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TEST FACILITY FOR REWETTING EXPERIMENTS AT CDTN

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ABSTRACT

One of the most important subjects in nuclear reactor safety analysis is the reactor core rewetting after a Loss-of-Coolant Accident (LOCA) in a Light Water Reactor LWR. Several codes for the prediction of the rewetting evolution are under development based on experimental results. In a Pressurized Water Reactor (PWR) the reflooding phase of a LOCA is when the fuel rods are rewetted from the bottom of the core to its top after having been totally uncovered and dried out. Out-of-pile reflooding experiments performed with electrical heated fuel rod simulators show different quench behavior depending the rods geometry. A test facility for rewetting experiments (ITR - *Instalação de Testes de Remolhamento*) has been constructed at the Thermal Hydraulics Laboratory of the Centro de Desenvolvimento da Tecnologia Nuclear (CDTN), with the objective of performing investigations on basic phenomena that occur during the reflood phase of a LOCA in a PWR, using tubular and annular test sections. This paper presents the design aspects of the facility, and the current stage of the works. The mechanical aspects of the installation as its instrumentation are described. Two typical tests are presented and results compered with theoretical calculations using computer code.

1. INTRODUCTION

One of the fundamental subjects in Light Water Reactor (LWR) safety analysis is the rewetting of the reactor core after the postulated Loss-of-Coolant Accident (LOCA). Following the blow-down phase of a LOCA in a LWR, the fuel clad temperature may rise quickly to a high value (around 930ºC at PWR), so that the injected water from the emergency core cooling system (ECCS) during reflooding phase may not wet the clad immediately due to its sudden evaporation. As fuel cladding temperature decreases and Leidenfrost temperature is reached, water wets the surface and a more effective cooling takes place. Quenching the clad is essential for effective heat removal by the emergency coolant. The worst postulated condition would result from core melt down due to degradation on heat removal from the very hot fuels [1] [2].

Rewetting is the condition when liquid re-establishes contact with the hot solid surface. Associated processes called reflooding, quenching and refilling have also different industrial applications such as steam generators, metallurgical treatment, jet of rockets, start-up of liquefied natural gas pipe lines, and filling of vessels with cryogenic liquids at room temperature.

A great effort has been taken to study rewetting phenomena during the last two decades. These efforts include both experimental and analytical works [8] [9]. Accident Analysis techniques simulate the course of the accident using computer codes for verification of the nuclear reactor safety conditions. Older Accident Analysis computer code used very conservative assumptions and thermal-hydraulics models to compensate the lack of knowledge for the phenomena involved, which means that very large safety factors were used. More recent codes, which were developed within an international effort to allow the reduction of the high costs involved in building nuclear power plants, are able to describe more realistically the behavior of these plants.

The Centro de Desenvolvimento da Tecnologia Nuclear (CDTN) has a long lasting program to study experimentally and numerically several phenomana that affect reactor safety. Several conditions have and are being studied including the rewetting phenomena. In the past decade at CDTN a new group of codes have been used to study reactor safety. These are Computational Fluid Dynamics – CFD codes that solve the 3D flow and require validation so they can be properly used for accident analysis and licensing purposes.

Although there are experimental data on several phenomena, the results obtained in the past emphasized information that could lead to average behavior empirical correlations. For modern 3D code validation, a new set of more detailed measurements are required. Therefore there is a necessity to perform new experiments. This work describes the existing system for the experimental evaluation of rewetting in a PWR reactor and the improvements being performed.

2. THE TEST FACILITY FOR REWETTING (ITR)

2.1. Description of the Test Facility for Rewetting (ITR)

The Test Facility for Rewetting (ITR) is an experimental facility designed and built in the thermal hydraulics laboratory of the Centro de Desenvolvimento da Tecnologia Nuclear (CDTN). It simulates the conditions that occur in the reflooding phase of a LOCA in a pressurized water reactor using a tubular test section and it could also use an annular test section. Experiments can be carried on varying the initial wall temperature, the injected cooling water temperature and velocity, the heat flux dissipated in the testing section and the system pressure [5]. ITR is now being recovered to simulate the LOCA reflooding phase with more detail so that its data can be used to validate complex 3D CFD code simulation.

Table 1 provides the range of each experimental parameter. The maximum design pressure (6 bar) corresponds approximately to the predicted equilibrium pressure between the primary loop and the containment of a PWR, during a LOCA due to a major disruption. The heat flux in the test section and the injected water temperature and velocity correspond to the range expected during a LOCA. The maximum initial wall temperature was set up in 600 °C due to limitations with the thermocouple brazing to the test section wall. Figure 1 shows an installation photograph and Fig. 2 shows its flowchart.

Table 1: Experimental parameters range for ITR [3]

Parameter	Range	Step
Initial wall temperature	300 to 600 ºC	50 ºC
Injection water temperature	40 to 100 ºC	20 ºC
Injection water velocity	2 to 12 cm/s	2 cm/s
Heat flux through the test section internal surface	2 to 6 W/cm^2	1 W/cm^2
System pressure	2 to 6 bar	1 bar

Figure 1: Test Facility for Rewetting (ITR).

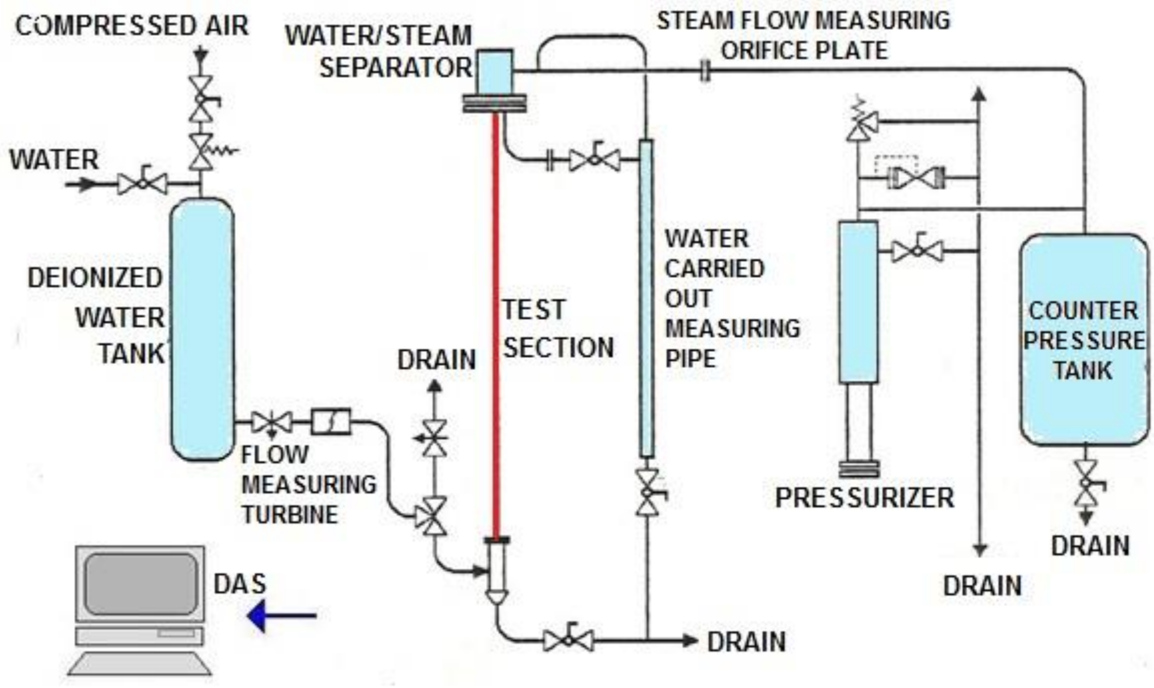

Figure 2: Diagram of the Test Facility for Rewetting (ITR).

The ITR is divided into three distinct segments (see Fig. 2):
- the water injection line;
- the pressurization line;
- the measuring line of the water carried out by the steam.

The water injection line consists of the piping section from the demineralized water tank to the entrance of the test section. This is a stainless steel AISI 316 piping, 50 mm in diameter, serial 40. The injection turbine flow meter and a globe valve for adjusting the flow are located in this line. After the turbine a three-way valve allow to direct the injection flow to the test section or to a draining piping. An experiment starts up by turning on these three-way valve to direct the flow to the test section. The draining piping has another globe valve for adjusting the upstream pressure at about the experimental pressure set up for the test section, so that after turning on the three-way valve the flow rate does not presents significant changes.

The pressurization line consists of the whole piping from the steam/water separator to the pressurizer. The piping is also in stainless steel AISI 316, 25mm in diameter. Close to the pressurizer there is a counter pressure tank to stabilize the system pressure. A relief valve installed over the pressurizer maintains the set up pressure. The safety valve has a higher set point (9 bar). The pressurizer has a set of resistors with an electrical power of 11 kW. A protective thermocouple welded at the top end of the resistance triggers the shutdown of them if they come to find out, thus preventing its burning. The volume of water above the resistors at the tests beginning allows carrying a complete test without the resistors becoming dry.

The measuring line of the water carried out by the steam consists of the water/steam separator, two vertical tubes for collecting and measuring the water carried out by the steam and precipitated in the separator and the collecting ducts. The water/steam separator is a camera above the test section where the entrained water precipitated due to the steam expansion. The steam leaving the test section undergoes an expansion upon entering the separator, causing it to precipitate the water droplets entrained in the steam. A piece of a tube 80mm in diameter, positioned within the separator, is another cause of the precipitation of entrained droplets of water, either by forcing the steam to make a sudden change in direction as to impact the tube. As the flood front is moving in the testing section increases the volume of water carried out. Thus, when the first tube is full, the second filling starts. The second tube a greater diameter and is able to contain a greater volume of water. A differential pressure transducer measures the volume of the precipitated water collected in the vertical tubes.

The ITR test sections were built with commercial tubes, but with hydraulic diameter close to Angra II fuel elements. It consists of an AISI 316 stainless steel tube 13,72mm external diameter and 9.24mm internal diameter. The tube has a total length of 4040mm: 3900 mm of heated length and 140mm upstream to get a fully developed flow. The test section is heated directly by Joule effect, using the tube itself as the heating resistance. The electric supply lower connection is directly on the test section tube, 140mm over the lower flange. The upper electric connection is on the test section upper flange.

At its upper portion the test section go through the flange of the water/steam separator ending 50mm above it to prevent the precipitate water to flow back into the test section (see Fig. 3). An outer tube, larger in diameter and wall thickness, makes the connection between the test section end and the flange.

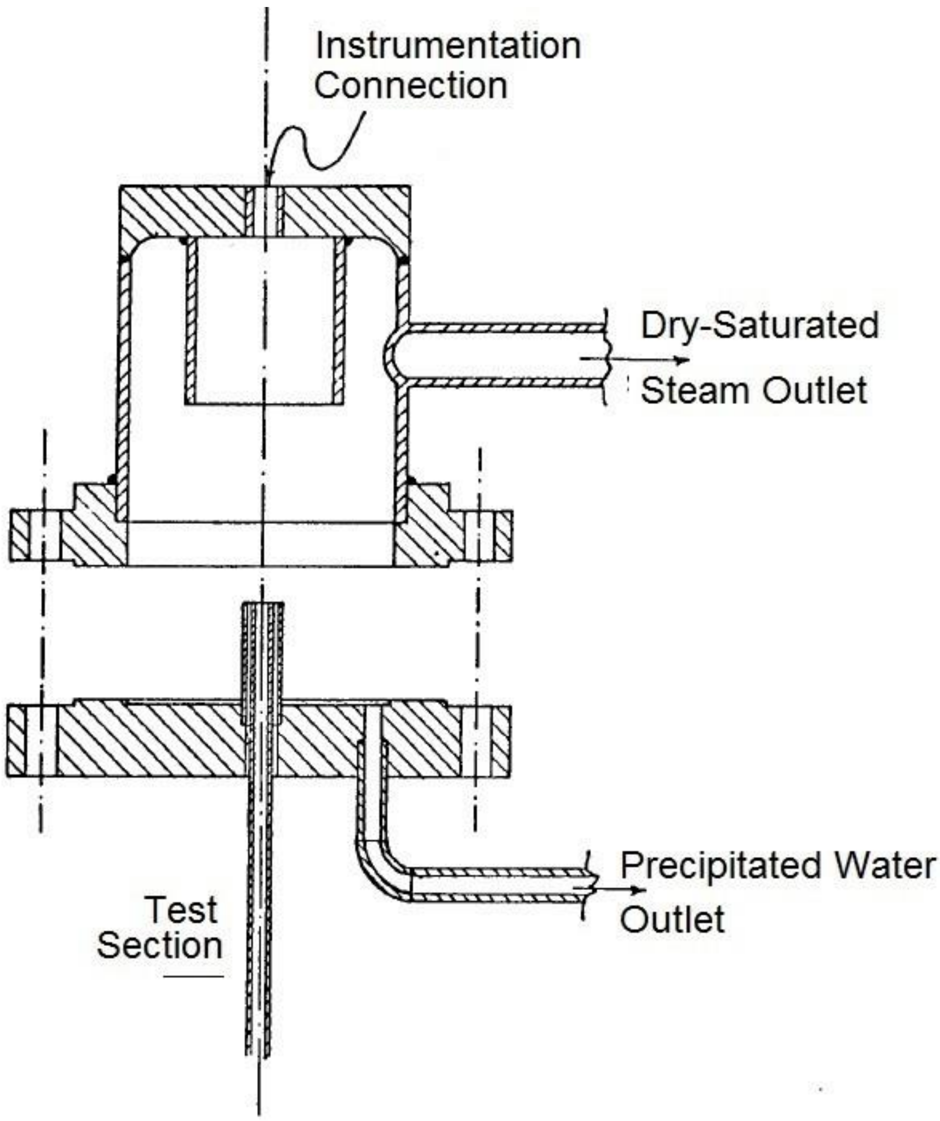

Figure 3: The Upper End of the Test Section and the Water/Steam Separator

2.2 The Instrumentation

Ten thermocouples were brazed in small slots along the test section for wall-temperature measurements. Their vertical positions are shown in Figure 4. There were used type K (cromel-alumel) thermocouples 1.5 mm in diameter with mineral insulation and stainless steel cladding. Each thermocouple is 60° radially phased from the previous thermocouple to obtain a spiral distribution.

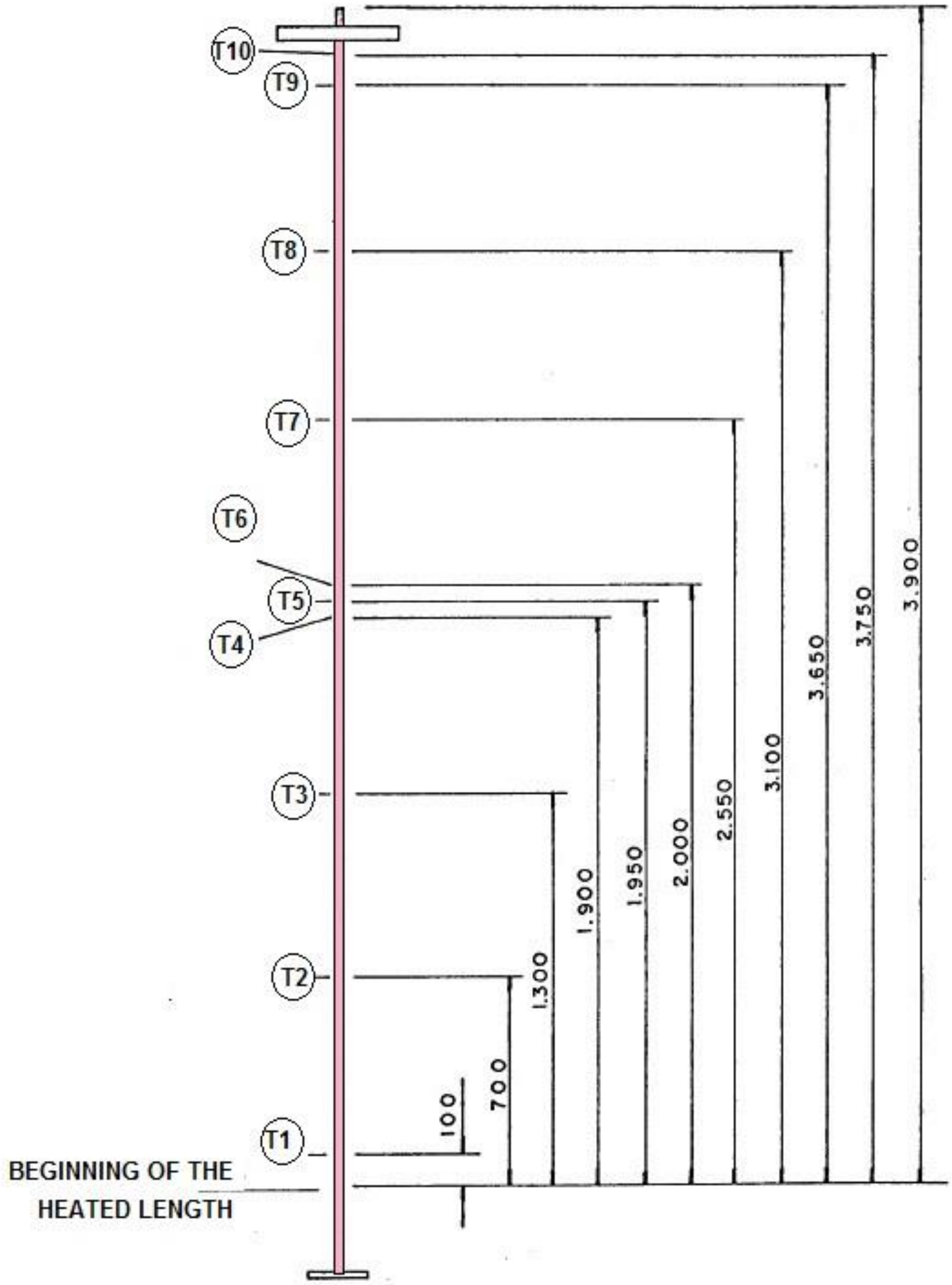

Figure 4: Thermocouple Distribution along the Test Section

The electric power on the test section is obtained as a function of the voltage, measured with a Hewlett-Packard digital voltmeter. The electrical resistance depends on temperature, but it is supposed to be linear with the average temperature on the test section. Out of the test section, pressure, level, flow rate and fluid temperature are measured along the experimental facility. For safety reasons the measured values are sent to a Data Acquisition System (DAS) installed remotely in a control room where it is necessary the presence of an operator. A new Data Acquisition System will be used for new experiments.

A thermo-resistor in the range of 0 °C to 100 °C measures the injection water temperature. Two gauge pressure transducers measure pressure on two different position on the injection line. An orifice plate and differential pressure transducer system is used to measured steam flow rate on the outlet of the water/steam separator. This system was defined according to Burton [6]. The methodology used to evaluate the propagation of uncertainty was based on Kline and McClintock [7].

3. TYPICAL TESTS

Table 2 shows the parameters values of two tests carried out in the past at ITR from the master's work of Rezende [5]. Figures 5 and 6 shows some results of these tests and compare these results with the results predicted by the German computer code Hydroflut [8] [9]. They show the evolution of the wall temperature in the positions of three thermocouples on the test section.

Table 2: Values of Parameters for Two Tests Performed with ITR

Parameter	Test 1	Test 2
Pressure	4 bar	1 bar
Injection Velocity	6.2 cm/s	6.2 cm/s
Water Injection Temperature	400 °C	400 °C

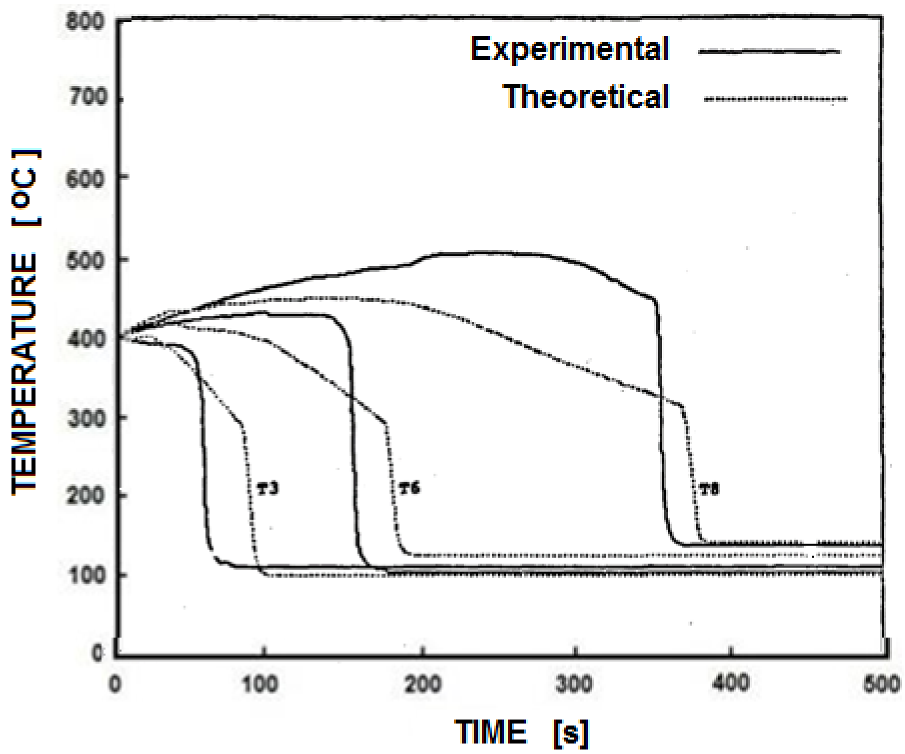

Figure 5: Experimental and Calculated Evolution of Wall Temperature in Test 1.

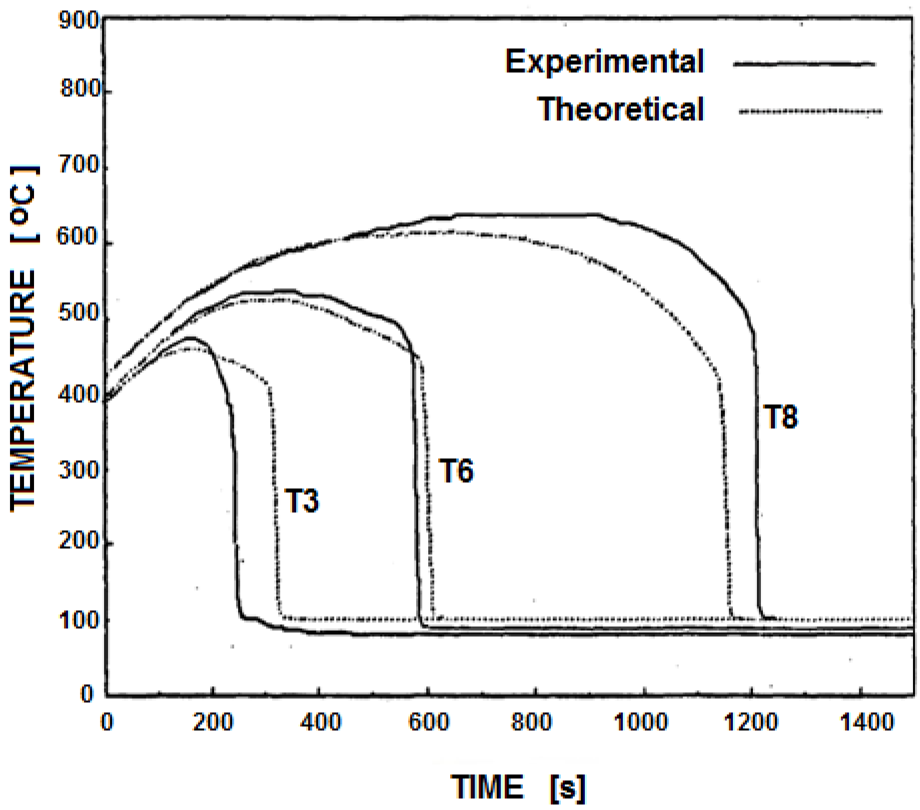

Figure 6: Experimental and calculated evolution of wall temperature in Test 2.

4. CONCLUSIONS

The ITR installation for studying rewetting of PWR nuclear reactors was presented. At the time when the original work was performed a few thermocouple results were enough to determine satisfactory empirical correlations.

Work is being performed to update the test section and increase instrumentation and lower uncertainty of the results. Also a better Data Acquisition System will be used in the facility. These efforts are being made so that a new experimental campaign can be performed to obtain results for complex 3D code validation, as CFD.

New experiments with ITR include a planned experimental study on the effects of nanoparticle deposition on rewetting.

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REFERENCES

1. TODREAS, N. E.; KAZIMI, M. S. *Nuclear Systems Vol 1: Thermal hydraulic Fundamentals*. CRC Press, New York, 1002p. 2012.
2. MULYA, J.; ANHAR, R. Antariksawan, J. P.; ISMU, H.; EDY, S. & KISWANTA, B. Experimental Study of Quenching Process During Bottom Reflooding using “Queen” Test Section**.** Risk Analysis and Accident Mitigation Division. Nuclear Safety Technology Development Center (P2TKN) – BATAN. Indonesia. (2005).
3. LEE, D. W.; NO, H. C.; KIM, H. G. & OH, S. J._An experimental study of thermal-hydraulic phenomena in the downcomer with a direct vessel injection system of APR1400 during the LBLOCA reflood phase. *Journal of Nuclear Science and Technology*, **Volume 41, Nº 4**, pp. 440-447, (2001).
4. CHIKHIA, N.; NGUYENB, N. G.; FLEUROTA, J. “Determination of the hydrogen source term during the reflooding of an overheated core: Calculation results of the integral reflood test QUENCH-03 with PWR-type bundle”, *Nuclear Engineering and Design*, **Volume 250**, pp. 351-363, (2012).
5. REZENDE, H.C. Projeto, Montagem e Comissionamento de uma Instalação de Testes de Remolhamento. Dissertação de Mestrado. Escola Engenharia da UFMG, Belo Horizonte. 126p. (1985).
6. BURTON, J. Pratique de la mesure et du controle dans 1'industrie, 10ª. ed., Paris, Dunod, (1965).
7. KLINE, S. J.; McClintock, F. A. Describing Uncertainties in Single-Sample Experiments. Mechanical Engineering. 3-8, (1953).
8. KÜHLER, W. Precalculation of CSNI LOCA standard problem no.7 on reflooding with the Hydroflut code. Erlangen, Kraftwerk Union, 21p. (Arbeitsbericht KWU/R513/9/79), (1979)
9. HEIN. D. Modellvorstellung zum Wiederbenetzen durch Fluten. Hanover, Thesis, University of Hanover, 134p, (1980).