Analysis of thermal-hydraulic transients of the water-cooled eutectic LiPb blanket of NET

H.Th.Klippel
Netherlands Energy Research Foundation ECN, Petten (NH), Netherlands

ABSTRACT

For the liquid LiPb blanket concept of NET the response of the first wall and the water cooled blanket to coolant tube blockage, loss of pumping power and coolant tube break events has been evaluated. Thermal-hydraulic calculations were carried out with the code RELAP followed by 2-dimensional transient non-linear heat transfer calculations with the code MARC. The main conclusion is that the concept of two parallel coolant loops provides a good safety measure since a flow coast-down in one coolant loop while the plasma is still burning does not lead to any significant temperature increase of the structures. Also passive shut-down cooling by natural convection flow or thermal radiation is sufficiently adequate to remove the decay heat in first wall and blanket.

1 INTRODUCTION

For the liquid LiPb breeder of NET a water-cooled first wall and blanket design has been developed (NET 1985), with the following main features:

a) breeder modules in the form of 6m long poloidal pressure tubes with two parallel helical coolant tubes passing through slowly circulating liquid LiPb, see fig. 1;

b) separate first wall panels with poloidal coolant tubes contained in a secondary first wall box structure with either liquid LiPb or solid Pb as thermal bound and buffer for leaks, or welded directly to the stainless steel structure, see fig. 2. Also a wall with protective tiles of graphite is being considered;

c) first wall and blanket are provided with two independent coolant circuits.

An important aspect for the evaluation of these concepts is the analysis of conceivable coolant accidents. For the different first wall proposals and the outboard breeder blanket module the following transients have been analysed under the assumption that the plasma shut-down will occur after a certain time delay:

- a single coolant tube blockage;
- a loss of coolant flow due to pump failure in one and both of the main circuits;
- a total loss of coolant in one loop due to rupture of the main manifold at the segment inlet in one of the two parallel main circuits.

The calculations have been carried out with the thermal-hydraulic analysis code RELAP (1984) followed by transient non-linear heat transfer calculations with the finite element code MARC (1983). In RELAP the phase change of the water coolant is properly handled. Single phase and two-phase flow equations and the best correlations for nucleate boiling, film boiling and dryout are applied. Thermal-hydraulic characteristics of the water coolant and the resulting temperature histories in the first wall and the breeder module during the transients have been evaluated and will be summarized in this paper. Dynamic calculations concerning a tube break within the breeder modules have been presented elsewhere (Klippel 1986).

2 THERMAL-HYDRAULIC TRANSIENTS OF THE PRIMARY CIRCUIT

2.1 Nodalization scheme of the coolant circuit

Two independent loops are foreseen in the coolant circuit of NET (1985) to strengthen the safety of the whole concept. Both the inlet and the out let headers of the blanket segments are located at the top. The pressurizer is connected to the hot leg of the coolant loop and via a heat exchanger the heat is transferred to a secondary water loop. Part of the nodalization scheme as used by the RELAP code is given in fig. 3 (only one loop is given, the other loop is nodalized identically). The power producing components are connected to both coolant loops. Between the inlet and outlet plenums of the blanket five parallel power producing components have been modelled, the one as given in fig. 3 representing the whole blanket with a power production of 600 MW, one representing a module from the first row in the outboard blanket (producing 1320 kW) and the others representing the first wall types of fig. 2, marked by FW1, FW2 and FW3. The main data of these components as used in the calculations were taken from Dänner et al. (1986).
By lack of detailed information on the other loop components, e.g. pumps, main coolant lines, heat exchanger and pressurizer, data from existing components in light water reactors have been assumed. A heat exchanger of the once-through type with a total height of the pipe bundle of 10 m has been assumed. In the reference case the mid-plane of the heat exchanger lies 11 m above the midplane of the blanket.
The main coolant lines have a diameter of 0.30 m and a total length of 60 m. The total volume of primary coolant, including pressurizer and storage tank is about 80 m^3 per loop.
At the inlet of the blanket the water temperature is 265°C and the pressure 115 bar. At nominal flow condition the outlet temperature of the blanket modules will be 305°C and of the first wall ca 280°C. The subcooling margin for the first wall is much larger than that for the blanket (saturation temperature is $T_{sat} \approx 315$°C). In vertical direction a power distribution according to the poloidal power distribution calculated by the FURNACE code (Dänner et al. 1986) has been applied. The result is that the maximum temperature in the blanket and FW will occur just above midplane. This location has received special attention in the further calculation and has been chosen for the calculations given in section 2.3. After shutdown the initial decay heat is taken as 0.75% of nominal power production in the blanket module and as 1.5% of nominal power production in the first wall. These values were derived from time dependent decay heat densities given by Ponti (1985).

2.2 Summary of thermal-hydraulic calculations

a) **Coolant flow blockage of a single coolant tube** in one blanket module or in one first wall panel.
The time constant to reach a new steady state where the heat removal is taken over by the intact coolant tubes is about 50 s, see fig. 4.
In the intact coolant tube of the blanket module some nucleate boiling of the water starts after about 100 s. This nucleate boiling enhances the heat transfer coefficient to more than twice its nominal value. The critical heat flux above which coolant dryout might occur, is 5 to 7 times the actual maximum heat flux. The average blanket temperature rise at midplane is about 80°C. The coolant temperature rise at midplane is 25°C. In the first wall panel nucleate boiling occurs only in the upper sections of the panel due to the larger subcooling margin. The average temperature rise of the first wall structure at midplane is max. 90°C.

b) **Loss of pumping power in one of the coolant loops while the fusion power is going on.** The pressure in both loops is assumed to be maintained at constant level. The mass flow in the interrupted loop drops from nominal value (1500 kg/s) to natural circulation (250 kg/s) in about 50 s. The time constant of the flow coast-down is ca 10 s. The heat transfer coefficient in the breeder module initially drops from 34000 W/m^2K to 5000 W/m^2K, but due to the coolant temperature rise nucleate boiling starts already after about 15 s which significantly enhances the heat transfer coefficient, see fig. 5. Therefore the total heat fluxes to both loops do not change very much. After reaching natural convection flow in the interrupted loop ($t \approx 100$ s) the total heat transfer to that loop has been decreased by only 3% while the heat transfer to the intact loop has been increased by the same amount. The average temperature in the blanket and FW panels increases only by about 20°C.

c) **Flow coastdown in both loops with delayed plasma shutdown.** In both loops the mass flow reaches natural circulation condition after 50 s, and nucleate boiling at midplane starts within 15 s, see fig. 6. On the longer time scale also in the lower part of the blanket and first wall near-saturation temperature will be reached. After an initial drop the heat transfer coefficient will become high. At the exit of the blanket the steam quality will be 65%. The critical heat flux value still remains a factor 5-7 higher than the actual maximum heat flux to the coolant. Obviously a loss of pumping power in both loops does not require very acute measures. The coolant reaches near saturation temperature, the heat transfer coefficients to the coolant remain high and the average blanket temperature increases by about 30°C within the first 30 s and thereafter, due to a 0.3% lower heat flux to the coolant, the blanket temperature increases by about 0.02°C/s.
Assuming a controlled plasma shutdown 150 s after the onset of the pump trip the maximum average temperature rise in blanket and first wall is 40°C. The afterheat removal by natural convection cooling is quite satisfactory. The mass flow decreases gradually to 210 kg/s for both loops together (at $t = 1000$ s), which is 7% of the nominal flow. The heat transfer coefficient in the coolant tubes drops to 3000 W/m^2K.

d) **Total loss of coolant in one loop.**
This event which might occur after e.g. a large break in the inlet main coolant line is most severe when plasma power production is not stopped. After shutdown the intact loop guarantees heat removal even in natural circulation condition. Before shutdown all the nuclear power produced is transfered to the intact loop. Assuming no shutdown and no action on the secondary flow to increase the heat transfer through the heat exchan-

gers, the coolant and blanket temperatures rise considerable, see fig. 7. The coolant inlet temperature rises up to near saturation within about 150 s. Then the mass flow decreases drastically due to the considerable steam formation. Then also departure from nucleate boiling (DNB) and transition to film boiling might occur in the upper part of the blanket. Consequently, the heat transfer coefficient drops below 2000 W/m^2K in the upper half of the blanket. Before this point has been reached the reactor must be scrammed.

2.3 Two dimensional temperature development in first wall and blanket

Using the code MARC (1983) x-y plane-symmetric calculations for the first wall and r-z rotational-symmetric calculations for the blanket module have been performed to evaluate the local temperature behaviour in the structures after coolant disturbances as described in 2.2.
The power densities used in the first wall and blanket are representative for the midplane in the outer blanket and were taken from FURNACE calculations (Dänner et al. 1986).
The neutron wall loading at midplane is 1.2 MW/m^2, resulting in a maximum volumetric nuclear heat production in the stainless steel of the first wall of 18 MW/m^3. For the radiation heat flux from the plasma on the wall surface a value of 0.2 MW/m^2 has been taken.
For the different coolant accidents as described in 2.2. the temperature histories within the structures of first wall and blanket have been calculated taking the time dependent heat transfer coefficients and coolant temperatures from the RELAP calculations as input for the MARC calculations. The MARC calculations specifically show the spatial temperature development at midplane. In table 1 a summary is given of the maximum and minimum structural temperatures reached in FW and blanket as far as the transients have been followed in time.

4 CONCLUSIONS

The present calculations have shown that the thermal response of the first wall and the blanket of NET to the most conceivable coolant disturbances are relatively slow so that safety systems can be actuated in due time (up to 150 s) before damage to the first wall and blanket occurs.

The basic concept of two parallel coolant loops provides a good safety measure, since the maximum local temperature rise in the first wall and the blanket in case of loss of the coolant flow in one of the coolant loops is limited to 40°C while the plasma is burning.

Passive shutdown cooling by natural convection is sufficiently adequate to keep the structural temperatures low.

ACKNOWLEDGEMENT

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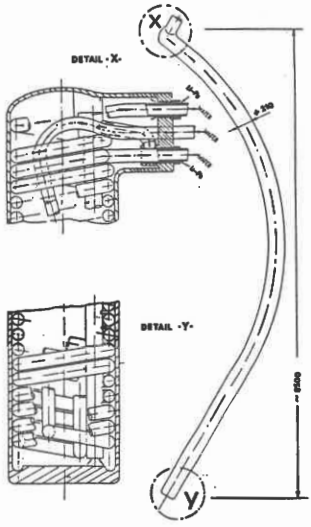

Fig.1. Liquid LiPb blanket module (vertical cross-section)

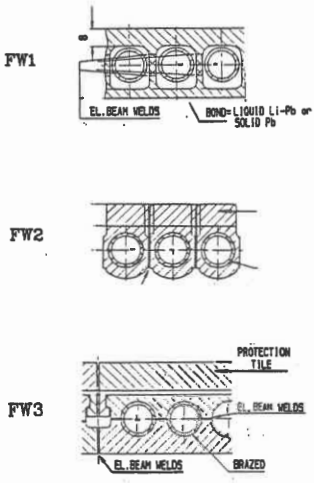

Fig.2. First wall proposals NET (horizontal cross-section)

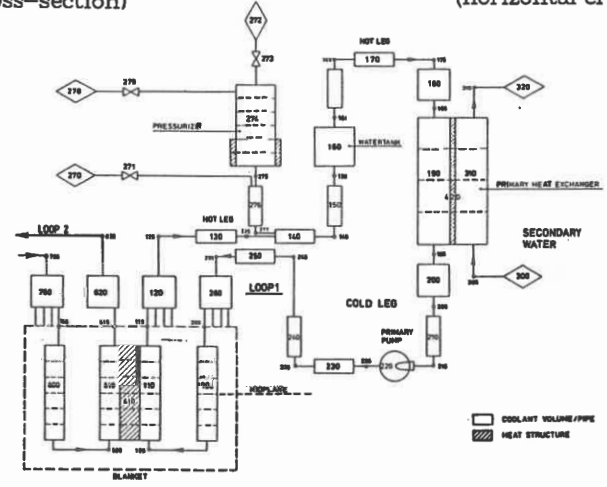

Fig.3. Nodalization scheme NET coolant circuit
(LOOP 2 IS IDENTICAL TO LOOP 1)

Fig.4. Blanket response to single coolant tube blockage

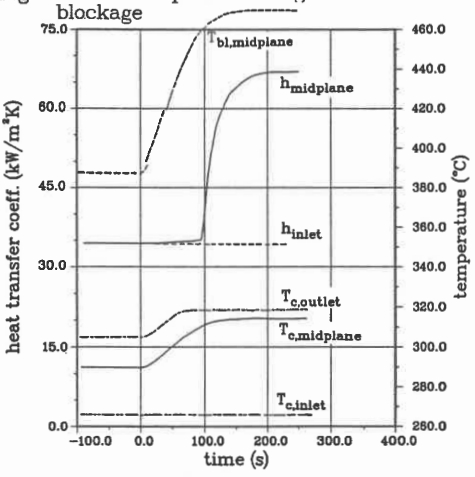

Fig.5. Blanket response to loss of pumping power in loop 2

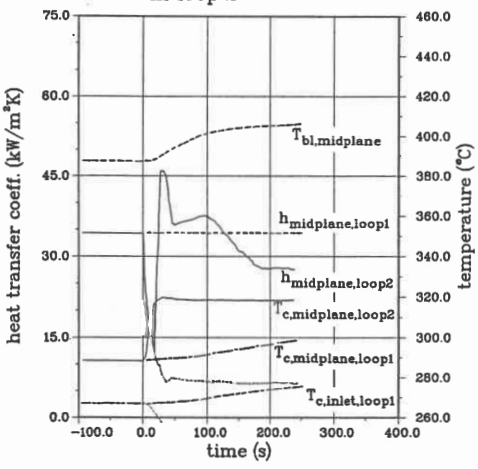

Fig.6. Blanket response to loss of all pump power plasmascram at 150 s

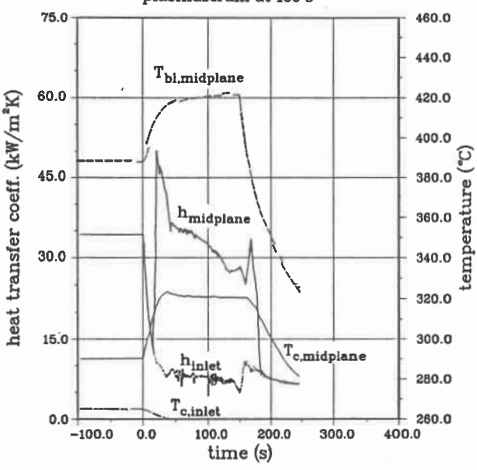

Fig.7. Blanket response to loss of cooling in loop 2

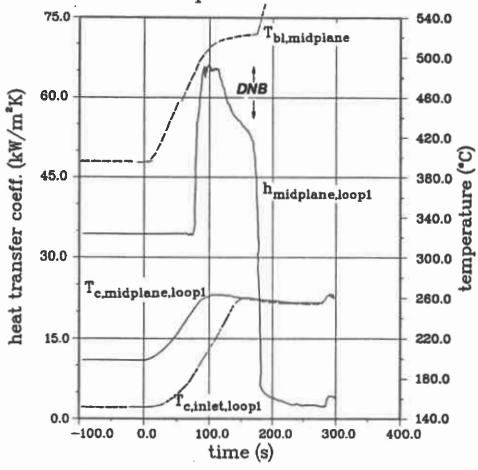

Table 1. Maximum/minimum structural temperatures (°C) in coolant transients.

	Blanket	FW1	FW2	FW3
continuing power				
steady state (reference)	389/340	508/315	488/288	517/290
single coolant tube blockage (t>>200 s)	542/387	743/387	670/367	703/370
flow decay in one loop (t>>200 s)	420/360	535/362	520/320	548/328
flow decay in two loops (t=150 s)	410/370	530/350	514/324	548/318
total blockage one loop (t=150 s)	550/390	760/395	680/372	720/376
shutdown				
natural convection in two loops ($T_{coolant}$ = 200°C)	202/201	208/206	210/207	209/206