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PPPL- 4885

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June, 2013

Prepared for the U.S. Department of Energy under Contract DE-AC02-09CH11466.

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Systems Analysis Exploration of Operating Points for the Korean Demo Program

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***Abstract*—The Korean DEMO program is pursuing a steady state tokamak configuration to develop a fusion energy producing facility. Systems analysis is performed to determine its geometry and operating space available. After the plasma major radius and elongation is chosen, and the maximum toroidal magnetic field at the coil is established, the operating space can be explored with a range of assumptions. A database approach for the systems analysis is used that generates a large number of solutions, that can be used to examine sensitivities and parameter uncertainties.**

Keywords—tokamak; demonstration power plant; systems analysis; fusion physics and engineering

I. INTRODUCTION

The Korean DEMO (KDEMO, demonstration power plant) project is pursuing a tokamak facility to develop the technical basis for fusion power production [1]. The combination of the KSTAR steady state high performance plasmas and ITER burning plasma physics, and technology development, will provide the needed basis for the KDEMO. In addition, the Korean technology program will develop advanced Nb_3Sn superconducting coils, capable of reachng a maximum field of 16 T [1]. The purpose of this study is to determine the plasma geometry and operating space for this facility, and identify sensitivities to motivate needed research in plasma physics and engineering. The plasma power and particle balance is solved given a series of inputs, providing a large database that is passed through engineering and inboard radial build assessments. Filters are used to identify the most relevant solutions and the corresponding operating space can then be determined.

II. IDENTIFICATION OF PLASMA AND DEVICE GEOMETRY

Systems analysis is used to identify possible configurations for the KDEMO device. The plasma power and particle balance is solved given a series of inputs including R (major radius), A (aspect ratio), B_T (toroidal field), q_{95} (edge safety factor), κ (elongation), δ (triangularity), α_T (temperature profile), α_n (density profile), n/n_{Gr} (Greenwald density ratio), τ_p^*/τ_E (global particle confinement time ratio), η_{CD} (current drive efficiency), f_{imp} (impurity fractions), and Q (fusion plasma gain). These parameters are scanned over a given range, generating large numbers of viable physics operating points that satisfy the balance equations [2]. Several quantities are determined including the helium content, the energy and particle confinement and current diffusion times, bootstrap current fraction, radiated powers from bremsstrahlung, line and cyclotron, profile peak to average values, fast particle β, effective charge, current drive and heating powers, fusion power, and plasma stored energy. The benefits of evlauatng many operating points in this approach is that all dependences are immediately available. Optimization algorithms only provide the final point, and are not effective in providing information on sensitivites or the impact of uncertainties. This database is then passed through an engineering algorithm that determines heat fluxes to the first wall and divertor, power components and power balance to generate electricity, first wall, blanket, shield/support, and vacuum vessel radial builds on the inboard, the toroidal field (TF) coil build, bucking cylinder or other TF superstructure build, and central solenoid build. These depend on a series of inputs, some from detailed analysis, e.g. neutronics blanket and shield thicknesses, and others that carry significant uncertainty, e.g. radiated power fraction in the divertor. Only the inboard radial build is determined since it is limiting to the configuration, while the top, bottom and outboard builds are not. This analysis will eliminate several viable physics operating points because they will not satisfy engineering limits. The surviving points are then filtered for the desired features such as net electric power output, β_N, n/n_{Gr}, q_{div}^{peak}, H_{98} (energy confinement scaling), or Q_{engr} engineering gain.

KDEMO will have two primary phases, with a low (~200 MW_e) and a high (~600 MW_e) electric power. These imply lower or higher plasma perfomance to reach the associated fusion powers, characterised by β_N, H_{98}, Q, and $\langle N_w \rangle$ (ave neutron wall loading). In order to determine the major radius regime that can access both these conditions, while remaining within reasonable physics and engineering limits, a large scan was performed. The major radius was varied from 5.5-7.5 m, toroidal field from 5.5 to 7.5 T, Q from 5-35, β_N from 2.5-4.5%, q_{95} from 3.5-8.0, f_{Ar} = 0.1-0.3%, and n/n_{Gr} from 0.8-1.25. The aspect ratio is fixed at 3.25, τ_p^*/τ_E = 5, li = 0.70, η_{CD} = 0.175, α_T = 1.7, and α_n = 1.7. In the engineering module the assumptions are that the neutron multiplication is 1.1, 3% of fusion power is used for pumping, 3% of gross electric power is used for all other subsystems, heating and current drive wall plug efficiency is 0.4, thermal conversion efficiency is 0.35, and 90% of pumping power is recoverable. A Fundamenski formulation [3] is used for the power scrape-off width in determining the peak heat flux in the divertor, typically giving 3-4.5 mm for these cases, and up-down symmetric double null is assumed. The inboard radial build, tentatively determined by separate analysis is, 10 cm for the scrape-off layer, 3.1 cm for the first wall, 86 cm for the blanket and shield, 10 cm for the support structure, 13 cm for the vacuum vessel, about 26 cm for various gaps (assembly gaps, thermal insulation, etc.) between components, and 8 additional cm to reach the winding pack in the TF coil where the maximum toroidal field occurs. For the present studies 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 plasma will have its current driven completely non-inductively, so that the central solenoid is sized to only provide the flux swing in the current rampup.

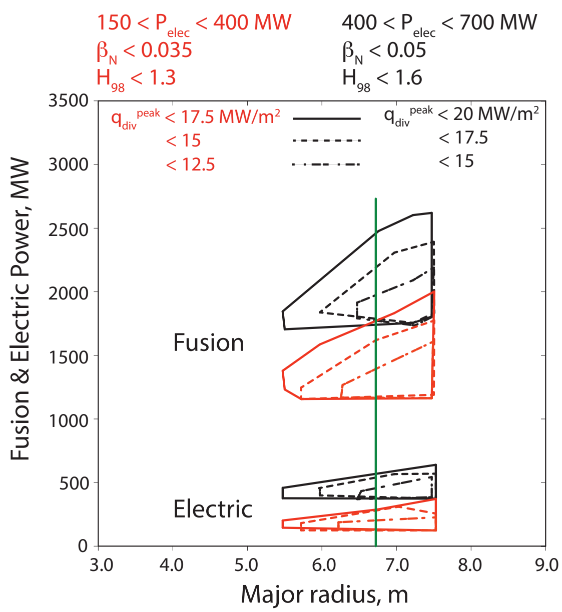

Figure 1. Fusion and electric power versus plasma major radius, with black for high power and red for low power phases. Solid contours show highest divertor peak heat flux, dashed is medium, and dash-dot is lowest. R = 6.75 m is chosen as the lowest major radius allowing access to the lowest heat fluxes.

Utilizing filters for electric power (Phase I) of 150-400 MW, $\beta_N^{total} < 3.50$, $H_{98} < 1.3$, and $q_{div}^{peak} < 17.5$ MW/m^2, along with $f_{rad,div} = 90\%$, configurations with major radii as low as 5.5 m were found. When reducing the allowed divertor peak heat flux to 12.5 MW/m^2, the smallest major radii rises to 6.25 m. A significant number of solutions could be found with major radii from 6.5 to 7.0 m. It should be noted that there are always more solutions at larger radius and the 7.5 m limit is due to not scanning above this value. At the lowest divertor heat flux and toroidal fields at the TF coil of 14.5-16 T, plasma toroidal fields were 7.0-7.5 T, plasma currents were 10.5-12.2 MA, β_N^{th} from 2.5-2.75%, q_{95} were 7.5-8.0, n/n_{Gr} at 1.15-1.25, Q all at 10, H_{98} at 1.25-1.3, auxiliary powers (for heating and current drive) from 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 powers were 150-200 MW$_e$. The operating spaces are shown in Fig. 1, for both the low and high electric power results, with varying divertor peak heat flux values.

Utilizing filters for the electric power of 400-700 MW, $\beta_N^{total} < 5.0$, $H_{98} < 1.6$, and $q_{div}^{peak} < 20$ MW/m^2, along with $f_{rad,div} = 90\%$, configurations with major radii as low as 5.5 m were again found. When reducing the allowed divertor peak heat flux to 15 MW/m^2, the smallest major radii rises to 6.5 m. A large number of solutions could be found if the major radius included the range 6.75-7.5 m. At the lowest divertor heat flux and toroidal fields at the TF coil of 14.5-16 T, plasma toroidal fields were 7.0-7.5 T, plasma currents were 11.7-13.0 MA, β_N^{th} from 3.25-3.5%, q_{95} were 7.0-8.0, n/n_{Gr} at 1.25, Q at 17.5-20, H_{98} at 1.53-1.6, auxiliary powers (for heating and current drive) from 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 powers were 400-520 MW$_e$. The larger net electric power requires larger fusion power and somewhat larger plasma radii than the lower power cases. Higher thermal conversion efficiencies could allow smaller major radii, although these were difficult to justify based on present helium gas cooling demonstrations, and the fact that KDEMO would be a first of a kind facility. Based on multiple factors, including the desire to keep the facility cost low, it was concluded that a plasma major radius of 6.8 m, and a maximum toroidal field at the plasma of 7.4 T would become the baseline. Fig. 1 indicates this choice relative to the operating spaces accessible. The plasma elongation was also chosen to be 2.0 in order to avoid very small operating space. 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. This choice for the major radius would result in lower net electric power in the high power phase, < 600 MW, but this was considered a sufficient level for demonstration of reliable power production to utilities.

Additional observations from the systems analysis scans included that the plasma elongation at the X-pt should not fall below 2.0, or the viable operating space of solutions would be severely reduced. The plasma density was always above the Greenwald density, which has been demonstrated experimentally [4] with core fueling, but is not routinely accessed in experimental operations, especially with high non-inductive current fraction. The plasma major radius required for significant fusion power was aided by the higher allowed toroidal field parameters compared to restricting solutions to ITER CS/TF parameters, 13 T at 14 MA/m^2 for the central solenoid or 11.5 T at 12 MA/m^2 for the toroidal field coil.

III. Operating Space for Parameter Variations at Fixed Geometry and Toroidal Field

The plasma geometry is fixed at R = 6.8 m, a = 2.1 m, κ_x = 2.0, δ_x = 0.63, and B_T = 7.4 T at R. The inboard radial build is left the same as before, although it is being refined, and will be updated as appropriate. It is desirable to determine the operating space associated with this configuration to understand the impact of assumptions in plasma and engineering systems. In order to address this, high resolution scans of the β_N, q_{95}, temperature and density profiles, n/n_{Gr}, Q, and f_{Ar} were performed. Fixed parameters include plasma geometry, li, τ_p^*/τ_E, η_{CD}, η_{th}, η_{aux}, $f_{rad,div}$, P_{pump}, P_{sub}, and $\langle j_{TF} \rangle$ and $\langle j_{CS} \rangle$.

A reference operating space is identified where the fusion power ranges from 850-2250 MW, and the net electric power ranges from 50 to 600 MW_e. Fig. 2 shows the low and high power operating spaces combined in fusion power versus net electric power. Viable operating points are inside the red contour, and the neutron wall loading is shown as an independent x-axis. These solutions are filtered by $q_{div}^{peak} < 15$ MW/m^2, $\beta_N^{total} < 4.25\%$, $H_{98} < 1.6$, $2.0 < T_0/\langle T \rangle < 2.75$, and $1.0 < n_0/\langle n \rangle < 1.5$. The density weightd volume average temperature ranges from 13-28 keV, while the volume average density ranges from $0.6\text{-}1.2 \times 10^{20}$ /m^3. The Greenwald density ratios span 0.8-1.25, with energy confinement multipliers (H_{98}) range from 1.05-1.6. Neutron wall loads range from 0.75-2.4 MW/m^2, q_{95} from 5.25-8.0, Ip from 11-16 MA, Q and Q_{engr} from 7.5-30.0 and 1.2-3.3, bootstrap current fraction from 0.45-0.95, and peak divertor heat fuxes from 5-15 MW/m^2. Table I provides parameters for one point in the lower electric power and one in the higher power regions, also marked on the plot in Fig. 1.

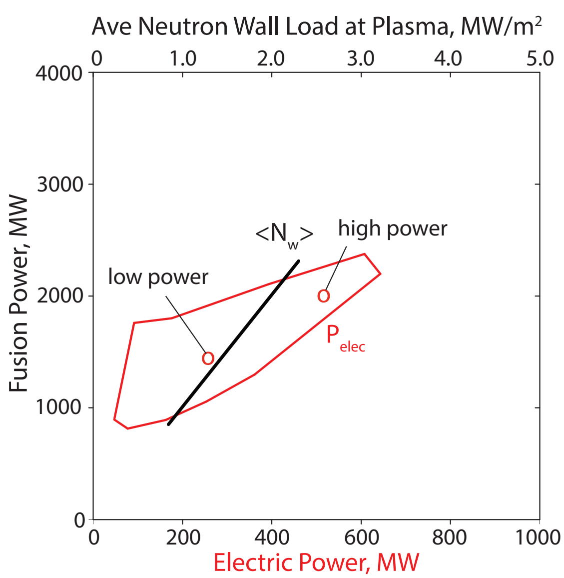

Figure 2. Fusion power versus electric power, and versus averge neutron wall loading, for combined high and low power operating space, showing the individual operating points listed in Table I. This serves as the reference operating space.

In Fig. 3, the low power operating space is separated out and shown by black contours, recall that the red contours include both low and high power operating points. This filtering is done by limiting the total normalized beta (thermal plus fast) to less than 3.25% and the global energy confinement multiplier to less than 1.3. From the black contours the maximum electric power has dropped to 350 MW_e, and the correspnding fusion power range has shrunk to 1000-1750 MW. In addition, the operating space for both the total and low power regimes are shown when the peak heat flux on the divertor is restricted to 10 MW/m^2 or less, compared to 15 MW/m^2 for the reference. The maximum electric powers and fusion powers tend to decrease in order to reduce the power load to the divertor, however electric powers are still significant even for the low power regime with > 100 MW_e.

The uncertainty in the divertor heat flux is significant, in spite of intense experimental research on tokamak facilities around the world [5-7]. Estimates of the power scrape-off width can range from sub-millimeter to centimeters, which would lead from nearly impossible engineering to straightforward design solutions. The integrated solution for the divertor will require focused research in the present experimental devices, the long pulse devices like KSTAR, and ITER.

Table I. KDEMO parameters at low and high electric power operating points.

	P_{elec} = 254 MW	P_{elec} = 496 MW
R, m	6.8	6.8
a, m	2.1	2.1
κ_x	2.0	2.0
δ_x	0.63	0.63
Ip, MA	12.3	12.7
B_T, T	7.4	7.4
B_T^{max}, T	16	16
β_N^{th}, β_N^{total}, %	2.53, 2.95	3.25, 3.92
q_{95}	7.25	7.0
n/n_{Gr}	1.15	1.2
Q, Q_{engr}	12.5, 1.70	23.8, 2.68
H_{98}	1.29	1.6
$T(0)/\langle T \rangle$	2.10	2.56
$n(0)/\langle n \rangle$	1.44	1.44
f_{BS}	0.67	0.83
Z_{eff}	2.04	1.50
$\langle N_w \rangle$, MW/m^2	1.49	1.97
q_{div}^{peak}, MW/m^2	11.9	13.2
λ_q, m	0.0044	0.0042
P_{aux}, MW	119.0	83.0
P_{brem}, MW	45.6	46.5
P_{cycl}, MW	44.5	95.3
P_{line}, MW	22.3	8.0
P_{fusion}, MW	1488	1966

The thermal conversion efficiency, fusion plasma gain, and recirculating electric power requirements are critical parameters in defining the efficiency in electric power production. As indicated in Sec. II, there are fixed assumptions for the wall plug power for heating and current drive, pumping power, and subsystems power. A demonstration power plant will need to show the efficient operation of relevant blanket concepts, moving away from the conventional approaches of water at low temperature (ITER), toward helium cooling and high temperature. Projected thermal conversion efficiencies for such systems [8] can be as high as ~ 45%, however first of a kind systems would likely be lower, since they would lack the experience and optimization. Fig. 4 shows the combined low and high power operating space for four values of the thermal conversion efficiency, 0.25, 0.30, 0.35, and 0.40. All other power terms are fixed at values given in Sec II. As the efficiency decreases the accessible electric powers decrease and the fusion powers shift to higher values consistent with the need to generate more fusion power to produce a given amount of electric power. The benefits of enhancing the thermal conversion efficiency are signifcant, and although scans are not shown for the recirculating power components, they have a similar impact. The heating and current drive recirculating power typically dominates in tokamak power plants. Although the current drive efficiency in the plasma is set by physics, and may be enhanced by optimizing source types and launching locations, the source, transmission, and coupling components require more focused research to raise the wall plug efficiency of such systems.

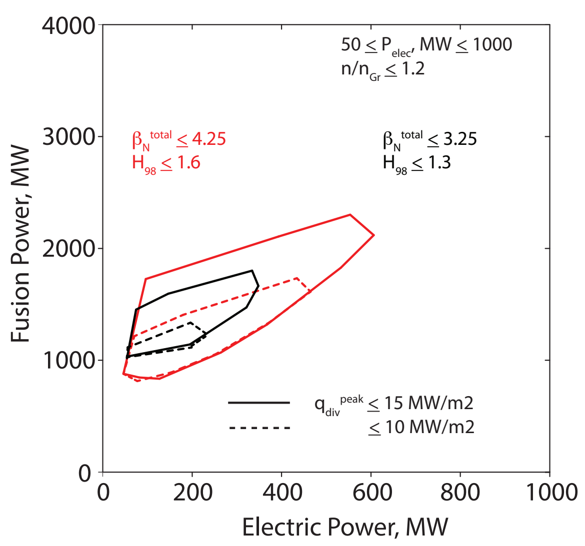

Figure 3. Fusion power versus electric power for low power regime and the combination of high and low power regime, with 15 MW/m^2 and 10 MW/m^2 divertor peak heat flux limitations.

The capability of radiating power in the divertor is critical to exhausting the combination of alpha particle power from fusion reactions and the auxiliary power injected into the plasma, the combination of which can range roughly from 350-450 MW. KDEMO is assumed to be an up-down symmetric double null, so that a divertor structure exists at both the top and the bottom of the device. The peak heating power in the divertor is determined by a simple model [9] which assumes that the conducted power inside the power scrape-off width is preserved as it enters the divertor and strikes the divertor target, but the power in that channel is reduced by the assumed radiated power fraction. The reference assumption is a high radiated power fraction of 90%, which requires more simulation and experimental examinations to confirm its accessibility self-consistently with a stable divertor regime and high perfromance steady state core plasmas. The impact of reducing this fraction on the operating space is shown in Fig. 5, where the fraction of power entering the divertor that is radiated to the target and side walls is scanned from 0.7 to 0.9.

With the constraint that the peak heat flux be less than 15 MW/m^2 the reduction in operating space with reduced radiated power fraction is strong. ITER assumes a radiated power fraction of 70% with a partially detached regime (strong reduction in temperature near the divertor target and along strike flux line). In Fig. 5 the apparently small operating space for 70% radiated power shows a range of solutions with Ip = 11.5-13.2 MA, f_{BS} = 0.56-0.74, n/n_{Gr} = 0.8-1.15, H_{98} = 1.5-1.6, Q = 8.8-18.8, $\langle N_w \rangle$ = 0.85-1.12 MW/m^2, β_N^{total} = 2.64-2.95% and q_{95} = 6.75-8.0. Reducing both fusion power and auxiliary power is required to reduce the divertor heat load since the geometry factors are fixed. Establishing high radiative divertor configurations is a high priority for all tokamak power plant configurations.

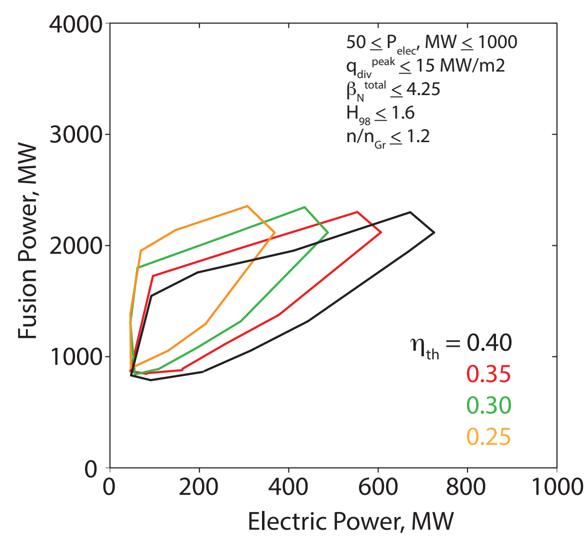

Figure 4. Fusion power versus electric power showing the operating space under varying thermal conversion efficiencies, demonstrating the change in accessible electric powers and required fusion powers.

Operation at values of the Greenwald density ratio, n/n_{Gr}, larger than 1.2 (not shown) was found to expand the operating space under all constraints, gives access to lower divertor heat fluxes, and higher fusion plasma gains. With central fueling (pellets) it is expected that surpassing the Greenwald limit will be possible, as demonstrated on present day tokamaks, however raising the scrape-of layer density simultaneously may cause the same edge cooling seen on present tokamaks with fueling dominantly from the edge. Consistency experiments integrating the divertor regime, the high performance core, and the high density regime will be necessary to develop confidence in such operating conditions.

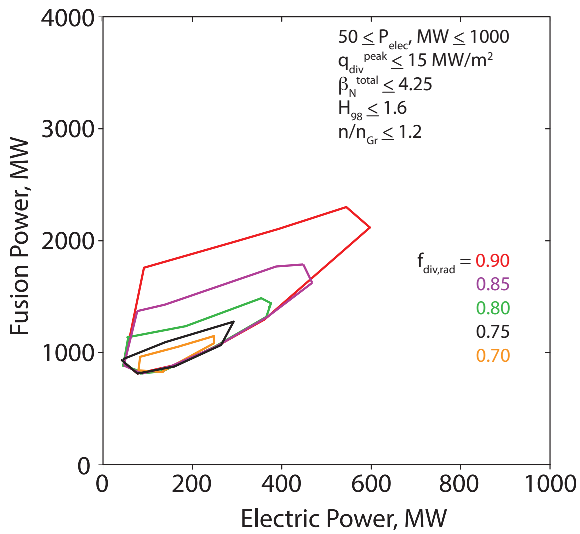

Figure 5. Fusion power versus electric power operating space, varying the fraction of power entering the divertor that is radiated to the target and side walls.

IV. Conclusions and Future Work

Initial systems analysis of the KDEMO facility have been performed, based on input from detailed magnet engineering, preliminary neutronics blanket build, and configuration analysis. The operating point geometry has been defined to provide the lowest size facility giving access to significant operating space for both low electric power (Phase I) and a high electric power (Phase II) phases. The major radius is R = 6.8 m, a = 2.1 m, κ_x = 2.0, and δ_x = 0.63. In addition, the maximum toroidal field on the TF coil is taken to be ~ 16 T, making the reference toroidal field at the plasma center ~ 7.4 T. This is the target of advanced Nb_3Sn superconducting magnet research in Korea. It is found that within reasonable assumptions electric powers of at least 200 MW_e can be produced in Phase I, and electric powers up to 450-600 MW_e can be produced in Phase II with improved plasma beta and energy confinement.

Virtually all configurations have high plasma density, approaching or exceeding the Greenwald density, and this regime must become more routine on experimental tokamaks to confirm operation of high performance steady state plasmas at such densities. High radiated power fractions in the divertor are required to disperse the combination of alpha and injected power from the plasma. Divertor regimes like this are not well established experimentally, and increased emphasis on both experiments and simulations are needed. Operating at higher normalized beta can allow a more compact tokamak with higher fusion power, and research into the requirements to access such regimes needs to continue.

The engineering of fusion power producing facilities is also subject to a number of uncertainties. The thermal conversion and heating and current drive wall plug efficiencies contribute significantly to the facility size and power flow in the plant. Higher steady heat flux capability in the divertor can also expand the available operating space, although more accurate loading descriptions are required to optimize such systems. The continued development of low temperature Nb_3Sn superconductor will enhance the accessible operating space and magnet reliability. As the KDEMO device is defined in greater detail, systems analysis will continue to produce more accurate configurations and more critical R&D can be defined.

Acknowledgments

This work was performed under contract with the National Fusion Research Institute of Korea and Princeton University, and the DOE under DE-AC02-09CH11466.

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