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Predictive maintenance in nuclear power plants through online monitoring

The nuclear power industry is working to reduce generation costs by adopting condition-based maintenance strategies and automating testing activities. These developments have stimulated great interest in online monitoring (OLM) technologies and new diagnostic and prognostic methods to anticipate, identify, and resolve equipment and process problems and ensure plant safety, efficiency, and immunity to accidents. This paper provides examples of these technologies with particular emphasis on a number of key OLM applications: detecting sensing-line blockages, testing the response time of pressure transmitters, monitoring the calibration of pressure transmitters online, cross-calibrating temperature sensors in situ, assessing equipment condition, and performing predictive maintenance of reactor internals.

Keywords: Nuclear power plants; Noise analysis; Sensor response time testing; Sensing-line blockages; Calibration monitoring; Reactor diagnostics.

Х. М. Хашеміан

Прогнозне технічне обслуговування АЕС із застосуванням оперативного контролю

У сфері атомної енергетики проводяться роботи зі зниження витрат на виробництво електроенергії шляхом прийняття стратегій технічного обслуговування за поточним станом обладнання та автоматизації випробувань. Ці роботи викликали великий інтерес до технологій оперативного контролю та нових методів діагностування та прогнозування для виявлення і вирішення проблем, пов'язаних з устаткуванням і технологічними процесами, а також забезпечення безпеки, ефективності та стійкості до аварій. У статті наводяться приклади технологій, заснованих на ряді ключових напрямків застосування оперативного контролю: виявлення блокування вимірювальних ліній, тестування часу відклику датчиків тиску, контроль калібрування перетворювачів тиску, взаємне калібрування температурних датчиків на місці, оцінка стану обладнання та виконання профілактичного обслуговування внутрішньокорпусних пристроїв реактора.

Ключові слова: атомні електростанції, аналіз перешкод, тестування часу відклику датчиків, блокування вимірювальних ліній, контроль калібрування, діагностика реактора.

The term online monitoring (OLM) describes methods, such as the noise analysis technique, for evaluating the health and reliability of nuclear plant sensors, processes, and equipment from data acquired while the plant is operating. Although OLM technologies typically apply to all types of nuclear power reactors, this paper uses pressurized water reactors (PWRs) as the reference plant since they are the type most commonly used in the Western hemisphere. To control the PWR plant and protect its safety, several different kinds of sensors are employed to measure the process parameters (see Table 1). Figure 1 shows a simplified schematic of the primary loop of a PWR plant and its important sensors. Depending on the plant design and manufacturer, a PWR plant has 2 to 4 primary coolant loops; however, Russian PWRs (called VVERs or WWERs) have up to 6 loops. The normal output of these sensors can be used both to establish the health and condition of the plant and sometimes to verify the performance of the sensors themselves.

Table 1. Typical Population of Important Sensors in Pressurized Water Plants

Sensor	Measurement	Typical Population in a Reactor
RTDs [a]	Temperature	16 e 60
CETs [b]	Temperature	50 e 100
Pressure Transmitters [c]	Pressure, Level, and Flow	500 e 2500
Neutron Detectors [d]	Neutron Flux	10 e 20

[a] Resistance Temperature Detectors
[b] Core-Exit Thermocouples
[c] Including Differential-Pressure Transmitters
[d] Ex-core and Some In-core Neutron Detectors

Figure 2 shows the output of a process sensor as a function of time during plant operation. Normally, while the plant is operating, the output of the sensor will have a steady-state value that corresponds to the process parameter indicated by the sensor. This steady-state value is often referred to as the static component or DC value. Figure 2 also shows a magnified portion of the sensor's output signal to illustrate that, in addition to the static component, a small fluctuating signal is naturally present on the sensor output. The fluctuating signal, which is known as the signal's dynamic or AC component, derives from inherent fluctuations in the process parameter as a result of turbulence, random flux, random heat transfer, vibration, and other effects. It has long been known that the condition of a nuclear power plant can be effectively monitored by analyzing these small fluctuations in the process variables, such as reactivity coefficients, vibration amplitudes, and response times, around their stationary value. This technique, commonly known as noise analysis, noise diagnostics, or reactor diagnostics, makes it possible to discover the abnormal state of the system, which registers either as a shift of these parameters into non-permitted regions or the appearance of a changed structure of the noise signatures, usually the frequency spectra. The advantage of the noise analysis technique is that it non-intrusively measures process variables under normal operation without requiring any external perturbation.

One idiosyncrasy of the noise analysis technique is that a change in the measured signal characteristics may be caused either by a change in the transfer properties of the system or by a change in the driving force; that is, the fluctuation of the parameter that induces the measured noise. Hence, performing

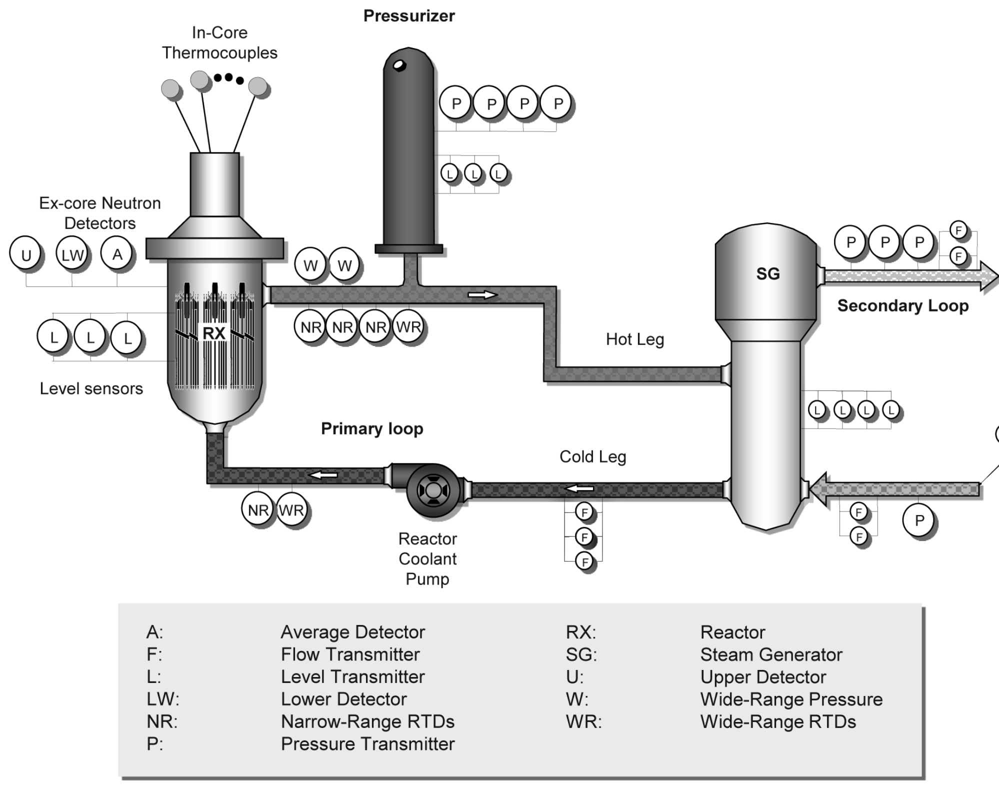


Figure 1. Primary Loop of a Pressurized Water Reactor (PWR)

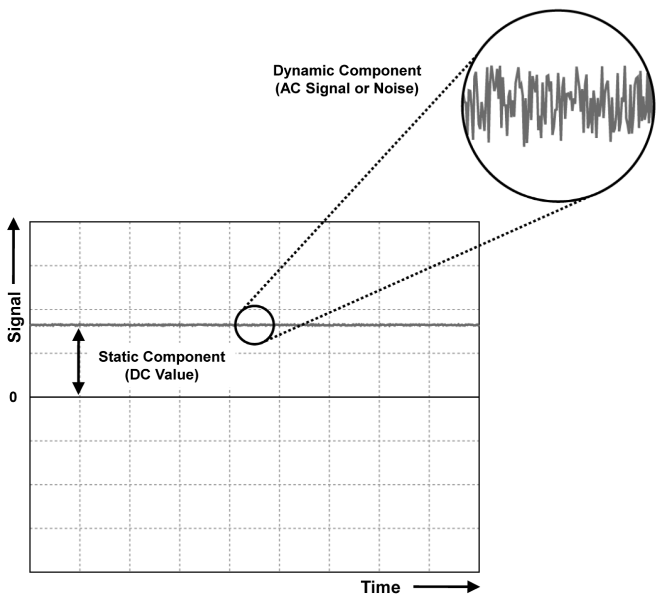


Figure 2. Normal Output of a Process Sensor with Illustration of the DC and AC Components of the Output

a proper diagnosis requires sufficient expert knowledge to choose the appropriate model on which the diagnostic algorithm is based. The noise analysis technique also involves an additional ambiguity: deteriorating sensor characteristics can change the measured noise signature. In the Three Mile Island accident in 1979, for example, a role in the accident sequence was played by a failed sensor and the control room personnel's inability to realize its failure. Sensor malfunction, or just de-calibration, can also occur under much less dramatic circumstances, through fouling, drift, response time degradation, and aging. As it turns out, noise analysis can be used even for sensor health analysis, by differentiating between sensor degradation/failure and system malfunction/anomaly.

Because the static (DC) and dynamic (AC) components of the sensor output each contain different information about the process being measured, they can be used for a wide range of monitoring applications. For example, applications that monitor for gradual changes in the process over the fuel cycle, such as sensor calibration monitoring, make use of the static component. In contrast, applications that monitor fast-changing events, such as core barrel motion, use the information in the dynamic component, which provides signal bandwidth information. Figure 3 illustrates how existing data from process sensors is used for these applications. Note that in this figure, the static data is analyzed using empirical and physical modeling and averaging techniques involving multiple signals, while dynamic data analysis entails time domain and frequency domain analysis based on single signals or pairs of signals. For example, the dynamic response time of a nuclear plant pressure transmitter is identified by Fast Fourier Transform (FFT) of the noise signal. The FFT yields the auto power spectral density (APSD) of the noise data from which the transmitter response time is calculated. In applications where pairs of signals are used (e.g., core barrel vibration measurements), the cross power spectral density (CPSD), the phase, and the coherence data are calculated to distinguish the vibration characteristics of various constituents of the reactor internal.

The types of OLM applications used in nuclear power plants are in large part determined by the sampling rates available for data acquisition. Static OLM applications, such as resistance temperature detector (RTD) cross-calibration and online calibration monitoring of pressure transmitters, typically require sampling rates up to 1 Hz, while dynamic OLM applications such as sensor response time testing use data sampled in the 1 kHz range. Other, high-frequency OLM applications, such as measuring the vibration of rotating equipment and monitoring loose parts, may use data sampled at up to 100 kHz. Figure 4 shows examples of OLM applications

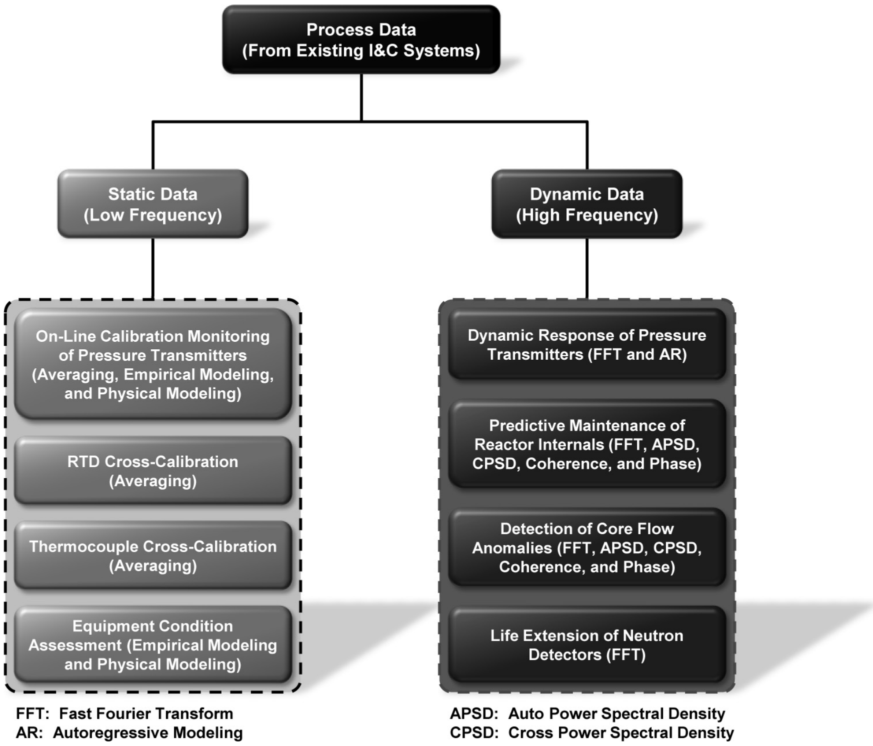


Figure 3. Online Monitoring Application of Static and Dynamic Data Analysis in this Paper

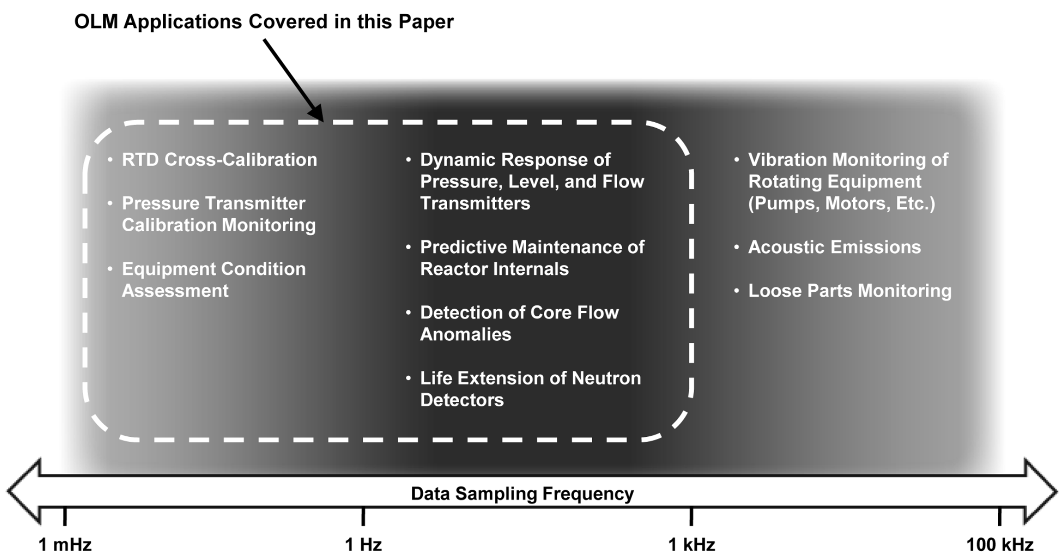


Figure 4. Online Monitoring Applications Versus Sampling Frequency

that can be used in nuclear power plants, with their range of data sampling frequency. For low-frequency (DC) data analysis, averaging techniques are typically used for redundant sensors, and empirical and physical modeling techniques are used for non-redundant sensors.

Because I&C sensors that measure temperature, pressure, level, flow, and neutron flux up to data sampling frequencies of around 1 kHz represent the majority of measurement devices in nuclear power plants, focusing this paper on the OLM applications that monitor these sensors will show to best advantage the potential benefits of OLM for nuclear plants. Other OLM applications, such as measuring the vibration of rotating equipment and monitoring loose parts, which primarily rely on high-frequency acquisition of data from accelerometers, are not discussed in this paper because they don’t acquire data from the existing process sensors of the plant.

OLM APPLICATIONS IN NUCLEAR POWER PLANTS

The success of the noise analysis technique in nuclear power plant applications stimulated the industry to examine the feasibility of implementing an online monitoring (OLM) system that incorporates the noise analysis technique in both the current and next generation of nuclear reactors for the purpose of dynamically testing sensors, measuring the vibration of reactor internals, and performing a variety of diagnostic applications. This OLM system will also give plants the capability to verify the calibration of pressure, level, and flow transmitters as well as RTDs and thermocouples. The system will have built-in signal validation, noise analysis, and OLM algorithms that will enable nuclear power plants to check for: (1) calibration and response time of process instruments; (2) identify sensing-line blockages; (3) monitor the reactor coolant flow, and (4) alert the reactor operator of excessive vibration of reactor internals.

Such an OLM system can provide plants with the information they need to evaluate I&C sensors by providing applications that identify drifting instruments, alert plant personnel of unusual process conditions, and predict impending failures of plant equipment. Moreover, operating nuclear power plants can use OLM technologies to improve their efficiency. For example, nuclear power plants are required to calibrate important I&C instruments once every fuel cycle. This requirement dates back 40 years to when commercial nuclear power plants were first licensed to begin operation. Based on calibration data accumulated over these four decades, it has been determined that the calibration of some instruments, such as pressure transmitters, does not drift enough to warrant calibrating all transmitters as often as once every fuel cycle. OLM allows calibration efforts to be focused on the instruments that have drifted out of tolerance, thereby saving plants a significant amount of the time and manpower.

Online Detection of Sensing-line Blockages

Chief among applications of noise analysis in nuclear power plants is detecting sensing- line blockages. Sensing-lines (also called impulse lines) are small diameter tubes that bring the pressure signal from the process to the pressure sensor. Depending on the application and the type of plant, pressure sensing-lines can be as long as 300 meters or as short as 10 meters. The isolation valves, root valves, and bends along their length make them susceptible to blockages from residues in the reactor coolant, failure of the isolation valves, and other problems. Sensing- line blockages are a recurring problem in PWRs, boiling water reactors (BWRs), and essentially all water-cooled nuclear power plants. They are an inherent phenomenon that causes the sensing-lines of nuclear plant pressure transmitters to clog up with sludge, boron, magnetite, and other contaminants. Typically, nuclear plants purge the important sensing-lines with nitrogen or backfill the lines periodically to clear any blockages. This procedure is, of course, time consuming and radiation intensive, and more importantly, not always effective in eliminating blockages. Furthermore, with the exception of noise analysis, no way exists to know ahead of time which sensing-lines may be blocked. Also, unless the noise analysis technique is used, it's not possible after purging or back filling a sensing-line to verify that the line has been cleared.

Figure 5 shows the cutaway of a partially blocked sensing-line of a nuclear power plant pressure transmitter. It's clear from the figure that this blockage can reduce the flow path in this sensing-line by about 40 percent. A blockage like this hampers the dynamic response of the pressure sensor at the end of the sensing-line. In particular, depending on the design characteristics of the pressure transmitter, a sensing-line blockage like this can cause the response time of the affected pressure transmitter to increase by an order of magnitude. The degree of increase in the dynamic response depends on the "compliance" of the pressure transmitter. Compliance is a pressure transmitter design parameter that relates to the physical displacement of the sensing element of the transmitter per unit of input pressure. Some transmitters, such as those with sensing elements made of "bellows," have a large compliance and are therefore affected strongly by sensing-line blockages. On the other hand, transmitters with sensing elements made of stiff diaphragms have smaller compliances and are therefore less affected by sensing-line blockages.

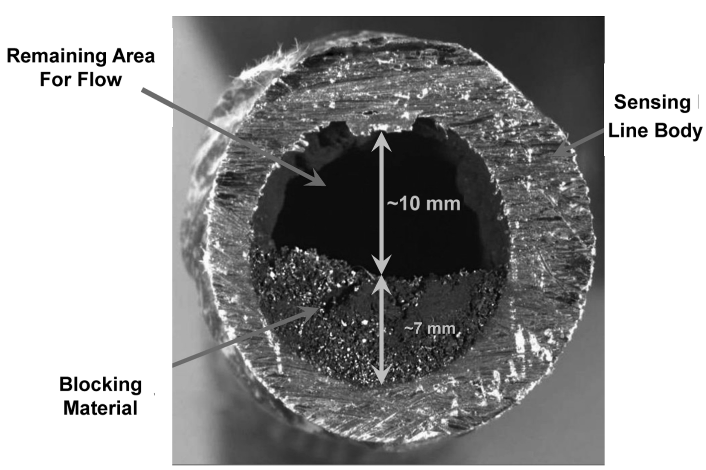
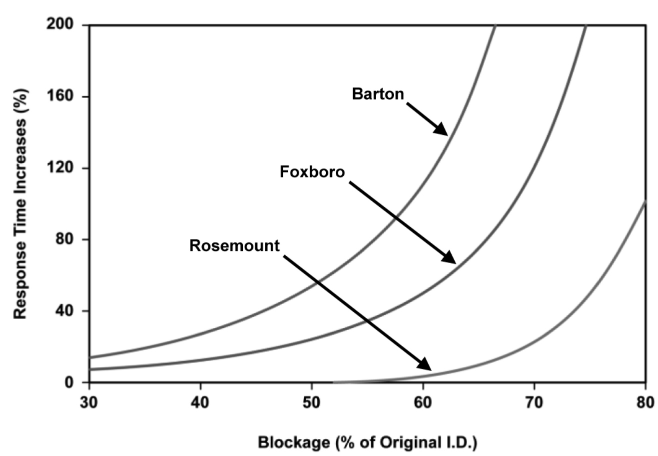


Figure 5. Photograph of a Nuclear Plant Sensing-line with a Partial Blockage

The effect of compliance on the dynamic response of a pressure transmitter was revealed in a research project performed by the author for the U.S. Nuclear Regulatory Commission (NRC) in the early 1990s (Hashemian, 1993). The goal of the project was to characterize the effects of normal aging on the performance of nuclear plant pressure transmitters by illustrating the effect of compliance on the response time of representative nuclear-grade pressure transmitters from three manufacturers: Barton, Foxboro, and Rosemount (see Figure 6). The data in Figure 6 is from laboratory tests measuring the response time of the transmitters using a pressure ramp signal.

A significantly blocked sensing-line can render the pressure sensor essentially useless or even dangerous. The danger here is that, due to a total blockage, the operating pressure may get locked in the transmitter and cause its indication to appear normal. Then, when the pressure changes, the transmitter will not respond and will continue to show the locked-in pressure, which will confuse the reactor operators and potentially pose a risk to the safety of the plant.

If a blocked pressure transmitter happens to be a part of a redundant safety channel, it can trip the plant during a transient. More specifically, the indication of a blocked transmitter will obviously not match the other redundant channels, creating a mismatch that could trigger a reactor trip.



Transmitter Manufacturer	Model Number	Compliance (cm^3 / Bar)
Barton	764	9.51
Foxboro	E13DM	0.12
Rosemount	1153RC7	0.01

Figure 6. Research Results on the Effect of Sensing-line Blockages onResponse Time of Nuclear Plant Pressure Transmitters

In fact, this problem has occurred in France where partial blockages in flow transmitters caused two French PWRs to trip during load flowing episodes in the mid-1980s (Meuwisse and Puyal, 1987).

Some sensing-line blockages are so severe that the sensing-line has to be drilled to clear the blockage. This type of problem is the reason why measuring the response time of pressure transmitters is so important and why it is so surprising that even today, some nuclear power plants measure the response time of their safety-related pressure transmitters using conventional procedures that exclude the sensing-lines. These plants typically use a hydraulic pressure generator to input a pressure signal to the transmitter and measure its response time. In doing this, the sensor is isolated from the sensing-lines. The research work documented in the U.S. Nuclear Regulatory Commission report NUREG/CR-5851 uncovered this flaw in the maintenance of nuclear plant pressure transmitters. As a result, many plants have recognized that they must measure the response time of both their pressure transmitters and their sensing-lines. These plants have accordingly switched to the noise analysis procedure to verify the dynamic characteristics of their pressure sensing systems.

Response Time Testing of Pressure Transmitters

Pressure, level, and flow transmitters in nuclear power plants behave like filters on the natural plant fluctuations that are presented to their inputs. That is, if one assumes that the input to the transmitter exhibits wide-band frequency characteristics (which is typically the case for nuclear power plant fluctuations), information about the sensor itself can be inferred by measuring the transmitter output. This is the basis of the noise analysis technique that is used to determine the dynamic response of pressure, level, and flow transmitters in nuclear power plants (Thie, 1981).

The noise analysis technique is used to remotely measure sensor response time from the control room area while the plant is online. These measurements do not require the sensors to be disconnected from the plant instrumentation or removed from service for the tests. That is, the tests are passive and do not cause any disturbance to plant operation. This reduces test time and helps to reduce radiation exposure of the test personnel who would otherwise have to enter the reactor containment to make the response time measurements.

Dynamic response analysis is performed in the frequency domain and/or time domain, and is based on the assumption that the dynamic characteristics of the transmitters are linear and that the input noise signal (i.e., the process fluctuations) has proper spectral characteristics. Frequency domain and time domain analyses are two different methods for determining the response time of transmitters. It is usually helpful to analyze the data with both methods and average the results, excluding any outliers.

In frequency domain analysis, the APSD of the signal is generated first, usually using an FFT algorithm. After the APSD is obtained, a mathematical function (model) that is appropriate for the transmitter under test is fit to the APSD to yield the model parameters. These parameters are then used to calculate the dynamic response of the transmitter. The dynamic response of the transmitter can then be analyzed to determine the response time of the transmitter in situ. Figure 7 shows an example of process noise that enters a pressure transmitter and is subsequently filtered by the transmitter. The response time of the transmitter can be inferred from the APSD with the proper analysis tools.

Under normal plant conditions, the APSDs of nuclear plant pressure transmitters have characteristic shapes that can be baselined and compared with the APSDs of similar transmitters operating under the same process conditions. Figure 8 shows examples of a few typical nuclear plant APSDs for steam generator level, reactor water clean-up flow, and pressurizer pressure transmitters.

Through laboratory experiments, the noise analysis technique was validated for in-situ response time testing of pressure transmitters. This validation work involved directly measuring response time and then using the noise analysis technique to compare the ramp input signals with the response time results. Table 2 shows representative results of this validation work for seven different transmitters from various manufacturers of nuclear-grade pressure transmitters. For each transmitter, the results of the direct measurement of response time (ramp test) were compared with the results of the noise analysis test; the difference between the two results is shown in Table 2. The details of the validation of the noise analysis technique for response time testing of nuclear plant pressure transmitters are documented in a comprehensive report, "Long Term Performance and Aging Characteristics of Nuclear Plant Pressure Transmitters," published by the NRC in March 1993 as NUREG/CR-5851 (Hashemian, 1993).

Online Calibration Monitoring of Pressure Transmitters

Online calibration monitoring refers to monitoring the normal output of nuclear plant pressure transmitters during plant operation and then comparing this data with an estimate of the process parameter that the transmitter is measuring. At most plants, the plant computer contains all the data with an estimate of the process parameter that the transmitter is measuring. At most plants, the plant computer contains all the data that is needed to verify the calibration of pressure transmitters, including data from plant startup and shutdown periods used to verify the calibration of instruments over their operating range. Using the online calibration method, transmitter outputs are monitored during process operation to identify drift. If drift is identified and is significant, the transmitter is scheduled for a calibration during an ensuing outage. On the other hand, if the transmitter drift is insignificant, no calibration is performed for typically as long as eight years. This eight year period is based on a two year operating cycle and a redundancy level

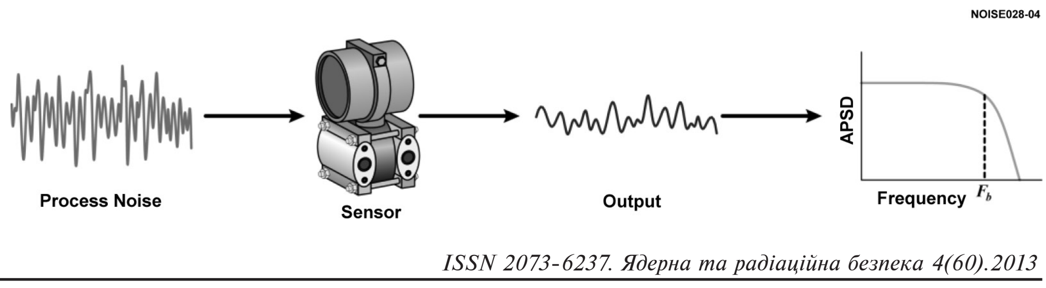


Figure 7. Example of a Pressure Transmitter Filtering the Process Noise

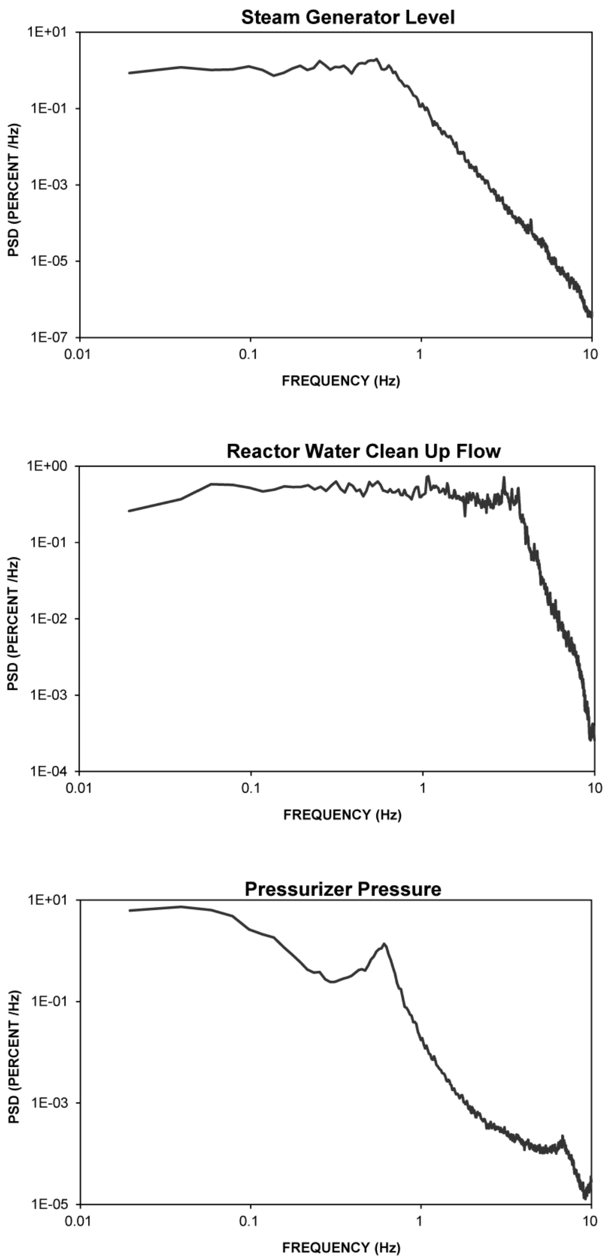


Figure 8. Examples of Auto Power Spectral Densities of Nuclear Plant Pressure Transmitters

Table 2. Representative Results of Laboratory Validation of Noise Analysis Technique for Response Time Testing of Nuclear-grade Pressure Transmitters

Number	**Response Time (s)**		
	Ramp Test	**Noise Analysis**	**Difference**
Barton			
1	0.05	0.09	0.04
2	0.17	0.20	0.03
3	0.17	0.25	0.08
4	0.12	0.15	0.03
5	0.12	0.20	0.08
6	0.11	0.15	0.04
7	0.12	0.18	0.06
Foxboro			
1	0.13	0.16	0.03
2	0.21	0.18	-0.03
3	0.16	0.13	-0.03
4	0.09	0.12	0.03
5	0.29	0.30	0.01
6	0.25	0.15	-0.10
7	0.28	0.25	-0.03
Rosemount			
1	0.05	0.06	0.01
2	0.32	0.28	-0.04
3	0.07	0.05	-0.02
4	0.10	0.07	-0.03
5	0.11	0.08	-0.03
6	0.09	0.08	-0.01
7	0.09	0.09	0.00
Other manufacturers			
1	0.15	0.15	0.00
2	0.21	0.18	-0.03
3	0.02	0.08	0.06
4	0.03	0.07	0.04
5	0.08	0.11	0.03
6	0.15	0.27	0.12
7	0.33	0.37	0.04

of four transmitters. In this application, OLM is not a substitute for traditional calibration of pressure transmitters; rather, it is a means for determining when to schedule a traditional calibration for a pressure transmitter.

Reviews of the calibration histories of process instruments in nuclear power plants have shown that high-quality instruments, such as nuclear-grade pressure transmitters, typically maintain their calibration for more than a fuel cycle of about two years and do not, therefore, need to be calibrated as often (Hashemian, 1995, Hashemian 1998). The validity of the OLM approach in verifying the calibration of nuclear plant pressure transmitters was researched in the mid-1990s using both theoretical work and laboratory experiments and in-plant trials. The results are documented in a comprehensive NRC report published in November 1995 as NUREG/CR-6343: “Online Testing of Calibration of Process Instrumentation Channels in Nuclear Power Plants” (Hashemian, 1995).

To perform online calibration monitoring, the output of redundant sensors is averaged. The average value, called the process estimate, is then used as a reference to determine the deviation of each sensor from the average of the redundant sensors and to identify the outliers. For non-redundant sensors, averaging obviously cannot be used to determine a reference value. Therefore, if there is insufficient instrument redundancy, the process estimate for calibration monitoring is determined by analytical modeling of the process. Both empirical and physical modeling techniques are used in this application, although empirical models are preferred because they can be adapted to various processes and different operational envelopes.

In particular, empirical modeling techniques involving neural networks have been researched for online calibration monitoring applications in nuclear power plants (as documented in NUREG/CR-6343). Although neural networks are effective, the nuclear industry does not currently favor this application because of difficulties in determining the uncertainty of their results. As such, other methods, such as averaging or analytical modeling techniques, have been developed for monitoring the calibration of pressure transmitters.

In-situ Cross Calibration of Temperature Sensors

PWR plants often employ 20 to 40 RTDs to measure the fluid temperature in the reactor coolant system. The temperatures measured by the RTDs are used by the plant operators for process control and to assess the operational status and safety of the plant. As such, the calibration of the RTDs is normally evaluated at least once every refueling cycle. Each RTD must meet specific accuracy requirements in order for the plant to continue to produce power according to its design specifications. There are also about 50 core-exit thermocouples (CETs) in PWRs to provide an additional way to monitor reactor coolant temperature. Accuracy for CETs is not as important as for RTDs because CETs are used mostly for temperature monitoring. Nevertheless, CETs are sometimes cross calibrated against RTDs to ensure that their output is reliable.

In each loop of a PWR plant and for each core quadrant, redundant RTDs and CETs are used to minimize the probability of failure of any one RTD or CET affecting the safety of the plant. This redundancy of temperature sensors is the basis for a method of evaluating the calibration of RTDs and CETs called cross calibration. In cross calibration, redundant temperature measurements are averaged to produce an estimate of the true process temperature. The result of the averaging is referred to as the process estimate. The measurements of each individual RTD and CET are then subtracted from the process estimate to produce the cross- calibration results in terms of the deviation of each RTD from the average of all redundant RTDs (less any outliers). If the deviations from the process estimate of an RTD or CET are within acceptable limits, the sensor is considered in calibration. However, if the deviation exceeds the acceptance limits, the sensor is considered out of calibration and its use for plant operation is evaluated.

Traditionally, cross-calibration data has been acquired using data acquisition equipment that is temporarily connected to test points in the plant instrumentation cabinets. The traditional cross-calibration method, while highly accurate, causes the plant to lose indication when the data is being acquired, and costs the plant time during shutdown and/or startup to restore the temperature indications. Now, with new and more advanced plant computers, RTD and CET measurements can be collected in the plant computer, which provides a centralized location for monitoring and storing the measurements. Using online data from the plant computer for cross calibration can save plants startup and shutdown time, while producing results that are comparable to the traditional method.

Equipment Condition Assessment

Static analysis methods may be used for other purposes besides evaluating the health of individual sensors as in online cross-calibration and transmitter calibration monitoring. Equipment condition assessment (ECA) applications take the idea of online calibration monitoring a step further by monitoring for abnormal behavior in a group of sensors. An example of ECA is illustrated in Figure 9, which shows a simplified diagram of a typical chemical and volume control system (CVCS) in a PWR. The primary functions of a typical CVCS in a PWR are:

1. Controlling the volume of primary coolant in the reactor coolant system (RCS)
2. Controlling chemistry and boron concentration in the RCS
3. Supplying seal water to the reactor coolant pumps (RCPs)

Several transmitters are typically used to monitor various parameters related to the operation of the CVCS. Figure 9 highlights the normal operation of a few of these parameters:

1. Charging Flow — measures the flow rate of the coolant being provided from the volume control tank (VCT) to the RCS and RCP seals
2. Reactor Coolant Pump Seal Injection Flow — measures the flow rate of the coolant provided to the RCP seals
3. Seal Return Flow — measures the flow rate of the coolant returned to the VCT from the RCP seal injection
4. Letdown Flow — measures the flow rate of the reactor coolant as it leaves the RCS and enters the VCT

During normal operation, the measurements of these parameters will fluctuate slightly, but should remain at a consistent relative level. However, in abnormal conditions such as a RCP seal leak, some parameters may exhibit upward or downward trends, indicating a problem in the plant. For example, Figure 10 shows the four flow signals mentioned above during normal operation of a PWR plant (the actual flow rates are scaled to simplify this example). As Figure 10 shows, the flows remain at relatively constant rates relative to one another.

Figure 11 shows how these flow signals may appear at the onset of a RCP seal leak in this PWR plant. In this example, the onset of the RCP seal leak is first indicated by a downward trend in the seal return flow measured at time T1. This is followed by an increase in charging pump flow at time T2 as the charging pump compensates for the loss of coolant due to the RCP seal leak.

Of course, an abnormal trend in an individual parameter such as seal return flow could mean that the sensor is degrading; however, abnormalities in related parameters that occur close together in time are more likely to indicate the onset of a system or equipment problem. Early warning of these types of failures is thus the key benefit of ECA. More specifically, early warning of impending equipment failures can provide nuclear plants with increased plant safety through early recognition of equipment failures and reduced downtime stemming from timely repair of affected equipment.

Predictive Maintenance of Reactor Internals

A research project published in NUREG/CR-5501 (June 1998), "Advanced Instrumentation and Maintenance Technologies for Nuclear Power Plants," investigated such OLM applications as noise analysis for measuring the vibration of reactor internals and other components such as RCPs (Hashemian, 1998). Typically, vibration sensors (e.g., accelerometers) are located on the top and bottom of the reactor vessel to sound an alarm in case the main components of the reactor system vibrate excessively. However, neutron detectors have proved to be more sensitive than accelerometers in measuring the vibration of the reactor vessel and its internals. This is because the frequency of vibration of reactor internals is normally below 30 Hz, which is easier to resolve using neutron detectors than accelerometers. Accelerometers are more suited for monitoring higher-frequency vibrations.

Figure 12 shows the APSD of the neutron signal from an ex-core neutron detector (NI-42) in a PWR plant. This APSD contains the vibration signatures (i.e., amplitude and frequency) of the reactor components, including the reactor vessel, core barrel, fuel assemblies, thermal shield, and so on. It even contains, at 25 Hz, the signature of the RCP rotating at 1,500 revolutions

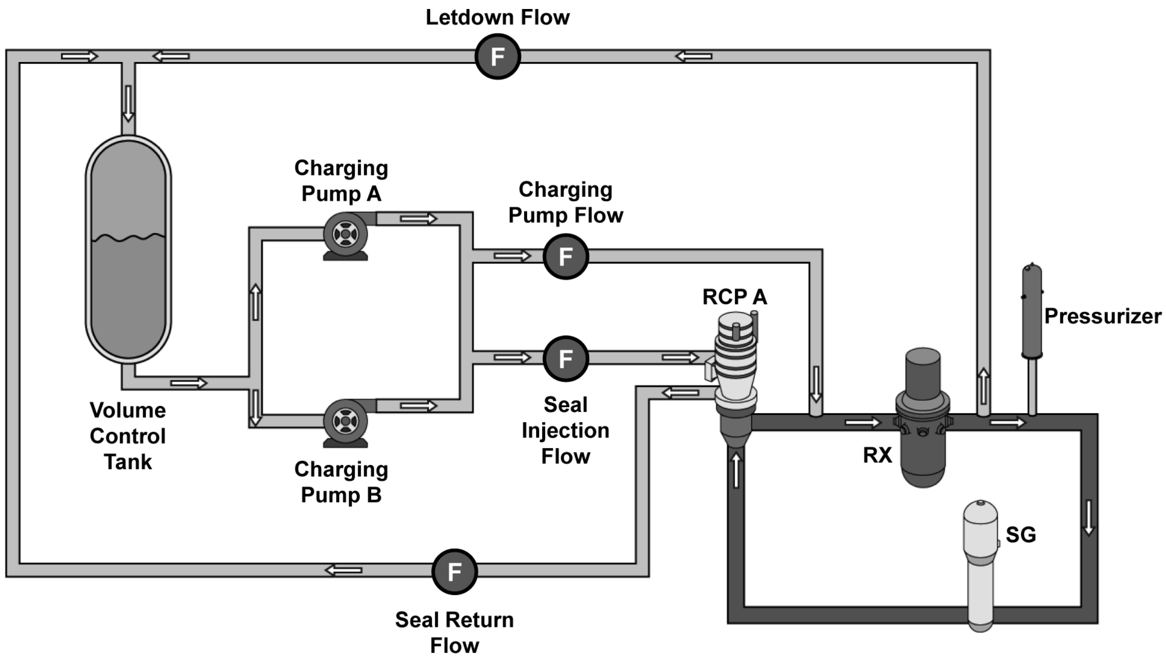


Figure 9. Simplified Diagram of Chemical and Volume Control System Components

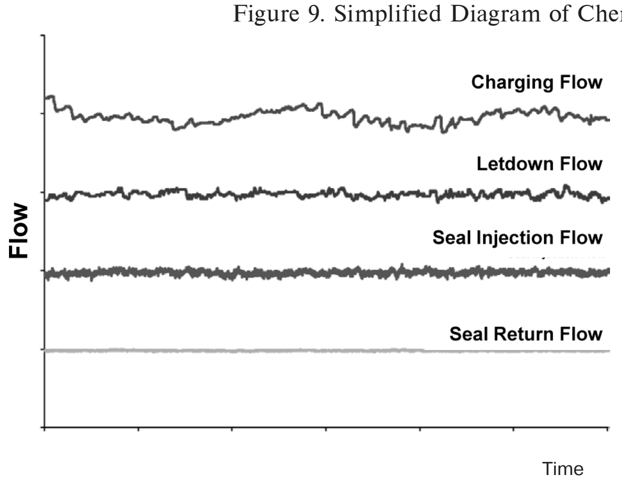


Figure 10. Normal Operation of Chemical and Volume Control System Flow Parameters

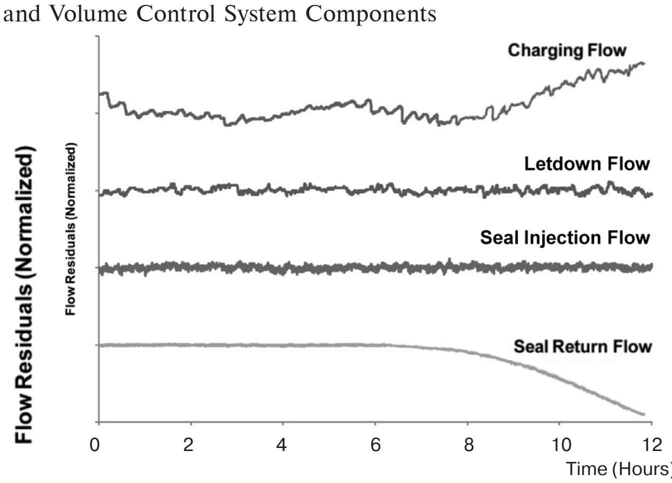


Figure 11. Chemical and Volume Control System Flow Parameters at the Onset of a Reactor Coolant Pump Seal Leak

per minute, which corresponds to 25 Hz. These signatures can be trended to identify the onset of aging degradation, which can damage the reactor internals. The neutron detection approach has been recognized as a predictive maintenance tool that can help plants guard against vibration-induced mishaps that may be encountered as plants age, making them more vulnerable to challenges to their structural integrity.

Over the last ten years, numerous plants have begun programs to measure reactor internal vibration using neutron noise analysis and then trend the results so as to identify changes and signs of degradation. Table 3 shows average values for the resonant frequency of vibration of reactor internals of PWR plants. The resonant frequency of the RCP also shows up on the neutron noise signal, as shown in Table 3, at 25 Hz corresponding to 1500 RPM (revolutions per minute).

The results in Table 3 are the average of neutron noise measurements made by the authors and others in 15 PWR plants around the world representing ABB, Westinghouse, Babcock and Wilcox (B&W), Areva (i.e., Framatome and Siemens), and Mitsubishi Heavy Industries (MHI) plants. The details are presented in NUREG/CR 5501 (Hashemian, 1998).

Summary and Conclusions

Over the past 40 years, an array of techniques has been developed for equipment and process condition monitoring. Because of regulatory constraints, cost of implementation, and other factors, these techniques have been used in nuclear power plants mostly on an “as-needed” basis rather than for routine condition monitoring applications. Now, with the advent of fast data acquisition technologies and proliferation of computers and advanced data processing algorithms and software packages, condition monitoring can be performed routinely and efficiently using dedicated equipment installed at the plants.

This paper reviewed a class of condition monitoring technologies that depend on data from existing process sensors during all modes of plant operation including startup, normal operating periods, and shutdown conditions. The data may be sampled continuously or periodically depending on the application. The steady-state (DC) component of the data is analyzed to identify slowly developing anomalies such as calibration changes in process sensors. The fluctuating (AC) component of the data is analyzed to determine such parameters as the response time of pressure

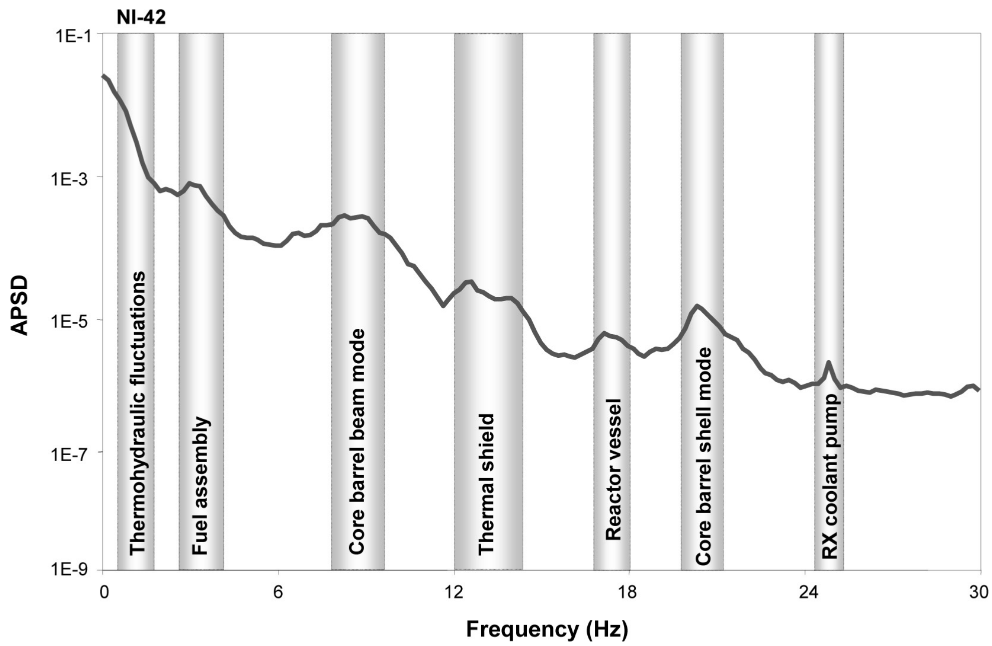


Figure 12. Auto Power Spectral Density Containing Vibration Signatures of Reactor Internals

sensors or to measure the vibrational characteristics of reactor internals, check for blockages within the reactor coolant system, identify flow anomalies, and provide other diagnostics.

The AC and DC data acquisition and signal processing techniques described in this paper can be integrated together to provide an online monitoring (OLM) system for nuclear power plants. This paper introduced the key applications of this system together with the requirements for implementing it in nuclear power plants. Such OLM systems should be built into the design of the next generation of reactors to contribute to optimized plant maintenance by providing automated measurements, condition monitoring, and diagnostics. In fact, an OLM system is currently under development by the author and his colleagues at AMS Corporation for the Small Modular Reactors (SMRs) currently under design and development in the United States. One such reactor, referred to as mPower, is currently slated to be built by Babcock and Wilcox (B&W) Company and will be constructed in Oak Ridge, Tennessee. The SMR plant will belong to Tennessee Valley Authority (TVA) which is a Federal Utility in the United States. TVA already owns and operates six nuclear power plants, and is in the process of completing its seventh conventional PWR. The SMR plant to be built in Oak Ridge, Tennessee will consist of two to four modules of about 200 MWe power, to be completed by 2022.

Table 3. Typical Frequencies of Motion of Reactor Internals at Pressurized Water Reactor Plants

Reactor Component	Average Resonant Frequency (Hz)
Fuel Assembly	3.0
Core Barrel Beam Mode	9.7
Core Barrel Shell Mode	23.1
Thermal Shield	12.5
Reactor Vessel	18.5
Reactor Coolant Pump	25.0

Source: NUREG/CR-5501

As for the current generation of reactors, they should be retrofitted with OLM systems as utilities begin to appreciate their benefits and as regulators realize the added benefits of OLM to nuclear reactor safety.

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