LOSS OF FLOW INCIDENT – SIMULATION AND MEASUREMENTS IN THE MPR

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ABSTRACT

As part of the Probabilistic Safety Analysis of the Multi Purpose Reactor, MPR, the list of Postulated Initiating Events was analyzed and one of these PIEs corresponds to the Loss of Coolant Flow. It is well known that during the operation life of a research reactor a LOFA could eventually occur and, once this event takes place, in time detection and automatic actions, thanks to the engineering safety features of the system, will mitigate the incident evolution. The postulated event corresponds to a loss of flow due to a total loss of power supply. The goal of the present work is to provide a general description and the engineering safety features of the MPR, as well as describe the sequence of scenarios during a LOFA. Temporal evolution of main parameters is presented, also. During Stage A of the Commissioning Program measurements of the core cooling system pump coast-down were performed in order to validate previous simulation results, as well as, flap valves opening time. In this way it was verified that engineering safety features worked properly. On Stage B of the Commissioning Program the upward natural convection flow was verified and results comparison against analytical calculation, showed that the reactor core was cooled within the adopted design goals.

MPR DESCRIPTION

The Egyptian second research reactor (ETRR-2) is a 22 MW_{th} open pool Multi Purpose Reactor (MPR) [1]. It is located at Inshas site of the Atomic Energy Authority. It aims for research in neutron physics, material science, nuclear fuel research and development, radioisotope production, neutron radiography, activation analysis, Silicon Doping, boron neutron capture therapy and training in nuclear engineering and reactor operation. The reactor has several beam tubes, hot cells, high-pressure test loops and other research equipment. Figure 1 shows ETRR-2 general view.

Fig. 1 ETRR-2 General View

The core consists of a 5 x 6 grid surrounded by a Zircaloy chimney and it is placed 10 meters below the pool surface. The fuel elements are low enriched Uranium type with aluminum cladding (19.75 % Uranium 235). Each fuel element has 19 fuel plates and they are cooled and moderated by light

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water. Six Ag-In-Cd alloy control plates are part of the First Shutdown System and four of these six plates control, also, the core reactivity.

Beryllium reflectors are positioned around the core and outside the reactor chimney. The chimney structure not only provides independence between the core and the main pool but also, a double wall of Zircaloy in the lower part, houses the Second Shutdown System once it is triggered.

Figure 2 shows the core and rods configuration.

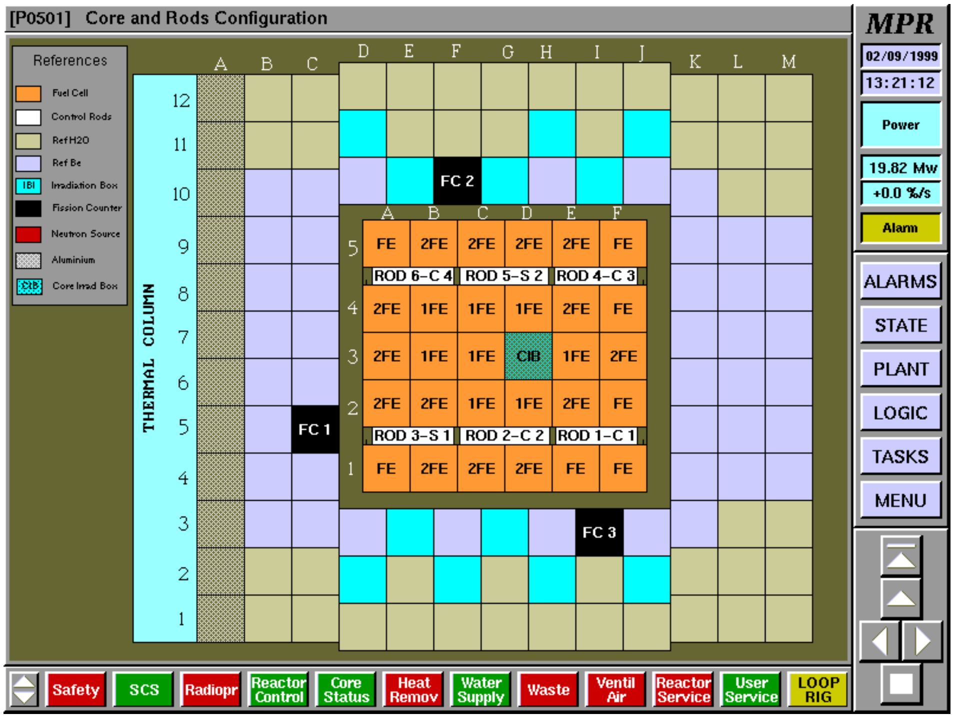

Fig. 2 Core and Rods Configuration

CORE THERMALHYDRAULICS

Primary core cooling system

It removes the fission power generated in the core by an upward forced flow of demineralized water. Outside the pool, the primary cooling piping splits into two loops. Each loop contains one heat exchanger and two parallel pumps (one in stand-by) and has capacity to absorb 50 % of the reactor power.

It is worth mentioning that there are three different Operational Regimes (OR):

OR-0: coolant in the natural convection regime and very low power (400 KW).
OR-I: one loop under operation and 50% of full power
OR-II: two loops operation and full power.

Figure 3 shows the core cooling system while main data is summarised in Table 1.

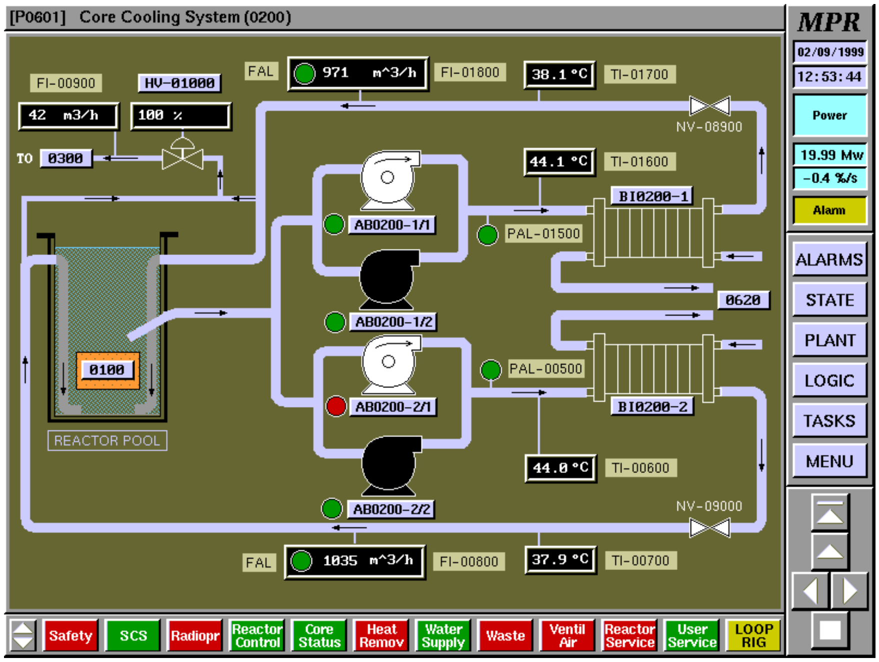

Fig. 3 Core Cooling System

Table 1: Main Data

Initial power	22 Mwatts
Nominal flow rate	1900 m^3/h
Number of fuel elements	29
Initial inlet core temperature	40 °C
Pool water temperature	40 °C
Power shape	Cosine
Radial power peaking factor	1.92
Total power peaking factor	3

Thermohydraulic Engineering Safety Features

As previously mentioned core coolant flows in the upward direction and driven by pumps. Once that pumps stop, natural convection must establish to avoid a dangerous wall temperature increase. For this reason special devices or ***flap valves*** are provided, not only to perform the siphon breaker-effect function but to "built" the new natural convection cooling circuit, as well.

Another engineering safety feature is defined, the ***flywheels*** for the pumps to increase the forced convection time before natural convection establish.

Besides the previous Engineering Safety Features, the ***chimney*** and the ***Chimney Water Injection System (CWIS)*** are included here to complete the list.

The structural chimney provides physical independence between the core and the reactor pool, as already mentioned so, for the case one of the irradiation beams breaks and the end of it is opened, pool water will drain through this break and the water level inside the pool will decrease while the core remains under water thanks to the chimney structure. Anyway, the water level inside the chimney will also decrease with a slower rate due to evaporation, as decay heat is present and due to the leaks through the metallic couplings in the chimney and with the core grid. The CWIS compensates this level decrease.

LOFA DESCRIPTION

After a total loss of energy supply or some malfunction in the core coolant pumps, a core flow coastdown takes place, according to the pump inertia flywheels. As a consequence the first shutdown system is triggered by the following signals, shutting down the reactor:

- Unavailability of electric power
- Low core pressure drop
- High temperature difference across the core
- Low flow of the core cooling system
- High core outlet temperature
- Opening status of the flap valves

The sequence of events is as follows:

- At **t = 0.0 s** a Loss of flow incident occurs, i.e., pump coast down begins*.*
- The First Shutdown System triggers and shuts down the reactor.
- The two flap valves open due to the low flow rate through the core
- Natural convection establishes.

Several scenarios have been analysed according to the different operational regimes in this study, however, only results for the operational regime **II** are presented, while results for the rest of the ORs are given in [1].

This analysis was performed using the RETRAN code [2] and some important features, like coast down flow and opening time of flap valves, were compared against measurements in order to validate the program models.

COMPUTATIONAL MODEL

For RETRAN simulation a simple discretization was adopted defining no more than 28 control volumes, 32 junctions and 9 heat conductors, as shown in Figure 4.

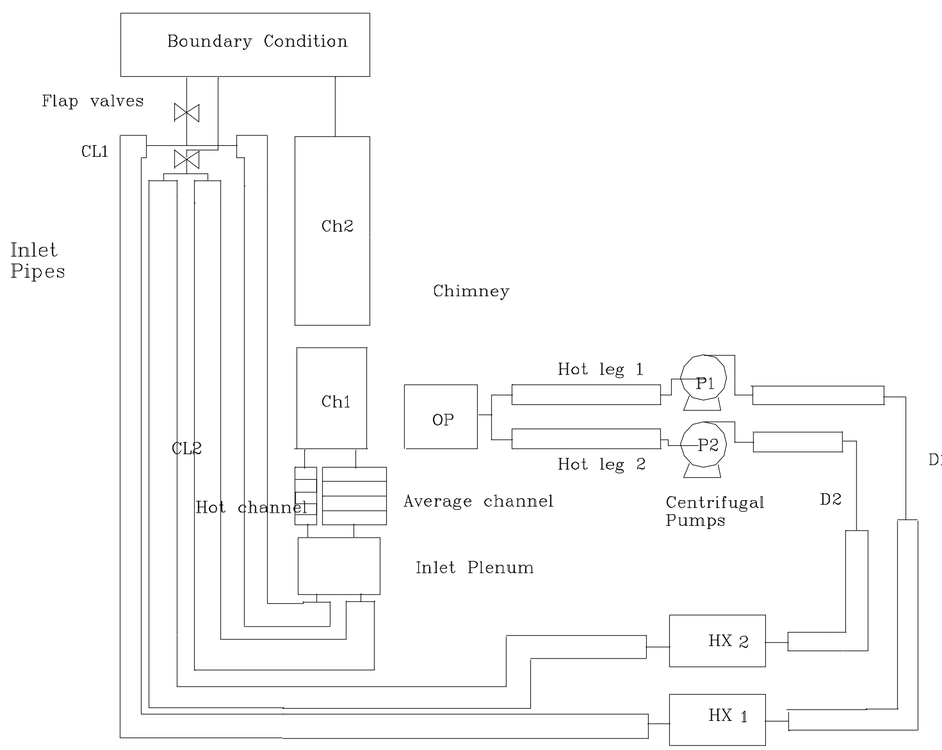

Fig. 4 Primary Loop Generic Nodalization

Some special features have been adopted to best estimate maximum temperatures. They are:

- **Core**: The average and hot channels have been simulated. Both types of channels have been divided axially taking into account the cosine power shape.

- ***Chimney closure flow:*** The downward closure flow inside the chimney was not modeled, as it is irrelevant for the flow coast-down simulation.

- ***Temperature transport delay model:*** This model presented in RETRAN is specially suited to be used where needed to follow a front such as cold water flowing down a pipe. This model has been adopted for both, the hot and cold legs of the core cooling system.

- ***Flap valves opening***: It is important to notice that the core cooling system has been simulated as a forced convection closed loop considering the core, hot legs, pumps, heat exchangers and cold legs. Once that flap valves open natural convection establishes and a "new" loop can be built with the core, the chimney (hot leg), the pool (cold sink) and the two 12"pipes entering the core (cold legs). When the flow through the 12"pipes entering the core is low enough, flap valves open in 1sec.
The flap valves were designed to open when the relation between the flow rate and the decay power give a ONBR ≥ 1.3.

In order to simplify calculations and to be more conservative the following assumptions have been adopted:

- A power cosine shape.

- Decay heat was calculated from a normalized power versus time table with the ANS curve specified in [3] adopting a safety factor of 1.2 to take into account not only fission products but actinides decay also.

- There is no heat conduction across the core structures.

- Although there are several SCRAM signals only the low core flow has been considered as conservative.

- A low flow signal (90% of nominal value) triggers the SCRAM control plates and no delay time was considered.

Using previous discretization and considering those special features and conservative assumptions a steady state was run and reached after 50 seconds before the transient simulation begins. This was done to avoid numerical instabilities due to initialization matching. Table 2 summarizes the most important steady state variables and their values.

Table 2: Main steady state parameters.

Inlet core temperature	39.5 C
Δ Core temperature	10.0 C
Hot channel outlet temperature	57.0 C
Maximum cladding temperature - Hot channel - Average channel	 95.0 C 66.7C
Maximum fuel temperature - Hot channel - Average channel	 117.8 C 76.5 C
Nominal flow rate	1867 m^3/h
Core pressure drop	0.8 bar

As presented in Table 3 these values show a good agreement with those obtained with TERMIC, [3],

Table 3: Comparison of steady state values

Parameter	RETRAN	TERMIC*	%
Coolant outlet temperature	57.0 C	60.0 C	5.2
Cladding temperature	95.0 C	88.5 C	7.3

(*) TERMIC calculations were performed without uncertainty factors.

RESULTS COMPARISON

- ***Flow coast down:*** it was simulated according to flap valves dimensions, pump and fly-wheel characteristics as well as loop characteristics, resulting in the decreasing flow shown in Figure 5.

Measurements performed during the Stage A of commissioning program are also shown and it can be seen that there is a good agreement between calculated and measured data (< 5% for the first two points and < 15%, and conservative, for the rest).

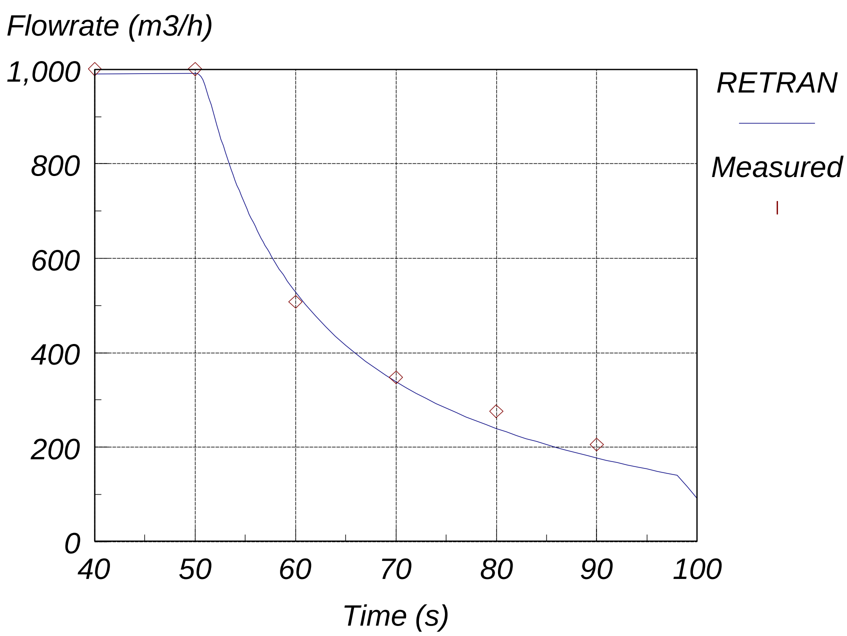

Fig. 5 Pump coast down curve

- ***Flap valve opening time:*** during the commissioning stage A, the flap valve opening time was checked and it results equal to **76/75. s**, depending on the which pumps are selected, [4]. According to RETRAN simulation and depending on core flow value, flap valves open at **68.s** after the transient begins. It means that although this value is ≈ 10% lower than the measured ones it is a conservative value.

- ***Natural convection flow:*** once that flap valves open natural convection flow starts and, although this flow was not measured for LOFA conditions, measurements performed during Stage B of the commissioning program, for fission powers ≤ 400. KW show that results agreement with calculated values are within 13%, [5].

LOFA RESULTS

The main goal of this simulation is to calculate maximum temperatures in the coolant, cladding and fuel in order to be sure that fuel element integrity is preserved and that the core is within safety goals.

Summarising, the sequence of events for Operational Regime II simulated with RETRAN is the following:

⇨ ***t = 0.0s*** Pumps coast down begins.

⇨ ***t = 0.5s*** Heat exchangers are disconnected.

⇨ ***t = 2.0s*** The First Shutdown system triggers and shuts down the reactor.

⇨ ***t = 68.s*** Flap valves open due to the low flowrate and natural convection establishes.

A special comment deserves the decay power by the time that natural convection establishes, at **t = 68.s** ≈ **940 Kwatts**, it is higher than the one defined for Operational Regime 0 (**400 Kwatts**). It must be remembered that power for OR 0 is a steady state value based on operative experience while during transient analysis, a maximum of ***1070 Kwatts*** could be removed in the natural convection regime without any fuel damage, according to CONVEC program calculations [6].

In Figures 6 to 9 the core flow evolution as well as maximum temperatures, for both the hot and average channels and core inlet and outlet temperatures, are shown.

It must be bore in mind that figures begin at t=50.s, that is, once that the steady state was reached.

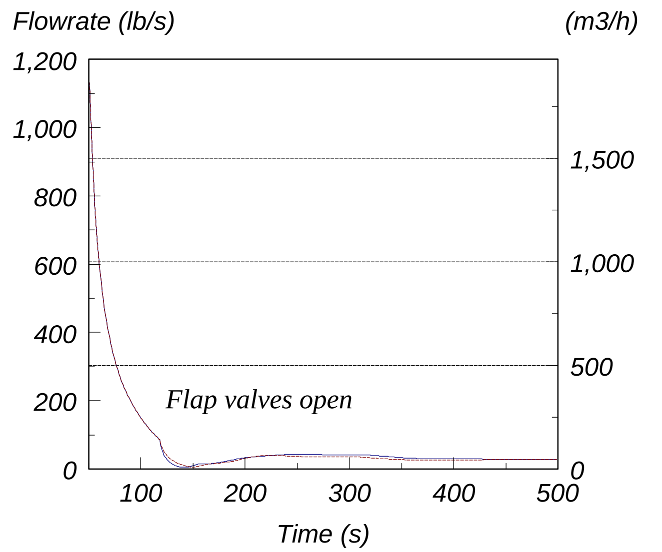

Fig. 6 Flow coast-down for operational regime II and the two running pumps

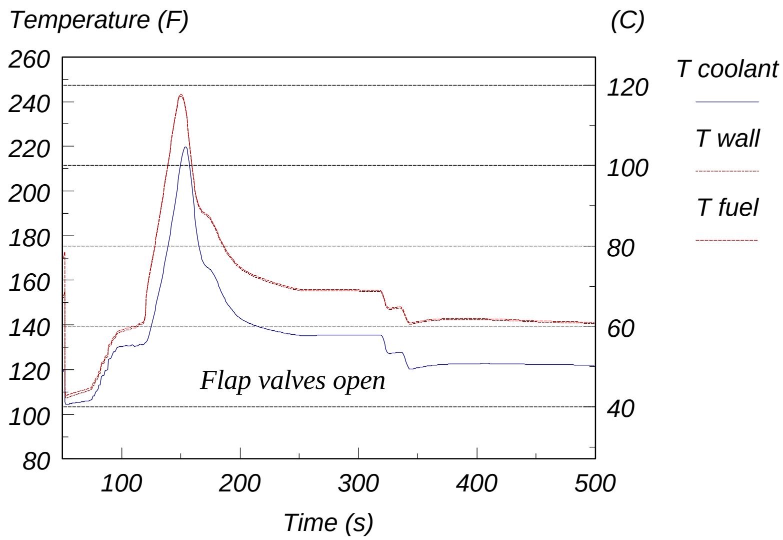

Figure 7: Average channel temperature distribution.

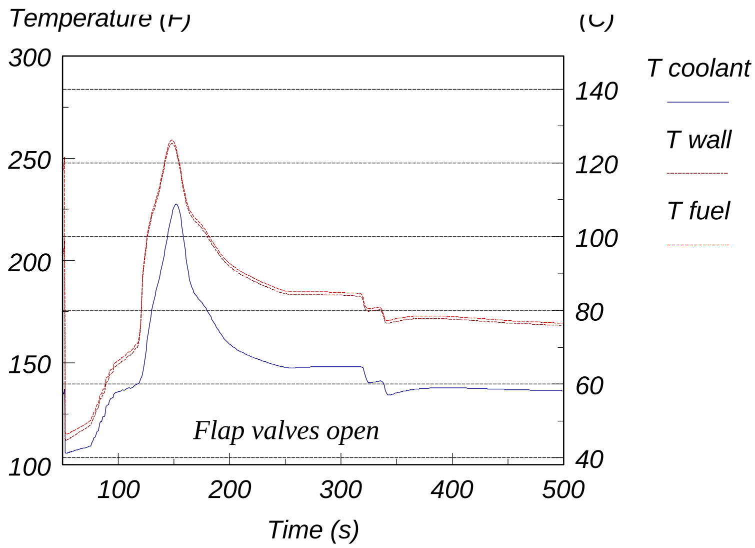

Figure 8: Hot channel temperature distribution.

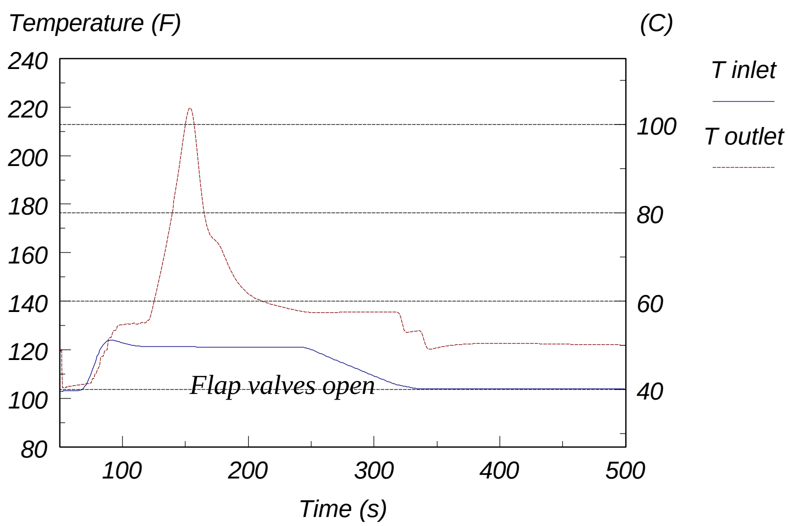

Figure 9: Core inlet and outlet temperatures.

Table 4 summarizes maximum temperatures for the average and the hot channels.

Table 4: Maximum temperatures

Fuel Temperature		Clad Temperature		Coolant Temperature	
Average	Hot	Average	Hot	Average	Hot
117.4	126.0	117	125.0	104.3	108.6

With reference to coolant temperature it is important to notice that the water saturation temperature, corresponding to a nominal pressure of 2.0 bar, is 120°C and, concerning the wall temperature design goal of 105 C it is exceeded but during a few seconds.

Some verifications have been done regarding DNB (Departure from Nucleate Boiling) and

with TERMIC code gave DNB and redistribution margins equal to 3.1 and 2.1, respectively, which are far above the 1.3 safety goal for abnormal situations.

A loss of energy supply, resulting in a loss of flow transient, has been simulated for Operational Regime II considering that the two flap valves open.

the reactor shuts down. Core flow decreases gradually due to the inertia flywheels until flap valves open and natural convection establishes.

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elements integrity during a LOFA incident, that is,

- The maximum wall temperature in the hot channel is well below the Al6061 melting point (560°C), and the Al6061 blistering temperature (450°C).

  Although maximum wall temperature in the hot channel exceeds, for limited periods, the

  integrity on the long term.

- Maximum coolant temperature in the hot channel for every regime does not exceed the water saturation temperature, 120°C.

  Safety margins to flow redistribution and DNB phenomena, in the forced convection

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REFERENCES

[1] ETRR-2 Safety Analysis Report

- EPRI NP- 1850- CCM- A. November 1988.

[3] TERMIC 1H v3.0: A Program for the calculus and thermal hydraulic design of reactor cores.

[4] Protocol n 767-COMM-047 Flapper valve opening time

[5] Protocol n 767-COMM-040 Power Operation in Regime 0

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