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INVESTIGATION OF THE DIFFERENT SCENARIOS
OCCURING IN A PWR IN CASE OF A TMLB ACCIDENT

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INVESTIGATION OF THE DIFFERENT SCENARIOS OCCURING IN A PWR IN CASE OF A TMLB ACCIDCNT

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1 - Introduction

Severe accidents in light water reactors fall into one of two main categories, depending on whether or not core meltdown is accompanied by a pressure buildup in the primary system. The way in which the accident develops is, in fact, largely conditionned by this pressure aspect : temperature distribution in the core and primary system resulting from natural convection gas streams; fuel clad failure mode, etc... One major effect of pressure buildup on the accident scenario is primary system failure under the combined actions of pressure and temperature. The purpose of the present paper is to present, after a detailed thermalhydraulic study, an analysis of the timing and location of the system failures in case of a TMLB accident on CPY french type reactor.

It is necessary to remember first that, for the French PWR's operated by EDF, there is a procedure, known as H2, whereby plant operators are required to depressurize the primary system before complete steam generator dryout. A core meltdown with pressure buildup should consequently not occur in French PWR's or at least, its probability of occurence is extremely low and it could doubtless be classified in the residual risk category.

It has nonetheless been decided to analyze this type of scenario, on the one hand to have comparisons, because it has been extensively investigated on an international scale, on the other hand, because this type of scenario is included in those which will be analyzed in the framework of the international program PHEBUS PF.

In order to make this analysis more exhaustive, it was also decided to include, a reactor coolant pump seal leak. Such a leak is particularly improbable at the beginning of the accident because French PWR plants are equipped with an emergency system, for pump seal feeding.

Several studies have been made, with no leak at the pumps (pump seal water injection operating) or with leaks (pump seal water injection failure). These leaks are not easy to quantify and this kind of phenomenon is certainly not reproduceable. Therefore a parametric study has been made covering a reasonable range. Based on best current estimation of leaks the range 1.4 kg/s to 10.5 kg/s has been explored.

These results are first analysed regarding the thermalydraulic behavior in function of pump seal leak rates. The emphasis is put on natural circulation and heat transfer mechanism, with the consequences on structure temperature. Mechanical evaluation is then analysed on the whole reactor primary circuit including steam generators, in view of localizing the weakest point.

2 - Hypotheses of thermalhydraulics calculations

2 - 1 Used codes :

First RELAP4 MOD6 then TRAC PF1/MOD0 (1) and finally for this paper TRAC PF1 version 14.3 (5) have been used to study the TMLB sequence. Some comparison have been made with the STCP code (4) and results (3). CATHARE 1 which does not include steam generator walls has not been used. In the future CATHARE 2, which provides a better representation of the steam generator, will be utilized.

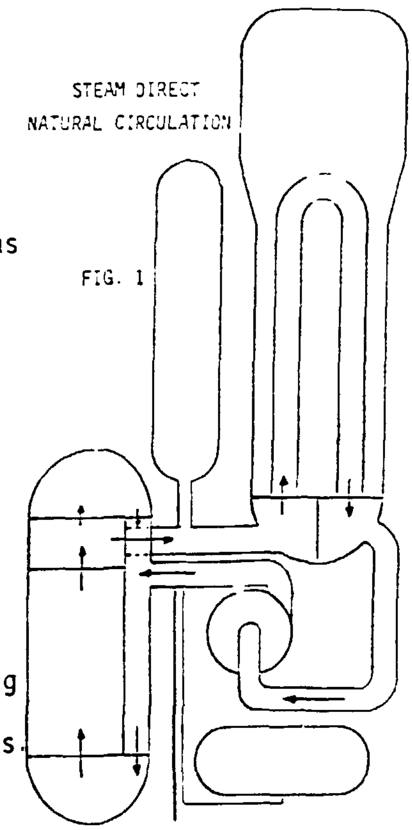

FIG. 1

2 - 2 Nodalization :

In order to save computer time a one dimensionnal, one loop scheme was choosen. The pump leaks are perfectly symmetric, the only assymetry is caused by the pressurizer. The basic noding for RELAP or TRAC is given figure 1. Following a 3d vessel calculation with TRAC on TMLB accident (2), showing a strong mixing by natural circulation between core and core by-pass, the one-d core include the by-pass.

2 - 3 *Initial and boundary conditions :*

They are identical for all the calculations. The power is 102% of nominal power, total heat losses represents 2 MW, and residual power is given by ANS+20% in order to make easier comparison with other studies. At time zero occurs the loss of on site and off site electrical power which induces automatic scram, loss of normal and auxiliary feedwater, loss of primary pumps, loss of HPIS. In addition the failure of the auxiliary water turbo pump is supposed and also the pump seal safety circuit may not be available. The SEBIM pressurizer safety valve caracteristics are used. No operator actions are taken into account. The importance of the leaks at the primary pump seals has been chosen as parameter. The values are no leak, 1.38 - 2.76 - 4.16 - 6.94 - 10.55 kg/s per pump for fluid initial conditions. Thermalhydraulic calculations are done without taking into account any rupture on the primary or secondary circuit or clad melt.

3 - Thermalhydraulic analysis of the pump leak influence

3 - 1 General sequence :

After the accident initiation the main phenomena which occur are : the steam generator dry out, the pressurizer safety relief valve opening, the beginning of core uncovery with the intermediate leg voiding, 2 φ natural circulation, the core complete dry-out, steam natural circulation with structures as heat sink, the lower plenum and pressurizer voiding and in certain case the accumulator injection.

3 - 2 Pressure evolution :

Following the complete loss of electrical power the reactor scram occurs which induce a pressure decrease of the primary side pressure (fig.2) during a small interval of time.

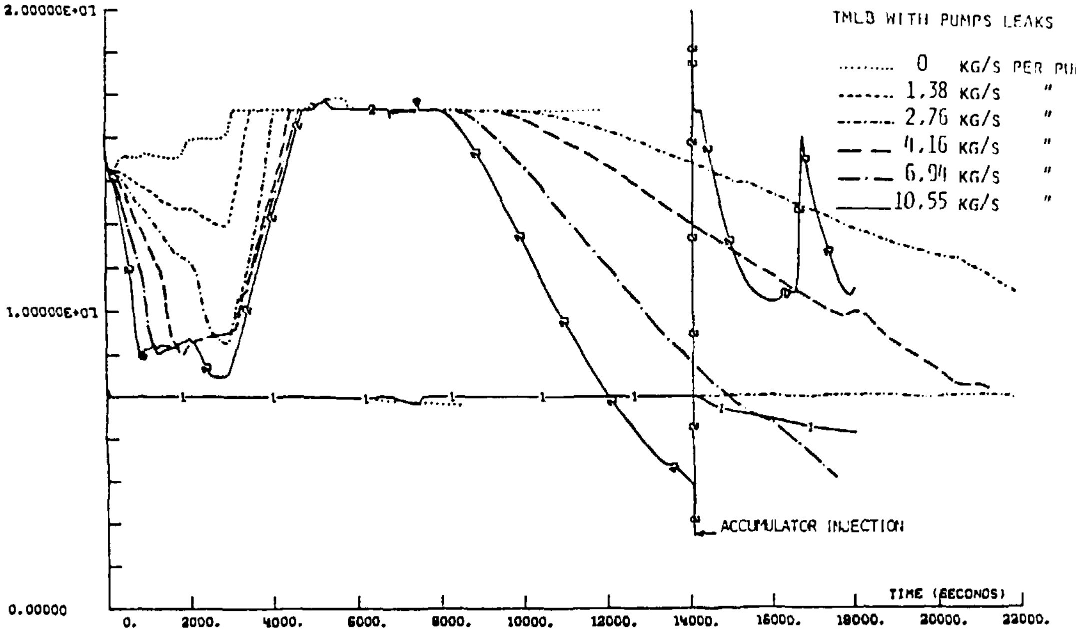

FIG. 2 1 PRESSURE (PA) : SG STEAM DOME
2 PRESSURE (PA) : PRESSURIZER

On the contrary due to the loss of the main feedwater and the steam generator (SG) secondary side isolation, the secondary pressure increases and reaches the safety valve opening pressure value. The SG pressure remains almost constant during all the transient except in case of accumulator injection. On the primary side there is a competition between fluid expansion and leakage. When there is no leak in the primary side the pressure increase. For the smallest leak in this study (1.38 kg/s per pump) the pressure decreases. After the SG dryout the pressure increases until the safety relief valve opening pressure is reached. After the primary pressure remains almost constant until the complete primary circuit voiding. This is followed by a decrease of pressure in case of leaks.

For the two largest leaks (6.94 - 10.55 kg/s per pump) the accumulator set point is reached. The accumulator injection phase has been calculated for the 10.55 kg/s leak. This injection causes a strong decrease of pressure due to condensation, followed by a large pressure pic, when water goes into the core. The pressure reaches again the safety relief valve opening pressure. The injected water is pushed out of the core and there is again a decrease of pressure, followed by a new increase and so on.

3 - 3 Intermediate leg voiding :

In old RELAP4 MOD6 calculations there was no dependence on the leak flow rate, in the calculations results, concerning the voiding of the intermediate leg. With the latest codes two different behaviors are observed. For the smallest leak or with no leak the complete voiding of the intermediate leg is not obtained (fig.3) before stopping the calculation (too high steam temperature).

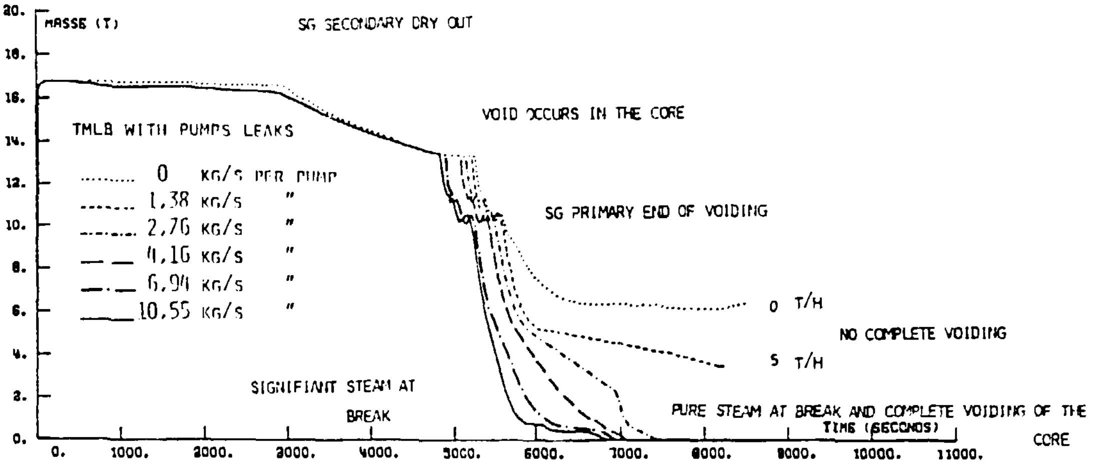

FIG. 3 PUMP AND INTERMEDIATE LEG LIQUID MASS

So the natural circulation of steam is largely blocked and there is a strong increase of all reactor upper part fluid and structures temperatures. This situation is consistent with hypothesis made in STCP (4) where the primary circuit is considered as one node with no possibility of natural circulation.

In the other cases the intermediate leg becomes empty so steam natural circulation can be established with structures as heat sink. In these cases a more detailed description is necessary to evaluate properly the temperatures in the different parts of the circuit.

3 - 4 Pressurizer behavior :

After the short transient following the scram, the pressurizer water mass decreases, the rate of decrease increasing with leak flow rate. After the SG secondary side dryout (at about 3 000s.), water feeds progressively the pressurizer (fig.4). For smallest pump leaks water goes to the safety valves.

After the SG primary side voiding, water mass decreases strongly until the hot legs voiding, stay almost constant at the begining of core voiding then decrease more and more strongly as the core voids.

Then different behaviors are observed.

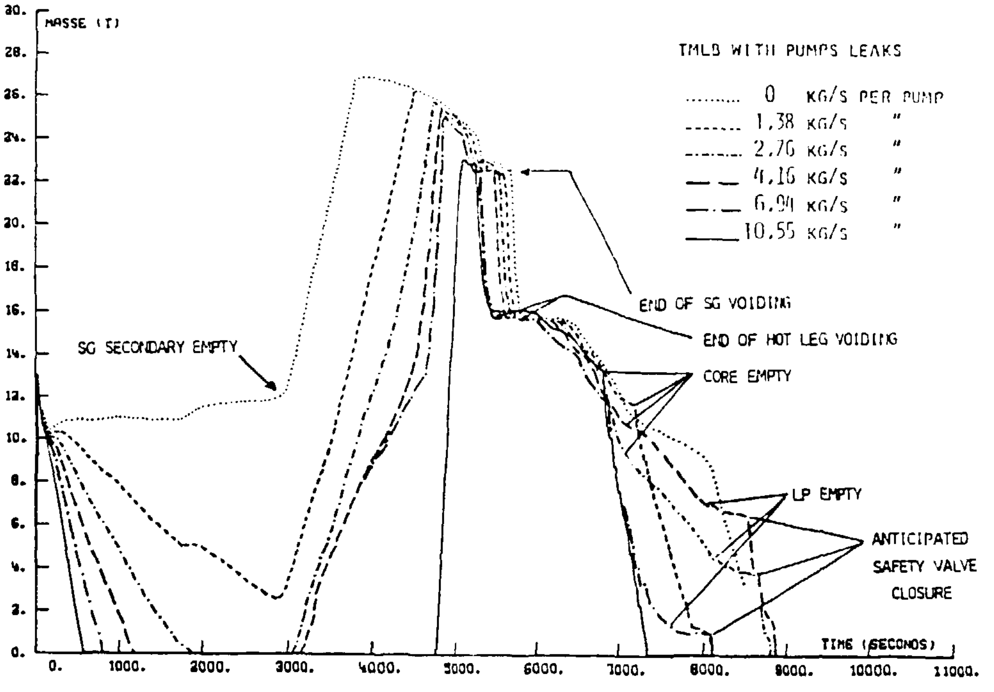

FIG. 4 PRESSURIZER LIQUID MASS

For the case without leak and for the smallest leak, it has to be mentioned that the vessel lower plenum stays practically full of water due to the absence of steam natural circulation. In this case the pressurizer will be empty before the lower plenum. For the largest leak, the pressurizer is also empty before the lower plenum due to the large leak draining. On the contrary for other leaks the pressurizer voids after the lower plenum.

For the 2.76 and 4.16 kg/s leaks per pump it happens that the pressurizer safety valves close shortly, a few hundred seconds after lower plenum voiding, when there is respectively 3.8 t and 6.6 t of water in the pressurizer. That induce a strong voiding due to the absence of counter current flow limitation at this time. The safety valves open again and close several times after, but the pressurizer voids. Saturated water flashes in the surge line, hot legs, and becomes saturated steam into upper plenum. That changes completely the driving forces for natural circulation and a completely reverse flow path occurs (fig.5) being the opposite of the preceeding one (fig.1)

STEAM REVERSE
NATURAL CIRCULATION

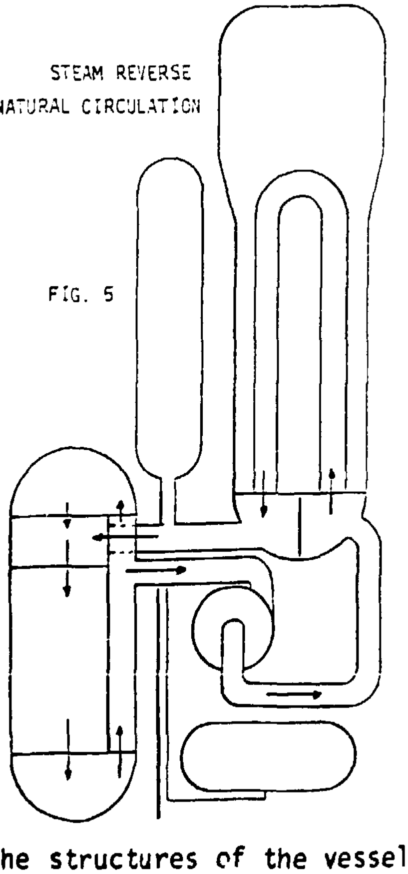

FIG. 5

This is again a stable situation because the structures of the vessel and cold leg are a better heat sink than the upper part of the circuit due to their lower temperature.

3 - 4 Fluid and clad temperatures :

The fluid temperatures at the upper part of the core show (fig.6) the above described phenomena. The lowest temperatures are obtained for 2.76 and 4.16 kg/s leaks.

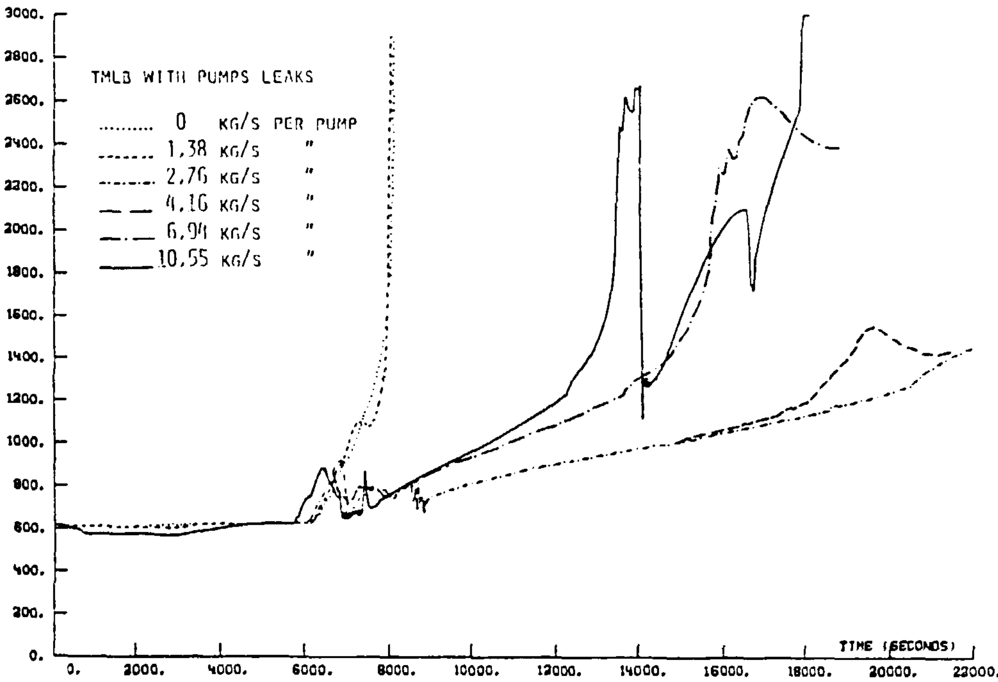

FIG. 6 STEAM TEMPERATURE AT THE TOP OF THE CORE

The figure 6 gives a clear classification of the different thermalhydraulic heat transport modes which are observed in this study :

1 - no leak or very small leak (1.38 kg/s) no steam natural circulation
2 - intermediate range of leak (2.76 - 4.16 kg/s) reverse steam natural circulation.
3 - larger leaks (6.94 kg/s) direct steam natural circulation.

Cladding melts first for mode 1 (at 8 000s.) then for mode 3 (6.94 kg/s 15 700s. - 10.55 kg/s 13 400s.) and finally for mode 2 (2.76 kg/s 20 400s. - 4.16 kg/s 17 800s.) For each mode larger is the leak, shorter is the time for melting.

4 - Mechanical behavior of the structures on primary circuit and S.G.

4 - 1 Hypotheses :

The components are submitted to high pressure and high temperature with some of them around 1200 K. In our study the stress is caused by pressure and temperature gradient. According to design codes primary stress is due to pressure and thermal gradient is no longer considered. The structure will adapt and stress will relax. Depending on temperature two kinds of limits are applied :

- Case of the temperature $\leqslant$ 700 K, unsignificant creep, application of the RCC - M codes

$Pm \leqslant 0.7$ Su with Su ultimate strength, for the vessel, the nozzle, the steam generator

$Pm \leqslant \min \begin{cases} 2.4 \text{ Sm with Sm the allowable stress} \\ 0.7 \text{ Su} \end{cases}$

for the tubes of the S.G. and the pipes.

- Case of the temperature $\geqslant$ 700 K, significant creeps, application of the RCC - MR code, calculation of the rupture usage factor for all components:

$$W\,(1.35\,\Omega\,Pm) \leqslant 1$$
$$W\,(1.35\,(Pl + \phi Pb)) \leqslant 1$$

ϕ takes into account the relaxation due to bending stress

Ω prevents elastic follow-up due to membrane stress.

In the case where the creep caracteristics are unknown the RCC-M rules has been applied with extrapolated ultimate strength and yield stress. When the allowable limits are not respected, the primary equivalent membrane stress has been compared to the ultimate stress.

4 - 2 Results (6) (7) (8)

Hot leg behavior (fig.7)

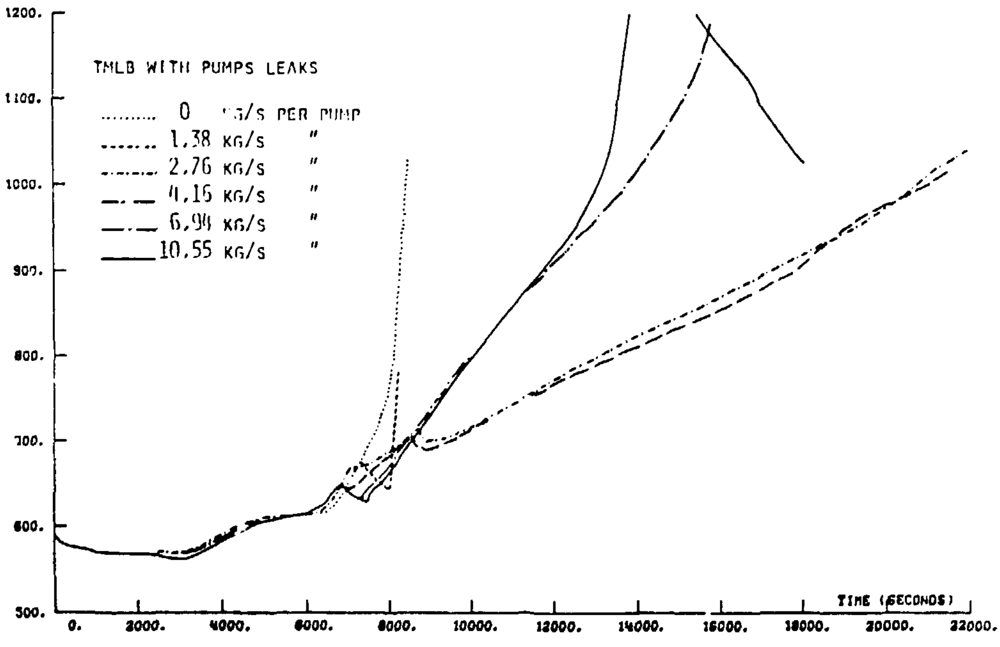

FIG. 7 AVERAGE T (K) : HOT LEG

Appying the RCC-MR rule gives a time to rupture of only a few minutes for no leaks and leaks of 1.38 kg/s per pump. The rupture will oppear around 8 700 s at the heterogeneous weld between the nozzle in 18 MWDS (AS08 CL3) and the pipe in Z2CND 17.12 (316).

A risk of rupture exists also for the largest leaks after the accumulator injection when the pressure spike occurs that is 14 000 s for 10.55 kg/s and probably 17 800 s for 6.94 kg/s.

Steam generator conical junction behavior :

A correction taking into account radiative transfer has been made to calculated values from TRAC (fig. 8).

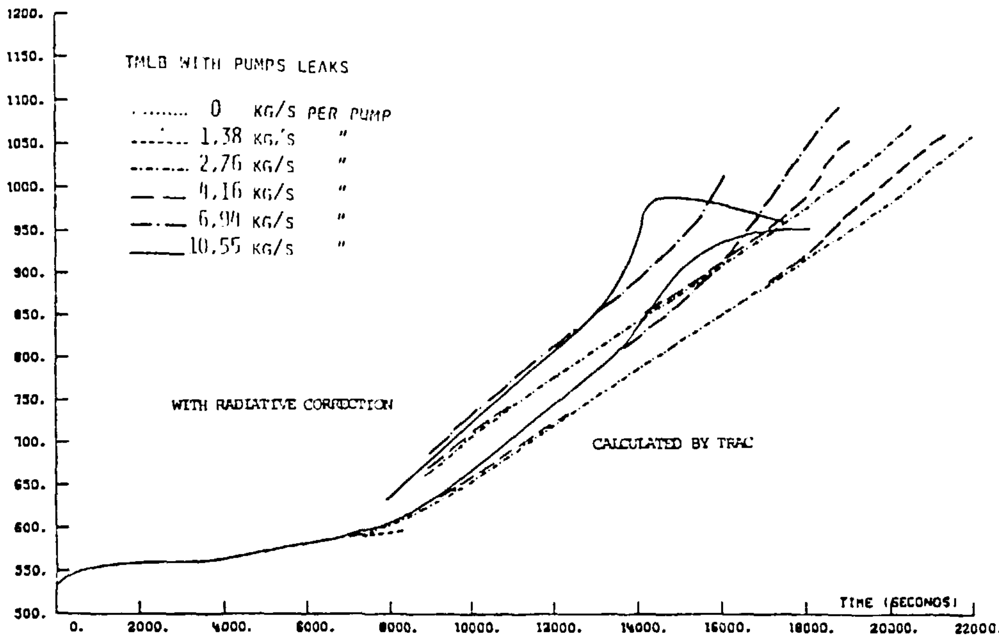

FIG. 8 AVERAGE T (K) ; S.G. CONICAL JUNCTION

The risk of rupture exists for leaks from 2.76 kg/s to 10.55 kg/s around 975 K.

Cold leg behavior :

For the two intermediate leaks 2.76 kg/s and 4.16 kg/s due to reverse steam natural circulation the heterogerons weld between the nozzle and the pipe is the weak point on the primary circuit.

The following table gives the most important results for this study :

leak kg/s	component	time s	T k	P MPa	Stress MPa	Su Mpa	time to rupture h
0	hot pipe	8700	1175	17	114		0.1
1.38	"	8700					
2.76	conical junction	18500	975	7.2	160	160	
	cold pipe	19300	1175	13	87		
4.16	conical junction	18100	975	7.2	160	160	
	cold pipe	18000					
6.94	conical junction	15800	975	7.2			
	hot pipe	17800	1175				
10.55	conical junction	14500	975	7.2			
	hot pipe	14000	1175				

5 - Conclusions :

- Second generation thermalhydraulics codes give a better understanding of the complex phenomena which occur during a TMLB sequence. They are able to produce new sensitivity studies on parameters which were not judged to be important with first generations codes.

- This thermalhydraulic study has shown, with the hypotheses made for the calculations, three different modes for heat transport depending on the leak flow rate :

1 - no steam natural circulation for leaks $\leqslant$ 1.38 kg/s
2 - reverse steam natural circulation for 2.76 $\leqslant$ leaks $\leqslant$ 4.16 kg/s
3 - direct steam natural circulation for leaks $\geqslant$ 6.94 kg/s.

- The second mode is due to the closure of the pressurizer safety valve when several tons of water are still in the pressurizer. Saturated steam enters the upper plenum and the flow reverses. This mode gives a large delay in temperature increase of all the parts of the circuit.

- For each full established mode larger is the break earlier is the clad melt.

- These results depend strongly on pressurizer and intermediate leg voiding as well as on CCFL, pump 2φ model, interfacial momentum transfer. Limitations are due also to steam properties and radiative transfer.

- For the first mode codes with a one node description of the primary circuit, like STCP, can be acceptable from the beginning of the accident, for other modes they should be used only for the end of the transient.

- The coupling between thermalydraulic results and mechanical analysis has shown that for french CP1 reactors the weak points in the circuit are the heterogeneous weld between nozzle and pipe for hot or cold leg and the conical junction for the wall of the steam generator.

- Ruptures are expected only on the hot leg for leaks $\leqslant$ 1.38 kg/s.

- Ruptures are expected both on SG and legs for leaks $\geqslant$ 2.76 kg/s. Depending of the scenario SG tubes can be damaged.

- Ruptures are expected in the legs not long after clad melt but before core melt or vessel damage. This leads to a follow-on accident at low pressure in the primary circuit.

- So it appears that for such a TMLB sequence, without any operator actions and with all the possible systeme failures, the core melting will occur definitely at low pressure for French CPY type reactors.

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