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Superconducting magnet and conductor research activities in the US fusion program

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Abstract

Fusion research in the United States is sponsored by the Department of Energy's Office of Fusion Energy Sciences (OFES). The OFES sponsors a wide range of programs to advance fusion science, fusion technology, and basic plasma science. Most experimental devices in the US fusion program are constructed using conventional technologies; however, a small portion of the fusion research program is directed towards large scale commercial power generation, which typically relies on superconductor technology to facilitate steady-state operation with high fusion power gain, *Q*. The superconductor portion of the US fusion research program is limited to a small number of laboratories including the Plasma Science and Fusion Center at MIT, Lawrence Livermore National Laboratory (LLNL), and the Applied Superconductivity Center at University of Wisconsin, Madison. Although Brookhaven National Laboratory (BNL) and Lawrence Berkeley National Laboratory (LBNL) are primarily sponsored by the US's High Energy Physics program, both have made significant contributions to advance the superconductor technology needed for the US fusion program. This paper summarizes recent superconductor activities in the US fusion program.

Keywords: Superconductor; Superconducting magnet; Fusion technology

1. Introduction

The largest proposed United States activity in magnetic confinement fusion is the International Thermonuclear Experimental Reactor (ITER). The United States has proposed to supply four of the seven modules (six in the assembly plus one spare) needed for the ITER Central Solenoid (CS). The ITER CS is a 840 tonnes system, requiring 138 tonnes of Nb_3Sn superconductor strand. The CS has a peak flux density of 13 T, a peak current of 45 kA and stores 6 GJ of magnetic energy. The CS will provide up to 277 Wb of magnetic flux to inductively drive 15 MA of plasma current in ITER. Key issues for the CS design have

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been identified as: prediction of Nb_3Sn superconductor cable performance in the presence of large transverse Lorentz loads, and verification of the structural integrity of the CS magnets during cyclic operation. The United States development effort for ITER has focused on these two issues.

Lawrence Berkley National Laboratory (LBNL), the Plasma Science and Fusion Center at M.I.T. (MIT-PSFC), Lawrence Livermore National Laboratory (LLNL), and the Advanced Magnet Laboratory (AML) recently produced and tested a magnet system capable of focusing intense beams of heavy ions for inertial confinement fusion. The prototype focusing magnet developed for the heavy ion fusion (HIF) program consisted of a superconducting quadrupole doublet integrated inside of a low heat-leak cryostat.

MIT-PSFC and Columbia University designed, built, and recently produced the first plasmas of the levitated dipole experiment (LDX). LDX is one of the OFES's innovative confinement concepts plasma science experiments. LDX is the largest levitated dipole experiment in the world. The core of the LDX machine comprises three superconductor coils. Each coil employs a significantly different superconductor technology, which is best suited to that coil's function in the device.

2. Superconductor technology for ITER

The International Thermonuclear Experimental Reactor (ITER) program is an international magnetic confinement fusion (MCF) project involving The People's Republic of China, the European Union, India, Japan, the Republic of Korea, the Russian Federation, and the United States of America [1]. ITER is a burning-plasma, engineering test reactor based on the tokomak configuration. It is designed to generate inductively driven plasmas producing 500 MW of fusion energy for durations of up to 500 s with a Q of about 10. The overall objective of ITER is to demonstrate the technological feasibility of fusion energy for commercial power production.

The conductors for the ITER magnet systems employ a cable-in-conduit (CIC) configuration. Conductors for the ITER CS contain approximately 1000 superconductor and copper strands that are combined into a multiple stage cable and encased in a structural metal jacket, which also serves as the pressure boundary for the cable's supercritical helium coolant. Subdivision of the conductor into large numbers of superconductor strands significantly increases its wetted perimeter, resulting in a marked increase in conductor stability.

Two large CIC superconducting "model" coils were built and tested during the engineering design activities (EDA) phase of the ITER program, which ran from 1993 through 2001. One coil was intended to simulate the operation conditions expected of the ITER CS [2], while the second coil was intended to simulate the operation conditions expected of the ITER toroidal field (TF) coils [3]. The US provided the 74 tonnes support structure and a 47 tonnes, 10 layer inner module for the CS Model Coil, and participated strongly in both the CS and TF Model Coil test programs. Although both magnet systems fulfilled all of their technical objectives, the measured conductor performance for each coil was significantly below the behavior predicted prior to the start of testing.

2.1. *Superconductor strand and cable investigations*

Most investigators attribute the discrepancy in observed conductor performance to a combination of high transverse electromagnetic loading of the CIC combined with its relatively low transverse stiffness [4]. As the cable in a CIC conductor deforms under electromagnetic loading, it typically compresses towards one side of the conduit. Transverse loads are concentrated at points where the strands cross over one another in the cable pattern. At the same time, the unsupported strand lengths between these cross-over points bend under the influence of the Lorentz force loading. To help distinguish the consequences from these two effects, an experimental program was implemented at the MIT-PSFC and at the Applied Superconductivity Center (ASC) to: examine the effect of pure bending on strand performance, observe filament cracking as a function of bending strain and strand type, and to evaluate cable performance at several transverse loads.

2.1.1. *Strand investigations*

Two test configurations were developed at MIT-PSFC to investigate the effect of bending on the critical properties of single Nb_3Sn superconductor strands. For the first configuration, the strands are reacted straight. Following reaction, the strands are mounted into grooved Ti–6Al–4V clamping fixtures which bend them to a pre-determined radius of curvature, with peak bending strains of up to 1.4% [5]. Preliminary results of the critical current versus fixed bending strain for various strands show significance differences in behavior. For some strands there is an initial increase in critical properties with bending, whereas other strands show deterioration of critical properties for even the smallest bending radius used for the study. Further work is needed to correlate the measured behavior with features particular to each internal strand configuration.

The second configuration seeks to examine strand behavior using the variable bending device shown in Fig. 1 [6]. The use of a variable bending device facilitates the search for irreversible changes in the critical property. Essential features of the variable bending experiments include: superconductor strand samples, which are reacted straight prior to mounting; a flexible support beam consisting of Ti–6Al–4V, which sets the radius of curvature during testing and supports the Lorentz loads on the strands; and a precision gear train, connected to a drive mechanism at the top of the test cryostat. Initial strand investigation produced questionable results due principally to poor sample mounting technique.

Microstructural examination of bent strand samples has been performed at ASC [7]. Sample polishing was carefully performed using a low-force automatic polisher, followed by 8 h of ion milling to eliminate any surface damage. Preliminary investigations revealed very little sign of damage up to peak bending strains of approximately 0.2%. A significant increase in crack density is typically observed at peak bending strains in the range 0.5–0.8%. Filament breakage generally begins on the tensile side of a sample and there is significant variation in the breakage pattern between strand types. So far, no direct correlation has been found between either the incidence or distribution of filament breaks in a strand and its observed critical current versus bending strain.

2.1.2. Sub-scale cable transverse load investigation

An apparatus was developed at MIT-PSFC to examine the effect of transverse loading on the critical current of a bare, sub-scale cable composed of 36 strands [8]. The cable uses a 3 × 3 × 4 cabling pattern, similar to that for lower stages of the ITER cables. The main components for the device include the 36 strand cable, which is wrapped in a single loop around a cylindrical, expanding collet and captured in a circular cross-sectioned groove in an external support ring, machined from Incoloy 908. The external ring reacts the transverse loads on the cable. A conical wedge located inside the internal collet can be driven axially to expand or contract the collet and mechanically

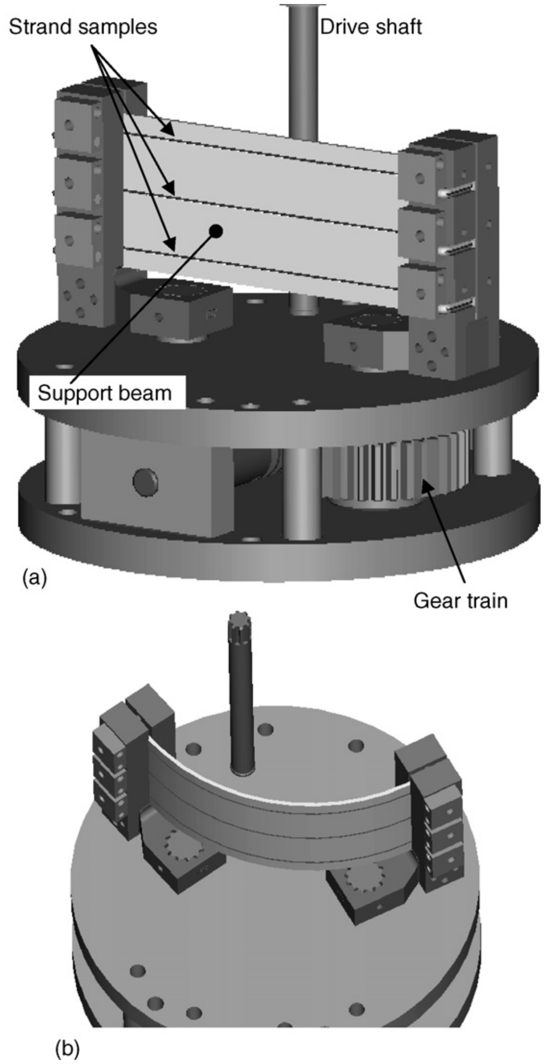

Fig. 1. Variable bending, strand test apparatus at (a) zero bending and (b) constant radius of curvature test positions.

increase or decrease the transverse load on the cable. Because of the relatively small size of the test cable, transverse loads for this experiment are generated by a combination of Lorentz loads and mechanical loads provided by the sliding wedge and collet. The apparatus can apply maximum transverse pressure to the cable up to 20 MPa, which brackets the design operation range for the ITER conductors. An initial experimental run of the test apparatus was performed at the National High Magnetic Field Laboratory during September 2005. A second experimental run is planned for early January 2006 to complete the intended range of test conditions.

2.2. Structural analysis for the ITER CS

Fig. 2 shows one of the ANSYS finite element models that was developed at MIT-PSFC to analyze the structural behavior of the pre-loaded CS modules. Time series outputs throughout the ITER plasma operation scenarios are used to quantify the maximum stresses, stress variations and stress intensity factors on the coil components, and to examine the effect of various design options on the resulting stress levels [9]. Several models of varying complexity are needed to quantify both the local and global behaviors of the CS stack and its individual component elements.

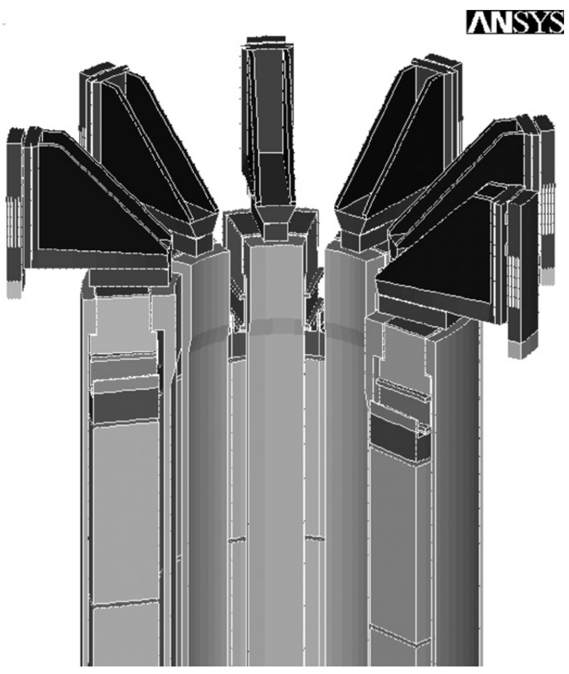

Fig. 2. Coarse mesh, 360° ANSYS model for the ITER CS with upper mounting structure.

The principal stress in each CS module is hoop tension, resulting from self-electromagnetic loading. The hoop tension is carried principally by the conductor jacket, with secondary contributions from insulation shear stresses. The jacket stresses vary cyclically during each plasma pulse, introducing a fatigue as a key concern for design verification. The turn-to-turn radial joggles, axial transitions between layers, transitions to the current feeders, and helium penetrations produce local peak stresses, which may require additional mitigation in the final CS design, depending on the results from the on-going analyses.

2.3. Characterization of jacket materials for the ITER CS conductor

The jacket for the CS conductor is required to sustain at least 60,000 full stress cycles during its projected lifetime. A fatigue estimate based on deterministic fracture mechanics was prepared at MIT-PSFC for potential ITER conductor jacket materials [10]. The analysis included all independent variables affecting fatigue crack initiation as well as fatigue crack growth. Several of the material parameters needed for accurate assessment of fatigue life were poorly defined at the start of the analysis. Thus, a supplemental materials test program was initiated to examine the static and cyclic performance at 4 K of candidate jacket materials, including Incoloy 908 and JK2LB. The test program revealed anomalously low, 4 K ductility following cold working and aging for the developmental alloy JK2LB, which renders it an unsuitable jacket material. Until this discovery, JK2LB had been considered a leading candidate for the CS jacket [11].

The critical current of Nb_3Sn CIC conductors is sensitive to strain, especially that due to the difference in coefficient of thermal expansion (CTE) between the cabled conductor and jacket. The most unexpected result from the Model Coil program was the significantly lower than expected performance for the CIC conductors jacketed with low CTE materials. A second unexpected result was the performance decrease observed for all conductors following repeated electromagnetic loading. To examine these two effects, a pair of sub-scale CIC samples was prepared and tested through collaborative effort of the US and European Union (EU) fusion programs [12]. The samples contained identical, 144-strand cables jacketed respectively in stainless steel (a high CTE material) and titanium (a low CTE material) and were electromagnetically cycled to loads comparable to those needed for the ITER coils. Results from this comparative study indicate that although the performance of a CIC conductor jacketed with low CTE materials is significantly less than expected, a CIC with low CTE material typically provide about 30% higher critical current at ITER relevant operating conditions than the same cable jacketed with a high CTE material.

3. Superconductor technology for heavy ion fusion drivers

Accelerated beams of heavy ions are a promising driver for inertial confinement fusion. The high current experiment (HCX) program at LBNL was initiated to test ion beam dynamics using a single channel configuration [13]. Four, NbTi-based superconductor quadrupoles were developed for HCX. Two design approaches were considered, taking into account their future use as component modules in large beam focusing arrays. This consideration favors the use of a flat, racetrack shaped winding for each pole of the quadrupole. Superconductor magnet development for HCX took place at LLNL and AML. LBNL and MIT-PSFC provided design and testing support.

The magnets produced at LLNL used an innovative design with a record breaking engineering current density in the winding pack of 550 A/mm^2 at 7 T. Racetrack modules were wound, inserted into recesses in winding plates, preloaded to their operational load with ferromagnetic wedges, and then impregnated. The modules were then joined together and clamped inside a four piece iron yoke and welded stainless steel shell. The magnets produced by AML were wound from continuous seven strand superconductor cables, by placing the cable in pre-machined grooves in insulating support plates. The grooves were positioned to optimize the field profile and support the cable without turn-to-turn load accumulation. All four prototypes achieved their design operation current, however, the quadrupoles built to the LLNL design required few or no training steps and all LLNL quads reached the short sample limit [14].

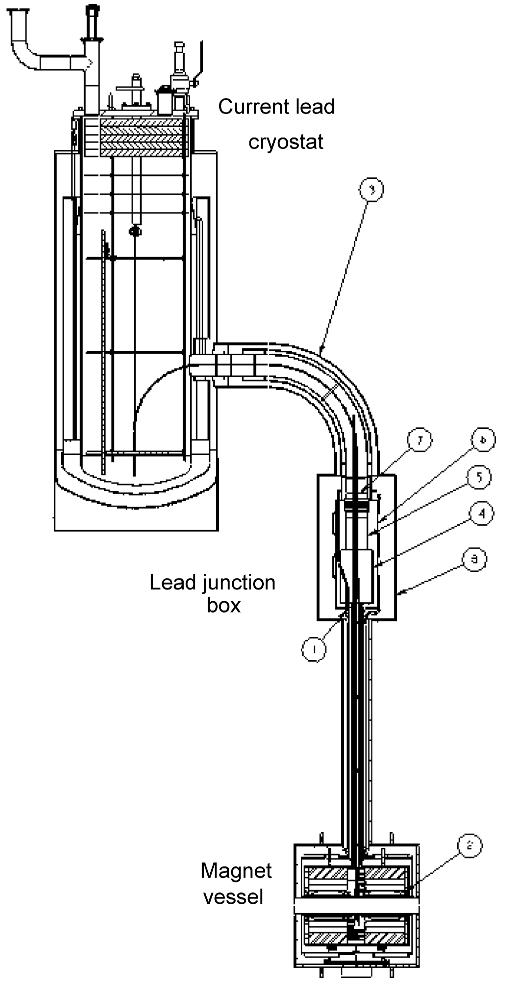

Fig. 3. Section view of cryostat to measure heat load on the prototype HCX quadrupole doublet beam focusing magnet.

The magnet prototype phase of the HCX program culminated in the design, construction and test of a cryostat housing a pair of the previously tested quadrupole magnets [15]. Fig. 3 shows a cross-section view of the cryostat. The cryostat consists of two main sections, an upper section equipped with a pair of 3 kA vapor cooled current leads that is connected through a small elbow to a lower section containing the quadrupoles. The purpose of the cryostat design was to permit precise calorimetric measurement of the heat load on the quadrupole doublet, especially that from its room temperature beam tube. The evaluated heat load on magnet assembly was less than 0.5 W, which was 50% of the design goal. The quadrupole doublet achieved critical current operation on the first attempt, but showed significantly higher ramp rate sensitivity that was observed during the previous, individual test of the quadrupoles.

4. Superconductor technology for LDX

The levitated dipole experiment (LDX) is an innovative confinement concepts plasma experiment designed and built by Columbia University (CU) and MIT-PSFC to investigate steady state, high-beta plasmas, with near-classical energy confinement [16]. Major factors that set LDX apart from earlier experiments are the absence of toroidal field coils, and its emphasis on magnetic flux expansion. Fig. 4 shows a cross-sectional view of the experiment. LDX consists of a 5 m diameter by 3 m tall vacuum chamber and three superconductor coils: a floating coil (F-coil) that provides the dipole field for plasma confinement; a NbTi charging coil (C-coil) that inductively charges and discharges the F-coil current; and a high temperature superconductor (Bi-2223) levitation coil (L-coil) that electromagnetically supports the weight of the 600 kg floating coil and controls its vertical position within the vacuum chamber.

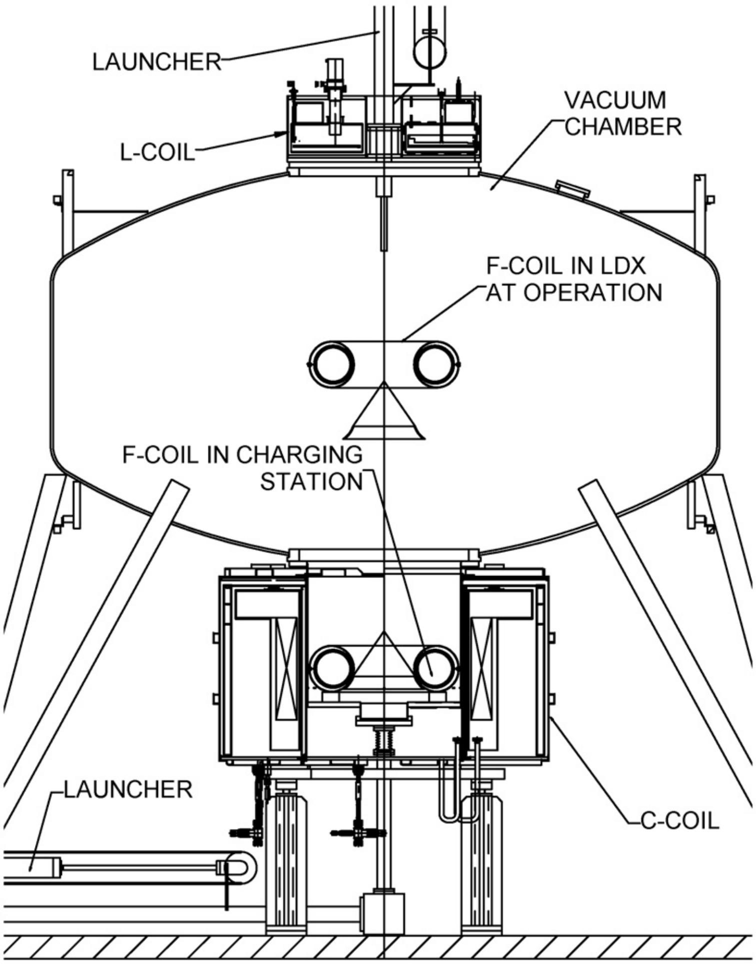

Fig. 4. Section view of LDX showing relative locations of the superconducting magnet systems.

4.1. LDX floating coil

The 600 kg, Nb_3Sn F-coil and cryostat, with on-board helium coolant, form the core of the LDX. Although the maximum magnetic flux density at the F-coil is only 5.3 T, Nb_3Sn conductor was selected because its relatively high critical temperature enables the coil to remain levitated for several hours as it warms from 4.5 to 10 K, without any external helium or current connections passing through the plasma volume. The electromagnetic, structural and cryogenic design of the F-coil was performed at MIT-PSFC. The coil was designed without current leads to minimize heat leak to the cryostat. A 2000 A maximum operation current was selected to minimize internal quench voltages.

The F-coil conductor consists of an 18 strand Nb_3Sn Rutherford cable that was reacted and then soldered into a structural copper channel before winding [17]. The strand for the cable was fabricated by Intermagnetics General Corporation (IGC). The cabling was performed at LBNL, followed by reaction heat treatment at BNL. The cable was returned to IGC and soldered into its half-hard copper channel. The performance of the cable-in-channel conductor was verified at BNL, followed by a slight reworking of the conductor and coil winding at Everson Electric Company under MIT-PSFC supervision. The strain state of the cable was continuously controlled during fabrication to minimize degradation of the critical current. The cryostat was fabricated and assembled around the F-coil at Ability Engineering under CU supervision.

4.2. LDX charging coil

A preliminary design for the 10 tonnes, 1160 mm warm bore, NbTi C-coil was performed at MIT-PSFC. Engineering design and manufacture of the helium-bath-cooled C-coil was performed at the Efremov Institute in Russia. The conductor for the C-coil contains three superconductor strands and six copper strands which were cabled together and then highly work hardened at the Bochvar Institute in Moscow to provide the necessary strength for the C-coil conductor. To start an experimental run, the F-coil is located in a charging port at the bottom of the LDX vacuum vessel. The C-coil is charged first to full current. The temperature of the F-coil is then reduced to below its superconducting transition temperature with the C-coil current held constant. The C-coil is discharged, inducing current into the F-coil. The F-coil is then mechanically raised to the center of the LDX vacuum vessel for plasma operation.

Initial, reduced current operation of the C-coil in LDX began during July 2004 and the first plasma experiments with inductively charged F-coil began during August 2004. By charging the C-coil to increasingly larger currents during successive test campaigns, the C-coil's maximum 400 A current was realized in July 2005. During the first year of plasma operation, the F-coil was operated in a supported mode, in which the coil is suspended from its lifting fixture at the center of the vacuum vessel. Preparations are presently underway for the next major phase of LDX operation, during which the L-coil will electromagnetically support the F-coil during plasma operations.

4.3. LDX levitation coil

The L-coil is a 2800 turn, 1.3 m outer diameter, double pancake winding that is mounted on top of the LDX vacuum vessel. For top levitation, the F-coil is stable to tilt and horizontal displacements, reducing control requirements for off-axis motions. The vertical position of the F-coil during levitated operation will be sensed optically and controlled within $a \pm 1$ cm band near the mid-plane of the vacuum vessel by varying the voltage applied to the L-coil. The LDX L-coil was the first coil in the US fusion program to use high temperature superconductor. Approximately 7330 m of three-ply narrow BSSCO-2223 tape was provided for the L-coil by American Superconductor Corporation under an OFES Phase II Small Business Innovative Research program grant.

The electromagnetic, structural and cryogenic design of the L-coil and L-coil cryostat was performed at MIT-PSFC. A new design code, LEVITATOR, was specifically written at MIT for the design of the L-coil [18]. The purpose of the code was to select the conductor configuration, operation temperature, numbers of turns, pancakes, and layers, and the inner and outer radius for the coil. According to analysis, the single largest heat load on the coil is magnetization hysteresis losses in the HTS conductor. The estimated conductor loss due to the estimated ac position control current ripple was roughly 10 W based on the assumption of full penetration of both the axial and radial field into the conductor. To minimize demands on the LDX operators, the L-coil was built as a conduction cooled coil, with cooling provided by a Cryomech AL-230 cryocooler that provides 25 W heat removal at 20 K.

The L-coil and cryostat were constructed at the Everson Electric Company under MIT-PSFC supervision. The coil and cryostat were tested as a stand alone system during May 2003. The coil and cryostat are installed on top of the LDX vacuum vessel in preparation for final integration into the LDX control scheme.

5. Summary

The implementation of superconductor devices in the US fusion program is a multidisciplinary activity that requires electrical, thermal, fluid, metallurgical, and structural design, verification and engineering. At present, superconducting devices comprise a small fraction of the total US fusion effort. The continued success of on-going superconductor activities coupled with the need for steady-state devices providing sufficient fusion power gain for commercial scale power plants will inevitable lead to increased dependence on superconductor technology for future US fusion program activities.

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