Conceptual Design of a Fast-Ignition Laser Fusion Reactor FALCON-D

T. Goto 1), Y. Ogawa 2), R. Hiwatari 3), Y. Asaoka 3), K. Okano 2, 3), Y. Someya 4), A. Sunahara 5), T. Johzaki 6)

1) National Institute for Fusion Science, Gifu, Japan
2) Graduate School of Frontier Sciences, The University of Tokyo, Chiba, Japan
3) Central Research Institute of Electric Power Industry, Tokyo, Japan
4) Graduate School of Engineering, Musashi Institute of Technology, Tokyo, Japan
5) Institute of Laser Technology, Osaka, Japan
6) Institute of Laser Engineering, Osaka University, Osaka, Japan

e-mail contact of main author: goto.takuya@LHD.nifs.ac.jp

Abstract. A new conceptual design of the laser fusion power plant FALCON-D (Fast ignition Advanced Laser fusion reactor CONcept with a Dry wall chamber) has been proposed. The fast ignition method can achieve the sufficient fusion gain for a commercial operation (~100) with about 10 times smaller fusion yield than the conventional central ignition method. FALCON-D makes full use of this property and aims at designing with a compact dry wall chamber (5~6m radius). 1-D/2-D hydrodynamic simulations showed the possibility of the sufficient gain achievement with a 40 MJ target yield. The design feasibility of the compact dry wall chamber and solid breeder blanket system was shown through the thermomecanical analysis of the dry wall and neutronics analysis of the blanket system. A moderate electric output (~400MWe) can be achieved with a high repetition (30Hz) laser. This dry wall concept not only reduces some difficulties accompanied with a liquid wall but also enables a simple cask maintenance method for the replacement of the blanket system, which can shorten the maintenance time. The basic idea of the maintenance method for the final optics system has also been proposed. Some critical R&D issues required for this design are also discussed.

1. Introduction

In the development of the fusion energy source, inertial fusion energy (IFE) plays an important role because it has quite different properties from magnetic confinement fusion (MCF) and can contribute to a flexible application of the fusion energy source. Thus it is required to carry out the research with the consideration of the difference and commonality between them. In the design of IFE reactors, one of the most critical problems is the protection of the reactor chamber wall from high energy, short pulse load by X-rays and charged particles. Then most of IFE commercial plant designs adopt a liquid wall concept, wall protection by a thin liquid metal layer, to reduce the size of the reactor chamber. However, such liquid wall concept has some difficulties. It is not easy to cover the wall perfectly with a liquid metal due to the properties of both materials (e.g., surface wettability, viscosity, etc.). Especially, at the ceiling or the edge of beam ports, some specific methods are necessary. The liquid wall also inevitably diminishes laser usability due to the deterioration of the chamber vacuum condition accompanied with the formation of metal vapor. Whereas a new method for the achievement of ignition and burn of the fuel, fast ignition, can reduce fusion yield about 10 times smaller than the conventional central ignition method. To make full use of this property, we have proposed a new design concept, FALCON-D (Fast-ignition Advanced Laser reactor CONcept with a Dry wall chamber), which aims at designing with a compact dry wall chamber. Such compact dry wall chamber has high engineering feasibility and enables a simple cask maintenance method, which can shorten the maintenance time. In this paper, core plasma analysis, dry wall and blanket system design, maintenance method, and some critical issues about each R&D element are reviewed.

2. Design Specification and Core Plasma Analysis

To realize a compact dry wall chamber, the fusion yield needs to be minimized to reduce heat and particle load on the wall with keeping sufficiently high fusion gain. A simple 0-D physics model predicted that fusion gain of 100 can be achieved with 400 kJ input energy (350 kJ for implosion, 50 kJ for heating). Then if we assume the use of tungsten as the first wall material, the design of a chamber with 5-6 m radius is considered to be possible. Table I shows main design parameters of FALCON-D based on these estimations. Moderate electric power (~400MWe) can be achieved by 30Hz repetition of the laser irradiation.

To examine the feasibility of such small yield, high gain target design, we performed implosion simulations by using 1-D hydrodynamics code ILESTA-1D [1]. Here we adopted a spherical-shell type cryogenic D-T fuel pellet surrounded by a plastic ablator layer. According to the prediction by the 0-D model, the implosion laser energy was fixed to be 350 kJ. FIG. 1 shows the density and ion temperature profiles of the compressed core obtained by the irradiation of the laser pulses shown in the small figure inside FIG.1 to the pellet with the initial aspect ratio A_0=4. It was shown that high density (>800 g/cc) and high areal density (~2 g/cm^2) of the core can be achieved by optimizing fuel/ablator thickness and laser pulse shape.

TABLE I: MAIN PARAMETERS OF FALCON-D.

Chamber radius [m]	5.64
Input energy [kJ] (implosion / heating)	400 (350 / 50)
Fusion gain	100
Target yield [MJ]	40
Average wall load [J/cm^2]	2.0
Laser repetition rate [Hz]	30
Plant fusion output [MW]	1200
Average neutron wall load [MW/m^2]	2.4

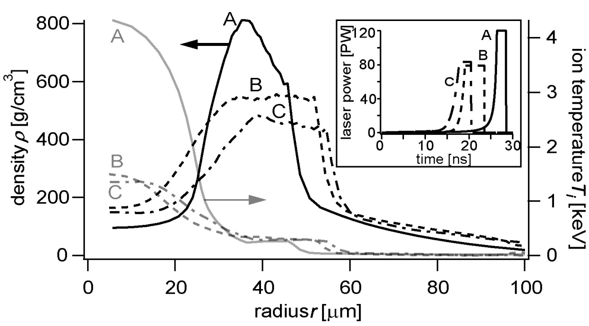

FIG. 1. Density and ion temperature profiles of compressed core and corresponding laser pulse shaping.

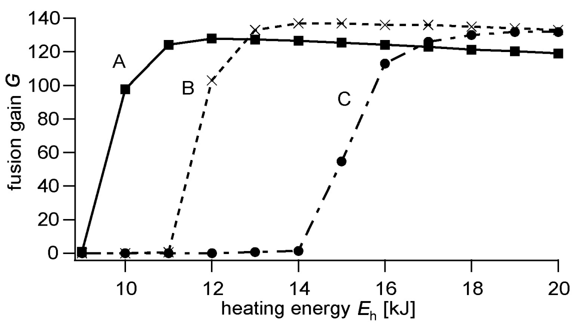

FIG. 2. Achieved fusion gain as a function of the core heating energy.

Then using these compressed core profiles, we carried out 2-D burn simulations. Here we assumed homogeneous electron heating over the cylindrical region (15 micron radius and the 1.8 g/cm^2 optical depth) at the core edge. The pulse width was set to be 10 ps. Here no use of a cone guide was considered to avoid the adhesion of the cone material on the dry wall. FIG. 2 shows the fusion gain as a function of the core heating energy. One can see that there is a threshold heating energy to achieve the high gain and it strongly depends on the core density profile. It was also found that the fusion gain around 100 can be achieved with the heating energy of 10 kJ (pulse shaping A). If 20% coupling efficiency from the heating laser to a compressed core (the same value experimentally obtained with a cone guide [2]) would be achieved, the heating laser energy of 50 kJ is sufficient for this design. Then fusion gain of 100 can be achieved with total laser energy of 400 kJ, i.e., 40 MJ fusion yield. To ensure the design of such small yield target, further engineering and physics studies are required, e.g., accurate injection and tracking of the pellet, effective heating of the core without a cone guide.

3. Design Feasibility Analysis of a Dry Wall Chamber System

3.1. Dry Wall Design

The dry wall chamber undergoes several threatening effects due to the high heat and particle load. Here we adopted ferritic steel covered with 1mm-thick tungsten for the first wall and water for the coolant, because this pair of structural material and coolant has good compatibility with each other and plenty of databases and experiences in nuclear power plants are available. For the early realization of FALCON-D, the use of reduced activation ferritic steel (F82H), which is a leading candidate of the structural material, is favorable. However, considering the relatively small thermal output of the FALCON-D as a power plant, it is useful to examine the supercritical water concept as a possible option. This supercritical water concept provides a harsher condition on the first wall because of its high temperature. Thus we carried out thermomechanical analysis assuming this high temperature.

Here we assumed 5.64 m chamber radius (coincides 2 J/cm^2 heat load). First we estimated the temporal temperature evolution by solving the 1-D thermal conduction equation. The energy deposition profile was estimated from the photoabsorption coefficient and the ion stopping range with using the spectra of X-ray and energetic ions obtained from the hydrodynamic simulation described in the previous section. Here the inlet temperature of the coolant was fixed to be 623 K, expecting a possibility of the supercritical water, and the thermal convection coefficient was assumed to be 20 kW/m$^2 \cdot$K. FIG. 3 shows the temporal temperature evolution during the first shot. In case of an irradiation with 30 Hz repetition, after 10 sec the surface temperature saturates around 1600 K, which is much less than tungsten melting point (3680 K) and the threshold temperature of surface roughening (2400 K). Next we performed mechanical analysis with using this result as a temperature load. The stress-strain relation was modeled by a simple bilinear approximation. In elastic region, temperature-dependent mechanical properties of the material were considered. In plastic region, there are no reliable data and then tangent modulus of tungsten was fixed to be 667 MPa. FIG. 4 shows the stress-strain behavior of the surface layer during the first 5 shots. The surface layer undergoes large plastic deformation in not only the heating phase but also the cooling phase. To avoid a fatigue due to this cyclic plastic deformation, the enhancement of the yield stress is indispensable. A highly-engineered material, e.g., ultra fine-grained tungsten (UFG-W) [3] can be a candidate, because the yield stress of UFG-W is much higher than that of the standard tungsten. In addition, by taking the fast strain rate (~10^4/s) into account, the yield stress of UFG-W can reach to 3 GPa at room temperature [4], whereas ~1.2 GPa for the standard tungsten. Then the first wall design with no plastic deformation could be available.

The irradiation of energetic particles also causes other threatening effects. Especially, blistering and exfoliation due to helium accumulation is quite severe. Experimental study in the HAPL project [5] indicated blistering occurs at the helium fluence of 10^{21}/m^2 and exfoliation occurs at 10^{22}/m^2. Assuming that the layer with the thickness coincides to 3.5 MeV helium range is lost when exfoliation occurs, the loss rate of the surface layer of FALCON-D reactor chamber can be a few millimeters or more per one year, which is totally unacceptable. Here the materials that have micro-structure [6] or fine grain can provide a solution, because it could drastically suppress the concentration of helium, resulting in the loss reduction due to the exfoliation. Experimental study showed UFG-W has high resistance to blistering [7] and it is a possible candidate for the first wall armor.

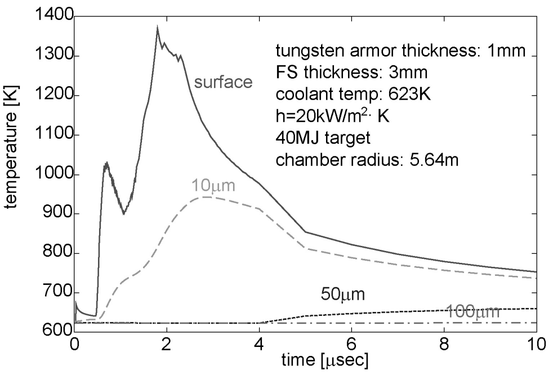

FIG. 3. Temporal temperature evolution during the first shot at various depths.

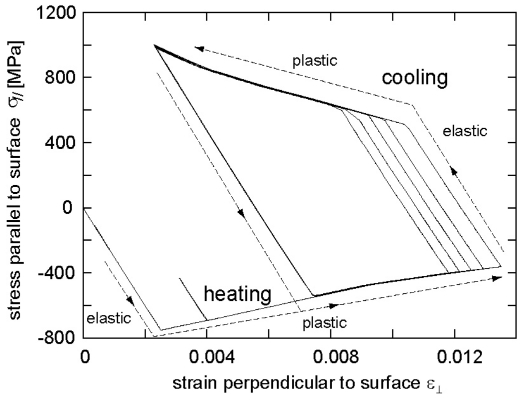

FIG. 4. Stress-strain behaviour of the surface of the first wall during the first 5 shots.

Since this design has relatively low wall load, other materials that have melting temperature greater than 2000 K can be a candidate for the armor material. There are few data of irradiation effect and thermomechanical properties in high temperature regime about such materials. Further experimental study is strongly expected to ensure this dry wall concept as a possible path to the design of laser fusion power plants.

3.2. Blanket Design

In the design of a power plant, it is quite important to show the feasibility of the blanket system. Thus we adopted water cooled solid breeder blanket using F82H, which has been adopted for the test blanket module (TBM) of ITER, for the first step analysis of the blanket system of FALCON-D. Taking accounts of a sufficient margin of tritium for a stable operation of the fusion power plants, the requirement of net tritium breeding ratio (TBR) has been estimated to be approximately 1.07 for the first generation of fusion commercial plants [8,9].

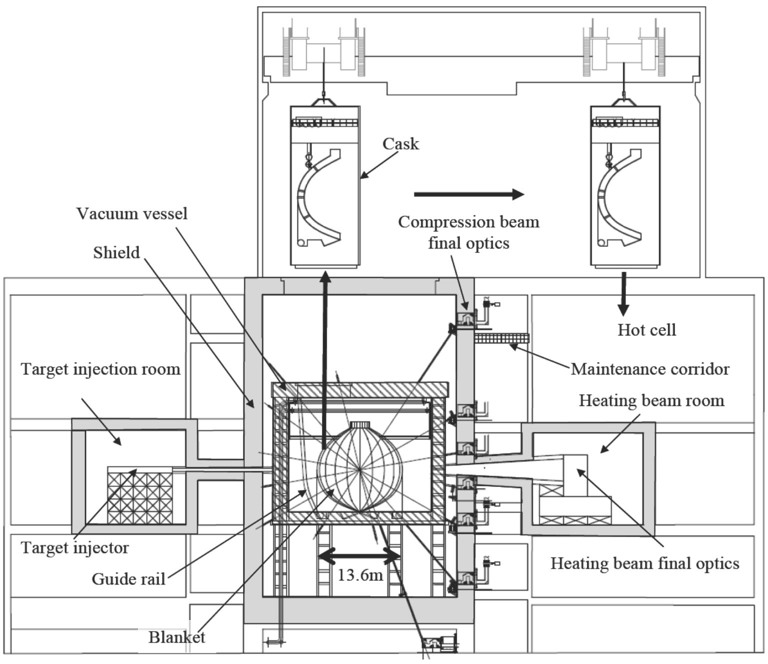

FIG. 5. Elevation view of FALCON-D and replacement process of the blanket sectors. The vacuum vessel and the neutron shield correspond to the hatched and the shaded region, respectively.

FIG. 5 shows an elevation view of the FALCON-D. The first wall and blanket system are divided into 10 sectors, and 31 implosion beam lines and the heating beam line go through the division between the blanket sectors. To protect toroidal field coils from the neutron and gamma-ray, the blanket system of the MCF power plant need to set radiation shielding. Whereas, in the case of FALCON-D, most of reactor core components except the final optics system of the heating beam are placed in the reactor room. Therefore, in case of FALCON-D, the reactor room wall serves as the radiation shielding.

In the neutronics analysis of FALCON-D, the three-dimensional Monte Carlo N-particle transport code MCNP-4C [10] with the nuclear data library ENDF/B-V [11] was used. The blanket system was modeled in spherical shape with a point source at the center of the chamber ($1.42x10^{19}$ neutrons per shot with the energy of 14.06 MeV). FIG. 6 shows the cross-sectional view of the blanket system. The blanket system consists of 4 Li_2TiO_3 pebble (90% Li-6 enrichment) beds, 3 Be pebble beds, cooling water panels and the structural material (F82H). It has 37.1 cm thickness and is located 564.0 cm from the center of the chamber. FIG. 7 shows the achievable initial local TBR and Li-6 consumption rate after 3.0 full power year (FPY) operations. TBR increases towards the both side of breeding layers, and Li-6 consumption is also large at there. The initial TBR was 1.20 in the breeding zone. After a period of 3 FPY, the TBR at the center of the breeding layers slightly increases as the consumption of the Li-6 at the both sides of the breeding layers increases. As a result, the decrease of TBR is about 0.3 % after 3.0 FPY operations. Therefore, assuming the blanket coverage ratio of 90 %, the net TBR of more than 1.07 is attainable.

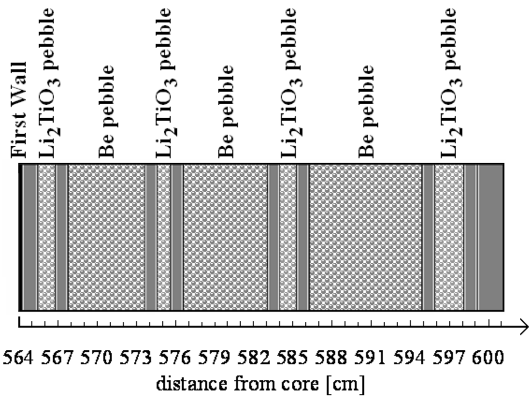

FIG. 6. Cross-sectional view of the blanket system of FALCON-D.

The cooling plumbing in the blanket system consists of 8 cooling panels of the equator direction. The temperatures of the breeding and multiplier materials are kept below these temperature limit; 900℃ and 600℃, respectively. The coolant pressure of the blanket system is selected to be 15 MPa. FIG. 8 shows the temperature distribution in the blanket system for the FALCON-D. As a result, inlet/outlet temperatures of the water coolant was 180/342 ℃, and total pressure loss was estimated to be 1.3 MPa. Moreover, the weight of the blanket system per a sector was estimated to be 39 ton. This blanket weight enables the blanket maintenance method described in the following section.

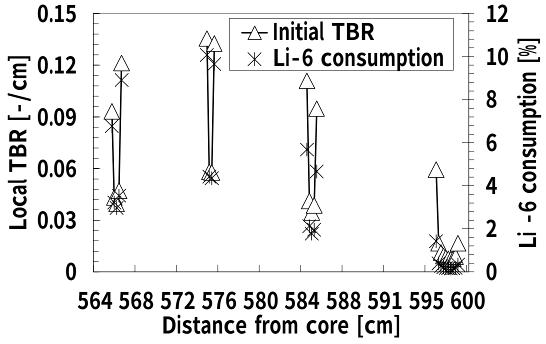

FIG. 7. Local TBR and Li-6 consumption distribution of the FALCON-D blanket system.

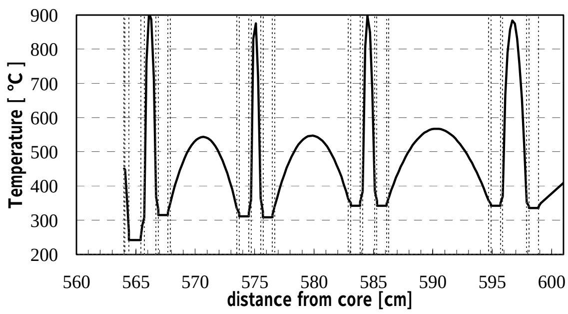

FIG. 8. Temperature distribution of the FALCON-D blanket system.

4. Maintenance Approach

4.1. Importance of maintenance for FALCON-D

The blanket system of FALCON-D has to be replaced periodically. Generally, there are two methods for the blanket replacement approach: one is to use a remote manipulator arm with a maintenance rail as in ITER (in-situ method) and the other is to use a large cask with a large maintenance port (cask method). The latter concept is favorable from the view point of the achievement of high plant availability. To apply the cask method to FALCON-D, the path space wide enough for the large blanket sector to pass is needed to be kept between the multiple beam lines and target injection line.

In addition, the final optics of 32 laser beams are directly exposed to the high energy neutrons. To alleviate the irradiation damage, the long distance from the reactor core to the final optics is required. For example, KOYO-Fast has 20 m and 30 m distance from the reactor core to the final optics for implosion beam and heating beam, respectively [12]. On the other hand, a grazing incidence metallic mirror (GIMM), which prevents the final optics from the direct expose by the high energy neutron, has also been proposed in the HAPL program [13]. Nevertheless, the final optics is supposed to be damaged by the neutron irradiation, and the easy replacement method of the final optics is required and essential.

4.2. Maintenance method of blanket system

The blanket system, divided into 10 sectors, is extracted and installed through the upper large port. Correspondingly, 10 maintenance ports exist on the upper side of the vacuum vessel. The cask accesses to those maintenance ports, and the door of the port opens to the inside of the cask together with the cask door, after the cask door is attached to the port door. In FIG. 5, there is no beam duct inside the vacuum vessel so that the blanket sector is easily lifted up into the cask. This is why the vacuum vessel is separated from and is located outside the blanket system. Then 10 blanket sectors do not have to be completely sealed each other. Therefore each blanket sector can be extracted, just after the cutting of connections for the heat transfer and tritium recovery system by a remote handling device. To avoid the sector’s shaking during extraction, the guide rails are located corresponding to each blanket sector. Each blanket sector has a joint on its bottom side to stand by itself.

The layout of implosion beam lines corresponds to 32 lines between the center and the vertex or the center of gravity in each surface of the regular icosahedrons (20-polyhedron). There should be the beam injection from the upper pole (corresponding to the north-polar position) to the reactor core, but this beam line is probably an obstacle to the maintenance. In the theory, more than 32 laser beam lines improve the laser uniformity for target implosion [14]. However, the fast ignition method does not require the complete laser uniformity, which is required for the central ignition method. This may enable the removal of the upper pole beam line, and this beam line is eliminated in FALCON-D. Consequently, there is no intersect of the implosion beam line on the upper side of vacuum vessel, and maintainability is improved.

4.3. Maintenance method of final optics system

FIG. 5 shows the locations of the final optics for the implosion and heating beams. The locations of the final optics for implosion beams are divided into 6 groups by the polar angle of the beam lines, and we proposed the build-in style module in the shield for them. To access the final optical modules, 6 access corridors are placed along to the shield, and all final optical modules for implosion beam is withdrawn and replaced through those corridors by a remote handling device. There is neutron streaming through the final optical module, and the total neutron flux just behind the final optical module is estimated at 2.4×10^{7} n/(cm$^{2}\cdot$s). This is ten times less than that just behind the shield of the tokamak reactor concept such as CREST [9] and the remote maintenance system considered for several tokamak reactor concepts is also applicable to the replacement of the final optical module for implosion beam of FALCON-D.

The shortest distance from reactor core to final optics is about 15 m near the equatorial plane of the reactor core, and the neutron flux above 100 keV in front of that final optical module is 1.55×10^{13}n/(cm$^{2}\cdot$s). The final optics of implosion beam is considered as transmission diffraction grating made of quarts as proposed in KOYO-fast [15]. According to the design criterion of the diagnostics window for the ITER made of quarts [16], the limit of the fast neutron fluence is assumed to be 3.0×10^{20}n/cm^{2}. If we assume this value as the limit under the condition that irradiation damage by low energy neutrons would be recovered by annealing, the final optics must be replaced every 6 months. In order to keep one FPY operation without outage for maintenance, we propose a sliding changer concept in the final optics module shown in FIG. 9, which has two set of final optics. This method could be adapted for the replacement of one FPY cycle in the final optics system.

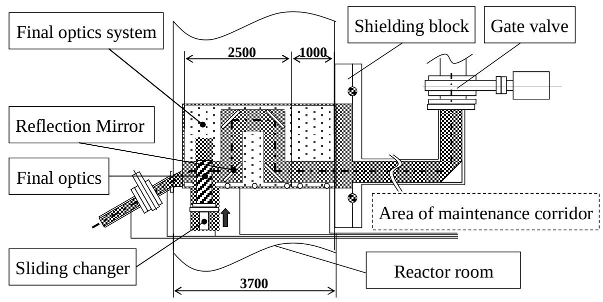

FIG. 9. Configuration of the final optics system for FALCON-D.

The final optics of the heating beam is located at 30 m distance from the reactor core in the heating beam room of FIG. 5. That final optics is also directly exposed to the neutron. In addition, the target injector room is in the same condition as for neutron streaming. Hence, the heating beam room and the target injection room also require the neutron shield as shown in FIG. 5. The neutron flux at the entrance of the heating beam room is estimated at $2.25\text{x}10^{11}$n/(cm^{2} s). The remote handling device is also required for the replacement of final optics of the heating beam. The radiation dose rate in the heating beam room is supposed to be higher than that of the maintenance corridor for implosion beams due to this large neutron flux. This fact implies that the handling device for the final optics of the heating beam probably needs to be equipped with the highly radiation resistant system, and the replacement time of the final optics of the heating beam is restricted to as short period as possible. In order to figure out the effective maintenance method for the final optics of the heating beam, the detail design of final optics and the effect of neutron damage on the heating performance have to be made clear.

The neutron load on the blanket system is about 2 MW/m^{2}, and the blanket sectors have to be replaced by every several years. On the other hand, the final optical modules have to be replaced every year at least in the present FALCON-D concept. Hence, the lifetime of the final optics for both implosion and heating beams determines the required maintenance frequency and the plant availability, and these analyses remain to be done in the near future.

5. Concluding Remarks

A new concept of the laser fusion reactor that utilizes the property of the fast ignition has been proposed. Numerical simulation by 1-D/2-D hydrodynamics codes showed the possibility of the achievement of the sufficient fusion gain (~100) with a 40 MJ fusion yield by optimizing the pellet design and the laser pulse shaping. Thermomechanical analysis and considerations about the effect of energetic particle irradiation showed that the design of a compact dry wall chamber (radius of 5-6 m) is possible with the use of some highly-engineered materials for the first wall armor. Neutronics analysis showed the feasibility of consistent design of the blanket system. This compact dry wall chamber enables a simple cask maintenance method. By separating the blanket system and vacuum vessel, the cask can access the upper port of the vacuum vessel without interference with multiple laser beam ducts.

This study indicated a design possibility of laser fusion power plants with a compact dry wall chamber. Of course there still remain several critical R&D issues to ensure such design concept; stable implosion and effective heating of the small fuel pellet without a cone guide, a large scale production and testing of the first wall armor, and further detailed consideration of the maintenance method. However, reactor design with a dry wall chamber and solid breeder blanket system has high engineering feasibility and gives a great impact on the laser fusion development. Thus further physics and engineering studies are strongly expected.

Acknowledgements

The authors would be appreciate Drs. K. Tanaka, Y. Ueda, T. Norimatsu in Osaka University, Dr. N. Yoshida in Kyushu University, Dr. H. Kurishita in Tohoku University, Dr. A. R. Raffray in UCSD, Dr. S. Sharafat in UCLA for giving valuable advices.

References

[1] TAKABE, H. et al., Phys. Fluids **31** (1988) 2884.
[2] KODAMA, R. et al., Nature **418** (2002) 933.
[3] KURISHITA, H. et al., J. Nucl. Mater. **367-370** (2007) 1453.
[4] WEI, Q. et al., Acta Materialia **54** (2006) 77.
[5] GILLIAM, S. B. et al., J. Nucl. Mater. **347** (2005) 289.
[6] SHARAFAT, S. et al., J. Nucl. Mater. **347** (2005) 217.
[7] KURISHITA, H. et al., J. Nucl. Mater. **377** (2008) 34.
[8] ASAOKA, Y. et al., Fusion Technol. **30** (1996) 853.
[9] ASAOKA, Y. et al., Fusion Technol. **39** (2001) 1018.
[10] BRIESMEISTER, J. F., “MCNP-A General Monte Carlo N –particle Transport Code, Version 4C”, LA-13709-M (2000).
[11] MAGURNO, A. et al., ENDF-201 Supplement 1, ENDF/B-V.2 Summary Documentation, BNL-NCS-17541, 3rd Edition (ENDF/B-V), National Nuclear Data Center, Brookhaven National Laboratory, Upton, Long Island, New York 11973, January 1985.
[12] KOZAKI, Y. et al., J. Plasma Fusion Res. **83** (2007) 19 (in Japanese).
[13] SETHIAN, J. D. et al., Nucl. Fusion **43** (2003) 1693.
[14] MURAKAMI, M. et al., J. Appl. Phys. **74** (1993) 802.
[15] MIYANAGA, N. et al., J. Plasma Fusion Res. **83** (2007) 3 (in Japanese).
[16] NISHITANI, T. et al., Fusion Eng. Des. **42** (1998) 443.