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A Research Overview for Developing Enhanced Reactor Operation Strategy Through Improved Sensing and Control at NPPs

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Abstract

The Chosun University (CU), the Korea Atomic Energy Research Institute (KAERI) and the Cheju National University (CNU), the Oak Ridge National Laboratory (ORNL) and the Ohio State University (OSU) collaborate to examine, develop, and demonstrate how modern sensing and control can improve the operation of nuclear power plants.

The project consists of three tasks. The objective of the first task is to evaluate the basis for current reactor operation strategies including assessment of the state-of-the art for primary system measurement, investigation of the effects of measurements limitations on operational performances of existing NPPs, and identification of potential operational/safety improvements resulting from improved measurement and control. The objective of the second task is to develop three advanced sensors; a solid-state in-core flux monitor, a Johnson noise thermometer and a magnetic flow meter. The objective of the third task is to take advantage of the benefits of improved sensors by devising advanced reactor operational strategies that optimize core performance and permit reduced operational margins.

1. Introduction

New sensor technology will allow inexpensive monitoring of more of the plant than previously possible because of small footprint inexpensive sensors and the ability to place some sensors after the plant is built (using wireless technology). The use of optical techniques for improved accuracy and reliability in measurements and first principle-based sensors will allow much more accurate and reliable sensing. Faster, more powerful computer processors and display technologies will allow better, more reliable control and operational strategies. Newer generation nuclear power plants will take advantage of several new features as shown in Fig. 1.

The objective of this research project is to examine, develop, and demonstrate how modern sensing and control can improve the operation of nuclear power plants. A more precise knowledge of the reactor system state (e.g., primary coolant temperature, core flux map, primary and feedwater flowrates) can facilitate operation closer to design margins, support improved thermal efficiency, and permit extended fuel burn-up. As a result, advanced control methods (e.g., innovative control algorithms) need to be developed to realize these benefits offered by improved sensing capability.

Nuclear power plants are usually licensed to operate at power levels up to a specified thermal power rating. Safety analyses and evaluations are performed at conditions selected to account for uncertainties in determining thermal power. The NRC in Regulatory Guide 1.49, Rev. 1, December 1973 provides guidance regarding the amount of margin needed to account for uncertainties. Guidance provided in Regulatory Guide 1.49 recommends that analyses and evaluation be made assuming thermal power is equal to 1.02 times the licensed thermal power. The reason that analyses should be performed at two percent above the licensed thermal power is to allow for possible instrument errors.

Nuclear power plants are required to operate at a specified thermal power level. Margin is introduced to account for measurement uncertainties. Therefore an accurate spatial measurement of thermal power in the reactor core is of significant importance for enhanced safety, reliability and increased power output of both current and future nuclear power plants.

In current operating nuclear power plants neutron flux is measured to provide a continuous indication of thermal power. In the case of pressurized water reactors (PWR) these flux measurements are external to the core thus providing little spatial information. In addition the nuclear instruments used to measure neutron flux must be calibrated using a heat balance to measure thermal power. The accuracy with which the thermal power is calculated is primarily dependent on the accuracies of the following direct measurements: feedwater flow, feedwater temperature and steam pressure.

This project is directed to methods to improve these measurements and to control algorithms to use these measurements to improve control of thermal power. Expected benefits of this work include a roadmap for deciding what level of improvements are possible, the resulting advantages in operation, and examples that can be used in future designs.

The proposed project is collaboration among ORNL, Ohio State University (OSU), Chosun University (CU), Cheju National University (CNU), and Korea Atomic Energy Research Institute (KAERI). The research proposed for ORNL and OSU will be funded by the U.S. Department of Energy (DOE). The research proposed for CU, CNU and KAERI will be funded by ROK Ministry of Science and Technology. ORNL will have management responsibility for the US portion of the project and CU will have management responsibility for the ROK portion of the project.

2. Background

Instrumentation and Control (I&C) systems in current operating nuclear power plants have not changed appreciably since their original design in the 1950's. These systems depend on a variety of traditional process and radiation sensors for the measurement of safety and control variables such as temperature (RTDs, thermocouples), pressure (diaphragm, piezoelectric), flow (differential pressure across a flow-restricting orifice) and neutron flux (Fission chambers, ion chambers).

The I&C systems in the advanced light water reactor designs, i.e. Generation III NPPs, do employ more advanced technology than current plants, however, they do not incorporate new technology on a broad scale. This in part is a consequence of the ALWR design philosophy that discouraged use of advanced technology if current technology was adequate. As a result the I&C systems in the ALWRs continue to make use of current technology. There are two exceptions, however, which are the broad use of software-based digital systems and fiber optics for signal isolation and data transmission in nonradioactive areas.

As we consider I&C systems in Generation IV reactors we have the opportunity to take a much less "timid" design philosophy than was taken in the design of I&C systems in the ALWRs. Since there will be increased use of fiber optics for data transmission there should also be increased use of optical based sensors. We should also take advantage of microprocessors, which provide opportunities to embed "intelligence" in the sensor that can be used to increase accuracy, stability and tolerance to external stressors (i.e., radiation, humidity, smoke, and high temperature) (Hashemian, 1998; Miller, 1999) and wireless sensors, which provide opportunities for innovative sensor location.

Currently the resistance temperature detector (RTD) is the sensor most often used to measure feedwater temperature in a PWR. Although RTDs are extremely accurate, the 0.3 C accuracy design requirement for RTDs in the hot/cold legs in PWRs leaves little tolerance for drift (bias) errors. As concluded in studies by (Donahoe, 1992) and (Hashemian, 1990) drift in RTDs is dominated by thermal aging due to long term exposure to high temperature that leads to change in resistance-thus change in indicated temperature or a bias error. This may result from insulation degradation or change in resistance of the sensing element from increase in tension stresses. As discussed in Task 2 description Johnson noise is insensitive to the material condition of the sensor and consequently is immune to contamination, transmutation, and thermo-mechanical response shifts. In the proposed measurement application, an RTD is used to detect the “base” temperature. Then, every few seconds, the Johnson-noise first principles measurement is used to correct the RTD value for drift errors.

Until recently, the Johnson noise thermometry (JNT) has been a thermodynamic technique relegated to either very low temperatures (T<1° K) or very high temperature (T>1200° K) where relative accuracies of $1x10^{-3}$ are sufficient. However, with the advent of higher speed, higher accuracy, analog-to-digital converters (ADCs), and the resulting growth of digital-signal-processing techniques, it is now becoming feasible to accurately measure Johnson noise over larger bandwidths at higher accuracies than ever before possible. In addition, the recent advances in intrinsically accurate, ac waveform synthesis via the Josephson ac voltage standard now permit the construction of a calculable noise source (NIST, 2000).

Liu and Miller (2000) describe use of a Fabry-Perot fiber optic temperature sensor which was selected for performance evaluation and for potential application in nuclear power plants because of its unique interferometric mechanism , embedded data processing technique, and its commercial availability. They have shown that this sensor configuration is very tolerant of internal degradation and external stressors, and can potentially be qualified for use in safety-related applications in nuclear power plants. In addition optical sensors are inherently immune to EMI/RFI.

Primary loop flow measurements are used to determine the core heat rate in PWRs that is a primary parameter in plant thermal efficiency. These measurements are conventionally made using flowmeters based on differential pressure. A differential pressure flowmeter consists essentially of a flow-restricting orifice with pressure measurement devices located on either side. Such differential pressure flowmeters have several fundamental performance limitations. Over time, contamination products can build up on the orifice, thereby changing its calibration. Also, some differential pressure transmitters have a failure mechanism (oil leak) that cannot be readily detected while in service (Weiss, et al., 1990). Moreover, differential pressure measurements have anaccuracy under accident conditions as low as ±10% (Weiss, et al., 1990). In addition, in differential-pressure-based flowmeters, the pressure change varies non-linearly with flow rate, thereby limiting the range of the measurement and reducing its accuracy.

A 1% error in primary loop flow can result in a 1% reduction in unit net load if the error is in the high direction. In order to avoid errors in the low direction (and exceeding the licensed plant thermal power) a margin is built into the control system. Improved accuracy of primary flow measurement allows reduction of this margin. EPRI has reported that typical power plant primary flow measurement errors are ~3-5% (Iverson, et al., 1995).

In the past several years the use of acoustic methods, either transmission timing or correlation methods have been developed to the point where they are being introduced as a backfit in operating plants. The advantage these methods offer is increased accuracy, which translates into increased reactor power. (Regan 2000) compared three methods for flow measurement in light water reactors in terms of accuracy.

As discussed in Task 2 description magnetic flow meters offer a potential solution to limitations currently encountered in differential pressure flow meters. Magnetic flow meters are highly accurate (±0.5%), respond linearly, and have no obstructions (no fouling; consume no pumping power). Also, the transmitter for magnetic flowmeters can be located remotely (tens of meters) from the point of measurement, thus reducing environmental exposure. Magnetic flow meters operate on the principle that whenever a conductor (in this case, the coolant water—conductivity greater than the required 5 μS/cm due to the chemical buffering) is passed through a magnetic field a voltage is generated, this voltage being proportional to the velocity of the conductor. Physically, a magnetic flowmeter consists of signal processing apparatus (the transmitter), magnetic coils, and electrodes (to measure the potential across the coolant). The magnetic coils and electrodes are typically implemented as part of a short segment of pipe made of non-magnetic material that has a non-conductive inner surface pierced by electrodes.

Reactors are provided with safety-grade power sensors for one reason – to protect thermal limits to fuel melt (LOCA LHGR limits) and fuel cladding failure (CPR limits). Allowing these limits to be exceeded through excess energy deposition could result in a lack of core coolability and release of radioactivity during a design transient. In US power reactors, these thermal limits are protected by quantum-principle sensors that are designed to measure neutron flux rather than energy deposition. The three neutron flux sensors most commonly used for making power measurements are in-core fission chambers, ex-core boron lined ion chambers and non-safety-grade self-powered neutron detectors. To protect fuel thermal limits, it is necessary to relate the signal provided by these sensors to the energy deposited in the fuel.

Fission chambers most closely approximate energy deposition in the fuel. Their signal is produced with fission capture that occurs in a fissile film. If the film is very thin, fission products escape the film with a small loss of kinetic energy and enter a gas space where each particle interacts with the gas and produces ion-electron pairs. Applying a voltage differential across the gas space allows the ions and electrons from each interaction to be collected, providing a signal. Other particles entering the gas space from outside the sensor can also interact directly with the gas to produce ion-electron pairs in a manner similar to non-fission energy deposition in the fuel pellet. Therefore, this sensor responds to the same energy that is deposited in the fuel, however it is not a direct measurement of the deposited energy because the relative interaction rates of fission fragments, electrons and gammas are not the same in a gas as in the solid fuel pellet.

Neutron-sensitive ion chambers used for PWR are ex-core sensors, which are similar to fission chambers in that ion-electron pairs are generated and collected. However, the ion and electrons that result from the neutron flux are produced by alpha particles and to a lesser extent lithium ions emitted from neutron interactions with the ^{10}boron in the sensor lining rather than from fission fragments.

The solid-state, in-core flux monitor described in Task 2.1 is a poly-crystal ceramic of aluminum nitride. Electrical contacts and leads are applied to the surfaces of the ceramic plate and a bias voltage is applied across the compact. The measured signal is a change in the sample resistivity with neutron flux. The major advantages of this type of sensor as compared to traditional flux measurement technologies are its extremely small size, its ease of applicability to high temperature environments, and the reduced levels of voltage required.

Self-powered neutron detectors (SPNDs) are even farther removed from the processes leading to energy deposition in the fuel. In SPNDs, thermal neutrons are captured, transmuting the active sensor wire (typically rhodium or vanadium) into another element that subsequently emits beta particles thereby generating an electric current. The sensor must be electrically insulated, and nuclear interactions with the insulation can lead to ionization adding to the original signal. Measurements from this sensor differ from deposited energy in the fuel for the same reasons given for boron-lined ion chambers.

Ruddy et. al. (2000) describes the use of neutron sensitive silicon carbide semiconductor detectors as ex-core neutron monitors for pressurized water reactors. The authors identify several advantages of these detectors, which includes combining the functions of the current three-range system into one system and eliminating the need for gamma compensation. They have two additional attributes that could potentially make them useful for incore measurements. Unlike the long ion chambers currently used for power range monitoring in PWRs silicon carbide detectors can be configured to provide many discrete spatial measurements of neutron flux. This could be useful in improving the measurement of axial offset in PWRs. They offer the possibility for gamma spectroscopy, which for example through the use of tomographic methods, could provide useful incore information.

There is one notable exception to the use of quantum instruments. During the mid-1980s the RADCAL gamma thermometer (Knoll, 1989) was developed and tested in European reactors. The gamma thermometer is an energy deposition device in which photon interactions heat a piece of metal and the heat flux developed is measured by a pair of thermocouples. This non-quantum sensor has two advantages; it is non-depleting and it can be calibrated *in situ* using an electrical heating element. The gamma thermometer, however, has two characteristics that make in unacceptable for use in safety-related power measurement. It is insensitive to neutron flux and its response time is limited by the physics of heat conduction.

A new method for local measurement of reactor power is being developed (Support, 1996-2003, provided by EPRI, DOE NEER and DOE NERI) at Ohio State (Radcliff, 1999, 2000; Liu 2000). This power sensor concept is based on maintaining a constant temperature in a small mass of actual reactor fuel or fuel analogue by adding heat through resistive dissipation of input electrical energy. Sensors of this type can provide a direct measurement of the nuclear energy deposition rather than neutron flux. The constant temperature power sensor (CTPS) concept can be introduced by considering the energy balance on a fuel pellet. Energy is deposited through many different nuclear interactions while energy is removed through conductive and convective heat transfer to the reactor coolant. Since the heat removal rate is a function of temperature difference, the fuel pellet will achieve thermal equilibrium at a temperature above the surrounding bulk coolant given a particular energy deposition rate.

Now consider an electric heating element placed within the fuel pellet. Introducing a current through this element will deposit energy in the fuel through resistive dissipation in addition to any nuclear energy deposited. Proper control of the dissipated electrical energy maintains the fuel pellet temperature above that resulting from the deposited nuclear energy. Note that the resistance of the heater element is directly related to average temperature by its material coefficient of resistivity. The heater element, then, allows both control and measurement of the fuel pellet temperature.

This becomes a useful innovation when we place the heater resistance in a feedback control loop that varies energy addition to the heater to maintain a constant heater resistance, and therefore temperature, regardless of the nuclear energy deposition in the fuel pellet. If the heat transfer and environmental temperature remain constant, the electrical power added to the pellet becomes an inverse measure of the deposited nuclear energy.

This inverse signal is unique in reactor power sensing. The instrument signal will be large at zero reactor power and must be designed to decrease to zero given a deposited nuclear energy somewhat greater than the highest expected local value. This type of constant temperature control has been well-developed in the field of hot-wire anemometry.

The constant-temperature power sensor has three attributes that make it potentially superior for the measurement of power in modern or future reactors when compared with the quantum or classical instruments currently available. First, the CTPS provides a direct measure of local deposited energy with good sensitivity. All of the components of the signal, whether from fission fragments, neutron, beta or photon interactions, are in the correct proportion. Second is the potential for a time response sufficient for safety-related applications. With a fission pile or a gamma thermometer, it is necessary that energy deposition in the heated element and heat transfer to the environment come into equilibrium through a change in the element temperature. This is an inherently slow, asymptotic process. In the CTPS, the thermal inertia of the sensor is much less important because the temperature is held nearly constant. The third attribute is that the two-wire sensor is in a feedback control loop. Control manipulations to extract information about the instrument dynamic response *in situ*, introduces the potential for sensor self-diagnostics and local heat transfer characteristics.

In order to improve the performance of nuclear power plants and to make them more robust many plant control systems have been upgraded from analog to digital, however, most of them continue to utilize traditional single input single output architecture. This project contains development of new protection logic and control algorithms using improved sensing techniques to be developed and thus development of enhanced reactor operation strategy by integrating improved sensing and control at nuclear power plants. Thus, to accomplish complete plant control and protection systems of which digital systems do not mean that only its appearance is digital, which means to utilize up-to-date computer-based I&C technologies, we will develop advanced algorithms to be applied to major control and protection problems in NPPs. The control problems include core power level and distribution controls and the protection problems include Departure from Nucleate Boiling (DNB) and Local Power Density (LPD) protections.

3. R&D Tasks Description

The project consists of three tasks designed to (1) evaluate the basis for current reactor operation strategies, (2) develop advanced measurement capabilities to more precisely determine reactor conditions, and (3) devise advanced reactor operation strategies that optimize core performance and permit reduced operational margins.

TASK 1—EVALUATION OF THE BASIS FOR CURRENT REACTOR OPERATION STRATEGIES

Task 1.1 Assessment of the State-of-the-Art for Primary System Measurements (OSU/CU/CNU)

The objective of this task will be to complete a comprehensive assessment of pressure, temperature, flow and in-core flux measurement systems that may be used in nuclear power plant (NPP) instrumentation and control (I&C) systems. This assessment will be accomplished through a detailed review of the state of the technology (e.g., sensor specifications, existing performance documentation, recent research publications, etc.) and a comparative analysis of the performance characteristics for key advanced sensors and measurement systems.

A review of all relevant documentation will be conducted. We will include a review of the open literature, relevant conference proceedings and reports written on behalf of national research labs both domestic and international. The assessment will categorize specific measurement systems in terms of: a) developed with extensive operational experience; b) developed with no or limited experience; and c) still in the research and development stage. The scope of the assessment will include traditional process instrumentation and advanced measurement systems such as optical and wireless sensors and measurement systems with embedded microprocessors, i.e. “smart sensors.”

Also, this task will involve analysis of the characteristics of candidate measurement methods and systems. Specifically, it will include but not be limited to the following:

- A detailed accuracy comparison between commercially available ultrasonic flow meters and the magnetic flow meter described in Task 2.3 description.
- A performance assessment of the capabilities of optical and wireless temperature measurement systems compared to the Johnson noise temperature measurement method that is described in Task 2.2 description.
- A comparative investigation of the informational content available from advanced computational methods applied to ex-core flux measurements with the flux distribution information available from advanced in-core flux measurements. For this comparison, it will be assumed that ex-core flux monitors have the operational characteristic of being located at many different discrete spatial locations and the capability of doing real time gamma-ray spectroscopy. The advanced in-core flux sensors considered for this comparison, such as the solid state, in-core flux monitor described in Task 2.1 description, can provide spatially discrete in-core flux measurements. In addition to a comparison between in-core and ex-core flux sensors we will also consider the benefits of simultaneous use of spatially discrete in-core and ex-core flux sensors.

Task 1.2 Effect Analyses of Measurements Limitations on Operational Performance of Existing NPPs (CNU/CU)

The objective of this task will be to investigate the effects that limitations in accuracy and reliability of sensors have on the current operational performance of NPPs. This investigation will involve an assessment of current operational strategies and safety approaches employed in existing NPPs, a determination of design basis and technical specification constraints related to system state information and/or measurement deficiencies, and identification of focus areas for potential improvement.

This task will involve selection of an existing NPP and review of its operational procedures, safety analysis report, and technical specifications. From this investigation, the consequences of incomplete or inadequate system state information can be identified. For example a review of technical specification limits will identify key measurements and margins that are imposed because of uncertainties.

Task 1.3 Identification of Potential Operational/Safety Improvements Resulting from Improved Measurement and Control (CU)

The objective of this task will be to determine the most promising operational approaches for the focus areas identified in Task 1.2. These operational approaches will take advantage of enhanced knowledge about the reactor system state provided by advanced measurement technologies. In this evaluation, NPP operational performance will include safety, reliability and plant efficiency. The opportunities that exist may involve reducing margins incorporated in technical specification limits, introducing advanced control algorithms, and/or increasing the level of automation for selected operational regimes. Specifically, this evaluation will focus on the potential impact on operational performance by improvements in temperature, flow, and flux measurements.

In addition, this task identifies safety improvements that will be possible through better measurements and addresses possible improvements in operational safety margins related to the measurement improvements. If various primary system measurements including in-core flux, pressure, temperature and flow will be improved, then the set points for safety system actuation (or operator action) may be raised even for existing protection systems. Also, improvements in measurement accuracy should allow reduced operational margins and expand operational regions by developing new protection logic. Some part of this task will be used as inputs of Task 3.1.

Improvements in various system measurements, including in-core flux, pressure, temperature and flow, should support the use of advanced control algorithms and approaches. This task examines potential operational performance improvements (more power, fewer operational upsets, less stress on operating components) through system measurement improvements coupled with advanced control strategies. This task will be used as inputs of Task 3.2. Operational regimes examined in the study will be start-up, normal operation, load-following, shutdown, and operational upsets. Emphasis will be placed on the regime shown by early examination to be most fertile for improvement.

TASK 2—SENSOR AND SIGNAL PROCESSING DEVELOPMENT

The objective of this task is to design, prototype, and test improved sensors and their requisite readout electronics and algorithms relevant to increasing knowledge of the nuclear plant status. Three separate sensing systems are proposed. The first of these sensors is a solid-state in-core flux monitor based upon an aluminum nitride poly-crystal ceramic. The second sensor is a digital implementation of a Johnson noise thermometer. The final sensor type is a magnetic flow meter for PWR primary flow loop measurement.

Task 2.1 Solid State, In-Core Flux Monitor (ORNL/OSU/KAERI)

The proposed solid-state, in-core flux monitor is essentially a nitride crystal. Electrical contacts and leads are applied to the surfaces of the ceramic and a bias voltage is applied across the crystal. The measured signal is a change in the sample resistivity with neutron flux. Figure 3 illustrates the situation. The major advantages of this type of sensor as compared to traditional flux measurement technologies are its extremely small size, its ease of applicability to high temperature environments, and the reduced levels of voltage required.

In order to function as a neutron flux sensitive resistor a material must possess particular material properties. First the material must have high electrical resistance. The sensor signal consists of the neutron-induced current. Non neutron induced current is a noise signal, so the sensor resistance must be large enough to allow the extraction of the neutron induced current from all other electrical currents. One of the electrical currents of particular concern is that thermally induced current. Since nuclear reactor core operational temperatures can approach 1000 °C, the sensor material must continue to exhibit high electrical resistances at these temperatures.

Another electrical property the sensor must exhibit is a high free carrier mobility lifetime product. The first step in the process by which the neutrons generates an electrical current in the sensor is through creating free-carriers via the slowing down of the energetic daughter particles resulting from a nuclear reaction. These free carriers (electron-hole pairs) must be free to move appreciably to serve as a current. In many wide band-gap materials such as silica or sapphire the free-carrier lifetime is less than a picosecond, so the carriers do not move appreciably under an applied voltage and thus not produce a measurable current. All of the group III nitrides, to the extent that their properties are known, exhibit high carrier mobility. The properties of the group III nitrides are reviewed in the recent compilation "Properties of the Group III Nitrides".

A further electrical property that the sensor material must exhibit is good electrical field penetration. In many materials one of the constituent atoms or an atomic impurity is mobile with applied electric field. In this case, the mobile species moves toward to the electrode of the opposite polarity and the rearrangement of the charge within the material neutralizes the field within the sensor. Essentially the material functions as an electrolytic capacitor with near zero internal field away from the region immediately adjacent to the electrodes. Note that the figure-of-merit most commonly used when comparing the electrical properties of various detector materials is the schubweg which is the product of the carrier mobility, the electric field, and the carrier lifetime. Another required property for the flux sensitive resistor material is that it must have a significant neutron interaction probability. Nitrogen-14, which composes 99.6% of naturally occurring nitrogen, has a significant inverse neutron velocity proportional energetic proton production cross-section. This means that all of the group III nitrides have a significant neutron interaction cross-section. Furthermore the boron-10 isotope of boron nitride has a very high neutron induced alpha particle production cross-section. Boron nitride is therefore particularly interesting for situations requiring higher sensitivity.

The neutron interaction within the sensor material must yield energetic charged particles to generate electron-hole pairs as the energetic charged particles slow down in the material. Both ^{10}B and ^{14}N have neutron capture reactions that yield energetic charged daughter particles as shown in equations 1 & 2.

$$^{10}B+{}^{1}n \rightarrow \begin{cases} {}^{7}Li+{}^{2}a\,(6\%) \\ {}^{7}Li+{}^{2}a\;+g\begin{pmatrix} & E_g &= 480keV \\ 94\% \leftrightarrow & E_{Li} &= 840keV \\ & E_a &= 1.47MeV \end{pmatrix} \end{cases} \tag{1}$$

$$^{14}N+{}^{1}n \rightarrow {}^{14}C+{}^{1}p\begin{pmatrix} E_C = 42keV \\ E_p = 585keV \end{pmatrix} \tag{2}$$

Both aluminum nitride and boron nitride appear to possess all of the required properties to function as neutron flux monitors. Boron nitride will have a significantly higher sensitivity, but will suffer from more rapid burn-up as well as very high thermal loading under high fluxes. Whereas aluminum nitride will have a lower overall signal that may be low enough to require gamma response compensation, but should function for long times under full power neutron fluxes.

The basic device response estimation involves applying standard physical models for each of the physical process involved in the detection process to yield an estimate of the electrical current produced per unit neutron flux. The electrical resistance of high purity aluminum nitride as well as boron nitride remains very high for temperatures up to and above 900 °C.

In the case of aluminum nitride, the annual burn-up at power for such a device would be roughly six parts per thousand. It is not expected that the device will have a significant radiation damage response because the physical form that is originally used is a disorganized polycrystalline solid. This is very likely the same structure that will reform under intense irradiation.

One potential concern about this device is that the intense gamma flux within the reactor will also produce a response that may be of the same magnitude as the neutron response. While in general the gamma flux at power in a nuclear reactor core is itself considered a measure of overall reactor power and hence may not be considered parasitic, it may be required to compensate for the gamma response of the device. This is technologically possible in much the same fashion as a compensated ion chamber functions. Nitrogen-15 does not exhibit the thermal neutron absorption cross section of ^{14}N. Hence an otherwise identical device composed of ^{15}N based aluminum nitride would be expected to respond just as a ^{14}N aluminum nitride sensor, but without the neutron sensitivity. Therefore subtracting the currents from the two devices would yield the sensor current due to the thermal neutron flux. The only caveat to this technique is that isotopically separated aluminum nitride is not currently a commercial product and would have to be produced specifically for the sensor. While this is technologically feasible, it is cost prohibitive until the other performance aspects of the sensor have been demonstrated experimentally.

Since the most valuable neutron measurements of reactor power should be made at power, directly within the core, this project is focusing its efforts on aluminum nitride as a sensor material. Additionally we will only examine ^{15}N based gamma compensation, as we are successful with the initial neutron detectors.

Task 2.2 Drift-Free Temperature Measurement (ORNL/OSU/KAERI)

Johnson noise is a fundamental representation of temperature. It is the vibration of the electronic field surrounding atoms as they thermally vibrate. Since temperature is merely a convenient representation of the mean translational kinetic energy of an atomic ensemble, measurement of the electronic vibrations yields the absolute temperature. Johnson noise is inherently drift-free, moreover.

Resistance thermometers that employ Johnson noise measurements for drift correction have been developed over the past 25 years by ORNL to the point where it is a practical laboratory measurement and can be performed in the field with effort and skilled personnel up to 1100 °C. ORNL and others have proposed and to a limited degree applied Johnson noise thermometry to the nuclear industry (Shepard, et al., 1992; Von Brixy and Kakuta, 1996). The primary reasons why Johnson noise thermometry has not won widespread industrial acceptance are its sensitivity to electromagnetic noise and the continuing requirement for skilled operators. Modern advances in digital signal processing greatly improve the ability of Johnson noise thermometry to withstand electromagnetic noise. It is the specific intent of this project to create a device that does not require expertise to implement, operate, or maintain.

The technique and laboratory electronics necessary for Johnson noise thermometry have previously been developed under DOE, NASA, and EPRI sponsorship (Shepard, et al. 1993). Johnson noise temperature measurement requires sophisticated high-speed electronics including dedicated amplifier and high-speed digital signal processing chips. The analysis of the Johnson noise signal from the sensor is done with a combination of very low-noise analog electronics coupled into digital data acquisition and spectral analysis hardware. The preamplifiers are discrete, low-noise, high frequency, bipolar analog components, and the digital system consists of integrated high-speed digital circuits. Two nominally identical high-input impedance voltage preamplifiers simultaneously measure the open-circuit noise voltage of the sensor. The sensor is connected to the preamplifiers by a low-capacitance four-wire cable. This allows simultaneous measurement of the resistance of the sensor by a four-wire technique and of the noise voltage by two preamplifiers simultaneously. The two outputs from the preamplifiers are then converted to digital form, and the cross power spectral density (CPSD) is computed with the high-speed digital signal processor. The processor also analyzes the CPSD for the presence of any narrow band interference signals (as is likely to occur in an industrial environment) and eliminates them before converting the CPSD into sensor temperature. Knowledge of the CPSD and the transfer functions of the preamplifiers are sufficient to compute the sensor temperature.

To conduct this task, the first is to design and fabricate suitable Johnson noise preamplifiers as well as to establish the specific digital signal-processing algorithm to be implemented. To the extent possible, ORNL's prior work on Johnson noise amplifiers will be leveraged to produce extremely low noise, wide bandwidth preamplifiers. The central concept underlying the use of digital signal processing to eliminate electromagnetic interference to the noise thermometer is to subdivide the Johnson noise power spectral density into frequency bands and employ the known shape (essentially flat) of the Johnson noise power spectral density to eliminate those frequency bands in which added noise (EM interference) exists.

High accuracy, Johnson noise thermometry requires sophisticated electronics and signal processing to be located within a few meters (<30) of the high-temperature environment. This will require creation of an electronics protection housing capable of withstanding many years of use by field personnel in an aggressive manufacturing environment (high vibration, mechanical shock, electromagnetic noise, temperature cycling, smoke, humidity, etc.). Sufficient engineering efforts will be devoted to this and the general system integration to produce a sensor system that is practical and easily used. ORNL will be responsible for this algorithm and amplifier development

The digital algorithm developed will be implemented in hardware. With the recent and on-going advances in digital signal processing, it is intended to perform the wide bandwidth CPSD directly using Field Programmable Gate Arrays (FPGAs) as opposed to developing a custom chip-set. Since the speed of the measurement for a given accuracy is directly, linearly dependent on the bandwidth of the signal processing (for example a 0.03% uncertainty can be obtained for a 100 second integration time with a frequency bandwidth of 100 kHz) as rapid as possible CPSD computation is desired. KAERI will be responsible for this digital signal processing implementation.

Last, the performance of the developed Johnson noise thermometers will be assessed under hostile operating conditions. Specifically the accuracy and drift of the measurement will be measured under both reactor conditions (at the OSURR) and under harsh electromagnetic conditions. The electromagnetic compatibility testing will be performed at ORNL's environmental testing compatibility laboratory.

Task 2.3 Magnetic Flow Meter for Primary PWR Flow Loop (ORNL/OSU/KAERI)

Magnetic flow meters offer a potential solution to limitations of existing flow meter measurement techniques. The major limitation to the immediate application of magnetic flow meters to nuclear power plants is the radiation sensitivity of the non-conductive inner pipe liner. Ceramic pipe liners are currently available for pipe diameters up to 30 cm. However, for larger pipes only radiation sensitive materials such as Teflon ™or rubber are available. The major technical objective of this project would thus be to develop and demonstrate (including fabrication techniques) radiation tolerant large diameter, non-conductive pipe liners. Several different material and manufacturing techniques are potentially suitable. The initial material/fabrication technique attempted will be to flame spray alumina on the interior of the pipe wall. ORNL will be responsible for this task.

The primary unknown for magnetic flow meters at PWRs is the long-term accuracy of measurement under the radiation and temperature environment characteristic of a PWR with the relatively low conductivity primary coolant water. This task is directed towards assessing these characteristics. For this task, Ohio State University personnel will irradiate the prototype magnetic flow meter using the OSURR and characterize its performance and stability under low flow conditions with typical PWR water chemistries.

Following the low-flow and radiation tolerance testing at The Ohio State University and any required modification to the liner material or technique, the prototype device will be performance tested at large size and scale at Integrated Thermohydraulic Loop (ITL) of KAERI. Since magnetic flow meters rely on a known material (stainless steel) as the primary pressure boundary and very significant expense can be incurred under pressure qualifying a design, it is not the intent of this project to demonstrate a magnetic flowmeter at full PWR pressure and temperature.

TASK 3 — DEVELOPMENT OF CORE OPERATION STRATEGY

This task addresses improvement of operational margins mainly related to the protection system and addresses improvement of operational performance mainly related to the control system. This task contains development of new protection logic and control algorithms using improved sensing techniques developed by Task 2 and development of enhanced reactor operation strategy by integrating improved sensing and control at nuclear power plants. To accomplish this objective, studies on improving operational margins will be focused on protection systems with relatively complex algorithms among protection logics such as Departure from Nucleate Boiling (DNB) and Local Power Density (LPD) protection. Also, advanced control algorithms will be developed to improve operational performance. To begin, a 3-dimensional reactor core kinetics model integrated with a thermo-hydraulic model of a reactor core will be developed and implemented. Two advanced control methodologies will be used to design important control systems for a primary system and partly for a secondary system: robust control and model predictive control methods.

Task 3.1 Improvements of Operational Margins (CU/CNU)

This task addresses monitoring and protection limiting conditions for operation by employing new sensing techniques. The core protection philosophy is to define a region of permissible operation in terms of power, pressure, temperature, flow rate and 3-D power distribution, and to trip the reactor automatically when the limits of this region are approached. The protection system of the conventional pressurized water reactor designed by Westinghouse is an analog system. However, the Korea Standard Nuclear Power Plant (KSNPP) and the recently designed nuclear reactors employ a digital protection system. The CE-type nuclear power plants, on which the KSNPP is based, employ the Core Protection Calculator System (CPCS). The CPCS continuously calculates Departure from Nucleate Boiling Ratio (DNBR) and Local Power Density (LPD) to assure that the specified acceptable fuel design limits on DNB and centerline melt are not exceeded during anticipated operational occurrences. The CPCS has approximately 6,000 constants and the CPCS is designed by deciding the CPCS constants (Auh, et al., 1990). This large number of constants makes the software V&V (Verification and Validation) more difficult.

The onset of nucleate boiling is characterized by extremely high heat transfer rates. However, if the fuel rod is operated at a high enough power density, the surface temperature of the clad may eventually reach the point where the liquid can no longer wet the surface and the heat transfer mechanism becomes film boiling with severely reduced heat transfer ability, which is called Departure from Nucleate Boiling (DNB). The local boiling heat transfer rate is suddenly reduced and this reduced heat transfer rate converts the nucleate boiling into film boiling. As a result, the clad and fuel pellet overheat if the reactor is not immediately shutdown. Therefore, it is very important to monitor the system and predict the margin to DNB to ensure we prevent the boiling crisis and clad melting (Na, 1999).

The DNB correlations provide the expected value of fuel rod surface heat flux that will cause DNB for various coolant conditions and flow geometries. The ratio of the expected DNB heat flux to the actual fuel rod heat flux at a particular time during an incident is called the DNBR at that time. A correlation limit DNBR (e.g., 1.3 for W-3 correlation or 1.22 for ERB-2 correlation) is established based on the variance of the correlation such that there is a 95 percent probability at a 95 percent confidence level that DNB will not occur when the calculated DNBR is at the correlation limit DNBR. The conservative design method that the calculated DNBR is greater than the correlation limit DNBR on the limiting power rod, is established by considering all parameters at fixed conservative values. The variable value design method to be used in this project establishes a DNBR less than the normal conservative correlation limit DNBR on the limiting power rod by statistically combining the effects of uncertainties of the input parameters. Therefore, the design limit DNBR (e.g., 1.54) applicable to all Condition I and II events is determined by utilizing the DNBR sensitivities and variances in three input parameter categories: plant operating parameters, nuclear and thermal parameters and fabrication parameters (Chelemer, et al., 1975). It is needed to predict the DNB according to operating conditions by using improved sensors with small uncertainty and examine the benefits.

In addition, we will develop the prediction technique of local power density for assuring that centerline melt fuel rod temperatures are not exceeded during anticipated operational occurrences, and we can use it to monitor nuclear plant margin. We will predict the LPD according to operating conditions by using improved sensors with small uncertainty.

Task 3.2 Improvements of Operational Performances (CU/CNU)

Figure 4 shows major control systems in current pressurized water reactors (PWRs). This task will improve operational performances by developing advanced control algorithms to be applied to important control systems using the developed sensors.

Reactor core modeling combined with in-core sensor signals is required first to support development of core control methodologies. We will provide a reactor kinetics model based on two-group diffusion theory to calculate the steady state and transient core conditions. The reactor core kinetics model is updated by using signals from the developed sensors.

The conventional reactor control system consists of a temperature deviation "channel" (the difference between the programmed coolant temperature and the average coolant temperature) and a power mismatch "channel" (difference between the turbine load and the nuclear power). The conventional control method drives the control rods by compensating and filtering these two "channels". This method has the advantages of easy implementation and well-proven technology. However, it is difficult to optimally design compensators and filters for controllers because of variations in nuclear system parameters, nonlinear reactor dynamics, and complex temperature feedback effects. Techniques for the optimal control of nuclear reactors were studied extensively in the past two decades. But it is difficult or often impossible to design optimal controllers for nuclear systems because of variations in nuclear system parameters and modeling uncertainties.

The model predictive control (MPC) methodology has received much attention as a powerful tool for the control of an industrial process (Kwon, et al., 1977; Richalet, et al., 1978; Clarke, et al., 1991; Garcia, et al., 1989; Kothare, et al., 1996; Lee, et al., 1998). The basic concept of the model predictive control is to solve an optimization problem for a finite future at current time and to implement the first optimal control input as the current control input. That is, at the present time k the behavior of the process over a horizon N is considered and the process output to changes in the manipulated variable is predicted by using a mathematical design model. The moves of the manipulated variables are selected such that the predicted output has certain desirable characteristics. However, only the first computed change in the manipulated variable is implemented and at each subsequent instant, the procedure is repeated. This is its main difference from conventional optimal control methods that use a pre-calculated optimal control law. This method has many advantages over the conventional infinite horizon control because it is possible to handle input and state (or output) constraints in a systematic manner during the design and implementation of the control. In particular, it is a suitable control strategy for nonlinear time varying systems because of the receding horizon concept and recently, the problem of controlling uncertain dynamical systems has been of considerable interest to control engineers (Na, 2001). Therefore, in this project the model predictive control method is applied to the reactor power level control.

In addition, since it is inevitable that the reactor model has some degree of uncertainty, the controller designed with the classical method may have performance deficiencies under realistic conditions. Hence the robust technologies such as mu-analysis or H-infinity method will be used. The three-dimensional reactor core model will be used for the design of robust controller. The controller designed by these methods is expected to provide better performance under realistic (i.e., uncertain) operational conditions. As key design factors, the performance and the stability of the system will be considered. Since these factors conflict each other, the optimized controller should be developed. In addition, the parametric uncertainty approach by use of the Kharitonov theorem will be made for the validation of the designed controller. Usually, a controller designed using robust methods has a large order (number of equations), resulting in implementation difficulty. So, model reduction will be used in the design process as necessary. All the control designs will be done using numerical simulations. And upon completion of the controller design, it will be verified by the application to the KAERI's mockup system.

The load-following operation should consider four categories to be evaluated; overall power maneuvering performance, core control methodology, NSSS operation performance, and the impacts of load-following operations. This task is focused on core control methodology using new sensing technologies. Load-following operation induces xenon-induced spatial power oscillation. Xenon oscillation is particularly important because of the large thermal absorption cross section of xenon. Its effects in the reactor are delayed because only a small fraction of xenon is produced directly by fission but the major portion is formed by the decay of the iodine precursor. The axial xenon oscillation in nuclear reactors is a highly nonlinear phenomenon that is a function of several time-variant parameters such as boron level, rod position and power level. Axially non-uniform buildup and removal of xenon cause the core power distribution to oscillate between the core top and its bottom with a period of 20 to 30 hours.

Maintaining the local core power within acceptable limits is a common objective for control problems. The core power distribution is usually manually regulated by the control rods and the control rods are inserted in radially symmetric groups. The radial power shape can be changed by moving independent rod groups. Control of the core power distribution is mostly concentrated in the axial direction. Axial power shaping in pressurized water reactors is achieved by insertion or withdrawal of groups of full-length and part-length control rods and changes in boron concentration in the coolant.

The fact that there is no direct way of measuring the xenon concentration often causes operators a great deal of difficulty in anticipating the amplitude, direction and the rate of change of the xenon imbalance that is closely related with axial power shape. Since the power distribution control has been one of the most challenging control problems in the nuclear field, there has been extensive research in this area, especially using optimal control methods. While some tracking controllers use only the current tracking command, the receding horizon control can achieve better tracking performance because future commands are considered in addition to the current tracking command. Therefore, in this project the model predictive control method will be applied to the reactor power distribution control that is a part of load-following operation. In addition, due to the actuators and valves as well as piping, there is a lot of uncertainty and it is almost impossible to obtain the exact model. To overcome this limitation, another control method, a robust control method will be used in designing the power distribution controllers.

Finally, this task contains developing the pressurizer pressure and level control systems, steam dump control system and steam generator level control system closely related with developed sensors and sensing techniques. Robust control and model predictive control methods will be applied to the control of the aforementioned systems.

Task 3.3 Integrated Reactor Operation Strategies (CU/CNU)

In this task, the foregoing methodologies for protection and control systems are integrated and implemented to provide advanced reactor operation strategies. Then their performances will be tested and verified by simulation.

4. Conclusions

I&C systems in existing nuclear power plants have not changed appreciably since their original design in the 1950's. The I&C systems in the advanced light water reactor designs employ more advanced technology than current plants, however, they do not incorporate new technology on a broad scale.

The objective of this research project is to examine, develop, and demonstrate how modern sensing and control can improve the operation of nuclear power plants. Major tasks of this project are to develop solid-state in-core flux monitor, drift-free temperature sensor and magnetic flow meter for primary PWR flow loop. In addition, to realize a variety of benefits offered by improved sensing capability, enhanced reactor operation strategies that apply advanced control and protection methods will be developed. Development of these techniques can facilitate operation closer to design margins, support improved thermal efficiency, and permit extended fuel burn-up.

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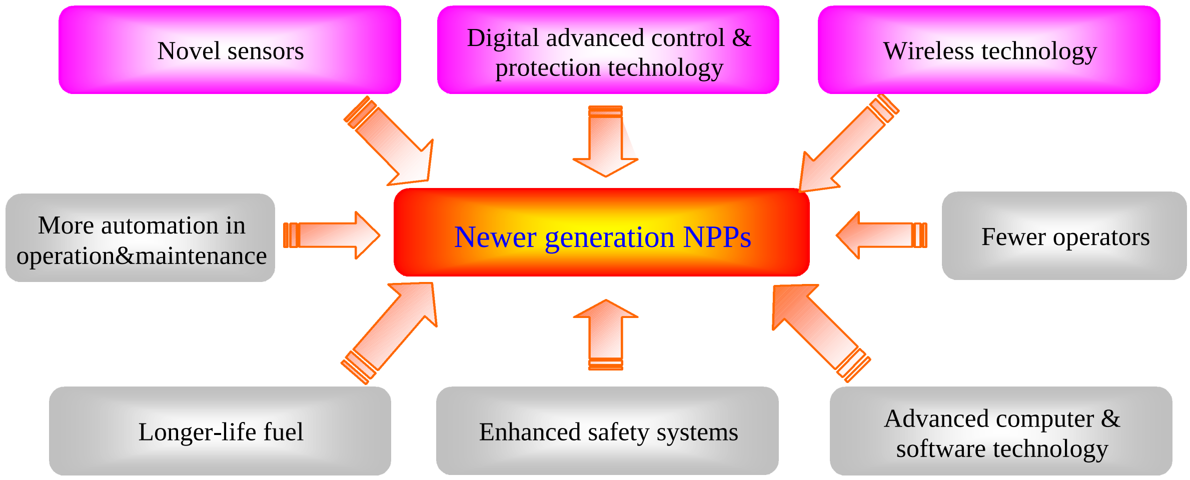

Fig. 1. Main features of newer generation nuclear power plants.

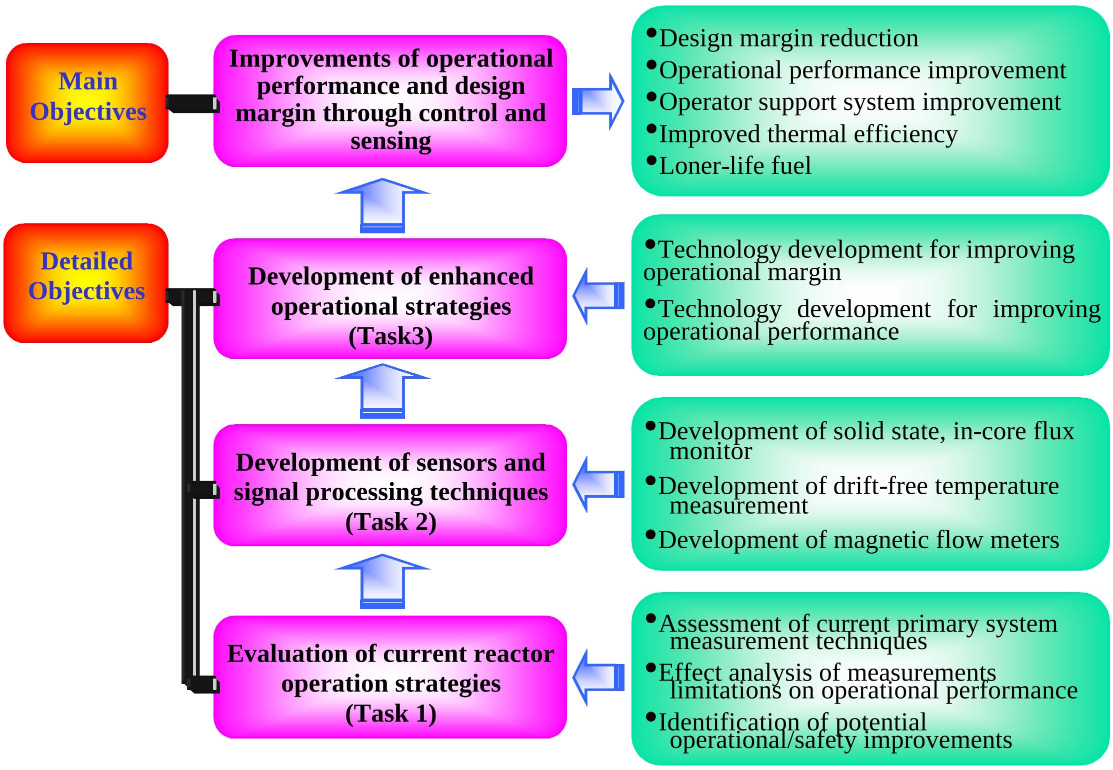

Fig. 2. Objectives of the research.

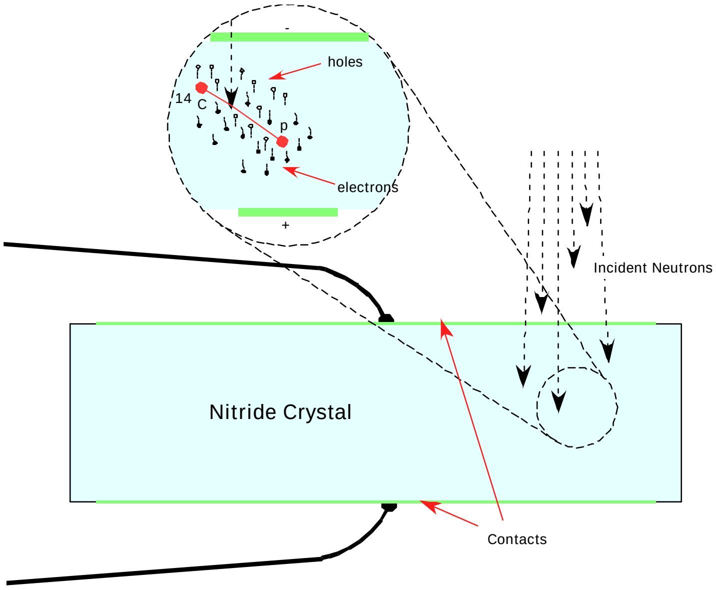

Fig. 3. Conceptual diagram of a solid-state flux monitor.

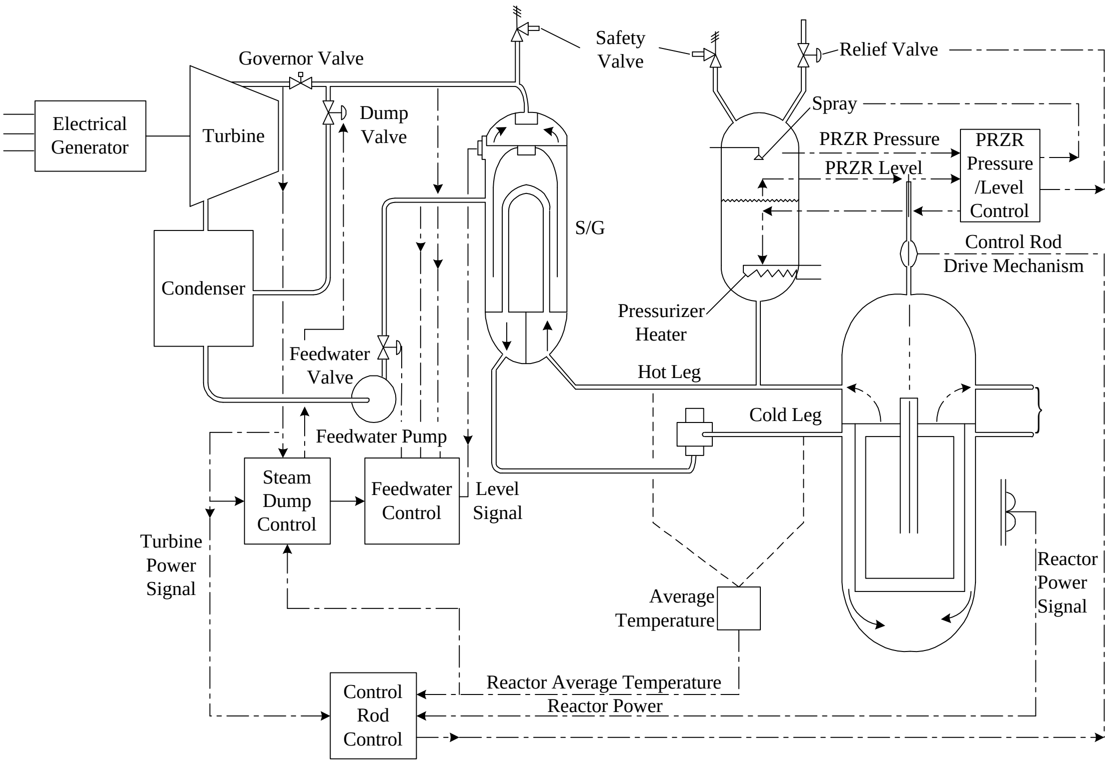

Fig.4. Major control systems in PWRs.