LIQUID METAL COOLED DIVERTOR FOR ARIES

by

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Russian Scientific Center Kurchatov Institute

Small Enterprise TORUS

"Approved"

General Director of TORUS

" 30.08 1993

LIQUID METAL COOLED DIVERTOR FOR ARIES

Final Report of the Design Study

Principal Investigator

" 30.08 1993

Work has been done by a group of specialists
from Institute of Nuclear Fusion of
Russian Scientific Center Kurchatov Institute
as a temporary working team of Small Enterprise TORUS
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ABSTRACT

A liquid metal, Ga-cooled divertor design was completed for the double null ARIES-II divertor design. The design analysis indicated a surface heat flux removal capability of up to 15 MW/m^2 , and its relative easy of maintenance. Design issues of configuration, thermal hydraulics, thermal stresses, liquid metal loop and safety effects were evaluated. For coolant flow control, it was found taht it is necessary to use some part of the blanket cooling ducts for the draining of liquid metal from the top divertor. In order to minimize the inventory of Ga, it was recommended that the liquid metal loop equipment should be located as close to the torus as possible. More detailed analysis of transient conditions especially under accident conditions was identified as an issue that will need to be addressed.

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PREFACE

This report represents the design study of a divertor system with liquid metal coolant (gallium) related to ARIES project. The work has been conducted in 1993 by a group of specialists from Institute of Nuclear Fusion of Russian Scientific Center Kurchatov Institute as a temporary working team of Small Enterprise TORUS within the scope of subcontract No.E212601 with General Atomics, San Diego, CA, USA. According to the Contract as the results of Milestones 1,2 and 3 completion the general description of the divertor concept, the key design decisions for the divertor targets and the divertor system development results have been presented in Preliminary Concept Description, Preliminary Design Report and in Intermediate System Design Report. For the Russian team this Final Report completes the first attempt of a contractual work with an American Institution and we hope it will stimulate further collaborative efforts of this kind in Fusion Reactor Design activities.

Principal Investigator

E. Muraviev

1. INTRODUCTION

The divertor design has always presented a challenge for physicists and engineers. The major concerns identified at the very early stages of tokamak reactor design studies were high thermal and particle loads resulting in thermal stresses, fast erosion and short life time of the divertor targets. In this connection attractive ideas of using liquid metal (LM) to form a stress-free and erosion-insensitive working surface of the divertor target have appeared. The first proposals of that kind implied, along with the blanket studies, the application of lithium, an important rational for this originating from experimental data [1.1] about its high absorption ability with respect to hydrogen isotopes under the divertor conditions. The divertor targets were suggested in the form of a liquid film flowing over a solid back plate [1.2-1.5] or just freely falling [1.6], though in these proposals no detailed calculation models for MHD film flow were presented. Later in the Soviet Union an assessment of various LM devices for a tokamak reactor was performed, their basic MHD parameters evaluated and a number of candidate liquid metals compared [1.7-1.10]. In particular in these studies gallium was named as the most promising alternative to lithium. Another concept has been also put forward concerning the feasibility of a LM limiter located directly within the plasma chamber as an alternative to the divertor target. With respect to MHD flow patterns the film, jet and droplet flows have been established as the pertinent types for the plasma facing devices.

The mentioned works have stimulated theoretical and experimental efforts in studying MHD flows with free surface. A review of literature on the film MHD flows is given in [1.11]. The idea of using LM droplet flow in the divertor was independently proposed by W.Wells [1.12]. And after some years the concept of LM plasma facing devices was supported by a number of countries involved into the controlled fusion research that was reflected in a few proposals for the tokamak reactor innovations [1.13-1.19]. Several calculation and design studies were carried out within the scope of international INTOR program [1.20-1.22] and later with respect to ITER project [1.23-1.25].

The investigations in support of the droplet divertor target concept developed successfully in the USSR having as a good basis the well-established technology for producing metallic granules by means of electromagnetic granulators [1.26, 1.27]. This technology has been adopted for developing the jet-droplet flow generator (JDFG) operating

in the magnetic field of a tokamak reactor [1.28, 1.29]. At the same time an experiment on T-3M tokamak was planned [1.30, 1.31] and successfully realized [1.32].

Further research efforts were done with respect to development of both droplet and film type MHD devices in the Institute of Nuclear Power Plants (with participation of the Institute for Electrodynamics of Ukrainian Academy of Sciences) [1.33], in Physics Institute of Latvian Academy of Sciences [1.34], in I.I.Polzunov Central Boiler and Turbine Institute [1.35, 1.36], St. Petersburg Technical University [1.37, 1.38], G.M.Krzhizhanovskij Power Institute [1.39, 1.40]. In the most part these activities were supported by the State Committee of Atomic Energy through Kurchatov Institute (Moscow) and Efremov Institute (St. Petersburg). Sufficient data have been accumulated to prepare experiments on large tokamaks and start designing of full-scale modules for ITER type reactor.

Nevertheless the major concern with respect to the open LM surface devices in the tokamak plasma chamber remained with the problem of plasma impurity influx that might result from the plasma contact with LM. The existing models and computer codes can not solve this problem for any target material with the atomic charge number higher than that for carbon ($Z=6$). E.g. for Ga ($Z=31$) it is far beyond any practical possibilities. So the answer should be obtained in experiments. But then experiments are needed on a large tokamak with a divertor since there were many comments from plasma physicists on impossibility to extrapolate the only existing experimental results from T3-M machine (small tokamak without divertor) towards larger machines. And no team running a large tokamak all over the world seems to be eager to start experiments of this kind because they are afraid to intervene with their current programs. Thus a closed circle is formed.

Meanwhile at the stage of ITER Conceptual Design Activities (CDA) that was a major effort of the international fusion community lately, the divertor problem was recognized as the most crucial one for the entire project. A closer assessment of the problem taking into consideration reliability and safety issues has resulted in the conclusion that the CDA divertor design is not acceptable for further development. A large list of serious unresolved problems has been given in [1.41] from which several points are as follows:

- the feasibility of peak heat fluxes up to 30 MW/m^2 considering sweeping limitations, disruption life time, fatigue life of the braze joint and critical heat flux limits;
- protection of the heat sink against run-away electrons;

- frequent conditioning of the carbon armor between pulses with low coolant temperatures;
- demonstration of the manufacturing feasibility and performance of prototypical mock-ups, with emphasis on the integrity of the bonded joint armor-heat sink under normal loads and disruptions.

In this context the advantages of LM coolants in general and Ga in particular seemed to be very important for the specific divertor environment and worth trying to employ them. Taking into account the pointed above difficulties with experimental proving of the free LM surface divertor target concept, a new divertor concept where the LM (Ga) is to be used only as a coolant has been proposed [1.42-1.44]. In this case the plasma facing surface can be made of the pertinent solid material. Luckily enough all the considered candidate materials for the divertor plate plasma facing surface, namely graphite, tungsten and beryllium turn out to be the best materials compatible with Ga that is generally known as highly corrosive LM (especially above 300°C). E.g. the corrosion rate in Ga at 400°C is 10^{-6} g/(m²·hour) for W and 10^{-4} g/(m²·hour) for Be while it is 7.1 g/(m²·hour) for Fe and 1100 g/(m²·hour) for Cu (see Section 3).

This new concept has been applied in the current design study for ARIES project with the following major objectives:

- to enlarge the scope of the project adding a viable alternative for the divertor design;
- to insure higher permissible thermal loads to the divertor target and provide thereby greater safety margins for the machine operation regimes;
- to demonstrate the feasibility of a divertor design with easy maintenance and high duty cycle potential;
- to establish the most important mutual requirements between the liquid metal cooled divertor and other tokamak reactor systems and give recommendations for further design efforts;
- to reveal specific problems and find their solutions for the liquid metal cooled divertor concept application for a commercial fusion reactor, evaluate the prospects for the proposed concept.

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2. GENERAL CONCEPT DESCRIPTION

2.1. Liquid Metal Selection

In general selecting LM for using in the tokamak divertor the following requirements should be taken into account: low Z number, high boiling temperature in vacuum, a wide temperature interval between melting and boiling points, high heat accumulation capacity, compatibility with structural materials, safe contact with air and water, low activation by neutrons, availability of industrial technological basis.

As it was mentioned in Section 1, from our earlier studies [2.1] a conclusion has been drawn that along with the traditionally considered lithium a new candidate LM - gallium can be recommended. The major advantage of lithium is its low Z number, but if we put limits on the vapor pressure then the working temperature range and heat accumulation capacity for gallium turn out to be much larger.

Lithium has a unique feature: except the tritium production it is not activated by neutrons. The gallium activity decays strongly in two weeks after the end of irradiation (See Sec. 8.2). Though in both cases some problems may arise due to activation of corrosion products.

Besides these general considerations some additional arguments in favor of Ga coolant related specifically to ARIES-II machine with the lithium cooled blanket can be given.

A natural question arises: why should we have a different cooling loop for the divertor with another LM, different pressure and temperatures. But in fact it would be a real problem to try to have a *single* cooling loop for the entire reactor. For on one hand it is just rather large and it's difficult to combine all the coolant outlets around the torus into a single loop. Due to this only reason the blanket cooling system itself must have several separated loops. Then the conditions in the inboard and outboard blanket segments are quite different, and it's difficult to have equal pressure and temperature in them. The same arguments are valid for the divertor system. Considering the possibility of combining blanket and divertor modules one should keep in mind the difference in working conditions (assumed non-uniformity of the thermal load in the divertor). All this confirms the expedience of having a number of separate loops both within blanket and divertor system. Thus the actual issue is not the number of loops

with various parameters but switching part of them to different technology. And this can be justified by the feasibility of a more reliable divertor which remains still a critical reactor component.

In comparison of Ga versus Li two levels of considerations should be addressed: a narrow and a broad one. In the narrow level only the working temperature range and vapor pressure can be taken into account that are key parameters for the design with open LM surface in the vacuum chamber. Note that we want to use the leak-tolerant joints (without welding) for easier assembly and maintenance operations, so there may always be a free LM surface irrespective of jet or tube type cooling scheme. One should consider two characteristic higher temperatures of the coolant: T_1 - the average outlet temperature and T_2 - maximum LM/wall interface temperature (the latter can be "seen" on the surface of LM flowing down out of cooling cells). For Ga $T_1=230^\circ\text{C}$ (see Section 2.5) and T_2 is near 450°C (at 15 MW/m^2 heat flux). For Li, with its melting temperature 150°C higher, $T_1=380^\circ\text{C}$ and T_2 is about 600°C . According to the vapor pressure curves taken shown in Fig. 2.1.1 in case of Li the vapor flow from the open LM surface (proportional to the equilibrium vapor pressure) can be 8 orders of magnitude higher than for gallium. If we assume that the plasma impurity danger is proportional to Z^2 then taking into account the charge numbers ratio (31 for Ga and 3 for Li) the impurity influx from lithium turns out 10^6 times more dangerous. Of course this is a very rough estimate but still the tolerance of the main plasma to various impurities influx from the divertor has not been established yet, and now we can only be certain that in the proposed concept Ga vapor flow into the plasma can be kept at a very low level while for lithium it can not.

Besides this there is additional problem of pressure gradients in magnetic field. We can rely on preliminary experimental results related to the formation of electroinsulating films on V-alloy in Ga that show the feasibility of the contact resistance at $10^{-4}\text{ Ohm}\cdot\text{m}^2$ while for Li the corresponding results give only $10^{-5}\text{ Ohm}\cdot\text{m}^2$. That means substantial difference in the pressure level in the feeding tubes for the jet cooling option and in the entire cooling duct for the tube cooling option. And finally, it is much more difficult to provide the LM self-draining for Li than for Ga because of 12 times lower mass density and higher MHD-resistance to the flow with poorer electric wall insulation.

In the broad consideration we would like to draw attention to Ga cooling option as free from concerns related to Li (large stored chemical energy, fire hazards and tritium release danger). The suggested divertor design option remains viable for ARIES-II-like reactor with

Fig. 2.1.1. Vapor pressure for Li and Ga versus temperature (A.N. Nesmejanov. Davlenie para khimicheskikh elementov. M., Izdatel'stvo AN SSSR, 1961)

V-based structural material and lead coolant [2.2] as well as for a reactor with helium cooled blanket like ARIES-IV. If some next commercial, DEMO or experimental reactor project comes up with a water or steam cooled blanket, this Ga-cooled divertor will be still compatible with it.

It should be mentioned also that gallium is a rather expensive material though the technological basis for its production exists and can be enlarged which may result probably in the cost reduction. According to some unofficial expert estimates in Russia dozens tons of gallium could be available within period of several years at a cost near \$100 per kg. Considering everything gallium has been selected as the working LM for this design study.

2.2 Key Features of the Concept

The key features of the proposed divertor design concept based on the specific LM coolant properties are as follows:

- the requirement of the vacuum tightness of the divertor cooling tract is dismissed;
- the pressurized coolant ducts can be separated from the plasma facing structure (PFS) elements which are subject to the thermal loads, and with this feature PFS can be replaced independently, without disturbing the cooling system; this is achieved with using free LM jets sprayed on the back side of the PFS elements, free LM film cooling and free LM draining under the action of gravity force; obviously this point is more suitable for single-null bottom divertor;
- additional LM inlet pressure (above the necessary for pumping through the feeding tubes in the magnetic field) can be used for either heat transfer enhancement via high velocity jets and/or PFS moving to reduce the local thermal/particle loads.

To appreciate the potential role of the listed basic features of the new concept it is rather important to take into consideration the real toughness of engineering challenge that is faced in the divertor design development not for a physical tokamak facility but for a power tokamak reactor. In this case we have an unprecedented combination of the strict vacuum requirements of tokamak with large cooling system spread over dozens of square meters of cooled surface in vacuum. Severe working conditions of the PFS elements can be compared (in a mild approximation) with those for the cladding of fuel elements in water cooled fission reactors. And it is known that for the latter permanent leakage for about 0.1 per cent of the total number of fuel elements is considered normal because it seems practically unavoidable. The question is: how are we going to operate a water or gas cooled divertor target in a tokamak if no such leaks can be afforded?

In this respect the very first basic feature of the proposed concept gives us the following major advantages against water/gas cooling options:

- non-sensitivity to a small leakage in the divertor target cooling tract within the vacuum chamber and therefore elimination of the possibility of a whole type of the internal LOVA that might have the strongest impact on the reactor reliability and availability;

- elimination of the possibility of another whole type of accidents with the vacuum chamber pressurizing due to internal LOVA with the cooling tract rupture;
- easy replacement of the divertor target modules without welding/brazing operations.

The second basic feature which is applicable to the full extent for the single-null bottom divertor configuration provides additional important advantages:

- feasibility of PFS elements replacement separately from the cooling system;
- longer PFS elements fatigue life in the absence of brazed joint with the heat sink structure;
- longer operating life of the pressurized coolant feed pipes due to absence of thermal stresses.

Besides these specifically for comparison with ITER-like water cooling such advantages should be emphasized as:

- no coolant interaction with hot plasma facing material in case of the internal LOVA and no hydrogen production;
- no need for the cooling system protection from the run-away electrons
- no critical heat flux limits.

The third basic feature allows in principle to increase greatly the permissible heat flux by means of moving the PFS against the incident plasma flow. With this purpose in [2.3-2.5] a concept of the divertor target in the form of a rotating drum with the outer surface facing the plasma and the inner surface cooled by free LM flow has been considered. LM in this case is used also as electric conductor forming within the drum a unipolar motor circuit for rotation drive in the external magnetic field.

The boundary between the steady and moving PFS options in the design window can be approximately defined by the heat flux level at 10-15 MW/m². For ARIES conditions where the gaseous divertor has been assumed with rather uniform thermal load distribution over a large area the simplest steady PFS option seems quite sufficient. So in this design study we stayed with it only.

We had to embed our system into a completed and self-sufficient reactor design affording no significant changes in it. In this respect we faced two specific points to which our concept had to be accommodated, namely:

- 1) double-null divertor configuration;
- 2) gaseous divertor configuration with specially organized "pockets" for the plasma flow moving along the separatrixes.

From the general lay-out of the divertor target system within the vacuum chamber of ARIES reactor shown in Fig. 4.1 one can see that in the top divertor LM from the central "hump" target area could not be self-drained under the only action of gravitational force as it was supposed in the initial proposal. Hence we could not apply to the full extent all the basic concept features listed above. Namely we had to use closed pressurized cooling ducts to remove the thermal load from the target. But still the demonstration of all the potential advantages of our concept seemed worth doing.

The divertor design has been developed formally as particularly applicable to ARIES-II reactor, the major reason for this being the choice of a vanadium-based alloy as the structural material compatible with gallium. Though there are some good prospects that carbon based materials including SiC-composite might be compatible with gallium as well. Then this concept could be used also in ARIES-IY and this possibility should be kept in mind for future.

Though the concept of gaseous divertor accepted in ARIES project assumes very low and uniform thermal loads to the divertor plates and no sputtering, some strong doubts about its viability still exist. E.g. in a new Russian project of a DEMO tokamak reactor [2.6] this approach has been rejected and for the plasma parameters close to those of ARIES the peak heat flux to the divertor along the separatrix was established at 500 MW/m² level in contrast to 2-3 MW/m² for ARIES. So for this study it was supposed expedient to try to apply the specific advantage of the new divertor concept, namely its tolerance to higher heat flux and shorten erosion life time of the plasma facing components, to ARIES conditions assuming that in the gaseous divertor some non-uniformity of thermal load might occur. Specifically the target value of the peak heat flux of 15 MW/m² has been established.

An important issue for this design study was adjusting the divertor target of free LM flow cooled cells' type to the specific double-null gaseous divertor geometry. The following basic decisions have been made:

- the entire plasma facing structure consisting of two type targets subject to low and high heat flux respectively;
- to apply the principles of leak tolerance and LM self-draining from the reactor special drain channels within the inboard and outboard blanket envisioned for draining LM from the top divertor;
- to realize a reliable design for the high heat flux target the easily replaced single-wall de pressurized cooling cells with free surface LM flow employed at the ends of the divertor "pockets" where the maximum heat flux can be expected;
- the low heat flux targets of large surface and various geometry made in the form of double-wall panels with internal pressurized cooling tubes and external de pressurized shell filled with slowly flowing LM as a thermal contact medium.

This approach allowed to minimize the diversity in the components design for the adopted condition of non-uniform thermal load and have almost identical design for the top and bottom divertors. The requirement for the draining ducts in blanket is of course a weak point, but it can be justified by the following two reasons:

- 1) the introduction of these ducts in itself is not a serious problem since no vacuum-tight welded or brazed joints are required; it is much simpler and more reliable than e.g. using an electromagnetic pump in situ to pump out LM from a pool in the vacuum chamber or having a target with pressurized cooling tubes in the high heat flux region;
- 2) the simplicity of self-draining scheme for the bottom divertor is so attractive that it seems worth demonstrating it in contrast to the difficulties it might bring for the top one; it should be taken into account probably next time when trading-off the single-null configuration versus double-null; e.g., ITER now turned from the double to single and the latter has been assumed also in [2.6].

As the structural material for the divertor system the same vanadium alloy is accepted as for the blanket design of ARIES-II. A protective (and may be renewable in situ) Be coating of the PFS can be considered if it is acceptable for the general approach to the ARIES plasma chamber technology. Should Be utilization has been adopted by ARIES the PFS elements with free LM flow cooling could be made entirely of Be.

Specifically for this design an electroinsulating coating is envisaged on all metallic surfaces in contact with LM. Its feasibility is assumed on the basis of experience of developing such coatings on vanadium alloys in lead and lithium [2.6,2.7].

2.3. Summary of the Design Parameters

The following basic design parameters of ARIES-II needed for this study have been obtained from General Atomics [2.8]:

Major Radius, R, m	5.8
Minor Radius, a, m	1.4
Plasma vertical elongation, b/a	2.03
TF Coils Number	16
The Reactor Operation Scenario:	steady state
Fusion Power, MW	1949
Toroidal Field on Axis, T	8
Plasma Current, MA	6.46
Divertor Plates Neutron Load, MW/m ²	2
Power Conducted to the Divertor, MW	188
to outboard divertor plates, MW	117
to inboard divertor plates, MW	71
Divertor plates area in gaseous divertor configuration:	
outboard side, m ²	58.4
inboard side, m ²	35.6
Number of Disruptions,	1

To obtain the poloidal magnetic field distribution the data [2.8] on PFC location and their currents were used that are shown in Table 2.3.1 (data are given for the upper set of PF coils, that is 3 central solenoid coils and 3 large ring coils, the other 6 PF coils of the lower set have exactly the same parameters except negative values of their heights over the reactor middle plane).

Table 2.3.1. Poloidal Field Coils Parameters

Equivalent current radius, m	Equivalent current height over middle plane, m	Current, MA
1.54	1.05	-4.227
1.54	3.82	-1.833
1.54	4.41	1.332
6.97	7.53	12.65
10.63	4.77	-12.38
11.03	4.11	0.884

The space-energy distribution of neutron flux necessary for the divertor coolant and structure activation calculations has been obtained from [2.9]. These data are presented in Section 8.2.

Besides this the following parameters have been assumed for the in-reactor part of the gallium loop:

inlet temperature, °C	80
outlet temperature, °C	230
Ga properties at average working temperature:	
density, ρ , kg/m ³	5900
specific heat capacity, c , J/(kg K)	346
electric conductivity, σ , MCm/m	3.3
thermal conductivity, λ , W/(m K)	27
dynamic viscosity, μ , kg/(m s)	$1.6 \cdot 10^{-3}$
expected resistance parameter for self-healing	
electroinsulating films on vanadium alloy, $\rho_i \delta_i$, Ohm·m ²	10^{-4}

The following main resulting design parameters for the divertor system have been determined:

Total thermal power removed by LM divertor system, MW	376
Total cooled surface area, m ²	278
Number of the divertor target modules in the top/bottom divertor	32/32
Total LM coolant flow rate, m ³ /s	1.3
Average LM coolant temperature rise, °C	150
LM inventory inside reactor, m ³ , including:	6.79
top divertor	1.36
bottom divertor, pressurized pipelines and targets	2.96
bottom divertor, drain system	2.47
Total LM inventory in the divertor system, m ³	28.2
Total required pumping power, MW	2.63
Estimated maximum tritium inventory in the LM loop, g	16-30

Some major parameters of separate divertor cooling lines are shown in Table 2.3.2.

Table 2.3.2.
Main Resulting Design Parameters for Separate Divertor Target Cooling Lines

Parameters	LHF inboard target	HHF inboard target	LHF central target	HHF outboard target	LHF outboard target
Surface area, m ²	23.65	5.79	68.16	7.84	33.71
Conducted power, MW	25.75	15.86	74.23	35.46	36.70
Coolant flow rate, m ³ /s	0.074	0.083	0.208	0.177	0.105
Coolant velocity, m/s	1.6	5-3	1.9-1.4	5-3	1.5
Heat transfer coefficient, kW/m ² K	30	135-65	18	135-70	30
LM temperature rise, °C	173	100	178	100	174
Pressure drop, MPa	1.0 (t) 1.2-(b)	1.0 (t) 1.2-(b)	1.0 (t) 1.2-(b)	0.7 (t) 1.0-(b)	0.7 (t) 1.0-(b)
In-reactor pressurized coolant line length, m	4.15 (t) 9.94-(b)	2.50 (t) 3.45-(b)	6.85 (t) 8.88-(b)	1.90 (t) 2.80-(b)	3.45 (t) 7.88 (b)

2.4. Critical Design Issues

The feasibility of the proposed concept will depend on successful solution of a number of problems that require certain R&D efforts. The most important issues in this respect are the following:

- technology of thin electroinsulating films or coatings formation on metallic walls in contact with Ga;
- stability and self-healing of the electroinsulating films under neutron irradiation;
- data base on corrosion and mass-transfer under the conditions of non-isothermal gallium loop with working section in magnetic field;
- data base for the LM heat transfer in MHD film flow;
- data base for MHD jet flow - wall interaction;
- data base for hydrogen isotopes solubility in Ga and its adsorption properties in the vacuum chamber environment;
- technology for Ga purification from PFS erosion products and tritium.

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3. MATERIALS REVIEW

3.1. General Materials Compatibility with Gallium

A serious drawback of gallium is the fact that it actively dissolves most metals and alloys at temperatures above 100°C. The most stable metals (mainly refractory ones) are shown in Table 3.1.1. In this Table the maximum permissible temperatures ($T^{\max}$) at which these metals can be recommended for long term application in contact with Ga, their solubility and corrosion rates at 500°C and 700°C are given [3.1-3.3].

Considering the divertor plate materials it should be pointed out that:

- the most stable are tungsten and rhenium which do not react with gallium up to 800°C and begin to react slowly at higher temperatures with formation of volatile oxides;
- metallographic analysis of tungsten samples exposed to gallium at 1100°C within 200 hours has shown a considerable growth of boundary grains that gives an evidence of inter crystalline corrosion [3.4].

Beryllium above 600°C forms with gallium an interaction zone, with the depth increasing with temperature. This zone strongly suppresses further process of inter diffusion and provide high stability of beryllium in gallium and its alloys. Different data on beryllium stability are available. According to [3.5] the beryllium solubility after 14 days exposure at 800°C was 6.6% (at.), while in [3.4] four times lower solubility, only 1.41% (at.) at the same temperature has been obtained.

Molybdenum and chromium begin to interact with gallium actively at temperatures above 600°C. Characteristic reaction zones have been observed on sample surfaces after few tens of hours exposure. The corrosion rates for these metals strongly depend on their micro structure and surface treatment. The most stable surface films could be obtained by diffusion chroming process.

Vanadium, titanium and hafnium could be used in liquid gallium at temperatures up to 450°C.

Table 3.1.1.
Stability of Some Pure Metals in Liquid Gallium

Metal	T _{max} , °C	Solubility, at. %		Corrosion rate, g/(m ² ·h)	
		500°C	700°C	500°C	700°C
W	800	10 ⁻⁶	6·10 ⁻⁵	10 ⁻⁴	2.6·10 ⁻³
Re	700	10 ⁻⁵	1.2·10 ⁻³	9·10 ⁻³	5·10 ⁻²
Be	600	10 ⁻⁴	6·10 ⁻²	3·10 ⁻²	0.2
Ta	600	10 ⁻⁴	4·10 ⁻³	0.5	1.4
Nb	600	10 ⁻⁴	4·10 ⁻³	0.6	1.8
Mo	600	2·10 ⁻³	10 ⁻²	0.8	2.8
Os	550	4·10 ⁻⁴	10 ⁻²	0.8	2.8
Ru	500	3·10 ⁻³	0.11	1.6	7.5
V	500	9·10 ⁻³	0.12	1.9	8.4
Hf	500	10 ⁻³	0.12	1.6	12.0
Ti	400	4·10 ⁻²	0.23	2.6	20.0
Ir	400	10 ⁻²	0.21	2.0	20.0
Cr	400	10 ⁻²	0.21	2.0	25.0

3.2. Corrosion Resistance of Structural Materials

The steels corrosion resistance has been evaluated according to a 5-point scale depending on the corrosion rate as it is shown in Table 3.2.1 [3.6].

Table 3.2.1.
5-Point Scale for Steel Corrosion Rate Evaluation

Resistance Group	Corrosion rate, mm/year	Corrosion intensity level
Rather resistant	<0.1	1
Resistant	0.1-1	2
Reduced resistance	1-3	3
Low resistance	3-10	4
Non-resistance	>10	5

It was established that austenitic steels (of 1X18H9T type which is analog to SS-316) do not react with gallium and gallium based alloys up to 250°C within 2000 hours under static conditions [3.4]. Though LM loop experience on T-3M-2 tokamak device has shown that gallium had different impurities from structural materials at lower temperatures.

Steels stability at 250°C and 400°C in gallium eutectic alloy prepared from high-purity components (with 25% In and 13% Sn and less than 0.05% of impurities) has been studied in [3.4] under dynamic conditions. In test sections the alloy velocity was ~0.5 m/s. Some results for chromium containing steels are shown in Table 3.2.2.

Table 3.2.2
Corrosion Rates and Resistance of Some Steels in Gallium Eutectic Alloy Under Dynamic Conditions

Steel	Temperature, °C	Exposure, hours	Corrosion rate, mm/year	Resistance Level
40X	250	200	1.2	3
35XM	250	200	1.3	3
38X2H3M	250	200	1.1	2
60X4A8H8B	250	500	0.8	3
34XH3M	250	200	1.1	3
Cr*	250	500	0.001	1
Cr*	400	500	0.008	1

* samples were electrolytically chromed

Table 3.2.3 presents the results of corrosion test experiments for different stainless steels classes in gallium eutectic alloy at 250°C and 400°C under dynamic conditions [3.7]. As it is seen the corrosion resistance depends on substitution solutions containing considerable amounts of chromium. The velocity of gallium eutectic alloy in these experiments was ~0.15 m/s. The following steels were related to the rather resistant corrosion class where corrosion rates are below 0.1 mm/year:

- at 250°C: X22, X25T, X18H9T, 4X15H7A7O2MC, 3X17A15H4AO2, 4X12H8A8MOA;
- at 400°C: X25T, X18H9T, 4X15H7A7O2MC, 3X17A15H4AO2, 4X12H8A8MOA.

Other steels can be considered only as low corrosion resistant materials. It was observed that thermal treatment may have a strong impact on steel corrosion rate. But more strongly corrosion resistance depends on chromium content. As it is seen from Table 3.2.3 the corrosion rate is decreased for steels with large chromium (X) contents and usually decreased with aluminum addition.

Table 3.2.3.
Corrosion Resistance of High Doped Steels in Gallium Alloy
Under Dynamic Conditions (0.15 m/s at 250°C and 400°C)

No	Steel	TT	Temperature °C	Exposure, hours	Corrosion rate, mm/year	RL
1	1X17H2	T	250	500	0.1	2
			400	200	1.3	3
2	2X13	T	250	500	0.17	2
			400	200	1.8	3
3	X22	A	250	500	0.06	1
			400	200	1.00	2
4	X25T	A	250	500	0.05	1
			400	200	0.8	2
5	1X18H9T	T	250	500	0.06	1
			400	200	0.8	2
6	X16H16	T	250	500	0.11	2
			400	150	1.30	3
7	4X15H7Ã7Ô2MC	T	250	500	0.08	1
			400	150	0.95	2
8	3X17H4Ã15AÔ2	T	250	200	0.08	1
			400	180	1.3	31
9	4X12H8Ã8MÔÁ	T	250	500	0.05	1
		A	250	500	0.1	2
		T	400	200	0.87	2
		A	400	200	1.65	3

Here: 1,2- martensitic steels,
3,4- ferritic steels,
5-9- austenitic steels;
TT- thermal treatment: T- tempering at low temperature;
A- the sample is tempered or annealed;
RL- resistance level.

Kinetics of steels corrosion in gallium eutectic alloy was studied under static and dynamic conditions. The experimental data for 1X18H9T and X22 steels are given in Table 3.2.4 [3.4]. Under static conditions the corrosion rate is decreased in time by a factor of two. This effect is less prominent under dynamic conditions depending on LM flow parameters and can be attributed to the following:

- restricted diffusion through the thicker film;
- decreasing of the solution rate due to LM saturation.

Table 3.2.4.
Kinetics of Steels Corrosion Rate in Gallium Alloy Under Static
and Dynamic Conditions (at 400°C and 0.15 m/s)

Steel	Exposure, Hours	Static Conditions		Dynamic Conditions	
		Sample Length Reduction, mm	Corrosion Rate, mm/year	Sample Length Reduction, mm	Corrosion Rate, mm/year
1X18H9T	30	0.004	1.2	0.005	1.5
	50	0.004	0.7	0.008	1.4
	70	0.006	0.7	0.008	1.0
	120	0.009	0.6	0.011	0.8
	200	0.013	0.6	0.018	0.8
X22	30	0.005	1.5	0.005	1.5
	50	0.008	1.4	0.009	1.6
	70	0.009	1.1	0.013	1.6
	120	0.014	1.0	0.018	1.3
	200	0.020	0.9	0.023	1.0

From the above mentioned experimental data it can be concluded that one of the weak gallium points would be its high corrosion activity. The LM temperature in a gallium coolant loop made of stainless steel must not exceed 300-350°C level. At these temperatures some austenitic stainless steels can be considered as corrosion resistant materials.

Vanadium and vanadium based alloys have better corrosion stability in liquid gallium than steels. The maximum temperature limit can be increased up to the 450°C.

3.3. Thermo-Mechanical Properties of the Selected Vanadium Alloy.

Vanadium alloys have been accepted as basic structural materials for ARIES-II design. In according with this we have selected V-15Cr-5Ti alloy for the divertor system. This alloy has acceptable mechanical characteristics at high temperatures [3.8,3.9], which are shown in the Table 3.3.1, along with thermal properties.

Table 3.3.1.
Thermo-Mechanical Properties of V-15Cr-5Ti Alloy

Temperature, °C	20	200	400	600	700
$\sigma_{0.2}$, MPa	560		340	340	330
σ_b , MPa	650		520	560	560
δ , %	32		26	25	28
λ , W/m K	18	23	26.8	29.5	
C, J/kg K	470	495	535	575	
$\alpha \cdot 10^6$, 1/K	10.0	10.1	10.2	10.5	
E, MPa	129	128	127	123	
μ	0.31	0.36	0.36	0.35	

where $\sigma_{0.2}$ -0.2% offset yield stress; σ_b -ultimate tensile strength; δ -elongation.

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4. DESCRIPTION OF THE DIVERTOR DESIGN

4.1 Double-Null Divertor Configuration Lay-Out

In the developed design the overall configuration of the liquid metal cooled divertor target system is practically identical to that of the reference gas divertor (Fig. 4.1, 4.2, 4.3). Both the top and bottom divertor incorporates LM feeding pipelines, targets and drain ducts (for sake of simplicity some pipelines are not shown in figures).

For convenience of assembly and maintenance each divertor structure is split into separate modules. The total number of modules in a divertor is 32. One module incorporates 5 targets of two different types: for low heat flux (LHF) and high heat flux (HHF). To provide continuous toroidal divertor target belt there are two types of modules arranged in pairs: one of rectangular and the other of trapezium shape in plane (Fig. 4.4). The module is fixed on a tube frame attached to the inner and outer blanket walls. One of the frame supports is moveable for allowing free thermal expansions and facilitating the assembly. Each module has its own system of feeding and draining pipelines. The LM feeding pipelines and some parts of the draining ducts are attached to the modules. LM draining from the top divertor is carried out by using part of the blanket ducts. In the bottom divertor special draining pipelines are envisaged that consist of arrays of tubes with comparatively small diameter and smooth bending. Both the feeding and draining pipelines are grouped into clusters that pass through the divertor access ports making room for other vacuum chamber communication lines.

The non-pressurized draining ducts inside the vacuum chamber have collective joints allowing easy connections and disconnections without welding or brazing operations. Small LM leakages from these joints are removed through discharge holes into the same draining duct. Some part of the LM pumped into the divertor is passed through the supporting frames providing removal of the heat released in the metal structure due to neutron and gamma radiation.

4.2. Divertor Targets Design

4.2.1. High Heat Flux Targets

The maximum heat flux from the plasma is expected within a narrow belt of about 60-80 mm width near separatrix. For the accepted maximum heat flux level of 15 MW/m^2 calculations have shown that only thin metal plates cooled from the back side by LM flow with high velocity can provide a viable engineering solution. Thus the HHF target design has been developed as a set of inclined plates (troughs) with the plasma facing wall thickness of 1 mm, 250 mm length (established with respect to possible oscillations of the separatrix) and 80 mm width, with the side stiffening walls 5 mm high. The troughs are fixed cantilevered by means of legs formed as some higher parts of the side walls (Fig. 4.5). The holder is combined with the manifold tube. There are T-shaped recesses in the holder wherein the trough legs enter. To increase the structure stiffness the manifold and the holder plate are connected by ribs. The target plates are prevented from falling out of the recesses by means of an angular piece attached to the holder plate. The shape of the target plates in plane is rectangular (Fig. 4.6) except those that are located at the module sides. These side plates have trapezium shape as shown in Fig. 4.7. Thus the continuous belt of plasma facing target surface along the major torus circumference is provided. The manifold tube has a set of orifices from which the LM jets are ejected on the cooled trough surfaces. Hitting the plates LM jets form a thin film that flows down due to its own inertia as well as under the action of gravity force. At the end of the trough the film enters a flatten receiver-manifold from which the LM flows into the draining ducts.

4.2.2. Low Heat Flux Targets

The LHF targets are made in the form of curved panels consisting of a set of flat wedged cooling tubes encased with clearances into a protective shell (Fig. 4.8). On the outer surface of the cooling tubes there are segmented longitudinal ribs of 2 mm width, 20 mm segment length, located with 20 mm pitch. The rib height is 1 mm on the plasma facing side and 2 mm on the back side of the panel (Fig. 4.9). The protective shell wall at the plasma facing side is 1 mm thick and the one at the back side is 2 mm. They are joined with the cooling tubes by diffusion welding and then the entire stack is curved by pressing in the vertical plane according to the required target configuration. Since the toroidal width of a single target is only about 1 m it does not need to be curved in this direction, thus a complicated technology for shaping double-curved panels is not necessary. The flat cooling tubes have branch pipes on both ends for LM inlet and outlet. The pressurized duct formed by the cooling tubes and

branch pipes is separated from the low pressure space between them within the protective shell (Fig. 4.10). The branch pipes are welded into the inlet and outlet manifolds. The LM inlet to the intertube space is located always at the bottom manifold (irrespective to the cooling flow direction) and the outlet is at the top manifold, so filling of this space with LM as the thermal conductance medium can be insured even during abnormal working regimes (e.g. at a loss of flow in this low pressure tract). In case of a cooling tube rupture the LM can flow through the intertube space to the drain system without spraying into plasma.

4.3. In-Reactor Module Design

4.3.1. Rectangular Module

The modules of the top and bottom divertors are almost identical, so the detailed design description is given here for a top module shown in Fig. 4.11. The module consists of two HHF targets (at the inboard and outboard side of the divertor) and three LHF targets (the inboard, the intermediate and the outboard ones) fixed on the common tubular frame attached to the blanket walls.

All the targets are made as independent units with their own LM feeding/draining pipelines. They are fixed on the supporting frame by means of special brackets at the pipelines near the manifolds. The only exclusion is the inner LHF target which is fastened by the same brackets to the blanket wall (Fig. 4.12).

The bracket is a support with a set of yoke inserts bounded by studs. Such a device allows to fix in one place (and in one plane) one or several pipelines providing the feasibility of easy remote dismantling of a target from the module. The brackets receive the mechanical loads caused by the thermal expansion of the LM pipelines.

Free thermal expansion of the HHF targets is allowed in longitudinal direction by their cantilever fixation and in lateral (toroidal) direction due to guaranteed clearances.

Compensation of thermal expansion of the inboard and outboard LHF targets curved in vertical plane is provided by sliding supports at their bottom ends free from the fixing brackets. The central LHF target with both ends fixed by the brackets has nevertheless sufficient ability for self-compensating due to change of its curvature.

The module as a whole is a complete unit to be installed in the reactor. The feeding pipelines of the module can be welded to the corresponding pipes of the external part of the

LM loop outside the reactor (the design of vacuum penetrations has not been considered in this study). The draining pipeline segments belonging to the module are attached to the permanent part of the drain system inside the vacuum chamber by means of shifting telescope junctions.

The module supporting frame structure consists of two bearing tubes with the ends rested on the blanket walls and three lateral elements on which the fixing brackets are placed (Fig. 4.13). The tubes are cooled by a fraction of LM flow fed to the targets. The frame structure has two pairs of hinged supports, one of which is supplied with telescopic movable devices to insure free thermal expansion and assembly clearances.

4.3.2. LM Sink Devices of the Module Drain Pipes

In accordance with the developed concept the LM tract within the vacuum chamber does not have to be leak tight. Such an approach allows to realize very simple design of the joints between the draining pipeline segments belonging to the module and the permanent part of the divertor drain system.

There are two types of sink devices: for the HHF and LHF targets. The sink device for troughs (HHF target) is a set of curved tubes with flattened rectangular tips that form an intake slot across the entire toroidal width of the target (Fig. 4.14). The rectangular inlet cross-section is changed smoothly to the circular one down the pipe length. In the LHF targets similar tubes with varying cross-section shape are connected directly to the outlet manifolds and belong to the target while the sink devices are made on the circular parts of the drain pipes. Such a device has a cylindrical cup with a labyrinth sealing at the bottom that can be moved on the end of the target drain pipe (Fig. 4.15). The diameter of the cup is larger than that of the target drain pipe and in the gap between them a hydraulic trap with LM column can be formed that will prevent LM leak from the duct.

The mentioned permanent parts of the divertor drain system are installed in the reactor before the divertor target modules. And after the latter are placed in the vacuum chamber they are connected to them by means of the described movable devices.

4.3.3. Trapezium Module

The design of the trapezium module (Fig. 4.16, 4.17) is quite similar to the rectangular one. The only differences are in length of lateral elements on which the fixing brackets are placed and in configuration of those brackets.

4.4. Overall Gallium Loop Layout

The scheme of typical LM cooling system is shown in Fig. 4-18. Heat from the primary LM coolant is transferred through heat exchanger to water secondary loop. Gallium comes from the divertor at 230°C and is cooled in the heat exchangers down to 80°C. There are six major subsystems in the primary loop listed below:

- LM cleanup/conditioning system (LMCS);
 - Heat exchangers (HE);
 - LM make-up loop (LMMS);
 - Tritium extraction/processing system (TEPS);
 - Main circulating/auxiliary pumps (MCP);
 - LM dump tank and storage (LMDT).
- DM - divertor module; LMMS - LM make-up system; TEPS - tritium extraction/processing system; HE - heat exchanger; MCP - main circulating pump; LMCS - LM cleanup system; TP - transfer pump; LMDT - LM dump tank

The main circulating pumps must handle about 1.3 m³ or 7670 kg of Ga per second. It has been assumed that the liquid metal pump and heat exchangers technology developed for LMFBF will be applicable for Ga coolant which is expected to be contaminated with tritium and radioactive corrosion products.

Tritium inventory in the LM coolant is a critically important parameter from safety viewpoint. Its low level must be insured by tritium extraction system. In fact only a small part (few per cents) of the main LM flow have to be passed through tritium extraction units.

As it was shown earlier [4.1] Ga is strongly activated by neutrons that produce some short-living isotopes. After their decay the residual activity is governed by various impurities in LM and corrosion products. The primary purpose of Ga cleanup system is to remove the

Fig. 4-1

Fig. 4-2.

Fig. 4-3.

Fig. 4-4.

Fig. 4-5.

Fig. 4-6.

Fig. 4-7.

Fig. 4-8.

A - A

Fig. 4-9.

Fig. 4-10.

Fig. 4-11.

B - B

Fig. 4-12.

Fig. 4-13.

Fig. 4-14.

Fig. 4-15.

Fig. 4-16.

Fig. 4-17.

Fig. 4.18. LM divertor cooling system

latter. It may be useful also for the control of oxygen content that is required in case of using oxygen-based electroinsulating films in the LM tract. Thus a small amount of LM has to be purified in a by-pass line.

The LM make-up system and storage represent a minor part of the overall LM loop and weren't considered in detail here. Basically this system would be used to compensate small Ga losses due to operation of the LM cleaning system or occasional leaks.

The general layout of the LM divertor loop must be closely related to the other reactor systems. Unfortunately in the course of this design study practically no data on these systems were available. Hence in the first approximation the authors used the results of an assessment performed for ITER [4.2] wherein a LM divertor system incorporated into a well developed reactor auxiliary equipment layout had been considered. The main features of the assumed arrangement include the following: the central position of the reactor surrounded by annular structure housing the equipment closely related to the tokamak. The rooms adjacent to the torus require as a rule an inert atmosphere. It is true in particular for the room containing vacuum pumps and other equipment connected with the plasma chamber. This kind of premises closest to the tokamak is the most convenient place for the considered LM loop equipment because such a location allows to minimize the total LM inventory and eliminate safety problems that might be caused by Ga activation.

The entire LM divertor system is split into 16 separate loops according to the number of toroidal field coils and the vacuum chamber access ports located between them. Each LM loop serves for both top and bottom divertor and incorporates the described subsystems.

Since the different cooling lines of the top and bottom divertors require different inlet pressure to circulate LM, the pump system of each loop is equipped with several pumps connected in series to permit intermediate coolant take-off.

Certain efforts were devoted to minimize the overall LM inventory. Two schemes of the coolant loop layout were analyzed. In the first case all LM subsystems were to be installed under the vacuum pumps room (exactly as for ITER [4.2]). In the second scheme all the equipment is installed at the same floor as the vacuum pumps and only dump tanks and storage systems remain below. The second option seems more favorable in terms of LM inventory that in this case can be reduced by about 20%. So it has been accepted as the reference design solution. The distribution of LM inventory between parts of the entire system is presented in Table 4.4.1.

Table 4.4.1
Gallium Inventories in ARIES LM Divertor System, m³

Component	Top divertor	Bottom Divertor
Module:		
Divertor targets	0.024	0.024
Feeding pipes and manifolds	0.129	0.129
Drain pipes	0.245	0.051
Total in module	0.398	0.204
Total in-reactor components	0.602	
External cooling pipelines:		
-feeding pipes	0.292	0.125
-drain pipes	0.125	0.125
Total in pipelines of one loop	0.667	
Heat exchanger	0.255	
Tank	0.061	
LM clean up system	0.08	
Tritium extraction system	0.097	
Total in one loop	1.764	
Total in reactor	28.224	

References for Section 4

- 4.1. E.V.Muraviev, V.S.Petrov, P.V.Romanov et al. Liquid Metal Cooled Divertor for ARIES. Intermediate System Design Report.
- 4.2. Technical Report. RSC Kurchatov Institute/TORUS. Moscow, May 1993.

5. IN-REACTOR THERMAL HYDRAULICS

5.1. Thermal Loads and Flow Rates Distribution.

In accordance with the initial data for the divertor design (Section 2.3) the power conducted to each divertor (top or bottom) is about 188 MW. The plasma facing area of the HHF target belts is 5.79 m² at the inboard side and 7.84 m² at the outboard side of each divertor. The LHF targets form the inboard, outboard and central panel belts. Their surface is 23.65 m², 33.71 m² and 68.16 m² respectively.

As it was told in Section 2 the efficiency of the gaseous divertor in this study has been assumed less than 100% which results in a highly peaked heat flux at the region of separatrix intersection with the divertor targets. The accepted heat flux distribution in the plane normal to the separatrix is shown in Fig. 5-1. The maximum heat flux is about 15 MW/m² for the outboard HHF targets. For the inboard HHF targets the maximum heat flux is supposed to be about 7.5 MW/m². Hence the power conducted to the inboard HHF targets in each divertor will be 15.86 MW and to the outboard HHF targets – 35.46 MW. Accordingly the average thermal load on the LHF targets is decreased down to 1.09 MW/m² that will determine the power conducted to each of the three LHF target belts. The calculated thermal loads and corresponding LM flow rates, average coolant velocities and coolant temperature rise values are shown in Table 5.1.1.

Table 5.1.1.
Thermal Loads and Flow Rates Distribution (for one divertor)

Parameters	Surface area, m ²	Power, MW	Flow rate, kg/s	Coolant velocity, m/s	Coolant temperature rise, °C
Inboard LHF targets	23.65	25.75	438	1.6	173
Inboard HHF targets	5.79	15.86	490	5-3	100
Central LHF targets	68.16	74.23	1225	1.9-1.4	178
Outboard HHF targets	7.84	35.46	1043	5-3	100
Outboard LHF targets	33.71	36.70	620	1.5	174

The average coolant temperature rise over the whole divertor is about 145°C.

Fig. 5-1.

5.2. Film Flow MHD-Braking

Assuming high resistivity of the electroinsulating film on the cooled surface, high Hartmann's numbers and smooth variation of the magnetic field along the plate the motion equation for the averaged flow velocity in a stationary film can be written as follows:

$$\rho u \frac{du}{dx} = -\rho g \sin \alpha - (j_t B_n + j_n B_t)$$

where subscript t denotes the toroidal direction and n the normal to the cooled plate surface.

The electric currents are closed via thin Hartmann's layers both at the cooled surface itself (interaction with the normal component of the magnetic field B_n) and at the side walls of the cooling cell (interaction with toroidal magnetic field B_t). Additional currents flow through the non-ideal electroinsulating films and the walls though at high film resistance the wall conductance value is not important [5.1]. Hence the current density within the flow core is:

$$j_n = \sigma u B_t (Ha_t^{-1} + C_t), \quad j_t = \sigma u B_n (Ha_n^{-1} + C_n),$$

where

$$Ha_t = \frac{w}{2} B_t \sqrt{\frac{\sigma}{\mu}}, \quad Ha_n = h B_n \sqrt{\frac{\sigma}{\mu}}, \quad C_t = \frac{h}{\sigma \rho_i \delta_i}, \quad C_n = \frac{w/2}{\sigma \rho_i \delta_i},$$

h is the film thickness, w is the cooling cell width in toroidal direction, ρ_i and δ_i are the electrical resistivity and thickness of electroinsulating film respectively. Substituting du/dx by $\Delta u / \Delta x$ where Δx is the cooling cell length, the film velocity reduction Δu can be evaluated. These relations have been used to determine the velocity reduction and heat transfer coefficients variations in the open cooling cells of the HHF targets.

5.3. Flow regimes and heat transfer coefficients

In the open cooling cell of the HHF target the impinging LM jets form a film flow. Since the jet orifices make a row in toroidal direction the film flows in quasi-coplanar (close to coplanar) magnetic field with the major toroidal component, B_t lying in the film plane and

perpendicular to the velocity vector. The additional poloidal magnetic field gives a longitudinal component parallel to the velocity and another one, B_n normal to the cooled surface. In the absence of adequate experimental data it has been assumed that, with respect to some loss of momentum, the impinging jets at 7 m/s will form a slower film of about 5 m/s velocity. The strong coplanar magnetic field should stimulate fast formation of a uniform film flow with the transition length of about the distance between the jets (the orifices pitch). For the selected jet diameter above 2 mm and 10 mm pitch the minimum film thickness of 0.6 mm can be expected. That was considered acceptable for our comparatively small cooling cells (only 80 mm wide). The fast thin LM films with almost exactly such a geometry in coplanar magnetic field have been obtained in experiments [5.2] (though without impinging jets at the inlet).

The coplanar magnetic field interacts rather weakly with averaged flow and in this respect is similar to the longitudinal magnetic field. Magnetic field suppresses turbulent pulsations and thus can reduce the heat transfer coefficients. Therefore in the absence of experimental data on heat transfer in coplanar magnetic field we'll use an empirical relation suggested for LM flow in longitudinal magnetic field [5.3] and modify it for our case:

$$Nu = Nu_L + (Nu_T - Nu_L) e^{-32Ha/Re}, \quad (5.3.1)$$

where

$$Nu_T = 8.6 + 0.029 \cdot Pe^{0.8} \quad (5.3.2)$$

is the heat transfer coefficient for a turbulent LM flow without magnetic field in a flat tube [5.4].

The heat transfer coefficient for laminar flow in a plane duct is $Nu_L = 8.2$. Formula (5.3.2) can be used for the flow which is highly turbulent ($Re > 10^4$) without magnetic field. If magnetic field is weak it doesn't suppress pulsations and dimensionless heat transfer coefficient $Nu = Nu_T$. When the magnetic field is strong enough then $Nu = Nu_L$.

Toroidal and normal components of magnetic field - B_t and B_n interact with averaged flow in channels and cooling cells. They cause additional hydraulic resistance to LM flow. So the coolant velocity will be reduced and the film thickness increased in downstream direction. Consequently the heat transfer coefficient will be reduced from the entrance of the open cell to its outlet. Fig. 5-2, 5-3 show the variations of coolant velocity, film thickness and heat transfer coefficients in the inboard and outboard HHF target cooling cells.

Fig. 5-2. Coolant velocity, film thickness, heat transfer coefficients in the outboard HHF cooling cells.

Fig. 5-3. Coolant velocity, film thickness, heat transfer coefficients in the inboard HHF cooling cells.

In Fig. 5-2, 5-3 the solid lines correspond to the rectangular (constant width) cells and the dashed lines - to the trapezium (varying width) cells. For the top divertor the major reduction of the heat transfer coefficient takes place in the trapezium inboard cells where the width decreases in downstream direction. Here changes from initial value of 138 kW/m²K down to 54 kW/m²K, where $\alpha = Nu\lambda/d$ (λ - thermal conductivity, $d=4h$ - hydraulic diameter, h - film thickness). The magnetic field components are: $B_t = 7.5$ T, $B_n = 0.5$ T at the outboard target and $B_t = 10.9$ T, $B_n = 0.5$ T at the inboard target.

The temperature drop from the wall to bulk LM in the middle of the outboard HHF target with maximum heat flux of 15 MW/m² will be about 220°C.

In the LHF target cooling tubes LM also flows in quasi-coplanar magnetic field (B_t is parallel to the longer side of the rectangular tube cross-section). In calculations of heat transfer coefficients for these tubes a formula similar to (5.3.1) but with different heat transfer coefficients for turbulent and laminar flow has been used [5.4]:

$$Nu_T = 5 + 0.020 \cdot Pe^{0.8} \quad (5.3.3)$$

$$Nu_L = 5.38$$

These formulas can be used for a flat duct flow heated from one side. In this case a is calculated using (5.3.1) and (5.3.3) but the hydraulic diameter here $d=4b$, where b is the half-height of the flat duct. The average velocity is determined by the accepted temperature rise, given heat flux and channel geometry. It was established at 1.6 m/s for the inboard and outboard LHF targets and 2 m/s for the central target. The corresponding heat transfer coefficients are:

$\alpha = 35$ kW/ m²K - for the inboard LHF panel target tubes

$\alpha = 18$ kW/ m²K - for the central LHF panel target tubes

$\alpha = 36$ kW/ m²K - for the outboard LHF panel target tubes

The temperature drop between the wall and the flow core at the heat flux of 2 MW/m² will be 120°C. The heat transfer coefficients only weakly depend on the coolant velocity (Fig. 5-4) because of the turbulence suppression by magnetic field and relatively low Peclet numbers. Thus the channel cross-section variation along the LHF target won't have any appreciable impact on heat transfer. It can be assumed that heat transfer coefficients are constant for each tube.

Fig. 5-4. Heat transfer coefficient versus coolant velocity in the LHF central target channel

5.4. MHD Pressure Losses and LM Self-Draining

5.4.1. General Points

The pressure drops within the reactor magnetic field have been calculated with the use of PUMPIN code that allows to take into consideration the detailed geometry of the liquid metal feeding pipelines. The mathematical model realized in this code is based on the quasi-one-dimensional approximation of MHD flow. That means splitting long segments of the duct into pieces for which the quasi-uniform flow approach is applicable. The total pressure drop for a line is found as sum of the pressure loss in each of such duct pieces and the local pressure drops in various bends and joints. In general both for the uniform flow pressure loss and local pressure drops the superposition principle is applied with respect to the impact from two transverse magnetic field components: the "coplanar" one directed across channel width and the "normal" one perpendicular to the channel top and bottom walls. This is of course not a very accurate approximation but nevertheless it can be justified by two following considerations:

- high accuracy of the pressure loss calculations is not the critical issue for the considered liquid metal duct concept (with the electroinsulating films on the walls that ensure essential pressure loss reduction);

- the basic simplicity of the model used on each step of calculations provides the unique feasibility of the real design analysis taking into account the complicated pipeline geometry with all its bends and cross-section variations in the non-uniform magnetic field of tokamak.

The last circumstance is of special value for the in-reactor part of the liquid metal loop optimization in terms of gallium inventory which is quite important due to its high cost. Typically this inventory is subject to trade-off versus the permissible total pressure loss. Preliminary calculations performed in [5.5] have shown that the major contribution to the total pressure loss can be made by the effects of non-uniform MHD flow at "sharp" bends (with zero length) irrespective of the quality of electroinsulating films on the walls. Even using pipes with comparatively large diameters and moderate LM velocities (not greater than 1 m/s) in actual complicated LM line geometry it is very difficult to achieve the total pressure loss level below 3 MPa. From these results the following recommendations for the final stage of the LM pipelines optimization were drawn:

- only "smooth" bends with relative length of 10-20 diameters should be used in the places where the pipes turn from poloidal to toroidal direction or vice versa (that allows to eliminate the local pressure drop caused by abrupt change in the magnetic field vector orientation with respect to the pipe axis);
- to prevent the above mentioned local pressure drops in the toroidal manifolds of the divertor targets the pipelines connected to each target must consist of a large enough number (10-30) of parallel pipes that can form a uniform inlet flow across the toroidal width of the target; the latter is better achieved with the use of flattened pipe tips making a slot-like openings at the joint with the target manifold.

The typical pipeline segments that were considered as single "elements" of the LM tract are shown in Fig. 5-5.

Fig. 5-5. Typical segment shapes:

- a) vertical straight circular tube; b) "sharp" bend from vertical to horizontal / poloidal to toroidal direction;
- c) "smooth" bend from poloidal to toroidal direction;
- d) wedged straight rectangular tube;
- e) transition segment from rectangular to circular cross-section (flattened tip)

5.4.2. Key Relations

The local components of the magnetic field are recalculated from the absolute ones: radial B_r , toroidal B_t and vertical B_z . The local components are defined in a coordinate system connected with the channel:

$$\text{longitudinal} \quad B_l = (B_r \cos\theta + B_t \sin\theta) \cos\alpha + B_t \sin\alpha;$$

$$\text{coplanar} \quad B_w = (-B_r \sin\theta + B_t \cos\theta) \cos\omega - B_z \sin\omega;$$

$$\text{normal} \quad B_h = [-(B_r \cos\theta + B_t \sin\theta) \sin\alpha + B_t \cos\alpha] \cos\omega,$$

where θ is the angle between the channel axis and major radius in toroidal coordinate system ($-\pi < \theta \leq \pi$),

α is the angle of channel axis inclination to horizon ($-\pi/2 \leq \alpha \leq \pi/2$),

ω is the angle reflecting orientation of the channel normal: if its projection on Z-axis is positive then $\omega = 0$, otherwise $\omega = \pi$.

For a given piece of the channel with Δl length wherein, according to imposed restrictions, the flow and magnetic field parameters can be assumed constant the pressure loss in the uniform flow approximation is calculated by the relation valid for a channel flow at high Hartmann's numbers and presence of a contact resistance between LM and channel walls (in the form of a non-ideal electroinsulating film) [5.1]:

$$\Delta p = -\rho g \Delta l \sin \alpha + \frac{\rho}{2} (u_{in}^2 - u_{out}^2) - \quad (5.4.2.1)$$

$$-u D l \left[\left(\frac{B_w}{w} + \frac{B_h}{h} \right) \sqrt{\sigma \mu} + \frac{\sigma_w \delta_w B_w^2 / w}{1 + \varepsilon_w \frac{w}{3h} + \frac{\sigma_w \delta_w}{\sigma} \left(1 + \frac{\rho_i \delta_i}{h / \sigma} \right)} + \frac{\sigma_w \delta_w B_h^2 / h}{1 + \varepsilon_h \frac{h}{3w} + \frac{\sigma_w \delta_w}{\sigma} \left(1 + \frac{\rho_i \delta_i}{w / \sigma} \right)} \right],$$

where w , h is the channel half-width and half-height respectively;

u_{in} , u_{out} , u is the inlet, outlet and averaged flow velocity respectively;

ρ is the LM mass density;

g is the gravitation acceleration;

σ , μ is the LM electric conductivity and dynamic viscosity respectively;

σ_w , δ_w is the channel wall electric conductivity and thickness respectively;

ρ_i , δ_i is the electric resistivity and thickness of the electroinsulating film;

ε_w , ε_h are form-factors depending on the channel cross-section shape: for a circular one ε_w , $\varepsilon_h = 0$, for a rectangular one ε_w , $\varepsilon_h = 1$ (the correction introduced for the rectangular cross-section was taken from [5.6]).

The local pressure drops due to sudden cross-section changes or the magnetic field component variations (in bends) are calculated using the following basic relation:

$$\Delta p = kN \frac{\rho u^2}{2}, \quad (5.4.2.2)$$

where

$$N = \frac{\sigma B^2 l}{\rho u},$$

l is characteristic length, it can be either w combined with B_h or h combined with B_w .

For a segment with non-uniform flow the parameters for relation (5.4.2.2) are taken from the "generating" cross-section (either inlet or outlet) which is determined by the maximum value of induced electric potential across the channel:

$$U_{\max} = uBl$$

and coefficient k in (5.4.2.2) is determined from a basic value k_{ref} (established experimentally for a well defined reference case) with respect to "completeness" of the reference case realization for the considered segment (e.g. a 30° bend against the reference 90° case) and "smoothness" of parameters variation in terms of the segment length in calibers $\Delta L/l$:

$$k = k_{\text{ref}} \Psi_L \Delta U / U_{\max},$$

where $\Delta U = |(uBl)_{\text{out}} - (uBl)_{\text{in}}|$ and for the reference case by definition

$$\Delta U / U_{\max} = 1,$$

$$\Psi_L = 1 - \frac{\Delta L/l}{(\Delta L/l)_{\text{ref}}} \quad \text{if} \quad \frac{\Delta L/l}{(\Delta L/l)_{\text{ref}}} < 1, \quad \text{otherwise} \quad \Psi_L = 0.$$

5.4.3. Calculation Results

For an example the detailed hydraulic parameters distribution along the feeding pipeline of the inboard LHF panel target is presented in Table 5.4.1. This way all 10 pressurized feeding pipelines (5 lines for 5 targets within the module of each of 2 divertors) were calculated. The calculation results are summarized in Table 5.4.2. For the both divertors it was found expedient to establish two different pressure loss levels: a larger one for the inboard LHF, HHF and central LHF targets that are subject to stronger magnetic field, and a lower one for the outboard HHF and LHF targets. These levels differ also for the top and bottom divertors due to two factors:

- 1) the inlet and outlet points at the targets are not symmetrical in the mirror-like structure of the top and bottom target belts, they are located so to provide within the targets of both divertors a downstream flow with respect to gravitation force;
- 2) the pressurized lines in the top divertor end closely to the targets at the open joints with the drain pipes built-in into the blanket while in the bottom divertor they are continued down to special movable joints (described in Section 4.3) with the permanent drain tract.

The LM drain lines have been analyzed with DRAIN code which is quite similar to PUMPIN. The major difference is that in this code the condition $p > 0$ is checked along the channel, which is critical for the non-pressurized self-draining ducts. Besides this each draining line is assumed consisting of an arbitrary number of subtracts connected in series with open joints (like an end of a tube over a sink). That means imposing $p = 0$ condition at such points. The calculations were performed for 4 drain lines located within the bottom divertor chamber:

- the inboard blanket drain line receiving the total coolant flow from the inboard LHF and HHF targets of the top divertor;
- the inboard HHF target drain line of the bottom divertor;
- the outboard HHF target drain line of the bottom divertor
- the outboard blanket drain line receiving the total coolant flow from the central and outboard LHF and HHF targets of the top divertor.

Table 5.4.1.
Hydraulic Parameters Distribution Along the Feeding Pipeline of the Top
Divertor Inboard LHF Panel Target

Seg. No.	L, m	Shape	D, mm	H, mm	u, m/s	R, m	θ°	α°	B_w , T	B_h , T	Δp , MPa
1	0.296	Circ	14	14	3.01	5.58	-91	-90	-0.58	-8.31	-0.0687
2	0.314	Circ	14	14	3.01	5.58	-89	0	-0.21	-0.20	-0.0479
3	0.579	Circ	14	14	3.01	5.56	-89	0	-0.22	-0.19	-0.0153
4	0.663	Circ	14	14	3.01	5.36	-175	-28	-8.65	-0.34	-0.1323
5	0.547	Circ	14	14	3.01	4.88	-174	-28	-9.48	-0.37	-0.1554
6	0	Circ	14	14	3.01	4.88	-176	0	-9.50	-0.27	-0.0040
7	0.461	Circ	14	14	3.01	4.42	-176	0	-10.48	0.08	-0.1543
8	0	Circ	14	14	3.01	4.42	-175	-48	-10.47	-0.48	-0.0051
9	0.269	Circ	14	14	3.01	4.34	0	-90	-10.70	0.17	-0.0930
10	0.134	Rect	150	1.6	1.87	4.43	0	-42	-10.46	0.05	+0.0178
11	0	Rect	870	14	0.19	4.43	0	-45	-10.46	0.06	-0.0103
12	0.021	Rect	870	14	0.19	4.45	0	-45	-10.43	0.06	+0.0008
13	0	Rect	26	2	1.65	4.45	4	-61	-10.41	-0.55	-0.0366
14	0.746	Rect	29	2	1.54	4.81	4	-61	-9.63	-0.57	-0.0748
15	0;	Rect	940	14	0.18	4.81	180	-90	-9.65	-0.02	-0.0186
16	0.033	Rect	940	14	0.18	4.78	180	-16	-9.71	-0.01	+0.0008
17	0	Rect	160	1.4	2.01	4.78	-165	-11	-9.37	-0.47	-0.0450
18	0.081	Circ	14	14	3.01	4.70	-165	-11	-9.52	-0.46	-0.0261
Total	4.145										-0.8681

Note: channel diameter (width) and height, LM velocity, local channel axis angles and magnetic field component values are given for the segment outlet cross-sections.

Segment description for Table 5.4.1.

- 1 - vertical straight pipe at the inlet port;
- 2 - bend from vertical to toroidal direction;
- 3 - horizontal near-toroidal straight pipe;
- 4 - bend from toroidal to radial direction;
- 5 - near-radial inclined straight pipe;
- 6 - bend from inclined to horizontal direction;
- 7 - near-radial horizontal pipe;
- 8 - bend from horizontal to inclined downward direction;
- 9 - curved pipe with increasing downward inclination;

- 10 - curved transition segment with the cross-section shape changing from circular to rectangular slot parallel to toroidal magnetic field;
- 11 - entrance to the inlet manifold;
- 12 - near-toroidal manifold with uniform poloidal flow;
- 13 - entrance to the divertor target cooling channels;
- 14 - flat wedged rectangular cooling channel of the divertor target inclined downward;
- 15 - entrance to the outlet manifold;
- 16 - near-toroidal manifold with uniform poloidal flow;
- 17 - entrance to the outlet transition segment;
- 18 - inclined straight transition segment with the cross-section shape changing from rectangular slot parallel to toroidal magnetic field to circular.

Table 5.4.2.
General Hydraulic Parameters of the Pressurized Divertor Cooling Lines

Top divertor					
Line No.	Number of Segments	Length, m	Δp, MPa	Required inlet pressure, MPa	LM volume m³
1	18	4.15	-0.87	1.0	0.0093
2	9	2.48	-0.94	1.0	0.0054
3	19	6.85	-1.00	1.0	0.0498
4	9	1.90	-0.70	0.7	0.0084
5	15	3.45	-0.61	0.7	0.0122
Total for one module					0.0851
Total for the top divertor					1.3616
Bottom divertor					
1	27	9.94	-1.19	1.2	0.0340
2	14	3.45	-1.20	1.2	0.0115
3	21	8.88	-1.19	1.2	0.0853
4	9	2.80	-1.00	1.0	0.0132
5	19	7.88	-0.97	1.0	0.0409
Total for one module					0.1849
Total for the bottom divertor					2.9584
Total for two divertors					4.3200

The obtained calculation results are given in Table 5.4.3.

Table 5.4.3.
General Hydraulic Parameters of the Non-Pressurized Drain Lines

Line No.	Number of segments	Number of parallel pipes	Length, m	Pipe diameter, mm	Velocity, m/s	LM volume, m ³
1	5	10	2.96	42	0.71	0.0410
2	5	8	3.18	30	0.92	0.0185
3	5	10	3.19	32	1.37	0.0257
4	5	36	2.55	31	1.13	0.0694
Total for one divertor port					0.1546	
Total for the reactor					2.4736	

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6. THERMAL-STRESS ANALYSIS

6.1. High Heat Flux Targets.

Temperature and stress distributions were calculated for the outboard target, where the thermal load is supposed to be by a factor of 1.5 higher than that for the inboard one. For this analysis temperature dependent properties of the vanadium alloy shown in the Table 3.6 have been used. All calculations were carried out by means of a finite element code.

The structural scheme of the HHF target cell is shown in Fig. 6-1. This picture represents one half of the plasma facing plate of the cell. The thickness of the vanadium alloy plate subject to heat flux is 0.9 mm while the other parts of the cell are 1 mm thick. The LM film flow velocity along Z axis decreases under the action of MHD forces from 5 m/s down to 3 m/s. The heat transfer coefficient is consequently reduced from 135 down to 70 kW/m².K (see Section 5). The LM inlet temperature is 80°C and the outlet temperature is 180°C. For the temperature field calculations the heat flux distribution has been used that is shown in Fig. 5-1. The power density due to neutron flux has been estimated at 10 W/cm³.

The thermal analysis has shown that the maximum temperatures are achieved in the heat flux peak area on the plasma facing surface of the cell. In Fig. 6-2 the temperature distribution across the plasma facing surface is shown. The maximum temperature is about 740°C. The obtained temperature distribution over the whole considered structure was used for the stress calculations. In stress analysis it was assumed that part 1 (see Fig. 6-1) of the target cell, fixed in the slots, has restricted permissible displacements. In X direction and Y direction the displacements $U_X=0$ and $U_Y=0$ for all nodes of this part. Besides this it was assumed that all nodes on the surface 2 (Fig. 6-1) have the permissible displacement in X direction $U_X=0$. For these conditions the elastic equivalent stress, σ_m (von Mises) and $\sigma_i = \sigma_1 - \sigma_3$ stresses have been calculated. The analysis has shown that the maximum stresses are localized at the maximum temperature area that is on the plasma facing surface of the cell under the heat flux peak. In Fig. 6-3 the stress distribution (von Mises) across the plasma facing surface of the target is shown. The maximum stress value σ_m is about 216 MPa and $\sigma_i=224$ MPa. For the mentioned temperature at this point of 740°C the yield stress is about 330 MPa and the ultimate tensile strength is 560 MPa. According to ASME Code the limit value S_m is the least of the following quantities: $1/3\sigma_b$ (average ultimate tensile strength) and $2/3\sigma_{02}$ (average

Fig. 6-1.

Fig. 6-2.

Fig. 6-3.

yield stress) at the corresponding temperature. In our case $S_m=187$ MPa. For the stress calculations based on a linear elastic material model the equivalent stresses σ_m and σ_i should be lower than $1.2S_m=225$ MPa. This condition is fulfilled in our case.

6.2. Low Heat Flux Panel Targets.

There are three LHF targets: the inboard, central and outboard one. The inboard and outboard panel targets have the same fixing scheme, similar cooling channels, LM flow velocity and coolant temperature rise. That is why the temperature-stress analysis has been carried out only for the inboard and central LHF targets. To determine the location of maximum stresses a simplified three-dimension models of these panel targets have been considered in the temperature-stress analysis. The following data (together with those from Table 3.6) have been used:

Heat flux	2 MW/m ² ,
Volumetric power density	10 W/cm ³ ,
Heat transfer coefficient	18-30 kW/m ² ·K
LM inlet temperature	80°C,
LM outlet (inboard target) temperature	253°C,
LM outlet (intermediate target) temperature	258°C,
LM pressure inside cooling channel	0.5 MPa.

The calculation scheme of the inboard panel target is presented in Fig. 6-4 and its cross-section is shown in Fig. 6-5. The temperature distribution over the plasma facing surface (Y-Z plane) calculated for this scheme is given in Fig. 6.6.

The stress analysis has been performed under the following assumptions:

- at the inlet area where the inlet manifold of the actual structure is located all nodes on lines A and B (Fig. 6.4) on the back surface have zero permissible displacements both in X and Y directions ($U_X=0$ and $U_Y=0$);
- all nodes localized at the outlet edge (line D) on the back surface have zero permissible displacements in Y direction ($U_Y=0$).

Fig. 6-4.

Fig. 6-5.

Fig. 6-6.

These conditions should approximate the actual fixing conditions. The elastic stress analysis has shown that the maximum stresses are located on the back surface of the target at the inlet area.

To obtain more information on the temperature-stress distribution in that region a two-dimensional temperature-stress analysis has been carried out for the structural scheme shown in Fig. 6-5, corresponding to the cross section along line A in Fig. 6-4. In this elastic-stress analysis it was assumed that all nodes between points 1 and 2 (Fig. 6-5) have zero permissible displacements both in X and Y directions ($UX=0$ and $UY=0$). And all nodes between points 2 and 3 and between points 4 and 5 have zero permissible displacements in X direction ($UX=0$). These conditions should approximate the actual attachment conditions (welding to the inlet manifold). The obtained temperature distribution is given in Fig. 6-7 and the stress (von Mises) distribution - in Fig. 6-8. The maximum temperatures on the plasma facing surface don't exceed 500°C . The maximum stresses ($\sigma_m=295$ MPa and $\sigma_i=340$ MPa) that can be classified as local ones are observed near point 6 (Fig. 6-5) of the cooling channel. The temperature in this point is about 160°C . The general stresses σ_i over plasma facing plate of this target don't exceed 100 MPa at maximum temperature of 500°C . The general stresses s_i over the cooling channel don't exceed 200 MPa at maximum temperature of 160°C . The limit values are: $S_m=187$ MPa at 500°C and $S_m=217$ MPa at 160°C . For elastic analysis, according to ASME Code, the local stresses should not exceed $1.8S_m=390$ MPa in our case. The general stresses should not exceed $1.2S_m=225$ MPa at 500°C and 260 MPa at 160°C . All these condition are fulfilled for the considered structural scheme of the inboard LHF target.

The scheme of a simplified model of the central panel target is shown in Fig. 6-9. The cross section of this target is similar to the cross section of the inboard target (shown in Fig. 6-5). The only difference is in the height of the channel which is 4.4 mm instead of 2 mm for the inboard target. The temperature distribution over the plasma facing surface (Z-X plane) for this target is given in Fig. 6.10. The stress analysis has been performed under the following assumptions:

- at the inlet area, where the inlet manifold of actual structure is located, all nodes on the lines A and B (Fig. 6-9) at the back surface have zero permissible displacements both in X and Y directions ($UX=0$ and $UY=0$);
- at the outlet area, where the outlet manifold is located, all nodes on the lines C and D at the back surface have the same zero permissible displacements ($UX=0$ and $UY=0$).

Fig. 6-7.

Fig. 6-8.

Fig.6-9.

Fig. 6-10.

These conditions should approximate the actual attachment conditions. The three dimensional stress analysis of the simplified target model has revealed the maximum stresses located at the outlet area. The detailed stress distribution in the corresponding cross-section has been obtained from two dimension temperature-stress analysis. Here the same assumptions as one for the inboard target were used. The temperature and stress (von Mises) distributions are presented in Fig. 6-11 and 6-12 respectively. The maximum temperatures on the plasma facing surface don't exceed 650°C. The maximum stresses ($s_m=280$ MPa and $s_i=323$ MPa) that can be classified as local ones are observed near point 6 (Fig. 6-5) of the cooling channel. The temperature in this point is about 337°C. The general stresses s_i over plasma facing plate of this target don't exceed 100 MPa at the maximum temperature of 650°C. The general stresses s_i over the cooling channel don't exceed 190 MPa at the maximum temperature of 330°C. The limit values are: $S_m=187$ MPa at 650°C and $S_m=200$ MPa at 330°C. For elastic analysis, according to ASME Code, the local stresses should not exceed $1.8S=360$ MPa in our case. The general stresses should not exceed $1.2S=225$ MPa at 650°C and 240 MPa at 160°C. All these conditions are fulfilled for considered structural scheme of the central LHF target.

Fig. 6-11.

Fig. 6-12.

7. LIQUID METAL LOOP EVALUATION

7.1 Some Requirements and Initial Data

LM cooling has its specific design requirements which have to be taking into consideration for LM cooling equipment layout:

- thermoinsulation of all external pipelines with temperatures above 45°C and application of special heaters for pipes, manifolds and tanks to keep the coolant in liquid state;
- each LM loop must have a bottom receiver tank capable to hold all Ga inventory in the events when the divertor module must be withdrawn;
- according to recommendations made after preliminary LM divertor system analysis [7.1] each LM loop has to incorporate an emergency top tank containing a store of "cooled down" (in terms of radioactivity decay) Ga to be used during LOCA/LOFA for maintaining the divertor targets cooling in self-draining regime and provide some time margin for the reactor shut-down;
- a zoned layout of the LM equipment to limit the spread of radioactivity due to possible LM leakages and/or accidents with LM spilling;
- since Ga is a rather expensive material the loop equipment lay-out should be done with respect to the total LM inventory minimization.

In designing the primary divertor cooling circuit the main question to be answered was the number of LM loops needed and the unit capacity for each loop. According to listed above requirements 16 separate identical loops were chosen though no optimization was done to trade-off the benefit of using multiple standard units, capital cost reduction, easy operation, maintenance and repair against the disadvantages of too many units of small power (e.g. 32 or 64 divertor cooling loops). The loops have a vertical orientation, each of them incorporates two couples of modules from the top and bottom divertor (the couple in each divertor is one rectangular plus one trapezium module). Each LM loop has its own heat exchanger, buffer and dump tanks, pumps, LM supporting subsystems (LM making-up, conditioning, tritium extraction).

The major parameters of the LM divertor cooling system are given in Table 7.1.1.

Table 7.1.1.
Parameters of LM the divertor cooling system

Power output, MW	188x2
Coolant	Ga
Inlet temperature, °C	80
Outlet temperature, °C	230
LM flow rate, m ³ /s	1,22
Number of LM loops	16
Total LM volume, m ³	28,2

7.2. Heat Exchangers

Heat from the primary LM coolant can be removed in the following ways:

- directly to gas/water in the secondary loop;
- through an intermediate circulating coolant to the third loop;
- through a specially organized protective layer.

A weak point of the simplest first option is the problems with reliable partitioning between the liquid metal and the secondary coolant. This is a very important issue not only for chemically reactive liquids (like Na and H₂O) but for comparatively inert coolants like Ga too, which is explained by the difficulties of impurity removing from the LM coolant and leakage detecting. The latter circumstance is of critical importance for the considered system where the primary LM coolant is allowed to contact with the vacuum chamber environment and may serve as water vapor or gas conductor coming from the high pressure secondary loop.

Thus the presence of some intermediate agent with low pressure between the primary and secondary coolants, that is suggested by the second and third options, is highly desirable. Any leak from both sides into this agent can be detected in time but without urgency. From the viewpoint of chemical interactions the best candidate agent for this purpose is again gallium. Though, taking into account its high cost, the choice between the second and third options should be made in favor of the latter. It's worth to mention also that this type of

heat exchangers will allow to minimize tritium diffusion into the secondary (water/vapor) coolant loop.

But of course this option has its own disadvantages:

- complicated design and technology;
- additional thermal resistance due to intermediate layer.

Two different designs of the heat exchanger with the intermediate protective layer were considered:

- with double-walled tubes;
- with two independent sets of usual tubes belonging to the primary and the secondary cooling loops and the space between them filled with the intermediate protective agent.

The first design has been finally selected. Calculations were performed for counter current flow type heat exchanger consisting of straight circular double-walled tubes. Gallium coolant flows into the inner tubes and water passes outside. The protective layer is filled with slowly circulating gallium. In order to minimize the gallium inventory some optimization has been done with respect to the gallium filled volume. Its value was calculated as a function of the water coolant outlet temperature. The obtained parameters of the heat exchanger are given in Table 7.2.1.

Table 7.2.1.
Heat exchanger parameters for the LM cooled divertor system

Primary (hot) coolant	Ga
Primary coolant temperature, °C:	
- inlet	230
- outlet	80
Temperature drop, °C	150
Thermal load per one LM loop, MW	23.5
LM flow rate, kg/s	459
Coolant velocity, m/s	1
Heat transfer coefficient, kW/(m ² ·K)	26.9
Secondary coolant	H ₂ O

Table 7.2.1.
Heat exchanger parameters for the LM cooled divertor system (Continued)

Water temperature, °C:	
- inlet	50
- outlet	180
Water pressure (not less), MPa	2
Water flow rate, kg/s	328
Heat transfer coefficient, kW/(m ² ·K)	33
Tube outer diameters, mm:	
- inner tubes	12
- outer tubes	15
Tube wall thickness, mm	1
Tube number	975
Total heat transfer surface, m ²	122.4
Tube length, m	2.65
Diameter of the heat exchanger vessel, m	0.7
LM inventory in one heat exchanger, m ³	
- in cooling tubes	0.2
- in protective layer	0.05
total	0.25
Total number of heat exchangers	16

All the obtained parameters seem feasible from different viewpoints. The present state of the art of heat exchanger technology developed for LMFBR's is quite suitable for the considered Ga/H₂O heat exchanger design.

But it should be mentioned that the low outlet water temperature can bring forth certain difficulties for a high efficiency power conversion system.

7.3 Tritium Processing

7.3.1. Initial Data

In the developed divertor design there is no direct LM coolant with plasma particles incident to the divertor targets.

Deuterium and tritium ions and fast atoms from the plasma will bury themselves at a certain depth within the target material from which most of them will diffuse back into the

vacuum chamber. But some fraction of these isotopes will stay in the targets and diffuse into the LM coolant. Then it is important to know the amount of tritium that can be lost this way.

Different models can be considered to estimate tritium flux penetrating into the LM coolant [7.2, 7.3]. In this study the most simple one has been employed which does not take into account any surface effects caused by presence of some layers that may serve as diffusion barriers. It seems to be a conservative approach in our case since from the plasma side subject to particle bombardment the existence of any permanent layers different from the target bulk material is quite improbable actually, while at the coolant side such a layer will be present for sure in the form of the electroinsulating film which is a basic feature of the considered LM circuit. Thus the actual diffusion flux through the wall should be lower than calculated by the accepted model.

The key parameter in the diffusion model is the depth of plasma particles burial that depends on their kind, energy and the target material. The values of diffusion flux back from and through the wall are in inverse proportion to the ratio of the burial depth and wall thickness. The results of tritium permeation estimates are given in Table 7.3.1.1.

Table 7.3.1.1.
Initial Parameters for Tritium Inventory Estimates

Parameter	HHF Targets	LHF Targets	Total
Power output, MW	51.3	136.7	188
Average load, MW/m ²	2.78/4.5	1.1	
Target surface, m ²	13.6	125.4	
D,T particle flux, 1/m ² -s	$3.7 \cdot 10^{23}$	$1.07 \cdot 10^{23}$	
Tritium diffusion flux, atom(T)/s	$2.66 \cdot 10^{19}$	$2.38 \cdot 10^{19}$	
Tritium permeation rate, atom(T)/s	$3.62 \cdot 10^{20}$	$2.98 \cdot 10^{21}$	$3.34 \cdot 10^{21}$

7.3.2. Tritium Accumulation in Gallium Coolant

To clean the primary coolant from tritium it is passed through a tritium processing unit (TPU). The total LM flow rate in one primary cooling loop is 0.0764 m³/s in each module but only a small part of it has to be branched into tritium processing unit to obtain acceptable tritium levels in LM. Estimates of tritium accumulation dynamics in Ga coolant are based on

the calculation model [7.4] where tritium processing unit is characterized by the following parameters:

- LM flow rate fraction branched for tritium removing;
- inlet C_{in} and outlet C_{out} tritium concentrations in LM before and after tritium processing unit which can be written through tritium recovery efficiency as

$$\varepsilon = 1 - C_{out}/C_{in}$$

The calculations were made for continuously working reactor with duty cycle equal 1.0. It was assumed that tritium removing process starts simultaneously with the reactor start-up.

The results of tritium dynamic accumulation in the primary Ga coolant loop are shown in Fig. 7-1. The calculations were performed for two pressure levels in the gaseous divertor (10^{-4} and 10^{-3} Torr) and constant value of LM flow branched for cleaning (5%). The value of tritium removing efficiency has been varied from 0.1 up to 0.95. As it is seen from Fig. 7-1, if high tritium extraction efficiency (above 50%) is available the equilibrium tritium inventory in LM coolant will not exceed 16-30 g for the divertor with pressure level 10^{-4} Torr and 50-100 g for 10^{-3} Torr.

Some basic parameters for the tritium extraction system (TEPS) are given in Table 7.3.2.1.

Table 7.3.2.1.
Tritium Extraction System Parameters

LM flow rate in one loop, m ³ /s	0.0764		
Flow fraction branched into TPU	0.05		
Flow rate through TPU, l/s	3.82		
Tritium recovery efficiency	0.1	0.5	0.95
Tritium concentration, appm:			
-inlet	21.4	4.3	2.26
-outlet	19.3	2.15	0.113
Tritium inventory in primary coolant, g:			
-in one loop	9.62	1.93	1.02
-total for the divertor system	154	30.93	16.3

a)

b)

Fig. 7-1. Tritium dynamic accumulation in gallium primary coolant loop with tritium extraction system:

1- extraction efficiency 10%; 2- 50%; 3-95%

a) pressure in gas divertor 10^{-4} Torr;

b) 10^{-3} Torr

It was supposed that these parameters can be obtained in tritium processing units using vacuum outgassing method for tritium recovery. The tritium inventory in the LM coolant will be moderate even for relatively low efficiency of tritium removing.

7.4. Liquid Metal Conditioning

7.4.1. Impurity Sources

During operation the gallium coolant will inevitably accumulate different impurities with the major contribution from corrosion products. In addition to this the LM will be contaminated by hydrogen isotopes permeating through the divertor target walls. Certain amount of special agents (e.g. oxygen) may be added to the LM to control the process of formation and sustaining the electroinsulating films on the walls of the LM tract. Thus the LM in the primary coolant loop has to be controlled with respect to impurities accumulation both in gaseous and solid forms.

7.4.2. Requirements to the Impurity Content in LM.

In this study the permissible impurity contents in gallium coolant have been considered mainly from the viewpoint of induced activity (see Section 8.2). It was obtained that impurities of Ni, Fe, Mn will give isotopes with long life induced activity and the long term activity level in wasted coolant will depend on both initial impurity contents and their following accumulation due to corrosion process.

In accordance with the activation analysis results for the primary loop $\bar{A}\bar{e}$ -0 gallium grade (in Russian classification [7.5, 7.6]) has been chosen. Table 7.4.2.1 presents the standard impurity contents in this product. The upper limits of impurities during operation must not exceed the initial levels by more than 10^3 - 10^4 times.

Table 7.4.2.1
Impurity Contents in Gallium $\bar{A}\bar{e}$ -0

Element	Concentration, wppm	Element	Concentration, wppm
Si	5	Cd	5
Al	4	Mg	5
Fe	4	P	5
Ni	4	Sc	0.5
Cu	3	Te	0.5
Cr	1	Zn	0.5
Sn	1	Mn	0.3
Pb	1	Bi	0.01

7.4.3. Gallium Cleaning

As it was shown earlier (see Sections 3.1,3.2) Ga is a highly corrosive LM. The primary purpose of the Ga clean-up/conditioning system is to remove corrosion products and control the oxygen and other gases content. Actually a relatively small fraction of LM flow has to be purified and then fed back. For the conditions of gallium loop in the fusion reactor an efficient method of Ga cleaning with the use of glass/ceramic filters [7.7] can be recommended.

LM is pumped through porous filters under vacuum conditions at about 40°C. Oxide films, other insoluble metallic and intermetallic impurities precipitate on the filter surface. The precipitations can be easily removed during filter regeneration by gas blowing. Practically total removal of the solid products during regeneration and high resistivity to chemical interactions insure long operation life of the ceramic filters. The typical pores size for processing high purity gallium is 30-50 m. Some experimental results of Ga filtration through glass filters are given in Table 7.4.3.1. As it seen after filtration the solid impurity content could be decreased down to the 10^{-3} per cent (except Pb and Zn impurities).

An additional cleaning effect can be obtained after intermediate crystallization stage of preliminary filtered Ga. Redistribution of solid impurities between Ga and first crystals is shown in Table 7.4.3.2.

Thus the Ga clean-up system can be devised including the following processes:

- settling at low temperature
- first stage filtration;
- crystallization;
- second stage filtration;
- filter regeneration (by gas blowing).

Of course additional experiments are required both for checking filter parameters and LM conditioning feasibility in a by-pass configuration of a permanently operating loop. Such investigations can be performed on existing Ga loops and experimental assemblies.

Table 7.4.3.1.
Impurity Content (wt.%) in Ga After Filtration Through Glass Filters

	Experiment N1			Experiment N2
Element	Gallium		Wastage	Gallium
	in LM	in first crystals	on glass filter	in LM
Al	not found	0.001	0.005	0.001
V	--"	not found	not found	not found
Fe	--"	0.001	0.001	0.001
Si	--"	not found	0.01	0.001
Ca	--"	--"	not found	not found
Mg	0.001	<0.01	0.01	~0.001
Mn	not found	not found	0.03	not found
Cu	not found	not found	0.01	0.001
Mo	--"	--"	not found	not found
Ni	--"	0.0005	--"	0.001
Sn	--"	not found	>0.003	0.001
Pb	0.01	0.01	~0.01	0.01
Ti	not found	not found	0.001	not found
Cr	--"	--"	<0.01	--"
Zn	0.01	0.01	0.01	0.01

Table 7.4.3.2.
Impurity Content (wt.%) in Ga Before and After Crystallization

	Impurity Content		
Element	Initial	In first Crystals	After Crystallization
Al	0.01	0.03	0.003
V	0.003	0.01	0.001
Fe	0.003	0.01	<0.001
Mn	0.005	0.01	<0.001
Pb	0.05	0.03	0.06
Zn	0.96	0.58	1.43

7.5 Pumps

7.5.1. General Considerations

It is assumed that LM pumping technology from LMFBR will be applicable for gallium primary coolant loops. Two classes of pumps could be used for this purpose.

The first class is the electromagnetic pumps (EM-pumps). Now they are widely used in different fields of technology: for LM circulating in FBR, in chemistry and metallurgy. The feasible levels of EM-pump flow rates and pressures are up to 4000 m³/h and 2 MPa respectively. For the considered LM primary cooling loops much lower flow rate capacity is required. Through each loop about 275 m³/h of LM have to be pumped. For such flow rates different types of EM-pumps can be used. The first one is cylindrical linear inductive pump (CLIP), the second one is asynchronous helical inductive pump (HIP). The CLIP-type pumps are produced for flow rates up to a few thousand m³/h and pressure heads up to 0.4 MPa. HIP-type pumps have higher pressure heads (~1.6 MPa) but considerably lower values of flow rates (below 20 m³/h) [7.8]. Typical parameters of a CLIP-type pump, namely 3-150Ī model are given in Table 7.5.1.1. Usually the pump life time fixed in specifications is limited to 30,000 operation hours though there are pumps that had operation time without failure up to 50,000 hours. The same EM-pumps are available for pumping heavy LM such as lead, bismuth and gallium.

If necessary EM-pumps can be connected in series and in parallel. In general EM-pumps have the following advantages against mechanical ones:

- ta fully hermetic structure without any seals;
- tno moving or rotating parts;
- thigh reliability;
- tsimple flow rate control.

The required LM flow rates can be provided by existing EM-pumps which are commercially produced [7.9, 7.10]. Their major drawbacks are:

- tlow maximum pressure head (< 0.5 MPa);
- tlow efficiency (below 30%).

Following the experience of LMFBR's [7.11] instead of EM-pumps the well developed leak-free centrifugal circulation pumps (CCPs) can be used in the primary LM coolant loop. In comparison with the EM-pumps CCPs have much higher efficiency (up to 70%), higher pressure heads (up to 1.6 MPa for Na), high flow rates. But it should be mentioned that each type of CCP has been designed for a specific project. The problems of CCP are also well known: potential leakage of highly activated LM coolant, gas leakage, large sizes and weights.

Table 7.5.1.1.
Basic Technical Parameters of Industrially Produced CLIP
(Model 3-150/i)

Pumped liquid metal	Na/K, Li, Hg, Ga
LM temperature, °C	up to 450
Pressure head, MPa	0.245
LM flow rate, m ³ /hr	150
Efficiency, %	28.5
Mass, kg	500
Sizes, mm:	
length	1340
diameter	500
Channel material	Stainless steel (1X18H10T)

7.5.2. Pumping Requirements

For the LM circuit evaluation it was assumed that all the primary loops equipment that serves for both top and bottom divertors will be located in an annular space within the main tokamak building. So the heat exchanger and circulating pumps rooms have to be arranged around the reactor hall but at some lower levels. The reactor hall diameter is mostly affected by the envisaged repair and maintenance operations. With respect to ITER project materials the radius of the circulating pumps room location can be estimated at 17-20 m and according to the known ARIES reactor dimensions the loop height from circulating pumps to the top divertor will be 20-22 m.

The LM flow rate in each loop is $\sim 275 \text{ m}^3/\text{hour}$ or $0.077 \text{ m}^3/\text{s}$. The required pumping power is determined as

$$P_p = Q_1 \sum \Delta p_\Sigma$$

where Q_1 is the LM flow rate in a primary coolant loop;

$$\Delta p_\Sigma \approx \Delta p_{\text{ext}}^f + \Delta p_{\text{ext}}^h + \Delta p_{\text{in}}^{\text{MHD}},$$

p_{ext}^f , p_{ext}^h , $p_{\text{in}}^{\text{MHD}}$ is the pressure loss due to friction in the pipes of external part of the loop, the hydrostatic head and MHD pressure losses inside the reactor respectively.

The required pumping powers are summarized in Table 7.5.2.1. The value of total pumping power needed for gallium circulation in primary cooling divertor loops (top and bottom) is 2.63 MW.

**Table 7.5.2.1.
Pumping Requirements**

MHD-pressure drop, MPa	Top divertor	Bottom divertor
Inboard LHF target	1.0	1.2
Inboard HHF target	1.0	1.2
Central LHF target	1.0	1.2
Outboard HHF target	0.7	1.0
Outboard LHF target	0.7	1.0
Pumping power in module, kW:		
- to compensate MHD losses	33.05	43.83
- to provide pressure head	50.84	23.37
- to compensate friction in the external pipes	8.38	4.66
Pumping power for one loop, kW	164.13	
Total power for LM divertor system, MW	2.63	

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8. SAFETY CONSIDERATION

8.1. Accident Scenarios

The analysis of probable accidents has been carried out in following sequence. At first a few accidents were qualitatively analyzed, then some accidents were selected and analyzed quantitatively.

8.1.1. Loss of Coolant Flow Accident

Loss of coolant flow accident (LOFA) in the divertor targets of various design may develop differently. In case of LOFA in the HHF targets made in the form of open inclined troughs the situation looks more favorable for the top divertor: here coolant from feeding pipes and headers located *above* the troughs will flow out on them under the action of gravity and cool the target partially for some time. But in the bottom divertor where the troughs are located at the highest point of the cooling tract the coolant flow will stop quickly, following exactly the inlet pressure decrease.

In case of LOFA in the LHF targets two probable paths of events exist. In the first case coolant will drain from tubes and headers and in the second case it will stay. Then it becomes an additional mass acting as a heat sink but at the same time, due to induced activity, it is an additional source of heat after the reactor shut-down.

LOFA can result from both internal and external events. An external event is a large scale failure generally, stop of the main circulating pumps for example. In this case the accident will spread over a number of the reactor components. A typical internal event is in-reactor plugging up of a pipe, manifold, jet orifices in the HHF targets or rectangular cooling tubes in the LHF targets.

8.1.2. Loss of Coolant Accident

For a LM system, in contrast to one with water or gas, the loss of coolant accident (LOCA) combined with the coolant ingression to the vacuum chamber may result in plasma disruption but won't be followed by the chamber pressurization. Thus, if the fusion reaction is shut down

in time, the subsequent accident development will be determined mainly by after heat in the reactor elements.

LOCA in the HHF targets resulting from an external event, e.g. rupture of the main pipeline will develop according to the external LOFA scenarios because in both cases the HHF targets will be left without coolant.

LOCA in the LHF targets resulting from an external event will cause coolant draining from the cooling tubes and if plasma is shut down in time the targets will be heated only by after heat in the structural material and in the stagnant LM within the inter-tube space of the protective shell.

LOCA may also take place in the heat exchanger. In this case the accepted design of the heat exchanger (see Section 7.2) should prevent the direct contact of the activated primary LM coolant (gallium) with the secondary loop coolant (water).

8.1.3. Other Accidents

1. Plasma disruption and melting of the HHF cells will cause coolant ingress into the plasma chamber. Large drops of LM will easily self-drain down the walls to the bottom of vacuum chamber and into the divertor draining system. Micro-droplets will be removed during subsequent chamber baking-out. In general no dangerous consequences are expected.

2. Spilling of gallium in the reactor hall may cause its heating up due to after heat and corresponding environment temperature and pressure rise. However heat removal to floor and walls of the room may prevent considerable effects.

Two cases of coolant spilling will be considered:

- 1) coolant spilling from a divertor target module during maintenance;
- 2) coolant spilling from cold leg of the coolant loop.

8.1.4. Some Preliminary Conclusions

The following accidents have been selected for further consideration:

1. LOFA in two types of targets (HHF and LHF).
 - 1.1. LOFA resulting from an external event, two versions:
 - coolant stays inside cooling tubes;
 - coolant drained from cooling tubes.
 - 1.2. LOFA resulting from an internal event:
 - plugging-up of one or a few cooling tubes in a LHF target;
 - plugging-up of one or a few jet orifices in a HHF target.
2. LOCA
 - 2.1. LOCA resulted in external events.
 - 2.1. Spilling of liquid gallium in the reactor hall.

The corresponding trees of events are presented below.

Fig. 8-1. LHF target LOFA resulting from an external event.

Fig. 8-2. LOFA resulting from an internal event

Fig. 8-3. LOCA resulting from an internal event.

Fig. 8-4. LOCA resulting from an external event.

8.2. Activation and After Heat Dynamics

8.2.1. Initial Points

Gallium activation characteristics have been calculated to obtain activation after heat data for accident analysis (see Section 8.3). From this viewpoint the most interesting parameters were maximum values of specific activation energy release during reactor operation and after heat dynamics. In accordance with ARIES-II neutronic parameters [8.1] the divertor target neutron load is 2 MW/m^2 and estimated fluence is $16.4 \text{ MW}\cdot\text{year/m}^2$. It corresponds to 8.2 years of a continuous work with availability 1. Such a scenario of the divertor operation has been considered here. 46-energy groups flux obtained from [8.2] was used in activation calculations. The flux data were available for 3 points in the middle plane of ARIES-II blanket: at the first wall of the outboard blanket; at 10 cm and 20 cm inside the blanket (see Fig.8-5.).

Fig. 8-5. Neutron 46-energy groups flux for three points in the blanket

FRINDA code [8.3] PC version has been used to calculate the activation energy release. Activation cross-section library and nuclear-physical data were based on ACTL-82 [8.3] and data from [8.3, 8.4, 8.5, 8.6, 8.7, 8.8]. In particular, the decay energy values were calculated with data from [8.8, 8.10]. Gallium loop parameters from Table 4.4.1 are used here to obtain averaged after heat values for flowing gallium. A simplified schematic design of the Ga loop is shown in Fig. 8-6. The explanations of symbols in the scheme and gallium inventories in different parts of the Ga loop are shown in Table 8.2.1.1.

Fig. 8-6. Schematic design of the Ga loop

Table 8.2.1.1.
Gallium Inventories in Divertor Loops

No.	Scheme Symbol	Part of Ga-loop	Ga inventory, m ³ (rounded)
1	T _t	Top divertor targets	0.4
2	T _b	Bottom divertor targets	0.4
3	FL _t	Top divertor feeding lines	2
4	FDL _b	Bottom divertor feeding and draining lines	2.9
5	DL _t inboard	Top divertor inboard draining lines	1.5
6	DL _t outboard	Top divertor outboard draining lines	2.5
7		External loop equipment	18.5
	V _{total}	Total divertor gallium loops volume	28.2

It was supposed in calculations that neutron flux for parts 1 and 2 corresponds to that at the first wall; for parts 3 and 4 to the level at 20 cm inside the blanket; for parts 5 and 6 to that at 10 cm inside the blanket. Part 7 of the Ga loop outside the reactor gives no contribution to Ga activation. Angular distribution around the plasma taken from [8.11] was used to obtain actual after heat values (Table 8.2.3.2, Fig. 8-14).

8.2.2. Composition

Material composition used in after heat calculations is shown in Table 8.2.2.1. Gallium composition is based on data for Å-1 type gallium [8.12] but Ni content is as in Å-0 type [8.12]. Basing on the earlier results for gallium activation characteristics [8.14] we expected that from the viewpoint of accident analysis (i.e. for period of time about 2 weeks) there is no significant contribution into after heat from impurities in gallium.

Table 8.2.2.1.
Material Composition, Weight Parts

Gallium	Ga	Al	Mg	Si	Zn	Ni	Pb	Cu	Fe
5.9 g/cc	0.998956	1.0E-05	1.0E-05	1.0E-05	1.0E-04	4.0E-06	6.0E-04	3.0E-04	1.0E-05
V-alloy	V	Cr	Ti	Fe	Ni				
6.1 g/cc	0.7984	0.15	0.05	0.0001	0.00001				

In Table 8.2.2.2 the activation reactions taken into account in calculations are shown.

Table 8.2.2.2.
ActivationReactions List

Al-27	n,p	Mg-27	9.462 m	Fe-54	n,g	Fe-55	2.68 y	Zn-64	n,2n	Zn-63	38.1 m
Al-27	n,g	Al-28	2.24 m	Fe-54	n,p	Mn-54	312.5 d	Zn-66	n,p	Cu-66	5.1 m
Al-27	n,a	Na-24	15.02 h	Fe-54	n,a	Cr-51	27.704 d	Zn-66	n,2n	Zn-65	243.9 d
Si-28	n,p	Al-28	2.24 m	Fe-56	n,p	Mn-56	2.5785 h	Zn-68	n,a	Ni-65	2.52 h
Si-29	n,p	Al-29	6.56 m	Fe-56	n,2n	Fe-55	2.68 y	Ga-69	n,2n	Ga-68	68.1 m
Si-29	n,np	Al-28	2.24 m	Fe-58	n,g	Fe-59	44.496 d	Ga-69	n,p	Zn-69	55.6 m
Si-30	n,a	Mg-27	9.462 m	Cr-50	n,g	Cr-51	27.704 d	Ga-69	n,p	Zn-69m	13.76 h
Si-30	n,g	Si-31	2.62 h	Cr-52	n,2n	Cr-51	27.704 d	Ga-69	n,p	Cu-66	5.1 m
Ti-46	n,p	Sc-46	83.83 d	Ni-58	n,p	Co-58	70.916 d	Ga-69	n,g	Ga-70	21.15 m
Ti-46	n,t	Sc-44	3.927 h	Ni-58	n,2n	Ni-57	36.08 h	Ga-71	n,2n	Ga-70	21.15 m
Ti-47	n,np	Sc-46	83.83 d	Ni-60	n,p	Co-60	5.271 y	Ga-71	n,p	Zn-71	2.45 m
Ti-47	n,p	Sc-47	3.345 d	Mn-55	n,g	Mn-56	2.5785 h	Ga-71	n,p	Zn-71m	3.94 h
Ti-47	n,d	Sc-46	83.83 d	Mn-55	n,2n	Mn-54	312.5 d	Ga-71	n,na	Cu-67	61.92 h
Ti-48	n,np	Sc-47	3.345 d	Cu-63	n,2n	Cu-62	9.74 m	Ga-71	n,a	Cu-68	31 s
Ti-48	n,p	Sc-48	43.7 h	Cu-63	n,a	Co-60m	10.47 m	Ga-71	n,g	Ga-72	14.1 h
Ti-48	n,d	Sc-47	3.345 d	Cu-63	n,p	Ni-63	100.1 y	Pb-204	n,2n	Pb-203	51.88 h
Ti-48	n,a	Ca-45	163.8 d	Cu-63	n,a	Co-60	5.271 y	Pb-204	n,p	Tl-204	3.78 y
Ti-49	n,np	Sc-48	43.7 h	Cu-63	n,g	Cu-64	12.701 h	Pb-204	n,n'	Pb-204m	66.9 m
Ti-49	n,p	Sc-49	57.4 m	Cu-65	n,2n	Cu-64	12.701 h	Pb-204	n,t	Tl-202	12.23 d
Ti-49	n,d	Sc-48	43.7 h	Cu-65	n,p	Ni-65	2.52 h	Pb-206	n,p	Tl-206	4.2 m
Ti-50	n,a	Ca-47	4.535 d	Cu-65	n,na	Co-61	1.65 h	Pb-206	n,t	Tl-204	3.78 y
V-50	n,2n	V-49	330 d	Cu-65	n,a	Co-62	1.5 m	Pb-206	n,a	Hg-203	46.6 d
V-50	n,a	Sc-47	3.345 d	Cu-65	n,a	Co-62m	13.91 m	Pb-207	n,p	Tl-207	4.77 m
V-51	n,na	Sc-47	3.345 d	Cu-65	n,g	Cu-66	5.1 m	Pb-207	n,na	Hg-203	46.6 d
V-51	n,a	Sc-48	43.7 h	Zn-64	n,p	Cu-64	12.701 h	Pb-208	n,g	Pb-209	3.253 h

8.2.3. Calculation Results

The calculation results are presented in Tables 8.2.3.1, 8.2.3.2 and in Figs. 8-7, 8-8, 8-9, 8-10, and 8-14. In particular in Figs. 8-7, and 8-8 the energy release in gallium in part 1 of the Ga loop (without taking into account the coolant movement and the neutron load poloidal distribution) is shown.

Fig. 8-7. Energy release in gallium at 2 MW/m² (during operation).

Fig. 8-8. Energy release in gallium after 8.2 year exposure at 2 MW/m² (cooling down)

One can see that the energy release value in gallium is rather high during operation and ~3 days after shut-down. At the moment of shut-down it equals to $\sim 4.2 \cdot 10^{13}$ MeV/s·cm³ (for neutron wall load 2 MW/m²). So if there are any zones with large amounts of stagnant gallium staying permanently under exposure during all the operation time it may result in serious problem with the after heat. In Table 8.2.3.1 the energy release in unflowing gallium after 8.2 year exposure for certain points of Ga loop (with respect to corresponding values of neutron wall load [8.11]) are shown.

Table 8.2.3.1.
Energy release in unflowing gallium after 8.2 year for the points of Ga loop

Scheme Symbol	Point in	E, MeV/s·cm ³
T _t	Divertor targets	1.8E+13
FL _t	Divertor feeding lines	8.7E+12
DL _t inboard	Top divertor inboard draining lines	2.8E+13
DL _t outboard	Top divertor outboard draining lines	4.5E+13

In accordance with the Ga loop parameters (see Table 8.2.1.1) an average energy releases (E_{av}) in flowing gallium coolant at shut-down is considerably lower:

$$E_{av} = \left(E_1 \cdot (V(D_t) + V(D_b)) + E_3 \cdot (V(FL_t) + V(FL_b)) + E_2^{inboard} \cdot V(DL_t^{inboard}) + E_2^{outboard} \cdot V(DL_t^{outboard}) \right) / V_{total}$$

where $E_{1,2,3}$ - energy release values in the corresponding points with respect to angular distribution around the plasma,

V - volumes of the gallium loop parts corresponding to Table 8.2.1.1.

So averaged energy release value in gallium at the moment of shut-down is $\sim 6.8 \cdot 10^{12}$ MeV/s·cm³.

For V-alloy of the divertor targets similar dependencies (for neutron wall load 2 MW/m²) are shown in Fig. 8-9, and 8-10. One should keep in mind that, unlike gallium, V-alloy stays in the blanket permanently, so the after heat values given in Figures are the actual ones.

Fig.8-9. Energy release in V-alloy of the divertor target at 2 MW/m² (during operation).

Fig. 8-10. Energy release in V-alloy of the divertor target after 8.2 year exposure at 2 MW/m² (cooling down)

In Fig. 8-11 a neutron flux normalized by total flux value for three points across the blanket width is shown. It is clear that we can not expect any considerable changes in gallium activation dynamic depending on the point because shape of flux spectrum differs appreciably only for high energy neutron groups while major contribution to the after heat is due to thermal reaction Ga-71(n, γ)Ga-72. But for V-alloy it is just the high energy part of neutron spectrum that gives main contribution into after heat value due to threshold reactions V-51(n, α)Sc-48, Ti-48(n,p)Sc-48, V-51(n,na)Sc-47. So the relative input of these reactions at the blanket periphery will be smaller. It results in more smooth cooling dynamic for points 2 and 3.

Fig. 8-11. Normalized neutron flux

The after heat in the divertor targets has been estimated in 2-layer composition approximation: 3 mm of gallium and 3 mm of V-alloy. Let us consider the situation with activated products dominating during first 10 days after shut-down. This time interval is most important from the view point of accidents dynamic. Data presented in Figs. 8-8 and 8-10 show that we should take into account the following reactions:

for gallium: Ga-69(n,2n)Ga-68, Ga-71(n,2n)Ga-70, Ga-71(n, γ)Ga-72;

for V-alloy: V-51(n, α)Sc-48, V-51(n,n α)Sc-47, Ti-48(n,p)Sc-48.

In fact for each case there is only one obviously dominating reaction:

for gallium: Ga-71(n, γ)Ga-72;

for V-alloy: V-51(n, α)Sc-48.

Gamma-rays produced by Ga activation products have rather high energy (see Fig. 8-12) and since the divertor target structure is relatively thin only a small fraction of energy released in material will be actually transformed into after heat in situ. An estimated transformation factor is approximately 0.1 (according to that estimate such factor is valid up to ~0.8-1.0 cm of total target thickness). For V-alloy the situation is similar (see Fig. 8-13) except Ti-48(n,a)Ca-45 reaction. Ca-45 isotope has 100% β -decay with energy release in

Fig. 8-12. Decay gamma ray spectra for gallium activated products

Fig. 8-13. Decay gamma ray spectra for V-alloy activated products

form of β -particles. So, all this energy is absorbed in V-alloy material. But this reaction becomes dominating in energy release after ~ 1 month and can not effect the after heat dynamic within the considered time interval of 10 days.

Taking into account the relative contributions from gallium and V-alloy activated products into the total value of energy release for the first 10 days after shut-down a transformation factor of 0.1 can be adopted for the entire divertor target composition. The final after heat data (with transformation factor for Ca-45 equal 1) are shown in Table 8.2.3.2 and in Fig. 8-14.

Table 8.2.3.2.
After heat in Ga-V-alloy layer of the divertor target, W/cm³

Time	0	1h	3h	12h	1d	3d	30d	1y	3y	10y	30y
Total	1.2E-01	8.7E-02	6.9E-02	4.4E-02	2.7E-02	5.8E-03	1.1E-04	1.4E-05	7.4E-07	4.6E-08	3.7E-09
Ga	1.1E-01	7.5E-02	5.7E-02	3.4E-02	1.9E-02	1.8E-03	3.4E-07	1.7E-07	8.5E-08	3.0E-08	2.7E-09
V-alloy	1.2E-02	1.2E-02	1.2E-02	1.0E-02	8.4E-03	4.0E-03	1.0E-04	1.4E-05	6.5E-07	1.6E-08	1.1E-09

Fig. 8-14. After heat in Ga-V-alloy layer in the top divertor target.

8.3. Accident analysis

The goal of quantitative analysis of the selected accidents is determination of time until structure temperature is below some permissible level during an accident. The obtained results are presented below.

8.3.1. Loss of Coolant Flow Accident

The analysis of LOFA is based on calculations of two-dimensional transient temperature distribution in the cross section of LHF and HHF targets. The initial data for calculations are presented below:

Heat flux, LHF target	2 MW/m ²
HHF target (maximum)	14 MW/m ²
Inlet coolant temperature	80°C
Heat transfer coefficient, LHF target	17 kW/(m ² K)
HHF target	70 kW/(m ² K)
Nuclear heat density in all materials	10 MW/m ³
Thermal radiation emissivity	0.7
Specific view factor	0.31

The after heat decay curves for the structural material of the divertor targets and coolant are presented in Section 8.2.

The following versions of LOFA were considered:

- 1) LOFA in all channels of a LHF target without plasma shut-down at the accident beginning and without natural convection (version of LOFA resulting from an external event);
- 2) LOFA in one or two channels without plasma shutdown at the accident beginning and without natural convection (version of LOFA resulting from an internal event).

The calculation results have shown that in case (1) the structure temperature within 13 seconds will rise up to 1500°C. In case (2) the same temperature level will be reached within 18 seconds (with only one channel plugged-up) or within 14 seconds (with two).

If the reaction in plasma chamber is shut down at the beginning of accident and the chamber wall temperature stays near the normal level maintained by operation of the first wall cooling system, then heat from the fault divertor target will be removed by radiation to the first wall. Transient temperature of the LHF target is shown in Fig. 8-15. A maximum quasi-steady-state level at 460°C is reached in 3-4 hours after accident beginning and then the target temperature will decrease. Besides this in Fig. 8-15 the calculation results for a pure hypothetical conditions with no heat removal from the fault target are shown: in this case the maximum structure temperature of 1000°C will be reached within 5 hours.

All calculations have been done with an assumption that the heat flux on the target drops down to zero just at the accident beginning. However under real conditions it will take some time, may be about 10-30 seconds. In order to evaluate this effect calculations of the LHF target temperature have been done under LOFA conditions and the following assumptions:

- heat flux drops down to zero abruptly at the first moment, or decreases linearly to zero within 10 seconds, or remains constant;
- coolant flow rate drops down to zero at the first moment.

The calculations have shown that non-zero time of the thermal load decrease combined with a conservative assumption of abrupt cooling termination can result in somewhat higher maximum LHF target temperature (770°C) which is reached within 6 seconds after accident beginning and then the temperature will be determined by the after heat in structural materials and coolant and the available heat removal (Fig. 8-16).

8.3.2. Loss of Coolant Accident

First quantitative analysis of LOCA for the LHF target shows that the structure temperature level at 1500°C will be reached within a very short time (~5 s) if the reaction in plasma is not shut down. However if it is then heat removal by radiation to the cooled first wall will insure a safe temperature level (Fig. 8-15).

Spilling of liquid gallium in the reactor hall is a possible consequence of both external LOCA and some other accidents, e.g. during repair or maintenance operations.

The model for analysis of consequences caused by coolant spilling is based on calculations of temperature distribution across a coolant layer formed on some surface, e.g. on

Temperature of LHF target under LOFA and LOCA conditions, reactor shut down
 1- LOFA without any heat removal
 2- LOFA, LHF target is cooled by heat radiation
 3- LOCA, LHF target is cooled by heat radiation

Fig. 8-15

Temperature of spilled gallium. 1 – thickness of spilled gallium layer –
2 – thickness of spilled gallium layer – 10cm.

Fig. 8-16

a concrete floor. The coolant temperature is determined by heat release in it and heat removal to the floor (by means of thermal conductivity and natural convection) and to the walls and ceiling (by thermal radiation). The after heat decay dynamics in the coolant have been accounted for using data from the previous Section. The calculations have been done under the following assumptions:

- 1) the spilled coolant layer thickness is about 1 cm, thus the after heat level corresponds to that in the divertor target; or
- 2) the spilled coolant layer thickness is about 10 cm, then the after heat level due to much higher transformation factor (see Section 8.2) is greater by an order of magnitude;
- 3) temperature of ambient gas is constant (30°C).

The calculation results are presented in Fig. 8-17.

8.3.3. Conclusions

Both LOFA and LOCA in the LHF targets is not dangerous if the reactor is shut down within the first ten seconds after the accident beginning and the first wall cooling system is working. In this case the LHF target temperature will not exceed 500°C. LOFA with 10 seconds reactor shut-down time and no heat removal will lead to the temperature rise up to 1000°C within the first 5 hours after the accident beginning.

LOFA calculations taking into account various scenarios of the heat flux decrease during plasma shut-down have demonstrated this to be an important issue that must be included into further analyses. Spilling of the LM coolant won't lead to any appreciable temperature rise if thickness of the spilled coolant layer is about 1 cm, but there may be major consequences if it is about 10 cm.

8.4. Tritium Dose Estimate at Gallium Spilling Accident.

As it was noted in Section 8.1.4 the most serious events with gallium spilling are the following:

1. Coolant spilling from divertor target module during maintenance.
2. Coolant spilling from cold leg of the coolant loop.

Temperature of LHF target under LOFA conditions.
1- LOFA without any heat removal, under normal heat loading.
2- LOFA under linear heat flux dropping during 10s.
3- LOFA under momentarily heat flux dropping.

Fig. 8-17

Some preliminary estimates of tritium doses for these accidents were performed. For the first case tritium release into the reactor hall was estimated as 0.2-4 g depending on the efficiency of Ga-cleaning system during previous operation. For the second case the release value can be as high as 2-40 g. Let us consider the worst case, 40 g of tritium release. As a tritium dose characteristic a dose equivalent at 1 km distance from the release point was used here. The dose equivalent was determined according to the method described in [8.1]. In the scope of that approach we assumed no containment credit, quality factor (QF) 2 for tritium and ground release at worst weather conditions (category F by Pasquill). In such case the tritium dose factor at 1 km distance was 1.63 mSv/g and an accident dose was calculated as 85 mSv (accident dose limit is 100 mSv). For the very conservative assumptions made above we can consider this value as an upper limit estimate of the tritium dose due to an accident with Ga coolant. If we accept the ITER approach [8.1] (QF=1, 2% confinement release, tritium dose factor 0.5 mSv/g) then 50-yr. EDE (Effective Dose Equivalent) commitment at 1 km distance will be 0.4 mSv. This value is only 4 times higher than the public dose due to normal operation effluents (0.1 mSv/yr.) adopted in ITER.

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9. CONCLUSIONS

The performed design study was the first attempt of a complete divertor system analysis based on a new LM cooled divertor targets concept. It seems that such major objectives of this work as to insure higher permissible thermal loads to the divertor targets (up to 15 MW/m²) and to demonstrate the feasibility of a divertor design with easy maintenance and high duty cycle potential have been successfully achieved.

The other objective of establishing the most important mutual requirements between the LM cooled divertor and other tokamak reactor systems can be also considered fulfilled: on one hand, the compatibility of the LM divertor targets with the double-null gaseous divertor configuration has been demonstrated and, on the other hand, the major impact from the divertor system on the tokamak reactor revealed. In this respect the most critical issues are:

- the necessity of using some part (though not large) of the blanket cooling ducts for draining LM from the top divertor;
- the need for the rooms directly adjacent to the tokamak reactor for placing the divertor LM loop equipment to minimize gallium inventory.

The last two points should be added to the list of key design issues given in Section 2.4 that were envisaged from the beginning. Most of them require R&D work and the design solutions suggested in this study will allow to specify the corresponding tasks more exactly.

As to the recommendations for further design efforts, the following directions can be named:

- system analysis of the reactor with single-null bottom divertor configuration taking into account the specific advantages of the LM divertor system (in the first instance the feasibility of high thermal loads to the targets);
- more detailed analysis of transient regimes during accidents both in the divertor targets and in the entire LM loop, optimization of the loop design and equipment parameters;
- analysis of the divertor system operation in a pulsed plasma discharge regime (in a PULSAR type reactor).

In general, finishing this work we hope that the presented design will give start to the development of a viable divertor system relevant to use in various fusion reactors with high performance.

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