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1

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THE REVERSED-FIELD-PINCH (RFP) FUSION NEUTRON SOURCE: A CONCEPTUAL DESIGN*

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ABSTRACT

The conceptual design of an ohmically heated, reversed-field pinch (RFP) operating at ~5-MW/m^2 steady-state DT fusion neutron wall loading and ~124-MW total fusion power is presented. These results are useful in projecting the development of a cost effective, low input power (~206 MW) source of DT neutrons for large-volume (~10 m^3), high-fluence (3.4 MW yr/m^2) fusion nuclear materials and technology testing.

1. INTRODUCTION

A strong experimental database is evolving from a number of relatively small reversed-field-pinch (RFP) devices.[1] Consequently, the design and construction of the next-step RFPs are well under way in both the US and the European Economic Community.[2] A recent study of the commercial prospects of the RFP as a high-power-density, compact fusion reactor[3] has been completed, and a strong economic potential is indicated if the physics established by existing RFPs extrapolates through the next-step devices to the reactor regime. Preliminary scoping studies of RFPs with characteristics between these next-step devices and the reactor regime recently examined the potential of a RFP ignition/burn device as a steady-state source of DT fusion neutrons.[4,5] These results are used to characterize the RFP as a fusion test facility (FTF).

The steady-state FTF/RFP device is based on a low-to-moderate-Q, driven or marginally ignited plasma. The main goal of this device is the generation of fusion-relevant DT neutron currents (I_w = 4-10 MW/m^2) from plasmas that are sufficiently small to operate with a total fusion power of $\lesssim$ 100 MW without large expenditures in driver power. Central to the viability and/or feasibility of this compact approach is the ability to manage heat and particle fluxes in an RFP that differs little in size from the next-step RFP devices.[2]

The basic approach adopted by this FTF/RFP study first developed a quantitative understanding of the available operating parameter space. Cost estimates were also made in the early stages of these analyses to provide guidance. Upon selecting a design point from parametric analyses,[4,5] a two-dimensional vacuum magnetics computation established the size and position of the equilibrium-field (EF) and ohmic-heating (OH) coils, subject to the usual constraints imposed by equilibrium and startup (i.e., plasma breakdown, OH-coil stresses, and power) requirements. A one-dimensional RFP transport model was used to calculate radial density and temperature profiles in the impurity-seeded, highly radiating plasma needed to homogenize heat fluxes in this compact system. The detailed coil

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configuration and plasma profiles were then used in the plasma/circuit simulation to determine the ohmically-heated startup transient leading to steady state operation sustained by oscillating-field current drive (OFCD).[6] With the basic parameters for the plasma, magnetics (e.g., coil sizes, powers, forces), impurity and fueling control, and current drive established,[4,5] including all vacuum, shielding, and cooling requirements, the maintenance and testing requirements and capabilities were formulated. Combined with a conceptual but detailed picture of the testing geometry and the device cost, a procedure of evaluating the performance of the FTF/RFP relative to other approaches is devised, formulated, and evaluated. Given below are the results of each of these subsystem analyses, which combine to give a quantitative mechanical and operational definition of the RFP as a facility for fusion nuclear technology testing.

Before describing the parametric analysis in Section 3 and the design-point results in Section 4, the developing experimental basis for the RFP is summarized Section 2. Cost estimates and performance evaluations are reported in Section 5. Section 6 gives a brief summary and conclusions.

2. ESSENTIAL ELEMENTS OF THE RFP

The general characteristics of the RFP are summarized in Table I. The differences between this toroidal, axisymmetric device and a similarly configured tokamak is best illustrated by the comparison of magnetic-field and magnetic-shear profiles shown in Figure 1; whereas the relatively low-aspect-ratio tokamak is dominated primarily by toroidal magnetic fields, the poloidal and toroidal magnetic fields in the interior of the RFP are comparable in magnitude, with the poloidal field generated by toroidal currents flowing in the plasma actually being appreciably larger than the relatively low toroidal fields outside the plasma. The toroidal field actually reverses direction as its magnitude diminishes towards the outer plasma regions (Figure 1). The resulting high magnetic shear in plasma regions of steep pressure gradients provides high-β MHD stabilization to the RFP (β is the ratio of plasma pressure to confining magnetic-field pressure). It is this considerable reduction in the external-toroidal field and the toroidal-field (TF) coil requirements, and the magnetically efficient confinement of plasma (e.g. high β) by magnetic fields generated from currents flowing in the plasma rather than in external conductors, that leads to many of the positive attributes listed in Table I.

Table II summarizes the main parameters of existing RFP devices as well as those presently under construction; included in this table are parameters for conceptual designs of both the commercial power reactor, TITAN, and the FTF/RFP design being describe herein as a neutron source for fusion nuclear materials and technology testing. Figure 2 gives a comparison of toroidal cross sections for these devices.

The magnetic configuration depicted in Figure 1 represents one in which the energy is minimized.[1,7] These states of minimum energy have been eloquently describe[7] in terms of a phase space defined by the ratio F of the toroidal field at the plasma edge, $B_\phi(r_p)$, relative to the volume-averaged toroidal field, $\langle B_\phi \rangle$, within the separatrix and the ratio Θ of the poloidal field at the plasma edge, $B_\theta(r_p)$, again normalized to the volume averaged toroidal field; the parameters $F = B_\phi(r_p)/\langle B_\phi \rangle$ and $\Theta = B_\theta(r_p)/\langle B_\phi \rangle$

are called, respectively, the reversal and pinch parameters. The minimum-energy states of a pressureless plasma are described by a locus of points in this F-Θ space, which is shown in Figure 3. Typical experimental discharge trajectories are also shown in Figure 3 as they diffuse in time through F-Θ space toward the stable and quiescent RFP configuration where the reversal parameter F is slightly negative ($F \simeq -0.1$-0.5) and Θ characterizes the poloidal-field-dominated nature of this unique configuration. The actual time dependence of $\Theta \propto I_\phi$ and $F \propto B_\phi$ for a range of discharges is illustrated in Figure 4, where I_ϕ is the plasma current. The plasma processes responsible for maintaining a high toroidal field within the plasma in the presence of a low and directionally reversed toroidal field external to the plasma are related to complex current/field fluctuations, which together are called the "plasma dynamo," in that they very much reflect a generator-like phenomenon sustained by continually operative relaxation processes that steer the plasma to a near-minimum-energy state.

Considering the relatively low level of funding being devoted to the world-wide RFP program, progress has been significant in the relatively small devices described in Table I. A synopsis of experimental results[1] is given in Table III. Most of these quantitative results have been used to extend the RFP into the regimes of interest to neutron-source[4,5] and reactor[3] applications. Specifically, the following observations are used in these conceptual design studies:

- Constant-beta scaling ($NT \propto I_\phi^2$, where $N \equiv \pi r_p^2 n$ is the plasma line density).
- Ohmic heating alone is used to bring the plasma to near ignition conditions, with appropriate profile and Z_{eff} adjustments for an impurity-seeded plasma to enhance radiative loss of excess plasma energy, thereby uniformly spreading the heat flux over the first wall.
- Operation at a critical beta limit, $\beta_{\Theta c}$, is enforced, above which energy confinement rapidly degrades.
- An ohmic-heating transport/confinement scaling is used wherein the electron confinement time scales as follows, where $\nu \simeq 0.8$-1.5:

$$\tau_{ce} = C_\nu I_\phi^\nu r_p^2 f(\beta_\theta) \quad (1)$$

$$f(\beta_\theta) = e^{(\beta_\theta/\beta_{\theta c})^n} \text{ if } \beta_\theta > \beta_{\theta c} \quad (2)$$

- Operation of a robust RFP dynamo is invoked to assure the following:
  - "matched-mode" startup, wherein the fields inside and outside the conducting first wall are maintained equal.
  - slow rampup of current with toroidal flux generated primarily by the RFP dynamo sustained/driven from the poloidal-field (PF) coils.
  - oscillating-field current drive (OFCD) is invoked for steady-state operation.

To varying degrees, all these processes have been demonstrated or indicated as potentially possible on existing devices; the main uncertainty is the veracity of these effects in higher-current, hotter plasma. In addition to dynamo-sustained startup illustrated by the experimental result given in Figures 3 and 4, Figures 5 illustrates the constant-beta scaling and the existence of a critical beta limit, and Figure 6 indicates the ohmic transport scaling used in the present study.

These experiment-based results are used to project the FTF/RFP into a plasma and current regime that represents a step beyond the RFX and ZTH devices presently being constructed and scheduled for $I_\phi = 1\text{-}2\ MA$ operation sometime in the early 1990s. Hence, the FTF/RFP devices being proposed herein would not have the necessary physics database until after the year 2000. Current-drive and divertor experiments are not planned for RFX or ZTH until the present >1995 experimental program is completed, although preliminary divertor experiments on the Los Alamos ZT-P experiment have been proposed.[8] The ambiguous OFCD results[9] from ZT-40 and the complete lack of RFP divertor experience represent the main unaddressed physics uncertainties, since both OFCD and TF divertors are essential to the operation of a steady-state, high-power-density FTF/RFP.

3. PARAMETRIC SYSTEMS STUDIES AND FTF/RFP DESIGN-POINT DETERMINATION

The FTF/RFP design-point estimate uses the previously described physics database in a parametric systems analysis based in turn on a steady-state version of a zero-dimensional profile-averaging plasma/circuit simulation code used primarily to model RFP reactor startup phenomena.[3,10,11] These preliminary parametric studies were not based directly on a cost minimization, but the main sensitivity studies examined the dependence of key performance characteristics on two parameters that are directly related to cost: plasma radius (size) and plasma current (power supplies and coils). The preliminary designs that emerged from this procedure where analyzed with a detaiied two-dimensional magnetics model to determine the crucial equilibrium-field (EF), ohmic-heating (OH), divertor-field (DF), and TF-coil designs and related coil geometry. Since the desirable characteristics of a high neutron wall loading coupled with and constrained by a minimum total fusion power depend critically on achieving highly radiating plasma in order to spread heat/particle fluxes and allow the divertor to operate (survive) as primarily an impurity-control system rather than a power-handling system, one-dimensional (steady-state) plasma transport simulations of an impurity-seeded (Xe, high radiation fraction, f_{RAD}), pellet refueled RFP were carried out using a code applied also to the TITAN reactor study and described therein.[3] When combined with integrated plasma/circuit analysis, neutronics studies, and other subsystems studies conducted in parallel [impurity control, edge-plasma analyses, eddy-current (shell) analyses, current-drive studies], the above-described studies gave a firm basis for an integrated-facility conceptual design and detailed cost estimate. The details of the analyses and modeling that lead to these subsystem designs are described elsewhere.[12]

After summarizing in this subsection the parametric studies and design-point suggested therefrom, the mechanical design that results is described in the following Section 4. Cost estimates are then made in Section 5 along with a comparative figure of merit analysis.

Parametric results[4] are expressed as contours in a plasma current-radius (I_ϕ-r_p) phase space, which as noted previously is particularly useful in that it measures indirectly two major cost items: coils and power supplies (I_ϕ) and the torus (r_p). The parametrics model generates on a plot of I_ϕ *versus* r_p lines of constant neutron wall loading, I_w (MW/m^2), average first-wall heat flux, q_s (MW/m^2), total fusion power, P_F (MW), average electron or ion temperature, $T_{i,e}$ (keV), average electron density, n_e(m^{-3}), electron streaming parameter, $\xi = v_{De}/v_{the}$, average toroidal plasma current density, j_ϕ (MA/m^2), plasma loop voltage, $V_\phi(V)$, Lawson parameter, $n\tau_E$ (s/m^3), and an ignition parameter. Selecting the following range for the main variables defines the design window used to guide this study: I_w = 1-5 MW/m^2, $P_F \lesssim$ 100 MW, and $q_s <$ 5 MW/m^2. Figure 7 gives the design window in the r_p-I_ϕ design space for the following base-case parameters: Z_{eff}= 1.0, no anomalous ion heating (f_{OHM} = 0.0), aspect ratio A = 6.0, transport current exponent ν = 1.25 (Figure 6), poloidal beta β_θ = 0.1, full coupling of alpha-particle power to plasma (f_α = 1.0), $F = -0.11$, and Θ = 1.45. This window is set by average first-wall surface heat fluxes in the range q_s = 1-5 MW/m^2, a neutron wall loading $I_w >$ 1 MW/m^2 but below 5 MW/m^2, and a total fusion power $P_F \simeq$ 100 MW. The sensitivity of the size, shape, and location of the design window is shown in Figure 8 as the magnitude of poloidal beta, anomalous ion heating, and transport exponent (β_θ, f_{OHM}, and ν, respectively) are varied. Table IV lists the main parameters for a $I_w \simeq$ 4-5 $MW/m^2, r_p \simeq$ 0.3 $m, I_\phi \simeq$ 10 MA FTF/RFP that provides a "strawman" design for more detailed design and costing elaboration, described respectively, in Sections 4 and 5. This design is somewhat smaller in size than the ZTH experiment presently under construction, and represents an optimistic upper bound in terms of confinement and the operability of an efficient divertor for a highly radiating, impurity-seeded plasma.

4. DESIGN RESULTS

This section summarizes the results of design analyses of the magnetics configuration, the (divertor) impurity-control system, the current-drive system, and the overall maintenance and testing configuration and procedure based thereon. Cost estimates and a comparative evaluation of FTF/RFP performance as a fusion nuclear technology and materials testing device are given in Section 5.

4.1. Magnetics Configuration

A two-dimensional vacuum magnetics model was used to establish the details of closely coupled OH- and EF-coil sets subject to the usual equilibrium, stress, and power constraints. The PF coils were positioned a distance from the plasma, $\Delta b \simeq r_p$, that maximizes the system power density. Both OH and TF coils are fabricated from aluminum alloy to reduce activation at the expense of an ~50% increase in power consumption. Since the EF coils represent the main steady-state power requirement, water-cooled copper alloy was selected for the EF coils. Figure 9 gives a torus cross section that is representative of a near-optimal device. Routine (i.e., daily to monthly) maintenance and servicing generally would be conducted through horizontal motions on the outboard side of the torus, which is divided into relatively independent quadrants that are separated by four toroidal-field divertors (Section 4.2). Installation and longer-term maintenance of the OH and EF coils would occur by vertical access. The OH coils are positioned to exclude the back-bias magnetic flux from the plasma chamber to a level that meets the stray-vertical-field constraint for efficient plasma breakdown.[3] The main PF-coil parameters are listed on Table V. As seen from Figure 9, these constraints are met by a PF-coil set of sufficient outboard openness to allow horizontal removal of either divertor assemblies or torus quadrants, including the TF coils.

The TF coils are positioned at the minor radius immediately outside the shield/test-cell region. The TF coils generally operate at low magnetic fields ($0.9\,T$); during the startup phase, the PF-coil set provides most of the toroidal flux within the plasma through flux conversion by means of the RFP dynamo (Figure 4).[1] The magnitude of the radial magnetic-field ripple relative to the poloidal field, $\Delta B_R/B_\theta \leq 0.3\%$, is chosen to assure acceptably small magnetic islands relative to the distance between the toroidal-field reversal layer and the separatrix. Applying this constraint leads to the TF-coil design summarized in Table V and illustrated on Figures 9 and 10. The moderate centering and overturning forces on the TF coils would be supported by a toroidal strong-back that also serves as the outer surface of each blanket quadrant.

The DF coils represent the last major component of the FTF/RFP magnetics design. Each of four poloidally-symmetric TF divertors (Section 4.2) consists of a single TF nulling coil with flanking coils positioned at each side to minimize the perturbation of the reversed toroidal field. Table V also gives the main parameters for the DF coils.

4.2 Impurity Control

Central to the operation of a compact, high-power-density RFP, whether it is an ignition test device, an FTF, or a commercial reactor, is the control of high heat and particle fluxes. A survey[13] of impurity control options, which included armored and pump limiters, concluded magnetic divertors can operate with the highest heat and particle fluxes. Since the poloidal magnetic field is dominant in the plasma edge, poloidally symmetric limiters and divertors are required to provide magnetic field-line connection lengths that are sufficient for radial diffusion of energy and reduced peak heat fluxes. Furthermore, diversion of the minority (toroidal) field requires less power and minimally perturbs the plasma. A closed TF divertor was found to concentrate the heat and particle fluxes to the collector plate. The open divertor configuration shown in Figure 11 avoids this drawback by moving the collector plate closer to the field null; at this point magnetic-flux surfaces are expanded and poloidally symmetric, in contrast to the closed divertor. The open-divertor geometry also allows room for a larger collector plate.

The plate heat flux without plasma radiation is within a factor of three of the design heat flux of $q_{DIV} \leq 4\ MW/m^2$; a reduction in the plate heat flux is easily achieved by increasing plasma radiation losses by impurity injection (e.g., 0.1% Xe). As a result of the "soft" beta limits[1] observed in RFPs, the plasma parameters and global energy confinement are unaffected by the addition of impurities. This behavior is in marked contrast to devices not operating at a beta limit in which injecting high-Z impurities increases both the radiation and the total energy loss, thereby degrading the global energy confinement. The divertor for a highly radiating plasma need only remove sufficient impurities so that the impurities in the core plasma can be controlled at the design levels.

While an open-divertor configuration does not entrain impurities, it must nevertheless physically isolate the hot core plasma from the collector plate to protect both the plasma from neutral atoms and a possibly uncontrolled source of impurities, as well as protecting the plate from erosion. The minimum separation distance needed to isolate the collector plate from the core plasma is about four neutral-atom ionization mean-free-paths. Typical parameters for the plasma in front of the collector plate ($T_e \simeq T_i \simeq 10\ eV$ and $n_e \simeq 10^{21}\ m^{-3}$) give a mean-free-path of 0.2 mm. The design shown in Figure 11 locates the plate a distance of 15 mm (i.e., 72 mean-free-paths) from the core plasma.

Radial core-plasma, radial edge-plasma, and axial edge-plasma transport calculations give the key parameters for the open-divertor design; edge-plasma density and temperatures midway between divertors at the separatrix are $n \simeq 1.2 \times 10^{20}\ m^{-3}$, $T_e \simeq 100\ eV$, and $T_i \simeq 250\ eV$, with core-plasma and total radiation fractions of 0.82 and 0.93, respectively. The density and temperature near a tungsten-coated divertor plate are $\sim 10^{21}\ m^{-3}$ and ~10 eV, respectively, which should result in negligible erosion; the heat flux normal to the plate is 3.5 MW/m^2.

Plasma density and temperatures near the water-cooled ferritic-steel first wall for the divertor configuration described above are $10^{20}\ m^{-3}$ and 1 eV, respectively, with negligible transported heat flux and a radiation heat flux of about 2 MW/m^2. Wall erosion by plasma particles is negligible; erosion by charge-exchange neutral atoms, however, is 0.44 mm/yr. Lowering the separatrix temperature (e.g., lower heat flux or higher density) has a large effect on the first-wall parameters. A high-Z coating on the first wall should reduce erosion rates to negligible levels.

A geometry calculation in conjunction with a heat-flux constraint determines the shape of a divertor plate (Figure 11), which is located at an 8° angle relative to a field line. The thermal-hydraulics design of the divertor plate was carried out on the inboard side, where the heat flux is highest and the space is minimum. The heat flux normal to the divertor plate accounts for transport along field lines, flux-surface expansion, and radiation. The flux-surface expansion includes the inverse radial dependence of the poloidal field. The radiation heat flux is modeled as consisting of two parts: (a) a radial flux consisting of the core-, edge-, and half of the divertor-radiated power; and (b) the other half of the divertor-radiated power being deposited directly on the collector plate area. The simplest divertor-plate coolant arrangement was selected, which has poloidally symmetric coolant headers located external to the divertor nulling coil (Figure 11). These headers supply and receive coolant water through tungsten-coated copper tubes that form the divertor surface. The total coolant path between headers is less than half a meter, and the tungsten-coated copper tubes form the plasma-facing collector plate surface.[3] The collector-plate thermal-hydraulic parameters are given in Table VI.

4.3 Oscillating-Field Current Drive (OFCD)

An inductive but osciliatory (i.e., a time-averaged constant electromagnetic flux) means of steady-state current drive has been proposed for the RFP.[6] Intrinsic plasma processes related to turbulence and/or resistive instabilities generate voltage and current within the plasma to increase or reduce poloidal flux so that the magnetic-flux linkage or helicity is held constant and the plasma resides in a near-minimum-energy state. This nonlinear coupling between plasma and magnetic fields is strong in RFPs and can be used to rectify current oscillations in external coils into a net steady-state toroidal plasma current, I_ϕ. A power balance imposed at the plasma surface, a definition of the plasma internal magnetic energy, and a positive Faraday's Law ($V_\theta = d\phi/dt$) yield an expression for the toroidal plasma voltage, V_ϕ, in terms of the poloidal voltage, V_θ, and the plasma geometry (r_p, R_T).[3,6] Oscillations of $V_{\phi,\theta}$ in proper phase at frequency less than $\sim 2\pi/\tau_R$ can give a net time-averaged current, $\langle I_\phi\rangle$, with $\langle V_\phi\rangle = 0$ (i.e., no net flux change), where τ_R is the instability relaxation time responsible for poloidal-flux generation. Hence, a non-intrusive means to drive current using primarily the main confining coil system to drive low-frequency, low-amplitude plasma-current oscillations becomes possible.

An assessment of the OFCD engineering efficiency requires the modeling of the circuit elements external to the plasma to account for all power dissipation. Circuit equations are derived [3,6] for poloidal and toroidal current paths and are labeled (θ, ϕ) according to the current direction. The circuit elements simulated are the plasma, first wall, TF coils, OH coils, the blanket, a primary EF-coil set, and a secondary EF-coil set. Calculations with an electrically continuous first wall indicate a need for resistive breaks or gaps in order to assure acceptable levels of power dissipation in surrounding structure for a given plasma current.

The OFCD simulations compute the reactive and dissipative powers as a function of $\delta\phi/\phi_0$. The operating window of $\delta\phi/\phi_o$ is bounded above and below because of a loss of field reversal. The upper bound is the result of oscillations in ϕ becoming to large in amplitude at a shallow reversal ($F = -0.1$). The lower bound is the result of oscillations in I_ϕ becoming too large ($\gtrsim 5\%$) and, hence, the oscillation in the pinch parameter, Θ, being so large as to result in a loss of TF reversal because of the required adherence to a near-minimum-energy state (i.e., the F-Θ curve, Figure 3). The $\delta\phi/\phi_o$ operating window completely disappears below a driver frequency of 25 Hz. A summary of typical OFCD parameters is given in Table VII.

4.4 Maintenance and Testing Geometry

4.4.1 General Layout. A number of key design choices determines the main features of the FTF/RFP test geometry. As seen from Figure 12, the torus is dominated by the PF-coil system, with the TF-coil geometry shown being determined by: (a) field-ripple constraint; (b) TF-coil power consumption ($\lesssim$ 10 MW); and (c) maximized outboard openness for purposes of testing accessibility. The electrically close-coupled PF-coil geometry shown in Figure 9 was chosen to minimize the capital cost, EF-coil power requirement, and mechanical forces under both operational and fault-mode conditions. Lastly, the choice of four divertor sections equally spaced toroidally and the relatively low mass of even the dominant PF-coil set (22.6-tonne inner OH coils, 154.3-tonne EF coils plus outer OH coils) combine with the other constraints to suggest: (a) vertical installation and maintenance of the PF coils and total torus; and (b) horizontal maintenance of individual torus quadrants, divertor subassemblies, and torus-quadrant subassemblies. Details of the blanket and divertor quadrant submodules are shown in Figure 13.

The horizontal (radial, outboard) maintenance and testing scheme requires an outboard access gallery that is sufficiently broad to allow removal and transport of an entire (6.7 tonne) torus quadrant (first wall, blanket/shield, and TF-coils), with divertor assembly (0.8 tonne) and quadrant blanket/shield maintenance/servicing being the more frequent operation. The need to support both normal (+187 MN) and off-normal (-56 MN, no plasma current) EF-coil forces combines with the desire for an outboard toroidal service gallery to require vertical tension/compression bars to support the outboard EF-coil forces; these bars would be of 0.26-m^2 cross section and are located at the outboard side of each of N_{TF} = 28 TF coils. As shown in Figure 9, N_{TF} = 28 EF-coil bulkheads of 70-mm thickness and 1.5-m radial extent assures EF-coil interspace deflections of less than 1 mm; the vertical tension/compression bars would react the EF-coil forces through each of the corresponding bulkheads, shown in Figures 10 and 11.

The choice of both long-term and short-term maintenance schemes and the means to react the EF-coil forces strongly influence the placement of the vacuum boundary. Initial estimates focused on a vacuum chamber into which the entire torus (including all coils) would be placed and in which all short-term (routine) maintenance/servicing operations would occur. The need to react the operational loads and to make and break coolant lines under vacuum reduced the perceived advantages of a large vacuum chamber: eliminating or reducing frequent exposure of the torus to air and eliminating the complexities related to making vacuum connections directly on the torus. Overly large vacuum chambers were also projected if the operating forces had to be reacted under vacuum. Consequently, a close-fitting vacuum geometry was selected in which all vacuum connections are made primarily at the toroidal shell forming the outer boundary (i.e., TF-coil strongback) of each test quadrant and within the TF-coil set. As illustrated in Figure 10 and in more detail in Figure 11, breakable/reweldable seams between divertor sections and torus quadrants would allow removal/replacement of each without disturbing the other. Each replaceable divertor assembly includes divertor plates and associated coolant header/manifold, the captive nulling coil, and associated vacuum-pump shielding. All TF coils, flanking coils, test-cell quadrants, and first-wall tube banks and associated coolant headers and manifolding together form each toroidal quadrant. The cryogenic vacuum pumps attached to each of four divertor sections would be removed for replacement of either divertor or test quadrants. Similarly, diagnostics and fuel-pellet injectors would have to be disconnected and removed.

4.4.2 Test Arrangement. A given toroidal quadrant test space could be comprised of: a) two "clam-shell" hemi-quadrants; b) single testing spaces fabricated between and under each TF coil; or c) an arrangement of insertion test rods or "drawers" tailored to support irradiation of multiple specimens or small single-effects tests, some of which would be instrumented. While the latter approach interferes least with the machine operation, only the outboard portions of the test regions surrounding the plasma would be utilized.

Although not described in great detail or for specific tests arrangements at this point in the conceptual FTF/RFP design, the preliminary test configuration described above suggests a number of broad testing categories. These categories are summarized below.

- **Plasma-Materials-Interaction (PMI) Tests:** These tests would probably be limited to the few divertor sectors, each giving approximately $A_{DIV}/4 = 1\ m^2$ of area where direct plasma-material interactions could be observed. The overall design suggested above allows for relatively frequent, non-interfering horizontal maintenance/service of each divertor section. Since the divertor, as well as the first-wall, functions directly affect overall device operation (e.g., availability), it is likely that tests involving these components will be selected to minimize the impact on the overall machine availability and, therefore, on the bulk radiation tests.
- **Bulk Radiation Tests:** The use of the fusion neutrons can be broadly classified into integrated blanket/shield tests performed within a given test quadrant *versus* smaller, less-integrated tests, a number of which could be conducted in a given quadrant. Each of the quadrants could in principle be used independently to test fully integrated blanket/shield concepts. The available volume between the first wall and TF-coil array amounts to 8.0 m^3 or 0.3 m^3 per TF coil, with the between/under-coil testing volume being divided in proportion of 2:1. Hence, it may be possible to dedicate two or three quadrants to fully integrated blanket shield tests ($I_w = 4.3\ MW/m^2$, $B_\theta = 7\ T$, $B_\phi = 0.5\ T$, $q_s = 1.7\ MW/m^2$), and one or two quadrants dedicated to finer-scale tests ranging from breeder/coolant tests of a few per quadrant to single specimen materials test of one per TF-coil sector (~0.1 m^3, highly instrumented/interactive) to many per TF coil (high-fluence, multiple specimen, passive). The flexibility in individual TF-coil design (i.e., vertical outboard return leg, breakable conductors, etc.) to accommodate a range of active, subsystem tests should be noted.

A useful relationship for evaluation and intercomparison is the dependence of (uncollided) neutron flux, $I_w(MW/m^2)$ on available experimental volume, $V_{exp}(m^3)$. This relationship is shown on Figure 14. When combined with the costing results described in Section 5, Figure 14 provides a means to assess performance in terms of the cost of the FTF/RFP primary product: radiation damage, as measured in terms of total displaced atoms.

5. COST ESTIMATE AND FIGURE-OF-MERIT COMPARISON

5.1 Costing

A cost database was assembled from the extensive work done initially for TFCX[14] and later extended to assess the on-going CIT design.[15] This database was supplemented and/or augmented using the experience derived from ZTH[16] and TIBER[17] assessments. Generally, the cost account break-down originally derived for TFCX was retained and modified for the FTF/RFP study, with estimates of cost being categorized as Hardware Engineering, and Installations cost components. Details of this costing methodology and database can be found in Reference 12. Table VIII gives a breakdown of this cost evaluation, with the basic-device, buildings/facilities/utilities, and project cost being, respectively 265, 46, and 25 M$ projected on the basis of 1988 US dollars. Project costs are based on 8% of basic-device/buildings/facilities/utilities costs, and site costs *per se* are not included. On the basis of these cost estimates and the geometry and performance described in Section 4, the following subsection formulates and evaluates a figure-of-merit (FOM) for purposes of future design optimization and intercomparison.

5.2 Figure-of-merit (FOM) Evaluations

It is illuminating to provide a quantitative comparison of the performance of the FTF/RFP with other sources of high-energy neutrons for the purposes of fusion nuclear materials and technology testing. Such comparisons, if taken alone, are of little positive value, and in fact can be dangerously misleading if used carelessly out of context. Because of the high-volume, high-power-density characteristics, the FTF/RFP is expected to fare well in such a comparison. The figure-of-merits used herein, however, do not reflect the status of the FTF/RFP as a >year-2000 option that must undergo at least one and probably two major device steps in order to resolve critical physics unknowns/uncertainties related to confinement, thin-shell physics, active and passive equilibrium control, divertor/separatrix physics, and current drive.

The FOM model is developed in sufficient detail to allow application to a number of neutron-source approaches, although it is applied here only to the RFP and the dense Z-pinch.[18] Generally, a comparative assessment of the spectrum of neutron-source possibilities as a minimum must include the following:

- energy spectrum and time-dependence (e.g., peak-to-average flux or "compression" ratio, pulse frequency) of the primary neutron source
- degree of extrapolation from present physics database
- degree to which present-day technologies must be extended (e.g., heat fluxes, magnet field strengths, current drive, accelerator efficiencies, etc.)
- costs
  - capital cost, CAP(M$)
  - annual operating cost, AC (M$/yr)
- device performance indices
  - rate of neutron-induced lattice displacements per atom, (dpa/yr)
  - uncollided "effective" 14-MeV neutron current, $I_w(MW/m^2)$
  - test area, A_{FW} (m^2), and/or volume, V_{eep} (m^3)
  - grid power requirement, P_E $(MWe) = P_F/Q_p$
  - fusion power, P_F (MW)
  - CW *versus* pulsed operation
- unit costs (e.g., M$/kg of neutrons, M$/m^3 of test space, M$ per total level of damage, etc.)

While the comparative and/or overall viability of a given approach cannot be assessed accurately by a single figure of merit (FOM), particularly when a variable degree of physics extrapolation must be considered, the use of a broad-based FOM nevertheless provides a useful but a preliminary intercomparison of approaches that as a minimum can point towards directions where a given design might be improved. The FOM chosen here is the product and ratios of important performance indicators. Four important performance indicators are: (a) the annual operating cost, AC(M\$/yr); (b) the experimental volume, $V_{exp}(m^3)$, available for irradiation; (c) the desired final dpa level, DPA, and (d) the time needed to achieve DPA, T(yr). Hence, the following FOM is suggested and adopted:

$$FOM(M\$/dpa \cdot m^3) = \frac{AC * T}{DPA * V_{exp}} \quad , \tag{3}$$

which is the cost of producing a given number of (total) lattice displacements. A possible "normalization" or (maximum) target goal for FOM might be that for the Fast-Flux Test Facility (FFTF), which charges ~0.7 M\$ for an assembly that provides approximately one liter of test space and a dpa rate of ~35 dpa/yr; in this case, FOM $\simeq$ 20 M\$/dpa m^3 for a one-year test. It should be emphasized that the 0.7 M\$ charge covers only operating cost and does not include an amortization of the capital cost of the FFTF itself.

The annual charge, AC(M\$/yr), used to evaluate FOM is comprised of the following: (a) payment on capital cost, $\lambda * CAP$, where λ (1/yr) is the annual pay rate on the total capital cost, CAP(M\$); (b) cost of personnel, $N * COP$, where N is the number of people needed to operate the facility and COP(M\$/person yr) is the cost of a person-year of effort; (c) cost of power, $8.76(10)^{-3} P_E p_f * COE$, where COE (mills/kWeh) is the cost of electrical energy and p_f is the fraction of the year the neutron source operates; and (d) the annual cost of tritium used at a rate $P_F p_f/18.0$ (kg/yr) and at a cost COT(M\$/kg). Hence, the following expression for AC results:

$$AC(M\$/yr) = \lambda * CAP + N * COP + p_f P_F \left[\frac{COE}{114.2 * Q_p} + \frac{COT * (1 - TBR)}{18.0}\right] \quad . \tag{4}$$

A possible credit for tritium breeding within the device (at the expense of V_{exp}, but not including this effect) has been included in Eq. (4) in the form of a tritium breeding ratio, TBR. A physics Q-value, $Q_p \equiv P_F/P_E$, is introduced as a measure of efficiency in producing the fusion power, P_F, from a given electrical input power, P_E. It is noted that the cost of a neutron, CON(M\$/kg), is obtained by dividing this expression for AC by the neutron production rate, $p_f P_F/54.1(kg/yr)$, to give

$$\begin{aligned} CON(M\$/kg) &= 3 * COT * (1 - TBR) + 0.47 * COE/Q_p \\ &+ (5.41/P_F p_f)(\lambda * CAP + N * COP) \quad . \end{aligned} \tag{5}$$

This expression emphasizes the need to operate a high-Q_p system at reasonable fusion power to achieve an acceptable cost of neutrons. For instance, if $CAP \simeq 200 M\$$ and

$P_F \simeq 100MW$, and using the base-case parameters listed in Table IX, the cost of neutrons amounts to 74.4 M\$/kg (i.e., 3-4 times the tritium cost), and broken out as fractions of tritium/electricity/capital/personnel charges amounts to 0.40/0.25/0.27/0.08, respectively. To relate this unit neutron cost to a "product" (e.g., total displacements or the dpa), if an $I_w\tau = 1MW \cdot yr/m^2$ exposure creates 10 dpa, then a kilogram of neutrons passing through 1 m^2 of test area will generate 432 dpa; the cost of this test would be 0.17 M\$/dpa m^2, or for DPA = 100 the cost would be 17 M\$ per m^2 of test area, or \$1700 for each test specimen of area 1 cm^2 that received 100 dpa.

Using the capital cost reported in Table VIII to evaluate that component of the annual charge in Eq. (4), and the more generic base-case values summarized in Table IX, the relative FOM can be expressed with the aid of Figure 14 as a function of experimental volume. Those results are shown in Figure 15 along with a comparison with the DZP neutron source.[18] Both the RFP and the DZP compare favorably with present-day sources because of economy of scale (e.g., the RFP with its large test volume but large fusion power,) or efficiency (e.g., the DZP with its intense, but pulsed, neutron fluxes and low fusion power).

6. SUMMARY AND CONCLUSIONS

The physics required to realize the high-fluence (3.4 $MWyr/m^2$ per year at 80% availability), high-test-volume (10 m^3), moderate-power (124 MW fusion power, 206 MW input power) neutron source will not be available until well after the RFPs presently being constructed have operated for at least 5-6 years. The database for this fusion nuclear-testing facility, therefore, would not be available until after the year 2000. Given that high-current RFPs in relatively compact geometry can be demonstrated within this time frame, however, an efficient and cost-effective neutron source that is capable of providing a majority of small-sample and integrated-blanket test needs for fusion could be available for the period after the year ~2000. The unique confinement characteristics of the poloidal-field-dominated RFP along with the potential to combine or eliminate major plasma support systems favorably project[19] to a compact, high-volume, low-to-moderate-power fusion neutron source that is capable of providing a full spectrum of fusion nuclear materials and technology testing needs (e.g., small scale →fully integrated tests; surface→volumetric tests). This symbiotic combination of heating, confinement, impurity-control, and current-drive functions into a single, generally understressed and relatively low-technology system also projects to a superior commercial reactor product[3] that is not a significant extrapolation from the FTF/RFP system being proposed herein.

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Figures

TABLE I
General Characteristics of RFP

- High-aspect-ratio torus, $A \geq 6$
- Ohmic heating in high-current-density plasma
- High toroidal field inside plasma, low field outside plasma
- High poloidal field at plasma surface, low at coils ($B_\theta \propto 1/r$)
- Low coil fields, copper or aluminum coils possible without large power consumption.
- Coupling of toroidal and poloidal fields (currents) in near-minimum-energy plasma gives possibility of non-intrusive current drive (OFCD)
  - Low driver frequency (30-60 Hz)
  - Low amplitude plasma-current oscillations ($\leq 1.5\%$)
- Moderate-to-high beta
- Beta limit with possibility for highly radiating plasma
- Localized, low-field TF divertors with high-density, high-recycle, low-temperature plasmas at divertor plate
- Confinement scaling as $\tau_E \propto I_\phi^\nu R_p^2 (\nu \simeq 0.8 - 1.5)$
- Simplicity through combined systems (confinement, heating, current drive, impurity control)
- Few or single-piece fusion-core maintenance

TABLE II
Main Parameters of Existing, Planned, and Conceptual RFPs

Device	Status[a]	Laboratory	Major Radius $R_T(m)$	Minor Radius $r_p(m)$	Plasma Current $I_\phi(MA)$	Plasma Current Density $j_\phi(MA/m^2)$	Electron Temp.[b] $T_e(keV)$	Average Density $n(10^{20}/m^3)$	Poloidal Beta[c] β_θ	Transport $\chi_E(m^2/s) \equiv (3/16)r_p^2/\tau_E$
ZT-P	E	LANL/USA	0.45	0.068	0.095	6.5	0.25	1.5	0.3	43.4
TPE-1R(M)15	E	ETL/Japan	0.70	0.135	0.135	2.4	0.65	0.18	0.2	15.5
TPE-1R(M)	E	ETL/Japan	0.50	0.09	0.09	3.5	0.60	0.3	0.2	15.2
ETA-BETA II	E	Padova/Italy	0.65	0.125	0.15	3.0	0.08	1.0	0.1	-
HBTX1B	E	Culham/UK	0.80	0.26	0.22	1.0	0.20	0.7	0.2	25.4
OHTE/RFP	E	GA/USA	1.24	0.20	0.50	4.5	0.4-0.6	0.5-3.0	0.3	30.4
ZT-40M	E	LANL/USA	1.14	0.20	0.44	3.5	0.3-0.5	0.4-0.9	0.2	10.7
MST	E	Univ. of Wisc.	1.5	0.52	0.4-1.0	0.5-1.2	0.1-1	0.3	0.1-0.2	-
RFX	P	Padova/Italy	2.00	0.48	2.0	2.8	0.5-2.0	0.3-2.0	0.1-0.2	3.-4.[d]
ZTH	P	LANL/USA	2.40	0.40	4.0	8.0	0.5-5.0	0.3-5.0	0.1-0.2	2.[d]
FTF/RFP	C	LANL/USA	1.80	0.30	10.4	37.	10.-20.	6.0-9.0	0.1-0.2	0.4[d]
TITAN	C	UCLA-led Study	3.80	0.6	18.2	16.	10.-20.	9.0	0.2	0.3[d]

(a) Existing (E), Planned (P), Conceptual (C)
(b) Centerline temperature.
(c) Based on centerline temperature and $T_e \simeq T_i$
(d) Extrapolation based on a $\tau_E \propto I_\phi$ scaling, leading to $\chi_E \simeq 3.8/I_\phi$.

TABLE III
Synopsis of RFP Experimental Results

- RFP profiles are routinely achieved, are sustained, stable, quiescent, and appear to reside near a minimum-energy state
- Toroidal-field reversal dramatically decreases plasma resistance
- Near-minimum-energy RFP achieved by numerous routes
- RFP sustained for many field diffusion times, indicating internal toroidal-field regeneration ("dynamo")
- Slow ramping of toroidal field
- Control of density pump-out (pulsed discharge cleaning, pellets)
- Temperature, density, and pressure scale favorably with plasma current
- Conditions of (high) constant beta demonstrated
- Confinement time scales favorably with toroidal current
- RFP formation "windows" being understood
  - Burn-through sets upper limit on n and upper limit on j_ϕ/n
  - Fluctuations decrease with increasing temperature ($S = \tau_\eta/\tau_A$)
- RFP profiles/relaxation robust to forced field oscillations, giving potential for unique, low-frequency, nonintrusive current drive

TABLE IV
Device Parameters for the FTF/RFP

Parameter	Value
Plasma major toroidal radius, $R_T(m)$	1.8
Plasma minor radius, $r_p(m)$	0.3
First-wall surface area, $A_{FW}(m^2)$	23.45[(a)]
Blanket/shield thickness, $\Delta b(m)$ [(b)]	0.30
Blanket/shield volume, $V_{BLK}(m^3)$	10.23[(c)]
Pinch parameter, $\Theta = B_\theta(r_p)/\langle B_\phi \rangle$	1.52
Reversal parameter, $F = B_\phi(r_p)/\langle B_\phi \rangle$	-0.12
Poloidal/toroidal field at plasma edge, $B_\theta/B_\phi(T)$	6.8/-0.52
Safety factor, $q(r_p) \simeq \|F\|r_p/\Theta R_T$	~0.013
Average poloidal beta, β_θ	0.10
Average electron/ion temperature, $T_e/T_i(keV)$	9.00/8.53
Average electron density, $n_e(10^{20}/m^3)$	6.87
Effective plasma atomic number, Z_{eff}	1.69
Toroidal plasma current, $I_\phi(MA)$	10.20
Lawson parameter, $n\tau_E(10^{20}s/m^3)$	0.78
Ohmic dissipation in plasma, $P_{\Omega p}(MW)$	24.7
Fraction of alpha-particle energy to plasma, f_α	1.0
Fusion power, $P_F(MW)$	124.2
Power consumption, $P_E(MW)$	206
• Coils	
- OH (forward biased)	85
- (OH (back biased)	193
- EF	54
- TF	9
- DF (nulling)	30
- DF (flanking)	12
• Current drive (0.11 A/w, "wall plug")	76
- coils	39
- first wall/blanket/shield	19
- power supply	18
• Plasma	25
First-wall heat flux, $q_s(MW/m^2)$	1.72
Divertor peak heat flux, q_{DIV} (MW/m^2)	3.5
Fusion neutron first-wall loading, $I_w(MW/m^2)$	4.3
Plasma loop voltage, $V_\phi(V)$	2.42
Streaming parameter, $\xi = v_{De}/v_{the}$	0.0058
Poloidal flux, $L_p I_\phi(Wb)$	64.14
Transport scaling parameter, $\nu(\tau_{ce} \propto I_\phi^\nu r_p^2)$	1.25

(a) Theoretical value; divertors reduce first-wall coverage to 0.87.

(b) Assumed for purposes of subsequent magnetics calculation.

(c) Theoretical value; divertors and first-wall coolant headers reduce blanket coverage to 0.78.

TABLE V
Mean Steady-State Coil Parameters for the FTF/RFP

Parameter	Value				
	OH Coils	EF Coils	TF Coils	DF Coils	
				Nulling	Flanking
Current (MA)	22.6[a]/-34.0[b]	11.5	4.70	-1.54	1.54
Volume (m^3)	14.7	18.3[c]	2.24	0.03	0.10
Mass (tonne)	43.4	133[c]	6.65	0.19	0.74
Joule losses (MW)	84.8[a]/103[b]	54.3	8.80	29.9	11.8
Peak field (T)	10.8[b]	4.12	0.85	1.6	2.3
Current density (MA/m^2)	8.5-24.1[b]	9.0-11.8	9.2	200.	50.
Vertical field index, n	--	$0.62 (0 \leq n \leq 0.65)$	--	--	--
Stray vertical field (mT)	1.20(<1.33)[b]	--	--	--	--
Ripple, $\Delta B_R/B_\theta$(%)	--	--	0.07(< 0.3)	--	--

(*a*) Forward-bias values.
(*b*) Back-bias values.
(*c*) Includes EF trim coil.

TABLE VI
Collector-Plate Thermal-Hydraulics Parameters for the FTF/RFP

Coolant pipe inner diameter, $d_{id}(mm)$	10.0	
Tungsten coating thickness, $\Delta r_W(mm)$	2.0	
Water coolant velocity, $v(m/s)$	2.5	
Water inlet temperature, $T^{in}_{H_2O}(^\circ C)$	100.0	
Degrees below saturation temperature, $\Delta T_{SAT}(^\circ C)$	20.0	
Volumetric heating in metal and water, $Q(MW/m^3)$	67.0	
Pipe material	Cu	SS
Pipe thickness, $\Delta r(mm)$	1.0	0.5
Water outlet temperature, $T^{out}_{H_2O}(^\circ C)$	110.	109.
Water inlet pressure, $P^{in}_{H_2O}(MPa)$	0.25	0.24
Water pressure drop, Δp (MPa)	0.06	0.06
Critical heat flux, $CHF(MW/m^2)$	4.0	4.0
Peak pipe temperature, $T(^\circ C)$	280	355
Normalized total stress, σ/σ_Y	0.05	0.34

TABLE VII
OFCD Parameters for the FTF/RFP

Parameter	Value
Average Plasma Toroidal Current, $I_\phi(MA)$	10.20
Drive Frequency, $f(Hz)$	60
Toroidal-flux swing, $\delta\phi/\phi_o$	0.035
Θ Variation	1.466 – 1.581
F Variation	-0.043 – -0.196
Toroidal/Poloidal Circuit (MW):	
Plasma Poynting power, P_P^*	2,826.76/114.86
Plasma dissipation, P_{Ω_p}	23.28/0.0
First-wall dissipation, P_{FW}	0.00/0.00
Blanket dissipation, P_B	1.56/17.68
OH/TF/EF/Trim-Coil Terminal Reactive Power, $P_i^*(MW)$	3.26/944.3/67.19/817.6
OH/TF/EF/Trim-Coil Dissipation, $P_i(MW)$	0.00/10.64/54.63/26.28
OH/TF/EF/Trim-Coil Real (lost) Terminal Power, $P_i^T(MW)$	0.06/44.15/52.61/37.27
TF-coil dc power, $P_{TF}^{SS}(MW)$	8.82
EF-coil dc power, $P_{EF}^{SS}(MW)$	52.61
Power-supply dissipation, $P_{PS}(MW)$ [a]	18.32
Total dissipation, $P_D(MW)$	152.41[b]
Current-drive power, $P_{CD}(MW)$	90.98
Current-drive "efficiency," $I_\phi/P_{CD}(A/W)$ [c]	0.11

(*a*) Assumes the OFCD power supplies are 99% efficient.

(*b*) Excludes 41.7 MW associated with the four divertors.

(*c*) This efficiency is based on total power consumed in the system. An equivalent estimate for rf current drive in tokamaks is $\sim 0.06A/W$ assuming a conversion efficiency of 0.3.

TABLE VIII
FUSION-DEVICE COST EVALUATION (M$ 1988)

Cost Acct.	Hardware	Engineer.	Install.	Total (k$)
Divertor/Limiter (DL)	2.34	0.47	0.23	3.04
Primary Torus Assembly (PTA)	4.78	1.87	0.94	7.59
Vacuum/Fueling (VF)	5.63	1.27	0.69	7.55
Shielding System (SLD)	1.04	0.10	0.10	1.24
Toroidal-Field Coils (TFC)	0.52	0.05	0.05	0.62
Ohmic-Heating Coils (OHC)	5.14	0.51	0.51	6.16
Equilibrium-Field Coils (EFC)	25.48	5.10	2.55	33.13
Torus Assembly Mechanical Support System (TAM)	0.46	0.05	0.05	0.56
Auxiliary-Heating System				
Current-Drive System (CDS)	1.58	0.32	0.16	2.06
Magnetic-Divertor System (MDS)	0.11	0.01	0.01	0.13
Energy Supply and Distribution Systems (ES&D)	73.24	19.70	17.73	110.67
Diagnostic Devices (DD)	3.98	1.99	0.80	6.77
Maintenance Services (MS)	22.52	5.63	5.63	33.78
Central Data Acquisition, Control, and Processing (CDA)	1.99	0.99	0.40	3.38
Water Cooling and Heat-Rejection System (HRS)	18.74	7.50	7.50	33.73
Cryogenics (C)	0.00	0.00	0.00	0.00
Tritium Fueling System (TFS)	9.01	1.80	1.80	12.62
Building, Facilities, and Utilities (BFU)	46.13	0.00	0.00	46.13
Cleanup, Disposal, and Monitoring Systems (CDM)	1.10	0.33	0.27	1.70
Project (P)	0.00	24.87	0.00	24.87
Total	223.78	72.56	39.42	335.77

TABLE IX

Typical Parameters Used to Evaluate the Figure of Merit (FOM) for a Generic Fusion Neutron Source

Annual cost of money, λ (1/yr)	0.15
Number of people required to operate device, N	50.
Cost of people, $COP(M\$/person\ yr)$	0.16
Availability, p_f	0.8
Cost of electricity, $COE(mills/kWeh)$	40.
Cost of tritium, $COT(M\$/kg)$	10.
Tritium breeding ratio, TBR	0.0
Tritium burn-up fraction, f_B	0.05
dpa goal value, DPA	100.
Irradiation time to achieve DPA, $T(yr)$	1.0
Normalized dpa rate, $DPA/I_wT(dpa\, m^2/MW/yr)$	10.