

**RESEARCH AND DEVELOPMENT ON EQUIPMENT AND SYSTEMS MONITORING AND
DIAGNOSTICS AT THE INSTITUTO DE PESQUISAS ENERGÉTICAS E NUCLEARES**

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Abstract : A brief summary of the main ongoing research and development activities at the Division of Monitoring and Diagnostics, IPEN is presented. These include advanced techniques for detection of faulty instruments like RTD's, pressure transmitters as well as field actuators, vibration analysis of rotating machinery, leak detection of pressurized pipings and vessels, systems condition monitoring, motor operated valves monitoring, automatization of Eddy current testings and other applications. The main mathematical tools being studied include time frequency transforms like wavelets transforms, higher order spectral analysis, neural network and fuzzy logic.

Introduction:

The availability of a equipment is of utmost importance for the economical equation of a given installation. In the case of a nuclear power plant the reliability and safety are additional parameters to be met. Since the requirements the safety of the general public are very restrictive. These requirements can be achieved by a proper predictive maintenance with advanced monitoring and diagnosis of critical equipment and systems. To improve the availability, reliability, capacity and safety of a plant to reduce capital and operational costs as well as to better manage the equipment ageing extending their life, several research and development activities are under way in our division and they will be summarized in this paper:

- 1) Time response measurement of temperature and pressure sensors.
- 2) Plant condition monitoring and fault detection.
- 3) Aging study of resistance temperature detectors (RTD).
- 4) Development of a system for monitoring and diagnosis of steam generator tubes using artificial intelligence techniques and wavelets on eddy current test signals.
- 5) Monitoring and automated diagnosis of defective bearings using advanced signal processing.
- 6) Sub cooled boiling detection using signals spectral analysis.
- 7) Design and construction of an isolating amplifier for noise analysis.
- 8) Development of an integrated condition monitoring system for motor operated valves.
- 9) Leak detection and localization in pipings and vessels.

1) Time response measurement of temperature and pressure sensors.

The determination of response time of temperature and pressure sensors from the Reactor Protection System is a requirement that must be satisfied. Furthermore is an indicative of its degradation and its remaining working period. In general one can say that there are two basic methods to determine the response time: direct and indirect methods. The direct method used to determine the response time of Resistance Temperature Detector (RTD) is the **Loop Current Step Response (LCSR)**, and the method used to determine the pressure sensor response time is the **Step Test or the Ramp Pressure**. The indirect method is **Noise Analysis Technique** and consists in analyses the dynamic behavior of the sensor, where one can obtain the information with a minimal interference during the normal operation of the nuclear power plant.

The **Loop Current Step Response (LCSR)** method consists in applying a small current to the RTD leads. The temperature transient due to a step change that heats the sensor filament is analyzed to determine the response time that would have followed a fluid temperature change. The test can be implemented in a plant by connecting a test instrument at the point where the sensor leads are normally connected to their in-plant transmitter. Laboratory results show that the time constant estimates by this method are within 10 to 20 per cent of their true values.

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A small suitcase was made to hold the I.CSR circuit. The circuit produces a step current and measures the resistance variation of the RTD. Basically the circuit is a Wheatstone bridge in which the RTD is one of its resistors. The bridge output is acquired using a data acquisition board along with software developed using Labview®.

Step Test or the Ramp Pressure of pressure transmitters - Two methods can be classified as direct methods: pressure ramp test and pressure step test. These tests are called "direct methods" because they must be done "in-situ", i.e., directly applied to the sensor. The pressure ramp test basically consists in applying to the sensor under investigation and simultaneously to a fast response sensor a pressure ramp input signal and compare the responses of both sensors to determine the time response. This method was developed in the 70's by the Nuclear Services Corporation supported by the Electric Power Research Institute (EPRI), and together with the methodology uses an experimental hydraulic signal test unit, described below.

The dynamic pressure test unit was designed to generate positive and negative step pressures and positive pressure ramps. The test unit is composed by two accumulator flasks and a solenoid valve and a throttle valve in sequence for initiating and controlling the rate of pressure equalization between the accumulators. Generation of increasing ramp waveforms is accomplished by pressurizing the system to the driving pressure with compressed nitrogen or air and then, with the solenoid closed, bleeding accumulator A1 to the transient starting pressure. The throttle valve is then adjusted to obtain the desired signal ramp rate, and the solenoid valve is opened to initiate the signal. Analyzing the output of both sensors is possible to determine the "time delay" of the sensor under test.

Noise Analysis Method Description - The methodology basically consists on using a noise analysis based in time-series. Auto-regressive models are adjusted to the data and the impulse and step responses are then calculated from the adjusted auto-regressive coefficients.

To determine the response time of 20 RTD sensors and 36 pressure sensors from the Angra - I nuclear power plant, computer based programs were developed using Labview® and Matlab®. The response time results were then compared to those obtained using direct methods showing good accordance.

Once acquired, the data from the sensors were analyzed using a Matlab® program that:

- calculates the power spectral power density (PSD) of the signal;
- calculates the auto-regressive coefficients;
- calculates the power spectral signal from the impulse and step response, using the coefficients from the auto-regressive model.
- from the step response one can calculate the time constant of the sensor.

2) Plant condition monitoring and fault detection.

During the last two decades, *model-based fault diagnostics* methods have received increasing attention in both research and application. This approach is based on the concept of *analytical redundancy* as opposed to *physical redundancy* (hardware or parallel), that uses measurements from redundant sensors for fault diagnostics purposes. Analytical redundancy makes use of the prediction of signals generated by the mathematical model of the system being considered. These predictions are compared with the actual measurements from system sensors. The comparison is made using the residual quantities, which provide the difference between the measured signals and signals generated by the mathematical model.

In this work the models will be obtained by applying the Group Method of Data Handling (GMDH), which uses data generated for different normal conditions. These data are referenced as "fault-free database". The GMDH is based on sequential learning networks, which are networks of mathematical functions that characterize complex, nonlinear relationships in a compact and rapidly executable form. Such networks subdivide a problem into manageable sub-problems and then automatically apply advanced mapping techniques to solve each of these simple problems.

For the current application, data are acquired from the IEA-R1 reactor Data Acquisition System (SAD). The monitoring function is independent of the Instrumentation and Control panels indications installed in the Reactor Control Room. The SAD consists of a signal conditioning and processing module and a PC-based human-machine interface software. A total of 57 operational variables are monitored by the SAD. The monitoring system developed in this work was used to predict the IEA-R1 reactor hall dose rate, based on

three process variables: nuclear power, rod position and temperature. The details of model prediction and reactor measurement techniques will be presented in the next sections.

3) Aging study of resistance temperature detectors (RTD)

The main purpose of the study is the understanding of aging mechanisms and the correlation with RTD operational parameters such as **Self-Heating Index (SHI)**, **Time Constant** and other calibration parameters including *Callendar-Van Dussen* equation parameters α , δ and ice-point resistance (R_0) in order gather experimental information and improve safety and maintenance in our nuclear power plants.

The following tests will be used in order to monitor the aging effects in a temperature sensor. **Self-Heating Index (SHI)** The steady state temperature rise due to steady state I^2R heating of the sensor filament is used to detect changes in the sensor's overall heat transfer coefficient and consequently changes in its response time. The results are expressed in terms of the self-heating index (ohm of resistance change/watt of I^2R power). An increase in the self heating index indicates an increase in response time. The test can be implemented in a plant by connecting a test instrument at the point where the sensor leads are normally connected to their in-plant transmitter. The test data are analyzed by constructing a plot of resistance change versus I^2R power and measuring the slope. The measurement of SHI does not require the knowledge of temperature changes of sensor, and for this reason SHI is a more acceptable parameter in terms of monitoring than the dissipation constant. The objective of this research is to set an experiment in order to monitor RTD temperature sensor aging. The temperature sensor will be installed in a high temperature furnace, and will be submitted at temperatures above their normal operating range (500 to 900 °C), accelerating the aging process. Experimental routines involving the loop current step response test and self-heating test will be performed to monitor the sensor performance.

4) Development of a system for monitoring and diagnosis of steam generator tubes using artificial intelligence techniques and wavelets on eddy current test signals.

The Ipen/Cnen -SP and The University of Tennessee, Knoxville (UTK) cooperation project to develop an automated diagnosis system has attained many important objectives. The original project scope is very large and includes many steps that vary from data acquisition to the final decision (to plug or not the steam generator tube). Other main features are: filtering, calibration, defect simulations, "intelligent" system training for automatic recognition and characterization of steam generator tube faults. UTK Nuclear Engineering Department has been the pioneer in the integration of different artificial intelligence methods [1,2] to get an automated and precise steam generator tube defects diagnosis.

A set of feature extraction methods is applied to different tube defect type ECT signals and each obtained feature vector is projected into a bi-dimensional map obtained by a Self-Organizing Map (SOM) neural network. These maps show different clustering distributions that depend on the importance that each of the features has to characterize the defect type. An estimate of the characterization importance of each feature is done through the comparison between the clusters average distance on those maps. This methodology allows an optimal feature extraction methods selection for the defect type classification. The set of methods consists of Linear Predictive Coding (LPC), Fourier Descriptors, Wavelet Zero-Crossings (WZC), Center of Gravity, Signal Segments, Phase, and different combinations of these feature vectors.

5) Monitoring and Automated Diagnosis of Defective Bearings Using Advanced Signal Processing.

Faults detection and classification in defective bearings has been a constant concern in rotating machinery, mostly in those which execute safety functions and in those that are part of more complex system. Although the bearings have low cost, their failure can cause damage to the machines or long maintenance periods, which can generate great economical losses. Thus, monitoring and diagnosis techniques have presented a fast evolution with the incorporation of automation techniques using artificial intelligence such as fuzzy logic and neural networks. It was initiated and presented in this paper a research project aiming the development of

automated bearings monitoring and the diagnosis system using fuzzy logic for the detection and classification of defects.

With the collaboration of the Department of Mechanical Engineering of the University of São Paulo, which supplied a database, containing defective bearings signatures obtained in a experimental setup, where it was fastened an accelerometer that measured the mechanical vibrations produced by the spheres of the ball bearing when passing by the existing defects in this ball bearing, these defects were produced in a controlled way in the outer race of rolling elements. Four types of controlled defects were introduced: scratched (S), minor corrosion (C1), severe corrosion (C2) and pitting (P). Studies were performed on this database and although the results are preliminary, all the existing defects were successfully identified.

6) Sub cooled boiling detection using signals spectral analysis.

The optimization of heat exchangers, steam generators and nuclear cores has the common objectives of minimizing flaws occurrence with increasing thermal efficiency and power densities thus reducing costs. These equipments are extensively used in process and power plants and increasing efficiency and durability will certainly represent an important capital and energy saving for the modern industry. In the subcooled boiling regime where the heat transfer coefficient is highly improved, bubbles implosion with the resulting pressure pulses are accompanied by acoustic noise and sometimes structural vibrations. These three phenomena can be detected by the use of dynamic sensors (accelerometers, piezoresistive pressure sensors) and spectral, correlation and wavelet analysis of their signals. Detection of the onset of subcooled boiling will allow operation of heat transfer equipment to operate in this regimen.

In order to extend the experiments conducted at the Institut für Kerntechnik und zertörungsfreie Prüfverfahren von Hannover-IKPH- Germany-[2,3,4] to analyze the effects on subcooled boiling with parallel heated rods, a new circuit and test section named Apollo, Figs. 1 and 2, was designed and constructed at IPEN, São Paulo. This experimental facility is already operational and the experiments should be initiated soon. The circuit is typical for this type of application and its main operational characteristics are: pressure ranging from atmospheric up to 1.5 bar, maximum heat flux up to 200 W/cm² and maximum demineralized water flow rate of 20 m³/hr. The circuit control system allows fine control of the inlet temperature, heat flux and flow rate.

7) Design and construction of an isolating amplifier for noise analysis.

The use of noise analysis techniques allows us to obtain information about the dynamic process of the components in a given plant. Particularly, in a nuclear reactor, those techniques have great usefulness, once its primary circuit is not usually accessible, due to the nuclear radiation and to the financial burden of an unnecessary shut down. The objective of this work is the design and the construction of a modern isolating amplifier to perform dynamic signals conditioning of the signals with noise components coming from process sensors of a nuclear reactor. Since this equipment is specific for such purpose, we should have in mind some very peculiar characteristics, not found in similar equipments commercially available.

Firstly, this equipment should have a high isolation impedance in order not to interfere in the sensor electronic chain. Because an interference could jeopardize the operation, or even, to shut down unnecessarily the reactor if the signal is part of the plant protection system. Another fundamental amplifier characteristic is to have an excellent signal/noise ratio. The AC component of the signal being analyzed which in general has very low amplitudes should be amplified. This amplification should avoid influence introduced by the noise present in the input stages of electronic equipments where this problem is important. For this, we should design the equipment by choosing components as resistors, capacitors and operational amplifiers that are used in this stage which generate little noise like thermal noise and shot noise. Finally, the design should operate in frequencies range of the order of 0,01Hz-10KHz and amplitudes ranging from mV to V. We intend to apply this equipment for a wide range of operational conditions, facilitating the processing of signals from several different types of sensors and process.

8) Development of an integrated condition monitoring system for motor operated valves.

Motor-operated valves are located in almost all nuclear power plant fluid systems. The purpose of MOV's is to control the fluid flow in a system by opening, closing or partially obstructing the passage through itself. The readiness of nuclear plants depends strongly on the operational readiness of valves specially MOVs, they are applied extensively in control and safety-related systems. Their failures have resulted in significant maintenance efforts and occasionally, have led to the loss of safety system operational readiness. When motor-operated valve degradations are noted during plant operation, they led significant operational or economic impact on overall power plant production. In recent years MOV's have received considerable attention by the nuclear power industry, due the tremendous amount of money being spent on valve safety system and valve control system maintenance. Therefore, an effective MOV maintenance program is a critical element for the safe and continued operation of the nuclear power plants. Non intrusive diagnosis methods have provided the ability to detect mal functions in plant components during normal operation.

The aim of this work ongoing at the IPEN consists of development of an integrated condition monitoring system for motor-operated valves for Brazilian Nuclear Power Plants. It is a PC based measurement and diagnostic system for detecting valves failures in advance. The methodology used in this research is the motor power signature analysis during open-to-close and close-to-open stroke time. Once the baseline diagnostic measurements of the MOV is taken, it is possible to detect long-term deviations during valve lifetime. Motor power monitoring is considered to be the most promising for a real predictive maintenance approach. The system is still in development, the tests and validation will be performed using MOV data from operating pressurized water reactor plants and data from bench tests with MOV known anomalies.

9) On-line detection, localization and sizing of faults in pressurized tubes using array of sensors and higher order spectral analysis of acoustic waves.

The first target application for this system will be the steam generator tubes. The presently installed generation of PWR reactors presented severe damage history in their steam generator tubes caused either by improper design, material/water chemistry specification or manufacturing/operational procedures. In many cases these faults resulted in the full replacement of the equipment resulting in a heavy burden on the capital cost of the installation. After several design and operational upgrades the replaced steam generators still pose a potential failure problem since no significant operational history is available.

Such a monitoring system would also give the designers flexibility to reduce the design margins regarding the number of reserve tubes to be plugged, in case of faulty condition allowing for better thermal efficiency. In addition, the use of an on-line monitoring system will allow a more direct and precise safety action to be taken, such as the isolation of the faulty steam generator, thus improving the safety of the plant.

The objective of this project is the investigation, development and verification of an on-line automated detection, localization and sizing of defects, applied to steam generator tubes using an array of sensors and higher order spectral analysis of acoustic wave signals. This steam generator tubes fault detection system is based on the phenomenon of acoustic wave propagation in the material. The acoustic wave can be produced either by external calibrated point sources or by a leakage itself. The pressure gradient at the faulty position will produce a sonic jet, whose vortices will induce an acoustic wave propagating along the tube. The controlled externally induced point source excitations can be used to calibrate the system and may be used to monitor incipient defects even before a leakage occurs.

An array of sensors will be used to capture the propagating waves. The signals will be analyzed using the beam forming technique in conjunction with direction of arrival and time delay estimation algorithms based on higher order spectral analysis (HOSA). Basically, these techniques consider the third and fourth order cumulants and their correspondent spectra allowing for selective suppression of Gaussian noise and non-Gaussian noise, reconstruction of phase and magnitude response, and characterization of non-linearities in the signal waveforms. These methods are not new, and are being successfully used in other engineering applications such as sonar, seismology, telecommunication, biomedicine, and others.