A Core Pressure Drop Monitoring System: Design and implementation at the RA6 Research Reactor

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Abstract

The safe design of a nuclear reactor requires the installation of measurement systems that monitor those parameters which protect the integrity of the fuel assemblies (matrix / cladding, the main barriers against fission product release). These parameters are considered to be vital for the nuclear safety of the reactor, and their measurement must to be fast and reliable.
In the reactors that use forced circulation to remove the heat produced in the core, the primary coolant flow measurement is usually a safety parameter, since it gives a fast indication of degradation of the capability to remove the energy produced in the reactor. Given that the core is a flow restriction, there is also another protective signal (known as Core Coolant Differential Pressure – core DP) to ensure that the coolant is flowing in the system. The core DP has the advantages of its high sensitivity to small variations in core flow.
This paper presents a simplified design to implement a core pressure drop monitoring system in a MTR pool type research reactor with no changes in core components and supporting structures, simple and reliable operating principle, easy to mounting and inexpensive.
A characterization of certain core pressure drop changes that are expected to happen during nominal operating conditions, versus others arising from some experimentally simulated LOFA events, are compared in absolute and relative manner, in order to appreciate the system sensitivity to detect anomalies. Finally, the upper and lower core DP thresholds are set, which are intended to demand the reactor automatic shutdown whether certain specified values are exceeded.

1. Introduction

The majority of research reactors operating today were put into operation many years ago. Some of them have been recently converted from HEU to LEU and have been upgraded in power to meet requirements for higher neutron fluxes. In these upgraded facilities, in order to be consistent with modern safety standards and regulatory requirements, it might be mandatory to provide additional protective instrumentation systems to improve reactor´s safety.

In reactors that use forced circulation to remove the heat produced in the core, given that the core is a flow restriction, there is a protective signal (known as Core Coolant Differential Pressure) to ensure that the coolant is flowing in the system. The core DP has the advantages of its high sensitivity to small variations in core flow.

Usually, pressure drop through the core is measured by means of differential pressure transmitters. In old reactors, where assembly connections for differential pressure transmitters were not originally considered, the required modifications to do so (in the core components and its supporting structures) are very often complicated and expensive to implement.

This paper presents a simplified design to implement a core pressure drop monitoring system in a MTR pool type research reactor.

1.1 RA-6 Reactor

It is a pool-type reactor with a power output of 1 MW, cooled by light water and moderated by graphite. It has plate-type fuel assemblies with aluminum cladding. It is owned and operated by the National Atomic Energy Commission of Argentine. In 2010 it was converted from HEU to LEU (silicide fuel) and it was upgraded in power. It is used mainly for: Teaching of Nuclear Engineering, Neutron Activation Analysis, Boron Neutron Capture Therapy, training of nuclear operators, Research in Reactor Physics & Instrumentation.

The RA-6 core is assembled on a support grid with 80 available positions, 15 of which are occupied with normal Fuel Elements and 5 of which are occupied with control Fuel Elements. The rest are occupied by graphite reflectors, irradiation boxes, aluminum plugs, nuclear instrumentation and other devices, but there is not coolant flow circulation through them. The array of fuel assemblies is shifted from the support grid center due to neutron flux requirements of the experimental facilities. (*see FIG.2*)

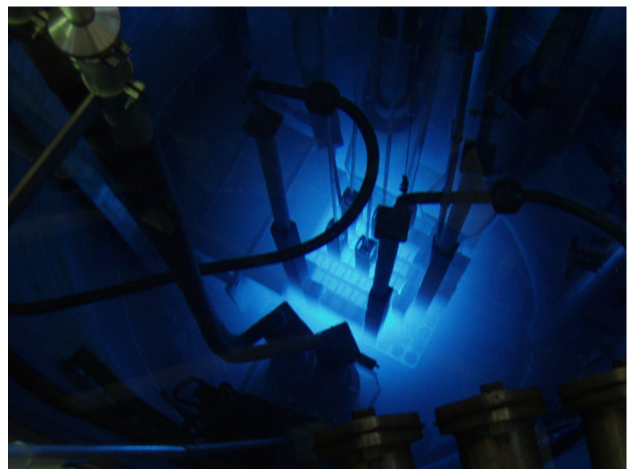

Figure 1: RA6 at full power

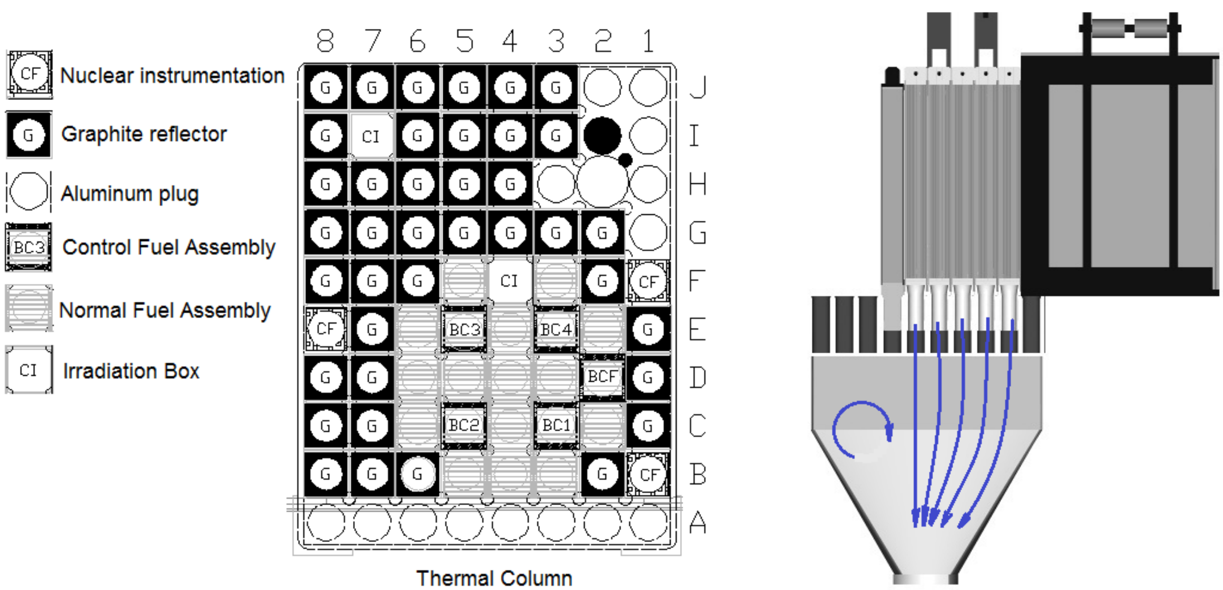

Figure 2 - Core configuration - top view and cross sectional view.

1.2 Normal Operating Conditions

Normal operation is defined as the reactor functioning within specified operational limits and conditions. From shutdown condition to nominal operating condition, the performed actions that produce expected core DP changes are described below:

Reactor Shutdown: The initial condition is subcritical with the control rods fully inserted into the reactor core. When the cooling pump is started, a steady cooling flow rate in the primary circuit is established and the corresponding nominal core pressure drop rises from 0 up to 370 mmH2O approximately.

Reactor Start up: To achieve the critical condition in the reactor, it is necessary the withdrawal of the control rods which are fully inserted in the core. This operation causes a decrease of core DP, since the geometric configuration of the core becomes less restricted to the coolant flow pass.

Reactor Power: The increasing reactor power leads to a heating of the water flowing through the inner fuel channels, which causes a reduction of the core hydraulic resistance and the corresponding decrease of the pressure drop.

Reactor's pool water temperature: In research reactors which have not a control system to maintain a steady pool water temperature (i.e. a fixed core coolant inlet temperature), a wider range of operating conditions must be permitted, due to that both coolant temperature and the hydraulic losses will vary depending of the meteorological condition prevailing at the reactor site, at the moment of the operations.

1.3 Postulated Initiating Events

In open pool type research reactors, where the core is cooled by forced circulation water in downward direction, there are abnormal or unforeseeable events that could lead to a LOFA (Loss of Flow Accident), whose main consequence is the reduction or loss of the coolant flow through the core and could jeopardize the fuel assembly integrity[1]. Some of these Postulated Initiating Events PIE are described below:

LOFA 1: Fuel channel blockage

There is some likelihood of a fuel assembly blockage by objects dropping into the pool. A partial blockage of the cooling channel constitutes a severe accident. The rapid loss of flow occurring in the obstructed channel leads to sharp two-phase flow conditions with rapid increase of coolant and fuel temperatures. Under these conditions, a large amount of vapour is produced leading to local dryout and eventually to loss of the fuel cladding integrity.

[1] Fuel safety temperature limit is the one corresponding to the blistering phenomenon, which is conservatively assumed to be equal to 400ºC.

Partial blockage of the core must be detected during normal operation of the reactor by the DP measurement system; a significant increase of signal should be seen owing the coolant flow redistribution through the rest of the fuel assemblies.

LOFA 2: Coolant reduction due to core bypass

In pool type research reactors is required to install flap valves and other primary cooling piping components in order to commute between the two cooling modes: forced circulation in power condition and natural convection in shutdown condition. Additionally siphon breakers (floating valves type), are installed in the primary circuit to prevent total pool water drainage in case of LOCA accidents.

When the primary cooling pump is started, a failure in closing these mechanical components might constitute a serious incident, since it allows the circulation of the coolant through an alternative pathway, thereby decreasing the coolant flow through the fuel assemblies.

Core bypasses must be detected during normal operation of the reactor by the DP measurement system; a significant decrease of signal should be seen owing the circulation of the coolant through an alternative pathway.

LOFA 3: Improper Assembling - Human error

Human actions in themselves are not PIEs; rather, human actions can be a cause of an PIE. Assembly and disassembly of the core in research reactors is made, inter alia, to fulfil different user requirements, to replace spent fuel assemblies, to obtain large subcritical configurations in extended shutdown, etc. Therefore the operators must carry out multiple remote handling movements underwater. In view that the core has several components besides fuel assemblies such as graphite reflectors, irradiation boxes, aluminum plugs, etc, there is a reasonable likelihood of omissions at performing such kind of assembly maneuvers. (ej: forgetting to place a core component)

An open hole in the core support grid, due to a component forgotten during assembly, can constitute a serious incident because it allows the circulation of the coolant through an alternative pathway (bypass) thereby decreasing the coolant flow through the fuel assemblies.

Human errors during assembly must be detected by the DP measurement system before reactor's start-up; a significant decrease of signal should be seen owing the coolant flow bypass through an open hole in the core support grid.

2. Core DP Measurement System

2.1 System Description

Two Immersible Pressure Transmitters (strain gauge sensor) [4] are used to accurately and reliably measured the hydrostatic pressure from the liquid column over them. Analog signals from the loop-powered pressure transmitters are processed by an electronic module where both

4-20 mA current loops are filtered and subtracted, to obtain an output linearly dependent to the core pressure drop.

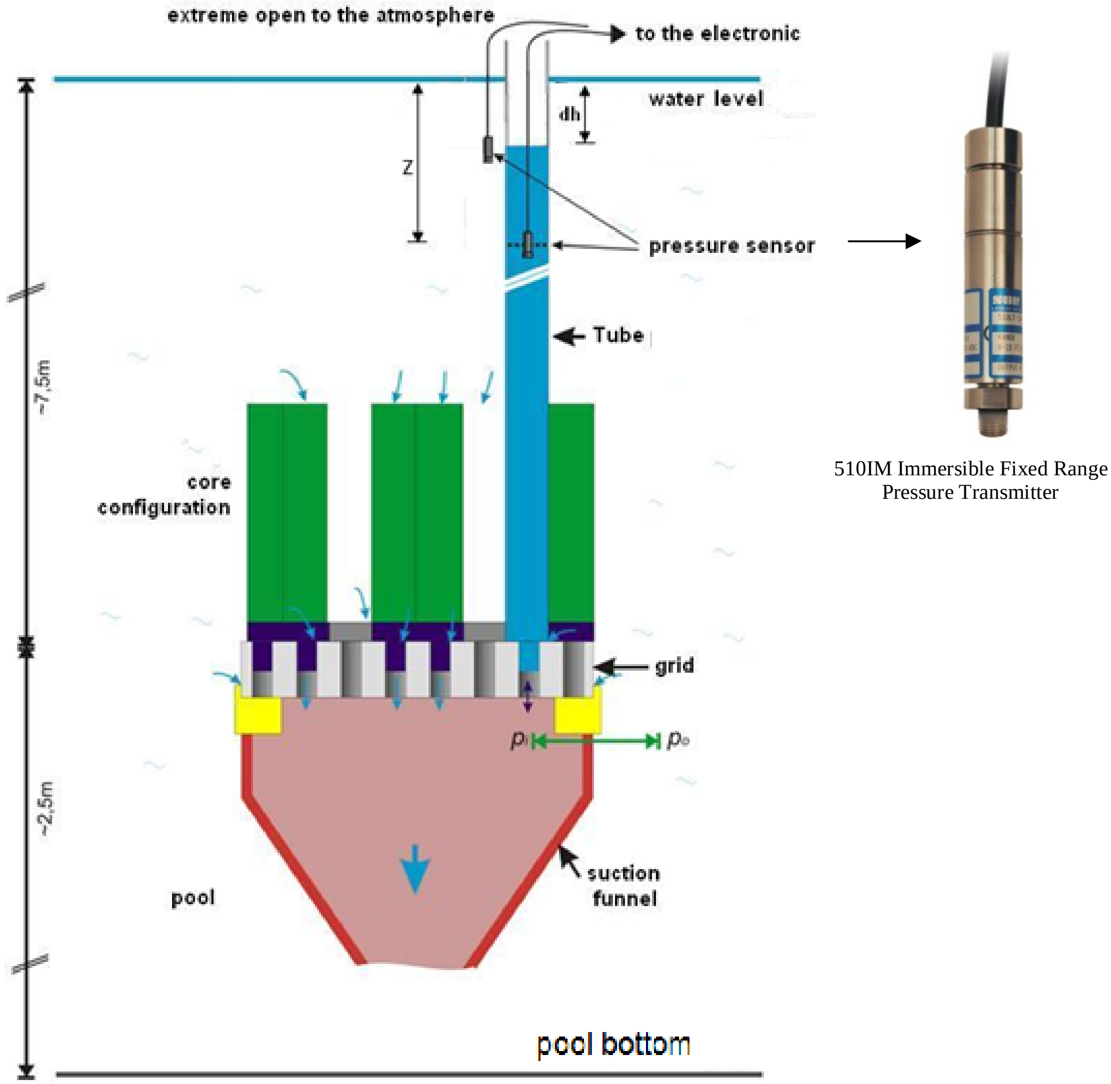

Figure 3 – schematic design of the core pressure drop measurement system.

The basic design consists of a long vertical tube connected to the core lower plenum through the core support grid (*see FIG.3 above*). At the top of the reactor pool the two pressure transmitters are placed underwater at suitable levels and fixed to the tube (*see FIG.4 below*). One sensor (outside) measures the steady level of the pool water, while the other (inside) is used to measure the variable level of water within the tube, which is related to the core pressure drop produced by forced circulation of the primary coolant.

When the primary cooling pump is in shutdown, there is no coolant flow rate through the core, and the inner sensor output signal is proportional to the height of the water column Z. When the pump is started, a steady flow rate is established in the primary cooling circuit, the water level inside the tube will descend and the inner sensor output signal will be proportional to the height of the water column (Z-dh). The height difference "dh" is the Core Pressure Drop.

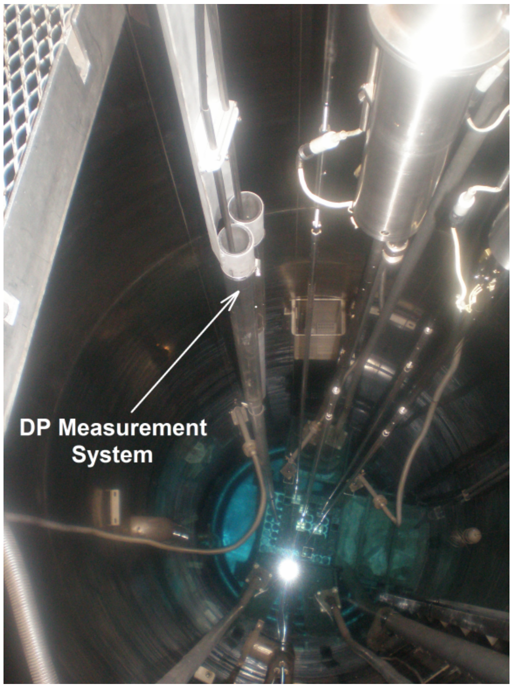

Figure 4 – top view of the mounting.
The long tube connected to the core support grid is shown

Analog signals from the pressure transmitters are processed by an electronic module, where both electrical current loops are filtered and subtracted in order to obtain an output that is linearly dependent of the core differential pressure. This analog signal is sent to an electronic comparator where the upper and lower threshold can be adjusted (safety system settings). The output signal from the comparator is sent to the Reactor Protection System which monitors all the safety parameters and demands the automatic reactor shutdown whether specified values are exceeded.

In the main control room, the value of core DP in engineering units [mmH2O] is displayed. Moreover this signal is sent to an Electronic Data Acquisition System (SEAD), where all the reactor operating variables are registered.

2.2. System Sensitivity

Pressure transmitters utilized have a range from 0 to 3600 mmH2O and they deliver a proportional 4-20 mA electrical current, i.e., the calibration factor of these sensors is about 225 mmH2O/mA. Therefore, from the electrical current signal (rounded to the second decimal place) an intrinsic resolution of about 2 mmH2O in the determination of the water column over the transmitter is obtained.

With the primary coolant flow rate at nominal steady state, the DP signal shows a characteristic oscillating behavior associated with pressure fluctuations in the core lower plenum (*see FIG.5 below*). It is consider that these fluctuations are associated with the turbulent movement of the fluid within the suction funnel. Figure 5 shows a typical acquisition of DP signal (about 5 minutes). These fluctuations generally do not exhibit peak-peak amplitudes greater than 10 mmH2O.

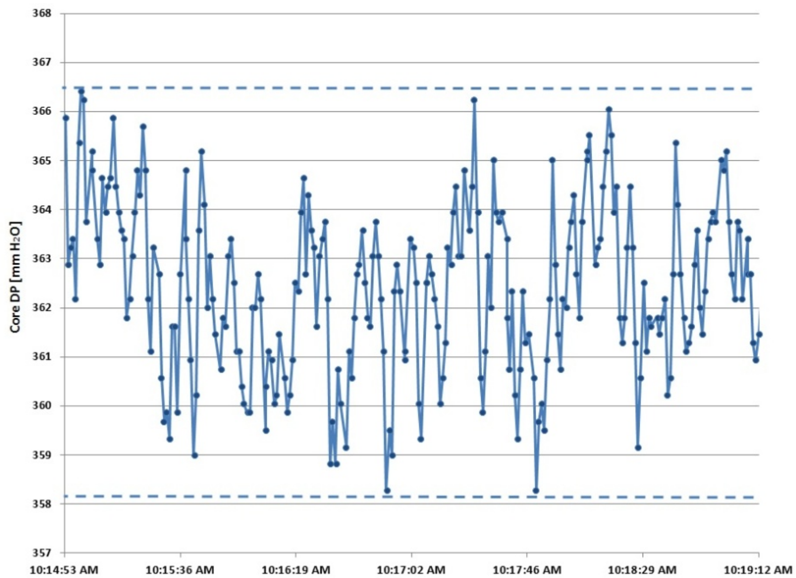

Figure 5 - Typical core DP signal acquired in nominal steady state condition.
It can be seen a variability caused by slow pressure fluctuations in the lower plenum.

3. Data Analysis and Results

3.1 DP changes at Normal Operating Condition

As has already been mentioned in section 1.2 some actions that produce expected core DP changes are associated with different states of the reactor, namely: cold start-up, critical state and full power state.

In order to evaluate the performance of this measurement system, files of DP signal acquired and registered throughout several months, which correspond to some 1MW steady continued operations, were analysed. Short time operations, where a thermal equilibrium is not reached in the primary cooling circuit were not considered.

Figure 6 shows the typical DP signal evolution from the cold start-up to full power state:

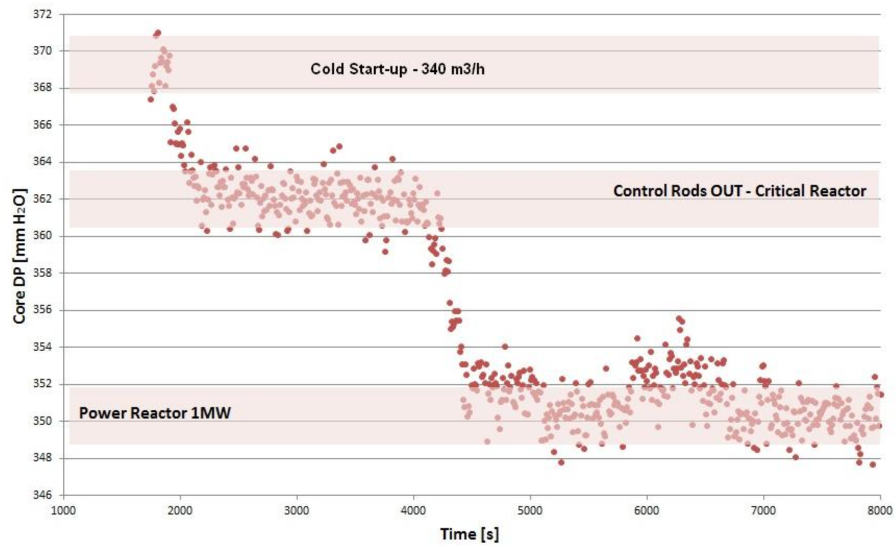

Figure 6 - Typical DP signal evolution during cold start-up and power increase
Red bands indicate the three steady state conditions.

It can be seen sensible changes (~10 mm H2O) when the reactor goes from cold start-up to critical condition and then from critical condition up to full power. This behaviour is observed in all operations analysed.

In order to compare the different operations registered, an average value was calculated and its corresponding uncertainty (mean and standard deviation of the data) for each DP steady state.

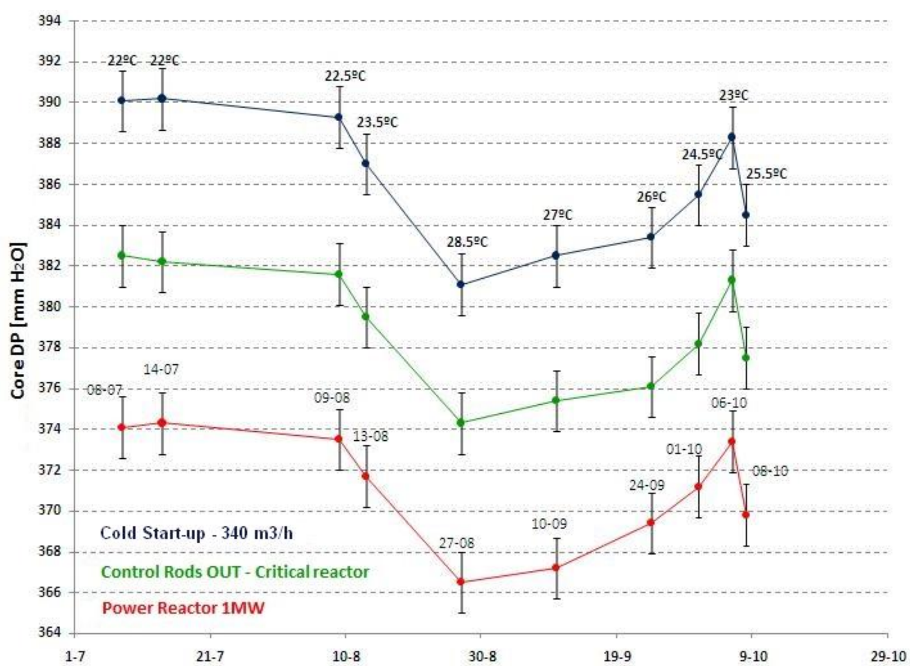

Figure 7 - core DP changes registered in several operation dates.
Each operational condition is plotted

Figure 7 shows the values of the core DP registered with different pool water temperatures owing to different meteorological conditions prevailing at the reactor site in those dates.

Figure 8 shows the same data of the previous figure, but they are plotted as a function of pool water temperature. A linear relationship between core DP versus pool water temperature, for each steady operating condition, is obtained.

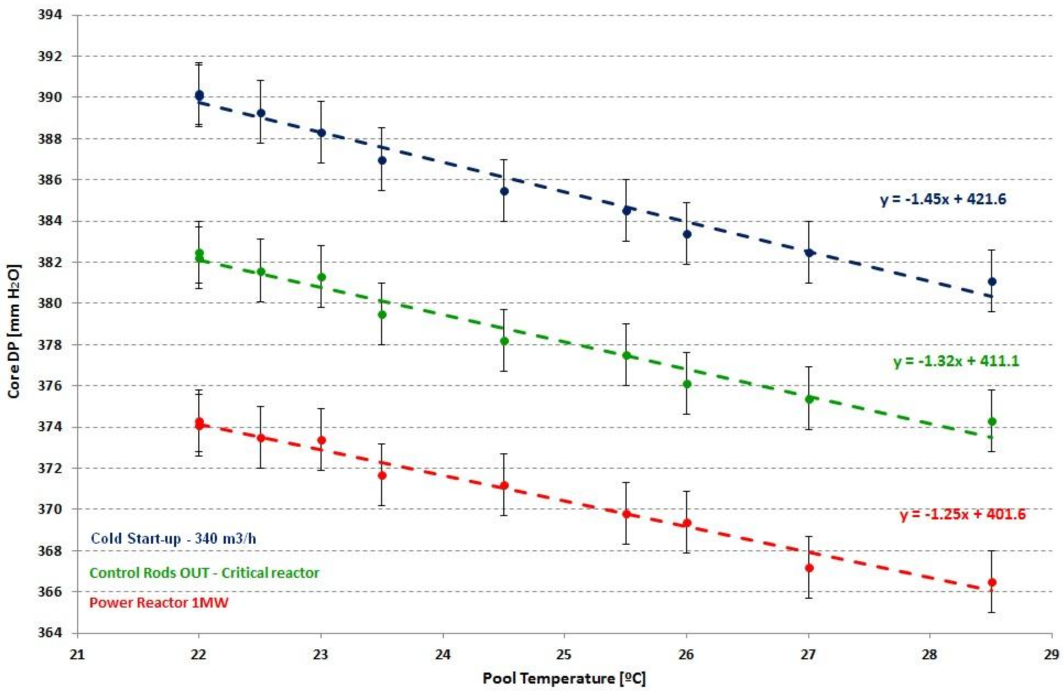

Figure 8 - core DP vs. pool water temperature relationship.

It can be seen that the increasing of pool water temperature leads to a core DP decrease of 1,3 mmH2O/ºC. Therefore, it can be established that the range of permitted operational conditions for pool water temperature (20ºC - 35°C) implies a core DP maximum variability of 20 mmH2O, approximately. Nevertheless, the pool water temperature in the RA-6 normally experiences a narrower range of variability throughout the year.

Cause	% of nominal core DP
Lower plenum pressure fluctuations	± 1,5 %
Control rods withdrawal	- 3 %
Reactor power increase	- 3 %
Pool water temperature variations	- 0,4 % / ºC

TABLE I – Summary of core DP variations in normal operating conditions.

Considering that the nominal core DP in the RA-6 is in the order of 350 mmH2O, expected variations of core DP can be expressed in a relative manner for each operational condition. (See Table I above).

3.2 DP changes at simulated LOFA events

As has already been mentioned in section 1.3 there are some postulated initiating events that could lead to LOFA (Loss of Flow Accident). A set of experiments were performed to determine the magnitude of DP changes caused by this kind of events. They were experimentally simulated at reactor shutdown condition, with an isothermal primary cooling circuit at nominal flow rate.

LOFA 1: Fuel channel blockage

Measurements were made at different locations of the core and the following results were obtained:

Grid Position	DP change [mmH2O]
B3	20 ± 4
D4	21 ± 4
F5	26 ± 4

TABLE II - Core DP change caused by the blockage of a single fuel assembly in different locations of the core.

The smallest DP changes are obtained in the peripheral locations of the core support grid. This is because the flow rate distribution is strongly dependent on the relative position of the fuel assemblies array with respect to the core support grid center and to the lower suction funnel geometry.

Partial core blockage can be precisely detected by the RA-6 DP measurement system. A significantly increase on the DP signal is obtained caused by the redistribution of coolant flow through the rest of fuel assemblies.

LOFA 2: Coolant reduction due to core bypass

Table III shows the magnitude of DP changes caused by this kind of undesirable malfunctioning of primary circuit mechanical components.

Flow bypasses can be precisely detected by the RA-6 DP measurement system. A significantly decrease on the DP signal is obtained caused by the loss of coolant flow through the fuel assemblies.

Condition	DP change [mm H2O]
Full open flap valve	240 ± 4
Full open siphon breaker	77 ± 4

TABLE III – core DP changes caused by opening the flap valve and the siphon breaker.

LOFA 3: Improper Assembling - Human error

Table IV shows the magnitude of DP changes caused by the presence of an open hole in the core support grid. Measurements were made at different locations of the support grid and the following results were obtained:

Position	DP change [mm H2O]
B3	16 ± 4
D4	26 ± 4
F5	32 ± 4

TABLE IV – Core DP changes caused by an open hole in different locations of the core support grid.

The smallest DP changes are obtained in the peripheral locations of the core support grid. This is because the flow rate distribution is strongly dependent on the relative position of the fuel assemblies array with respect to the core support grid center and to the lower suction funnel geometry.

Human errors during assembly can be precisely detected by the RA-6 DP measurement system. A significantly decrease on the DP signal is obtained caused by the coolant flow bypass through an open hole in the core support grid.

Once again, considering that the nominal core DP in the RA-6 is in the order of 350 mmH2O, expected variations of core DP can be expressed in a relative manner for each experimentally simulated LOFA events. (See Table V below).

Event	% of nominal core DP
LOFA 1	Increase 6 % to 7 %
LOFA 2	Decrease 20 % to 65 %
LOFA 3	Decrease 5 % to 9 %

TABLE V – Summary of core DP variations for each LOFA event

4. Safety System Settings

The set point for protective actions is defined as the safety system settings. In determining a safety system setting, the process uncertainties and measurement uncertainties, the response of instrumentation and uncertainties in calculations should be taken into account. It shall be ensured in the design that the set points can be established with a margin between the initiation point and the safety limits such that the action initiated by the reactor protection system will be able to control the process before the safety limit is reached.

Starting from nominal steady state condition (reactor critical at zero power) and in order to establish a working window for core DP signal, the following expected variations should be considered:

- There will be a decrease in core DP signal when the reactor goes from critical condition to full power (-10 mm H2O).
- It can be expected pool water temperature variations throughout the year (~ 10 mm H2O).
- There will be some variability in the core DP signal owing to pressure oscillations in the lower plenum (+/- 5 mmH2O).

Therefore, a minimum working window of 30 mmH20 have to be allowed in order to accommodate expected variations of core DP signal during normal operating conditions.

With this information, a first approximation for lower and upper limits for the core DP safety window can be established.

An exhaustive analysis of monthly registered operations (acquisition database) allows fine tuning of the thresholds for this protective action, considering all the variations formerly described from a statistic point of view.

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5. Conclusions

- A simplified design to implement a core pressure drop monitoring system, in a MTR pool type research reactor, was presented. It needs no changes in core components and supporting structures, has a simple and reliable operating principle, easy to mounting and inexpensive.

- The installation of this core pressure drop monitoring system in the RA-6 reactor provides a signal that is utilized in forced cooling mode, to prevent that the temperature of the fuel assemblies inadvertently exceed certain upper limits.

- The nominal value of the core pressure drop in the RA-6 reactor is approximately 350 mmH2O. The precision of the core DP Measurement System is ± 4 mmH20, which is about ± 1% of its nominal value.

- Some LOFA initiating events can be detected by this measurement system depending on the magnitude of coolant flow blockages or coolant flow derivations considered.

- Minimum working window of 30 mmH20 have to be allowed in order to accommodate expected variations of core DP signal during normal operating conditions.

- A minimum safety window of 35 mmH2O should be established whether spurious demands to the Reactor Protection System want to be avoided.

- With this safety system setting any sudden change exceeding 10% of nominal core DP value will cause the automatic shutdown of the reactor.

- Future actions. Safety signals which are sent to the Reactor Protection System (RPS) must be redundant. Therefore, after completing the characterization of this prototype, the triple redundancy of the RA6 core DP measurement systems will be implemented, in order to generate the corresponding RPS trip signal processed in a 2 out of 3 coincidence logic.

6. References

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Research Reactor Instrumentation and Control Technology, IAEA-TECDOC-973, Vienna (1997)

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Analysis for Research Reactors, IAEA Safety Reports Series No. 55, Vienna (2008)

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Research Reactors, IAEA Safety Standards Series No. NS-R-4, Vienna (2005).

[4] http://www.sorinc.com 510IM Immersible Fixed Range Pressure Transmitter