Burning Plasma Simulation and Environmental Assessment of Tokamak, Spherical Tokamak and Helical Reactors

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Abstract. Reference 1-GWe D-T reactors; tokamak TR-1, spherical tokamak ST-1 and helical HR-1 reactors, are designed using PEC (Physics Engineering Cost) code, and thier plasma behaviors with Internal Transport Barrier (ITB) operations are analyzed using TOTAL (Toroidal Transport Analysis Linkage) code, which clarifies the requirement of deep penetration of pellet fueling to realize steady-state advanced burning operation. In addition, economical and environmental assessments were performed using extended PEC code, which shows the advantage of high beta tokamak reactors in COE and the advantage of compact spherical tokamak in lifetime CO_2 emission reduction. Comparing with other electric power generation system, the cost of fusion reactor is higher than that of fission reactor, but on the same level of oil thermal power system. The CO_2 reduction can be achieved in fusion reactors same as in the fission reactor. The EPR of high-beta tokamak reactor TR-1 could be higher than that of other systems including fission reactor. These systematic design and comparative simulation analyses on both tokamak and helical reactors can be done by the help of the above two codes.

1. Introduction

For realization of future attractive fusion reactors, integrated design assessments are requisite to search for a compact, economic and environmentally reliable reactor system. Within toroidal magnetic confinement systems, comparative design studies of tokamak, spherical tokamak and helical systems are helpful to search for an optimized steady-state high-beta reactor. The comparative system studies using the PEC (Physics, Engineering and Cost) code [1-2] have been done so far by adopting the simple equivalent design constraints for tokamak and helical reactor systems.

In this paper, we define three reference reactors by simplified design assessment including plasma energy balance, coil, blanket and wall engineering issues through PEC code. One-dimensional plasma transport simulations coupled with two- or three-dimensional equilibria have been described in section 3 to reconfirm advanced plasma operations with internal transport barrier (ITB). The economical and environmental analyses with respect to cost of electricity (COE), CO_2 emission amounts and energy payback (profit) ratio (EPR) are given in section 4, and the summary and discussions are in the final section.

2. Reference Parameters of Tokamak, Spherical Tokamak and Helical Reactors

The design assessments of toroidal magnetic fusion reactors have been performed using PEC code, and related physics, engineering and cost results have been published so far [1-2]. As for physics modeling, the reactor plasma performance can be determined by beta and density limits. Moreover, confinement scaling laws are utilized for checking the burning plasma operation regimes. Because of the difference in the plasma confinement scaling and the density limit scaling laws, optimal operational regimes of both systems are found to be different. Optimal access to the ignition regime depends on the detailed radial plasma profile, which requires one-dimensional plasma transport analysis coupled with two- or three-dimensional high-beta equilibrium configuration effects.

As for engineering design, important design parameters to realize compact reactors are the

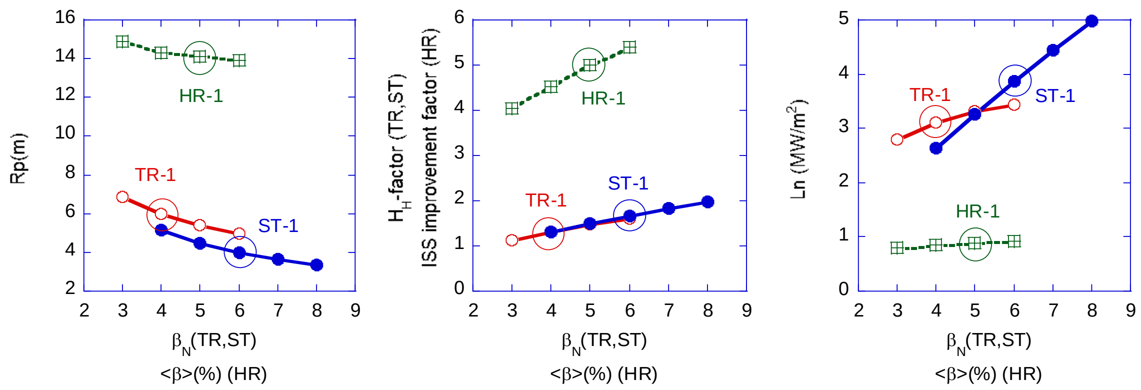

FIG. 1 Parameter dependence of TR-1, ST-1 and HR-1
as a function of normalized beta β_N and averaged beta <β>.

maximum magnetic field strength, the blanket and shield thickness, the wall loading, and so on. Comparative assessments of tokamak and helical reactors have been carried out with constraints of magnetic field strength of 13T and the inboard-side blanket-shield thickness of 1.2 meters. The blanket-shield thickness is a critical parameter determining the inboard-side radial build of reactor. In this paper, we assume this thickness $t_{blanket} + t_{shield}$ as a function of neutron wall load L_n as follows;

$$t_{blanket} + t_{shield} = 0.1L_n + 0.8 \quad \text{(m)} \tag{1}$$

The ratio of $t_{blanket}$ and t_{shield} is assumed depending on blanket designs; Flibe/FS, Li/V and LiPb/SiC. In this paper we focus on the LiPb/SiC design only, and defined typical three reference plant designs; tokamak reactor TR-1, spherical tokamak reactor ST-1 and helical reactor HR-1 with 1GW net electric power output.

In this paper, maximum field strength of superconducting coil is assumed 13 T, instead of maximum normal conductor field strength of 7.4T in ST-1 reactor. The tolerable neutron wall fluence is assumed 20MWYr/m^2 in the case of LiPb/SiC blanket system, which determines the replacement cycle of blanket modules. In Fig.1, designed plasma radius, required confinement improvement from ITER Elmy-H mode scaling [3] or ISS95 stellarator scaling [4] laws, and neutron wall load are given as a function of normalized beta for TR-1 and ST-1 or averaged beta value for HR-1. Helical reactor might become larger and higher confinement improvement is required. However, low neutron wall load is helpful for adopting

TABLE I: TYPICAL REFERENCE REACTOR DESIGN PARAMETERS

Parameters *:input	Tokamak TR-1	ST ST-1	Helical HR-1
R_p / a_p*	**3.06**	**1.62**	**5.7**
R_p /< a_p>*	**2.50**	**0.87**	**(7.8)**
T_0 [keV] *	**30**	**30**	**20**
<β>[%] *	**(5.3)**	**(22.6)**	**5**
β_N*	**4**	**6**	**-**
ellipticity κ*	**2.0**	**3.5**	**2.0**
triangurality δ*	**0.5**	**0.5**	**-**
B_{max} [T] *	**13 (SC)**	**7.4 (NC)**	**13 (SC)**
Electric Power[GW] *		**1.0**	
F_{wall} [MWYr/m^2] *		**20**	
Thermal Efficiency (%)		**50**	
Availability (%)*		**75**	
Operation Period (Yr) *		**30**	
R_p [m]	**5.97**	**4.00**	**14.0**
a_p [m]	**1.69**	**2.46**	**-**
<a_p> [m]	**2.39**	**4.62**	**2.1**
<n_e> [10^{20}m^{-3}]	**1.43**	**1.02**	**0.97**
$n_{e,crit}$	**1.50**	**1.20**	**1.17**
B [T]	**6.03**	**2.46**	**4.16**
I_p [MA]	**13.4**	**22.9**	**-**
f_{BS} [%]	**49**	**95**	**-**
τ_E [s]	**1.63**	**2.26**	**3.8**
H_H-factor	**1.31**	**1.67**	**-**
ISS improvement factor	**-**	**-**	**5.01**
P_{fusion} [GW]	**2.62**	**3.21**	**1.87**
P_α [GW]	**0.52**	**0.64**	**0.38**
P_{CD}[GW]	**0.12**	**0.01**	**-**
$L_{neutron}$ [MW/m^2]	**3.11**	**3.87**	**0.89**
Blanket Thickness [m]	**0.85**	**0.90**	**0.69**
Shield Thickness [m]	**0.36**	**0.39**	**0.30**
Wall Lifetime (Yr)	**4.6**	**3.7**	**16.0**

long-life-time blanket design [5]. ST-1 reactor becomes compact, but rather high H_H factor is required and frequent wall replacement should be performed because of short wall lifetime. The reference parameters of three 1GWe reactor designs, TR-1, ST-1 and HR-1, are shown in Table I.

These assessments are done using zero-dimensional power balance with parabolic radial temperature and density profiles. The designs critically depend on the plasma profile. Especially, high-beta plasmas should be achieved based on ETB (edge transport barrier) or ITB (internal transport barrier) operations. Therefore one-dimensional analyses are required for accurate design studies.

3. Burning Plasma Simulation

We focus on the burning plasma control scenario to access to the ignition regime and to sustain the steady-state burn including radial plasma profile change. Here, 2 or 3-dimensional (2-D or3-D) equilibrium and 1-dimensional (1-D) predictive transport code TOTAL (toroidal transport analysis linkage) [6-8] for tokamak and helical systems has been used. In this case the profile effect on the ignition attainment is very important, and the transport analysis coupled with equilibrium analysis should be carried out.

3.1. Simulation Models

The fusion reactor power strongly depends on the radial profile of core plasma density and tempearture. Especially, plasma confinement improvement with internal transport barrier (ITB) formation, density control by the high-field-side (HFS) pellet injection and the bootstrap (BS) current fraction at high beta are crucial parameters for the evaluation of tokamak ignition condition . The neoclassical ripple transport with ITB is also important for helical reactors. We developed the TOTAL-T (Toroidal Transport Analysis Linkage - Tokamk) code coupled with GLF23 and NCLASS tokamak codes, and TOTAL-H (Helical) code with multi-helicity helical ripple transport analysis code for burning plasma simulation. The effectiveness of the ITB transport coefficient is checked using experimental data of JT-60U and LHD.

The TOTAL code is composed of tokamak part TOTAL-T including 2-D equilibrium code APOLLO [9] and helical part TOTAL-H including 3-D equilibrium code VMEC [10]. The transport equations are solved using the equilibrium flux coordinates at each simulation time step to get time evolution of radial plasma profiles. When the plasma beta is changed beyond certain percentage (typically >0.1 %), the self-consistent equilibrium is re-calculated at that time step in the simulation. In the helical plasma simulation code TOTAL-H, the neoclassical ripple transport has strong effect on the reactor operation, and ambipolar electric field and multi-helicity magnetic components in high beta equilibrium are included. In the tokamak code TOTAL-T, the external current drive, bootstrap current, sawtooth oscillation, ballooning mode and the neoclassical tearing mode (NTM) analysis with the modified Rutherford equation are included [11]. In both codes, the steady-state burning plasma operation is achieved by the feedback control of density fuelling and external heating power control. The impurity dynamics [12] is also included in both codes.

In TOTAL code, ITB operations can be simulated by several transport models: simple local transport reduction model, GLF23 model, current diffusive ballooning mode, Bhom/GyroBhom mixed model with ExB shear flow effect and so on.

In this paper we used Bohm/GyroBohm mixed model [13,14] relevant to ion temperature gradient (ITG) mode with ExB shear flow stabilization and adopt a thermal diffusion coefficient χ in the form

$$\chi_{e,i} = \chi_{neoclassical} + \chi_{anomalous}, \qquad (2)$$

$$\chi_{anomalous} = \alpha_1 \times \chi_{Gyrobohm} + \alpha_2 \times \chi_{Bohm} \times F\left(\omega_{E\times B} / \gamma_{ITG} \right), \tag{3a}$$

or

$$\chi_{anomalous} = (\alpha_1 \times \chi_{Gyrobohm} + \alpha_2 \times \chi_{Bohm}) \times F\left(\omega_{E\times B} / \gamma_{ITG} \right), \tag{3b}$$

where transport reduction factor is given by

$$F\left(\omega_{E\times B} / \gamma_{ITG} \right) = \frac{1}{1 + \left(\omega_{E\times B} / \gamma_{ITG} \right)^2}. \tag{4}$$

The above coefficients α_1 and α_2 are determined using typical experimental data of JT-60U H-mode data and LHD e-ITB operations. Equation (3a) is adopted for tokamak data fitting, and Eq.(3b) is for helical data treatment.

Using this model, the analyses have already been carried out focusing on high-field-side pellet injection effects to make easy access to ITB regime [8]. The neoclassical tearing mode and impurity transport are also investigated using this TOTAL code [11-12].

3.2. Tokamak Reactor Simulation

In reactor systems, starting from low density low temperature plasmas alpha particle power is feedback controlled by the adjustment of both heating power and gas puffing rate. Even if the electric power output is same in both reactor designs, the required tokamak alpha particle power should be greater than helical one, because the current drive power is required in tokamak systems, especially in spherical tokamak system. The higher temperature operation is feasible in this tokamak design TR-1 than in the helical reactor HR-1.

Figure 2 shows the typical reversed shear burning tokamak operation with ITB formed by the deep penetration of high-field-side (HFS) pellet injection without unfavorable neoclassical tearing modes. The shallow pellet penetration of the low-field-side injection did not lead to the ITB formation. The radial profile of plasma parameters at t=100s are also shown in Fig.2. The ITB in this reversed shear case is formed around at ρ~0.5 where there is q-value minimum in this model analysis.

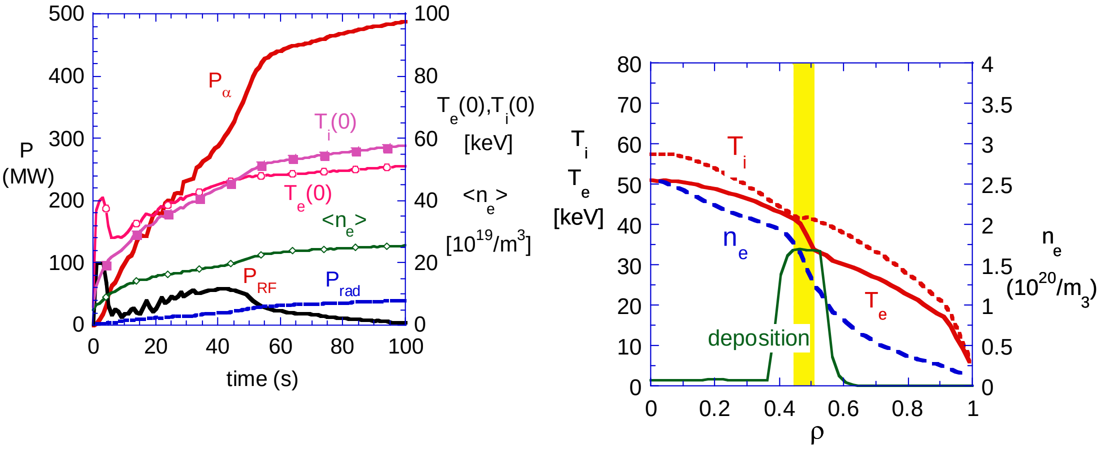

FIG.2 Start-up operation (left), and radial profile at t=100 s (right) of TR-1 reactor

In the tokamak reactor, the NTM decreases the temperature and make the 1GW electric power production impossible. The detailed analysis of these effects has been shown in Ref [11]. The impurity effects are also evaluated [12] using this TOTAL code.

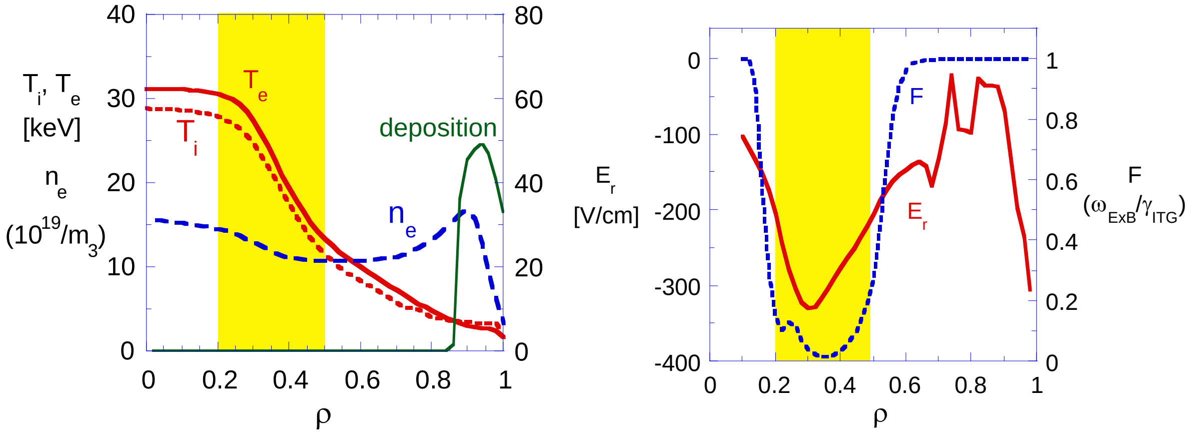

FIG. 3 Radial plasma profile of ITB operation in helical reactor HR-1.
(Left): density and temperature profiles, and pellet deposition profile
(Right): ambipolar radial electric field and transport reduction factor defied in Eq.(4)

3.3. Spherical Tokamak Reactor Simulation

For spherical tokamak reactor, the bootstrap current fraction is one of key parameters and is calculated in TOTAL code. The n-infinity ideal ballooning mode beta limit is calculated by Apollo code [9] and the operation simulation for ST-1 is confirmed in the 1.5-D simulation like TR-1 plasma.

3.4. Helical Reactor Simulation

In helical plasmas, neoclassical transport is a key to access to the ignition regime. In this system, the radial electric field can reduce neo-classical ripple transport loss and might reduce anomalous loss. In this paper, we simulate advanced plasma confinement of helical reactor using the ExB shear stabilization model by the ion-root negative electric field. The evaluation has been done using LHD e-ITB data [8,13].

Figure 3 shows typical simulation results of HR-1 plasma. Different from tokamak plasmas, the stabilizing ExB shearing effects are derived from ambipolar electric field due to neoclassical ripple transport. In addition, the HFS pellet injection effects are not expected in helical system, and rather edge fuel deposition might be performed eve in high speed pellet injection. However, the ITB-like structure is realized in the present simulation model.

The features obtained by the system code and zero-dimensional energy balance code, are re-confirmed by these 1.5 and 2.0 transport analyses. In this simulation, rather low beta equilibria are assumed, but the high beta multi-helicity effects on helical plasma confinement are large, and it is impossible to ignite plasmas in the present scale reactor including high beta equilibrium and ripple transport effects.

4. System Assessments

For the engineering design of DT reactors, blanket thickness, maximum magnetic field strength and neutron wall loading are crucial for determining the reactor size. In the code, four blanket designs; Li/V, Flibe/FS(Ferritic Steel), LiPb/SiC, FF(Fission-Fusion) Hybrid, can be evaluated in three type reactors. In the present analysis, high-thermal-efficiency LiPb/SiC blanket is mainly considered. Other blanket designs are evaluated and published somewhere in the future. Economic and environmental assessments are performed evaluating cost, CO2 emission and energy investment on several tens of reactor components using input-output table[15,16]. The obtained beta dependences of COE, CO2 emission and EPR are summarized in Fig.4 for 1 GWe power reactors with high heat efficiency (50%) LiPb/SiC blanket system.

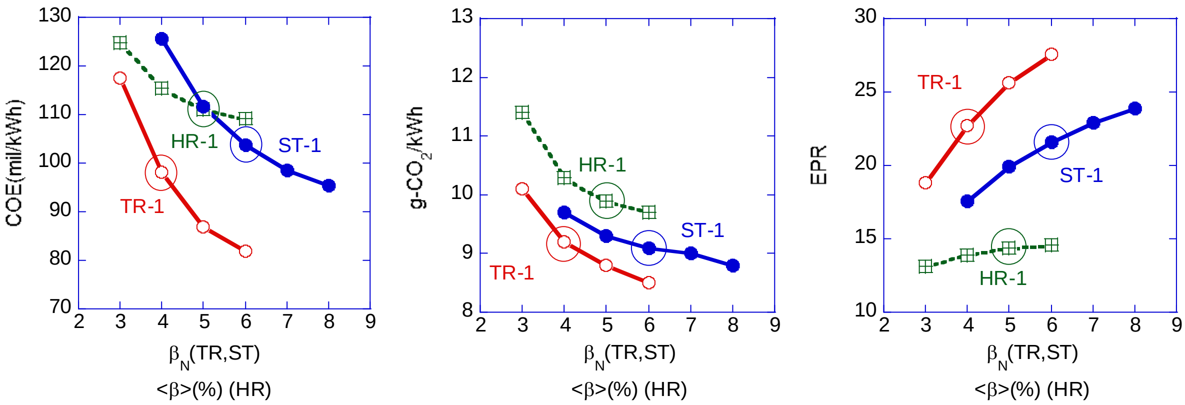

FIG. 4 COE, CO2 emissions and EPR of three-types of reactors as a function of beta vale.

4.1. Economical Assessment and Cost of Electricity (COE)

The beta dependence of COE is shown in Fig.4. The strong dependence on COE in Tokamak-type reactor is shown here, but in helical reactor rather week dependence is obtained.

The fusion island (FI) cost of ST-1 is lowest. However, its required fusion thermal power is largest and the TR-1 is superior in cost of electricity (COE) as shown in Fig.4 and Table 2. The BOP cost and CO2 emission in HR-1 is rather small but total COE and CO2 emission rate is not relatively small.

Among four blanket designs, Flibe/FS is superior in cost, because ferritic steel (FS) is much cheaper than vanadium (V). Fusion-Fission hybrid blanket has high neutron energy multiplication ratio, and it is possible to be constructed in relatively low cost.

TABLE II: COST, CO_2 EMISSION AND ENERGY ANALYSIS FOR THREE REACTOR DESIGNS GIVEN IN TABLE 1.

Parameters *:input	Tokamak **TR-1**	ST **ST-1**	Helical **HR-1**
< Cost [M$] >			
Fusion Island	**1056**	**1065**	**1823**
Balance of Plant	**1583**	**1817**	**1400**
Total Capital Cost	**5112**	**5582**	**6239**
< COE [mil/kWh] >			
capital cost	**75.6**	**82.5**	**92.7**
operations	**13.0**	**14.4**	**11.0**
fuel	**0.04**	**0.04**	**0.04**
replacement	**8.89**	**6.18**	**6.9**
decommissioning	**0.6**	**0.6**	**0.6**
Total COE	**98**	**104**	**111**
< CO_2 emission [kt] >			
Fusion Island	**288**	**129**	**439**
Balance of Plant	**628**	**692**	**577**
Construction Total	**926**	**820**	**1016**
< Rate [g-CO_2/kWh]>			
fusion island	**1.47**	**0.66**	**2.25**
balance of plant	**3.21**	**3.53**	**2.96**
operations	**3.16**	**3.16**	**3.16**
fuel	**0.24**	**0.25**	**0.45**
replacement	**0.34**	**0.74**	**0.26**
decommissioning	**0.78**	**0.78**	**0.78**
Total CO_2 Emission Rate	**9.2**	**9.1**	**9.9**
< Energy Investment [TJ] >			
fusion island	**2.2**	**2.2**	**3.1**
balance of plant	**10.8**	**11.4**	**18.2**
operations	**16.1**	**16.9**	**25.7**
fuel	**0.6**	**0.8**	**0.4**
replacement	**5.5**	**7.5**	**4.2**
decommissioning	**0.01**	**0.01**	**0.01**
EPR	**22.7**	**21.6**	**14.4**

4.2 Energy Analysis and Energy Profit Ratio (EPR)

The energy efficiency has been evaluated using Energy Profit Ratio, or Energy Payback Ratio (EPR) defined by the ratio of net energy output to energy input including energy related to material production, machine construction, operation, fuel, and decommissioning. Typical energy intensity used here is from input-output table [14,15]. The CO_2 emission intensity described in the next section are also shown in this table. The high-beta advanced TR-1 reactor might be better from the view point of EPR as shown in Table 2.

4.3. Environmental Assessment and CO_2 Emission Amount

Related to global warming, green-house gas emissions relevant to energy productions are serious problems. The life-cycle CO_2 emission amount per output electric power from fusion reactor is also evaluated in Fig.4. The ST-1 hig-beta reactor is favorable in CO_2 emission reduction, because rather compact and simple normal conducting coil system is adopted. The ST-1 and TR-1 need more frequent blanket exchanges than HR-1 with lower neutron wall load. However, HR-1 is still expensive and has low energy payback ratio (EPR) and higher CO_2 emission within the present evaluation model.

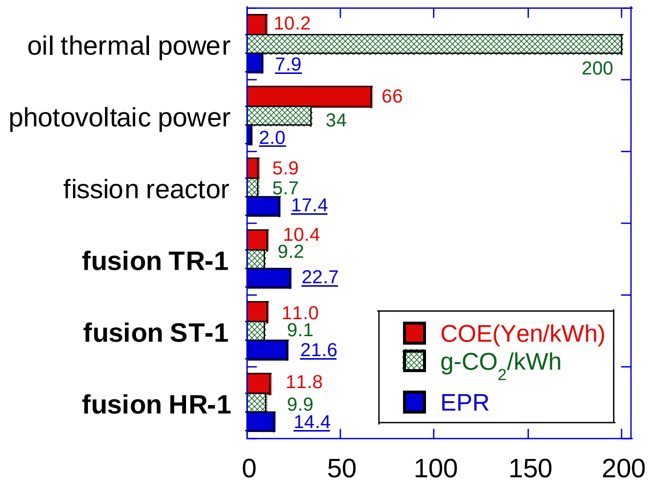

FIG. 5 Environmental comparisons among fusion reactors and other existing electric power plants which data is obtained from Ref.[15]

When we compare fusion power plants with other electric power plants [15], fusion reactors emit less CO_2 than fossil fuel thermal power plant and oversea Uranium enrichment fission reactors. In comparison with fission reactors with domestic Uranium enrichment, the fusion power plants have an advantage in EPR, but have disadvantages in COE and CO_2 emission.

The comparisons among three fusion reactors and several existing electric power generation systems in Japan are shown in Fig.5 related to COE, CO_2 emission and EPR. The exchange rate of US Dollar to Japanese Yen is considered as 1$ =106¥ based on the 2003 cost date in Ref.[2]. The cost of fusion reactor is higher than that of fission reactor, but on the same level of oil thermal power system. The CO_2 reduction can be achieved in fusion reactors like fission reactor. The EPR of high-beta tokamak reactor TR-1 could be higher than that of other system including fission reactor.

5. Summary and Discussions

System analysis of typical 1GW-electric Reactors, tokamak (TR-1), spherical tokamak (ST-1) and helical (HR-1) reactors were carried out using PEC code, and 1.5D or 2.0-D transport simulations are performed focusing on Internal Transport Barrier (ITB) operations using TOTAL (Toroidal Transport Analysis Linkage) code, which shows the requirement of deep penetration of pellet fueling to realize steady-state advanced burning tokamak operation in the present model. The advantage of high-beta tokamak reactors in COE and the advantage of compact spherical tokamak in lifetime CO_2 emission reduction are clarified in the economical and environmental assessments,.

The present system analysis assumes LiPd/SiC blanket system, and the assessment of other blanket including fission-fusion hybrid blanket will be published somewhere. In the comparisons with various existing power plants, thermal plant equipped with Carbon Capture and Storage (CCS) system and such advanced systems will be evaluated and effects of carbon tax on the COE of various electric power generation systems will be published in the near future.

References:

[1] YAMAZAKI, K., et al., "System assessment of helical reactors in comparison with tokamaks", 19th IAEA Fusion Energy Conference (Lyon, France, 14 - 19 October 2002) FT/P1-20.
[2] YAMAZAKI, K., DOLAN, T. J., "Impact of plasma, magnet and wall performances on tokamak and helical reactor economics", Fusion Engineering and Design **81** (2006) 1145-1149.
[3] ITER Physics Basis Editors et al., Nucl. Fusion **39** (1999) 2137.
[4] STROTH, U., et al., "Energy confinement scaling from the international stellarator database", Nucl. Fusion **36** (1996) 1063-1077.
[5] SAGAR, A., et al., "Conceptual design activities and key issues on LHD-type reactor FFHR", Fusion Engineering and Design, **81**(2006) 2703-2712
[6] YAMAZAKI, K., AMANO, T., "Plasma Transport Simulation for Helical Confinement Systems" Nucl. Fusion **32** (1992) , 633-644.
[7] YAMAZAKI, K., et al., "Burning plasma simulation of TR-1 tokamak and HR-1 helical reactors", Fusion Engineering and Design **81** (2006) 2743-2748.
[8] HIGASHIYAMA, Y., "Internal transport barrier simulation with pellet injection in tokamak and helical reactor plasmas" J. Physics: Conference Series **123** (2008)012032.
[9] YAMAZAKI, K., et al., "Ballooning beta limit of dee and bean shaped tokamaks", Nucl. Fusion 25 (1985) 1543-1554.
[10] HIRSHMAN, S.P., et al., "Three-dimensional free boundary calculations using a spectral Green's function method" , Comput. Phys. Commun. **43** (1986) 143-155.
[11] TAKAHASHI, Y., H-mode plasma transport simulation in ITER with effect of neoclassical tearing mode" J. Physics: Conference Series **123** (2008)0012036.
[12] YAMAZAKI, K., et al., "Simulation Modeling of Impurity Transport in Toroidal Fusion Plasmas", J. Plasma Fusion Res. SERIES, **7** (2006) 102-105.
[13] GARCIA, J., et al, "Analysis of Bifurcation Phenomena in the Electron Internal Transport Barrier in the Large Helical Device", Phys. Rev. Lett. **96** (2006) 105007.
[14] TALA, T., et al., "ITB formation in terms of $\omega_{E\times B}$ flow shear and magnetic shear *s* on JET", Plasma Phys. Control. Fusion 43 (2001) 507-523.
[15] Y. Uchiyama, "Life cycle analysis of power generation systems, Socio-economic Research Center", Y94009, (1995) (in Japanese)
[16] NANSAI, K., et al., "Embodied Energy and Emission Intensity Data for Japan Using Input-Output Tabled (3EID)", CGER-D031-2002 Report, Center for Global Environmental Research, National Institute for Environmental Studies, Japan (2002), http://www-cger.nies.go.jp/cger-e/db/d031_e/3eid_e.html