COMPUTER CONTROL OF POWER IN A NUCLEAR REACTOR

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Abstract

A signal validation methodology has been simultaneously applied to the on-line fault diagnosis of neutron flux detectors, primary coolant flow and temperature sensors, and to the computer control of power in an operating nuclear reactor. A CRT display presents the validated data and sensor diagnostics. The feedback signal to the controller is a digitally processed, weighted average of several valid power sensor signals where the weights are not a priori fixed but dependent on the a posteriori probabilities of failure of individual sensors computed on the basis of past and current observations.

Introduction

Sensor redundancy is generally provided for measuring safety-related variables in nuclear power plants. If such redundancy is not adequate, the set of sensors can often be augmented by analytical measurements generated from the physical relationships that exist among the plant variables. Computer-aided signal validation schemes can be designed to exploit these redundancies for reliable information display and control of plant parameters. Signal validation techniques,[1,2,3] when combined with the memory and computational capability of a minicomputer, provide estimates of plant conditions and feedback signals for plant control that are more accurate and more reliable than those available from single sensors. In a continuing series of experiments, signal validation methodologies have been developed and demonstrated for on-line fault diagnostics, information display, sensor calibration, and measurement estimation of power-related instrumentation such as neutron flux detectors, primary coolant flow, and temperature sensors in the 5 MWt fission research reactor, MITR-II, that is operated by the Massachusetts Institute of Technology. Presently, the reactor power is being digitally controlled, under steady-state operations, via feedback of a validated estimate obtained from a set of power sensors.

The objectives of this paper are to report (1) how the (MITR-II) reactor power has been digitally controlled on-line, (2) what safety procedures have been adopted to protect against possible computer malfunction and other accidents, and (3) some potential applications of the signal validation and control techniques. While these techniques are being demonstrated on a research reactor, they are applicable to both commercial power reactors and other industrial processes.

Background of the Signal Validation Methodology

The signal validation methodology provides a unified procedure for (1) fault detection and isolation (FDI), and (2) sensor calibration and measurement estimation. The FDI decisions are made on the basis of relative consistencies among all redundant measurements; the major assumption is that a measurement is normal if it does not exceed the true value by a specified error bound. The task of sensor calibration and measurement estimation is performed on-line in the framework of the aforesaid FDI technique via sequential tests that rely on both current and past measurements.

If the process variables being measured are time-dependent and if either the redundant sensors are installed in different spatial locations (e.g., neutron flux detectors) or if analytic measurements[4] are used to supplement the sensor redundancy (e.g., thermal power balances), the measurements of the given process variable may exhibit deviations from each other after a length of time even though the sensors are functioning normally. In a nuclear reactor, these differences could be caused by changes in the spatial flux distribution, changes in moderator temperature, time delays, etc. Consequently, some of the sensors may be erroneously deleted unless they are periodically recalibrated. On the other hand, failure to isolate a degraded sensor may adversely affect the estimate of the measured variable. These difficulties can be circumvented in a multiply redundant sensor system as follows:

(1) All consistent measurements are simultaneously compensated on-line such that their residuals are minimized.

(2) The weights of individual measurements for computing the validated estimate are updated on-line on the basis of their respective a

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posteriori probabilities of failure instead of being a priori fixed.

In the event of abrupt disruptions in some sensor(s) in excess of the specified error bound(s), the respective sensor(s) are isolated by the FDI algorithm, and only the remaining sensors are calibrated and used to provide an estimate. If a gradual sensor degradation occurs, the faulty sensor may not be immediately isolated but its influence on the estimate and the calibration of the remaining sensors is diminished as a function of its degradation because its weight decreases with an increase in the a posteriori probability of failure. Thus, if the error bounds of the measurements are appropriately increased to reduce the probability of false alarms, the resulting delay in detecting a gradually degrading sensor could be tolerated because an undetected fault, as a result of the reduced weight, does not significantly affect the accuracy of calibration and estimation. Moreover, since the weight of a gradually degrading measurement for computing the estimate is smoothly reduced, the eventual isolation of the fault does not cause any abrupt change in the estimate. This feature is very desirable in feedback control systems.

Description of the Reactor and Its Instrumentation

A description of the system configuration and instrumentation of the 5 MWt fission reactor is given in the MITR-II Reactor Systems Manual.[5] The reactor is heavy-water reflected, light-water moderated and cooled, and functions as a research and educational facility. It is a tank-type reactor, similar to a PWR except that it operates at atmospheric pressure and its coolant temperature is 55°C or less. A simplified diagram of the process and instrumentation is given in Figure 1.

The nuclear instrumentation used for the research described in this paper consists of three neutron flux sensors and a gamma-ray sensor that correlates neutron power with the radioactivity (N-16) of the primary coolant. All four sensors are linear over the power range. Four independent measurements of primary coolant flow are obtained from the pressure differences across orifices. Primary coolant temperatures are measured as follows: two sensors for the hot leg, two sensors for the cold leg, and one sensor for temperature difference between the legs. In effect, three measurements are available for the temperature difference. The noise and statistical characteristics of the MITR-II's flow, temperature, and neutron flux instrumentation are similar to those in commercial reactors. These sensors are hard-wired to a portable LSI-11/23 minicomputer through appropriate isolators, signal conditioners, and A/D converters. None of the sensors that form the MITR-II's safety system were used for this research.

Description of the Control Scheme

Coarse control of the power in the MITR-II is achieved by manually positioning a bank of six shim blades. Once critical, the neutron flux is normally maintained constant by an automatic analog controller that is monitored by the reactor operator. The controller adjusts a fine-control regulating rod according to the feedback signal of a single power sensor, as shown in Figure 2. If the error signal, i.e., the difference between the reference and the sensor signal exceeds a specified bound (typically around 2% of the rated reactor power of 5 MWt), "automatic" control is tripped to the "manual" mode in which control is maintained directly by the reactor operator. The analog controller in Figure 2 is a standard proportional-

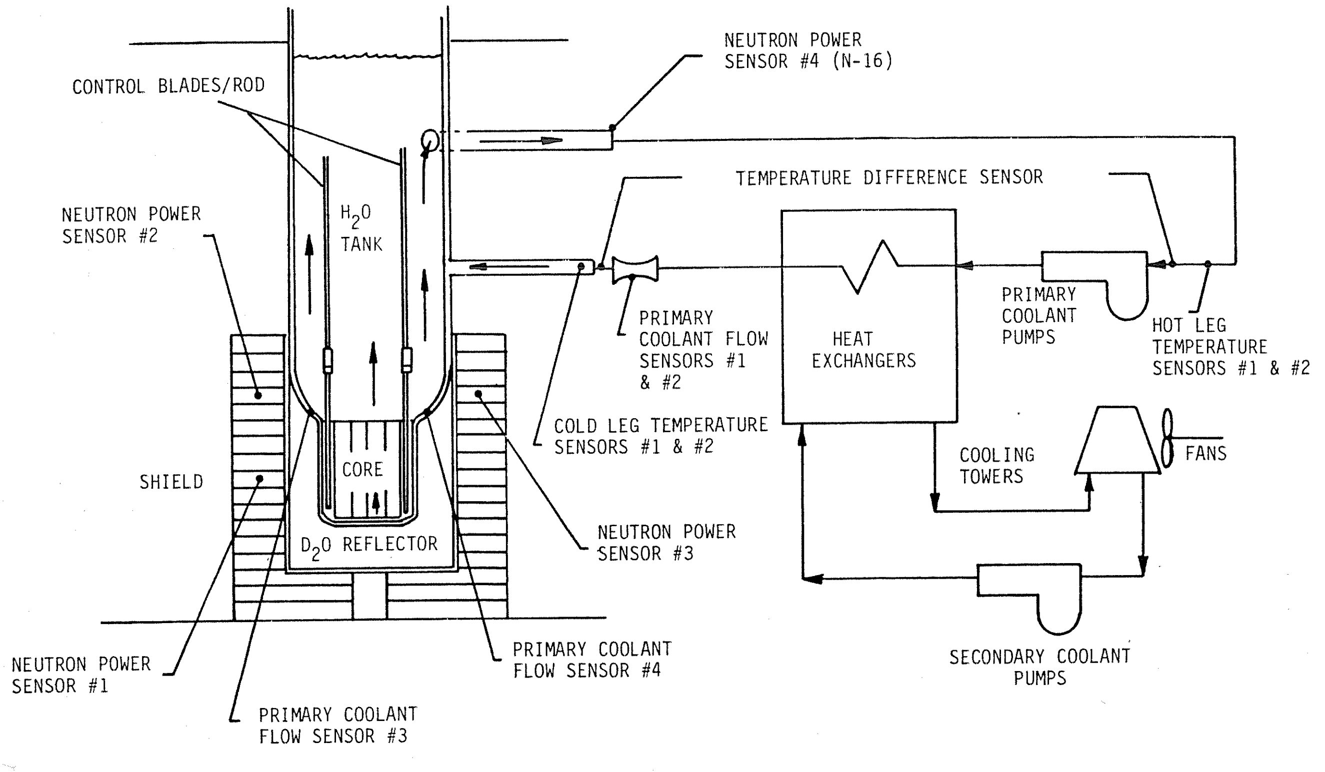

Figure 1. Simplified Process and Instrumentation Diagram for MITR-II

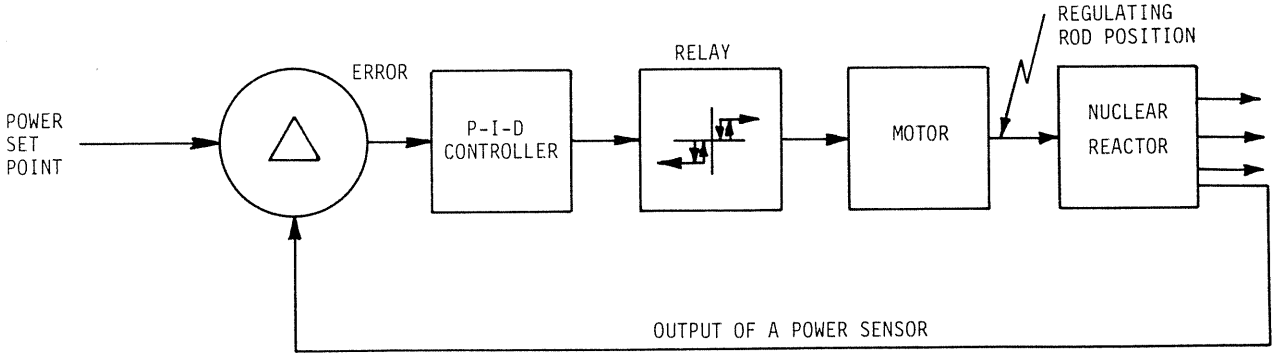

Figure 2. Analog Control System in MITR-II

integral-derivative (P-I-D) type. The controller output energizes a 3-position relay to drive a constant speed motor that moves the regulating rod up or down at 108 mm/minute, which corresponds to an average of reactivity change of about 1.1×10^{-5} ΔK/K per second. The signal validation system, which has been verified by on-line operations during the last several months, replaces the feedback signal from that single power sensor with an estimated measurement of power that is a digitally processed, weighted average of several power signals. The first phase of the experimentation[6] retained the analog controller and only involved the replacement of the sensor signal in the feedback path. In the experiments now in progress, the single sensor and the analog controller in Figure 2 have been successfully replaced by a digital control system as shown in Figure 3. Control algorithms, including the existing analog one, have been tested. The goal is to develop and demonstrate fault-tolerant, robust, digital control schemes for the various operational modes of power reactors.

Safety Features

The computer control scheme in Figure 3 can be activated by one or more power sensors, and is not restricted to a specific structure such as P-I-D in the analog controller. Functionally, this control scheme is more flexible and, from the point of view of sensor failures, more reliable than the original one (Figure 2). The experiments have been designed so that possible hardware and/or software failures in the computer system will trip the controller to the manual mode as required by the MIT Reactor Safeguards Committee. A transfer from "automatic" to "manual" control is initiated if any one of the following events occurs:

(1) The output of a particular power sensor exceeds the reference signal by a preset bound, typically 2% of full power. This trip retains an existing safety feature of the analog system, which causes a trip to "manual" if the difference between the power sensor and the reference is exceeded by a similar amount.

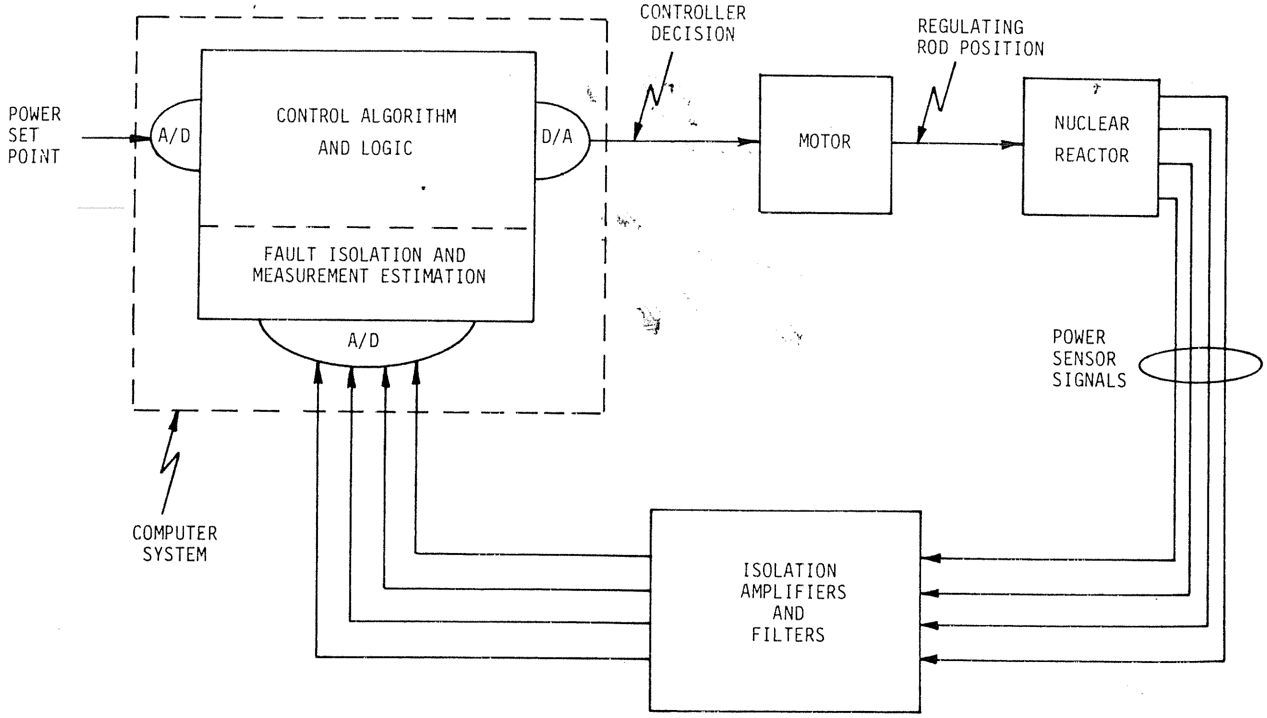

Figure 3. Computer Control System in MITR-II

(2) The computer changes its state from "run" to "halt." This action provides protection against a hardware failure.

(3) The computer does not function in a cyclic fashion, i.e., the A/D and D/A converters subprograms are not periodically called. This action protects against software failures by insuring that the program is being traversed in the proper sequence.

(4) If, due to simultaneous multiple sensor failures, a valid estimate of the reactor power cannot be generated, a trip to "manual" is initiated. This feature protects against an indecision.

(5) If the reactor period is shorter than a conservatively set value, a trip to "manual" is initiated. This action protects against an unforeseen situation that might drive the regulating rod out continuously.

Systematic procedures have been formulated for off-line and on-line testing of the system software. A checklist that tests the safety features must be performed prior to initiating any form of digital computer control.

Results of the Experiment

The digital control system was tested during steady-state and slow reactor transients induced by xenon, temperature, and fuel depletion. A sampling interval of 0.2 seconds was used for most of the tests. The immediate observations from the experiments are as follows:

- The digital controller is completely capable of maintaining reactor power constant during transients involving peak xenon burnout and rapid temperature changes.
- The digital control scheme is tolerant of one or two failures of the power sensors. This fact has been verified by both natural and induced failures.
- The digital control system (Figure 3) is less sensitive to sensor degradation and noise than the original system (Figure 2).
- The digital control system exhibits sustained oscillations in reactor power in excess of the process noise as the sampling period is increased beyond 1 second. This difficulty can be partially circumvented by replacing the on-off controller and the fixed speed motor by a continuous action controller and a variable speed motor.
- The digital controller maintains the reactor power closer to the desired value than does the analog controller.

The data taken from one of the neutron flux detectors and the switching actions of the regulating rod drive motor are shown in Figure 4 to demonstrate the performance of the digital controller in MITR-II. The controller generates a signal at every sample that either holds the regulating rod stationary or moves it up or down at a constant speed such that the reactor power is maintained at the desired level within a prescribed dead band. Initially, the power was held steady well within ±1/2% of the rated value of 5 MWt, i.e., ±25 KWt as depicted in Figure 4 for the first two minutes. Figure 4 also displays the actions of the digital controller under perturbations caused by the insertion and extraction of a sample, that amounted to step reactivity changes of approximately -0.015% and +0.015% $\Delta K/K$, respectively. Due to the sample insertion, the power decreased by approximately 1.5%. The controller promptly responded by an upward movement of the regulating rod that lasted for 10 seconds. Later, when the sample was extracted, the power sharply increased by approximately 1.5%. The controller caused a continuous downward motion of the regulating rod for 10 seconds resulting in a small undershoot that was compensated by an upward movement of the rod for 4 seconds. The power then settled down to the original state within 25 seconds. Although the performance of the digital controller conforms to the industrial practice for process control,[7] research efforts are directed towards the development of advanced control algorithms and modifications of the hardware that are expected to yield superior controller actions.

Potential Applications in Power Reactors

The work reported in this paper is a part of a

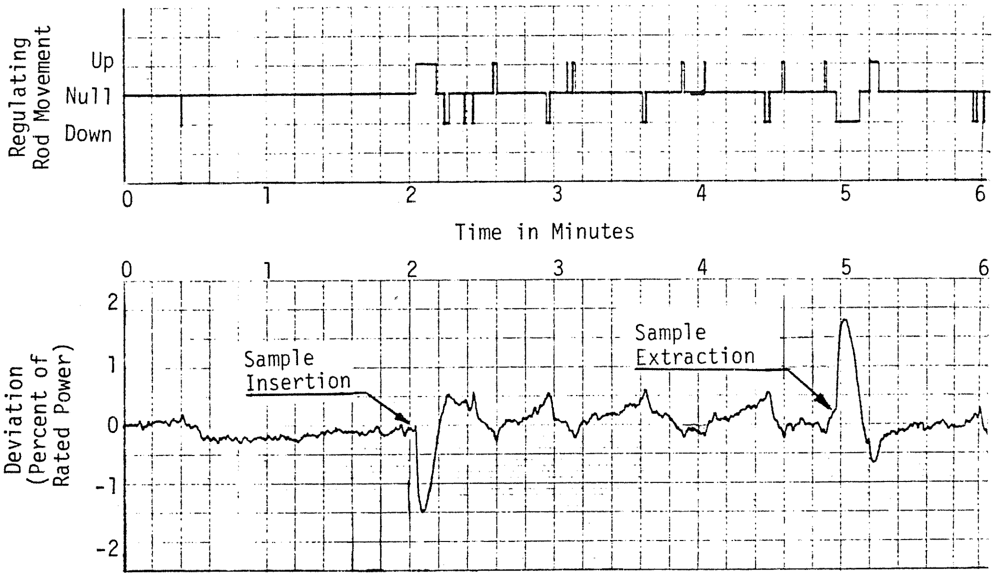

Figure 4. Reactor Power Profile Under Computer Control

general research program on the development of information display and control systems. The concept of the program builds upon the computer-aided management of the information available from the sensors and real-time analytic redundancies.[4] The scope of the program includes diagnostics of sensor faults and system malfunctions, and the validated data can be utilized in the feedback control systems. Two of its potential applications in nuclear power plants are discussed below.

Detection and Control of Xenon Oscillations

Oscillations in the spatial distribution of fission product poisons such as xenon may occur in large reactors.[8] The detection and damping out of such transients are particularly important to the safe operation of nuclear power plants in load-following modes. The control system, presented in this paper, is suitable for preventing such oscillations since it combines on-line fault diagnosis, sensor calibration, and control. The MITR-II, having a compact core, is not affected by spatial xenon oscillations. However, it is equipped with several water-filled shutters that are drained to admit neutron beams to various test facilities. The cycling of these shutters perturbs the leakage flux and influences the neutron sensors, although on a much shorter time scale, in a manner similar to what xenon oscillations would do in a power reactor. Experiments have shown that the digital control system maintains the reactor power constant while the cycling of these shutters causes some of the power sensors to vary to the extent of 15%.

Information Display Systems Design

The signal validation techniques play an important role in the design of man-machine interfaces. On the human side, the reliability is improved by increased confidence in the valid data and reduced confusion about failure messages as well as alarm system prioritization. On the machine side, the information system reliability and the accuracy of the measurements are improved by the utilization of all redundant information rather than relying on single sensors. The process parameters, such as the NPSH at the pump suction and the subcooling margin of the primary coolant in a pressurized water reactor (PWR), that are not directly measurable can be determined on-line from analytic redundancies.

One specific application at the man-machine interface is the Safety Parameter Display System (SPDS) that is now being developed for nuclear power plants.[9] The SPDS is designed to give an operator a quick and reliable review of the parameters that are necessary for the safe operation of the plant. The on-line fault diagnosis, analytic redundancy, and sensor calibration capabilities of the signal validation system form an ideal basis for the SPDS installation.

Conclusions

The benefits of on-line fault diagnosis, analytic redundancy, and sensor calibration have been incorporated in the development of a computer control system that has been successfully demonstrated for real-time information display and feedback control of power in a nuclear research reactor. The control scheme is tolerant of sensor failures, less prone to sensor noise, and more flexible than the original analog controller.

The information display and control system is readily adaptable to detection and control of xenon oscillations and to Safety Parameter Display System (SPDS) design in nuclear power plants. This technique can be applied to on-line signal validation and feedback control in fossil power plants and process industries as well.

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