Development of I²S-LWR Instrumentation Systems

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Abstract - *The Integral Inherently Safe Light Water Reactor (I²S-LWR) is a novel Pressurized Water Reactor (PWR) concept being developed by a multi-institutional team led by Georgia Institute of Technology, under the Department of Energy's Nuclear Energy University Programs Integrated Research Projects (DOE NEUP IRP). The I²S-LWR aims to deliver approximately 1 GWe power output, comparable to other large scale LWRs, while simultaneously delivering improved safety over other Generation III+ LWRs. This reactor concept aims to achieve these goals by incorporating inherent safety features in the design. These features include an integral design of the primary coolant and control rod drive systems which eliminate the possibility of large break loss of coolant and control rod ejection accidents. It also features a fully passive decay heat removal system (DHRS) designed to providing indefinite core cooling as long as the core remains submerged in coolant. These features impact the design of the instrumentation system in a variety of ways. The University of Tennessee is developing instrumentation and control design concepts for the I²S-LWR. The development of instrumentation systems in the I²S-LWR, and the challenges and technology gaps to be addressed, are presented.*

1. Introduction

A large number of measurements have to be taken to operate a nuclear power plant. Even though we are only concerned with the ones that necessarily have different implementation in the I²S-LWR from any operating pressurized water reactor (PWR), those are still several important measurements. Primary system coolant temperatures and flow rate are the two most obviously different in an integral design because there are no pipes of coolant flowing from component to component within the primary loop. Additionally, the large reactor pressure vessel (RPV) size poses challenges in locating neutron detectors for different power levels.

Pipes are very convenient structures for making process measurements. They provide uniform access to the process with respect to gravity, facilitating multiple measurements of the same property for averaging to obtain an accurate value of the property. This is particularly important in accurate temperature measurement of flowing water because of density stratification of the coolant. Another convenience of pipes is that they usually have

some bends in them. Bends in pipes increase the frictional energy losses the flowing fluid experiences as its mass changes direction. This shift in momentum is reflected in a small, but measureable, pressure decrease. The magnitude of this pressure change varies with the velocity of the coolant. The faster the coolant is moving, the more energy it loses in the bend, thus the greater the magnitude of the pressure drop. The change in pressure about a bend in a pipe is essentially what low flow reactor trips are based on in a typical PWR.

Additional areas of study are core power monitoring, routing of in-vessel instrumentation cables out of the RPV, steam flashing drum process monitoring, and instrumentation related to components normally outside the RPV, such as the control rod drive mechanisms, primary heat exchangers, and the pressurizer. The use of a steam flashing drum is novel to nuclear power systems in the United States, and thus the licensing burden on the instrumentation and control systems must be considered carefully for new technology. Space limitations are also a factor.

Further, the particular goals of this project add design constraints beyond the challenges

posed by the integral nature of the reactor. The major design constraints which interact with the instrumentation systems are the restriction of penetrations in the RPV to elevations greater than four feet above the top of the core, and the less definite design goal of minimizing size and number of penetrations. When considering routing instrumentation and other cabling out of the RPV, penetration size and number of penetrations compete with each other.

First, the instrumentation of systems that are mostly outside of the RPV, are considered. These are the secondary coolant loop, including steam flashing drum, and the decay heat removal system (DHRS). The subsequent sections will discuss the instrumentation inside the RPV, identifying solutions to some challenges, and approaches for solving others. To conclude, a brief review of the work remaining in the development of the instrumentation design of this novel reactor concept is presented.

2. Instrumentation for Secondary Coolant and DHRS Loops

2.1. Secondary coolant loop instrumentation

The secondary coolant loop includes eight microchannel heat exchangers (MCHX) inside the RPV, with the following components outside the RPV: four steam flashing drums (SFD), four secondary coolant pumps, and the piping between components as well as that for feed-water delivery and steam removal. The balance of plant (BOP) portion of the secondary coolant loop should not be significantly different from a typical PWR, so it is not addressed. The secondary side instrumentation is shown in Figure 1. The secondary loop piping has instrumentation similar to the primary loop of a typical PWR.

For measuring secondary coolant temperature, each loop consists of four resistance temperature detectors (RTD) on both the hot and cold legs of the secondary circuit. Three RTDs on each leg will be distributed evenly about the circumference of the pipe, in thermo-wells at 120 degrees apart. Distribution of the RTDs in this fashion provides an accurate temperature measurement from the average of

the signals, despite stratification of the different layers of the pipe flow due to temperature differences. These sensors are narrow range RTDs. The fourth RTD on each leg is a wide range RTD suitable for coolant temperature monitoring during startup and shutdown. The accuracy and sensitivity of the averaged narrow range RTDs are only necessary when the system is at power. Normal power operation is when small changes in the temperature are important for detecting changes in reactor power as measured by the power removed by the secondary coolant.

Differential pressure transmitters and orifice/venturi meters are used for measuring pressure and flow rate of the secondary coolant before and after it passes through the heat exchangers. If more precise measurement of flow rate is necessary, ultrasonic flow meters would be used. These are discussed more in a later section. Radiation detectors will also be on the secondary coolant piping to monitor the integrity of the isolation of the primary coolant.

Figure 1: Instrumentation for the secondary coolant loop and steam flashing drum of I^2S -LWR [1]

Steam pressure measurement uses pressure transmitters. Steam temperature is measured with three or four RTDs. Steam flow rate is measured using an orifice plate or a vortex meter. Multivariable vortex flow meters are

capable of measuring pressure, volumetric flow rate, and temperature. They then calculate the mass flow rate. It is advantageous to use such an instrument because it reduces the number of cables which must be routed out of the containment to the instrumentation cabinets in the control building. If the flow, temperature, and pressure measurements are taken separately and the mass flow rate calculated by the plant computer, then each instrument requires its own signal lines. If a single instrument is used that measures multiple parameters of the process, performs a calculation, then sends this data to the plant computer, cabling capital and maintenance costs can be reduced.

The flashing drum coolant level would also be monitored by differential pressure measurement. This is important because drum coolant level is a controlled parameter. Much the way the level of a U-tube steam generator is maintained to a set point, so is the water level in the flashing drum. This is done with four differential pressure transmitters located in the lower region of the drum. The drum level is controlled by adjusting the feedwater flow rate using a proportional controller. The total error input to the controller (E_{TOT}) is calculated by combining the normalized error in the secondary coolant level with the normalized error between the feedwater and steam flow rates. This is shown in Equation (1). If the feedwater flow rate is too low, both the drum level and flow rate terms would be positive, lending a positive sign to the controller output, thus increasing the feedwater flow rate.

$$E_{TOT} = \frac{D_{LSP} - D_L}{D_{LSP}} + \frac{W_S - W_{FW}}{W_S} \quad (1)$$

D_{LSP} = Drum level set-point.

D_L = Drum level.

W_S = Steam flow rate.

W_{FW} = feedwater flow rate.

Feedwater temperature is controlled to a set point by the feedwater heaters, so it too must be measured. The measurement does not require four RTDs because the accuracy is not as important as in the power related temperature measurements. Three RTDs, two narrow range and one wide range, are adequate for each

feedwater line. If the calibration or functionality of one of the narrow range RTDs is compromised, the other may be used exclusively until the next scheduled outage, or replaced in service. RTDs have very good records of maintaining calibration as long as they are not adversely affected by maintenance operations (such as being jostled by a worker, moving the assembly slightly out of position within its thermowell, increasing response time significantly and altering calibration). The flow rate of the feedwater must be measured as well. This is necessary for drum level control as well as for reactor operation.

2.2. DHRS loop instrumentation

The instrumentation needed for the decay heat removal system must be able to verify proper operation of the system by determining the energy extracted from the primary coolant by each DHRS for post-accident monitoring. The DHRS operates completely by natural circulation, and is designed to maintain the core at safe temperatures indefinitely in a post-accident cool down scenario, even in the event of loss of off-site power and backup electricity.

The primary DHRS heat exchanger (DHRS-HX) for each of four DHRS loops is inside the RPV. It consists of a helical coil tube for primary coolant within a larger pipe for secondary coolant, which enters and exits the RPV at the same elevation as the penetrations for the primary, power operation heat exchangers. The core barrel features fail-open valves just above the core to allow the DHRS primary intake access to the coolant rising out of the core if the reactor coolant pumps (RCP) are offline. There are four DHRS units in the I²S-LWR system, for a total of eight secondary coolant pipes that conduct coolant to and from the final heat sink, air cooled heat exchangers. Figure 2 shows the placement of the DHRS-HX on the outside of the core barrel, in between two primary heat exchanger units. The primary and secondary inlet and outlet locations are shown along with the associated instrumentation.

In order to monitor the overall operation of the DHRS, flow meters suitable for the slower, natural circulation flow in the secondary coolant loops are employed along with RTDs to monitor the temperature and flow rate of the coolant

entering and leaving the pressure vessel, from which the power removed may be readily calculated. The terminal heat exchanger, utilizing air as the ultimate heat sink, is monitored similarly.

Verifying the operation of the internal components is not straightforward. Measuring inside the RPV is restricted by space limitations and harsh operating environment. The parameters of importance for the primary coolant are its temperature and flow rate at the intake and outlet of primary loop. The lower plenum contains RTDs for normal operation 'cold leg' monitoring that can provide exit temperature. If space permits, RTDs can be placed around the outside of the core barrel, proximal to the 'hot' inlet, to monitor inlet temperature. If space does not permit, core exit thermocouples (TC) contained in the in-core instrumentation assemblies may be sufficient for post-accident monitoring.

It may be possible to include a flow meter on the primary coolant 'cold' pipe of the DHRS-HX if space permits and a suitable instrument is available. If primary flow cannot be measured directly, it has to be inferred from the coolant temperatures and the power removed from the secondary side. However, that approach assumes the system is otherwise the same as at the time of calibration.

Figure 2: DHRS heat exchanger and associated instrumentation (Figure from [2]).

The primary coolant could fail to flow through the tube side of the heat exchanger for a number of reasons. The primary coolant level could drop below the hot inlet, near the elevation of the top of the core, though the entire safety system is designed to prevent core uncover. Only a beyond design basis accident could result in such circumstances. It may also be possible for the tube to become sufficiently fouled under some accident condition to resist natural circulation.

If correct power is being extracted by the secondary coolant, then the primary coolant is necessarily flowing, so it is not critical to measure it directly. If correct power is not being extracted from the primary coolant and dumped through the air sink, the information from the instruments may help identify the cause.

If coolant level were to fall and approach the DHRS-HX intake, it would be detected by the primary coolant inventory monitoring system, discussed later. If fouling is the issue then the flow rate through a particular exchanger would be necessarily slowed, reflected in less power extraction measured on the secondary side of that particular unit. If there is some issue with the terminal heat exchanger, which dumps the decay heat to air, the instrumentation which brackets that equipment would show a decrease

in power extracted from the secondary coolant by the terminal heat sink.

3. In-Vessel Instrumentation Development

3.1. In-core instrumentation

For calibration of ex-vessel nuclear instrumentation and to monitor reactor performance, it is valuable to measure neutron flux throughout the core, and coolant temperature at the core exit. In current generation PWRs designed by Westinghouse Electric Company (WEC), periodic measurement is accomplished with moveable detector assemblies that are driven into position within the core through guide tubes that penetrate the bottom of the RPV. The I²S-LWR will not have any penetrations in that region to help prevent the possibility of core uncover, so that approach is not viable. Self-powered neutron detectors (SPND) in fixed in-core detector (FID) strings are used in Combustion Engineering (CE) designed PWRs and would be part of the in-core instrumentation in the new WEC AP1000 PWRs under construction at the Vogtle and V.C. Summer sites in Georgia and South Carolina, respectively.

FID assemblies typically consist of multiple SPNDs and a core-exit thermocouple. SPNDs operate based upon the interactions of neutrons and/or gamma rays with the emitter element of the detector to produce a current proportional to the local flux. Typical emitter elements are rhodium, platinum, and vanadium. Various properties of emitter elements are compared in Table 1. FIDs have the added benefit of providing continuous monitoring as opposed to the periodic monitoring capability of movable in-core detectors.

Table 1: Comparison of attributes of three elements as emitters in SPNDs for use in FID assemblies [3]

Emitter	Rhodium	Vanadium	Platinum
Source	Neutron	Neutron	Gamma
Response delay	Small	Medium	Prompt
Physics model	Medium	Simple	Complex
Signal level response	Large	Small	Small
Response proportional to	Surface	Volume	Volume
Sensor depletion	Large	V. Small	V. Small
Sensitivity change	Large	V. Small	FP Buildup
Sensitivity variation among detectors	Medium	V. Small	Small
Lead-wire compensation	Easy	Critical	Critical

The standard CE design uses rhodium self-powered detectors. AP1000 reactors employ vanadium based detector elements in a design called OPARSSEL, an acronym for “optimized proportional axial region signal separation, extended life.” The number of neutron detectors in each string depends mostly upon the height of the core. Sensors that are too close together generate indistinguishable signals. WEC AP1000 has a 14-foot active core and uses seven SPNDs. The I²S-LWR has a 12 foot core and we anticipate using six SPNDs per instrument tube. WEC reports that the OPARSSEL instrument assemblies provide reduced uncertainty and better fault tolerance than the standard CE style rhodium based design [3]. A sketch of the AP1000 in-core instrument thimble assembly (IITA) is shown in Figure 3. The orientation of the detectors within the thimble assembly is shown along with the dimensions of the thimble.

Figure 3: WEC vanadium based SPND and TC implementation in IITA adopted by AP1000 Reactors. This design is representative of proposed I2S-LWR in-core instrumentation [3]

The downside to vanadium only based detectors is that they exhibit too slow of a response to changes in neutron population for core protection applications. The next generation design of Westinghouse in-core detectors may incorporate both vanadium and platinum detectors to improve response time for direct use in the reactor protection system. The I²S-LWR takes advantage of the latest licensable technology at the time of deployment.

The proposed distribution of the in-core instrumentation assemblies throughout the core of the I²S-LWR is based upon the core fuel assembly map and the distribution patterns found in CE designs [4]. Figure 4 shows the distribution of 34 IITAs within the 121 fuel assemblies of the core. Each thimble assembly goes inside an instrumentation tube at the center of the indicated fuel assembly.

Figure 4: Proposed locations of 34 IITAs in I2S-LWR core map.

3.2. Ex-core instrumentation

As in all the areas of instrumentation development for this reactor, it is desirable to use existing technology whenever possible. Petrovic and Flaspoepler recently evaluated, among other things, the neutron fluence at full power for the I²S-LWR in a SCALE model in order to evaluate the nuclear instrumentation environment and the application of silicon

carbide (SiC) based neutron detectors for source range applications in this reactor [10]. This study was subsequently revised in the latest quarterly report [11] for the project to reflect a 10 cm reduction in the vessel diameter. Both analyses concluded that the larger size of the downcomer of the reactor, compared to common operating PWR designs, reduces the ex-vessel neutron fluence sufficiently that it would not be feasible to locate source range neutron detectors outside of the pressure vessel, as is common in operating PWRs in the U.S. Consequently, they propose using the same strategy for source range detectors proposed for the IRIS reactor, which is to utilize SiC based neutron detectors located inside the reactor vessel in the downcomer region.

The SiC matrix alone interacts with fast neutrons, but not thermal neutrons. Since the population of thermal neutrons in the downcomer is much greater than the population of fast neutrons and also of more direct interest to the monitoring of the core power during startup, it is beneficial to measure the thermal neutron population by means of a ^6Li converting layer surrounding the SiC matrix of the detector. The converting layer undergoes the following reaction:

The energized alpha and tritium products enter the SiC matrix, depositing energy and promoting electrons to the conducting band, which produces a proportional current when a reverse bias voltage is applied to the detector. Fast neutrons interact directly with the nuclei of carbon and silicon in scattering interactions, but with smaller cross sections than the thermal neutron absorption cross section of ^6Li .

The downside to using ^6Li conversion mechanism is that it consumes the lithium converting layer, which equates to detector lifetime. Such a detector suitable for detecting source range levels of neutrons in the downcomer is rapidly consumed under the fluence levels of the reactor at full power. The radial position of the detector in the downcomer can be varied to increase the lifetime, but this adversely affects detector sensitivity. Another possible solution to the lifetime versus

sensitivity problem is to include a moveable shield in the detector assembly that would partially or completely cover the detector once the reactor power level increased above the range monitored by the in-vessel SiC detectors.

Petrovic and Flaspoepler's work found ex-vessel fluence levels in the cavity between the vessel and the concrete biological shield to be greater than 10^6 neutrons/cm²-s at full power. This analysis, coupled with operating ranges obtained from WEC [12], indicates that nuclear instrumentation used in typical loop-type PWRs for intermediate and power range monitoring will be suitable for the I²S-LWR.

3.3. Pressurizer instrumentation

The I²S-LWR pressurizer is located in the top of the pressure vessel (see Figure 1) and requires pressure transmitters and differential pressure level sensors to monitor and control the pressurizer. Four level sensors and four pressure transmitters are sufficient for three-out-of-four reactor trip logic. Typical pressurizer pressure monitoring instruments have a range of zero to 3000 PSI (~20.68 MPa) and a calibration span of 1700 to 2500 PSI (~11.72-17.24 MPa). Qualified lifetime of these pressure transmitters depends strongly on ambient temperature of the instrument, and ranges from less than three years to upwards of 30 years. The electronics generally have a shorter lifetime than the pressure transmitting module itself. Response times are on the order of tenths or even hundredths of a second with accuracies around 0.25%, though ambient temperature and pressure can create an additional 0.50% uncertainty. There is no reason why the I²S-LWR would require better performance than currently operating PWRs enjoy from currently licensed nuclear grade pressure transmitters.

3.4. CRDM instrumentation

The CRDMs in the I²S-LWR are located in the primary coolant riser section. Enclosing the control rods and CRDMs inside the pressure vessel eliminates the possibility of a full rod ejection event. However, the position of every control rod must be monitored. The CRDMs include rod position indicator equipment inside each CRDM, bundling the power, control, and

instrument cabling together for each CRDM. The diameter of each cable is approximately 0.8 inch (20.32 mm) [5].

3.5. Primary coolant temperature instrumentation

The primary coolant ‘hot leg’ and ‘cold leg’ temperatures must be measured accurately and with rapid response time. These values are critical for reactor protection and control. RTDs are the instrument of choice, but access and placement are challenges. The ‘hot leg’ measurements may be made with RTDs inserted into thermowells that are supported by vessel internal structure. The lower plenum temperature (‘cold leg’) is measured using submersible RTDs. We are not aware of a suitable submersible RTD assembly, but conceptually, encasing an RTD in its own small ‘pressure vessel’ should be feasible. Placement and number of sensors within the available space will be addressed to ensure an accurate average coolant temperature measurement. Temperature stratification of the coolant in the plenum volumes will have to be investigated to aid in sensor placement. The hot and cold leg RTDs should have a calibration span of 70 °C surrounding the nominal values of the respective coolants at full power. Typical nuclear grade RTD accuracy is around 0.18 °C, with response times less than seven seconds. The accuracy and response time of these temperature sensors satisfy the coolant temperature monitoring requirements of the I²S-LWR. RTD response times are very slow in comparison to pressure transmitters, due to the fact that heat must transfer from the process, through the thermowell material, then to the actual transmitter before a change in temperature can be detected. In a pressure sensing line the pressure travels at the speed of sound in the medium to a diaphragm or cantilever based pressure transmitter which detects the change in pressure very quickly via its mechanics of operation. The rapid response times of pressure transmitters provides rapid detection of a partial or total loss of flow accident (LOFA). This means that any indirect measure of primary coolant flow based upon temperature

measurements is too slow to base the reactor safety systems on.

3.6. Primary coolant inventory monitoring

Two techniques are proposed for primary coolant inventory monitoring to ensure core submersion in post-accident scenarios. They are thermal probe level sensors (TPLS) and the torsional ultrasonic wave-based in-vessel level measurement system. Differential pressure transmitters are not applicable to RPV coolant level monitoring in the I²S-LWR because they would require pressure tap penetrations in the pressure vessel in the lower portion of the pressure vessel, violating a design constraint of the I²S-LWR concept.

TPLS monitor the presence of water or vapor at their location based upon the difference in thermal conductivity of liquid versus gaseous water. The sensitivity of such an implementation depends on how close together the sensors are placed. For example, if they are one foot apart, then the coolant level in the reactor would be known to be at least the level of the highest submerged sensor, but less than one foot above that sensor.

The torsional ultrasonic wave-based in-vessel level measurement system is a system designed at Oak Ridge National Laboratory [13] specifically for monitoring coolant level in integral LWRs. The system estimates the regional density throughout the RPV to determine the elevation of the surface of the coolant [6]. Either of these is a suitable approach to post-accident primary coolant inventory monitoring in the I²S-LWR.

3.6. Primary coolant flow monitoring

The flow rate of primary coolant in a typical PWR is monitored by the change in pressure of the coolant across the U-tube steam generator. The low flow reactor trip is activated when the magnitude of the pressure drop falls below the set point. However, this is not an extremely accurate method of primary coolant flow measurement. It is extremely fast, which is most important in responding to LOFA. Improved measurement of the primary coolant flow rate lends itself to improved safety and performance.

Pursuant to this end, several potential techniques for measuring the primary coolant flow rate are presented as candidates for use in the I²S-LWR. The objective is not necessarily to select a single, best, technique, but to select the best combination of methods to arrive at a flow rate measurement with the lowest possible uncertainty and fast response time. The use of existing instrumentation for flow rate estimation is also investigated. First, ultrasonic flow measurement is discussed.

3.6.1. Ultrasonic flow meters

The proposed implementation of ultrasonic flow meters in the I²S-LWR, as well as the principle of operation is described in [1]. A figure from that paper is reproduced here for reference in Figure 5. The proposed implementation involves locating an array of ultrasonic transceivers, or transmitter and receiver in one unit, in pairs, around the outside of the RPV, at the elevation of the down-comer region, where penetrations in the vessel wall are prohibited by design. Primary coolant flows through this region after exiting the primary heat exchangers. By placing transceiver pairs spaced evenly around the reactor, a more accurate average coolant flow rate may be determined. There are, however, issues pertaining to the physical interactions of sound waves in high temperature water and when transitioning between media with large differences in density that create practical concerns for the proposed implementation.

One practical issue is that the turbulence of the flow affects the path the sound takes, and thus the transit time. This can be resolved in some applications by measuring the same flow at different diameter sections of pipe. Of course, this solution is not applicable to the I²S-LWR concept due to the lack of primary coolant pipes. It may be possible to solve this challenge by having up and down stream signals triggered by receiving the signal from the other transceiver, then monitoring the frequency of the up and down signal cycle, rather than the time difference between the up and down stream pulses.

Figure 5: Side profile of conceptual implementation of UFM in down-comer of I²S-LWR for measurement of primary coolant velocity. Velocity may be calibrated to mass flow rate [1]

When the sound pulse passes between media of differing density, a fraction of the signal is reflected back into the origin media. The remaining fraction is refracted into the new medium and continues. The greater the difference in density, the larger the fraction of the energy that is reflected. There is also signal attenuation within each medium. Merely being able to successfully send and receive the ultrasonic pulses is not guaranteed in the proposed application. While it is possible to calculate the fraction of the original pulse that will arrive at the destination transceiver in the proposed arrangement, instrument reception sensitivity and pulse emission strength data is not available.

Further, the fraction of the pulse reflected into the RPV when the signal initially transitions from the RPV to the coolant will continue to travel through the RPV toward the other transceiver. This signal will reach the transceiver sooner, and with greater strength than the signal which traverses the coolant. This hurdle can be overcome with careful pulse timing so that the errant signal can be ignored by the electronics.

The speed of sound through water at high temperatures varies strongly with temperature, causing changes in the transit time, refracted angle, signal path, and actual annulus dimensions as compared to the design dimensions. While these issues may be alleviated with proper experimentation and calibration, this calibration will not be valid as the reactor system changes power level [5].

Ultrasonic flow measurement has many positive characteristics if satisfactory solutions to the practical challenges to implementation can be resolved. Experimentation work at the University of Tennessee is planned to construct a representative flow loop upon which to evaluate the feasibility of using UFM's in the I²S-LWR. However, it is prudent to be prepared to proceed without the ability to employ UFM's in this reactor concept. To that end, a few other methods of determining the primary coolant flow are presented.

3.6.2. *Alternate and combined approaches to primary flow monitoring*

A few alternate methods of evaluating coolant velocity are possible with fast sampling rates of instrumentation that is already in the reactor system. Coolant passing through the core flows sequentially past neutron flux monitors and core exit thermocouples.

Cross-correlation algorithms can be employed to detect the same pattern in the signals of different detectors. The time delay between the signals of sequential detectors relates to the velocity of the coolant by the distance between the detectors [7].

Another method of inferring the primary flow rate is to balance the energy extracted by each heat exchanger with the change in temperature across the primary side of the same heat exchanger. These are related by the primary flow rate through the heat exchanger according to the following equation:

$$\eta(\dot{m}_P c_{pP} \Delta T_P) = \dot{m}_S c_{pS} \Delta T_S \quad (3)$$

In Equation (3), $\dot{m}$ is coolant mass flow rate, c_p is the heat capacity (at constant pressure) of the working fluid, and ΔT is the change in temperature of the working fluid due to heat

transfer in the heat exchanger. The overall efficiency of the heat exchanger is specified by η . The upper-case subscripts S and P denote secondary and primary working fluids. Since the working fluid experiences a pressure drop across the exchanger, this simplified relationship is not exact. However, this discrepancy can be accounted for with experimentation and data based model generation, treating η as a correction factor rather than a strict efficiency of heat transfer.

Reactor coolant pumps (RCP) are typically powered by three-phase induction motors. Motor power exhibits a direct relationship to pump flow rate, though not necessarily a linear relationship [8]. Motor power can be measured continuously based upon the current and voltage drawn by the motor. The same data is also used for monitoring the operational integrity of the reactor coolant pumps for signs of wear or malfunction. Early detection of RCP degradation allows for maintenance or replacement during scheduled outage time as opposed to a sudden pump failure requiring an unscheduled outage. There is current work underway on motor degradation at the University of Tennessee.

It may be possible to measure pressure of the primary coolant before and after it passes through the primary heat exchangers. If this proves feasible, the pressure drop across the heat exchangers can be correlated to flow rate. This is the same measurement used for primary coolant flow rate based trips in operating PWRs. Improvements over this technique benefit the system performance and safety by providing more precise evaluation of the flow rate, which improves the capability of monitoring and diagnostic algorithms to detect and isolate anomalous conditions more rapidly.

Ultimately, the uncertainty of each of these methods must be characterized so that a combined uncertainty of the aggregate of all of these methods may be established. The response time of the methods is difficult to characterize. Calorimetric methods are inherently too slow because they rely on temperature sensors with multi-second response times. Pump motor power monitoring should, in theory, have a very fast response time to measure the actual change in power, but the sensitivity of this technique, along with how quickly a change in flow is

reflected in the power level, have not been characterized. The response time of cross-correlation methods has also not been well characterized for non-bubbly flows. As far as determining the response time actually required of primary coolant flow instrumentation to maintain the plant within safety margins, the development of the I²S-LWR concept has not reached this stage.

4. Instrumentation Cabling

4.1. In-core instrument cable routing

Each in-core instrument cable encompasses the instrumentation wiring for one instrumentation string. Each string contains six self-powered neutron detectors and one core-exit thermocouple. The outer diameter of an in-core instrumentation string cable is reported as 0.367 inch (9.32 mm) [3]. Engineering experience at WEC indicates the bending radius of the cable is approximately 5 inch (12.7 cm). The in-core detector cables would exit each assembly at the bottom of the core, bend around the bottom of the core barrel, traversing up to an appropriate elevation for exiting the pressure vessel. This may be at the lowest allowable elevation to reduce the length of the cable run, improving signal to noise ratios. It may also be at a different elevation or relation to the pressure vessel support cone, connecting flange, or both. The space within the RPV for cables to bend and exit through instrumentation plugs is found between the DHRS and primary heat exchangers, and between primary heat exchangers and the vessel wall. Additional details of in-core instrumentation cable routing in a commercial PWR are described in [14].

4.2. CRDM Cable routing

The control rod drive mechanism cables contain power and control cables for the CRDM as well as the cabling for the rod position instrumentation in a bundle. It is estimated that the diameter of the complete cable would not be greater than 0.8 inch (20.3 mm), with vessel penetrations of 1.5 inch per cable, according to expertise passed from WEC. The CRDM cables are routed over the top of the core barrel, across the top of the heat exchangers so as not to

interfere with flow, and out of the RPV in the same regions as described for the in-core instrumentation cables. Pathways for cabling to the outside of the RPV are shown in Figure 6.

Figure 6. Cable exit pathways for in-core instrumentation and control rod drive (RCCA) and position indication hardware.

5. Accident Monitoring

Instrumentation becomes even more important in a nuclear power plant if abnormal conditions arise. The instrumentation provides all the information that is available to operators and emergency response personnel in dealing with an accident scenario. Consequently, it is important to consider the information that is critical to have when responding to an accident when designing an instrumentation system and to optimize this design with accident mitigation in mind.

The International Atomic Energy Agency (IAEA) considers the terms *accident monitoring*,

and *post-accident monitoring* to be equivalent. In this paper, the term *accident conditions* refers to conditions or states that are less frequent and more severe than normal operational occurrences. Normal operational occurrences are events which the plant is intended to experience at least once during the lifetime of the plant. Accident conditions include two classes of unexpected conditions, design basis accidents (DBA) and design extension conditions (DEC). DBAs are postulated events the plant is designed to recover from, to a safe condition, according to conservative methodologies. DECs are considered in the design of the plant according to best estimate methodology. They are not extensions of the DBA scenarios but general extensions of states the plant may enter resulting in accident conditions. In some cases, the effects of DEC are limited by the described scenarios of one or more DBAs. In other cases, DEC results in significant fuel damage. This subset of DEC is referred to as severe accidents [9].

The monitoring of an accident is performed with both installed, permanent instruments used for normal operation, installed, permanent instruments not normally in service, and portable instruments and equipment that are incorporated into an accident response plan. This section discusses the instrumentation used for normal operation and how it is applicable to accident monitoring.

There are several ways in which the proposed instrumentation systems support accident monitoring. The primary system instrumentation is designed to provide the necessary information to operators to respond quickly and appropriately to any accident which leaves the instrumentation systems operational. The primary coolant inventory monitoring approach provides continuous monitoring of the inventory of primary coolant in the event of a loss of coolant accident (LOCA). This information, coupled with core power monitoring by core-exit TCs and the DHRS instrumentation, which verifies the energy extracted from the primary coolant during DHRS operation, yields a complete picture of decay heat removal process. These instruments also aid in isolating any malfunctions which may occur in the DHRS systems.

Appropriate response to an accident scenario is predicated largely on knowledge of the malfunction or failure which leads to the accident condition. If a reactor is losing primary coolant and no one knows why, the response is limited to indefinite replacement of the coolant. When prepared, ultra-pure makeup water runs out and must be replaced with water from a general purpose supply, such as the nearest potable water supply, the rates of reaction for corrosion processes will increase as the water chemistry of the reactor is unbalanced by the uncontrolled ion content of the replacement coolant. This increases the probability of fuel failure, as well as providing the potential to exacerbate the original leak or cause new leaks to initiate. Consequently, it is desirable to be able to isolate the source of equipment failure resulting in the accident such that the failure may be mitigated to bring the reactor system to a stabilized, safe condition. One of the principal benefits of the ultrasonic level monitoring system developed at ORNL is that if a penetration in the RPV is spraying primary coolant out of it, it should be spatially isolatable from the instrumentation as a difference in density as compared to both the pressure vessel wall and the air and insulation surrounding the pressure vessel. If a leak can be found, it can potentially be stopped.

The integrity of the fuel is another important factor for accident monitoring [9]. If the integrity of the fuel can be maintained, large release of radionuclides can generally be prevented. Hydrogen monitors in the pressurizer can provide indication of fuel integrity based upon the presence of hydrogen gas, a corrosion product. If nucleate boiling or significant hydrogen gas bubble production is occurring in the core, it may be detected by cross-correlation of the signals from in-core flux monitors and core-exit TCs. This cross correlation algorithm will already be running on the appropriate signals for core flow mapping, so gas detection becomes a matter of implementing an appropriate monitoring algorithm to look at the cross-correlation data for patterns indicative of gas bubble formation.

Ultimately, the majority of the information which may be extracted from the instrumentation data will come from rapid, on-

line analysis of the data. This analysis capability should be automated and incorporated into the original design of the plant, so that the efficacy may be taken credit for in the calculation of the core damage frequency expected from the reactor protections systems.

6. Concluding Remarks and Future Work

There are many areas related to the instrumentation of this reactor concept, both general and specific, that need to be addressed if this concept is developed into a complete design that can be licensed and constructed. Not all of them will be ultimately addressed by this phase of the development of the I²S-LWR because resources are limited. Instrumentation topics this phase will investigate include:

- optimal instrument allocation, by number and location
- application of wireless transmission of instrumentation data from containment to control room, as well as offsite for post-accident monitoring of severe accidents
- application of fault tolerant and resilient control systems to the I²S-LWR instrumentation approach

Other instrumentation issues which arise will be addressed at an appropriate level for the scope of the work.

Spent fuel pool monitoring is being addressed by the research group at Virginia Polytechnic Institute.

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