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ARCHITECTURE OF THE ETR SYSTEMS CODE*

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Abstract

TETRA, a tokamak systems code capable of modeling experimental test reactors (ETRs), was developed in a joint effort by participants of the fusion community. The first version of this code was constructed to model devices similar to the Tokamak Ignition/Burn Engineering Reactor (TIBER) in configuration and design. A major feature of this code is its ability to perform optimisation studies. Future work will include broadening the scope of the code, particularly in the area of materials selection, to more accurately simulate tokamak configurations such as the Next European Torus (NET) and the Fusion Engineering Reactor (FER).

Introduction

TETRA was developed through a cooperative effort of the fusion community† for application to TIBER/ETR system studies. This code is similar in concept to that described in Ref. 1 in that (1) the code is modular (i.e., each module describes a tokamak system or component) and (2) the modules are executed and controlled by a driver routine. In addition, a feature has been added to allow for optimization. This optimizer is similar in concept to that applied to an all-copper toroidal field (TF) and poloidal field (PF) coil version of the tokamak systems code used in the recent Compact Ignition Tokamak (CIT) studies described in Ref. 2.

Code Architecture

TETRA consists of a series of modules, each describing a system or component of a tokamak reactor, controlled by a driver or optimizer routine. The modules are shown in the Tokamak Reactor Systems flow diagram (Fig. 1). The hierarchy of module execution is designed such that the modules downstream depend on the upstream modules to supply input (e.g., the energy storage module requires input from the magnet modules and the plasma heating modules).

The overall ETR systems code schematic is shown in Fig. 2. The systems code is controlled by an optimizer or nonlinear equation solver. All modules from Fig. 1, which describe the tokamak reactor, serve as a function generator and are represented as a single block (Tokamak Reactor Systems) in Fig. 2. This block returns to the driver computed values of physics and engineering quantities, which are used in evaluating specified constraints. If constraints are not satisfied within tolerance, the driver changes specified variables for iteration and recalls the Tokamak Reactor Systems modules until convergence criteria are satisfied. When the optimizer option is used, the variables are iterated within prescribed bounds until the constraints are satisfied and a maximum or minimum selected figure of merit (such as cost or major radius) is achieved.

Constraints that link the tokamak reactor modules include

1. beta limits,
2. plasma density limits,

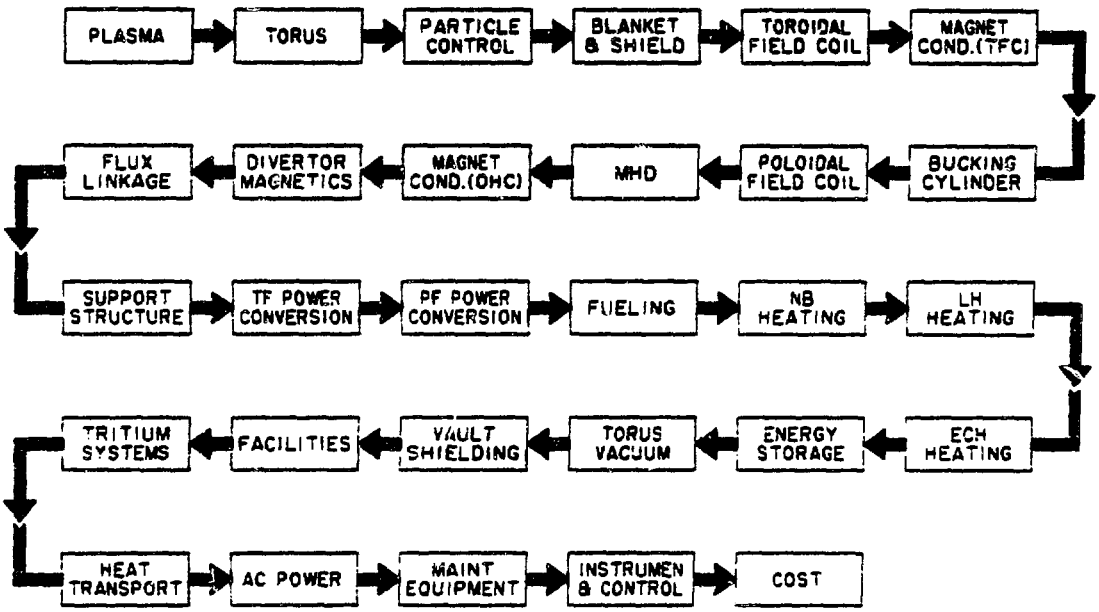

Fig. 1. Tokamak reactor systems flow diagram.

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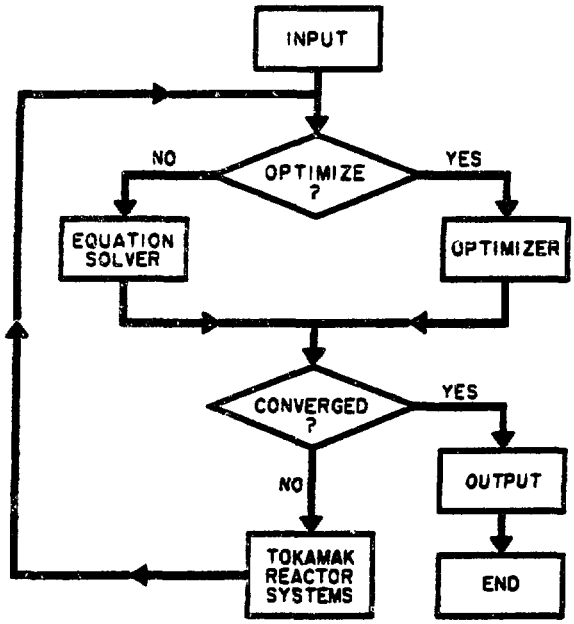

Fig. 2. ETR tokamak systems code schematic.

3. ignition margin,
4. neutron wall loading,
5. magnetic flux capability of the PF coil set,
6. coil stresses,
7. superconducting coil current limits,
8. shielding requirements for superconducting coils,
9. shielding requirements for biological considerations,
10. plasma power balance,
11. maximum field at the TF coil conductor, and
12. Q (the ratio of fusion power to current drive power).

Code Formulation and Usage

The TETRA code is organized in a standard "equation solver" format. The equation solver consists of a numerical software package, which iterates prescribed variables in order to satisfy a set of prescribed equations (or constraints). In this case, the engineering and physics modules, shown in Fig. 1, serve as the "function evaluator" of the equation solver, which provides information used in the solution of the equations.

Two equation-solving packages are included in TETRA: *VMCON*[3] and *HYBRID*.[4] *VMCON* is an "optimizer" package, or more specifically, a generalized nonlinear programming subroutine. This option requires (1) iterating more variables than constraints being considered, (2) incorporating upper and lower bounds on all variables being iterated, and (3) specifying a prescribed figure of merit to be maximized or minimized. As an example, this option is useful for finding the minimum-cost machine that satisfies physics and engineering constraints when parameters such as plasma size, coil sizes, magnetic field strengths, etc., are allowed to vary.

HYBRID is a nonlinear equation solver with no "optimization." It iterates the same number of variables as constraints being satisfied and does not include any bounds on the variables. This option is useful for performing benchmark comparisons when parameters such as sizes, fields, plasma conditions, etc., must remain fixed (at the values of the comparison case) and one wishes to see how the calculated stress compares with the allowable stress, how the beta compares with the beta limit, etc.

In using the optimizer (VMCON), initial decisions must be made concerning which constraints should be incorporated for the task being considered and which quantities should be varied to satisfy these constraints. In addition, values for lower and upper bounds must be supplied. Some of the constraints are formulated as "inequality" constraints; these are arranged as

$$\text{Calculated quantity} = f\text{-value} \times \text{allowable value} .$$

Examples of "calculated quantities" are plasma beta and coil stress, and the corresponding "allowable values" are the beta limit and stress limit. (Note: Sometimes the allowable value is computed, as with the case of beta limits, and sometimes it is input, as with the case of stress limits.) The f-values can be used as variables, and when appropriately bounded, the constraints become inequalities. Generally, it is desirable for the calculated quantity to be less than the allowable value, in which case the f-value should have an upper bound of 1. For example, if the beta limit constraint is being used and if the variable f_β is used with an upper bound of 1, then a solution that satisfies the beta limit must be found. A possible exception is the wall-load constraint, where it may sometimes be desirable for the calculated quantity (wall load) to be greater than the allowable or reference value (wall-load limit), in which case the f-value should have a lower bound of 1. It is possible to use these constraints as equality constraints by keeping the f-value fixed.

An important consideration when using VMCON is the choice of a figure of merit (or objective function) to be maximized or minimized. In TETRA this is determined by the value of an input switch. When the absolute value of the input switch is positive, the figure of merit is minimized; when zero or negative, the figure of merit is maximized (i.e., if the input switch value = 1, the minimum major radius solution is found; if the value of the input switch = 0, the maximum ignition margin is always found). It is possible to incorporate different figures of merit in the code.

When using the equation solver HYBRID (non-optimizer), the number of constraints to be considered must be equal to the number of variables used. Also, when not optimizing, the bounds on the variables are not used, which precludes the use of inequality constraints as described above. As previously noted, the use of the non-optimizing equation solver is useful for benchmarking. Constraints formulated as inequalities for VMCON can still be used by letting the appropriate f-value be a variable, even though it is not bounded. This provides information about how calculated values compare to limiting values, without having to change the characteristics of the device being benchmarked to find a solution.

Several constraints should always be used when analyzing tokamaks: (1) the poloidal beta equation, (2) the plasma beta equation, (3) the radial build equation, (4) the relationship between the magnetic field at the TF coil and the field on axis, and (5) a plasma power balance relation—either the global balance or the separate ion and electron balance equations. Some additional constraints available in the code are listed in the "Code Architecture" section of this paper.

Example Module (Physics)

Space does not permit describing each module of the TETRA code. However, a description of the physics modules is included in this section as a typical example.

The physics module uses global, profile-averaged, steady-state plasma physics similar to that used in the MUMAK code[5] and in Ref. 6. Primary input includes plasma size and shape, toroidal field on axis, edge safety factor, plasma density and temperature (and profiles), and Z_{eff}. We note that the electron and ion temperatures in this code are volume-averaged temperatures, as opposed to the density-weighted volume-averaged temperatures used in Refs. 5 and 6. Key output includes the plasma current, plasma composition, power balance terms, volt-second requirements, limits on density and beta, and some other quantities used in other modules. The calculations are summarized here; detailed descriptions of the methods can usually be found in the references.

Plasma Current

The plasma current scaling can be chosen from several options. In all cases, the edge safety factor, plasma major and minor radius, and plasma shape are input. For limiter plasmas, one can choose from scalings derived for a spherical torus,[7] CIT physics panel recommendations, a fit to MHD equilibria,[8] or the GA scaling.[9] All use the safety factor (q) as the q_ψ at the outermost closed flux surface. For divertors, the scaling described in Ref. 10 can be used; it uses q as the mean safety factor at the separatrix. The cylindrical safety factor (q_c) is also calculated using the method described in Ref. 5.

Plasma Composition

The plasma ion population consists of fuel, alpha ash, impurity species, and hot neutral beam ions. The fractional makeup is determined by inputting the thermal alpha ash density fraction (relative to the electron density), the neutral beam fast ion density fraction, the charge of the impurity species, and the effective charge of the plasma. The fuel ion density, impurity ion density, and other mass and charge-averaged quantities are calculated. The plasma density limit can be chosen from either the Murakami or Greenwald limit. The beta limit is the Troyon limit [see Ref. 6, Eq. (27)], with the coefficient being an input parameter, or a simplified beta scaling suggested by the CIT physics panel.

Fusion Power

The fusion power is found by integrating over the plasma volume as described in Ref. 5. Profile effects can be explicitly accounted for in the fusion power because the integration is performed for each iteration. The fusion cross sections are taken from Ref. 11 for ion temperature less than 20 keV and from Ref. 12 for ion temperature greater than 20 keV. The fraction of the alpha power going to the electrons is taken from Eq. (3.12) in Ref. 5. A simplified fit to the fast alpha beta fraction, which does not account for profile effects, is presently used.[13] The average neutron wall load is found by dividing the neutron power by the first-wall surface area.

Other Power Balance Terms

The bremsstrahlung radiation power calculation is an analytic integration over the plasma volume (including profile effects) and is equivalent to that described in Ref. 5. No synchrotron radiation power is presently accounted for. The ohmic power term uses the average plasma temperature to calculate the plasma resistivity and includes a neoclassical correction to the resistivity.[14] The ohmic heating (OH) term is negligible for all steady-state cases that the code is presently constructed to examine. The equilibration power between the bulk ions and electrons uses the same formulation as Ref. 5 except that volume-averaged temperatures are used in the expression, and the equation is analytically volume averaged to account for profile effects.

Transport power losses for ions and electrons are modeled as $1.5(n \times T)/\tau$, where n is the density, T is the density-weighted average temperature, and τ is an energy confinement time. For ions, the confinement is neoclassical, as formulated in Ref. 6. The losses can be enhanced by increasing an input factor. Also, the ion confinement time can be forced to be equal to the electron confinement by setting an input switch. Electron confinement can be chosen from a variety of scalings: neo-Alcator (Ref. 15); Mirnov [Eq. (A.3) in Ref. 5]; Kaye-Goldston L-mode [Eq. (A.1) in Ref. 5]; ASDEX H-mode [Eq. (A.11) in Ref. 5]; and IAEA-ASDEX H-mode (Ref. 15). The Kaye-Goldston confinement time is multiplied by the input factor. An option also exists for combining the Kaye-Goldston confinement time with the neoclassical confinement time scalings via inverse quadrature.

Volt-Second Requirements

Plasma volt-second requirements are separated into three parts: inductive, startup losses, and flattop-burn losses. The inductive requirement is further separated into internal and external components, as described in Ref. 16, except that the normalized plasma internal inductance is now an input. The external inductance comes from Ref. 17. The startup resistive volt-second requirement is taken to be a fraction of the internal inductive volt-second requirement. The default value for the fraction is 0.5, as derived through comparisons with WHIST calculations. For the volt-second requirements during burn ($VS_{burn} = C_{sawth} \times$ loop voltage $\times$ burn time), the loop voltage is calculated using the plasma resistance, which is based on average plasma parameters and a neoclassical correction factor. The coefficient C_{sawth} is input and can be used to enhance the burn requirement to mimic the effects of sawtooth activity. Comparison with 1.5-D WHIST calculations showed that C_{sawth} should be 3 for CIT regime studies.[16,18]

Auxiliary Calculations

Some auxiliary calculations done by the physics module are for particle loss rates from the plasma. The convective particle loss rate is found using a particle confinement time $\tau_p = 5$ times the effective energy confinement time (where the effective energy confinement time is the average for the ions and electrons). An input recycling ratio (RECYCLE) is also included in the particle loss rate where particle loss rate = (1 − RECYCLE) × ion density × volume/τ_p. The fractional burnup is defined as the fusion burn rate divided by the convective particle loss rate. A maximum fractional burnup of 0.5 is set, and when this is exceeded, τ_p is lowered so that the specified maximum fractional burnup is met. The plasma volume (VOL) is found with the crescent-shaped model used in Ref. 5. The average poloidal field is found using Ampère's law and using the poloidal path length around the plasma perimeter as described in Ref. 5.

Example Code Execution

To demonstrate the use of this code, the following brief study was accomplished. First, the TIBER parameters were simulated (benchmarked) by using the nonlinear equation solver branch of the code (HYBRID). The fractions of the inequality limits for stresses, beta, shielding thickness, and conductor currents for the coils were found and held as upper limits for the subsequent run. The stability margins for the TF and OH conductors were also found but were not held as upper limits because these margins were less than the allowable values. The fractions of the inequality limits for the TIBER base case are shown in column 1 of Table 1. The optimizer branch of the code (VMCON) was then invoked to find the minimum major radius machine holding the TIBER inequality upper limits and achieving the reference TIBER values of neutron wall loading and Q.

Results from example executions of the systems code are shown in Tables 1–4. The cases presented are the benchmark TIBER case and a case optimized for minimum major radius holding the neutron wall loading, Q value, and major limits of the reference TIBER case. Column 2 of Table 1 shows that all inequality limits were pushed to the limiting value of 1 with the exception of plasma density, which was constrained by the more stringent condition of the limiting value of beta. Table 2 shows that a reduction in major radius from 3.0 to 2.72 m and in capital cost from $1.23 billion to $1.08 billion is achieved. Note that one of the parameters changed by the optimizer is plasma current, from the 10-MA TIBER value to 8 MA for the minimum major radius point. The reduced current has a substantial impact on current drive injected power and its associated capital cost. Table 3 shows that the elements of the radial build that change between these two cases are plasma size and OH solenoid size, both of which are smaller for the minimum major radius point. Table 4 shows that the capital

Table 1. Fractions of inequality limits

	TIBER	Minimum major radius
Plasma density	0.41	0.37
Plasma beta[a]	1	1
Neutron wall load[a]	1	1
Q[a]	1	1
Volt-seconds[a]	1	1
TF stress[a]	1	1
OH stress[a]	1	1
TF conductor current[a]	1	1
OH conductor current[a]	1	1
TF stability	0.61	1
OH stability	0.75	1
Shield (inner and outer)[a]	1	1
TF port size[a]	1	1

[a]Limits set at TIBER values.

Table 2. Parameters

	TIBER	Minimum major radius
Major radius, m	3.0	2.72
Neutron wall loading, MW/m^2	1.2	1.2
Q	5.0	5.0
Aspect ratio	3.6	4.0
Field on axis, T	5.55	6.4
Plasma current, MA	10.0	8.0
Fusion power, MW	307	232
Plasma beta, %	7	6
Direct cost, billions of dollars	1.23	1.08

Table 3. Elements of the radial build (meters)

	TIBER	Minimum major radius
Plasma minor radius	0.83	0.67
Inboard shield thickness	0.48	0.48
TF coil thickness	0.49	0.49
OH soleonid thickness	0.52	0.50
OH solenoid radius	0.55	0.45
Major radius	3.00	2.72

Table 4. Direct capital cost by system (millions of dollars)

	TIBER	Minimum major radius
Buildings	135.6	132.0
Torus	195.0	156.6
Magnets	175.1	165.4
Power injection system	294.1	221.8
Vacuum system	27.1	20.2
Power conditioning	56.8	54.5
Heat transport system	55.4	46.8
Fuel handling systems	68.8	67.3
Instrumention and controls	100.0	100.0
Maintenance equipment	60.0	60.0
Electrical plant	20.3	17.8
Miscellaneous equipment	25.0	25.0
Heat rejection	15.9	12.9
Total	1229.1	1080.3

cost reduction for the optimized minimum major radius case was primarily the result of reduced costs for the torus, magnets, power injection system, vacuum system, and heat transport system.

Summary

A tokamak systems code, TETRA, has been developed to model ETR devices similar in configuration to TIBER. The architecture of the code consists of modules, each describing a tokamak system or component, controlled by an optimizer driver. A description of the physics module is included as an example of the type of modules that make up this code. The modules were provided by scientists and engineers from throughout the fusion community and were integrated at the FEDC. The code has been checked out on the Cray computers at the National Magnetic Fusion Energy Computing Center and has satisfactorily simulated the TIBER II design. The code will be improved to more accurately model ETRs such as NET and FER by including a greater choice of blanket, shield, and coil materials and configurations. This code is expected to be a powerful tool for performing sensitivity studies that relate physics parameters, engineering parameters, technology limits, and cost for the International Thermonuclear Experimental Reactor design effort.

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