/m^3)	9270
Resistivity (Ω-m)	1.32×10^{-6}

OFHC

Property	Value
Resistivity (Ω-m) @ 100 °K[a]	0.36×10^{-8}
Yield strength @ 100 °K (MPa)	370
Ultimate Strength @ 100 °K (MPa)	470

LN2 [26]

Property @ 70 °K	Value
Density (kg/m^3)	840.0
Specific Heat, C_p (J/kg -°K)	2024
Viscosity, μ (μPa-s)	220
Thermal Conductivity, k (W/m-°K)	0.150

[a] A temperature-dependent model for the OFHC resistivity was used (P.Titus, MIT, personal communication)

APPENDIX B – PLASMA PHYSICS ANALYSIS

Confinement [19]

The ITER Database IPB98(y,2) scaling is used:

$$\tau_E = H\tau_E^{IPB98(y,2)} \tag{B.1}$$

where

$$\tau_E^{IPB98(y,2)} = 0.144 I_p^{0.93} B_0^{0.15} P^{-0.69} \bar{n}_{e20}^{0.41} M^{0.19} R_0^{1.97} A^{-0.58} \kappa^{0.78} \tag{B.2}$$

and the units are in s, MA, T, MW, 10^{20} m^{-3}, amu and m.

Greenwald Density Limit

$$\bar{n}_{e20} \le \frac{I_p(\mathrm{MA})}{\pi a^2} \tag{B.3}$$

L-H mode transition threshold [19]

$$P_{LH}(\mathrm{MW}) = (2.84/M) B_0^{0.82} \bar{n}_{e20}^{0.58} R_0 a^{0.81} \tag{B.4}$$

MHD Stability

$$\beta_t \equiv \frac{\langle n_e T_e + n_i T_i + p_\alpha \rangle}{\dfrac{B_0^2}{2\mu_0}} \le \beta_N \frac{I_p(\mathrm{MA})}{aB_0} \tag{B.5}$$

$$q_{95} = \frac{5a^2 B_0}{R_0 I_p} \frac{1+\kappa^2\left(1+2\delta^2-1.2\delta^3\right)}{2} \frac{\left(1.17-\dfrac{0.65}{A}\right)}{\left(1-\dfrac{1}{A^2}\right)^2} \ge 3 \tag{B.6}$$

Bootstrap Current Fraction [52]

$$f_{bs} = C_{BS}\left(\sqrt{\varepsilon}\beta_p\right)^{1.3} \tag{B.7}$$

where

$$C_{BS} = 1.32 - 0.235 q_{95} + 0.0185 q_{95}^2 \tag{B.8}$$

and

$$\beta_p = \beta_t \left(B_0/B_p\right)^2, \quad B_p = \frac{I_p(\mathrm{MA})}{5a\sqrt{\dfrac{1+\kappa^2}{2}}} \tag{B.9}$$

Fast Wave ICRF Current Drive Efficiency [22]

$$\gamma_{FW} \equiv R_0 n_{e20} \eta_{CD} = 0.062 T_e \left(\text{keV}\right)^{0.56} \tag{B.10}$$

where $\eta_{CD} = I_{CD}$ (MA) / P_{aux} (MW) is the current drive efficiency.

Volt-Second Analysis

Volt-seconds required for startup, $\Delta\Phi_{start} = (\Delta\Phi)_{ind} + (\Delta\Phi)_{res}$ where:

$$\left(\Delta\Phi\right)_{ind} = I_p L_p \tag{B.11}$$

$$\left(\Delta\Phi\right)_{res} = C_{Ejima} \mu_0 R_0 I_p \tag{B.12}$$

where the Ejima coefficient is assumed to be equal to 0.4 [52].

$$L_p = \mu_0 R_0 \left[\ln\left(\frac{8R_0}{a\sqrt{\kappa}} \right) + \frac{l_i}{2} - 2 \right] \tag{B.13}$$

and the internal inductance l_i is given by [53]:

$$l_i = \ln\left(1.65 + 0.89\left(q_{95} - 1\right)\right) \tag{B.14}$$

APPENDIX C - MAGNET ANALYSIS

Central Solenoid (CS) Coil

Volt-seconds:

$$\left(\Delta\Phi\right)_{CS} = \pi B_{OH} R_{fc}^2 \left[1 + \frac{\Delta_{OH}}{R_{fc}} + \frac{1}{3}\left(\frac{\Delta_{OH}}{R_{fc}}\right)^2\right] \tag{C.1}$$

B_{OH} : magnetic field at the central solenoid

R_{fc} : flux core radius

Δ_{OH} : radial thickness of central solenoid

Equation C.1 assumes linear decay of the magnetic field within the CS cross section.

Tensile Stress [54]:

$$\sigma_{CS} = \frac{B_{OH}^2}{2\mu_0}\left(\frac{R_{fc}}{\Delta_{OH}} + \frac{1}{3}\right) \le S_m \tag{C.2}$$

where according to the ASME code, S_m = min [1/3 ultimate stress, 2/3 yield stress]. For composite materials, the maximum stress S_m is estimated from:

$$S_m = \sum_i f_i \times S_{mi} \tag{C.3}$$

where S_{mi} is the maximum allowable stress of material i and f_i is the volume fraction of material i.

Toroidal Field (TF) Coils

Centering Force [54, 55]

$$F_R = \frac{\mu_0 N I_{TF}^2}{2}\left[1 - \frac{1}{\sqrt{\left(1-\varepsilon_p^2\right)}}\right] \tag{C.4}$$

N : number of TF coils

I_{TF} : current per TF coil

and $\varepsilon_p = R_{bore} / R_0$ where R_{bore} the radius of the magnet bore.

Bending Stress

$\sigma_{bend} = F_R / A_{in}$ where A_{in} is the area of the inner leg of the magnet over which the inward force acts.

Tensile Force [54, 55]

$$F_T = \frac{1}{2}\frac{\mu_0 N I_{TF}^2}{4\pi}\ln\left(\frac{1+\varepsilon_p}{1-\varepsilon_p}\right) \tag{C.5}$$

and the corresponding tensile (hoop) stress is equal to

$$\sigma_t = F_T / A_{tor} \tag{C.6}$$

where A_{tor} is the cross sectional area of conductor plus structure, but not including the coolant channels. According to the ASME code, $\sigma_t + \sigma_{bend} \leq 1.5 S_m$ where S_m is defined as above.

APPENDIX D - PUMPING POWER CALCULATIONS

The flow rate through each region is determined from the heat removal requirement

$$P_{th} = \dot{m}C_p \Delta T = \rho v A C_p \Delta T \tag{D.1}$$

where P_{th} = thermal power (W), $\dot{m}$ = mass flow rate (kg/sec), C_p = 1000 (33.77-0.00158 T(K))/173.156 = heat capacity of the coolant (J/kg-K), v = flow velocity (m/s), and A =cross sectional flow area (m^2)

The pumping power is determined for each component of the pressure drop using

$$P_{p,x} = \frac{\Delta p_x A v}{\eta} \tag{D.2}$$

where η is the pumping efficiency and Δp_x is the result pressure drop from losses from the x component.

The friction pressure drop is determined [55] from

$$\Delta p_{fric} = f L_c \rho v^2 / 2D \tag{D.3}$$

where $f = .0014 + .125(Dv\rho/\mu)^{-0.32}$, D =4 A/wetted perimeter = hydraulic diameter and $\mu = .187e^{11640/8.314T(K)}$ = viscosity (mPa-s)

The pressure drop from potential energy gains or gravity was determined using

$$\Delta p_g = \rho g h \tag{D.4}$$

where *h* is the elevation change of the vertical flow, through the region.

The magnetohydrodynamic (MHD) pressure drop is calculated using [55]

$$\Delta p_{MHD} = L_c V B_r^2 \sigma_f \frac{C}{1+C} \tag{D.5}$$

where L_c = flow length for the path, B_r = the magnetic field in region r perpendicular to the flow, σ_f = conductance of the cladding (1/Ohm-m), $C = \dfrac{2\sigma_s t}{\sigma_f D}$, σ_s = conductance of the liquid metal coolant (1/Ohm-m), and t = thickness of the cladding (m)

APPENDIX E - TRANSMUTATION ANALYSIS

The transmutation analysis is performed by two different code packages. The REBUS fuel cycle code [37] performs the FTWR transmutation calculations. The fission products are treated as several lumps in the REBUS calculations. The SCALE 4.4 code [44] was used to determine the composition of the fission product lumps and to calculate the LWR SNF composition.

The REBUS fuel cycle is run in two different modes. The first mode is the enrichment search for the equilibrium fuel cycle. The second mode is the non-equilibrium or depletion mode to determine the behavior of the FTWR over the equilibrium cycle.

For the equilibrium calculations, the beginning of cycle target k_{eff} (0.95) and all fuel cycle parameters (e.g., cycle length, power level, recovery fraction, SNF feed composition) are specified along with the initial guess at the equilibrium enrichment. The REBUS code calculates the flux distribution and reaction rates at the beginning of cycle, and depletes the fuel to the next time step. The flux distribution and reaction rates are then calculated at the end of the time step and the code then adjusts the transmutation matrix and new compositions are calculated for the end of the time step. This process continues until the end of cycle is reached. The end of cycle composition is then processed according to the external cycle parameters, the recovered material is combined with makeup from the LWR SNF feed and a new estimate of the beginning of cycle concentration and enrichment is made. The cycle transmutation calculations are then repeated. The code then iterates until the enrichment is determined for the equilibrium cycle and the concentrations of materials in the equilibrium fuel cycle have converged.

A smaller number of time steps are required in the equilibrium cycle iterations than is necessary to accurately integrate some of the time dependent parameters such as the tritium inventory. To reduce the calculation time, the equilibrium cycle enrichment search is run using a smaller number of time nodes and then a single depletion calculation is performed with a larger number of time nodes. The beginning of cycle fuel concentration is depleted under the same conditions as the equilibrium calculations, except more neutron transport calculations are performed over a cycle, which provides the fission rate, tritium production rate and other data at more points throughout the cycle.

To greatly reduce the calculation time, the number of isotopes in the transmutation matrix can be reduced by lumping the fission products. In a fast spectrum, there are not any fission products with huge cross sections like xenon and samarium in a light-water reactor. There are 10 fission product lumps used in this analysis. Each fission produces two fission product lumps. One for rare earth fission products, a fraction of which are recycled with transuranics, and one for the non-rare earth fission products. These fission product lumps are based on the equilibrium fission product concentration for ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, and ^{241}Pu in a fast spectrum. The fission of other isotopes is assumed to produce the fission products for the isotope with the closest mass. The equilibrium composition of the hundreds of isotopes in the lumps are estimated using the SCALE 4.4 code package [44] with its standard fast reactor cross sections. These compositions are then used to produce multigroup cross sections for the 10 fission product lumps. For the toxicity calculations, the toxicity is assumed to be that of the isotopic mixture used to produce the fission product lumps.

APPENDIX F - TRITIUM ANALYSIS

The time dependent tritium inventory is given by

$$\frac{dT(t)}{dt} = (1-\alpha)\dot{T}(t) - \dot{F}(t) - \lambda_T T(t)$$

$T(t)$ = Tritium inventory

α = tritium reduction factor

$\dot{T}(t)$ = Tritium production rate

$\dot{F}(t)$ = Fusion rate

λ_T = Tritium decay constant

Limit

$$T_{EOC} \geq T_{BOC} e^{\lambda_T t_{down}}$$

T_{BOC} = Startup tritium inventory

T_{EOC} = Tritium inventory at the end of cycle

t_{down} = Down time between cycles

The tritium production rate is reduced by the α term that takes into account all losses other than decay as well as uncertainties in the reactions rates and geometrical modeling errors. The methodology, model, and values used to estimate α were based on the model developed in reference [56]. The above equation expresses the cumulative non-radioactive losses for and infinite number of passes through the plasma. Table F.1 shows the values of the parameters defined in reference [56] used to estimate the non-radioactive loss of tritium. The total non-radioactive losses are estimated at 2.3%. There are significant uncertainties in the parameters used to estimate the losses.

Table F.1 Tritium Loss Model Parameters

Parameter	Value	Definition
ε_1	0	Loss in Blanket
ε_2	0.001	Loss in Breeder Processing
ε_4	0	Loss in Fuel Cleanup and Isotope Separation
β	0.05	Fractional burnup in plasma
f_l	0.0001	Leakage from plasma to limiter
f_{fw}	0.0001	Leakage from plasma to first wall
ε_6	0.001	Loss in plasma exhaust processing
α"	0.02346	Fraction of tritium atoms produced that will be lost

The tritium production rate also needs to be reduced by an amount to account for uncertainty in the calculated tritium production rate. This is very difficult to estimate because these errors result from errors in the neutron spectrum resulting from cross section errors. The geometry model is also a very simplified model of the actual geometry, which produced additional errors in the tritium production rate. A total reduction in the tritium production rate of 7% was used to estimate the tritium inventory. Since tritium self-sufficiency is a requirement, the uncertainty in tritium production rate translates into an uncertainty in the lithium enrichment required to achieve tritium self-sufficiency.

APPENDIX G DEFINITION OF TERMS

ANL	Argonne National Laboratory
ATW	Accelerator Transmutation of Waste
BOC	Beginning Of Cycle
B_4C	boron carbide
CS	Central Solenoid
EOC	End Of Cycle
FPD	Full Power Day
FPY	Full Power Year
FW	Fast Wave
FTWR	Fusion Transmutation of Waste Reactor
HT-9	a ferritic steel alloy
ICRF	Ion Cyclotron Range of Frequency
JET	Joint European Torus
k_{eff}	effective neutron multiplication constant of a fissioning assembly
LBE	Lead-Bismuth Eutectic
Li17Pb83	Lithium-lead eutectic 17 parts Li and 83 parts Pb
LN2	Liquid Nitrogen LN_2
LWR	Light Water Reactor
MHD	Magneto-HydroDynamics
MT	Metric Tonne
MTU	Metric Tonne of initial Uranium
OFHC	Oxygen-Free High Conductivity copper
OTC	Once-Through fuel Cycle
POPCON	Plasma Operating CONtour
Pyro	Pyrometallurgical
Q_e	electric power amplification factor (electric power produced/electric power consumed)
Q_p	plasma energy amplification factor (fusion power/external heating power)
SNF	Spent Nuclear Fuel
TF	Toroidal Field
TRU	Trans-Uranics
UREX	URanium EXtraction system

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FIGURES