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Conceptual analysis of a thermonuclear power station with the tokamak type reactor

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Abstract
This paper concerns a technology of nuclear fusion reactors with magnetic containment of plasma in a tokamak and about using it in production of electricity. The reaction of the thermonuclear fusion is occurring in very high temperatures of the order of hundreds of millions degrees Kelvin. Such reactions are occurring in stars and produce the considerable quantities of energy. The closest natural fusion reactor is the sun in our Solar System. A power station based on a fusion reactor seems to be within reach of current technologies, however requires considerable efforts and the cooperation of many nations. Elements from which the thermonuclear power station will be built will have to be of high quality and of the precise execution. Thermonuclear reactor will consist of many parts which must be made appropriately and have a significant resistance to thermal fluxes and the neutron radiation. The analysis presented in this work regards production of electricity with applying the thermonuclear fusion. Most important system and elements that make up the tokamak type thermonuclear reactor are characterized. A few blends of fuel are considered for "burning" in the reactor, and most probable for the application is a blend of deuterium and tritium. A subject of the production of the tritium in cooling jackets with liquid lithium is brought up. Power station under consideration has a turbine cycle with steam as the working fluid. High-temperature plasma is the source of heat driving the power plant and the heat is being collected from the blanket of the reactor which surrounds plasma. Calculations were performed for the conceptual power station in three technological variants (different parameters of fresh and reheated steam). Matter of using the product after the thermonuclear reaction is raised. A review of a state-of-the-art technologies allows to conclude that the ITER is an important first step in thermonuclear fusion. Technology is available, clean, environment friendly and it would be very effective. Building fusion power plants requires outstanding technical capabilities, commitment and vision.

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1 Introduction

Almost 80% of the energy production at present comes from fossil fuels and their burning, it generates huge amounts of carbon dioxide (CO_2) which pollutes the environment, induces the greenhouse effects and extreme climate changes [6]. Energy from the so-called renewable sources is very dilute and its power per unit of surface used is low, e.g., from solar energy – (5–50) W/m^2, from hydro-energy about 11 W/m^2, from on-shore and off-shore wind – (2–3) W/m^2 , and from biomass – 0.5 W/m^2 only [6]. During the recent four centuries the world's population has grown more than ten times to the 7 billions people that live today on our planet. For the next fifty years the world's population is expected to grow from present state to a level of 12 billions, so energy consumption is expected to double or more. In this case it is vital to provide for future generations with safe energy sources. Nowadays people are already using nuclear fission to produce energy and by this method do not generate carbon dioxide, reducing global emission and pace of global warming. Production of nuclear waste and post-fission heat still remains a big problem of this energy source. Moreover there are limited resources of uranium for nuclear fission.

Thermonuclear fusion is the nuclear reaction (Fig. 2 contains important reactions) which takes place in stars and during the reaction an enormous amount of energy is released. In this reaction, the lightest nuclei – hydrogen – fuse at extremely high temperatures (after reaching sufficient temperature, given by the Lawson criterion, the energy of accidental collisions within the plasma is high enough to overcome the Coulomb barrier [which is electric repulsion – nuclei have positive charge]) and pressures to form heavier nuclei – helium – alpha particle. Lawson criterion was derived on fusion reactors by John D. Lawson in 1955 [5] is an important general measure of a system that defines the conditions, which are needed for a fusion reactor to reach ignition. That the heating of the plasma by the products of the fusion reactions is sufficient to maintain the temperature of the plasma against all losses without external power input. Lawson criterion gives a minimum required value for the product of the plasma density n_e and the "energy confinement time" τ_E. Central concept of Lawson criterion is the energy balance for any fusion power plant, which uses a hot plasma. It is shown below:

$$\mathbf{Net\ Power} = \mathbf{Efficiency} \times (\mathbf{Fusion} - \mathbf{Radiation\ Loss} - \mathbf{Conduction\ Loss})$$

Where net power is the excess power beyond that is needed internally. Efficiency is how much energy is needed to drive the reactor. Fusion is rate of energy generated by fusion reactions. Radiation is the energy lost as electromagnetic wave, leaving the plasma. Conduction is the energy lost, as mass leaves the plasma.

There are three isotopes of hydrogen: hydrogen, deuterium and tritium. The nucleus of each of these isotopes contains a single proton, which characterizes the isotopes as a hydrogen element. The nucleus of deuterium has one neutron, whereas that of tritium has two. In each isotope the neutral atom has an electron outside the nucleus to compensate for the charge of the single proton.

Reaction that is the most likely to occur is between the deuterium and the tritium (shown in Fig. 1), generating a helium nucleus (alpha particle) and a neutron. Total mass of the products in this reaction is lower than that of the interacting particles and energy is released according to the mass-energy equivalence principle.

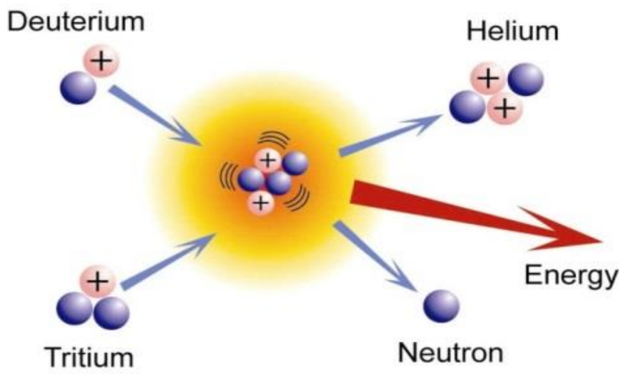

Fig. 1 Scheme of D+T reaction

For example:
AMU = atomic mass unit, proton=1 AMU (1 AMU also equals $1.67 * 10^{-27} kg$)
Mass of four hydrogen atoms: 4 x 1.007825 = 4.03130AMU
Mass of helium atom: 4.00268AMU
Difference: 0.02862AMU = 0.71% of original mass
If 4 grams of hydrogen are converted to helium, $2.8 * 10^{-2}$ grams are converted to energy:

$$E = mc^2 = (2.8 * 10^{-2} g) * \left(3 * 10^8 \frac{m}{s}\right)^2 = (2.8 * 10^{-5} kg) * \left(3 * 10^8 \frac{m}{s}\right)^2 = 2.6 * 10^{11} J \tag{1}$$

This tiny amount of lost mass is converted into a relatively large amount of energy because the small mass is multiplied by the speed of light squared.
In order to obtain controlled thermonuclear fusion with a positive energy balance in the laboratory, one have to heat a deuterium-tritium plasma to very high temperature (100 million degrees, i.e., more than six times the temperature of the sun's interior) and keep it confined in a limited space long enough for the energy released from the fusion reactions to compensate both for energy losses and for the energy put in to produce the plasma. But at such high temperature the problem is how to confine the plasma in a vessel.

$$^{2}D + {}^{2}D \rightarrow {}^{3}T + p + 4.0\ MeV$$
$$^{2}D + {}^{2}D \rightarrow {}^{3}He + n + 3.2\ MeV$$
$$^{2}D + {}^{3}T \rightarrow {}^{4}He + n + 17.6\ MeV$$
$$^{2}D + {}^{3}He \rightarrow {}^{4}He + p + 18.4\ MeV$$

$$^{6}Li + n \rightarrow {}^{4}He + {}^{3}T + 4.8\ MeV$$
$$^{6}Li + p \rightarrow {}^{4}He + {}^{3}He + 3.9\ MeV$$
$$^{6}Li + {}^{2}D \rightarrow 2\,{}^{4}He + 22.3\ MeV$$

Fig. 2 Important fusion reactions [10]

Different methods may be used to produce and heat up plasma. The most simple is a powerful electrical discharge between electrodes placed inside a vacuum chamber and supplied from a high-voltage condenser bank. This technique has been used in the so-called Z-pinch facilities, where plasma confinement is realized by magnetic field produced by a very intense current flowing through this plasma [11].
Another important method is based on the production of very dense and hot plasma from a small target (containing D-T mixture) irradiated by a very powerful laser or particle beams. Plasma produced in such a way can have a very high density (>10^{23} cm^{-3}) and temperature (>5 keV [1 eV corresponds to about 1.1×10^4 K].), and be confined by its inertial effects long enough to produce a large number of fusion reactions. The electromagnetic energy of high-

powered laser beams is uniformly transferred onto the sphere's surface, which evaporates and, according to the action-reaction principle, the fuel is compressed and heated. Such an 'inertial confinement' has been used in many laser–plasma experiments, e.g., in the very large laser system NIF (National Ignition Facility). It was made of 192 laser beams of the total energy equal to 1.9 MJ converted into X-rays (inside a gold 'Hohlraum' cylinder) and concentrated upon a miniature fusion target (a 2.3 mm diameter sphere containing a D-T mixture). During about 20-ns laser pulse, this target could absorb 8.5–12 kJ from laser radiation and produce 14.4–17.3 kJ energy from D-T fusion reactions [2].

The most effective method to realize controlled fusion reactions seems now to be a magnetic containment of plasma of thermonuclear parameters within the so-called tokamak. The tokamak is a quasi-toroidal device characterized by a hollow vessel or chamber in which the plasma is confined by a magnetic field and bound to force field lines along a spiral path. This type of magnetic configuration is obtained by combining an intense toroidal magnetic field, produced by magnetic coils placed around the doughnut, with a poloidal magnetic field, obtained by externally inducing a current in the plasma. The poloidal current also helps to prevent the plasma particles from migrating towards the vessel walls. Another set of external magnetic coils is used to provide auxiliary magnetic fields that control the position of the plasma in the vessel. The JET device produced about 16 MW during about 2 s (i.e., there was obtained about 1.4×10^{19} fusion neutrons) [16] in 1997 using a D-T mixture. In the described experiments for plasma heating there were applied microwaves of 3 MW power and neutral deuterium beams of 22 MW power. Unfortunately this means that the positive energy balance has not been achieved, but the obtained results have became the basis for acceleration of efforts to build a new large ITER (International Thermonuclear Experimental Reactor) facility.

The roadmap (Fig. 3) toward a fusion reactor relies on three devices: Joint European Torus (JET), its successor, an International Thermonuclear Experimental Reactor (ITER) and a demonstration reactor called DEMO. JET represents only a pure scientific experiment. The ITER project aims to make a long-awaited transition from experimental studies of plasma physics to full-scale thermonuclear power plants. Construction work on the ITER began in 2010, and the first plasma is expected to be produced in 2020. The ITER has been designed to produce 500 MW of output power with 50 MW of input only. The ITER has to demonstrate the principle of producing more energy from the fusion process than that used to initiate it, it is something that has not been yet achieved with previous fusion reactors. But it will be only a scientific demonstration, because the ITER will not generate any electricity. The next foreseen device, DEMO, is expected to be the first fusion plant to provide electricity to the power grid. Competitive electrical power production is the end goal for fusion energy [13]. The DEMO reactor will be a first-of-a-kind device. There are considered two reactor concepts (DEMO1: pulsed reactor and DEMO2 steady-state reactor). Validated ITER technology will be applied wherever possible to DEMO, nevertheless there will be areas of technological development that will have not been attempted previously (i.e., self-sufficient fuel cycle) in an integrated environment [1]. The DEMO must look like and act like the first commercial power plant, and its availability must be adequately high (>50%) for the power producers to commit to building a commercial fusion plant. To be competitive, the final goal for commercial Fusion Power Plant (FPP) should have high availability, preferably exceeding 80%, with very few unplanned shutdowns. To profit from economies of scale, the plants should be large (∼1600 MW electrical power) and total construction time should be less than 5 years [4].

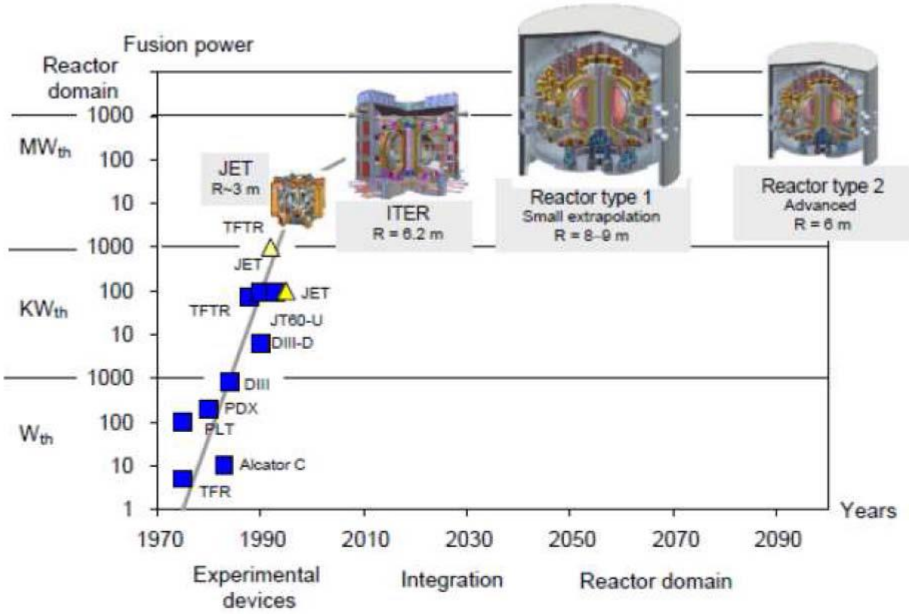

Fig. 3 History and plans for thermonuclear reactors in the coming years [10]

2 The purpose and the scope of work

The purpose of this conceptual analysis is to know the current available technology of the thermonuclear fusion, to determine abilities of the energy production with the help of the thermonuclear fusion and to know possibilities and parameters to yield energy using tokamak reactor type.

The scope of work:

- literature review treating about atomic physics, mainly about thermonuclear fusion,
- review of actual possibilities to obtain energy from thermonuclear fusion,
- establishing the best technology enabling obtaining this energy,
- data and literature analysis and processing,
- calculation of conceptual thermonuclear power plant.

3 Characteristics of the thermonuclear power station concept

Characteristics of the thermonuclear power station concept are based on the ITER (Fig. 4) and Russian DEMO-S reactor with continuous plasma burn assumptions. The main parts (some are shown in Fig. 5) of the reactor are the vacuum vessel (VV), blanket (BL), divertor, and magnetic system (MS) including the superconducting toroidal field (TF) and poloidal field (PF) windings. Dimensions, shape and number of TF coils (16) were chosen from conditions ensuring permissible field ripple, maximum possible outside diameter of the central solenoid (CS), and minimum possible radial dimensions of VV and BL. They were also determined by divertor dimensions and its position with respect to X-point. The major components of the machine are described below.

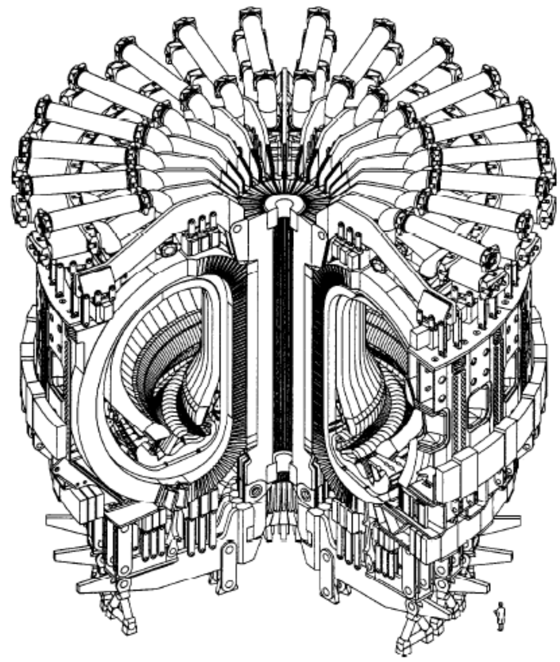

Fig. 4 General assembly of ITER [8]

The vacuum vessel is the mechanical structure supporting all internal components including shield blanket, divertor, antennae and pumps. In this role, it has to withstand forces from disruptions and abnormal events. The role of the vacuum vessel also establishes the first safety confinement barrier. The vacuum vessel provides high quality vacuum, and low electrical resistivity at high temperatures. The resistance is of order of 16 micro Ohms and limits the current induced in the vessel and consequently the stresses. The vacuum vessel design is a double-toroidal-shell structure joined by poloidal rings. The vacuum vessel is an integral part of shielding around the plasma; the inside of its double wall is filled with steel balls and cooled with water or an organic coolant, both of which are under consideration [8]. It also limits the nuclear heat flux to the toroidal coils to under 10 kW during the nominal operation of the machine. The basic vacuum vessel is composed of 24 sectors welded together. The different ports are welded to the vacuum vessel after a rotation of 7.5” of the basic vessel. There are 24 vertical ports allowing the assembly and disassembly of the shield blanket segments and 24 horizontal ports allowing introduction of antenna and blanket test modules and remote handling equipment. The horizontal ports also provide diagnostics access and are compatible with neutral-beam injection. The 24 lower ports are dedicated to assembly and disassembly of divertor cassettes and for vacuum pumping. Leaks, should they occur, can be repaired by welding throughout the basic performance phase, and during most of the extended performance phase.

Shield blanket can be a shield with or without the ability to breed tritium. The shield-blanket is composed of a first wall, shielding materials, a coolant system and a structural back plate. Its role is to be part of the neutron shielding for the superconducting magnets, to extract the fusion power produced by the neutrons and to maintain the integrity of the first wall under thermal and mechanical loads. Beryllium has been chosen as the facing element for its low 2 material properties (radiation and dilution), for its metallic behavior, its good thermal conductivity, its low chemical interaction with hydrogen and its gettering effect with oxygen. Experience has shown that discharges with a beryllium first wall are less prone to disruption owing to the radiation level. The shield blanket proposed here is a self-supporting structure made of welded poloidal segments. Each of them is constructed of stainless steel and is water cooled. The shield blanket is capable of withstanding, in normal operations, a mean neutron wall loading of 1 MW m^{-2}, peaking to 1.45 MW m^{-2} in the outboard region. The global electrical resistance of the shield blanket is low, about 12 micro Ohms. The plasma-facing surface of the first wall is coated with a renewable layer of beryllium and is protected from damage during off-normal events by multiple rows of poloidal limiters, cooled by contact, which could be easily replaced. The side walls, the back plate and the shielding material are made of stainless steel. The shield blanket coolant is water operating at a maximum operating temperature of 200 °C and pressure of 2 MPa. The outlet water temperature is kept constant at 200 °C, with the inlet temperature at 150 °C for a fusion power of 1.5 GW. Water enters and exits at the top ports and flows at a velocity of 2-5 m s^{-1}. It maintains the back-plate temperature at the outlet temperature in order to minimize thermal stresses and fatigue and to allow a proper outgassing. 12 primary heat exchangers for the shield blanked are located on the cryostat to minimize the length and volume of the piping. The shield material is activated by absorbed neutrons. The afterheat power after nominal plasma operation reaches a level of a couple MW and requires some provision for continuous cooling. At the end of the basic performance phase, the stainless steel and copper activation will be sufficient to require burial of these components after some period of intermediate storage. This shield-blanket system is arranged in modules which can be installed and removed through the vertical maintenance ports.

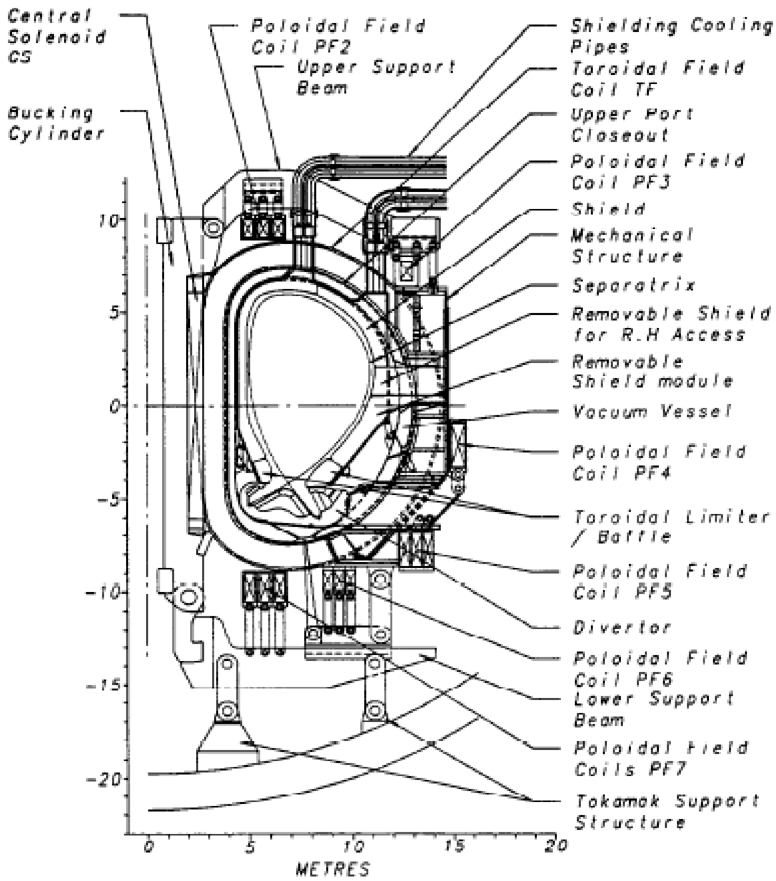

Fig. 5 Vertical cross-section of the ITER [8]

Tritium can be produced within the tokamak when neutrons escaping the plasma interact with a specific element—lithium—contained in the blanket. This concept of 'breeding' tritium during the fusion reaction is an important one for the future needs of a large-scale fusion power plant. Tritium could be breed inside Lithium self-cooled blanked and lithium breeder/coolant occupies about 90% of blanket volume. This means that thermonuclear fusion power plant could be self-sufficient in means of tritium, lithium need to be reloaded.

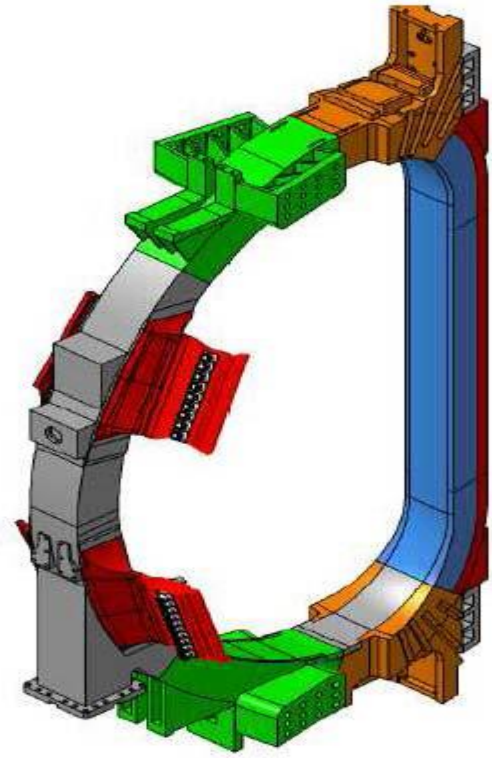

Fig. 6 Toroidal field coil of about 16 m in height and 9 m in width [7]

The magnetic system (MS) (shown in Fig. 6) includes 16 TF coils, 7 PF coils, central solenoid (CS) and structure elements. The ITER magnet systems are niobium-tin superconductor magnets. The TF magnet and central solenoid operate with a maximum magnetic field strength of 13 T up to 6 K. Niobium-tin allows operation at high fields and can withstand, with adequate margin, the heating associated with nuclear effects and losses.

The TF unit consists of the superconducting coil placed in a case made of stainless steel. The superconducting double-layer sections in the common case insulation. The main materials for TF winding are: stainless stabilized austenitic steel 316LN type, insulation of polyimide and glass tape, glass fiber and glass tape and epoxy compound. The CS is a multi-layer four-start cylindrical winding consisting of two sections with total active part length of 12 m and with electrical and cryogenic connections and leads brought out to the face parts.

The mechanical design aims at minimizing the material used in all the coils. This is done by balancing counteracting forces over the shortest possible path. The hoop tensile forces acting on each D-shaped coil are resisted by the structural steel contained in the coil winding and by a steel band running along the outboard portion of the coil, in combination with the bucking cylinder. The bucking cylinder, made of laminated steel plates, supports the central solenoid against the centripetal force generated by the TF coils, and also some of the tension in the TF coil inboard leg. For the TF coils, a laminated structure using stainless steel plates, which take most of the stresses, has been adopted. This structure distributes the path of the forces and eliminates the risk of catastrophic failure, resulting from crack propagation in thick structural elements. The conductor is designed to operate safely with losses induced in the cable at the beginning and end of the plasma pulse, and also during disruptions. For a toroidal magnetic field strength on the axis of 5.7 T, the magnetic field strength on the cable approaches 13 T for a current of 53 kA in the conductor. The reliability of the coil is essential; therefore adequate margins have been incorporated in the coil design to accommodate the level of stress in the conductor, structure and insulation. The insulation of the conductor to the plate is able to withstand the full operating voltage. In addition, each coil has a ground insulation also capable of safely withstanding the full voltage. Each TF coil has a mass of approximately 400 t and carries a total current of 9.7 MA in 182 turns; the TF magnet system can store energy up to 109 GJ. Each TF coil consists of 15 layers of conductor! (Parameters sum up in Table 1.)

Table 1. Principal parameters of the TF system [8]

Total mass per coil (t)	411
Overall height (m)	≈ 17
Overall width (m)	≈ 11
Number of layers	15
Number of turns per coil	182
Total current per coil (MA)	9.7
Current per conductor (kA)	52.5
Total stored energy (GJ)	109
Maximum toroidal field strength at conductor (T)	12.5
Maximum voltage between layers (kV)	1.33

The PF system includes the central solenoid and six individual coils distributed around the machine. The central solenoid (Fig. 7) has a mass of 928 t, carries a total current of 129 MA in 3212 turns and can store energy up to 12.9 GJ. The operating current is about 40 kA. The layers are wound along its entire height. Each layer is wound with four conductors in hand to reduce cooling channel lengths. The conductor for the poloidal coils is made of Nb_3Sn. These poloidal coils are supported vertically by the mechanical structure through links, which allow a variation in the radial dimension of the coils. Each trapped poloidal coil is subdivided into three elements which are individually connected and mechanically supported. This arrangement allows in case of a fault to eliminate the fault element and to operate with the remaining two provided so that there was some redundancy available.

Table 2. Principal parameters of the CS system [8]

Mass (t)	928
Height (m)	≈ 12
Outer radius (m)	≈ 2.8
Number of turns	3212
Total current (MA)	129
Current per conductor (kA)	40
Total stored energy (GJ)	12.9
Maximum toroidal field strength at conductor (T)	12.5

The conductors in the central solenoid are each in the form of a cable in conduit. They have a square outer cross-section and a circular cable cross-section. The conduit material is 908 Incoloy (nickel-based alloys with excellent strength at high temperatures). The cable uses Nb_3 Sn strand and features a cooling channel at the center. Cooling is provided by supercritical helium flow in the conductors. (Parameters sum up in Table 2.)

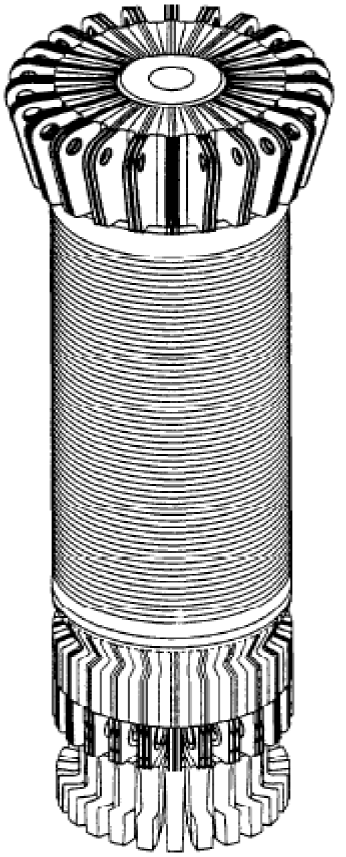

Fig. 7 Central solenoid and bucking cylinder [8]

The divertor (composed of 54 assemblies, about 8.7 tones each), due to the local inversion of magnetic field lines, is designed to collect heavy impurity ions from a near-wall plasma region, and to ensure efficient vacuum pumping through special slits and external cryopumps. Because of large thermal loads (up to 20 MW/m2 over 10 s, and 5–10 MW/m2 over longer periods) the inner and outer divertor targets as well as its central part (dome) will be made from tungsten. Experimental observations from DIII-D JET, ASDEX and JT-60U demonstrate that, with proper operating conditions, a cold dense plasma can be formed in the divertor chamber without loss of plasma performance [8]. This cold plasma can radiate most of the power flowing into the divertor thereby reducing the fraction falling on the divertor plates by an order of magnitude. The present divertor configuration [4], with an X point inside the vacuum vessel, channels particle and energy flows along the open magnetic field lines over a distance of about 25 m. The radiating surface is 2-300 m^2. The divertor geometry requires about a quarter of the available plasma volume. In normal operation, power flux exhausted perpendicular to the magnetic field is maintained below 3 MW m^{-2} and the power flux at the target plates could be limited to 5 MW m^{-2}.

One of the solution to heat up plasma in tokamak is ion cyclotron r.f. (radio frequency) heating (ICRH) of a minority which could be either deuterium or helium 3, or tritium at the second harmonic. The power will be coupled through two groups of nine antennae, each with two straps, each fed through three horizontal ports extending toroidally over 45”. Each group can be phased to produce a traveling fast Alfven wave (type of magnetohydrodynamic wave in which ions oscillate in response to a restoring force provided by an effective tension on the magnetic field lines) driving a toroidal current. The coupling resistance of the antennae can be actively controlled. These antennae resonate at a number of frequencies spread over a wide

frequency range from 20 to 90 MHz and do not include any electrical insulation inside the vacuum vessel. Each antenna is embedded in the shield blanket module similar to a blanket test module. It may be possible to double the r.f. power by upgrading the r.f. generator system and increasing the r.f. voltage on the antennae.
Electron cyclotron heating (ECRH) and drive can also be considered. This method can only be considered for the main heating of ITER if the continuous-wave gyrotron development is successful at frequency in the range of 170 GHz. ECRH, however, may be considered at a level of 2 MW at 140 GHz to assist the plasma start-up, although at the cost of added complexity [8].
Neutral-beam injection is a well-established heating and current drive technique using positive-ion technology, with an energy less than 160 keV. Neutral-beam injection is method which consist of a beam of high-energy neutral particles that can enter the confinement magnetic field. When these neutral particles are ionized by collision with the plasma particles, they are kept in the plasma by the confining magnetic field, and can transfer most of their energy by further collisions with the plasma. By tangential injection in the torus, neutral beams provide also momentum to the plasma and current drive, one essential feature for long pulses of burning plasmas. Neutral beam injection is a flexible and reliable technique, which has been the main heating system on a large variety of fusion devices.
Several types of power supply for ITER are required: to energize and control the toroidal coils and to protect them during an accidental quench; to energize the PF coils; to supply power to various systems, including auxiliaries (auxiliary heating, cryoplant, vacuum pumps, heat transfer systems etc).
Cryoplant building occupies approximately 40 percent of the covered space. It has three identical helium refrigerators which will work in parallel to provide liquid helium to the major clients for cooling. It is about 10.000 tonnes (in ITER) of superconducting magnets plus the cryosorption panels that ensure high-quality vacuum to the large volumes of the cryostat (8.500 m^3) and vacuum vessel (1.400 m^3). The three units will provide a total average cooling capacity of 75 kW at 4.5 K (-269 °C) at a maximum cumulated liquefaction rate of 12.300 litres per hour. The cryoplant will also produce liquid nitrogen (LN_2) to be used as a "pre-cooler" in the liquid helium plant and for the refrigeration of the ITER thermal shield. Lost nitrogen will be extracted directly from the atmosphere in a gaseous nitrogen generator with a production capacity of 50 tonnes per day, and then processed in the two liquid nitrogen refrigerators with a maximum capacity of 1300 kW at 80 K [14].

4 Analysis of the thermonuclear power station operation

4.1 Nuclear fuel

Appropriate amounts of fuel must be delivered in order to operate a thermonuclear reactor. The fuel can be D+T or D+D, tritium is rare on our planet - minimal amounts in the atmosphere and deuterium has a natural abundance in Earth's oceans of about one atom in 6420 of hydrogen. Thus deuterium accounts for approximately 0.0156% of all the naturally occurring hydrogen in the oceans. One could obtain deuterium from oceans, which have some amount of it (enough deuterium to produce energy for consumption for billions of years), and burn in fusion reactor, albeit D+D will grant one only 4 MeV. But from D+T one will obtain 17.6 MeV, it is more profitable. Large fusion power station generating 1500 MW of electricity will consume about 400 g of deuterium and about 600 g of tritium. Tritium (due to its radioactive

decay, with a half lifetime equal to 12.3 years) cannot be stored in big amounts and for a long time, but it can be obtained from tritium breeding modules. It is estimated that for 400 s operation of ITER one will need an input of 54 g of tritium, and if 1 g of that is burnt, the rest should be extracted through tritium breeding modules and the external tritium reprocessing plant [10].
Pure deuterium can be obtained from electrolysis of heavy water (D2O), which can relatively easily be separated from ordinary water, or from isotopic exchange in a hydrogen-sulphate gas. The separation of hydrogen isotopes can also be done by mean cryogenic distillation. One will need some power to obtain by this method deuterium, nowadays on account of establishing the CO_2 emissions reduction coal systems are being ruled out and is turning up for renewable energy sources. Renewable energy sources, which are working at night, are causing excess of electric energy (lower consumption at night) on grid. This energy could be used to divide H_2O or D_2O to H_2 or D_2 with O_2, it grants us a source of fuel for fusion power plant. This kind of source is better, because energy at night which is in excess is cheaper, giving a cheaper solution for yielding deuterium for fusion power plants.
Naturally occurring tritium is extremely rare on Earth, where trace amounts are formed by the interaction of the atmosphere with cosmic rays. One can get some tritium from nuclear reactions (tritium for American nuclear weapons was produced in special heavy water reactors). Some tritium can be obtained by removing it from heavy water from CANDU, it is a waste by-product of its commercial power production. During the Cold War era, there was a tremendous amount of nuclear materials produced – including tritium. The production supply was established, and excess material was stored until needed. One can get tritium from defense systems, it is a solution only for short time (limited amount of stored tritium and tritium inventory is decaying approximately 5.6% per year changing into He-3).
Tritium is produced in nuclear reactors by neutron activation of lithium-6 (reaction shown in equation 2). This is possible with neutrons of any energy, and is an exothermic reaction yielding 4.8 MeV. For applications in proposed fusion energy reactors, such as ITER, pebbles consisting of lithium bearing ceramics including Li2TiO3 and Li4SiO4, are being developed for tritium breeding within a helium cooled pebble bed also known as a breeder blanket.

$$ {}^{6}_{3}Li + n \rightarrow {}^{4}_{2}He + {}^{3}_{1}T \tag{2} $$

Lithium necessary for the tritium breeding can be extracted from several minerals and from clays. Its natural deposits are particularly found in South America. The Earth's crust contains enough lithium for thousands of years, and the world seas also contain a huge supply of lithium.
Lab experiments suggest that future fusion reactors could use helium-3 gathered from the moon (reaction shown in equation 3). There are plans to mine helium-3 for an ideal fuel for fusion reactors, but almost unavailable on Earth can be yielded from the moon's surface. So maybe there will be any space programs to the Moon to mine He-3. Russia claims about building a permanent moon base by 2015-2020 and it will be extracting He-3[15].

$$ {}^{2}_{1}D + {}^{3}_{2}He \rightarrow {}^{4}_{2}He + p + 18.4\ MeV \tag{3} $$

4.2 Fusion reactor operation

Plasma is supplied from two streams, deuterium and tritium, it is shown in Fig. 8. They both enters plasma and are “burned” inside in high temperatures in fusion reaction producing energy and helium-4. “Burned fuel”, helium-4 and remaining tritium, is pumped out to helium/tritium container, where tritium and helium-4 are separated. Tritium is returned to tritium container and carrying on to plasma and helium-4 is being stored in helium containers. In Fig. 8 lithium blanket is around plasma, which is bombarded by free neutrons and due to its reaction between Lithium and neutron (eq.3) lithium blanket produces tritium. Tritium from lithium blanket is transported to tritium container and from there further to plasma burn process. In blankets are high temperatures, which are caused by plasma electromagnetic radiation, heat is transferred to in heat exchanger to coolant (helium, sodium or water or combined stages) and there heat is used to produce steam in steam generator for steam turbine cycle. Steam turbine cycle is similar to steam turbine cycles used in conventional power plant. Example steam turbine cycle is analyzed in next subsection.

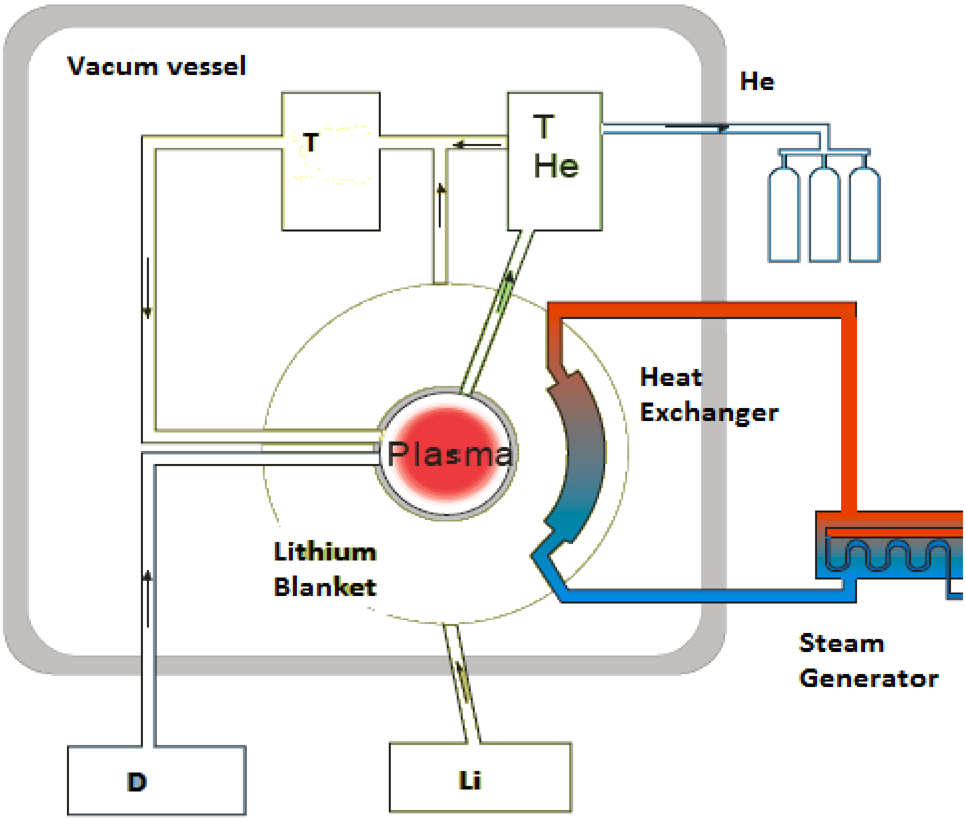

Fig. 8 Scheme of reactor functioning

There are two possibilities for fusion power plant reactor (depending on materials endurance and chosen building option), pulse reactors and reactors with continuous plasma burn. In both of them plasma is heated up by intense (up to several tens MW) electromagnetic pulses (ion and electron) of different frequencies (ranging from MHz to GHz) and/or intense high-energy (80–100 keV) neutral atomic beams (NBI), and with continuous burn bootstrap current could be used to maintain fusion reaction. Pulse reactor base on pulses and burn only for limited amount of time. Single ITER pulse involves a sequence of eight phases that take approximately 2200 s: a pre-magnetization (200 s), a plasma initiation (about 10 s) which requires a voltage per turn inside the vacuum vessel of 15 V (up to 30 V outside), a current ramp-up (90 s), the heating phase (50 s), the long burn (1000 s), the burn termination (100 s), the current ramp-down (200 s) and the PF rest and retool (550 s). The peak demand on the grid will be of the order on 1 GW and the power supply will use mainly thyristor rectifiers [8]. For fusion reactor with continuous burn, ignition could run as in pulse reactor, but in this kind of reactor plasma burn is maintained (fuel is supplied continuously – not only between pulses like as in pulse reactors) by plasma heat up systems. In Russian DEMO-S with continuous burn it is estimated for high (~60%) contribution of bootstrap current into the plasma current drive. The remaining fraction (~40%) of the driven current is to be provided by the neutral beam injection (NBI) system [3].

4.3 The power plant cycle

Fusion reactor could work with steam turbine system as it is shown in Fig. 9. Heat is transferred from coolant to water and steam generator produces steam, which is driving the turbine, which gives mechanical energy to generator. Generator is producing electrical energy from mechanical energy. Steam parameters that one could use depends on materials endurance only, because one can obtain any temperatures could want. The analysis focuses on steam turbine which is shown in Fig. 10. One's reactor should work with two turbines to ensure the reliability of power plant. Reactor in this analysis generates 2 GW of heat power (assumption). Analyzed reactor operates with two turbines. Thermal power generated in fusion reactor is divided into two streams for two steam generators (for two turbines) giving 1 GW of thermal power for each of steam generator.

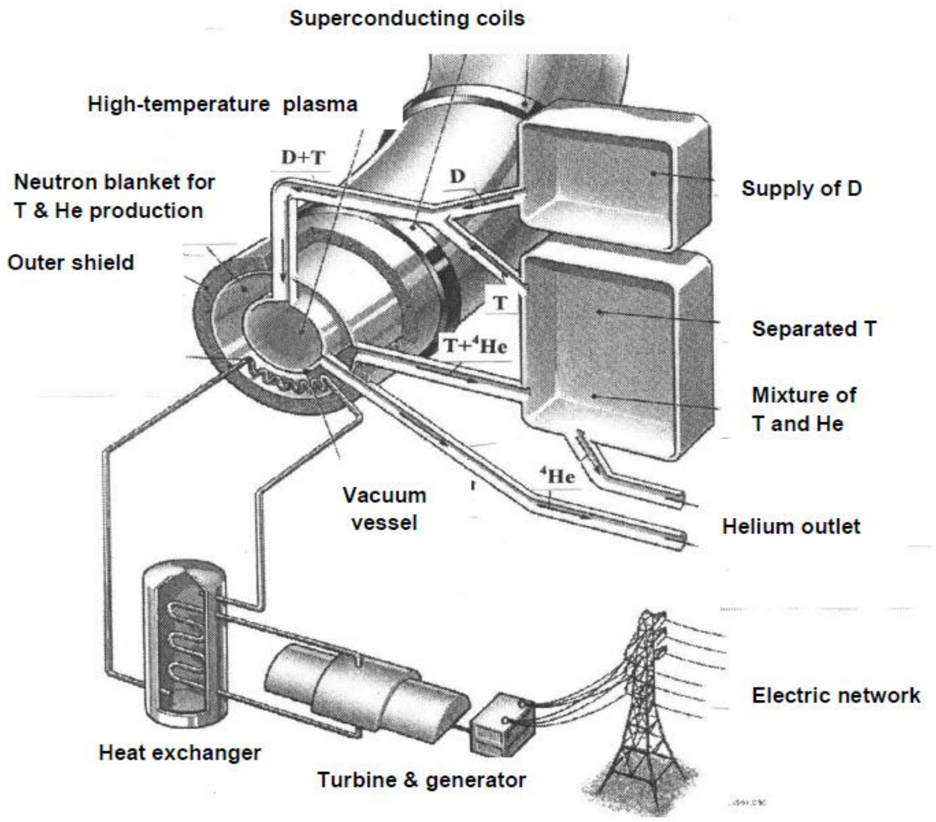

Fig. 9 Scheme of a future electric power plant based on the toroidal magnetic trap of the tokamak type [12]

Analyzed turbine (shown in Fig. 10) consist of three parts, high pressure, medium pressure and low pressure. Amount of generated power and efficiency in three variants were calculated (parameters and results in Table 4.) in this analysis. Calculations were made for conventional power plant, ultra-supercritical power plant and power plant with futuristic parameters of steam. Each of these variants uses steam reheat. Conventional variant parameters are 16/3.3 MPa, 540/540°C and 7 kPa pressure in condenser, inlet flow into steam generator has 200°C. Ultra-supercritical variant parameters are 30/3.9 MPa, 620/620°C and 5 kPa in condenser, inlet flow into steam generator has 240°C. Futuristic variant parameters are 35/4 MPa, 800/800°C and 5 kPa in condenser, inlet flow into steam generator has 245°C. Table 3. Contains important assumptions. Pressure for steam outlet flows from each part of turbine are different for each variant and are made as assumption. Processes in turbine are adiabatic non-reversible (equation number 5).

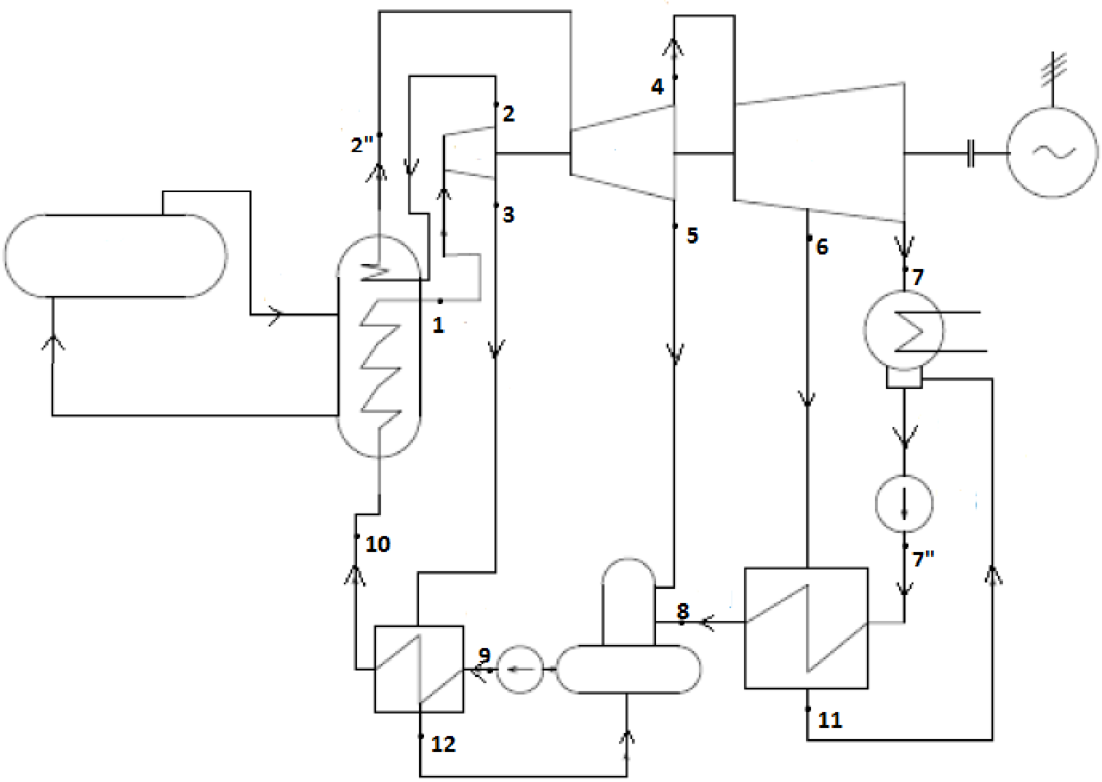

Fig. 10 Analyzed steam turbine

Table 3. Assumptions (2 GW heat power total)

Assumptions	
Power to steam generator, MW	1000
Efficiency of steam generator, %	95
Turbo generator efficiency, %	98
HP turbine internal efficiency, %	85
IMP turbine internal efficiency, %	88
LP turbine internal efficiency, %	80

Equations which were used are below:

$$i = f(p, t, s, v, x) \tag{4}$$

i – specific enthalpy can be obtained in function of two parameters (pressure, temperature, specific entropy, specific volume for overheated steam and additional parameter-quality in special cases when 1≥x≥0)

$$\eta_{it} = \frac{i_i - i_{i+1}}{i_i - i_{i+1s}} \tag{5}$$

η_{it} – turbine internal efficiency (processes in turbine are adiabatic non-reversible), where i is number of point in the scheme (Fig. 10)

$$N_i = m_i * (i_{in} - i_{out}) \tag{6}$$

N_i – internal power, $\dot{m}_i$ – mass flow, kg/s, i_{in} – inlet specific enthalpy of steam, i_{out} – outlet specific enthalpy of steam

$$N_{total} = \sum N_i * \eta_{me} \tag{7}$$

η_{me} – turbo generator efficiency, N_{total} – total electric power of one turbine

$$\eta_p = \frac{N_{total} * 2^*}{Q_r} \tag{8}$$

η_p – Power plant efficiency, Q_r – heat generated in reactor, * - there are two same turbines

Equation for mass flowing through steam generator (has to be iterated):

$$\dot{m}_1 = \frac{\frac{1}{2} Q_r * \eta_{he} - \dot{m}_2 (i_{2''} - i_2)}{i_1 - i_{10}} \tag{9}$$

η_{he}- efficiency of steam generator,

Energy balance equations for steam turbine (internal power):

$$N_{HP.internal} = \dot{m}_1 (i_1 - i_2) \tag{10}$$

$$N_{IMP.internal} = \dot{m}_2 (i_{2''} - i_4) \tag{11}$$

$$N_{LP.internal} = \dot{m}_4 (i_4 - i_6) + (\dot{m}_4 - \dot{m}_6)(i_6 - i_7) \tag{12}$$

Energy balance equations for heat exchangers:

$$\dot{m}_{7''} i_{7''} + \dot{m}_6 i_6 = \dot{m}_8 i_8 + \dot{m}_{11} i_{11} \tag{13}$$

$$\dot{m}_3 i_3 + \dot{m}_9 i_9 = \dot{m}_{10} i_{10} + \dot{m}_{12} i_{12} \tag{14}$$

Energy balance equation for deaerator:

$$\dot{m}_9 i_9 = \dot{m}_5 i_5 + \dot{m}_8 i_8 + \dot{m}_{12} i_{12} \tag{15}$$

Table 4. Parameters for conventional power plant

No.	Mass flow, kg/s	Temperature, ºC	Pressure, MPa	Enthalpy, kJ/kg
1	320	540	16	3412
2	246.5	318	3.3	3032
2"	246.5	540	3.3	3544
3	73.4	318	3.3	3032
4	238.3	305	0.57	3073
5	8.26	305	0.57	3073
6	0.25	174	0.14	2821
7	238	39	0.007	2444
7"	238.3	39	0.007	163
8	238.3	47	0.57	198
9	320	98	16	424
10	320	200	16	838
11	0.25	109	0.14	458
12	73.4	239	3.3	1002

Power plant with conventional parameters of steam (Table. 4) would produce 762 MW of electric output with efficiency level of 38.12%.

Table 5. Parameters for ultra-supercritical power plant

No.	Mass flow, kg/s	Temperature, ºC	Pressure, MPa	Enthalpy, kJ/kg
1	304	620	30	3509
2	266.2	317	3.9	3012
2"	266.2	620	3.9	3722
3	37.8	317	3.9	3012
4	235.1	403	0.9	3273
5	31.1	403	0.9	3273
6	37.25	268	0.26	3006
7	197.8	33	0.005	2469
7"	235.1	33	0.005	137
8	235.1	115	0.9	483
9	304	182	30	787
10	304	240	30	1006
11	37.25	129	0.26	539
12	37.8	249	3.9	1043

Power plant with ultra-supercritical parameters of steam (Table. 5) would produce 861 MW of electric output with efficiency level of 43.08%. This variant has higher parameters of steam but less steam flowing in power plant cycle.

Table 6. Parameters for futuristic power plant

No.	Mass flow, kg/s	Temperature, ^{o}C	Pressure, MPa	Enthalpy, kJ/kg
1	255	800	35	3996
2	232.8	442	4	3314
2''	232.8	800	4	4142
3	22.2	442	4	3314
4	213.3	564	1	3620
5	19.5	564	1	3620
6	53.6	409	0.3	3294
7	159.7	33	0.005	2626
7''	213.3	33	0.005	137
8	213.3	161	1	681
9	255	201	35	874
10	255	245	35	1027
11	53.6	133	0.3	559
12	22.2	250	4	1049

Power plant with futuristic parameters of steam (Table. 6) could not exist due to materials endurance (materials would not work long with 800 ^{o}C and 35 MPa). In this variant again there are higher parameters of steam, value of inlet mass flow is 65 kg/s less than with conventional parameters. Power plant could work with 924 MW of electric output with efficiency about 46.22%.

Table 7. Results table with steam parameters (*fresh steam and reheated steam, ** fuel for efficiency is heat generated in reactor)

	Conventional	Ultra-supercritical	Futuristic
Steam pressure*, MPa	16/3.3	30/3.9	35/4
Steam temperature*, ^{0}C	540/540	620/620	800/800
Generated electricity, MW	762	861	924
Power plant efficiency**, %	38.12	43.08	46.22

Much better results one could obtain with more heat exchangers and better carnotisation, what's more this is simplified scheme and efficiency is much higher than for coal or gas power plants. Results of electric power generated are for two steam turbines, which work at the same time with the same parameters. Moreover Russian DEMO-S is predicted to generate 1500 MW of electric power [14], so it is more than for analyzed variant with futuristic parameters.

Calculations were performed with the help of the Engineering Equation Solver. This is program that allow us to calculate lot of equations on engineering level. Significant advantage of the used program is a possibility of the regulation of equations of the model in the mathematical perspective. There isn't required no determined syntax or the manner of the writing of equations. Important feature are functions built in the program with basic parameters of working substances, like steam or helium.

4.4 Waste disposal

Waste management is considered for the fusion reactor option with He cooled ceramic blanket. The main materials are ferritic steel, Li_4SiO_4 breeder, Be multiplier and traditional structural and conducting materials for magnet system. Considerations don't include any specific divertor protection. The main goal was to estimate the maximum amount of the materials acceptable for recycling. Estimation for main reactor materials [12] shows that hands-on recycling of magnets (up to 60 wt% of reactor materials) should be possible after 30 years of cooling. Modified hands-on recycling is possible after 100 years of cooling for VV steel, ceramics and beryllium S-65C (about another wt25 %). Remaining 15 wt.% (mainly blanket ferritic steel structure) belongs to MLW class and require controlled disposal. Changes in material composition give small decrease in required cooling time for blanket and VV structural material, but causes significant change for the multiplier and breeder (details in [9]).
Waste product from fuel consumption is Helium-4. Helium is a colorless, odorless gas that is totally nonreactive and non-toxic. It is completely safe for environment, people and materials. After hydrogen, helium is the second most abundant element in the universe. It is present in all stars. It was, and is still being, formed from alpha-particle decay of radioactive elements in the Earth. Some of the helium formed escapes into the atmosphere, which contains about 5 parts per million by volume. Helium is used in many sectors of industry, for example as a cooling medium for the Large Hadron Collider (LHC), and the superconducting magnets in MRI scanners and NMR spectrometers, balloons, gas lasers. Helium in fusion reactor could be used as coolant in some blankets, so it would be a self-sustaining source of helium for fusion reactors. Moreover, helium is used in advanced gas cooled nuclear reactors, it is a coolant, which is used to transfer heat and produce mechanical energy on gas turbine and carrying on electricity on electric generator.

5 Summary and conclusions

Thermonuclear fusion technology probably would be able to meet demands for the process of the thermonuclear synthesis, so there are some chances of thermonuclear power plant availability in 30 years time or more. Technology would be clean, environmentally friendly and it would be effective method for energy production, far more than actual power generation technology. It would operate with post-burn non-waste production (helium is a product that can be used in another branches of industry and science) way of generating energy for future generations. The development of viable energy generation from fusion technology crucially depends on the effective heat removal from the reactor. Important issues in magnetically confined plasma fusion reactors are associated with the first wall and divertor targets. The steady-state and the plasma disruption heating loads will have to be accommodated and the absorbed fusion energy will have to be removed from the blankets and divertors. Technology is complicated, magnets are cooled to 6 K, when plasma has about 100 million K, heat loads on divertors are about 4-15 MW/m^2 and 1 MW/m^2 neutron wall loads, one has to assure best cooling for reactor blankets and divertors. This would be challenging task to complete. The ITER is currently under construction and thermonuclear fusion technology crucially depends on results of its operation. When something would go wrong with the ITER, development of thermonuclear fusion technology could stop.

Commercials reactor will probably have higher and better (in technological point of view) parameters. Moreover steady state operation mode permits to consider reliable design for most major reactor components (e.g. magnets, vacuum vessel, blanket, divertor) through their lifetime and heat loads of in-vessel components are limited to 2–5 full power operating years. One could obtain high amounts of heat energy and due to technology which is applied one could yield different amounts of electric energy with (different parameters of fresh steam and arrangement of steam turbine system). Nuclear fuel for fusion is a problematic topic, because of rare tritium, but it should be solved with tritium breeding blankets (solid or liquid). The ITER is an important first step where the conversion of fusion-to-thermal energy technology will be assessed, but it will not prove whether this technology is commercially viable. Building the ITER requires outstanding technical capabilities, commitment and vision. This can only be achieved by a dedicated and independent team, with the full support of the world fusion programs. And as for future fusion power plants, this world wide project should get all support as it is needed and operating conditions of this reactor are not as demanding on the materials, coolants, and tritium fuel production as those of the demonstration and more advanced commercial power reactors.

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Koncepcyjna analiza pracy elektrowni termojądrowej z reaktorem typu tokamak

Krzysztof Pajączek

Słowa kluczowe: tokamak, reaktor termojądrowy, elektrownia, reakcja fuzji

Streszczenie

W pracy zebrano informacje na temat technologii reaktorów fuzyjnych z magnetycznym utrzymaniem plazmy (typu tokamak) oraz o ich wykorzystaniu w produkcji energii elektrycznej. Reakcja fuzji jądrowej zachodzi w bardzo wysokich temperaturach rzędu setek milionów kelwinów, stąd też częste określenie fuzja termojądrowa. Reakcje takie zachodzą w gwiazdach i produkują znaczne ilości energii, najbliższy naturalny reaktor fuzyjny to Słońce znajdujące się w naszym układzie słonecznym. Elektrownia bazująca na reaktorze fuzyjnym jest w zasięgu aktualnych technologii, jednakże wymaga sporych starań oraz współpracy wielu narodów. Elementy, z których zostanie zbudowana elektrownia termojądrowa będą musiały być bardzo wysokiej jakości oraz bardzo precyzyjnie wykonane. Sam reaktor fuzji termojądrowej będzie składał się z wielu części, które muszą być odpowiednio wykonane oraz posiadać znaczną odporność na obciążenia cieplne czy promieniowanie neutronowe. Niniejsza praca dotyczy produkcji energii elektrycznej z zastosowaniem fuzji termojądrowej. Przedstawiono opis ważniejszych części wspomnianej elektrowni. Rozpatrywanych było kilka mieszanek paliwa do „spalenia” w reaktorze, przy czym najbardziej prawdopodobnym do zastosowania paliwem będzie mieszanka deuteru i trytu. Poruszony został również temat produkcji trytu w płaszczach chłodzonych płynnym litem. Rozważana elektrownia termojadrowa posiada obieg turbinowy z parą wodną jako czynnikiem roboczym. Źródłem ciepła jest wysokotemperaturowa plazma, a ciepło odbierane jest z płaszcza reaktora otaczającego tą plazmę. Wykonane zostały obliczenia dla koncepcyjnej elektrowni w trzech wariantach technologicznych (różne parametry pary świeżej i wtórnie przegrzanej) oraz poruszono sprawę wykorzystania produktu po reakcji termojądrowej. Przeprowadzona analiza studialno-obliczeniowa pozwala wnioskować miedzy innymi, że projekt ITER jest istotnym pierwszym krokiem na drodze ku fuzji termojądrowej. Technologia termojądrowa wydaje się być osiągalna, czysta i przyjazna dla środowiska, a co więcej bardzo efektywna. Budowa elektrowni opartej na fuzji jądrowej wymaga jednak wybitnych zdolności technicznych, obowiązkowości oraz nieco wizjonerskiego podejścia.