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HELIUM-COOLED SOLID BREEDER BLANKET FOR ITER

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ABSTRACT

This paper summarizes the latest results of a design study of a helium-cooled solid breeder blanket for ITER. Attractive features of this design include the following : 1) There is a significant design margin since only part of the allowable solid breeder temperature window needs to be used. 2) There is an expanding data base available from solid breeder experiments carried out internationally. 3) The solid breeder can be designed to operate at high reactor-relevant temperature, while the helium is kept at moderate temperature and pressure for safety and reliability. In addition, since helium is a gas, it can be run so as to optimize the structure temperature and accommodate long term power variation without incurring any substantial pressure penalty. 4) The use of helium, an inert gas minimizing any chemical reaction and corrosion, in combination with a low activation solid breeder, is a safety advantage. An extensive list of the blanket operating parameters is provided and key factors are discussed.

INTRODUCTION

For a total fusion power of 600 MW, ITER would require 100 kg of tritium for a three-year period of full-power operation. Only a portion of this can be available from Canadian heavy water reactors at a cost of about $10,000 per gram.[1] Thus, a tritium-producing blanket is required to supply ITER at least in part with its own tritium.

The tritium-producing blanket for ITER will occupy up to 90% of the space surrounding the first wall. This premium space in the only fusion engineering facility during the early part of the 21st century should be effectively utilized to provide as much information as possible. If the blanket is reactor relevant, then the data obtained from its operation can be extrapolated to the demonstration device beyond ITER. On the other hand, it must be recognized that the ITER blanket will have to be designed, constructed and operated based only on information from testing in non-fusion facilities, without prior fusion testing. Therefore, high performance and availability cannot be guaranteed.

A good strategy for the ITER blanket which carefully balances the above considerations is: 1) design the ITER blanket with reactor relevant materials and configuration, 2) operate parts of the blanket at reactor-relevant conditions where the information is most valuable and the penalty is minimum, 3) operate the rest of the blanket at the most reliable conditions, and 4) provide flexibility in the blanket design to accommodate a variety of ITER operating conditions.

This paper summarizes the final results of a design study for a helium-cooled solid breeder blanket which has been developed based on the above strategy. The blanket configuration and the breeder material and operating conditions are all reactor relevant. The attractiveness of this blanket concept is based on several factors, such as: 1) a significant design margin since such blanket configurations have been studied in detail in the past for high-fluence, high-wall-loading commercial reactor designs;[2,3] 2) the large data base available for solid breeders; 3) good tritium release from solid breeder operation at high temperature; 4) the added safety and reliability of operating the helium at moderate temperature and pressure; and 4) the flexibility provided by helium for temperature adjustment, for power variation accommodation and for first wall bake-out. This paper briefly describes the blanket configuration and summarizes the latest results of the analysis of the blanket operating parameters.

BLANKET CONFIGURATION

The proposed configuration is based on previous designs[2,3] and consists of a number of independent canisters, lying side by side in the poloidal direction on the ITER outboard region, as shown in Fig. 1. Each canister contains a bundle of staggered rods lying in a toroidal axis, as shown in Fig. 2. A rod bundle configuration is chosen for several reasons including the flexibility to allow for spatial variation in heat generation and to optimize the neutronics performance by tailoring the rod sizes and compositions within a canister; and the proven thermal hydraulic performance of this type of configuration. The helium coolant comes in along the side wall to provide for the cooling of the first wall and then enters the canister at the first wall and flows over the rod bundle before exiting the canister at the back. A porous mesh is provided along the inner side wall of the helium channel so that a virtually stagnant layer of helium inside the mesh will insulate the cooler side wall flow from the hotter rod-bundle flow.

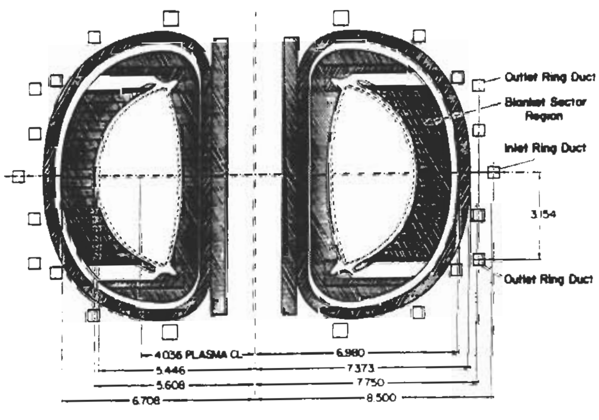

Fig. 1 Cross-section of ITER showing canister layout

Fig. 3 shows a typical solid breeder/multiplier rod. It consists of an inner solid breeder cylinder surrounded by an annular gap of sphere-pac Be and helium which can be tailored to provide the required thermal resistance between the hot solid breeder and cooler helium by varying parameters such as the gap thickness and the porosity. An helium purge flows through both the solid breeder and Be regions for tritium removal. Any of the major lithium ceramic solid breeders can be used in this blanket configuration. However, based on the solid breeder comparison study of Ref. 4, Li_4SiO_4 is chosen as the reference material here.

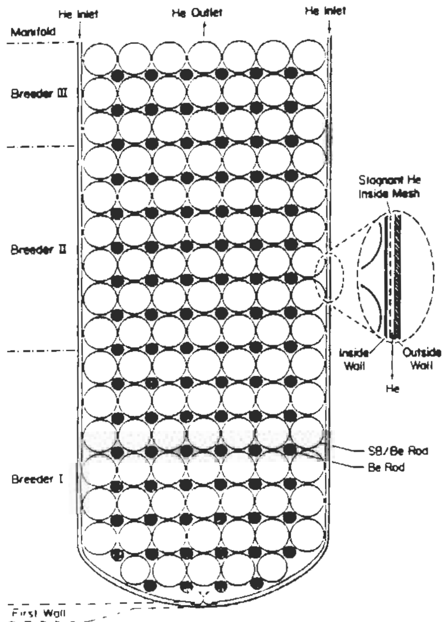

Fig. 2 Helium-cooled solid breeder blanket canister for ITER

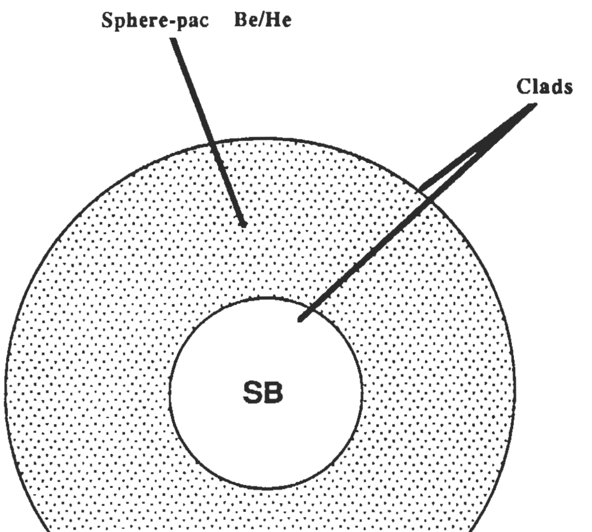

Fig. 3 Solid breeder/multiplier rod configuration

PERFORMANCE PARAMETERS

Table 1 shows an extensive list of the blanket performance parameters. Key factors associated with the blanket performance are discussed below. They include: 1) the neutronics performance, mostly in terms of the tritium breeding ratio (TBR); 2) issues relating to the analysis, predictability and controllability of the thermal conductance of the annular Be sphere-pac region; 3) possible ways of accommodating a LOCA; 4) the mechanical design of the canister 5) the leakage of the helium coolant through the first wall to the plasma; 6) the thermal-hydraulics, in particular whether the helium coolant can be operated at moderate pressure; 7) the ability to accommodate power variations; 8) the reliability of the rod bundle configuration; and 9) the tritium inventory.

Neutronics

Section 1C of the table indicates that a tritium breeding ratio of 1.35 is obtained from the 1-D toroidal model computations. The 3-D calculations yield a TBR of 1 for the outboard blanket. If a single-wall divertor configuration is used and if the blanket is placed so as to fill all the regions around the plasma except for a 6m high space-restricted inboard, the 3-D TBR increases to 1.21. Thus, there is enough margin to indicate that even after allowing for uncertainties, a TBR of unity can be obtained. A detailed analysis of the neutronics performance and of the trade off between the safety advantages and space penalty of using helium as compared to water for the separate inboard shield coolant system is given in a companion paper.[5]

Thermal Conductance of Be/He Sphere-pac Region

The concept of using a pebble bed of Be with He for the thermal resistance gap required between the hot solid breeder and cold coolant for ITER designs is applicable to water-cooled or helium-cooled solid breeder and also to water-cooled LiPb blanket options. This proposed gap configuration has several advantages including the flexibility of setting the desirable thermal resistance by tailoring the bed characteristics, the possibility of controlling the effective thermal conductivity by adjusting the purge flow through the bed. The use of a purge flow also provides for tritium removal in the Be region and thus creates an additional purge barrier against tritium permeation from the solid breeder region. Finally, the purge flow can also help to remove the afterheat during shutdown conditions or in the event of a LOCA, and could provide some flexibility in controlling the gap conductance by purge flow adjustment.

The maximum solid breeder temperature must be accurately predicted to allow for the most flexibility in case of power variation. The effective thermal conductivity of the proposed Be particle bed must then be accurately predicted and reproducible. Due to the poor extrapolation of existing models predicting the effective thermal conductivity of beds to conditions other than those of the experiments on which they are based, experimental data are required for ITER type conditions. Key conditions that should be reproduced in such an experiment include the porosity (20-30%), the thermal conductivities of the gas and solid (He and Be) and flowing He. More details about the analysis of the proposed Be/He sphere-pac concept for providing thermal resistance can be found in Ref. 6.

LOCA Conditions

Following a LOCA, the temperature of the first wall increases faster than that of the solid breeder rods for adiabatic conditions. If the maximum allowable temperature for reusable structure is assumed to be half of the melting point (which is about 1700 K for austenitic steels), then it takes about one hour for the first wall to reach this limit under adiabatic conditions for the solid breeder blanket design considered here. Under the same conditions, it takes about $1\frac{1}{2}$ days for the Li_4SiO_4 solid breeder to reach the maximum allowable temperature of 1000 K (based on thermal sintering). The afterheat values used for these calculations were based on 1 FPY of operation.

Including the effect of radiation heat transfer to the inboard enables the maximum temperature of the structure to stay below its limit. The solid breeder temperature also stays beneath its limit in this case. Use of the purge flow in the canister rods greatly helps to accommodate a LOCA. Even if radiation to the inboard is excluded, the solid breeder and structure would not reach their maximum allowable temperatures for a purge flow average velocity of about 0.005 m/s or more, assuming radiative heat transfer between the different outboard zones. The purge pressure drop for that case is reasonable, about 0.5 atm/m. Note that a system emissivity of 0.33 was assumed for all radiation calculations. Ref. 7 addresses the LOCA issues and discusses the results in more details.

Mechanical Design and Helium Leakage

The mechanical design of the canister is carried out with the following objectives: 1) Each canister must be structurally independent to avoid the "domino" effect in the event of one canister failing; 2) The canister shape, in particular in the high temperature first wall region, must be such as to fully utilize the space available for tritium breeding; and 3) The amount of structure used should be minimized to make maximum use of the available neutrons for tritium breeding. In addition, it is desirable to reduce the number and length of welds facing the plasma as much as possible to reduce any helium leakage to the plasma and for better structural integrity.

From Section 1B of Table 1, the reference dimensions of the canister are 1.2m long and 0.33m wide. The structural design consists of an outer wall 0.5 cm thick and an inner wall 0.2 cm thick. The clearance between the two walls is 0.3 cm and is used as a channel for the helium coolant to cool the walls. Thin ribs are located at intervals to join the two walls and to provide structural strength. Maximum space utilization is obtained with a rectangular canister. However, calculations show that the stresses in this case are up to an order of magnitude over the allowable. At the other extreme, low stresses (about 40 MPa) are obtained with a perfectly cylindrical geometry at the expense of minimum space utilization. The optimum design lies between these and the aim is to produce a design with the maximum possible radius of curvature of the first wall for space utilization and the minimum possible number and thickness of ribs for structure minimization.

Fracture mechanics calculations show that for ITER conditions, crack propagation is extremely unlikely to occur. An estimate of helium leakage to the plasma through the welds is difficult but, based on HTGR observations, the leak rate would be only a fraction of the helium production rate in the plasma (of the order of 1%), which is acceptable. Ref. 8 describes in more detail the mechanical design study and the helium leakage analysis.

Thermal-Hydraulics and Power Variation Accommodation

The helium coolant can be operated at moderate temperature and pressure because of the relatively low power density of ITER (the average outboard wall loading is assumed to be 1.5 MW/m^2). Fig. 4 shows the helium pumping power required for the canisters as a function of the inlet pressure for different helium temperature rises. A range of moderate pressure and temperature combinations is available for a reasonable pumping power. As shown in Section 1D of Table 1, the chosen reference pressure is 15 atm and the inlet and outlet temperature of the helium are 50°C and 300°C respectively. The total pumping power for the blanket (including the outside coolant circuit) is about 18 MW with a corresponding pressure drop of only 1 atm.

Even if the thermal conductivity of the Be sphere-pac region cannot be controlled, this blanket design can still accommodate increase in power of up to 50% above the reference value. This is achieved by making full use of the large solid breeder operating temperature window (from about 600°C to 1000°C for Li_4SiO_4) in combination with an increase in the helium flow rate with an acceptable increase in the helium pumping power. Note that, with helium as a coolant, any decrease in power can be accommodated (i.e. the solid breeder temperature can be kept high enough for tritium release) by reducing the helium rate of flow or increasing the helium inlet temperature.

Reliability

The blanket design consists of many thousands of solid breeder/multiplier rods. The rod design provides multiple barriers

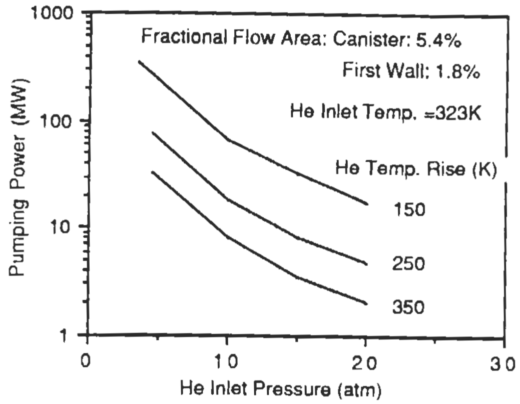

Fig. 4 Canister He pumping power as a function of the inlet pressure for different temperature rises

against contamination (from the breeder, tritium must penetrate the breeder cladding, the multiplier cladding, and ultimately the coolant containment). However, failure of these rods is a concern since a direct consequence would be tritium contamination of the main helium flow. An analysis was done to examine the probability of rod failure and the likely consequences.

The reliability of the blanket is often thought to decrease as the number of rods increases. The overall consequences of failure, though, do not necessarily follow this same trend, since the reduced consequences of failure per rod can outweigh the increased probability of rod failure in designs with a larger number of smaller rods. To address this issue, the probability of achieving a given level of consequences was investigated in designs with different numbers of rods. The calculations assume that tritium contamination is one of the most serious consequences of rod failure, since the breeder rods are not essential to the structural or coolant containment functions of the blanket.

Fig. 5 shows the probability of tritium release for different solid breeder volumes from which tritium escapes due to rod failure. A probability of 0.005 is arbitrarily assumed for 1 mm of crack appearing per meter of weld. The actual value is unknown, but does not affect the

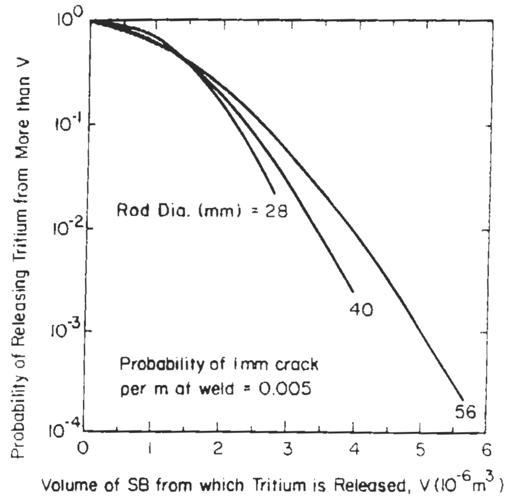

Fig. 5 Probability of releasing tritium from more than a given SB volume, V, for different rod diameters (i.e., different number of rods)

general conclusion. A different value will merely shift the point at which more rods are more reliable than fewer rods. Based on this analysis, if a small number of rod failures are acceptable, then the probability of exceeding contamination limits can actually be more in designs with fewer rods. This statement relies heavily on the assumption that a single rod failure can be tolerated without requiring replacement of the entire blanket module (or down-time of any kind due to repair).

The above discussion shows that, while there seem to be many opportunities for failure, the design is very robust, in the sense that most "failures" result in modest or insignificant consequences since such a design with many rods dilutes the source term. The rod cylindrical geometry is one which accommodates stresses well, and the relatively small rod size leads to minimal internal gradients; thus, the most likely rod failure mechanism is minor cracking, probably atwelds. For such rod failures resulting in minor contamination, it is shown that a larger number of rods is not necessarily worse.

Tritium Inventory

The tritium inventory was estimated as the sum of the diffusive, solubility and adsorption inventories. Each canister was first divided into three regions and the inventory components in each region were calculated for the corresponding temperature distribution and tritium production. The overall tritium inventory was then obtained by summing over the total number of canisters.

Expressions for estimating the diffusive and solubility inventory were based on Ref. 4. The diffusive inventory calculations assume a grain radius of 2 μm, a diffusion pre-exponential constant of 2.1×10^{-11} m^2/s and an activation energy of 64 kJ/mol for Li_4SiO_4. The solubility inventory was calculated from a form of Sievert's Law. However, since no data were available for Li_4SiO_4 and since it is believed that the solubility inventory in Li_4SiO_4 is less than that of Li_2O, the solubility was estimated as being 1/3 of the corresponding solubility in Li_2O.

The surface adsorption inventory was calculated by equating the expressions for the rates of adsorption and desorption[9], and solving for the resulting coverage in steady state, assuming a 1% hydrogen concentration in the purge. The heat of adsorption for Li_4SiO_4 was assumed to be 100 kJ/mol.

As shown in Section 1E of Table 1, the total tritium inventory is estimated as being 13.7g, with the major component being adsorption. The estimates are subject to large uncertainties (up to about a factor of 100) but are low enough as to be acceptable even with the inclusion of the uncertainties.

CONCLUSIONS

The helium-cooled solid breeder blanket concept is an attractive candidate for the basic tritium-producing blanket in ITER from several points of view, such as the well-studied configuration, the expanding data base, the safety, and the information it provides on reactor-relevant solid breeder operation at reactor temperatures. The design can allow a net TBR in excess of unity including a margin for uncertainties. The helium pressure and inlet and outlet temperatures are moderate (15 atm, 323 K and 573 K) and do not pose any significant penalty on the structural design. Helium cooling provides flexibility in selecting optimum temperature for structure and in allowing for power variation. A practical concept consisting of a Be packed bed has been proposed for providing the required thermal resistance gap between the solid breeder and the coolant. The concept is applicable to other blanket concepts and should be verified experimentally for key prototypical parameters (such as the porosity and the material thermal conductivities). The blanket is quite robust to failures assuming that the worst consequence is tritium contamination of the coolant. Helium leakage to the plasma does not seem to be a problem but more accurate estimates would be required for a definite conclusion. Other issues that need to be addressed include helium manifolding and the irradiation effects on solid breeder material properties, in particular those affecting the tritium transport. Note, however, regarding that last issue, that the estimated tritium inventory is low and uncertainties of up to a factor of 100 can probably be tolerated.

Table 1

Performance Parameter List of Helium-Cooled Solid Breeder Blanket Proposed for Outboard of ITER

1A. General Description

- Materials

Blanket	
Coolant	He
Breeder	Li_4SiO_4
Li-6 Enrichment	90%
Neutron Multiplier	Be
Structure	PCA
Reflector	C
Shield	
Shield(s)	B_4C
Structure	SS
Coolant	He

1B. Configuration and Dimension

Breeder/Multiplier rods in canister

- Canister	
Length	1.1 to 1.3m
Width	0.33m
Number	480
- Rod	
Diameter	0.041m
Spacing between Rods	0.002m
Number per Canister	106
- First Wall	
Thickness	0.005m
Channel Width	0.003m
Second Wall Thickness	0.002m

1C. Neutronics

- Average/Maximum Outboard Neutron Wall Load: 1.5/1.9 Mw/m^2

- Compositions and Dimensions for Neutronics Calculations

Region	Thickness (m)	Volume Fraction: Structure	Breeder (80% TD)	Multiplier (90% TD)	He	Shield
First Wall	0.02	0.4			0.6	
First Breeder	0.5	0.09	0.07	0.46	0.38	
Second Breeder	0.1	0.16	0.37	0.14	0.33	
Plenum	0.1	0.09			0.91	
Reflector	0.1	0.05			0.1	0.85 C
Shield	0.3	0.4			0.1	0.5 B_4C

- Tritium Breeding Ratio

1D-Poloidal (Cylindrical) Model with 100% Coverage (onedant)	1.6
1D-Toroidal Model with 100% Coverage	1.35
Net 3D with all Geometrical Details (estimation) (MCNP with ENDF Version V Rev.II Library)	1.0

- Maximum Nuclear Heating Rates

Breeder	$16W/cm^3$
Neutron Multiplier	$16W/cm^3$
Structure	$9W/cm^3$
Reflector	$0.27W/cm^3$
Shield Materials	$0.6W/cm^3$

- Energy Per Fusion Neutron

Blanket	14.2 MeV
Shield	0.38 MeV
Total	14.6 MeV

- Nuclear Responses in the TF Coils

Maximum Critical Current Change at 2.5 FPY	-27%
Maximum Fast Neutron Fluence in the Superconductor	$2.8x10^{18}n/cm^2$
Maximum Induced Resistivity in Copper Stabilizer @ 1FPY	$2.1x10^{-7}$ ohm/cm
Maximum Atomic Displacement in Copper Stabilizer	$8.63x10^{-4}$ dpa/FPY
Maximum Dose to Electrical Insulator (neutron and gamma)	$3.55x10^9$ rad
Maximum Nuclear Heating in the Winding Pack	$1.36x10^{-3}$ W/cm^3
Nuclear Heating in the Winding Pack at Midplane	$0.26x10^{-3}$ W/cm^3
Nuclear Heating in the Magnet Case at Midplane	$4.72x10^{-3}$
W/cm^3Total Nuclear Heating in TF Coils	1700W per TF coil

- Radiation Dose in the Reactor Hall at:

1 Day After Shutdown	1 mrem/h

1D. Thermalhydraulics

- Coolant

Inlet/Outlet Temperature	50°C/300°C
Inlet/Outlet Pressure	1.5 MPa/1.4 MPa

- Breeder

Minimum/Maximum Temperature	325°C/500°C

- Structure

Maximum First Wall Temperature	275°C
Minimum/Maximum Temperature at:	
Coolant Interface	50°C/300°C
Breeder Interface	325°C/400°C
Multiplier Interface	150°C/475°C
Reflector Interface	139°C/145°C
Shield Interface (All Matls.)	110°C/165°C

- First Wall/Blanket

Blanket Canister Length	1 m
Blanket Canister Surface Area (Front/Back)	$0.33m^2/0.39$ m^2
Total Thickness (incl. Manifolds and Gaps)	0.72m
Manifold Thickness	0.1m
Coolant Total Pressure Drop	0.1 MPa
Coolant Pumping Power	18 MW
Coolant Velocity (max/ave)	FW 62/47 m/s canister 31/20 m/s
Coolant Flow Rate	1.1×10^6 Kg/h

- Shield

Total Thickness (incl. Manifolds and Gaps)	0.4m
Manifold Thickness	0.0066m
Coolant Total Pressure Drop	0.43 MPa
Coolant Pumping Power	1.8 MW
Coolant Velocity	81 m/s
Coolant Flow Rate	$8.6x10^4$ Kg/h

- Heat Transport System

	He Side	Water Side
Maximum Pressure	1.5 MPa	0.1 MPa
Inlet/Outlet Temperature	50°C/300°C	20°C/40°C
Tritium Barriers	Oxide layer probably	

- Power Variation Capability

Minimum/Maximum Change from the Average Neutron Wall Loading %	50%/50%

1E. Tritium Removal

- Method	Purge Flow

- Average Blanket Tritium Inventory

Diffusion	0.3 g
Solubility	1.2 g
Adsorption	12.2 g
Total	13.7 g

- Purge Gas Parameters

Gas Constituents (inlet, outlet)	Helium + 1% H_2
Inlet/outlet Temperature	About 400°C
Pressure	1.4 MPa
Mass Flow Rate	590 Kg/h

1F. LOCA Conditions

- Integrated Total After Heat at:

1H After Shutdown	0.1 MW
1D After Shutdown	0.37 MW
1W After Shutdown	0.34 MW

- Possible Accommodation Methods include Radiation to Cooled inboard, Free Convection, Conduction, Flowing Purge

- Maximum Material Temperature for LOCA Conditions, C

Structure	< 577°C
Breeder	< 727°C

1G. Mass of Materials

Breeder	2.8×10^4 kg
Multiplier	8.8×10^4 kg
Structure	1.4×10^5 kg
Shield	3.4×10^5 kg

ACKNOWLEDGEMENT

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