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Physics and Engineering Assessments of the K-DEMO Magnet Configuration*

G. H. Neilson[1], T. Brown[1], K. Im[2], C. Kessel[1], K. Kim[2], P. Titus[1], Y. Zhai[1]

[1] Princeton Plasma Physics Laboratory, Princeton, NJ 08543 U.S.A.
[2] National Fusion Research Institute, Daejeon 305-333, the Republic of Korea

e-mail contact of main author: hneilson@pppl.gov

Abstract. Increased attention is being given now to studies of next-step fusion facilities with nuclear missions. Among these, South Korea's K DEMO is unique in its focus on a high toroidal magnetic field, large major radius, steady-state tokamak design for the core of a facility to test fusion nuclear components in Phase I and, after upgrades, produce 500 MW of electricity in a Phase II. Innovative features of the K DEMO magnet set include the use of two toroidal field (TF) coil winding packs with conductor grading and a machine configuration designed for vertical maintenance. The magnet arrangement features large TF coils and widely-spaced poloidal field (PF) coils to accommodate removal of in-vessel components as large modules. Physics and engineering assessments of the pre-conceptual K-DEMO magnet configuration are reported, including: 1) design point and operating space assessment, 2) conductor assessment, and 3) structural assessment. It is found that a reference design point at 6.8 m major radius and 7.4 T toroidal field provides sufficient operating margins for the 500 MWe Phase II mission. Analyses of candidate cable-in-conduit conductors provide predictions of critical current degradation, both in the initial load cycle and an additionally with cyclic loading. A first-pass global analysis of the magnet system found minimal out-of-plane deformations of the TF coil, but an overstress condition in the inner leg of the TF coil. However an analysis taking into account elastic-plastic behavior, frictional sliding, and displacement shows that the structure can safely carry the load. Although the design evolution is still at an early stage, these assessments support the design point choices to date and the expectation that a feasible solution for the high-field K DEMO magnet system can be found.

1. Introduction

Through the ITER project, the international fusion community has established a timeline for creation and study of a power plant-scale burning plasma. As a result, increased attention is being paid to the roadmap from ITER to a fusion DEMO, including studies of next-step fusion facilities with nuclear missions ranging from component testing and fusion nuclear science to electricity generation. Among these, South Korea's K-DEMO [1] is unique in its focus on a high toroidal magnetic field (B_T = 7.4 T on axis), large major radius (R = 6.8 m), steady-state tokamak design for the core of a facility to test fusion nuclear components in Phase I and, after upgrades, to produce 500 MW of electricity in a Phase II. Innovative features of the K-DEMO magnet set include the use of two toroidal field (TF) coil winding packs, each of a different design, and a configuration design framed by maintenance considerations. A Princeton Plasma Physics Laboratory team has supported the pre-conceptual design of the K-DEMO device with coupled physics and engineering analyses of the magnet configuration, providing feedback to guide the design evolution.

2. Magnet Configuration

The K-DEMO device reference point incorporates a double-null (DN) diverted plasma with strong shaping (high elongation and triangularity). The DN option promotes higher plasma performance, with an accompanying reduced machine size when compared with single-null (SN) designs. High availability requires large openings to remove and replace large in-vessel segments. The K-DEMO device incorporates a vertical maintenance approach, taking advantage of the overhead space and the tooling needed to assemble the device. The main features of the K-DEMO tokamak are shown in Fig. 1.

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The 16 toroidal (TF) coils are sized to provide sufficient space for in-vessel segments to pass between coils during maintenance operations. This consideration provides low ripple (0.08%) and wide maintenance access at the cost of a large TF coil that extends farther out in major radius than physics alone would require. The coil design incorporates two winding packs to

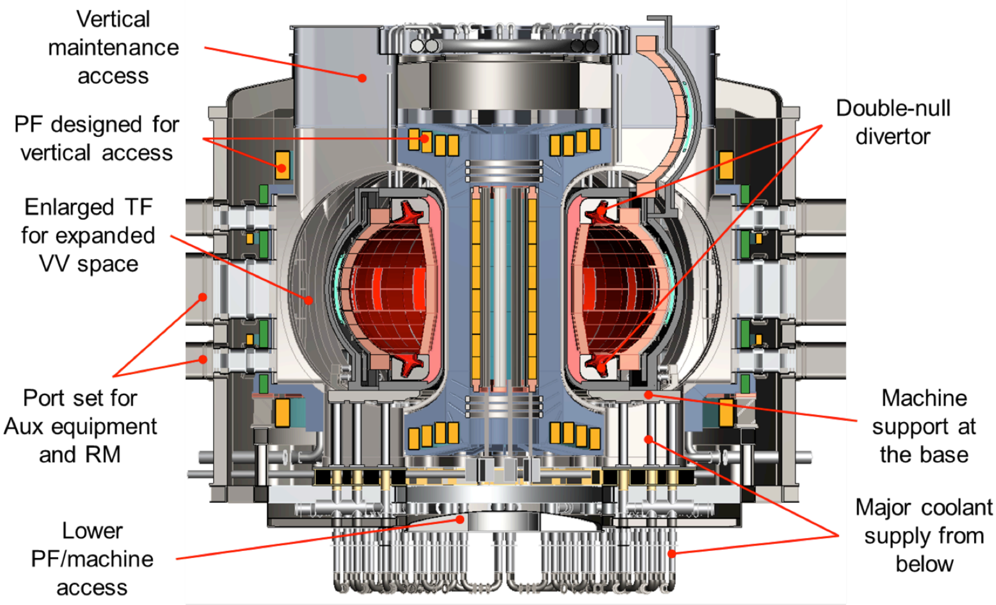

Fig. 1. K-DEMO device core design features

eliminate coolant pressure drop issues associated with the extended winding length of the coils and reduces the quantity of costly superconducting material in the low field winding. Different cable-in-conduit (CICC) conductor designs are used in the high- and low-field windings. A near continuous outer shell, compatible with the openings defined for the selected vertical maintenance approach, supports the TF overturning forces. Fig. 2 shows the basic TF winding arrangement of the 16-Tesla peak field magnet.

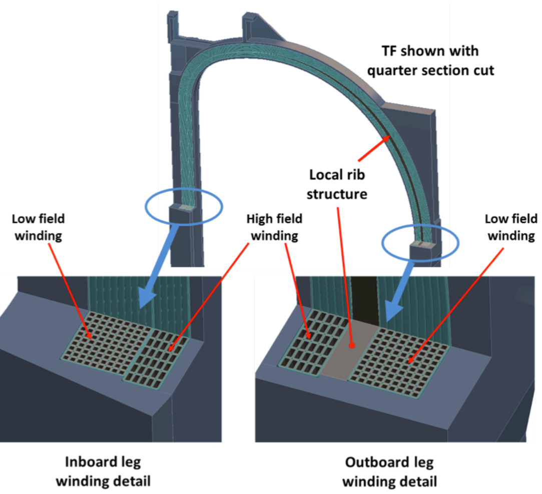

Fig. 2. Illustration of two-winding TF design.

The poloidal field (PF) coil positions are constrained by the large TF coil, which places them far from the plasma, and by the vertical maintenance concept, which requires wide radial spacing between PF coils at the top of the machine. Table 1 and Fig. 3 show a PF configuration that satisfies these constraints and supports a double-null plasma equilibrium, shown in the figure, that corresponds to Phase II conditions where 500 MW of net electricity is generated. The pre-charge poloidal flux is -100 Wb, scaled from the ARIES-ACT2 [2] design. The high-power condition has poloidal flux of +100 Wb, plasma current of 14 MA, and $\beta_N = 4.2$.

Table 1. PF Coil Locations and Currents for Reference High-Power Equilibrium

Coil	R (m)	Z (m)	Pre-charge currents (MA)	Steady-state currents (MA)
CS1	1.52	0.70	-3.11	-1.55
CS2	1.52	2.10	-3.11	-1.55
CS3	1.52	3.50	-3.25	-2.25
CS4	1.52	4.90	-3.25	-2.25
PF1	2.98	8.31	4.51	5.72
PF2	3.66	8.31	6.24	7.91
PF3	4.34	8.59	6.17	7.90
PF4	5.02	8.75	5.29	6.91
PF5	12.96	7.50	-13.70	-15.95
PF6	14.88	2.95	0.14	-0.22

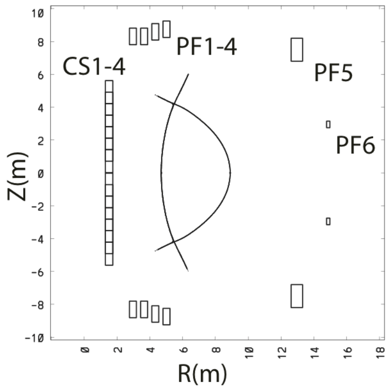

Fig. 3. PF coils and reference equilibrium.

3. Physics Assessment

The reference design point for K-DEMO was selected using a systems analysis approach in which a set of input parameters including R (major radius), B_T (toroidal field), and other variables was scanned to generate a large data base of viable physics operating points satisfying plasma power and particle balance equations [3]. Peak heat flux in the divertor is determined using a formulation by Fundamenski [4] for the power scrape-off width, typically giving 3-4.5 mm for K-DEMO cases, and a divertor radiation fraction of 90% is assumed. In all cases the plasma is assumed to have an up-down symmetric double null divertor, plasma elongation of 2.0 and triangularity of 0.63. An overall TF coil current density (including winding pack and all structure) of 15 MA/m^2 is used, which gives good agreement with more detailed magnet design analysis. A similar assumption is made for the central solenoid coil. The inboard radial build provides space allocations for the scrape-off layer, first wall, blanket, shield, vacuum vessel, gaps for assembly and thermal insulation, and structure.

The reference design point was chosen to support missions of P_{elec} > 150 MW and > 400 MW in Phase I and II, respectively, while reducing mission risks due to too limited an operating space or to divertor heat fluxes exceeding technology limits. A major radius of 6.8 m was chosen to keep the peak divertor heat fluxes to less than 12.5 MW/m^2 and 15 MW/m^2 in Phase I (150-400 MWe) and Phase II (400-700 MWe), respectively. The maximum magnetic field at the nominal plasma radius is 7.4 T, taking advantage of high-field magnet technology (i.e.., a field strength of 16 T at the coil and coil current density, including winding pack and structure, of 15 MA/m^2 for the TF coil, *vs.* 11.5 T at 12 MA/m^2 in ITER). The plasma elongation was also chosen to be 2.0 in order to sufficient operating space to provide margin against uncertainties in physics projections. The plasma triangularity was chosen to be 0.63 based on previous power plant studies to provide good ideal MHD stability, allow sufficient inboard shielding near the strike point, and avoid spatial conflicts on the outboard.

In Phase I, it is assumed that β_N^{total} and H_{98} are limited to < 3.5 and < 1.3, respectively, where β_N^{total} is normalized beta, including thermal and non-thermal components, and H_{98} is an energy confinement scaling multiplier. At the lowest divertor heat flux an operating space is available with toroidal fields at the TF coil of 14.5-16 T, plasma toroidal fields of 7.0-7.5 T, plasma currents of 10.5-12.2 MA, β_N^{th} of 2.5-2.75%, q_{95} of 7.5-8.0, n/n_{Gr} of 1.15-1.25, Q of 10, H_{98} of 1.25-1.3, auxiliary powers (for heating and current drive) of 120-155 MW, bootstrap fractions about 0.56-0.62, average neutron wall loads at the plasma surface of 1.2-1.48 MW/m^2, Z_{eff} of 1.4-2.0, and net electric power outputs of 150-200 MW.

In Phase II, it is assumed that β_N^{total} and H_{98} are limited to < 5.0 and < 1.6, respectively. At the lowest divertor heat flux an operating space is available with toroidal fields at the TF coil of 14.5-16 T, plasma toroidal fields of 7.0-7.5 T, plasma currents of 11.7-13.0 MA, β_N^{th} of 3.25-3.5%, q_{95} of 7.0-8.0, n/n_{Gr} of 1.25, Q of 17.5-20, H_{98} of 1.53-1.6, auxiliary powers of 96-118 MW, bootstrap fractions about 0.72, average neutron wall loads at the plasma surface of 1.6-2.0 MW/m^2, Z_{eff} of 1.45-2.0, and net electric power outputs of 400-520 MW. The operating spaces are shown in Fig. 4 for both the low and high electric power results, with varying divertor peak heat flux values.

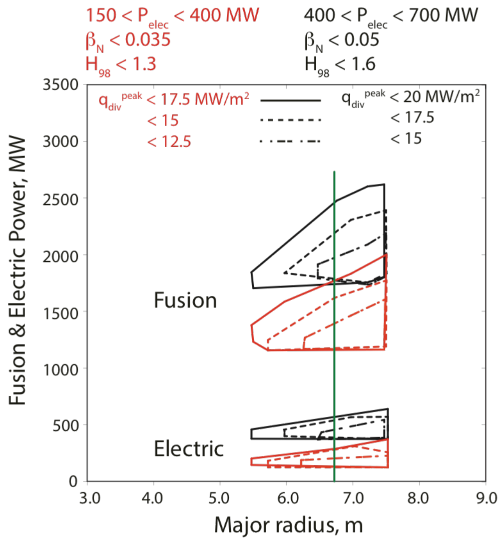

Fig. 4. Operating space for Phases I and II for different limiting divertor heat flux assumptions.

The Tokamak Simulation Code (TSC) [5] was used to simulate the time-dependent evolution from early startup (I_p = 500 kA) to t=4300 sec, when the current density profile is close to steady state. The TSC solves the free-boundary equations as a function of time. Flux surface-averaged transport equations are solved to obtain the temperature profiles, utilizing a modified Coppi-Tang [6,7] transport model with a prescribed temperature pedestal. The density profile is prescribed with a peak to volume average of 1.4-1.5 [8], with an edge pedestal and finite separatrix density at 0.35 times the central value. In the simulation it is found that the plasma requires 165 Wb of poloidal flux swing to reach full current at 12.3 MA from 0.5 MA, including resistive, internal and external inductive contributions. Assuming an additional 20 Wb are needed to bring the plasma to 0.5 MA, the total flux swing is 185 Wb. The thermal diffusivity is adjusted to provide sufficient global confinement to reach a target β level, as identified by the 0D systems code analysis. In order to raise the radiated power to a significant level argon and tungsten are included at 0.1 and 0.001%, respectively, of the electron density. It is found that the large toroidal field and high electron central temperatures lead to a significant cyclotron radiation loss. Profiles of plasma current density, input power, and radiation losses in the fully relaxed flattop plasma condition are shown in Fig. 5 and a comparison between the 0D and time-dependent analyses of plasma parameters is provided in Table 2.

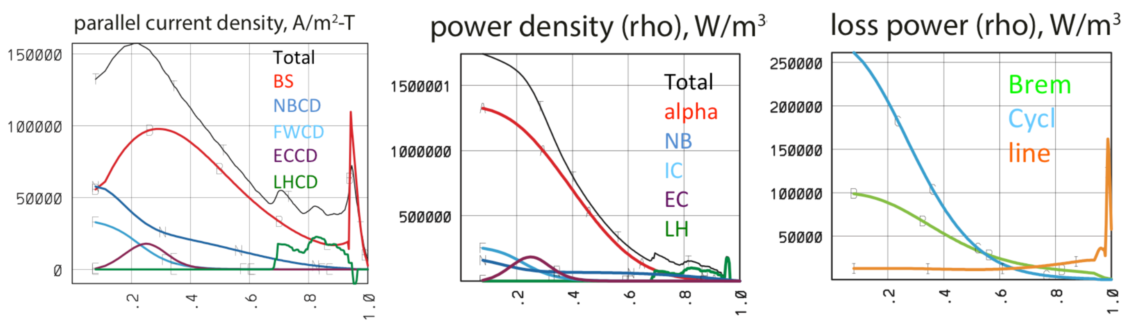

Fig. 5. Profiles of plasma current density, input power and radiation losses after full current relaxation.

4. Conductor Assessment

The Florida Electro-Mechanical Cable Model (FEMCAM) [9] was used to evaluate the design and provide performance assessments for candidate K-DEMO cable-in-conduit conductors (CICC). At issue is the potential for performance degradation and even fracture of the superconducting strands as a result of mechanical displacements. The analysis combines 1) the thermal bending effect during the cool down phase of the manufacturing process due to differential thermal contraction and 2) the electromagnetic bending effect due to locally accumulating Lorentz force during magnet operation. It also includes effects of filament fracture and related local current sharing.

Table 2. Plasma Parameters from Systems Analysis Reference and 1.5D Simulations

	0D Systems Code	1.5D time-dep't. TSC
Ip, MA	12.3	12.3
f_{BS}	0.67	0.68
B_{tor}, T	7.4	7.4
q_{min}, q(0)		1.6,1.8
li(1)	0.7 input	0.8
n/n_{Gr}	1.15	1.19
W_{th}, MJ	677	746
n(0), 10^{20} m^3	1.25	1.15
$<n>_v$, 10^{20} m^3	0.89	0.92
n(0)/<n>	1.44	1.54
b_N^{th}	2.53	2.88
$H_{98(y,2)}$	1.3	1.3
$T_{e,i}(0)$, keV	30.8	40
T(0)/<T>	2.12	2.7
P_{alpha}, MW	298	308
P_{aux}, MW	119	120
P_{cycl}, MW	44.5	42*
P_{line}, MW	22.3	28
P_{brem}, MW	45.6	28
Z_{eff}	2.04	1.48
n_{He}/n_e	0.063	0.08
n_{Ar}/n_e	0.003	0.001

*assuming first wall reflectivity of 90%.

The K-DEMO TF conductors are graded into two types of CICC based on the magnetic field distribution on the TF coils. The high field region TF CICC runs 65.5 kA net current using cable of 1800 Nb_3Sn strands inside a 316LN stainless steel jacket. The cable is cooled by forced flow helium in the 28.1% void space and a helical spiral cooling channel. Fig. 6 (left) presents the K-DEMO high field TF CICC cross-section. The peak field in the high field CICCs is ~16 T. The high field CICC has a rectangular cross section with an aspect ratio of 2.2 to reduce a transverse pressure load transferred by the cable during magnet operation. The low-field TF CICC has a square-shape cross section using cable of 360 Nb_3Sn strands and 402 copper strands as shown in Fig. 6 (right). Peak field in at this conductor is 12.1 T. The low-field TF CICC has ~27.1% void fraction and a central cooling channel.

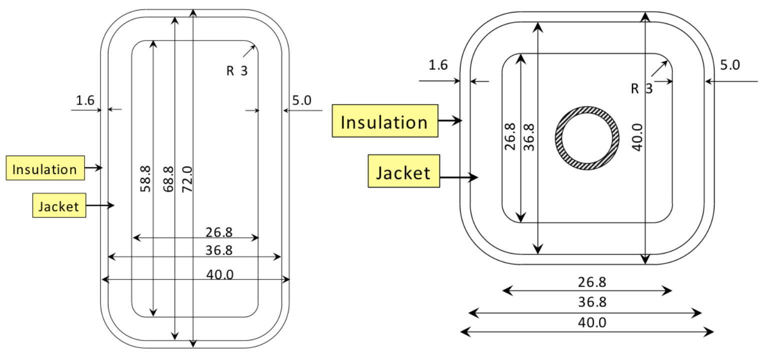

Fig. 6. K-DEMO high-field (left) and low-field (right) TF CICC conductor cross sections.

Advanced Oxford Superconducting Technologies (OST) dipole strands with J_c ~2600 A/mm^2 are used for the K-DEMO TF conductor performance evaluation. An assumed 0.15% hoops strain is included in the evaluation. The axial thermal compressive strain is assumed to be ~0.45% [10] and a bending wavelength of 7 mm is used, based on the long twist pitch used for TF cables. The strain sensitivity of OST dipole strand critical current is shown in Fig. 7 at the TF CICC nominal condition. Fig. 8 presents J_c degradation due to bending of the OST dipole strand with and without assumed filament fracture for both no current transfer and full current transfer cases. The blue curves show the effect of filament fracture where a conservative assumption of J_c is used. It clearly shows the J_c degradation due to bending of the advanced Nb_3Sn strands if filament fracture beyond the irreversible strain limit is introduced in the analysis. The analysis shows that K-DEMO TF conductors may have 10-30% critical current degradation in the initial load cycle. With cyclic loading, the TF conductors are predicted to have 5% additional degradation due to mechanical strains. These analyses are compatible with current design assumptions.

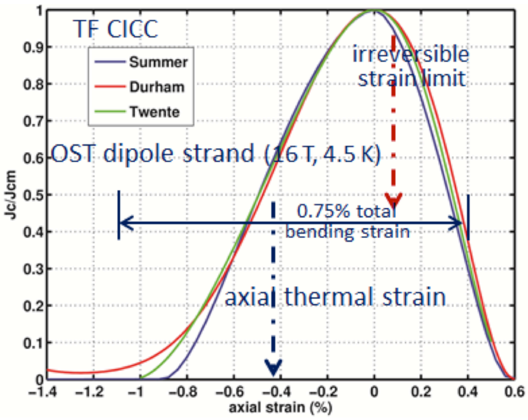

Fig. 7. Strain sensitivity of OST dipole strand for K-DEMO TF CICC evaluation. The peak total bending strain is ~0.75% for TF CICC strands.

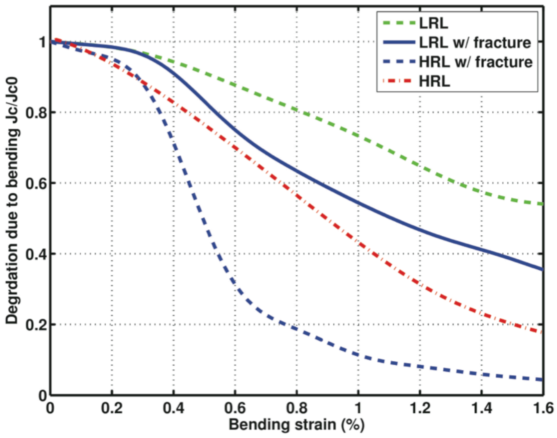

Fig. 8. OST strand bending characteristics with full filament current transfer (LRL) and no filament current transfer (HRL).

5. Structural Assessment

The global structural adequacy of the K-DEMO magnets is being assessed at this stage to guide the allocation of space between steel structure and other elements, as feedback to the iterative design development process. A finite-element model (Fig. 9) of the magnet system was created from the CAD design data, and magnetic fields and Lorentz forces were calculated from a reference equilibrium at full magnetic field and performance. Allowable stresses were developed based on structural properties of the proposed TF coil case material (316 stainless steel), and applying criteria according to the ITER Magnet Structural Design Criteria document. The primary membrane stress allowable at the 4 K operating temperature is 666 MPa, whereas the bending allowable is 1.5 times this value, or 999 MPa.

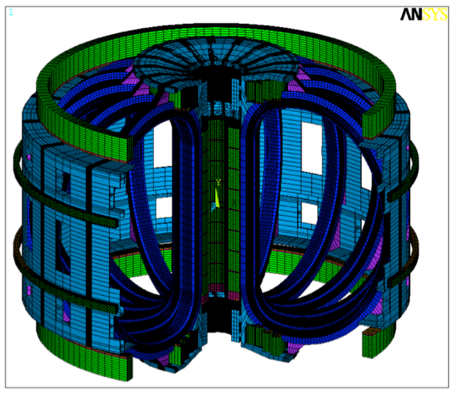

Fig. 9. Magnet system model with graphical symmetry expansion (Only one TF coil is actually modeled).

Stresses and deformations were calculated using the ANSYS code. Initial calculations showed the out-of-plane deformations of the TF coil to be only ~1 cm, a small value for such a large machine. Displacements at this level would likely not impose any problems with either component interferences or field quality. However, the inner leg of the TF coil was found to be over-stressed. This is shown in Fig. 10, where small local areas of the inner leg exceeding 900 MPa can be seen as gray areas in the upper picture. Bending stresses in the inner corners of the TF of up to 930 MPa should satisfy the bending allowable of 999 MPa. It is argued that the primary stress is around 860 MPa in the yellow and brown contours seen in the lower picture. This is a Tresca stress which is basically the absolute sum of the wedging or vault pressure and the vertical tension from the bursting load. This exceeds the 666 MPa allowable by about 30%.

Various options were investigate to resolve the static overstress condition. The addition of structural reinforcements to the outer leg and horizontal legs of the TF case did not sufficiently reduce the inner leg stress. Likewise, increasing the wall thickness in the wedged “nose” of the case by 10 cm (25%) was insufficient, an initially surprising result that is explained by the fact that the winding pack itself contributes substantially as well. Reduction in the toroidal field strength to 6 T would be needed to bring the stresses into accordance with the 666 MPa primary membrane allowable.

Structural design codes allow various options for assessing compliance with stress allowables and evaluating the load carrying capacity of a structure. One such option is to perform a detailed, non-linear analysis that accounts for elastic-plastic behavior, frictional sliding, and large displacement to determine the limit load on the structure. The limit load is that load which represents the onset of a failure to satisfy the normal operating condition. The structure is considered adequate if the limit load exceeds the normal load by a factor of safety greater than 2.0. To investigate this method for K-DEMO a test load of 2.0 times normal was applied. The primary stress increased by only 12% but the strains are substantially higher. **Error! Reference**

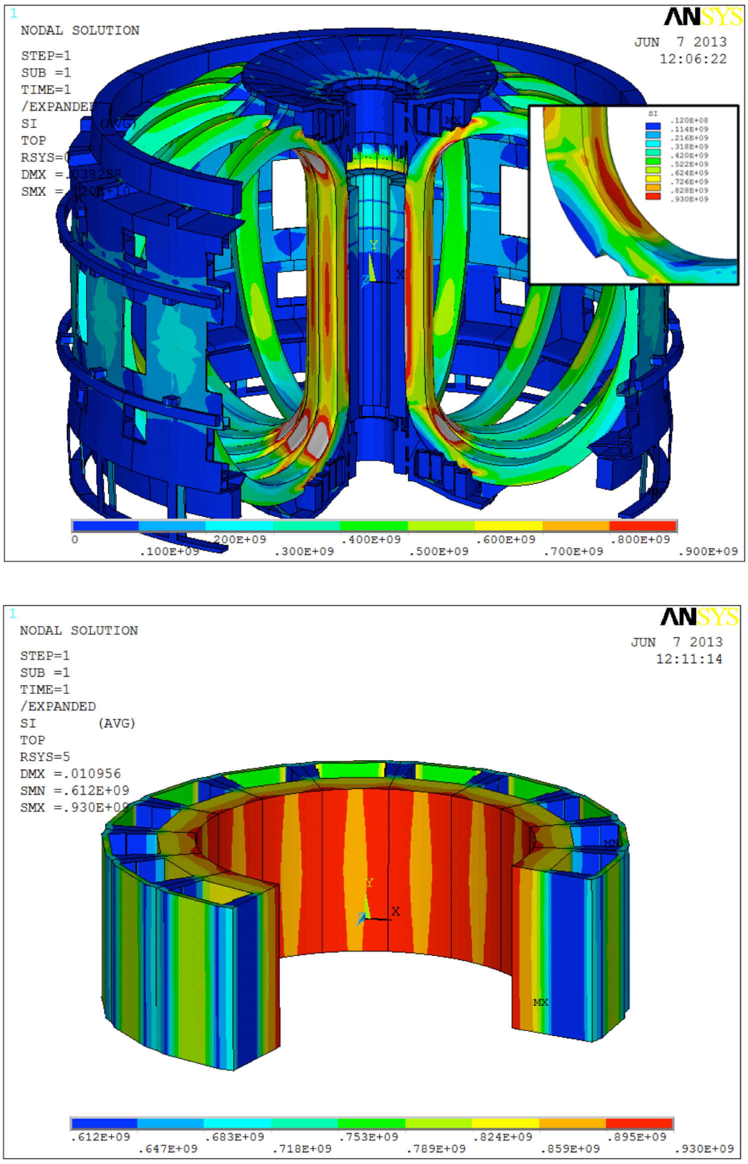

Fig. 10. Upper: TF coil case stress distribution contoured to 900 MPa maximum. Lower: Details of TF inner leg stress, contoured to maximum inner leg stress.

source not found. shows the case strains to be about 1.2%; the strains in the superconductor would be similar. The acceptability of this level of deformation in the superconductor would have to be confirmed. With the loads removed, a permanent deformation in the structure for this artificial loading scenario of up to 10 cm is seen. Significantly, the deformations are bounded and the analysis converges to a solution, meaning that while the structure deformed it did not fail. Such a result demonstrates adequate margin. Rigorously, the exercise would have to be extended to address insulation failure, supercondoctor breakage, and other failures.

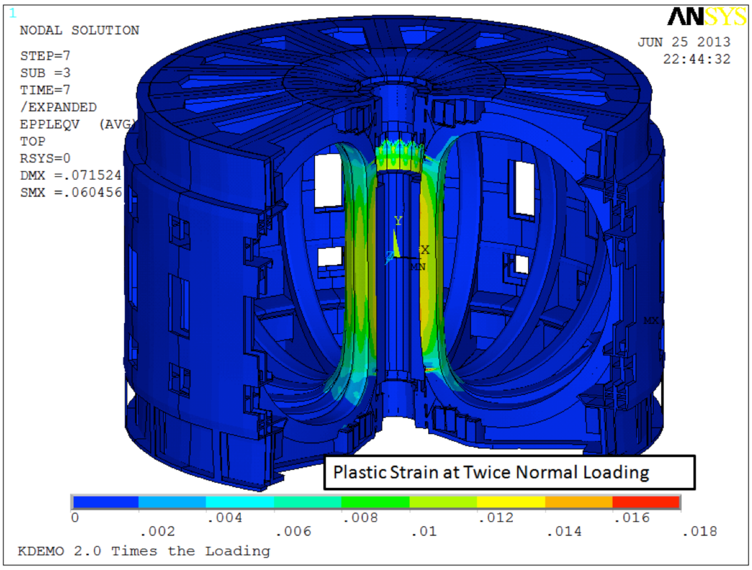

Fig. 11. Plastic strain in the TF inner leg with a load of 2.0 times normal.

6. Summary

A tokamak magnet configuration for the K-DEMO mission has been developed at a preconceptual level. A TF coil design with two winding packs of different design and a heavy steel structure address issues associated with coolant pressure drop, structural support, and cost. The TF and PF coils are sized and spaced to provide openings for vertical removal of large in-vessel segments for maintenance. The physics assessment shows that this magnet set can support double-null plasmas with a finite operating space, providing reasonable performance margin against physics uncertainties. Degradation of the TF conductor under load has been calculated and found to be consistent with design assumptions. The support structure is found to be adequate based on an elastic-plastic limit analysis. Although the design evolution is still at an early stage, these assessments support the design point choices to date and the expectation that a feasible solution for the high-field K DEMO magnet system can be found.

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