RESULTS OF SYSTEMS STUDIES FOR THE STARFIRE COMMERCIAL TOKAMAK*

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Summary

Extensive system and tradeoff studies were performed to support the selection process for the major parameters and design features of the STARFIRE commercial reactor. With a thermal power of 3800 MW, a neutron wall load of 3.5 MW/m^2 results in a relatively small-size reactor without imposing excessive requirements on the first-wall cooling capability, maximum toroidal-magnetic field, and frequency of structural material requirements. This moderately high-wall load requires that the first-wall coolant be liquid (water or lithium) and the lifetime of the structural material is >15 MW-yr/m^2. With moderate plasma elongation and beta the required maximum toroidal-field is ~11 T. STARFIRE is operated steady-state with no OH coil. The absence of an OH coil makes it possible to design the reactor with a low-aspect ratio (~2.5) and small major radius. However, higher aspect ratios (~3.5-4) are favored when the plasma current is driven with rf because the power required for the current drive, P_{rf}, is much larger at lower aspect ratio. Since P_{rf} increases at lower plasma temperature, the optimum design for STARFIRE requires operation with plasma temperatures higher than those normally selected for designs with OH-driven current.

1. Introduction

Previous fusion reactor design and systems studies have demonstrated the presence of a wide range of design parameters and a diversity of design concepts. A primary goal of the STARFIRE study[1] is to select, based on present knowledge, the most attractive set of design parameters and concepts that make tokamaks economically competitive and environmentally acceptable. In addition to experience gained from design and systems studies in the United States and worldwide, extensive tradeoff analyses were carried out to guide the selection process for STARFIRE. A primary tool for these tradeoff studies is a comprehensive systems computer program that is capable of predicting the performance characteristics and economics of the entire tokamak power plant. The ANL Systems Code[2] supplemented by the MDAC Code[3] was utilized for this purpose.

The major design parameters that chracterize a tokamak reactor are the reactor power, the neutron wall load, aspect ratio, plasma elongation, major radius, plasma beta, magnetic field, scrape-off region thickness, and blanket/shield thickness. A brief review of the considerations that were factored into selection of these major parameters is presented in this paper.

2. Reactor Power

It has been shown that tokamak reactors exhibit an economy of scale; i.e. larger power reactors have lower cost of energy. However, three considerations important to the utilities limit the desirable power rating of a plant. The first is the difficulty of raising the capital for larger power plants. The second relates to the cost of reserve electric power capacity that the utility must provide to compensate for scheduled and unscheduled outages. The cost of reserve capacity increases with the size of the individual power plant. The third is the maximum capacity of a single turbine generator. Based on recommendations by the Utility Advisory Committee for STARFIRE, the most desirable power rating at present is in the range of 3000-4000 MW for thermal power and ~1250 MW for electrical power. Therefore, the fusion power for STARFIRE was selected as 3200 MW. This corresponds to a nominal thermal power of ~3800 MW, based on a 21-MeV per fusion reaction, and a net electric power of ~1150 MW. The recoverable thermal power will be modified by the addition of rf power for current drive and the loss of low-temperature heat such as that in the limiter system.

3. Neutron Wall Load and Structure Life

A key parameter that has a substantial impact on the physical size of the reactor is the neutron wall load. The neutron wall load, P_{nw}, is related to the fusion power, P_f, as

$$P_f = P_{nw}A_w\left(\frac{17.6}{14.1}\right) = P_pV ,$$

where A_w is the surface area of the first wall, P_p is the average fusion power density in the plasma, and V is the plasma volume. For the same P_f, higher P_{nw} results in a smaller surface area, higher power density, smaller reactor volume, and potentially lower cost. This underlines the motivation for developing designs with higher wall loads. Figure 1 shows the relationship between the major radius and the neutron wall load for P_f = 3200 MW and plasma elongation of 1.6.

There are limitations, however, on both the ability to produce and the ability to use high-wall loads. The upper limits on the use of high-wall loads are dictated primarily by the first-wall cooling capability and the structure lifetime. Constraints such as the maximum operating temperature and thermal stresses place an upper bound on the allowable wall load. For typical structural materials such as ferritic steels in pulsed reactor systems, the neutron wall load should be limited to ~2.5 MW/m^2 for helium coolant. Higher wall loads are possible with water and lithium coolants. In general, the maximum allowable wall load is higher for reactors such as STARFIRE operating in a steady-state mode.

For a given fluence lifetime, the neutron wall load should be limited so that the frequency of structure replacement is not excessive. In order to limit the fractional increase in the cost of energy due to the plant downtime for replacement of the structural material to δ, the structure lifetime must be sufficiently long to satisfy the following inequality:[4]

$$t_w > \frac{t_d}{365\,\delta} ,$$

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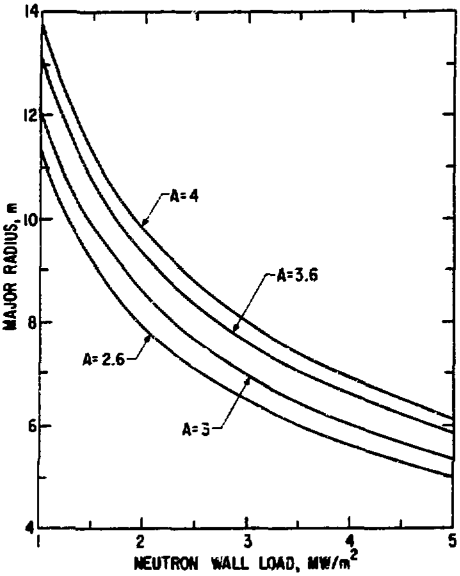

Fig. 1. Major radius as a function of neutron wall load at four values of the aspect ratio. Results based on: fusion power = 3200 MW and plasma elongation = 1.6.

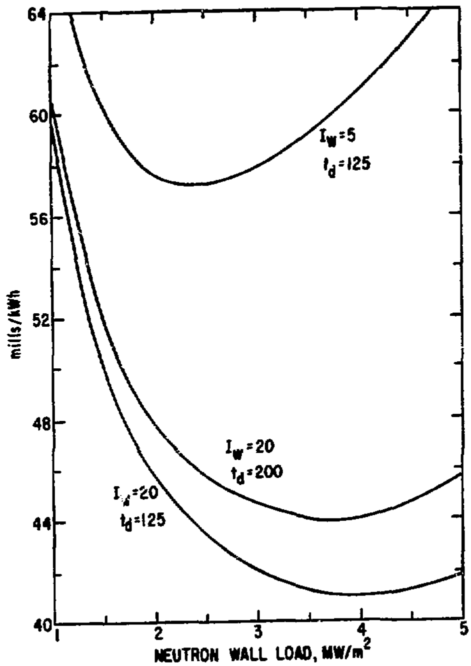

Fig. 2. Cost of energy as a function of neutron wall load. I_w is the integral neutron wall load in MW-yr/m^2 and t_d is the total downtime in days for replacement of the structural material. Results are based on fusion power of 3200 MW, aspect ratio of 3.6, plasma elongation of 1.6, and β_t = 0.067.

where t_w is the structure lifetime in years and t_d is the total cumulative downtime in days for replacement of the structural material. For example, in order to limit the increase in the cost of energy to 10% (i.e. $\delta = 0.1$) when the downtime is 125 days the structure lifetime must be greater than 3.4 yr.

For a given structural material and a fluence lifetime, the loss of energy production resulting from choosing high P_{nw} and short t_w must be weighed against the economic gain realized by designing a small size reactor. Figure 2 shows the cost of energy as a function of the neutron wall load at two values of the integral neutron wall load, I_w, of 5 and 20 MW-yr/m^2 and at two different values for the total cumulative downtime, t_d, for replacement of the structural material. For I_w = 5 MW-yr/m^2 and downtime of 125 days the neutron wall load should be kept in the range of 2-2.5 MW/m^2. For $I_w \sim 20$ MW-yr/m^2 the cost of energy decreases significantly as the neutron wall load is increased from 1 to 2 MW/m^2. A smaller, but significant, saving in the cost of energy (COE) is realizable by increasing P_{nw} from 2 to 3 MW/m^2. A slight change in COE is noticeable in the range $P_{nw} \sim 3$-4 MW/m^2. The reasons for the modest increase in COE as P_{nw} is increased beyond $\sim$4 MW/m^2 will become evident from discussions later in this paper.

It is clear from Fig. 2 that the achievable lifetime of the structural material has a significant impact on the cost of energy. By eliminating short plasma pulses and designing STARFIRE for steady-state operation, it is anticipated that a fluence lifetime of 20 MW-yr/m^2 or greater is obtainable with selected candidate structural materials. There are several important advantages for such long life: (a) the cost of energy is substantially reduced because of less frequent replacement and higher availability factor; and (b) when the frequency of replacement is substantially reduced, the cost of energy becomes less sensitive to moderate variations in the downtime. This is quite important as it permits flexibility in the reactor design not available otherwise for designs driven primarily by the need for achieving very short downtime; (c) a less frequent replacement of the structural material results in a lower inventory of radioactive materials for which storage has to be provided. (It should be noted that even for the candidate structural materials with no long-term activation, adequate radioactive storage is necessary for two to five decades); and (d) the demand on material resources is less with longer life.

For a given fusion power, plasma elongation and aspect ratio, a higher wall load implies a higher plasma density, P_p. This varies as $P_p \sim \beta^2 B^4$. Since the plasma β is limited by stability considerations, a higher P_{nw} is obtainable only by providing a higher magnetic field as shown in Fig. 3. This figure shows the maximum toroidal magnetic field, B_m, required as a function of aspect ratio and neutron wall load. The two different scales for B_m on the left and right of Fig. 3 correspond to two different plasma impurity control schemes as discussed in Sec. 5. For P_{nw} > 2 MW/m^2 there is $\sim$1 T increase in the required B_m for every 1 MW/m^2 increase in P_{nw}. The economically attractive range for P_{nw} of 3-4 MW/m^2 requires a maximum toroidal field in the range 10-12 T which is considered acceptable for the STARFIRE design as discussed shortly.

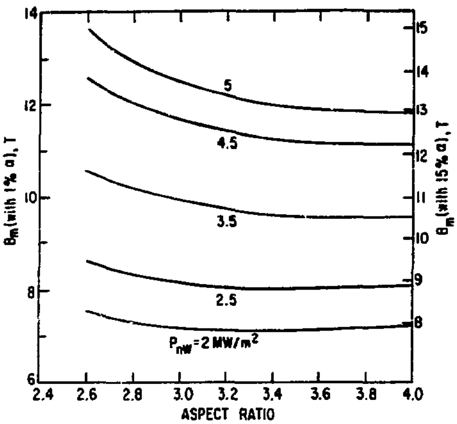

Fig. 3. B_m as a function of A at several values of P_{nw}. The LHS scale is for $n_\alpha/n_{DT} = 0.01$ and the RHS scale is for $n_\alpha/n_{DT} = 0.15$. (P_f = 3200 MW, β_t = 0.24 A, Δ^i_{BS} = 1.2 m, Δ_v = 0.1 m, $\bar{T}$ = 8 keV, κ = 1.6.

In light of the above considerations, a neutron wall load of 3.5 MW/m^2 has been selected for STARFIRE. This moderately high-wall load appears to be a reasonable choice that makes it possible to design a relatively small-size reactor without excessive requirements on first-wall cooling capability, maximum toroidal magnetic field, and frequency of structural material replacement.

4. Plasma Beta

Previous systems studies[5] have indicated significant economic benefits for operating at high-average plasma-toroidal beta, β_t. However, the maximum realizable β_t is limited by plasma stability considerations. Based on recent theoretical analysis, the relationship $\beta_t = 0.24/A$, where A is the aspect ratio, has been assumed for the STARFIRE study.

5. Toroidal Magnetic Field

Figure 3 shows the maximum toroidal field (B_m) required as a function of aspect ratio and neutron wall load for P_f = 3200 MW, κ = 1.6, β_t = 0.24/A, Δ_v = 0.1 m, and Δ^i_{BS} = 1.2 m. The scale on the left side of Fig. 3 shows the required B_m if an efficient plasma impurity control mechanism (e.g. divertor) is provided such that n_α/n_{DT} is ∿0.01. The scale on the right shows the B_m required if less efficient plasma impurity control mechanism (e.g. limiter/vacuum system, see Ref. 1) is utilized such that n_α/n_{DT} is ∿0.15. Figure 4 provides more details about the variation of B_m with the impurity level for the case of P_{nw} = 3.5 MW/m^2, aspect ratio of 3.6 and $T_i \sim T_e$ = 14 keV. The rf power required to drive the plasma current decreases with T_e as discussed shortly. The dependence of B_m on T_e is shown in Fig. 5.

There exists at present considerable experience with NBTi superconductor. However, it has been shown that a magnetic field of ∿10 T is the maximum practical limit for NBTi cooled to 4.2 K at atmospheric pressure. On the other hand, Nb_3Sn is capable of generating higher fields. Although present experience with Nb_3Sn is limited, the progress in the current technology development program indicates that the Nb_3Sn technology will be available in the STARFIRE time frame.

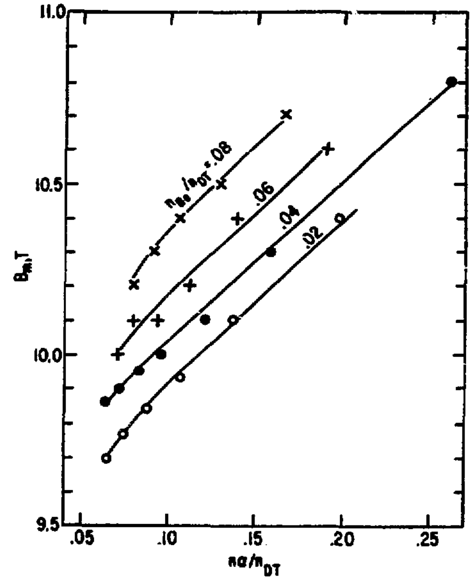

Fig. 4. Variation of the maximum magnetic field with the alpha-particle concentration, n_α/n_{DT}, and other impurity concentration, $n_{Be}/_{DT}$. (P_f = 3200 MW, P_{nw} = 3.5 MW/m^2, κ = 1.6, A = 3.6, β_t = 0.067, $T_i \sim \bar{T}_e$ = 14 keV, Δ^i_{BS} = 1.2 m, Δ_v = 0.1 m, R = 7 m.)

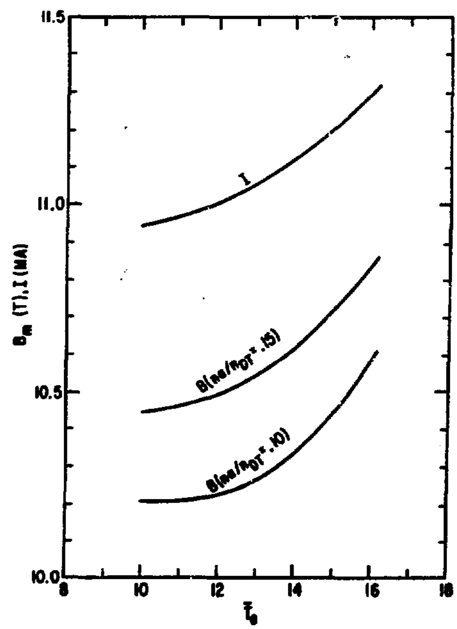

Fig. 5. Variation of the maximum magnetic field and plasma current with the average electron temperature, T_e. (P_f = 3200 MW, κ = 1.6, A = 3.6, R = 7 m, β_t = 0.067, Δ^i_{BS} = 1.2 m, Δ_v = 0.1 m.)

Therefore, the 10-T limit of NbTi was not imposed as a constraint in the STARFIRE study; the required value of the toroidal field was determined from engineering and economic tradeoffs for the overall reactor system.

6. Plasma Elongation

The elongation ($\kappa = b/a$) of a D-shaped plasma has a significant impact on the plasma performance, the reactor design characteristics, and economics. In particular, at higher κ the achievable β_t is higher but the required EF coil system becomes more complex and costly. Based in previous work,[2,6] a value of $\kappa = 1.6$ was selected for STARFIRE. This is believed to be nearly the upper limit on elongation if the important design goal of locating most of the EF coils external to the TF coils is to be achievable.

7. Power Requirements for Current Drive

Since STARFIRE will operate in a steady-state mode, a particularly important aspect of the design is the mechanism for plasma current drive. As discussed in Ref. 1, lower-hybrid rf is the selected option for current drive. Relativistic electron beams are also being examined as a backup option. An important impact of the current drive system on the reactor performance and economics is the electrical power requirements for this system. The dependence of the magnitude of this power for the rf system on key plasma and reactor design parameters is discussed below.

The theory of lower-hybrid-wave-drive currents[7,8] indicates the ratio of rf power density to current density is proportional to the electron density. Thus, rf power is reduced by operating at higher plasma temperatures (lower densities for a fixed beta). However, as the plasma temperature increases above $\sim$10 keV, the fusion power density starts to decrease and the ratio of rf power to fusion power is a minimum in the range 15-18 keV.[8,9] In surveying reactor operation at various temperatures, the desire to minimize rf power calls for considering temperatures above 10 keV, but larger toroidal fields needed to keep the total fusion power at 3200 MW must also be acknowledged at these higher temperatures (see Fig. 5).

In determining the dependence of reactor design on aspect ratio (A), for a fixed total power and wall load and for $\beta_t \propto A^{-1}$, the total current increases going to lower aspect ratio, while the major radius and plasma density decrease. Based on the analytic formula[8]

$$P_{rf} = C R \bar{n}_e I ,$$

where R is the major radius, $\bar{n}_e$ is the average electron density and I is the plasma current, the rf power required to drive reactors for a series of equilibria in the range $3 \leq A \leq 4$ was computed. The coefficient C is a function of the plasma profiles, spectral width, and degree of current penetration; for typical reactor parameters, and for both centrally peaked and surface current density profiles, the rf power increases by 30-50% if the aspect ratio is reduced from 4.0 to 3.0. Hence, large values of A are preferred from the rf power point of view.

8. Major Radius and Aspect Ratio

With the fusion power and neutron wall load selected, the surface area of the plasma is defined. For a given κ, the aspect ratio (A) or the major radius (R) should be selected in order to fully define the plasma geometry. As shown previously in Fig. 1, at $P_{nw} = 3.5$ MW/m^2, the major radius increases from 6 m to $\sim$7.3 m if A is increased from 2.6 to 4.0.

The size of the reactor building, length of piping, etc., are strongly affected by the size of the reactor, in particular, by the value of R + a, where a is the plasma minor radius. Notice that the variation of R + a as the aspect ratio is changed is less than the variation in R alone. Despite the reduction in the capital cost of several items sensitive to size when R is smaller, economic optimization does not necessarily favor the selection of minimum R. This is true for both pulsed and steady-state reactors although for different reasons.

For pulsed reactors, the optimum size is significantly affected by the central core radius, r_v. For a given magnetic field for the OH coil, decreasing r_v reduces the available volt-seconds and shortens the burn time resulting in a lower reactor electrical output.

For steady-state reactors with no OH coil the problem of the central core radius disappears. In this case, the plasma current has to be driven by external means (e.g. rf or REB). If the electrical power requirement for the current driver were negligibly small, then the smallest aspect ratio should be chosen so that the major radius is minimum, provided of course that there is adequate space on the inner side of the torus to accommodate the TF coils and support cylinder. This is illustrated by the case $P_{rf} = 10$ MW in Fig. 6 which displays the cost of energy as a function of the aspect ratio.

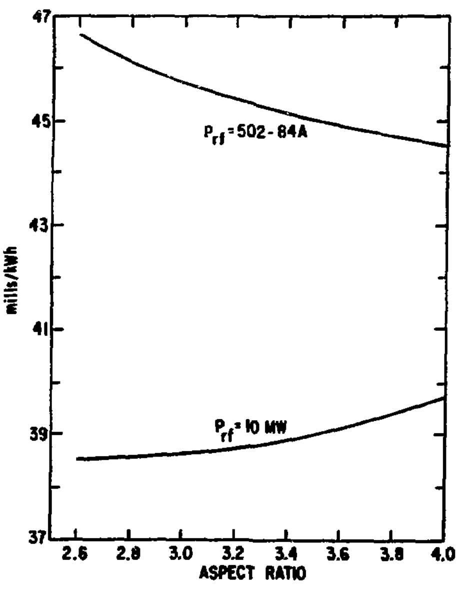

Fig. 6. Cost of energy as a function of aspect ratio for a steady-state reactor in two cases for the electrical power, P_{rf}, required for the current-drive system: $P_{rf} = 10$ and $P_{rf} = 502-84$ A where A is the aspect ratio and P_{rf} is in MWe.

As discussed earlier in Sec. 7, the electrical power requirement, P_{rf}, for the rf current drive system appears to be relatively large and increases with lower aspect ratio. Figure 6 shows the dependence of the cost of energy on aspect ratio for the case where $P_{rf} = 502-84$ A, with P_{rf} in MW electric. In this case the cost of energy decreases as A increases up to A $\sim$ 4.

It is anticipated, however, that as a result of further work the rf power requirements will be reduced significantly. This would reduce the optimum aspect ratio somewhat below A = 4. Therefore, A = 3.6 was adopted for STARFIRE. This results in a major radius R = 7 m. The maximum magnetic field required is nearly minimum at A = 3.6 as evident from results in Fig. 3.

A major incentive for considering relativistic electron beams (REB) as a backup option for plasma current drive is the much lower electrical power requirements compared to those needed for the rf system. Present estimates indicate that REB electrical power requirements are in the range 10-20 MW compared to ~150-200 MW for the rf system. Assuming that the capital cost is roughly the same for both systems, the cost per unit power is ~8% lower with REB compared to that with the rf system. However, the present state-of-the art for REB makes it difficult to quantitatively evaluate the technical and operational problems that might arise as the concept is investigated in more detail.

9. Inner Blanket/Shield Thickness

The thickness, Δ^i_{BS}, of the blanket and shield on the inner side of the torus, or more precisely the distance in midplane from the plasma side of the first wall to the location of the maximum toroidal magnetic field, has a substantial impact on the reactor size, the required strength of the magnetic field and reactor economics. A comprehensive investigation of the optimum value for Δ^i_{BS} has been carried out previously.[10] The details of this previous work will not be reported here, but the results are briefly stated.

For a given P_f, P_{nw}, β_t, and major radius, the advantages of a smaller Δ^i_{BS} are (1) lower maximum magnetic field; and (2) larger central core (OH) radius, r_v. In pulsed tokamaks, the impact of the value of Δ^i_{BS} on r_v is large and becomes critical for low P_f-high P_{nw} designs. In a steady-state tokamak with no OH solenoid, the primary incentive for reducing Δ^i_{BS} is to reduce the maximum magnetic field required.

On the other hand, there are several penalities for making Δ^i_{BS} too small. The increase in the radiation field when Δ^i_{BS} is decreased results in (a) increase in the resistivity of the stabilizer material and a need for increasing the amount of the stabilizer to satisfy the cryogenic stability requirements; (b) a decrease in the critical current density of the superconductor necessitating the use of more superconductor; and (c) an increase in the heat-generation rate in the TF coil since the increase in the power requirements for the TF coil cryogenic system can be so large that the reactor net electrical power output is seriously reduced.

For the reference design of STARFIRE, a value of Δ^i_{BS} = 1.2 m was selected based on careful considerations of the above tradeoffs. This value includes ~25% void to account for the vacuum gap in the TF coil and the use of helium coolant in a portion of the inner blanket, if necessary. The shield consists of a combination of tungsten, boron carbide, lead, and a structural material.

10. Outer Leg of the TF Coils

For a given major radius (R), plasma inner radius (a) and inner blanket/shield thickness (Δ^i_{BS}) the position of the inner leg of the TF coil is defined. In order to fully define the D-shape of the TF coil, the size of the vertical or horizontal bore must be selected. We will discuss this choice in terms of R_2, the major radius of the midpoint of the outer leg of the TF coil. R_2 is the sum of the major radius, first-wall minor radius, outer blanket/shield thickness (Δ^o_{BS}), clearance (Δ_c) in the midplane from the outer edge of the shield to the TF coil and half of the TF coil thickness. The choice of R_2 has a significant impact on many of the reactor characteristics and cost as discussed below.

With the reactor parameters defined in the previous sections (R, a, Δ_v, B_m, Δ^i_{BS}) the only remaining parameters that affect R_2 are Δ^o_{BS} and Δ_c. The necessary value of Δ^o_{BS} varies with blanket and shield material and coolant choices. As a goal, the materials in the shield should be chosen to have inherently low long-lived radioactivity even if they are less efficient in radiation attenuation. For typical material choices, the required Δ^o_{BS} is ~1.3 m and ~1.8 m for blanket with water (or lithium) and helium coolants, respectively. The clearance from the outer edge of the shield to the TF coils is required for several engineering reasons; the most dominant of which is to accommodate the coolant manifolds. The required R_2 is ~12 m for water or lithium coolants compared to ~13 m necessary for helium coolants.

The economics analysis shows a substantial penalty for increasing R_2. The cost of energy increases by ~3% for each additional meter increase in the value of R_2 beyond 12 m. The cause of this penalty is that the value of R_2 directly influences the size and weight of the TF coils and their support structure, the size of the reactor building, the length of the piping for the heat transport system, and the size of the externally located EF coils. For example, by increasing R_2 from 12 to 14 m, the ampere-turns and stored energy in the EF coils nearly double.

It should be noted that in order to keep the field ripple at the plasma to an acceptable level, the value of R_2 should be greater than a certain minimum. The minimum value for R_2 is larger for a smaller number of TF coils. A relatively small number, 12, of TF coils has been chosen for STARFIRE to enhance reactor maintainability. This makes extending the TF coils as a means of satisfying the field ripple criteria economically unattractive. Therefore, the alternative of an imposed field-ripple correction system (e.g. saddle coils or iron blocks) will be provided if necessary.

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