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Structural Design of High Field Tokamaks

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Abstract

Interest in high field non-superconducting tokamaks was strong after the US left the ITER project at the end of 1999. This class of tokamaks, including Alcator C-Mod, IGNITOR, and FIRE, has many challenging and interesting structural characteristics. Structural behavior of these tokamaks is discussed along with the special structural features employed to sustain the huge loads resulting from their high fields. This paper was originally intended to be a part of a special journal on high field tokamaks that was canceled with the renewed interest in ITER.

Introduction

The size of proposed fusion projects seems to follow a cycle, and small, low cost high performance machines were viewed with interest in 2001 and 2002. The pendulum is now swinging back to an ITER[18] scale project. "Low cost" translates to small, high field, copper, pulsed machines. The concepts and specific machine designs to satisfy these requirements have been available for a while. There is a long list of machine studies, not built, but which provide a wealth of design options. There are operating machines which fall into the category of high field tokamaks -MIT's C-Mod, and the Frascati tokamak, FTU are examples. IGNITOR ands FIRE are strong contenders now undergoing design. CIT, and BPX were high field machines that went through conceptual design but were not built.

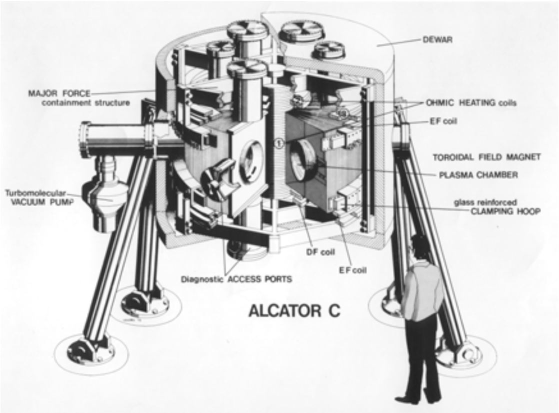


Figure 1 Alcator -C, An early example of a high field tokamak, and the precursor of It's "modification", C-Mod

The goal is to achieve the most performance as measured against a physics mission, given cost and thus size constraints. Because of structural and field limitations of present day superconductors, high field tokamaks are restricted to normal conductor materials. This is likely to change with improvements in high temperature superconductors, some of which can operate in 30 Tesla and higher.

To realize the advantages of high field, material science, structural mechanics, and mechanical design must be utilized effectively to optimize the magnetic performance of the machine. Each of the machines discussed here use slightly different interplay of these disciplines to achieve the different goals of the machines, and they have many common problems and solutions.

Material Science approaches attempt to use high strength conductors such as metal matrix composites, CuNb and Cu-Ag; precipitation hardened alloys such as Berylium Copper and alumina dispersion strengthened copper such as Glidcop. All normal conductors have the unfortunate characteristic that the higher strength materials typically have higher resistivities. Figure 2 plots the conductivity of some of these materials against their strength. The more advanced materials are not yet available in practical sizes for tokamaks.

Mechanical design solutions for these systems must limit the loads carried by the conductor, while leaving as much space for conductor cross section as possible. The intention is to allow the use of low strength, higher conductivity materials. Use of preloads and large external structures can off load some conductor stress, but since the normal conductors experience Joule heat, the thermal differential expansion of the conductor and external structures limits the constraint the external structures can provide.

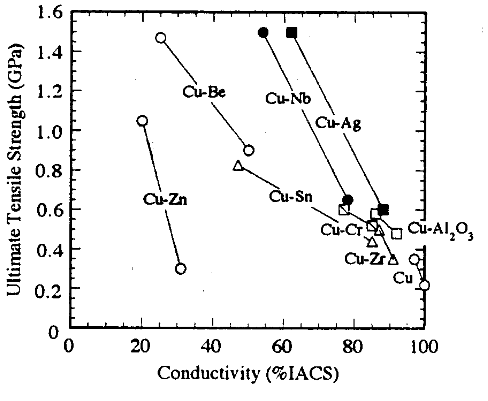


Figure 2 Strength and Electrical Conductivity of various copper based conductors. from [20]

Active preload systems have been proposed as a solution. An active radial magnetic press is used on IGNITOR. A vertical press was used on an earlier design. These back-off the preload while the coil is warmed to protect the coil from the combined effects of preload and thermal compressive strains. Without reductions in preload, stress sometimes is worst when the machine is turned off. Active cooling during the pulse would seem to be a solution, but active cooling trades conductor cross section for coolant media. There is a family of machines that can accept the lower conductor strength and benefit from active cooling. For the machines that will be discussed here, the primary cooling method is inertial.

For a typical compact low aspect ratio tokamak, the central column must be considered sacred territory for carrying current, and providing it's own support.

Introduction to forces

Free standing solenoids, and toroidal field coils which must resist direct tension can be thought of as membranes which contain internal magnetic pressure. In the case of the toroidal field coil, the internal magnetic field varies as the inverse of radius, and this produces a net centering load on individual TF coils. The non-uniform magnetic pressure changes the ideal pressure boundary shape from a circular form to a "D". Using a constant tension in the coil and mathematically matching the local radius of curvature of the coil to support the local magnetic pressure, added to an assumption that the inner leg force can be supported radially by other structure, produces the "D" form. The equations governing the behavior of the constant tension "D" may be found in [1], and the U. of Wisconsin paper by Moses and Young, [26]

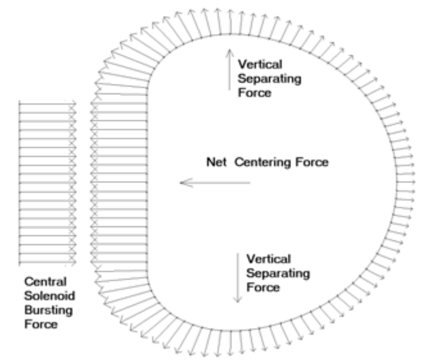
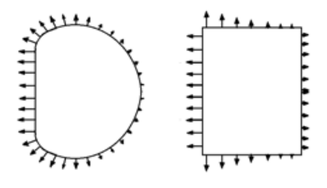


Figure 3



Difference between Constant Tension "D" and the Picture frame.

The primary load in the TF coil of tokamaks with a large bore, and a relatively low field is poloidally directed hoop tension, and "D" forms are typically used. Compact devices limit the geometric freedom available to support the TF coil, and for these, non-ideal shapes are typical. If you imagine a TF coil shape as a rectangle, you can logically separate the force components into inner leg centering, upper and lower leg separating and outer leg bursting forces. This rectangular "picture frame" coil produces a very different understanding of the structural support of the TF coil, and provides some insight into how loads might be supported in high field compact devices. Alcator C-Mod is the best example of this coil shape and allows a clear illustration of the mechanisms that are used to support the load components. C-Mod's cross section is shown in figure 4. In this reactor, the coil is structurally cut to force the individual load components to be supported by readily identifiable load paths.

Structural Concepts

All tokamaks adopt some philosophy for supporting the coil loads. One set of options relates to how the radial loading from the central solenoid (CS) , and the toroidal field (TF) coil are supported:

Free standing CS, Wedged (or "vaulted") free standing TF coils , The CS is loaded in hoop tension from it's radially outward loading, and the TF inner leg array is loaded in toroidal hoop compression from it's inward load. Examples of this support concept are BPX, FIRE and C-Mod; and the “low field tokamak, ITER FEAT [30,31]. The opposed nature of the loading of these two coil systems invites attempts to interact these loads.

"Bucked" - The TF coil bears, or is "bucked" against the CS. This supports the TF and loads the CS is in Compression, and offsets its bursting load and tensile stress. Examples of this approach are CIT; and the “low field” tokamaks, JET[28], ITER (from the first EDA), PCAST[6].

"Bucked and Wedged" The TF is wedged and Bucked against the CS. The CS may be reinforced with an inner bucking cylinder at its ID. IGNITOR, and the early CIT design are examples of this concept.

-And hybrids, for example, TF partially wedged and Bucked against the CS (employed in ATBX [13])

Bucked; and bucked/wedged designs require the use of special low friction material at the interface between the CS and TF[21]. Free standing segmented solenoids require the same kind of material between the segments of the solenoid. The bucked and wedged option has about half the stress of the free standing coils, but requires careful fit-up between the CS and TF. The same or higher level of precision is required of the fit-up of the wedge faces of the TF in the wedged-only concept. With modern laser surveying equipment, and numerically controlled machining equipment, fit-up of coils should not be an issue, but the bucked and wedged concepts retains the reputation as more difficult.

Other Structural support schemes relate to winding techniques, and placement of poloidal coils. Bitter Magnets are made up from stacked plates typical of the high field water cooled solenoids used in the magnet laboratories at MIT and the National High Magnetic Field Laboratory in Florida. When the concept is employed in a toroidal field coil in a tokamak, inertial cooling replaces the water cooling. Edge conduction from liquid nitrogen provides cooling between pulses. C-Mod, IGNITOR, and FIRE use Bitter plate TF coils. Wound conductor in an array is the alternative to bitter plates. Coolant passages are needed to cool the interior conductors. JET is an example of this arrangement. The PCAST proposed design had wound conductor in a case.

Placement of the central solenoid and poloidal field coils strongly effects the structural response. With some difficulty, poloidal field coils can be placed inside the bore of the TF. Internal coils are more effective in driving current in, and controlling the shape and position of, the plasma. Break-outs, terminals and leads have current components that cross the toroidal field, and for high field tokamaks, loads on these components may be difficult to sustain. Practical assembly of internal coils requires joints in the TF coil winding. This detail has proved a challenge. C-Mod and General Atomic's DIII-D experiment are examples of tokamaks with internal coils. Placement of PF coils outside the TF coil array requires higher PF coil currents and more out-of-plane loading on the TF coils, but this is the more typical arrangement of coils.

Support of vertical loading has a number of structural concepts as well, but before these are considered, metal failure mechanisms need to be discussed.

Stress and Failure

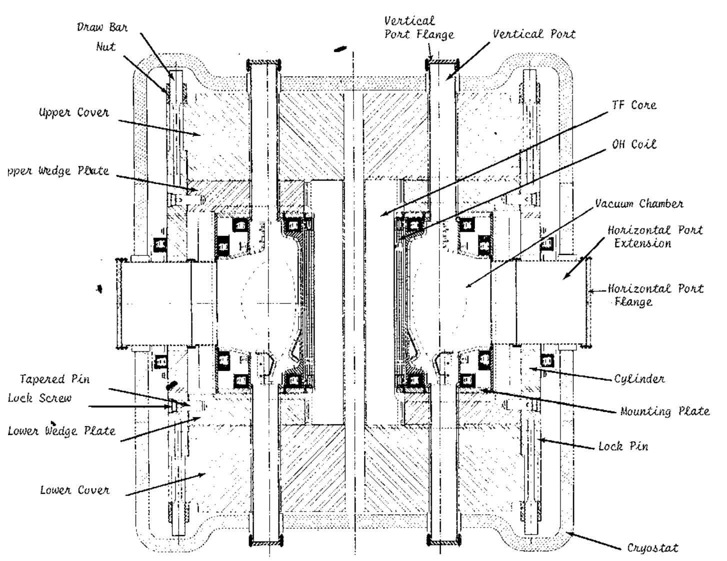
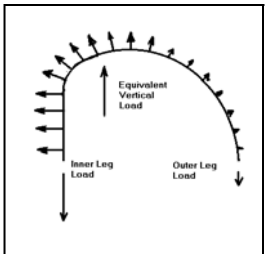


Figure 4 C-Mod Cross Section/Elevation View - Structural Elements

Failure in yield or ultimate breakage relates to the difference in the directional components of the stresses These differences appear in the equivalent stresses, Tresca, and Von Mises stresses which are used in the design codes to check the adequacy of a design. Balancing the stresses in the three directions an increase the overall load carrying ability of a conductor. An ideal condition is to achieve a hydrostatic stress state in which a uniform compression is experienced by the material in any and all three directions. Very large magnetic pressures can be supported in this way. In practice reduction in the differences between these stress components is the best one can do. An example is the loading in the inner leg of a TF coil in which radially inward loads can be supported by compressive stresses, but the vertical separating load is taken by tension. If some other way can be found for supporting the vertical separating force, then the stress difference, and thus the equivalent stress is reduced. A better stress state can be obtained if the vertical tension can be replaced by vertical compression. The goal is to obtain a nearly balanced three dimensional state of compression. Since free surfaces imply zero normal stress, and are an inherent violation of the hydrostatic stress state, one guideline for design is avoid free surfaces, or voids within the load carrying conductors. This is difficult to do if you are also obliged to provide cooling for the coils.

Vertical Separating Force Support Concepts



All the radial load support concepts load the TF inner leg in compression, either radial or hoop. To limit the equivalent stress of the TF inner leg, tensile stresses from the vertical separating forces should be minimized. This can be done by transferring the inner leg vertical tension to other structures. External support frames and presses have been suggested. A rather large frame and press concept was proposed for an early CIT arrangement. IGNITOR, FIRE and C-Mod also use substantial external structures for this purpose.

Tie rods, that pass through the center of the central solenoid have been used to resist the inboard leg vertical separating force. An early ITER EDA concept had a structural element that doubled as a bucking cylinder and vertical tensile element. A bucking cylinder in the bore of the CS is intended to support radially inward loads for toroidal field coils that rely on the central solenoid for support of centering loads. In a high field Tokamak, the present, most desirable aspect ratios limit the effectiveness of central tierod. Low aspect ratio machines leave little space in the bore of the central solenoid to resist much of the TF coil vertical separating force. Where bucking cylinders are not employed, the tendency is to dedicate most of the bore for leads and coolant channels. The most clever mechanism used to remove the vertical separating force from the inner leg is that used in IGNITOR. A large radial compression ring is used in concert with an external clamp/case, shown in Figure 5, which is selectively wedged to cause some of the radial ring load to compress the inner leg in a “pinching” action. An active magnetic press is used with the static ring loading. As the inner leg heats, the magnetic press load is backed off.

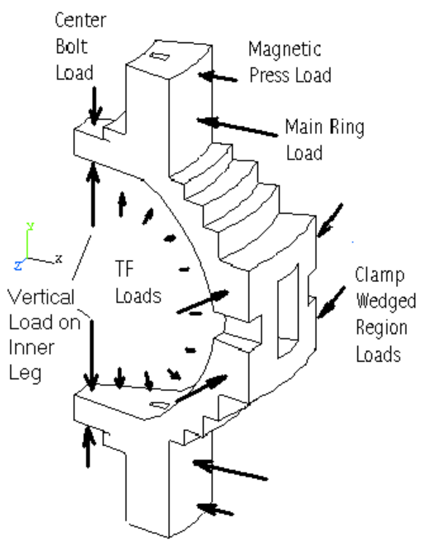


Figure 5 Force Balance on IGNITOR’s Steel "C" Clamp. The ring (not shown) is used to offset the vertical separating force through a "pinching action". It also maintains inner leg wedge pressure.

Out-of-Plane (OOP) Support

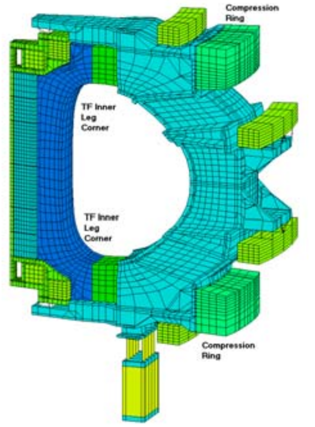
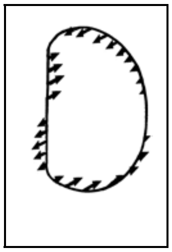


Figure 6. FIRE model with large Compression Rings.



Complicating the use of advanced structural concepts to support the loads from high field operation, is the need to support twisting loads from the interaction between TF coil currents and the poloidal fields. In most high field tokamaks, the inner leg TF centering force and resulting wedge pressure is used to frictionally couple the inner legs into a large heavy walled torque cylinder. For large wedge pressures the torsional capacity of this cylinder or central column is quite large. A difficulty arises when the outer leg radially outward load is communicated through the horizontal legs of the TF coil. This pulls at the upper and lower ends of the TF inner leg outward, diminishing the wedge pressures, and thus the OOP load carrying capacity of the ends of the large torque cylinder. This can be fixed with the addition of large compression rings. IGNITOR, FIRE and ITER-FEAT employ such rings. Mechanical jacks are planned for preloading IGNITOR and FIRE's large rings.

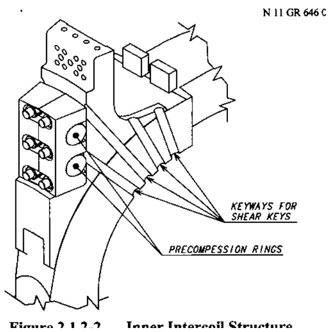


Figure 7. ITER FEAT's Glass Fiber reinforced ring

There is an interplay between the torsional shear distribution in the TF inner leg array, and the global torsional stiffness of the machine. Where outboard structures are stiff, TF inner leg torsional shear is higher near the ends of the inner legs, where there is a potential for the loss of out-of-plane frictional support.

End Of Flatop (EOF) TF Equatorial Plane Torsional Shear Stress - Comparison of Reactors

FIRE 10T, Wedged Inner Leg Torsionally Coupled	FIRE 12T, Wedged Inner Leg Torsionally Coupled	FIRE 10T Wedged, Only Mid-Plane Torsional Coupling	C-Mod	IGNITOR
14.0	19.9	37.1	22.8	33.3

legs into the inner corners of the TF coil. These radially outward loads "de-wedge" the corners, and if not compensated, can reduce the frictional torsional capacity to the point where the legs slip. One way to mitigate this effect is to de-couple the horizontal leg radial force from the inner leg by using C-Mod's solution, the sliding joint. A large radially inward preload can be used to offset this. Preloads can be provided by large compression rings. This is the solution used in FIRE. And This can introduce some limitation of PF Scenarios that can be run safely. A more detailed review of out-of-plane support mechanisms may be found in [11], and [23]

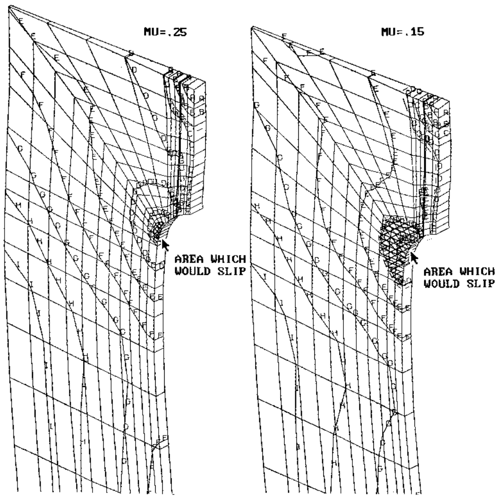


Figure 8 Local areas where slippage might occur in C-Mod

Even C-Mod, which has de-coupled the horizontal loads from the inner leg, and has reduced OOP loads due to the use of internal PF coils, exhibits some sensitivity to OOP frictional slippage. Inner leg joints allow all of the TF central column to be bonded unlike FIRE and IGNITOR which have frictional planes between inner leg assemblies, but because of the thermal cycling of the steel reinforced conductor used in C-Mod, failure of bond planes is considered. A friction coefficient (MU) for a fractured Polyimid Pre preg was measured as .3. Plotted in this figure is a post process of MU*wedge pressure minus the torsional shear, and where this goes below 1.0, slippage would be predicted. For the design field specification, no slippage was predicted for MU=.3. Small areas slip at lower friction coefficients. coil current scenarios that produce high poloidal fields, and low toroidal fields might be a problem, and operating current "windows" need to be specified to ensure adequate OOP load carrying capacity

Thermal Limits

Until superconducting solutions are available to reach the 15 to 20 Tesla range (at the conductor) needed for high field tokamaks, normal conductors are necessary. Peak temperatures allowed by insulation systems, and thermal strain interactions with support structures, limit how long an inertially cooled coil can be operated. Active cooling like that employed in the ARIES ST will be avoided in this discussion. As discussed before, active cooling tends to correlate with lower field designs. Current diffusion is a phenomena that is important in Bitter plate magnets. C-Mod, FIRE and IGNITOR use wide plates for their TF conductors. These allow currents to redistribute based on resistive and inductive effects. The inductive component results from the transient charging of the coil, and the tendency of the current to initially "hug" the bore of the TF.

As the stored magnetic energy increases, the current center moves outward. Resistive effects produces Joule heat, change the temperature, and then the local resistivity and causes currents to re-distribute to areas with lower resistivity. All these effects are coupled. Computer programs that solve the coupled problem include FORTE, used on IGNITOR; MAP used on C-Mod, and ANSYS used for FIRE in which two parallel transient solutions are used: one thermal and one electromagnetic. Nuclear heating and damage add to the complexity of the problem, and for high field magnets with low starting temperatures, like IGNITOR, magneto-resistive effects must be considered. For longer pulse lengths, the inductive part of the transient decays and the resistive component governs current redistribution. The effects of nuclear and magnetic resistance can then be simulated more simply.

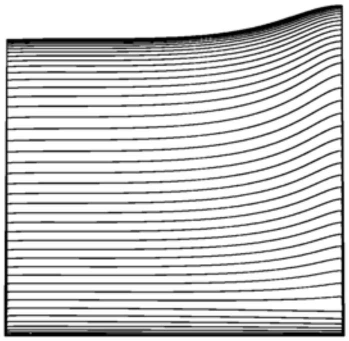


Figure 9 FIRE Temperature Profiles plotted with 1 second time intervals. with current redistributed based on resistance. The plasma side is at right and the higher temperatures result from the nuclear heat and resistivity changes.

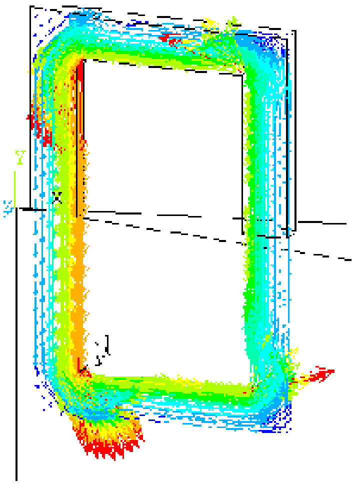


Figure 10 Current Density Vectors from an ANSYS Coupled-Thermal-Electromagnetic Analysis of the C-Mod TF Coils

Is it OK? -Design Criteria

In the past, Tokamak coil stress criteria have been selected for specific projects, and have been specified in project specific design guidelines. The requirements have been different depending upon the mission of the machine and what concessions were necessary to achieve the machine performance. All the criteria have basically evolved from ASME vessel criteria. They include load interaction equations, and monotonic stress criteria based on primary membrane, bending, discontinuity, and peak stresses. This assumes an identifiable set of primary loads and paths, and it also assumes that these stresses relate to the coil component failures that must be avoided. Fatigue and buckling criteria are added. Applying the vessel codes to tokamak coils has some fundamental weaknesses. Coils are electrical devices and fail differently than pressure vessels. The analogy of the coil "containing" a magnetic pressure is best applied to the free standing coil support options, but more efficient designs attempt to load the coil in three dimensions rather that the two dimensional membrane behavior that is the foundation of the ASME vessel codes. Primary loads in a tokamak include the central solenoid bursting forces, TF inner leg centering forces, TF vertical separating forces and TF outer leg outward loads. The choice of which structural components are assigned to support these loads may not be clear. A good example of this uncertainty is the assignment of the vertical separating force. For C-Mod, IGNITOR, and FIRE, the primary load path for the vertical separating force is the external structures, including the TF outer leg and outer case.

- The C-Mod inner leg is vertically cut by the finger joints, forcing nearly all the vertical load to the outer shell and drawbar structures.

- IGNITOR uses a system of heavy outer case structures and active and passive preload mechanisms to transfer the vertical load to the outer structures

- FIRE uses a heavy and relatively stiff outer case and thick TF outer legs to elastically pick up most of the vertical separating force. Limit analysis is used to demonstrate the self-relieving nature of the vertical tension stress in the inner leg.

To address the uncertainties in which stress corresponds with which code stress category, the FIRE criteria has included a limit analysis option to qualify the magnet system. A structural simulation is used to investigate failure mechanisms and determine a margin against failure that can be compared with the intent of the design code. Failure can be defined as exceeding the strain absorbing capacity of an insulation, excessive displacements, or tensile ultimate failure of some metallic component. The results of an extreme overload case of the FIRE tokamak is shown in figure ___. The TF field was increased from a nominal of 10 Tesla to 16 tesla. An elastic-plastic analysis was used, and the vertical separating force redistributed to the outboard structures as predicted. In the figure some slippage of the inner legs occurred and a small residual twist was locked into the TF inner leg array. strains were withiun the capacity of the insulation to accept without failure.

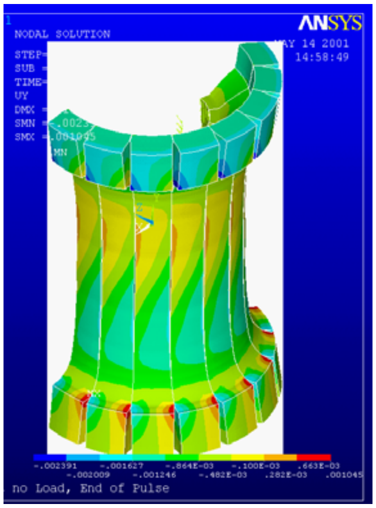


Figure 11 Toroidal Displacements, or Locked-In Twist, After two Shots with 16T TF. Displacement contours are continuous across inner leg boundaries if no slip has occurred.

Hyperstatic or Multiply Redundant Load Paths

Figure 12 Model of IGNITOR ULT at the 1992 review in Turin

Balancing stresses in three directions implies three load paths to support a load or set of loads, and this usually means a statically in-determinant or hyperstatic state. In simpler structures, designers attempt to avoid this, but by no means is this the only way to design a structure. Multiply redundant structures are as common as four legged chairs. In tokamak design, many examples may be found. At a time when the use of multiple load paths in IGNITOR's central column was being criticized, the statically in-determinant load share between the inner support structure and outer shear compression panels was the basis for support of TFTR's out-of-plane loads. As previously noted, for non-constant tension D TF coils, the vertical separating force in the TF coil is distributed to the inner and outer legs of the coil based on their relative stiffness. The vertical separating force has an inverse r distribution and the centroid of this distribution is often used in statics calculations to determine the loads in the inner and outer legs, but a much stronger effect is the relative stiffness of the inner and outer structures. To determiune the loading, a more elaborate structural analysis model is needed.

Persuading people the machine will work, or, How good is the analysis?

Critical in predicting the behavior of the multiple load paths is to be able to analytically model the tokamak in a reliable and believable way. For Bitter plate (and other large conductor concepts such as single turn tokamaks) electromagnetic diffusion needs to be modeled as well. The finite element codes used for these simulations have run many benchmark problems to qualify the features of the code. Simple problems with analytic solutions can and should be run to check out the code and the analyst's understanding of the code.

There is actually a wealth of published results regarding the use of, for example, ANSYS in design and analysis efforts that have had a high degree of predictive value. ANSYS periodically has conferences for users to present their results, and many typically relate to actual product development in which the predictive value of ANSYS is demonstrated commercially , (http://www.ansys.com/conference_2000/)

No tokamaks have been brought to commercial success, and few tokamaks have correlations between analysis and physical reality, A serious deficiency is the difficulty in matching analysis results with the behavior of the completed device. There is a reluctance to fully instrument the mechanical behavior of an electrical device because of the risk of shorts from the thermocouple and strain gage leads (fiber optic technologies offer some relief) . A common opinion, voiced recently at a FIRE PAC review is that "This is supposed to be a plasma physics experiment, not a structural mechanics experiment". Tokamaks are usually not well instrumented structurally, except as an after thought to understand a failure, as in some of the diagnostics added to C-Mod. Design and construction time scales typically add difficulty. The analytic state of the art for the present generation of engineer is much advanced over that available to the engineers that designed an experiment that typically takes 10 years from concept to full performance. A circumstance that offers some updated correlations is when a tokamak is to be upgraded in performance at the end of its life. This is the case with JET and was the case with TFTR. Both have been qualified for higher toroidal fields through more extensive structural modeling, and operation of the tokamaks has proved the validity of the approach [27],[28]. Recently there has been an effort to match structural analysis and test in the CS model coil program, with mixed results. Mis-behaving strain gages, and poor follow-on funding to maintain analysis models and evaluate measurements have made it difficult to do a proper structural "lessons learned" evaluation. Fiber-optic technology would appear to allow more extensive instrumentation in an operating tokamak which not only helps correlations with analysis, but would help minimize operator concerns and build confidence that the tokamak can be operated at its limits.

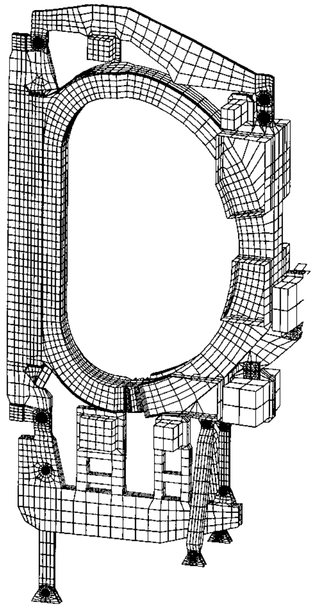


Figure 13 One of the more complicated models constructed by the author. This was an early (June 1994) ITER design that employed scissor keys that "stitched" the inner legs of the TF coil together, . These introduced a cross coupling between in-plane and out-of-plane forces that had to be modeled.

One of the most persuasive ways to approach this problem is to employ many analysts doing many analyses. During the ITER EDA period, the author had a challenging and rewarding experience testing his skills against the three other international participants[18]. At least it was persuasive for the analysts because we could converge on the correct behavior after modest adjustments in the models, but it seemed to frustrate those supervising the effort. For them the initial discrepancies seemed to taint the ultimate agreement. But complex behaviors relating to frictional behavior of the bucked system, and out-of-plane support were ultimately successfully analyzed (The scissor keys were especially interesting.) Good agreement was obtained (after some interations) for CS torsional shear stress, and TF case bending.

A good example of the effort to compare behavior of the ITER EDA reactor was the frictional simulation of the bucked interface between the CS and TF. This is a non-linear path dependent simulation that was effected by load profile and stiffnesses of the TF and CS. This is one of the more difficult analyses performed. There were many critical issues relating to this simulation. Frictional energy needed to be quantified to provide adequate cooling for the superconductors. Frictional constraint provided the primary support for the central solenoid. The simulation needed to determine the possibility that the CS might displace progressively with each succeeding shot and tear the terminal connections. It was intended to have the super-conducting TF energized without the CS being energized, and as the TF bore against the CS it would tend to frictionally expand the CS vertically , introducing tensile stresses in the CS. These were to be offset by an initial vertical compressive preload in the CS. This frictional "stretching" of the CS had to be simulated to qualify the initial preload. The simulation was first done with a non-linear ANSYS model. The solution for the problem was then formulated by the Russian Federation (RF) team using a finite difference solution of a matrix of springs and frictional gaps. Fundamental behaviors were confirmed.

IGNITOR is another example of multiple analysts coming to the same conclusions. The results of the initial European team of Toroni, Genachi, and Lanzavecchia were confirmed in Europe by the industrial team later charged with the R&D efforts, The results were confirmed by Princeton during the Ignition Studies program (which led to the Compact Ignition Tokamak - which initially was very similar to IGNITOR) Stone & Webster engineers confirmed the analyses of early compression versions of IGNITOR. At MIT, Becker and Dallesandro confirmed the analyses of a later version of the machine., and this author has done many confirmatory analyses of IGNITOR[8,16,17]. "Confirmed" is accurate, but results are not precisely the same. Most of the confirmations were done without the benefit of full integration into the project, and thus modeling details and material properties may not have been identical. Agreement has been good, and it is a testament to the robustness of the design that so many different analyses can produce essentially similar results.

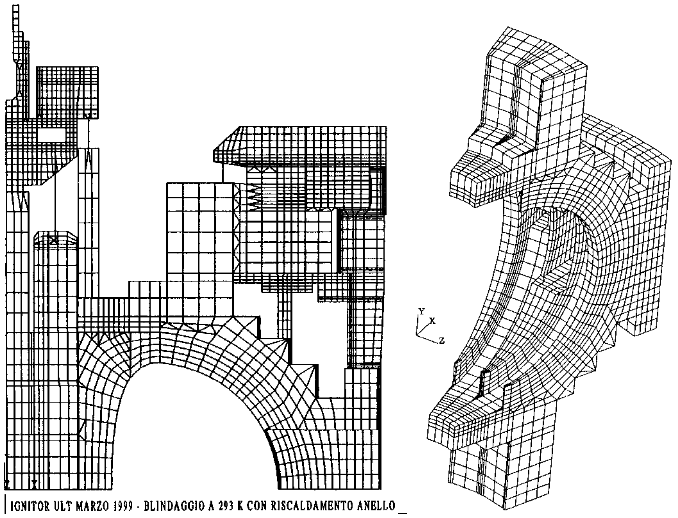


Figure 14, Ansaldo/Aldo Bianchi ANSYS Models

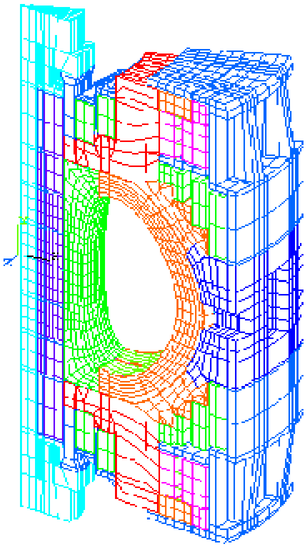


Figure 15, Author's ANSYS model

IGNITOR TF Equatorial Plane Stresses - Comparison of Results

				Inside TF EQ Plane Throat						Outside TF EQ Plane Throat				
Source	Time	Ring Ave Hoop	Ring Load (MN) *	SX Rad	SY Hoop	SZ Vert	VM	SYZ		SX Rad	SY Hoop	SZ Vert	VM	SYZ
Ansaldo**	EOF			-269	-126	-1	233			-34	-78	190	250	
Titus R#7,8	EOF	460	371	-272	-186	0	221	24		-36	-60	160	316	29.8

*Ansaldo results as reported in Ref[19] Table 5.111

Tokamaks follow attractive scaling laws. If you scale a tokamak up or down keeping the fields the same, and the temperatures the same, then the stresses are the same. Thermally limited pulse times increase with the square of the scale-change. A scale-up of IGNITOR was investigated [15]. Scale-downs are also possible. This raises the possibility of model testing to qualify a structural design and analysis. This was done for IGNITEX [14], and is proposed as an approach in ENEA's MODEL OMITRON project. The IGNITEX effort included solution of the electromagnetic thermal current diffusion problem using TEXCOR and the structural problem was solved with ABAQUS. Numerical results agreed with measured experimental results.

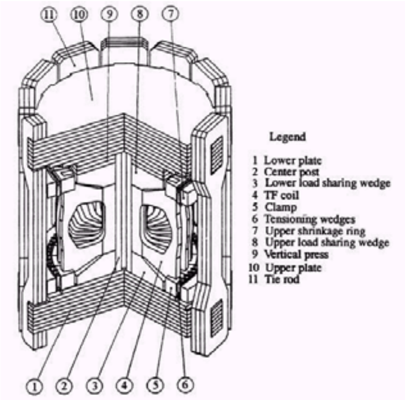


Figure 16 Ohmitron/Model Ohmitron

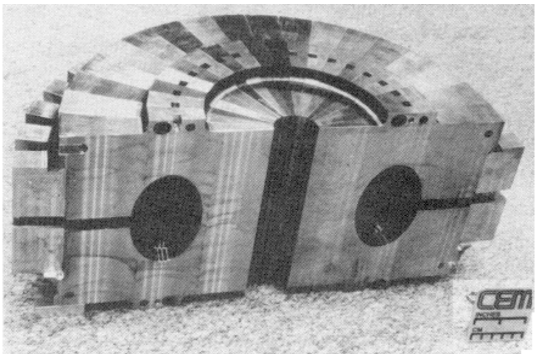


Figure 17 .06 scale test model of IGNITEX, ref 14, Support rings and bucking cylinder shown removed

C-Mod and it's Sliding Joints

Introduction of conducting joints in the picture frame TF coil has been partly motivated by an attempt to simplify the understanding of the structural response of the TF coil system. The attraction of joints in the TF is not just to manage loading, but also to achieve the advantages of de-mountable TF coils. This facilitates removal of vessel, blanket and divertor components and positioning of PF coils inside the bore of the TF, close to the plasma. Fixed (non-sliding) joints were implemented in DIII-D . These type of joints have substantial stress multipliers. For high field Tokamaks the sliding variety of joint is more attractive because these can be used in reduced equivalent stress configurations and ideally overcome the local higher stresses in the joint due to it's geometric complexity. This approach has been implemented in C-Mod and MAST, and studied for larger reactors, during the Ignition Studies Program in the mid-eighties [9] and later for the Steady Burn Experiment (SBX). The devil is in the details. These sliding joints swap a simplification of the global machine loading for the highly complex behavior of the joint. C-Mod's joint and some of the difficulties experienced with the joint are described in [10]. Variation in local current directions and densities due to current diffusion effects, out-of-plane loads due to the poloidal field, the difficult mechanical assembly, variations in the felt metal properties, -all make this solution more difficult than global forms of equivalent stress control such as those used in IGNITOR or FIRE. C-Mod's joint is an existence proof, at least for an 8 T machine (14T at the joint), with peak current densities of up to 9kA locally in the felt metal. C-Mod just completed a successful run with a few 8T shots and thousands of 5 T shots. Some improvements were made in the joints for this last campaign and the machine is currently shut down for inspection. The joints are in better shape than they were in 1998. If these types of joints are contemplated for another experiment such as the burning plasma experiment, a testing program with all the synergistic effects of combined field (and resulting out-of-plane loading) current density peaks, thermal effects, electrical polarity, sliding motion, and out-of plane deformations, have to be simulated

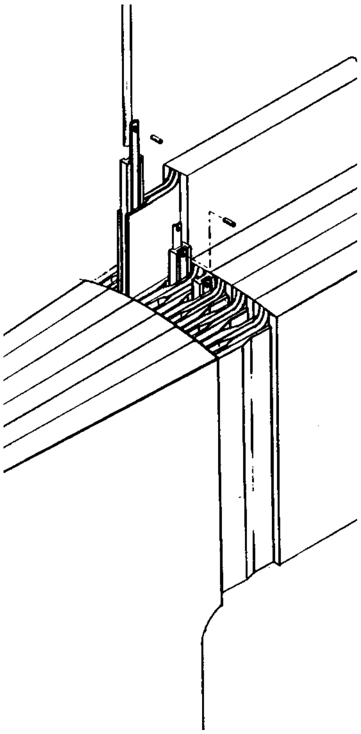


Figure 18 Assembly of the Original Finger Springs with G-10 "Carriers"

When the nature of the structure and loading make it difficult to assign the stress values to the stress categories defined in I 3.1, the limit capacity of the structure shall be determined by an analysis including elastic-plastic, frictional sliding and, large displacement phenomenon as appropriate. The limit load is that load which represents the onset of a failure to satisfy the Normal operating condition as described in I 2.6 The safety factor of limit load divided by the normal load shall be greater than 2.0

Joints Leads and Terminals

Sizing of a reactor during the conceptual phase needs to include allowance for the local details of the coil design. Stress analysis is typically based on “smeared” properties to which multipliers are applied to account for insulation, cooling and joint details. The stress multipliers due to the insulation and cooling channels and the stress multiplier for the joints need to be minimized. It is important to achieve these minimal factors for an efficient sizing of a reactor to be realized. Non-axisymmetric behavior of pancake wound solenoids was investigated in the CIT and BPX projects, and the phenomenon that produced ovality and higher stresses due to geometric effects were shown to be related to the distribution of voids around the transitions.[25] Pancake to pancake joints have a stress multiplier associated with them which is usually significantly larger than 1.0. There are two major sources of the multiplier. The geometry of the connection, including the effects of the offset, adds local stresses at the mechanical connection details. This usually requires addition of material to bring the stresses within the levels experienced by the rest of the turn. The increase in metal produces a stiffer region embedded in the coil and picks up more load than a single turn would normally take, adding further stresses to the mechanical details of the joint. The increase in metal also causes the second source of stress increase. Because the larger masses of metal run cooler in an inertially cooled coil, they don’t expand with the rest of the coil. The result is additional thermal tensile stresses in the vicinity of the joint.

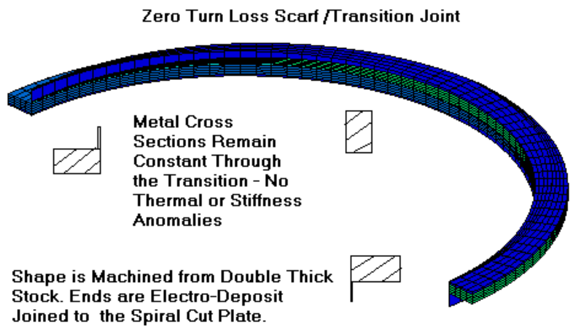


Figure 19, FIRE Inner Joint

. A shape similar to that shown in Figures 19 was developed for BPX after investigating many pinned or bolted or hooked joint concepts. The scarf/transition joint proposed for FIRE is a constant cross section design that eliminates both the stiffness and thermal anomaly. There is no void left by the joint, and no turn loss. In BPX, the joint was to be soft soldered over large lapped areas. Soft solder was chosen to avoid annealing out the cold work induced high strength of the OFHC copper. A better connection results using electro-formed joints at the butt ends of the scarf. Electroforming has been shown to produce strengths comparable with cold worked copper. Use of this detail means that the stress in the joint is the same as that computed by the larger models of the coil. This is especially advantageous at the ID of the coil. If some other joint concept is chosen, the coil stress allowable must be de-rated by the stress multiplier for the ID joint. This joint concept has similar advantages when used on the OD, but because the OD stresses in the CS are much lower than the ID, more conventional mechanical joints might be considered. For example, the double pancakes could be made an assembly with the scarf at the ID, then stacked and assembled mechanically at the OD. Since the coil segments are small enough, the scarf could be used at the OD as well. This would require electro-forming at the coil assembly, but C-Mod has shown that this is feasible.

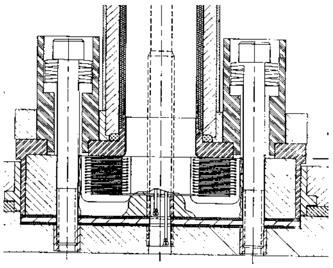


Figure 20, C-Mod's CS Coax Terminal Connection

This discussion is intended to illustrate the need to address local structural details of the coils at initial sizing of the machine. Joints, break-outs, leads and terminals very often are the limiting structural element of the design. In the ITER EDA CS model coil design effort, more effort was expended on the praying hand joints and buffer zone that surrounded and supported them, than the rest of the coil. Joints in High field regions are particularly challenging. Use of internal PF coils, i.e. coils in the bore of the TF is very challenging in a high field tokamak. The termination of C-Mod's EF3 coil failed and had to be re-worked with larger clamping details. Coax cables were thought adequate for C-Mod's CS connections, but loading internal to the coax, from the 14 Tesla field is so large, that it is hard to control the displacements at the coax connection to the coil "flag".[29] This connection carries 30, possibly 50 kA in up to a 14 Tesla field. Small uncompensated sections exist and these aggravate displacements. The "foot" that connects the CS lead to the coil through a felt metal pad, had to be re-manufactured with delicate EDM-Cut flexures to maintain contact of the "foot" while the central element of the coax deformed. This detail is still the subject of re-work in preparation for C-Mod's 2002 campaign. These difficult design details are usually cited as reasons to avoid internal PF coils in high field tokamaks. But it should be remembered that similar or worse conditions can be found in the central solenoid of a tokamak that employs external PF coils. The segmented solenoid used in the FIRE design compounds the difficulty of supporting the large loads with the requirement to allow differential radial motion of the CS segments. The axial field at initial magnetization (IM) is 18 T, and later in the pulse the axial field is 9 Tesla and the radial field is also 9 Tesla. Bucked and wedged designs such as IGNITOR have some advantage in that the CS is sandwiched between the TF and the bucking cylinder, and CS redial displacements are reduced. The bucking cylinder can also provide structural support for the leads and terminations.

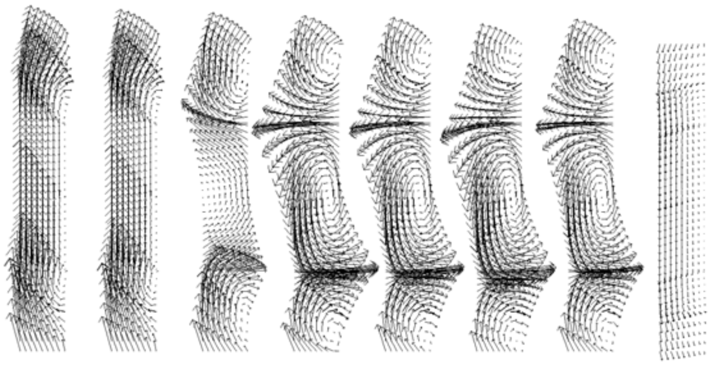


Figure 21 Field Vectors in the FIRE Central Solenoid, showing large radial fields that will cross the leads as they run up the bore.

Conclusion

The cross product of high fields and the high currents needed to achieve them produce loads that are a challenge to the fusion magnet designer. Physics advantages of high field tokamaks warrant finding designs that can survive the extremes in loads. A family of solutions, drawn from existing and conceptual reactor designs, is available to the designer of the next high field tokamak

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