The concept and option of the main parameters of fusion neutron source (TIN-1)

E.Azizov

- ***NRC “Kurchatov Institute***

- Hefei, May 2012

<u>Demonstration of</u> fusion neutrons source

The mission of demo TIN – <u>demonstration of the efficiency the main FNS systems</u>.

At what level and what should be demonstrated?

Demonstration of <u>physics</u>: steady state operation ($f_{BS} + f_{CD} = 1$)

$H = 1.2 – 1.4$; $\beta_N = 3 – 3.5$

R &D (physics) – T-15 (TIN – 0)

Investigation of the efficiency of the device under neutron irradiation.

Demonstration of the <u>nuclear fuel production</u> - in the fewl kg of produced fuel as the possible way for solution one the nuclear power problems.

TIN-1, the stages of development

The project TIN-1 (fusion neutron source based on tokamak) has been developing in Russia for long time and has the status of search for acceptable concept. It grounds on:

-interesting in Russia and other countries on hybrid systems (fusion-fission, projects JUST, JUST-T and others)

-the development of a new generation of thermonuclearв devices, one of which has already been realized (tokamak KTM). The peculiarity of these developments - moderate aspect ratio (A = 2 - 2.5) and the long pulse.

TIN-1 has passed several stages in its development. We briefly describe the main features of these stages. From them you can see the character and ideology of the search, the main modifications of the project and the gradual development concept of TIN-1.

The development TIN-1with warm magnetic system

The goal and the main restrictions on the parameters of installation were formulated.

To achieve the level of steady state flux of a 14-MeV neutron P_N> 5 MWt) under   restrictions: power from the network for EMS

P ≤ 200 MWt, additional heating systems less than 50 MWt,

the major radius of the torus not more than 2 m, aspect ratio c about 2, elongation k=1.7-1.9, toroidal field B<3T

EMS- warm (copper or copper alloys), miltiturn windings with insulation.

Blanket - only at the outer border of torus.

Evaluation of neutron fluence of materials

For production about 30 kg of fuel (U^{233} or Pu^{239}) we need approximately $7 \cdot 10^{25}$ neutrons. If the total area of the modules blanket ts ~ 6 м^2, the neutron fluence ~ 10^{21} n/cm^2.

Restrictions on fluence and the required degree of attenuation:

-organic isolation ~ $5 \cdot 10^{18}$ н/см^2 *,* the required degree of f the fluence release~ 200;

- nuclear heating. If we assume that the helium allowable heat level is no more than 1 kWt, the degree of attenuation (for heating), from the level of ~ 5 MWt (neutrons) will be $5 \cdot 10^3$

- **For save the compactness of FNS with SC EMS we should use protection materials such as ZrH_2. To release the flux on 3 of the order the desired thickness of the protection should be ~ 30 - 40 cm.**

Hold the power at level about 200 MWt is very difficult task.

Limitations for power from the national grid for other large projects:

- The Collider - 180 MWt (10% of the power consumed by the Canton of Geneva),

-**ITER – 500MWt**

We consider the possibility to use instead warm toroidal magnetic system <u>cryoresistive or superconducting one</u>, which could reduce the level of energy consumption. However it should be taken into account that «constant» component of the energy consumption associated with the injectors and all other devices, which in TIN (with the capacity of other systems) may reach 70 MWt. Also, the poloidal system and the central solenoid can be warm.

Radiation protection in fusion neutrons source

The critical point- protection the internal parts of EMS.

Options:

-without protection;

-protection with declining of the neutron flux ~ 10^3;

-full scale protection with decreasing neutron flux ~ 10^5 - 10^6.

The later option was accepted in ITER for superconducting coils.

Criteria for determining the allowed exposure level of irradiation of SC:

-change of the critical current in SC ~ 10^{18} n/cm^2;

-change of resistance of copper conductors ~ 10^{20} - 10^{21} n/cm^2;

-change of electric characteristics of isolation ~ (1–5)•10^{18} n/cm^2;

-acceptable value of specific nuclear heating at helium temperature ~ 10^{-4} Wt/cm^3.

<u>Irradiation with neutrons</u>, the nature of damage

The coefficients for rough scaling:

1 dpa ≡ 1 сна ~ 10^{21} n/cm^2

1 МВт/м2 per year~ 10^{21} n/cm^2

Power fusion reactor:

fluence ~ 10^{23} n/cm^2
damage ~ 100 сна

What happens with the materials under irradiation by fusion neutrons?

<u>The superconductor</u>. Change: critical current; resistance of copper; electric characteristics of isolation; nuclear heating.

Copper alloys. Change conductivity, mechanical characteristics, transmutation (Cu →Ni, Zn)

$\Rightarrow$ **For use of SC required <u>degree of attenuation</u> ~ <u>10^5</u>**

(2)

In the power reactor fluence on the first wall ~ 10^{23} n/cm^2;

fluence on SC coil ~ 10^{18} n/cm^2;

<u>degree of attenuation</u> ~ <u>10^5.</u>

The shield with thickness of 10 - 15 cm gives nothing, the flux does not go down: there is no neutron absorption (only slow down)

The shield 30 - 40 cm on the basis of ZrH_2 - the decreasing of neutron flux ~ 10^3 . Shield 80 - 100 cm (Fe-$H_2$0…) - the decreasing of neutron flux ~ 10^5 – 10^6.

<u>On the outer side</u> it is possible to use a full scale protection or blanket.

<u>On the inner side:</u>

1).Shield thickness 80 - 100 cm for the neutron source is unacceptable (leads to the size increase of tokamak R >3m).

2).Shield thickness 30 - 40 cm is desirable, but leads to tokamak with size R >2m.

3). The EMS without any protection sharply limits the possibilities of the neutron source by fluence. Restriction is associated mainly with acceptable fluence (insulation, copper alloys).

<u>Insulation</u> material for neutron source

The selection of winding insulation is especially important. Electrical engineering has accumulated detailed information about the characteristics of various kinds of insulation under irradiation. There are <u>three of insulation class</u>:

-organic insulation (epoxy resin, polyimides), capable to work up to neutron fluence ~ (1-5)•10^{18} n/cm^2

-inorganic insulation <u>with low-grade SiO_2</u> (mica, ceramic, silicate) with fluence to 10^{20} n/cm^2

-<u>nonsilicate ceramic</u> (Al_2O_3, MgO) with F to (0.5 – 1)•10^{22} n/cm^2

.

TIN-1 (expected fluence 10^{21} n/cm^2):two final classes on the basis of epoxy resin with Al additives (<u>R & D and mechanical properties</u>).

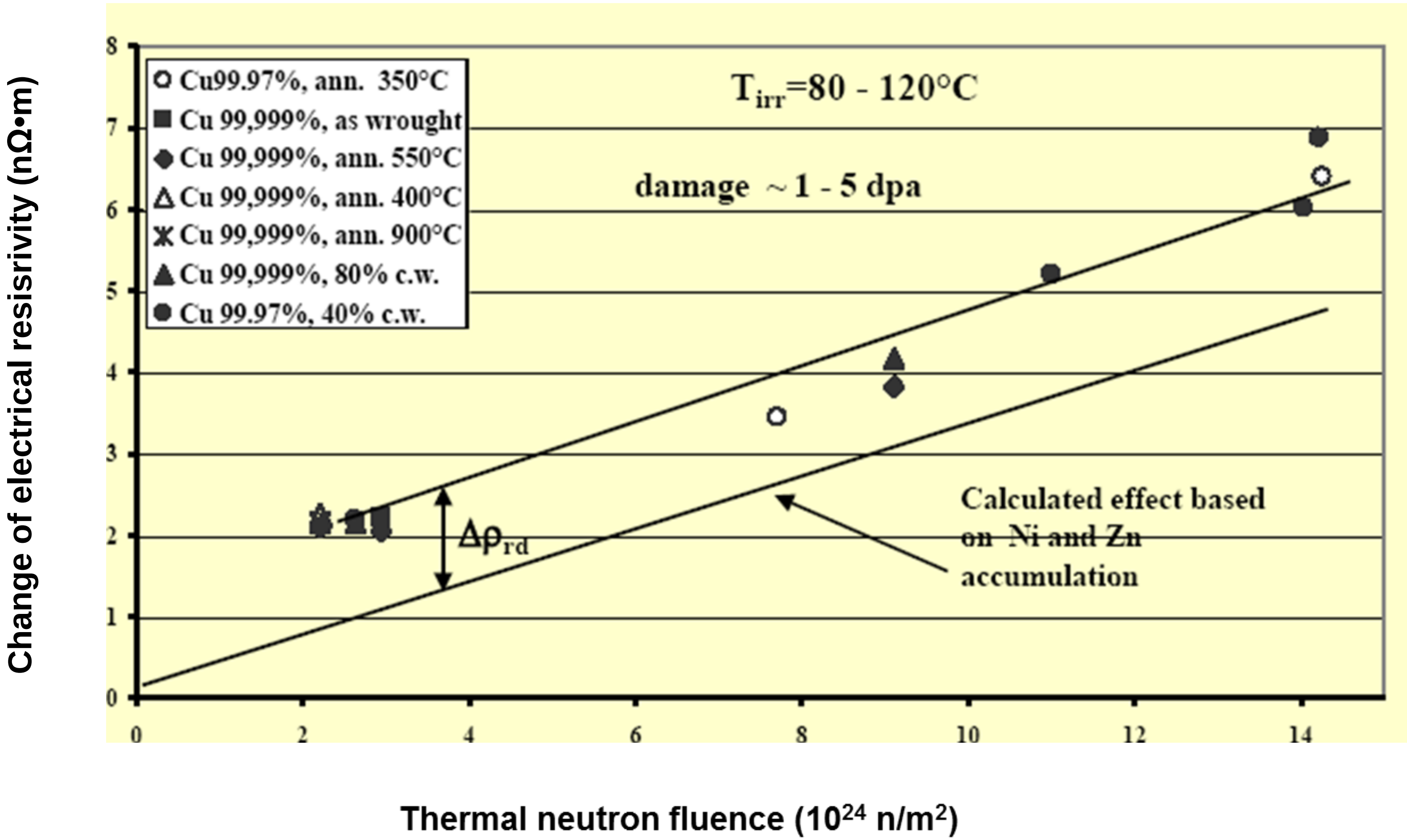
Insulation for EMS

Potential Insulation Solution	Status of Insulation Materials	Insulation Install. Proc.	Expected Rad. Limit	Range of Anticipated 'end-of-life' Properties
Glass/Epoxy	Existing, commercial available	VPI, Pre-preg Combination	10^9 Rads	Shear: >10 ksi(1) Compression: >180(1) kV/mm: > 60(1)
Glass/Aromatic Organic Resins (Polyimide and Bismalimides)	Existing, commercially available	Pre-preg	10^{10} Rads	Shear: > 16 ksi(2) Compression: >160 ksi(2) kV/mm: > 60(2)
Glass/Hybrid Aromatic Organic Resins (Polyimide, Bismalimides, Epoxies…)	Under Development	Pre-preg Combination	10^{10} Rads	Shear: >16 ksi(3) Compression: >180 ksi(3) kV/mm: > 60(3)
Inorganic/Organic Hybrids (reduced organic content)	Under Development	Pre-preg VPI Combination	>10^{10} Rads	Shear: > 9 ksi(3) Compression: > 160 ksi(3) kV/mm: > 70(3)
All inorganic Systems including ceramics	**Conceptual**	Pre-preg, VPI, Spray, Dip Coat, Combin.	**possibly ≥10^{13} Rads**	**Shear: N/A(4) Compression: N/A(4) V/mm: N/A(4)**

1) U.S ITER Insulation Irradiation Program Final Report
(2) Data extrapolated from testing performed by Harold Weber with ITER Insulation Characterization Data
(3) Engineering estimate by CTD
(4) Values for currently available materials not suitable, the hope is to develop new materials with better properties
(5) Gas Evolution from Potential ITER Insulating Materials, US ITER Insulation Irradiation Program

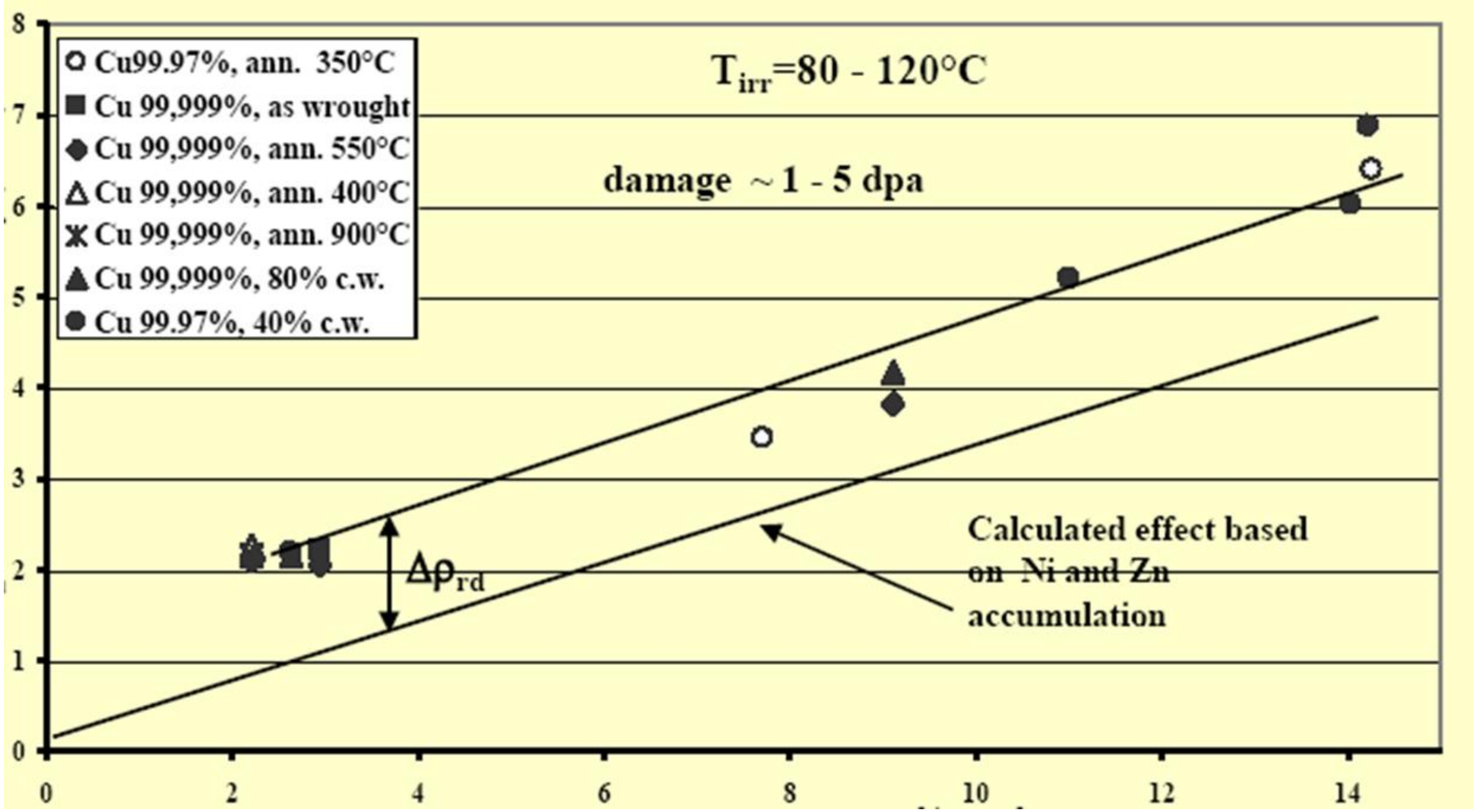
Electrical conductivity of copper alloys under irradiation

a). Pure copper (without irradiation at 20^0C : 17 nΩ•m)



Alloys based on copper

Change of electrical resisrivity (nΩ•m)



Thermal neutron fluence (10^{24} n/m^2)

•

About cryoresistive coils of toroidal field

Difficulties with warm EMS have resulted in a review of the different versions of the magnets cooled to cryogenic temperatures (superpure Al, ~20K) and SC magnets (NbTi and Nb_3Sn). Insulation and nuclear heating in such magnets has led to use a shield for its protection.

It seems that for pure aluminium (A999) on liquid hydrogen temperature (20 K) you can reach the gain according to the resistance of a few thousand times. But in magnetic fields of ~ 5 T this gain is reduced to 300 because of magnet-resistance , and taking into consideration the spending of refrigeration (70 Wt/Wt) – it will be only about three.

In addition, under neutron irradiation conductivity very rapidly degrades (already in fluence of fast neutrons 10^{16} n/cm^2 - resistance increases by 20%).

The lack of gains on power and degradation of the conductivity were led to refusal the options Al cryoresistive coils.

The thickness of the protection to decrease the neutron flux in 10^3 times

1). Δ = 60 см – for protection on the basis of SS – H_2O (B); SS – B4C;
Fe – H_2O;

Δ = 55 см – for protection on the basis of SS – TiH_2 .

2).: We need to use materials with a high density and a high content of hydrogen:

ZrH_2, TiH_2, UH_3

UH3: 30 cm shield is enough (density of the material - 10.9 g/cm^3, and the density of the hydrogen - 8•10^{22} nuc/cm^3).

TIN with winding on the basis of NbTi

ITER project data:

For s/c of NbTi constructive current density in the winding is close to 10 MA/м^2 in the field ~7 Tl.

This leads to a rather thick winding TMC, the aspect ratio for TIN A ~ 2.5 and major radius will be about 2.5 meters.

When we use more expensive superconductors on the basis of Nb_3Sn, constructive current density can be increased up to 25 - 30 MA/м^2.

TIN with winding on the basis of Nb_3Sn

For TIN with winding of Nb_3Sn major radius can be reduced to 1.9 m.

Main characteristics TIN: R=1.9m; A=2.5; B_{t0} = 3 Tl; k_{95} = 1.7; δ_{95} = 0.2 – 0.25; $p_n \geq 0.1$ MWt/$м_2$; the coefficient of improved confinement $H_{IPB(y,2)} \leq 1.2$; normalised beta $\beta_N \leq 3$; the additional plasma heating

$P_{aux} \leq 20$ МВт; the transition to a fully non-inductive current drive;

-the ideology of design: 12 toroidal coils from Nb_3Sn; the sectional double-shell vacuum vessel; solenoid with flux 3 Wb (magnetization from max. to zero); poloidal coils from NbTi; overall cryostat; solid-state shield with gas-cooling; 80 cm width blanket inside the vessel on its outer side ; the geometry of the divertor is like ITER, cooling the divertor by liquid metal.

The return to the warm EMS of TIN-1

The analysis of the FNS with SC EMS showed, that the total cost of the complex TIN-1 with the nuclear part, tritium system and needed R & D can be 400 - 500 million dollars, that is not acceptable. Therefore, steps have been taken to make a drastic reduction in price and return to the warm variant of the magnetic system:

-the suggestion is to exclude inner parts shield and the blankets will be put in some horizontal ports.

-the requirements to product: during operation of TIN in the blanket up to 30 kg of fuel - U^{233} or Pu^{239} (4 - 5 kg of fuel per year) are defined;

-- weight of the final product and area of modules (6 м^2 in 6 ports each size of 1.2 x 0.8 m) allow to find: the fluence of fusion neutrons for the period of exploitation (1 MWt•year/м^2), average neutron load on the first wall (0.14 – 0,2 MWt/m^2), neutron flux (8 MWt), the fluence on the isolation of EMS (to 10^{21} n/cm^2).

TIN-1 parameters with the warm EMS

R = 1.5 м, A = 2.3 – 2.5, B_t = 3 T, k_{95} = 1.7, δ_{95} = 0.2 – 0.25, $H_{IPB(y,2)} \leq 1.4$, β_N ~ 3, P_{aux} = 15 MWt, Δ_{in} = 0.15 м (the gap on the inner side between the plasma and the toroidal coillsOTP includes thickness SOL, FW and constructive gaps).

The vacuum chamber is sectional. Camera is double-wall, the total thickness of the camera is about 5 cm, thickness of each of the shells is 0.5 cm.

The geometry of the divertor is like ITER (with decreasing of sizes in 4 times). Divertor is water-cooled.

Neutrals beam injector energy is not above 160 KeV, power inject into plasma is 15 MWt. The size of the window of input is 1.0 x 0.3 m.

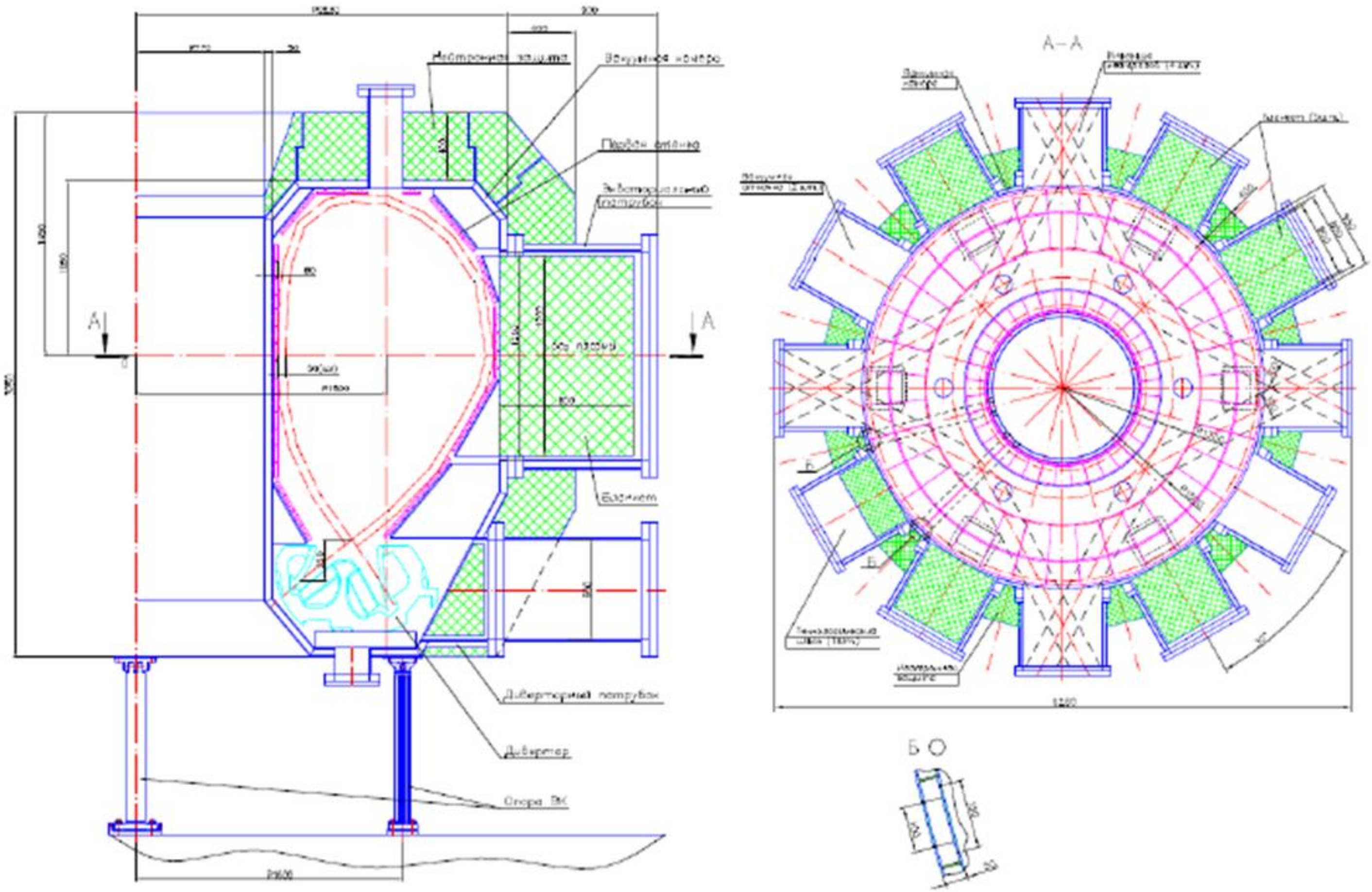
The test blanket modules are located in equatorial ports and has a thickness of 80 cm.

CS can has 2 Vs and use for current rump up to

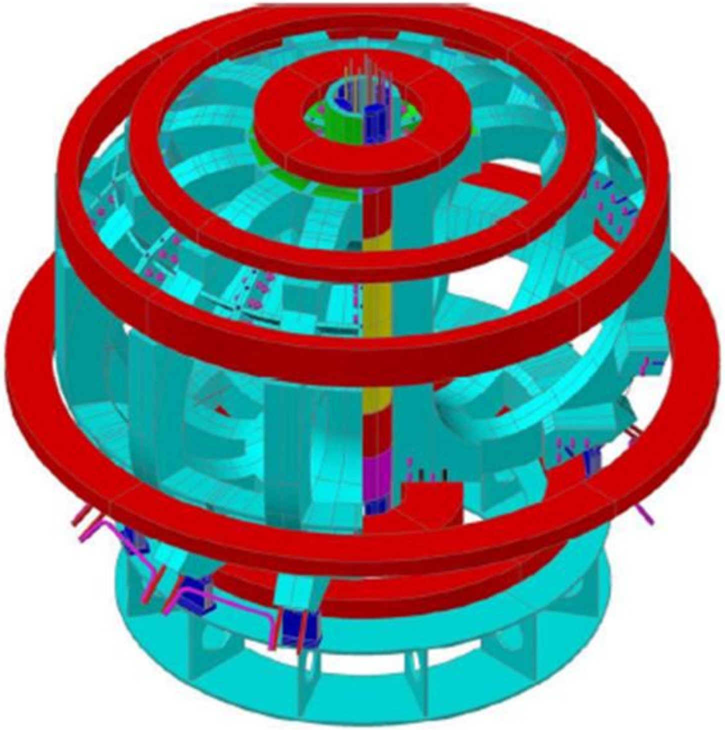
I_P = 500 кA (the further current rump up by current drive)).

A major radius of plasma core, R, m	1,5
A minor radius of plasma core, a, m	0,6
Aspect ratio, A	2,5
Plasma current, MA	2,5
Number of toroidal coils	12
A toroidal magnetic field on the axis of plasma	3,0
Elongation cross-section of the plasma	1,7
Plasma triangularity	0,25
The capacity of extra heating, MWt	15
Normalized beta	3
The coefficient of improved confined	≤ 1,4
Plasma gap – on the internal border	0,15

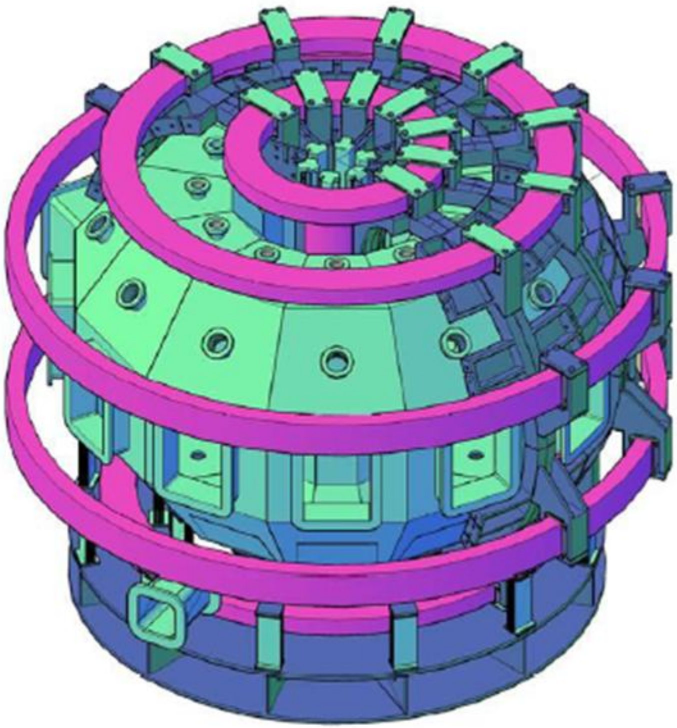
The vacuum vessel of TIN-1



Electromagnetic system of TIN-1



EMS



General assembly

R & D

The physics of TIN-1 is based oncommon accepted. But, a number of R & D are required:

1). All physics tasks will be investigated on TIN-0 (other names - T-15 with, modernized EMS and VV) ;

2). The isolation will be on the basis of Al_2O_3, MgO , capable to bear the fluence > 10^{21} n/cm^2 and mechanical loads.

3). Nuclear and tritium R & D (test blanket modules, tritium loop, plasma processes control, steady state injectors and so on).

In the analysis of parameters of TNS-1 important requirements from the end-product (nuclear fuel) are considered. This is defined dimensions of blanket and through it the size of the device. If such requirements are not set (neutrons without reference to what they are), they you can be reduced.

About vacuum vessel

VV – double-shell design. The thickness of the inner and outer shell - 5mm. The total thickness - 50 mm. Shells are linked by poloidal and toroidal ribs.

Shell VV - water cooled. In addition to basic functions of the cooling water in combination with metal shell partially changes the distribution function of the neutron.

VV - compact and is designed for creation of vacuum volume (with residual pressure - 5×10^{-8} Torr). Outer shell - sealed and must be able to withstand the pressure of the cooling water.

Shell VV - sectional in toroidal direction and consists of three sectors: the two main and one docking, sector will be connected by welded seams.

<u>Ports</u>

Equatorial ports will be distributed by following terms:

-for blanket modules – 5

-for neutrals injection (NBI) – 4

-vacuum pumping - 2

-technological - 1

<u>Divertor</u> ports (6) are attached to the outer cone bottom of the VV.

<u>Vertical</u> ports (12 upper and 6 lower) will be used for monitoring and diagnostics of plasma, electromagnetic and technological diagnostics, also foa gas injection.

<u>Lower ports</u>, if necessary, can be used for pumping of divertor zone.

It is possible to equip with VV 6÷12-th inclined pipes.

Divertor

Divertor is located. at the bottom of the VV and also performs function of the neutron protection the windings of EMS.

Constructively it is supposed to design TIN divertor the same as in ITER , but reduced by approximately in four times.

In toroidal direction divertor is sectioned on tape, fastening to the bottom of the VV and having independent water-cooling.

Installation and dismounting of the divertor plates is assumed to be performed remotely through divertor ports.

Neutron protection

To protect the windings of EMS from the neutron VV is equipped with a compact neutron shield with a thickness of 40cm and the decrease of the neutron flux on 3 of the order. Such shield will be made on basis of hydrides ZrH_2, TiH_2, UH_3.

Protection should be carried out by a modular principle. Modules are supposed will have proper design to avoid the penetration of the neutrons outside the protective structure.

On the subsequent development of design and after the thermal calculations shield modules will be supplemented with a cooling system. Water supply is independent of the cooling system VV shell and ports.

In the lower part of the VV the role of neutron protection partially plays divertor.

Blanket

Blanket TIN-1 is designed for output for over the years of operation up to 30 kg of uranium-233 or plutonium-239 (about 4÷5kg per year).

Blanket has a modular design. Blanket modules are placed in five equatorial ports. The total area of the blanket modules is determined by required end product:

-the area of the module, faced to the plasma - 1,2 × 0, 8m;

-the module width - 0, 8m.

Geometry and the design of five equatorial ports, intended for placement of blanket modules, should allow to carry out installation and dismounting blanket modules and their remote control and service.

Neutral beam injection

For additional heating of plasma in TIN-1 neutral beam injectors (100-140 keV) are used ,

They will be located in the 4-equatorial ports of the special design. The total power - 15 MWt.

Neutrals beams are injected tangentially to the plasma :

r = R – a / 2 = 1,5 – 0,6 / 2 = 1,2 м; a = 0,6м. A preview of the size of the window to enter the beam neutrals: 1,0 × 0, 3M.

The design of injector ports allows to enter the neutral beam co- and counter plasma current.

Conclusion

- In 2012 we have to develop concept design project of TIN-1. After discussion and corrections in 2013 we will start to develop design project of TIN-1.